RELATED CORRESPONDENCE

AUG 5 1981 Office of the Secretary Docketing & Service Branch

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before The Atomic Safety and Licensing Board

In the Matter of	)	
PENNSYLVANIA POWER & LIGHT COMPANY	)	Docket Nos. 50-387
and	)	50-388
ALLEGHENY ELECTRIC COOPERATIVE, INC.	)	
(Susquehanna Steam Electric Station,	)	
Units 1 and 2)	)	

RECEIVED AUG 06 1981 U.S. NUCLEAR REGULATORY COMMISSION

AFFIDAVIT OF GEORGE R. ABRAHAMSON IN SUPPORT OF SUMMARY DISPOSITION OF CONTENTION 7(a)

County of San Mateo)
SS:
State of California)

George R. Abrahamson, being duly sworn according to law, deposes and says:

1. I am Vice President, Physical Sciences Division and Director of the Poulter Laboratory, SRI International. My business address is 333 Ravenswood Avenue, Menlo Park, California. I give this affidavit in support of Applicants' Motion for Summary Deposition of Contention 7(a) in this proceeding. The facts and conclusions are true and correct to the best of my knowledge and belief. My professional qualifications are attached as Exhibit "A" hereto.

2. Contention 7(a) in this proceeding states as follows:

> The nuclear steam supply system of Susquehanna 1 and 2 contains numerous generic design deficiencies, some of which may never be resolvable, and which, when reviewed together, render a picture of an unsafe nuclear installation which may never be safe enough to operate. Specifically:
>
> a. The pressure suppression containment structure may not be constructed with sufficient strength to withstand the dyanmic forces realized during blowdown.

As will be shown below, it is my professional opinion that the Susquehanna Steam Electric Station (SSES) containment can withstand the dynamic forces realized during blowdown with ample safety margin.

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3. This affidavit describes the dynamic loads imparted to the containment structures during blowdown. These loads are the hydrodynamic loads generated during steam discharge into the water pool used to condense steam within the SSES containment. The affidavit compares the hydrodynamic loads to the containment design capacity and test level. I begin with a brief description of the SSES containment and a summary of the background on hydrodynamic loads. This is followed by a review of the only two events that produce hydrodynamic loads: steam relief valve discharge, and loss-of-coolant accident. Finally, I compare the hydrodynamic loads to the containment design capacity and containment test level. The comparison shows that the SSES containment can withstand the hydrodynamic loads with ample safety margin.

SSES Containment

4. Figure 1 shows the SSES containment structure. The primary containment completely encloses the reactor vessel. It consists of a base mat, a hollow cylinder, a hollow cone, and a domed cap. The base mat is reinforced concrete 7 feet 9 inches thick and rests on slate-like siltstone. The cylinder and cone are reinforced concrete 6 feet thick; all the inner surface of the concrete is lined with steel plate 1/4 inch thick. The cap is steel plate, 1-1/2 inches thick, held in place by 80 bolts 2-3/4 inches in diameter.

5. The containment is divided into two chambers by the horizontal diaphragm slab at the junction between the cylindrical and conical sections. The upper chamber is the drywell and the lower chamber is the suppression chamber or wetwell. The drywell contains the reactor vessel and associated piping, valves and equipment; the wetwell is filled to a depth of 24 feet with water used to condense steam. Steam can be discharged into the wetwell water pool by actuation of a steam relief valve (SRV) or by a loss-of-coolant accident (LOCA). These are the events that produce hydrodynamic loads in the containment.

6. Each of the 16 steam relief valves has a discharge line that extends downward from a main steam line to the bottom of the water pool, where it connects to a quencher discharge device (Figures 1 and 2). Actuation of one or more SRVs results in steam discharge from the associated quenchers.

7. There are 87 downcomers that connect the drywell to the water pool (Figures 1 and 13). The downcomers are 24-inch diameter open pipes with a deflector shield on the top; the deflector shield does not restrict the flow of steam into the pipe. During a LOCA, steam fills the drywell and is discharged into the water through all 87 downcomers simultaneously.

Background on Hydrodynamic Loads

8. In April 1972 at the Wurgassen Nuclear Plant, a boiling water reactor (BWR) built by the German firm AEG-Kraftwerk Union, a safety relief valve was opened during startup testing and failed to close. The reactor remained at full pressure, and the valve discharged steam into the containment suppression chamber until the suppression pool water heated from just above ambient to about 170°F, in approximately 30 minutes. Pulsating condensation developed and impulsive forces with substantial underpressure amplitudes acted on the containment, eventually causing leakage from the bottom liner plate. As a result, concern was expressed that the structural integrity of other BWR pressure containment systems could be sensitive to SRV induced dynamic loads.

9. The Atomic Energy Commission issued Bulletin 74-14 to all BWR owners on November 14, 1974 to alert them to the potential problems of condensation instability (Wurgassen effect) due to SRV operation. The Commission requested verification that BWR suppression pools had been designed to withstand loads similar to those which were being experienced. In January 1975 the General Electric - Nuclear Energy Program Division (GE-NEPD) identified the following dynamic load conditions which had not been fully considered in the design criteria of Mark II BWR containments:

a. SRV discharge thermo-hydrodynamic phenomena.

b. Design basis accident (DBA), loss-of-coolant accident (LOCA) hydrodynamic phenomena.

10. Following the GE announcement, the containment construction sequence for the SSES as altered to enable the Pennsylvania Power & Light Company (PP&L) and its architect-engineer, Bechtel Power Corporation, to ascertain the effect of these phenomena on the existing SSES design. A task force which included representatives from Bechtel-San Francisco, GE-NEPD, and PP&L was formed in March 1975 to evaluate existing design criteria with respect to the newly defined SRV and DBA-LOCA loadings. In May 1975 Bechtel completed a preliminary study incorporating the effects of the new phenomena in the design criteria for the SSES suppression chamber structures and safety related equipment. As a result of this investigation, it was decided that the following civil-structural modifications were to be incorporated immediately in the containment design to aid in load transfer and add additional conservatism to the existing design:

a. The number of reinforcing bars in the suppression chamber vertical walls was increased.

b. The number of embedments in the suppression chamber walls for downcomer/piping restraints was increased to accommodate future requirements.

c. Anchor bolts were placed on the underside of the diaphragm slab to accommodate additional supports for the SRV discharge piping for horizontal runs should they be needed.

d. Additional anchor bolts were placed within the drywell wall to allow installation of additional snubbers and pipe restraints, if required.

e. The diaphragm slab shear reinforcement was changed from a 45° to a 90° orientation (with respect to the horizontal plane) to accommodate the most conservative pool swell uplift loadings yet predicted.

11. It became evident that a complex technical issue existed for all Mark II plants, and PP&L sought to create a unified utility group to address the matter. A Mark II BWR containment owners group was formed in June 1975 to define precisely the suppression pool dynamic loads and explore ways to assess their impact. As the direct result of action taken by the Mark II containment owners, a generic Dynamic Forcing Function Information Report, NEDE-21061P Rev. 1, which was also known as the DFFIR, was issued jointly by GE-NEPD and Sargent and Lundy for the Mark II owners in September 1975.

12. Based on the analytical techniques included in the DFFIR, a preliminary SSES unique containment design assessment was submitted by PP&L on March 15, 1976.

13. As the body of the useful supportive data increased, Revision 2 of the DFFIR was issued jointly by GE-NEPD and Sargent and Lundy for the Mark II containment owners group on September 1, 1976, as NEDO/NEDE 21061, Rev. 2. It was at this time renamed the DFFR.

14. In addition to its participation in the Mark II owners group, PP&L in November 1976, selected Stanford Research Institute, now called SRI International (SRI), as a consultant to supplement PP&L in-house technical resources.

15. A review by PP&L and SRI of the Mark II program and discussions with others indicated that the SRV discharge loads that caused the problem at Wurgassen could be eliminated by using a quencher discharge device at the outlet into the water pool.

16. Although the Mark II owners group had quencher-related tasks in their program, these tasks were not sufficiently timely to satisfy SSES construction schedule needs. From the work done in Europe at ASEATOM, MARVIKEN, and Kraftwerk Union, PP&L discovered that all known quencher designs were based on data from Kraftwerk Union (KWU). Thus, in March, 1977, SRI, Bechtel and PP&L visited KWU for discussion and tour of quencher-related facilities. In late July 1977, PP&L employed the services of KWU to design a SSES-unique quencher device.

17. Kraftwerk Union provided PP&L design and test reports pertaining to the quencher development to demonstrate design adequacy and quality of their device. These documents were submitted to the Nuclear Regulatory Commission (NRC) in January, 1978. The quencher load specification was submitted to the NRC in April, 1978. To verify KWU's design approach, a full-scale SSES-unique quencher was tested by KWU for PP&L. The documentation of this test series and verification of the design specification was submitted to NRC in March, 1979.

18. The definition of LOCA loads for SSES is in basic accordance with the Mark II program. In addition, PP&L conducted a series of transient steam blowdown tests in a test tank in Mannheim, Germany, to provide data to resolve NRC concerns on the differences in downcomer length between the original facility used by GE to obtain LOCA loads and a prototypical Mark II containment, and to verify the LOCA load specification used in the SSES design.

19. The licensing documentation submitted to NRC for the SSES is summarized in Appendix A.

SRV Discharge Phenomena

20. There are sixteen SRV discharge lines running from the main steam lines at the top of the reactor vessel into the water pool (one is shown in Figure 1). Each line has a device at the discharge end to enhance steam condensation, as shown in Figure 2. These devices, called quenchers, were specifically designed for SSES to minimize hydrodynamic loads. Figure 3 shows a schematic drawing of the SSES quencher. The horizontal centerline of the quencher is about 4 feet from the bottom of the pool. Steam is discharged through about 1000 holes approximately 0.5 inch in diameter; this greatly enhances the water surface exposed to steam and thereby increases the rate of condensation and eliminates the types of loads encountered at Wurgassen.

21. Prior to SRV actuation, the steam discharge line contains air down to the water level. Upon SRV actuation, steam enters the line and compresses the air. As the water clears the discharge line, compressed air emerges into the pool, followed by steam. The air discharge phase is called the air clearing phase, and the steam discharge phase is called the condensation phase.

22. During air clearing, air enters the pool at a pressure substantially higher than the local hydrostatic pressure and forms a bubble adjacent to the quencher. The excess pressure causes the bubble to expand, giving an outward velocity to the water. Due to the inertia of the water, the bubble expands beyond its equilibrium volume at the local hydrostatic

pressure, and is eventually driven back again compressing the air in the bubble. Thus the bubble oscillates, producing a periodic pressure history.

SRV Loads

23. To determine the pressure on the pool boundary, which is the hydrodynamic load on the containment, an extensive test program was undertaken by PP&L at the Karlstein test facility of Kraftwerk Union, the German firm with extensive experience in nuclear reactor steam discharge phenomena that designed the quencher. Tests were performed using an actual SSES steam relief valve, actual steam line diameters and line lengths, and an actual quencher. The water pool used in the tests was the same depth as the SSES pool, but the area was decreased to 1/16 of the SSES pool area, which is the pool area per SRV line. This corresponds to all sixteen valves actuating simultaneously, which is the case that gives the highest loads on the containment structure.

24. To permit calculations of containment response to proceed in parallel with the test program, KWU provided PP&L with an SRV load specification based on data taken by KWU in previous in-plant quencher tests. The load specification gives the pressure amplitude and distribution on the pool boundary, and the frequency range of the oscillations. The pressure measurements obtained in the SSES quencher tests verified the validity of the load specification.

25. The test tank is shown in Figure 4. The tank contains concrete blocks to reduce the pool area to 1/16 of the SSES pool area. The quencher depth is the same as in the SSES pool. The symbols P5.1, etc., refer to locations of pressure gages.

26. The tests covered the range of reactor operating conditions. The test matrix for the air clearing tests is shown in Figure 5; the test parameters are listed on the left. Tests were performed with the longest and shortest discharge lines, with different temperatures of the air in the discharge line (temperature affects the total air mass in the line), different water levels in the discharge line, vacuum breaker open and closed, various pool temperatures, various steam pressures, and different numbers of actuations. There were five repeat tests.

27. Figure 6 shows the reactor operating conditions in terms of reactor pressure and water pool temperature, together with the points for Test Group No. 1 from Figure 5. During reactor operation, only conditions within the operation field can be reached. The tests were designed to determine the hydrodynamic loads over the operation field.

28. Figure 7 shows the location of the condensation tests in the operation field. The solid lines extending upward and to the left indicate the loci of combinations of reactor pressure and pool temperatures that occur during a test. Thus, reactor pressure decreases and pool temperature increases. Figure 8 indicates the different behaviors observed during the steam condensation tests.

29. The main data relating to hydrodynamic loads on the containment are the pressure measurements. The records shown in Figures 9 and 10 are typical of the data obtained. Figure 9 shows typical pressure histories for an air clearing test. The traces are for gages located throughout the pool (Figure 4). The peak overpressures are of the order of 1 bar (about 15 psig), and the main frequency is about 6 Hz.

30. Figure 10 shows typical pressure history for a condensation test. Figure 10a covers a long time period, Figures 10b to 10e show pressure histories on expanded time scales. The pressure amplitudes are small compared to those during air clearing.

31. For actuation of less than 16 valves, the pressure measurements must be adjusted for the larger area in the SSES pool to obtain the pool boundary pressures. This is done by a calculational method that was checked by tests in the SSES pool (see second paragraph below). The pressure histories on the pool boundary are used as input to a computer model of the containment, and the results of the calculations are compared with the allowable conditions to determine the safety margin.

32. In addition to the loads that occur on the pool boundary during air clearing, loads can also occur on submerged structures due to pressure gradients in the pool. When pressure gradients are present, one side of a structure experiences a higher pressure than the other side, and hence an unbalanced force results that must be sustained by the structure.

33. A test program to measure loads on submerged structures for SRV discharge in the SSES pool was undertaken for PP&L by SRI. SRI designed a device, called a bubble source, that simulates SRV air clearing. The source was calibrated in the same tank in which the SRV discharge tests using the SSES quencher were performed. The calibration consisted of matching the peak pressure and oscillation frequency of the bubble source to the values observed during SRV discharge. This assured that the submerged structure loads found by using the bubble source in the SSES pool would be the same as would result from SRV air clearing. The similarity of the most severe pressure histories from SRV discharge in the Karlstein tank and the pressure history from the SRI source in the same tank is shown in Figure 11.

34. Eight tests were performed in the SSES pool with the SRI bubble source. Five tests were performed with one bubble, and three were performed with two bubbles. The loads on the submerged structures were well below the design loads.

35. These tests in the SSES pool provided an opportunity to check the accuracy of the frequency shift that occurs when SRV air clearing occurs in the SSES pool instead of the Karlstein tank. As can be seen in Figure 12, the measured oscillation frequencies in the SSES pool match the calculated frequencies very well. The upper dot shows that the 6 Hz frequency of Figure 9 for the Karlstein tank shifts to about 9 Hz in the SSES pool. The other dot at about 6 Hz corresponds to a frequency of 4 Hz in the Karlstein tank.

LOCA Phenomena

36. When it is desired to release steam from the reactor, it is done by actuating steam relief valves that allow steam to enter the water pool through discharge lines. A loss-of-coolant accident (LOCA) is an unscheduled flow of water or steam from the reactor into the drywell, and then into the wetwell through the 87 downcomers shown in Figure 13. The cause of the opening through which the flow occurs is not defined, hence these are called "postulated breaks."

37. Of the postulated breaks, the one that produces the largest steam flow and hence the highest pressure in the containment is the design basis accident (DBA). The DBA for SSES is a postulated double ended rupture of a 28-inch diameter recirculation line. Such a break results in rapid pressurization of the drywell. This is accompanied by a downward acceleration of the water in the downcomers, followed by discharge of air into the water at the downcomer exit plane and an upward motion of the water above the downcomer exit plane. The upward motion of the water continues until air breaks through the water layer; the water then falls back to rejoin the water below the downcomer exit plane. This phase is called "pool swell."

38. During and after pool swell, the flow through the downcomers decreases in air content and increases in steam content. The steam condenses as it contacts the water, and the air forms small bubbles that rise to the surface in the pool. The flow into the pool continues until the pressure in the drywell decreases to the water pressure at the downcomer exit plane. The phase following pool swell is called the condensation phase.

LOCA Loads

39. Loads on the containment during pool swell were investigated by SRI under the auspices of the Electric Power Research Institute (EPRI), of which PP&L is a member. The apparatus used was a 1/13.3 scale model of a quadrant of the Mk II suppression pool, shown in Figure 14. In these tests, the critical load on the containment is the differential pressure across the diaphragm slab. Figure 15 shows the drywell and wetwell pressures, and Figure 16 shows the differential pressure. The critial loads occur when the differential pressure on the diaphragm slab is maximum downward and upward, corresponding to the maximum and minimum in Figure 16. Analyses of the containment show that at these points the stresses are within the allowable range.

40. As with the SRV discharge, a single cell approach was used to determine LOCA loads (pool swell and condensation) in full-scale tests. Figure 13 shows a horizontal cross section of the SSES pool. The 87 downcomers are distributed reasonably uniformly, hence to examine LOCA

loads we may allocate to each downcomer a certain area of the pool. Such an allocation is shown in Figure 17. The smallest pool area results in the highest LOCA loads, hence the smallest pool area is used in the single cell approach. The drywell volume for the single cell is taken as the same fraction of the total drywell volume as the single cell pool area is of the total pool area.

41. With the single cell pool area and drywell volume determined, the apparatus shown in Figure 18a and 18b was constructed. Except for the reduced area, everything is prototypical of SSES. The downcomer is the same size as in the plant, the water depth is the same, etc. Tests were performed with this apparatus by flowing into the drywell 1/87 of the flow that would result from the DBA in the plant. This is the fraction that would flow through each of the 87 downcomers. Measurements were made of the pressures in the drywell, wetwell air space, and wetwell water space. These measurements give the loads on the containment.

42. From the pressure measurements a load specification was derived that is used in calculating containment response. The critical aspect of the load for containment integrity is the pressure buildup in the drywell and the wetwell. As an indication of the safety margin in strength of the containment, the measured pressure buildup is compared below with the containment test pressure.

43. The matrix for the LOCA tests is shown in Figure 19. A total of 22 tests (11 test conditions, 2 tests each) were performed covering a range of pool temperatures and steam flows. In particular, break sizes correspond to the design basis accident, here called recirculation line (RCL) break, the main steam line (MSL) break, and 1/3 MSL and 1/6 MSL breaks. Other break sizes result in combinations of pool temperatures and steam flow that are reasonably close to the values that occur in these tests.

44. The other main parameter is pool temperature, which ranges from 75 to 130°F. Drywell air content was varied to show the effect of reduced air content on LOCA phenomena.

45. The steam flow was designed to be 1/87 of the calculated flow from the reactor vessel for the given break ize. Good repeatability was achieved, as shown for RCL tests 1 and 2 … Figure 20. The steam mass flows for the other breaks are shown in Figure 21.

46. The pool pressure at gage P6.4 for Test 5 is shown in Figure 22. Gage P6.4 is located near the end of the downcomer, where condensation occurs, and is in the region where the pressure is highest (Figure 18b). Test 5 was chosen because it gave high pool pressures; it was also one of the large breaks (Figure 21). The top trace in Figure 22 gives the pressure history for the entire test time. The bottom two traces are on an expanded time scale to show the details of the pressure variation. We see that about every second the pressure exhibits a decrease, followed by a sharp rise and a damped oscillation. The successive pressure signals at one second intervals correspond to the frequency of condensation events at the downcomer exit. A condensation event consists of several stages: (1) the growth of a steam bubble at the end of the downcomer, (2) rapid condensation of the steam bubble (pressure decrease), and (3) collapse of the surrounding water into the steam cavity (pressure rise). The damped oscillation is due to pressure oscillations excited in the pool and in the downcomer air by the collapse.

47. The pressure histories for the wetwell air space and the drywell during an RCL break are shown in Figure 23. The wetwell pressure rises to 1.74 bars (25.2 psig) and the drywell pressure rises to 2.60 bars (37.7 psig). The RCL break produces the most rapid flow of steam into the drywell, and the drywell and wetwell pressures for the RCL break are greater than for smaller breaks.

Comparison of Hydrodynamic Loads with Containment Design Capacity and Test Level

48. SRV discharge produces pressure loads on the pool boundary. LOCA produces pressure loads in the drywell, in the wetwell air space, and on the pool boundary. Computer code calculations show that the combined

SRV and LOCA loads (pressure and frequency) acting on the containment produce stresses in the containment that are within the design values.

49. Moreover, the design pressure for the containment is 53 psig for both the wetwell and the drywell. The SSES containment has already been tested by pressurizing it to 61 psig with air, which is more than 50% greater than the pressure of 37.7 psig produced in the containment by an RCL break (Figure 23).

50. I conclude that the SSES containment can withstand the hydrodynamic loads from SRV discharge and LOCA with ample safety margin.

George R. Abrahamson

George R. Abrahamson

Subscribed and sworn to before me this 31st day of July 1981.

June R. Wilson

Notary Public

OFFICIAL SEAL
JUNE R. WILSON
NOTARY PUBLIC - CALIFORNIA
SAN MATEO COUNTY
My Commission Expires July 2, 1983

Appendix A

SSES LICENSING BASIS

1. Mark II Containment - Supporting Program

Task Number	Activity	Activity Type	Documentation
A.1	"41" Test Program	Phase I Test Report Phase I Appl Memo Phase II & III Test Rept Application Memorandum	NEDO/NEDE 13442-P-01 Application Memo NEDO/NEDE 13468-P NEDO/NEDE 23678-P
A.2	Pool Swell Model Report	Model Report	NEDO/NEDE 21544-p
A.3	Impact Tests	PSTF 1/3 Scale Tests Mark I 1/12 Scale Tests	NEDO/NEDE 13426-P NEDO/NEDE 20989-2P
A.4	Impact Model	PSTF 1/3 Scale Tests	NEDO/NEDE 13426-P
A.5	Loads on Submerged Structures	LOCA/RH Air Bubble Model LOCA/RH Water Jet Model	NEDO/NEDE 21471-P* NEDO/NEDE 21472-P*
A.6	Chugging Analysis and Testing	Single Cell Report 4T FS1 Report	NEDO/NEDE 23703-P NEDO/NEDE 23710-P
A.9	EPRI Test Evaluation EPRI 1/13 Scale Tests EPRI Single Cell Tests	EPRI - 4T Comparison 3D Tests Unit Cell Tests	NEDO 21667 EPRI NP-441 EPRI Report
A.11	Multivent Subscale Testing and Analysis	Preliminary MV Prog Plan MV Test Program Plan & Proc. - Phase I Phase I Test Report MV Test Prog Plan & Proc. - Phase II Phase II Test Report CONMAP Tests MHM Verification 1/10 Scale	NEDO 23697 NEDO 23697 Rev 1 Report NEDO 23697, Rev. 1 Supp. 1 Report Report NEDE 25116-P
A.13	Single Vent Lateral Loads	Dynamic Analysis Summary Report Summary Report (Extension)	NEDO 24106-P NEDE 23806-P Report
C.1	DFFR Revisions	Revision 3	NEDO/NEDE 21061-P Rev. 3*
C.3	NRC Round 1 Questions	DFFR Rev. 2 DFFR Rev. 2 Amendment 1 DIFFR Rev. 3 Appendix A	NEDO/NEDE 21061-P Rev. 2 NEDO/NEDE 21061-P Rev. 2 Amend. 1 NEDO/NEDE 21061-P Rev. 3 Appendix A
C.5	SRSS Justification	Interim Report SRSS Report SRSS Exec. Report SRSS Criteria Appl. SRSS Bases SRSS Justification Suppl.	(NEDE 24010) NEDO/NEDE 24010-P Summary Report NEDO/NEDE 24010-P Suppl. 1 NEDO/NEDE 24010-P Suppl. 2 Report
C.6	NRC Round 2 Questions	DFFR Amendment 2 DFFR Amend 2, Suppl 1 DFFR Amend 2, Suppl 2 DFFR Rev. 3, Appendix A	NEDO/NEDE 21061-P Rev. 2 Amend. 2 NEDO/NEDE 21061-P Rev. 2 Amend. 2, Supp. 1 NEDO/NEDE 21061-P Rev. 2 Amend. 2, Supp. 2 NEDO/NEDE 21061-P Rev. 3 Appendix A
C.7	Justification of "41" Bounding Loads	Chugging Loads Justification	NEDO/NEDE 23617-P NEDO/NEDE 24013-P NEDO/NEDE 24014-P NEDO/NEDE 24015-P NEDO/NEDE 24016-P NEDO/NEDE 24017-P NEDO/NEDE 23627-P

*Only part of this report used as licensing basis.

Task Number	Activity	Activity Type	Documentation
C.8	S/RV and Chugging FSI	Prestressed Concrete Reinforced Concrete Steel	NEDO/NEDE 21936-P
C.13	Load Combination & Functional Capability Criteria	Criteria Justification	NEDO 21985
C.14	NRC Round 3 Questions	Letter Report DFFR, Rev. 3 Appendix A	Letter Report NEDO/NEDE 21061-P Rev.3, Appendix A
C.15	Submerged Structure Criteria	NRC Question Responses	Letter Report

II. KWU Reports

Document Number	Title	Documentation
1	Formation and oscillation of a spherical gas bubble	AEG - Report 2241
2	Analytical model for clarification of pressure pulsation in the wetwell after vent cleaning	AEG - Report 2208
3	Tests on mixed condensation with model quenchers	KWV - Report 2593
4	Condensation and vent cleaning tests at GKM with quenchers	KWV - Report 2594
5	Concept and design of the pressure relief system with quenchers	KWV - Report 2703
6	KKB vent clearing with quencher	KWV - Report 2796
7	Tests on condensation with quenchers when submergence of quencher arms is shallow	KWV - Report 2840
8	KKB - Concept and task of pressure relief system	KWV - Report 2871
9	Experimental approach to vent clearing in a model tank	KWV - Report 3129
10	KKB - Specification of blowdown tests during non-nuclear hot functional test - Rev. 1 dated October 4, 1974	KWU/V 822 Report
11	Anticipated data for blowdown tests with pressure relief system during the non-nuclear hot functional test at nuclear power station Brunsbuttel (KKB)	KWU - Report 3141
12	Results of the non-nuclear hot functional tests with the pressure relief system in the nuclear power station Brunsbuttel	KWU - Report 3267
13	Analysis of the loads measured on the pressure relief system during the non-nuclear hot functional test at KKB	KWU - Report 3346
14	KKB - Listing of test parameters and important test data of the non-nuclear hot functional tests with the pressure relief system	KWU - Working Report R 521/40/77
15	KKB - Specification of additional tests for testing of the pressure relief valves during the nuclear startup, Rev. 1	KWU/V 822 TA
16	KKB - Results from nuclear startup testing of pressure relief system	KWU - Working Report R 142-136/76
17	Nuclear Power Station Phillipsburg Unit 1 Hot Functional Test: Specification of pressure relief valve tests as well as emergency cooling and wetwell cooling system	KWU/V 822/RF 13
18	Results of the non-nuclear hot functional tests with the pressure relief system in the nuclear power station Phillipsburg	KWU - Working Report R 142-38/77

Document Number	Title	Documentation
19	KKPI - Listing of test parameters and important test data of the non-nuclear hot functional tests with the pressure relief system	KWU - Working Report R 521/41/77
20	Air oscillations during vent clearing with single and double pipes	AEG - Report 2327

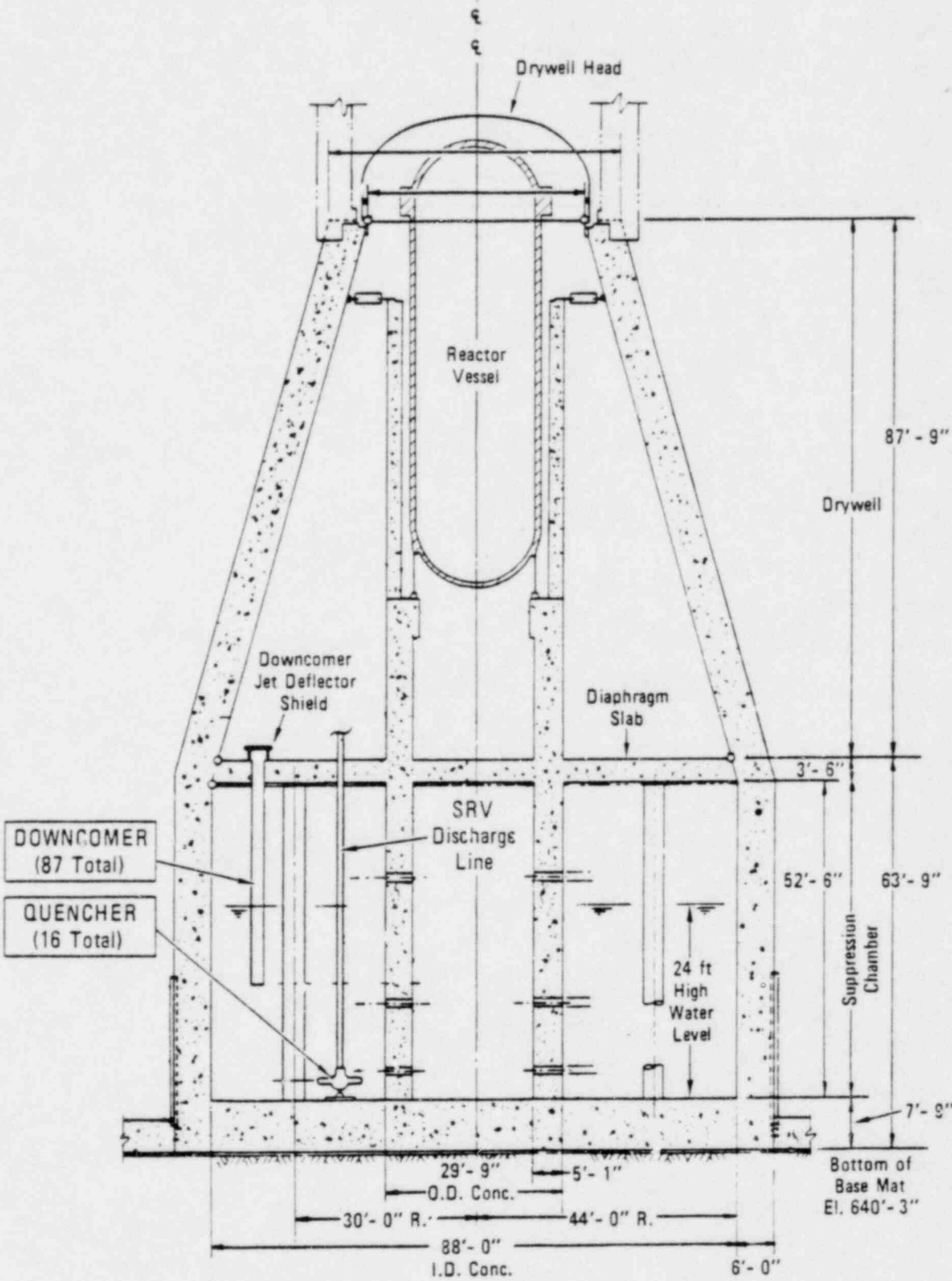


FIGURE 1 CROSS SECTION OF CONTAINMENT

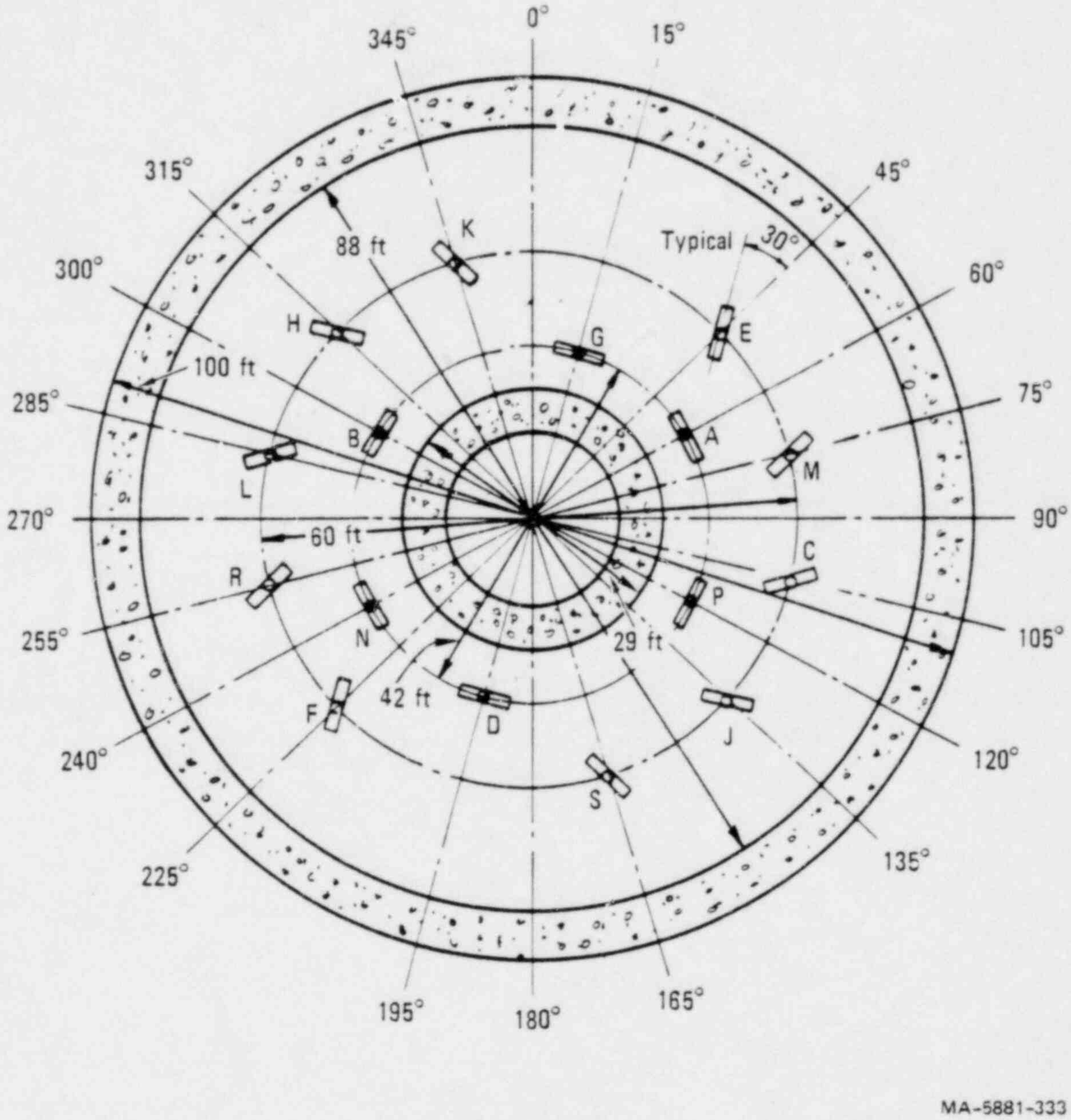
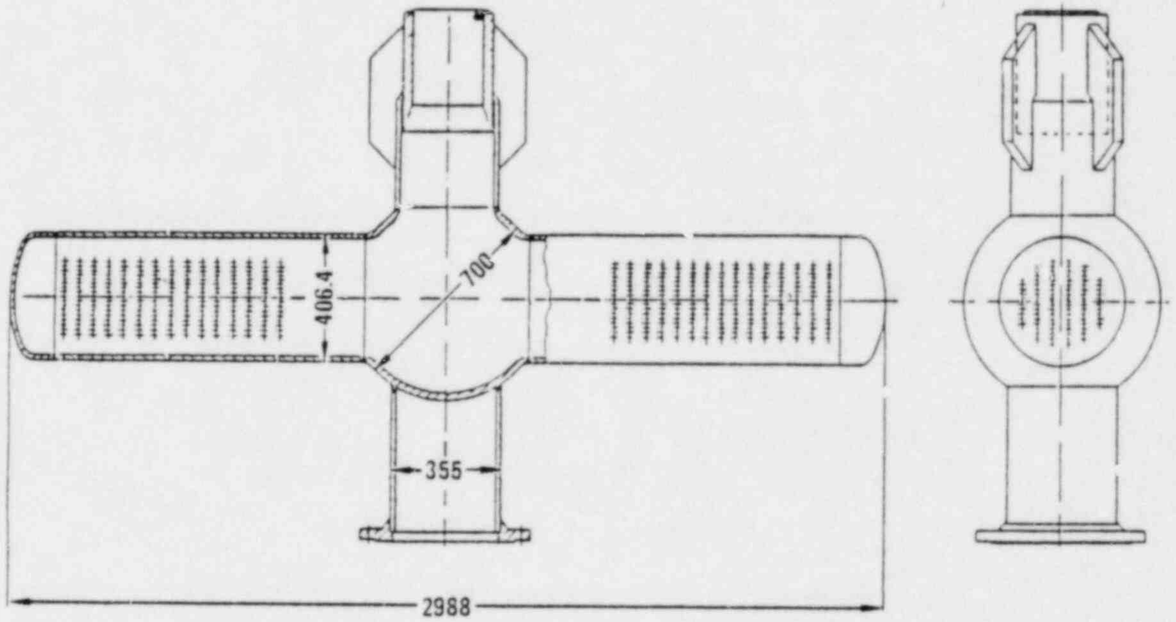


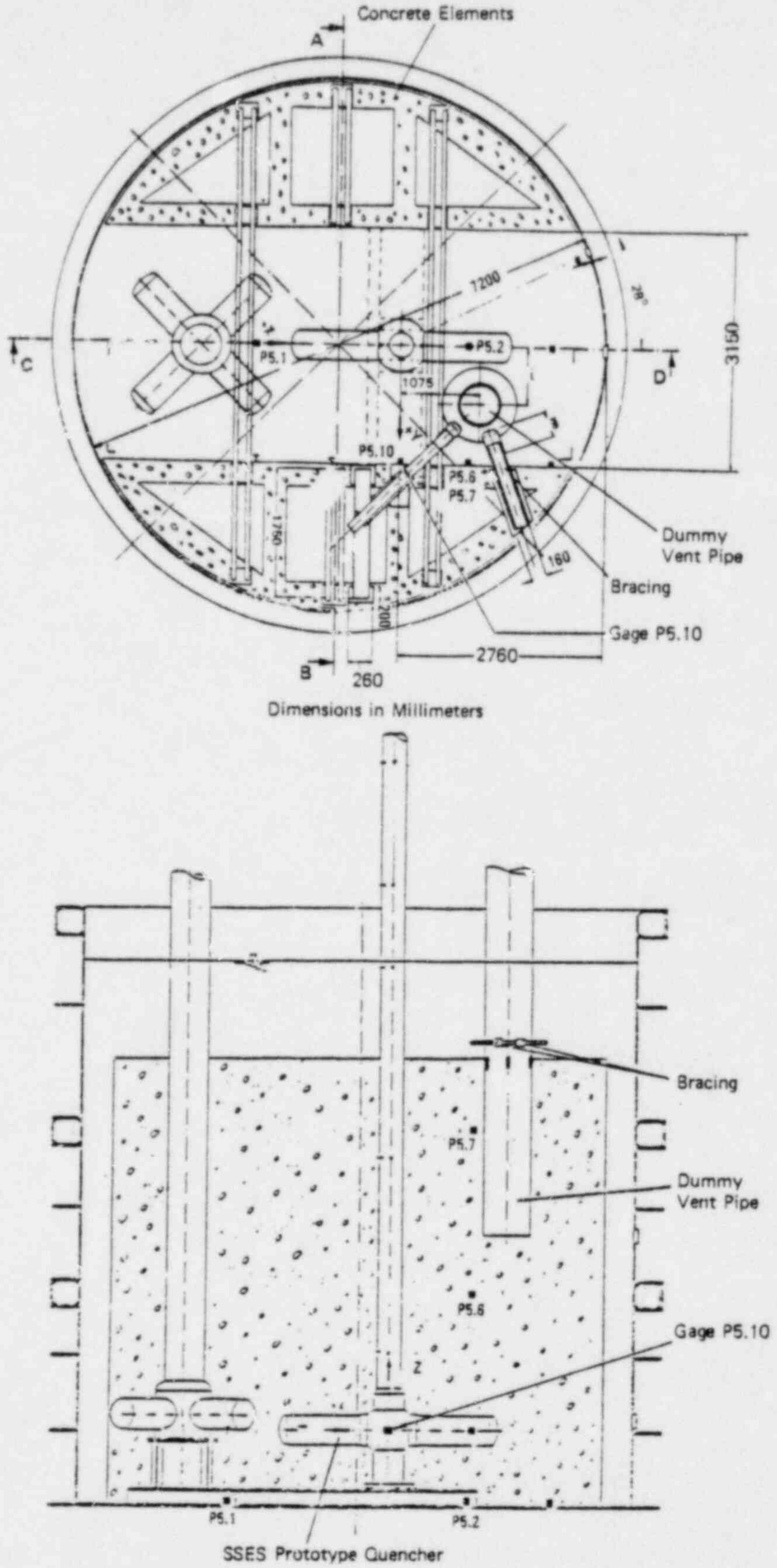
FIGURE 2 QUENCHER DISTRIBUTION



NOTE: All dimensions in mm.

MA-5881-334

FIGURE 3 SSES QUENCHER



MA-5881-114A

FIGURE 4 KARLSTEIN TEST TANK

PP&L-Quencher Verification Tests

-Testmatrix for the Air Clearing Tests-

Test Number		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
Discharge Line	long	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*											*	*	*				*
	short																*	*	*	*	*	*	*	*	*	*				*	*	*	
Discharge Line Air Temperature	50°C	*	*	*	*	*	*	*	*	*	*	*	*	*			*	*	*	*	*	*	*	*	*		*	*	*	*	*	*	*
	90°C														*	*										*							
Water Level in the Discharge Line	normal	*	*	*	*	*	*	*	*	*					*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*
	low										*	*	*	*																			
Vacuum Breaker No.2	unlocked	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*	*		*	*		*	*	
	locked																										*			*			*
Pool Temperature	24°C	*	*	*	*						*				*		*	*	*	*							*	*		*	*		*
	45°C					*						*				*					*								*			*	
	55°C						*	*														*	*			*							
	68°C								*				*											*									
	82°C									*				*											*								
Steam Accumulator Pressure	22.5 bar	*								*				*			*								*								
	45.0 bar		*						*				*					*						*									
	70.0 bar			*				*											*				*										
	80.0 bar				*	*	*				*	*			*	*				*	*	*				*	*	*	*	*	*	*	*
Number of Actuations					7	7									7	7				10	10												
		2	2	2			2	2	2	2	2	2	2	2			2	2	2			2	2	2	2	2	1	1	1	1	1	1	2
Repetition Tests					*											*				*	*					*	omitted*						
Test Group Number		1									2				3		4									5							
time between successive actuations		for 7 : 1.5/5/15/30/60/120s; for 10: 1.5/5/15/30/60/120/5/15/600s; for 2 : ≈ 600s; for 2*: 1[illegible]																															

*These tests were not performed because sufficient data were obtained from the other tests in the matrix.

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FIGURE 5 TEST MATRIX FOR AIR CLEARING TESTS

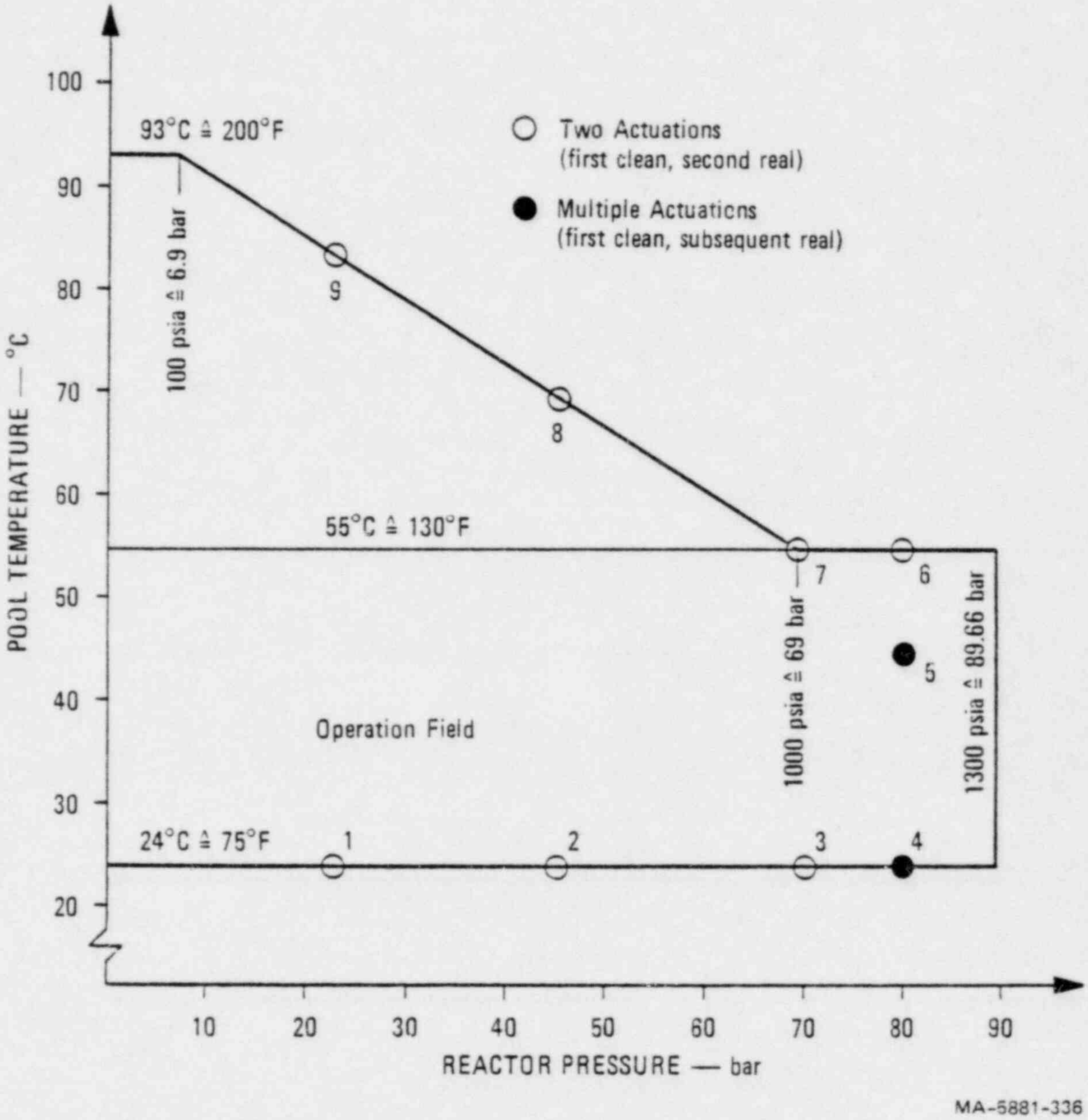


FIGURE 6 LOCATION OF TEST GROUP No. 1 IN THE OPERATION FIELD

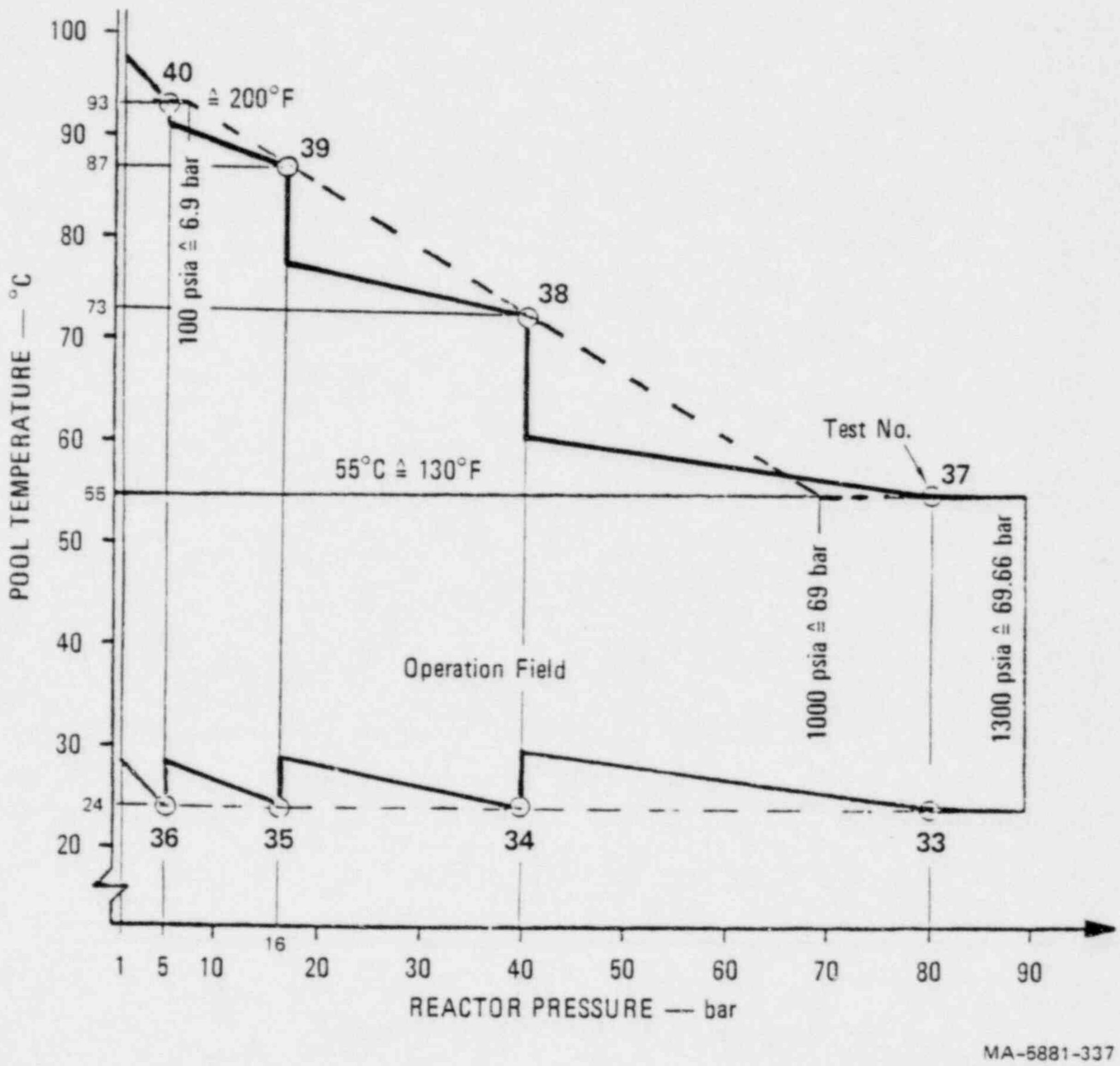


FIGURE 7 LOCATION OF CONDENSATION TESTS IN THE OPERATION FIELD

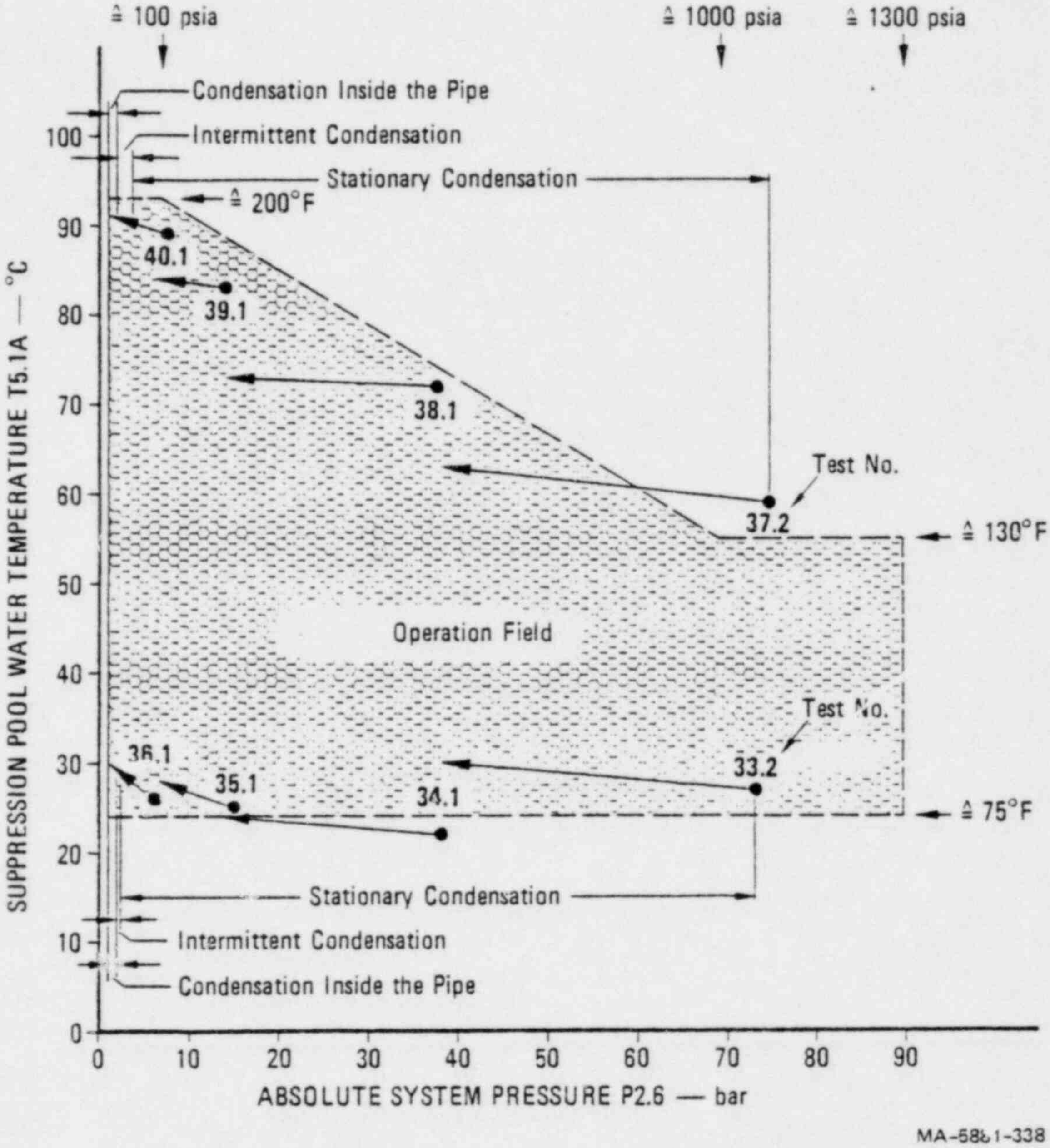
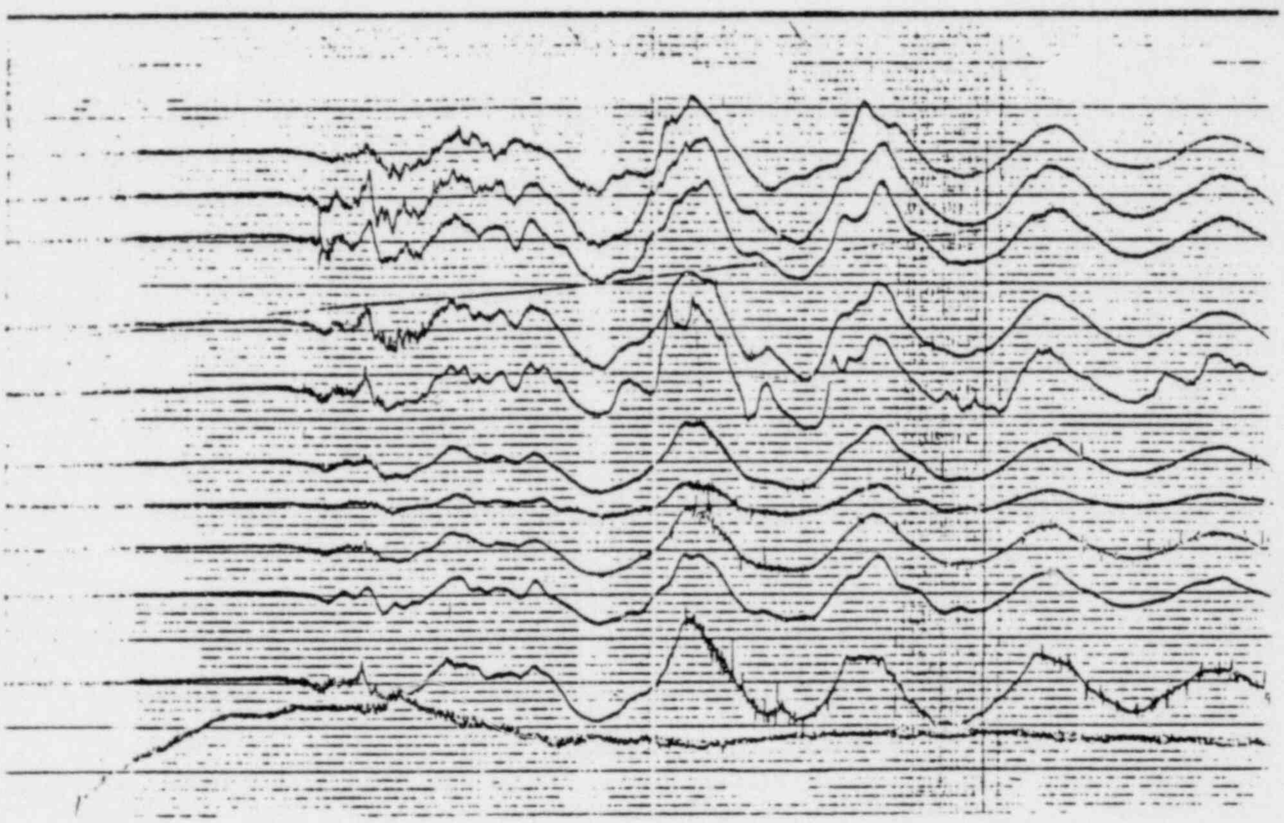


FIGURE 8 OBSERVED CONDENSATION PHASES DURING TESTS



TIME

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FIGURE 9 AIR CLEARING TEST 25.1 VISICORDER TRACE GAGES, P5.1-P5.10

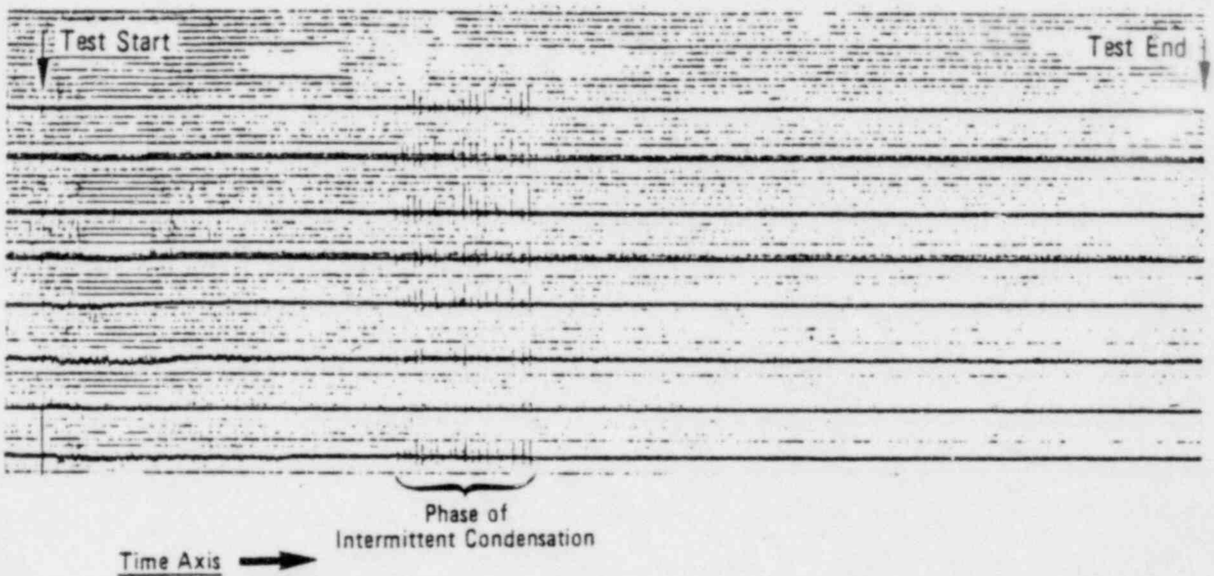

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FIGURE 10a CONDENSATION TEST 36.1 VISICORDER TRACE SHOWING INTERMITTENT OPERATION QUENCHER

Time Axis

MA-5881-341A

FIGURE 10b CONDENSATION TEST 36.1 VISICORDER TRACE SHOWING EXCERPT FROM INTERMITTENT OPERATION OF QUENCHER

280 seconds after start.

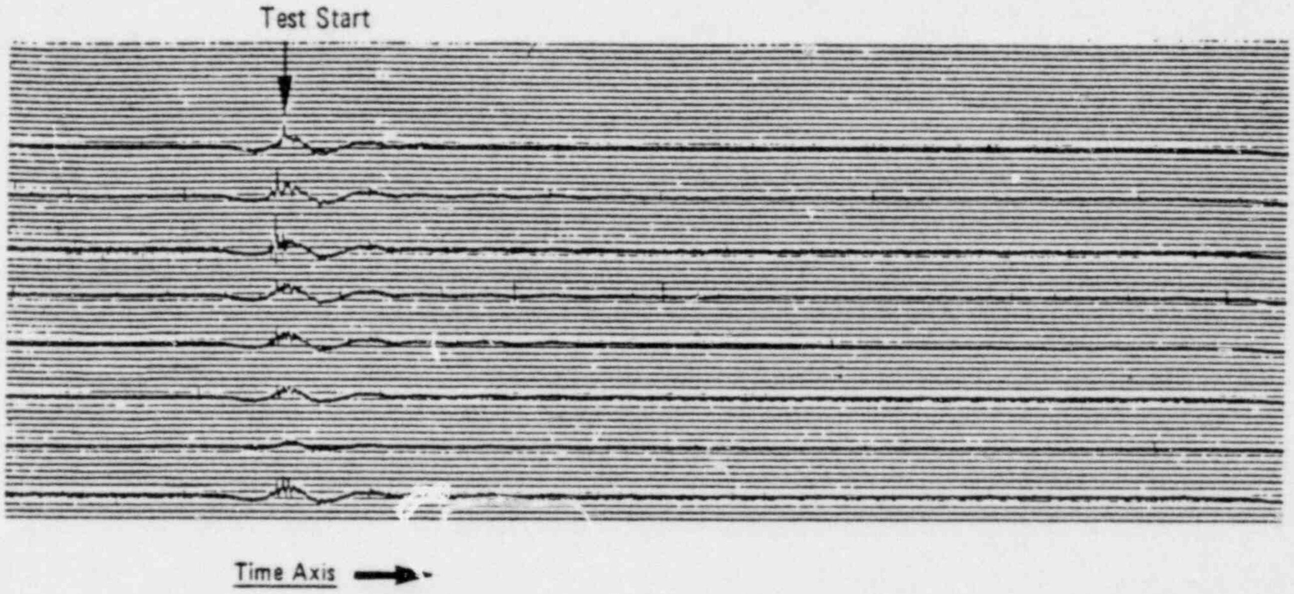

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FIGURE 10c CONDENSATION TEST 36.1 VISICORDER TRACE SHOWING SINGLE EVENT FROM INTERMITTENT OPERATION OF QUENCHER

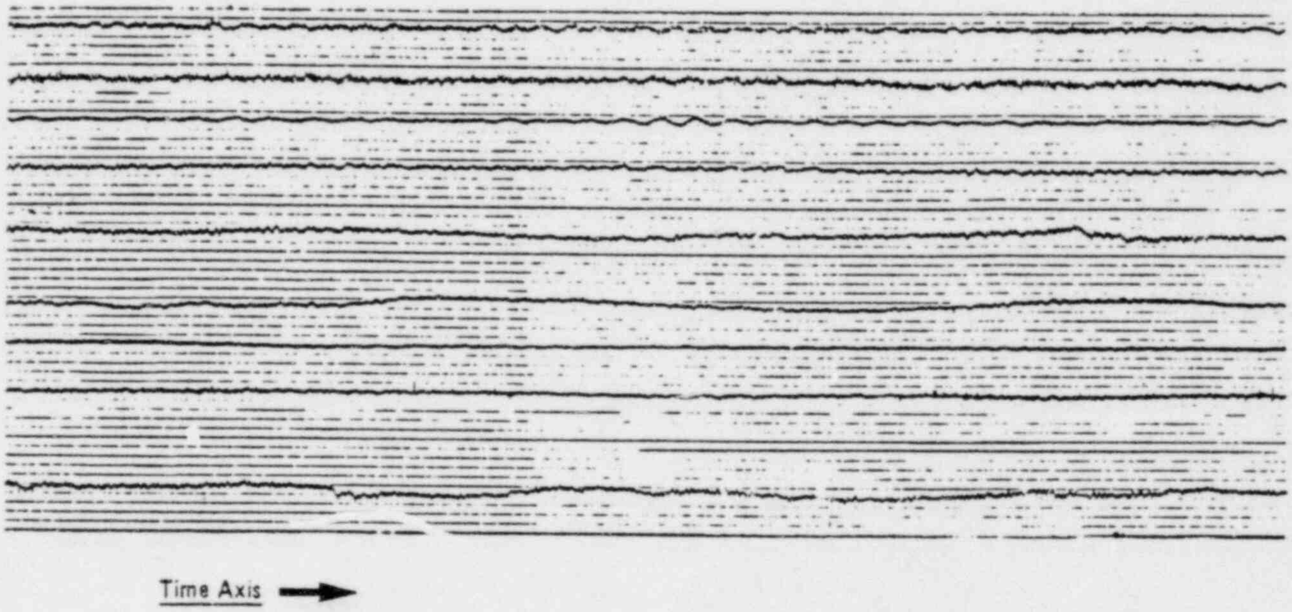

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FIGURE 10d CONDENSATION TEST 37.2 TYPICAL VISICORDER TRACE OF STATIONARY OPERATION OF QUENCHER

13 seconds after start.

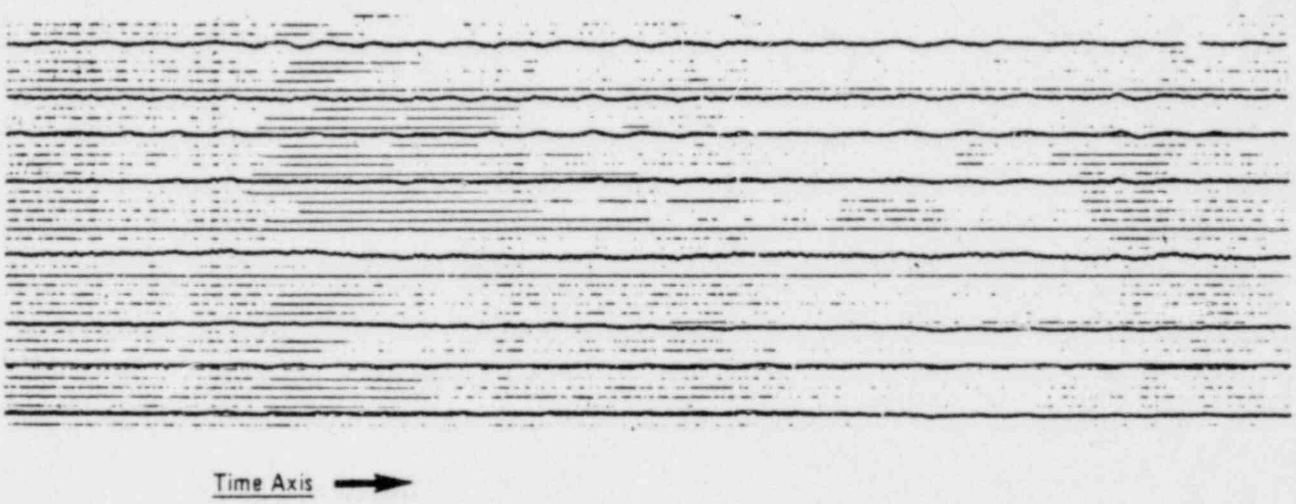

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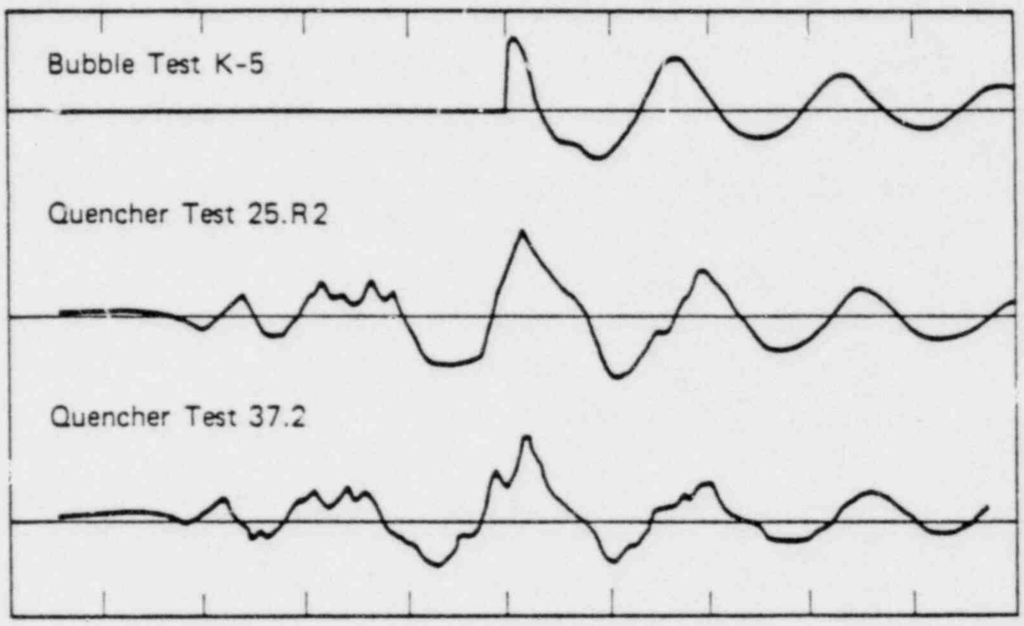
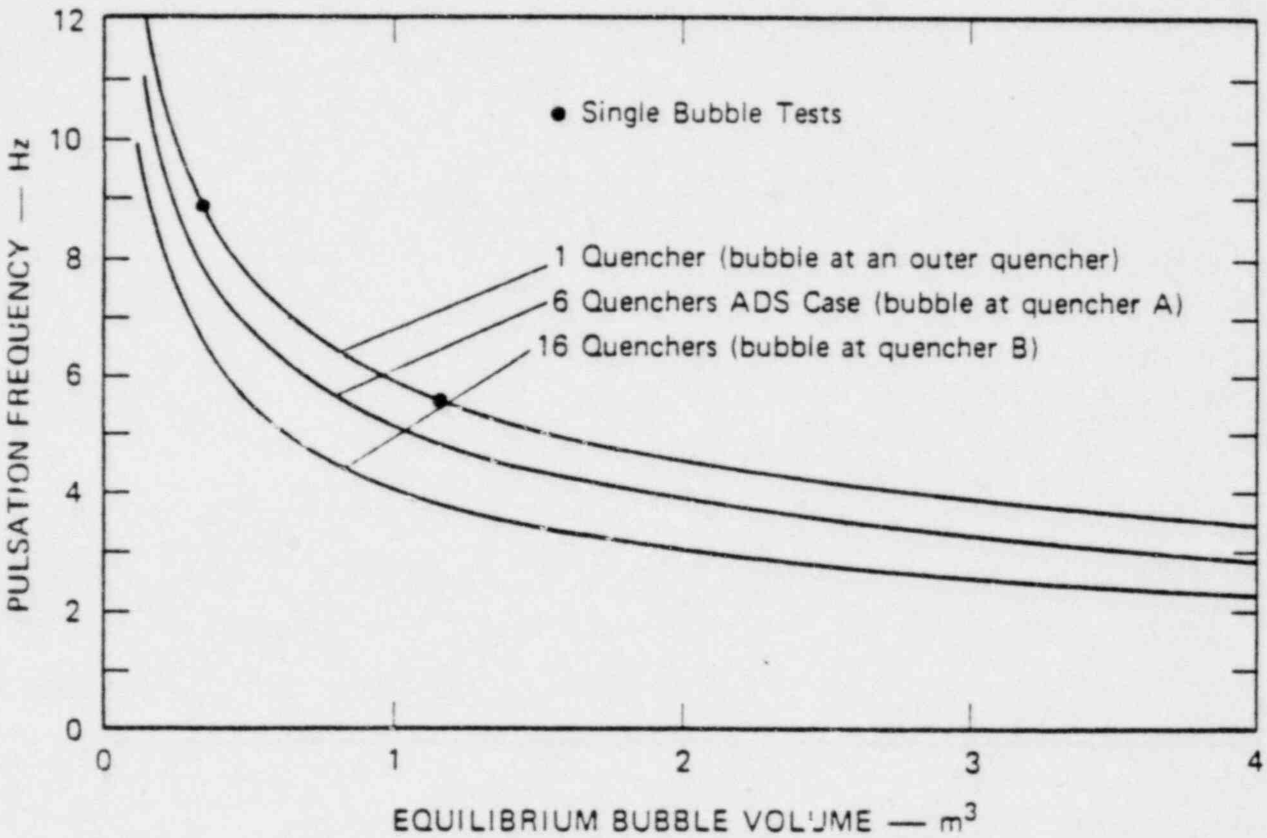
FIGURE 10e CONDENSATION TEST 39.1 TYPICAL VISICORDER TRACE OF STATIONARY OPERATION OF QUENCHER

10 seconds after start.



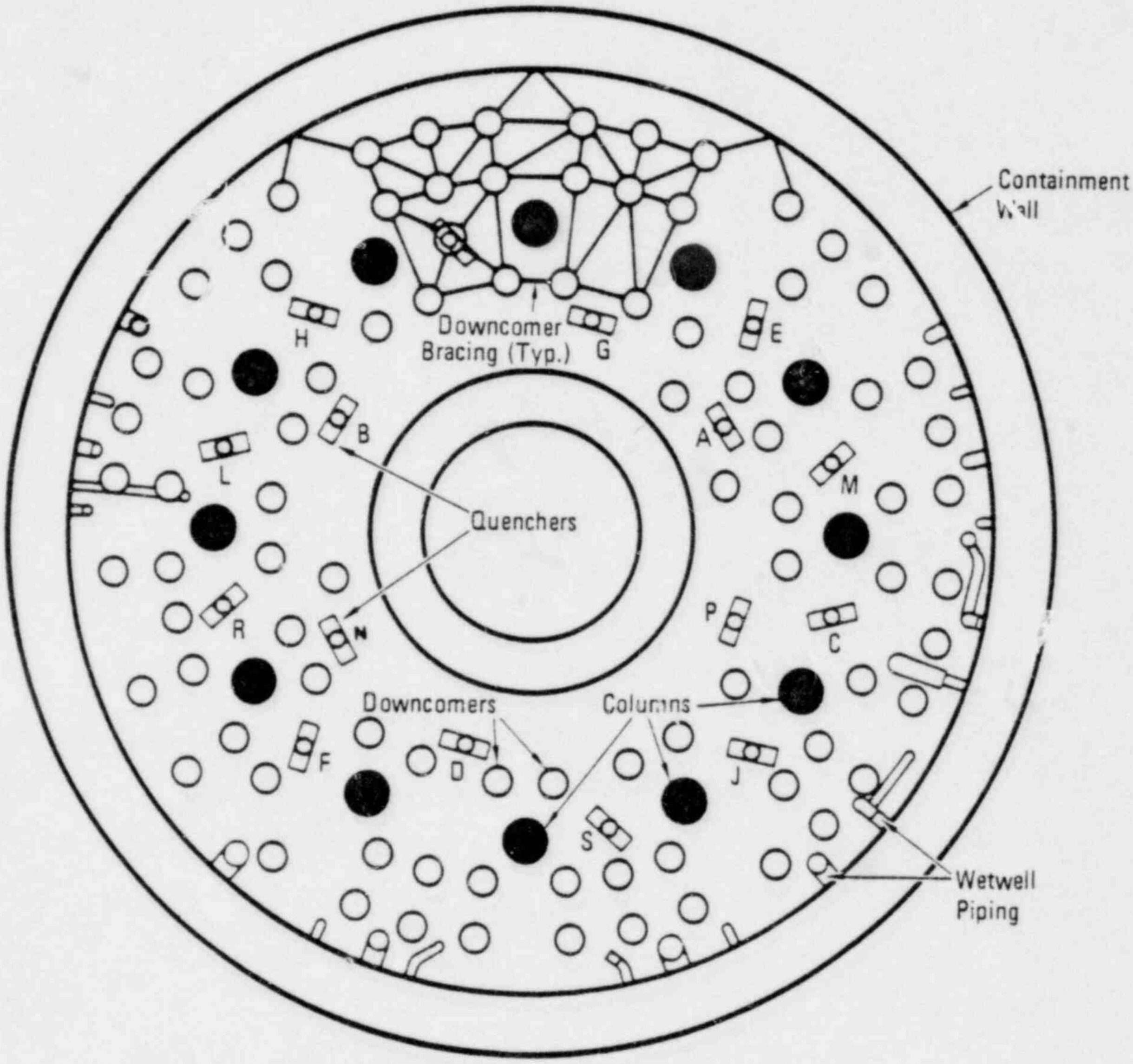
MA-5881-128B

FIGURE 11 COMPARISON OF BUBBLE AND QUENCHER PRESSURE TRACES FROM GAGE P5.10



MA-5621-117B

FIGURE 12 COMPARISON OF MEASURED AND CALCULATED BUBBLE FREQUENCIES



NOTE: Downcomer bracing is only partially shown in the interest of clarity.

Letters indicate SRV quenchers.

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FIGURE 13 CROSS SECTION OF SSES POOL SHOWING DOWNCOMERS

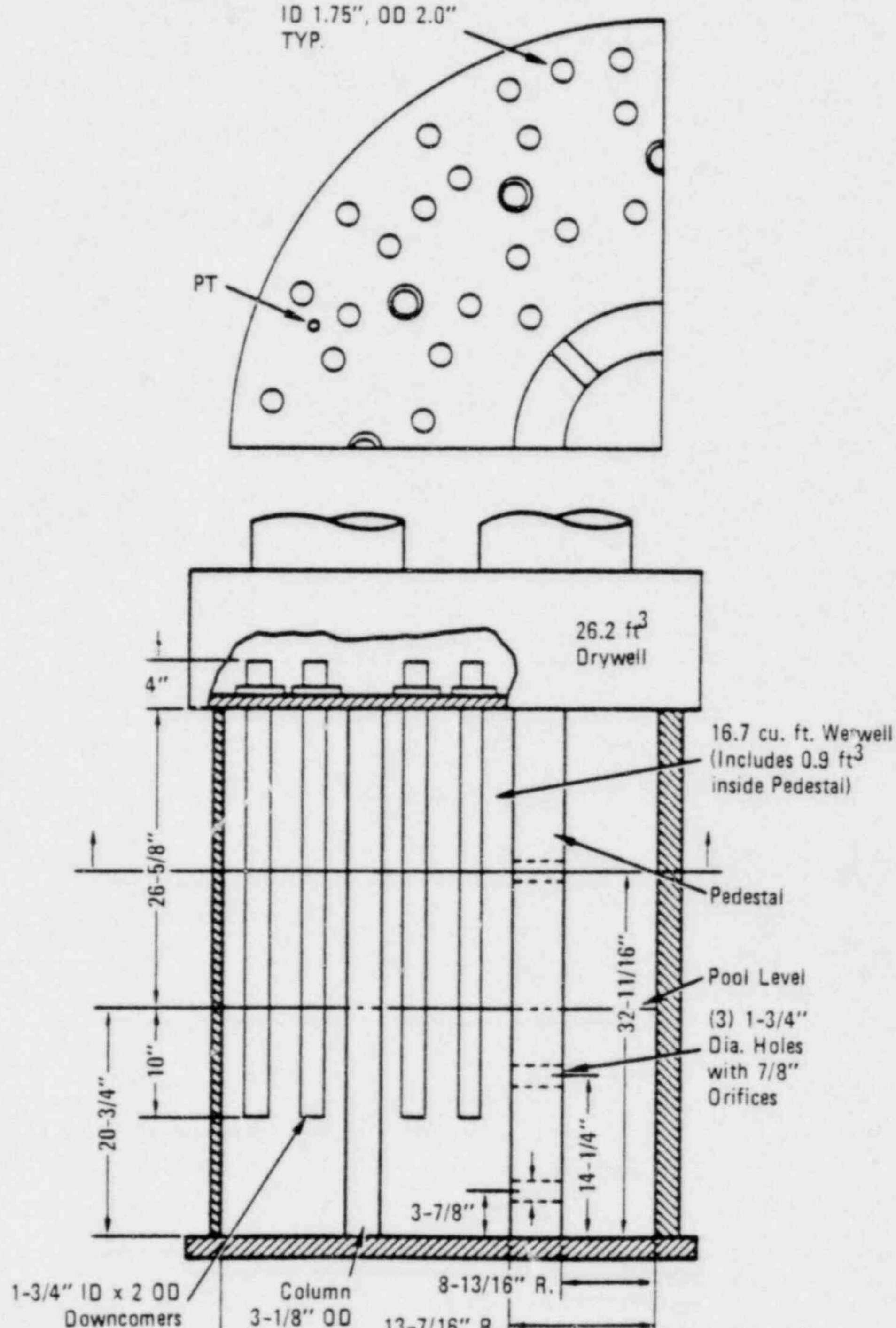
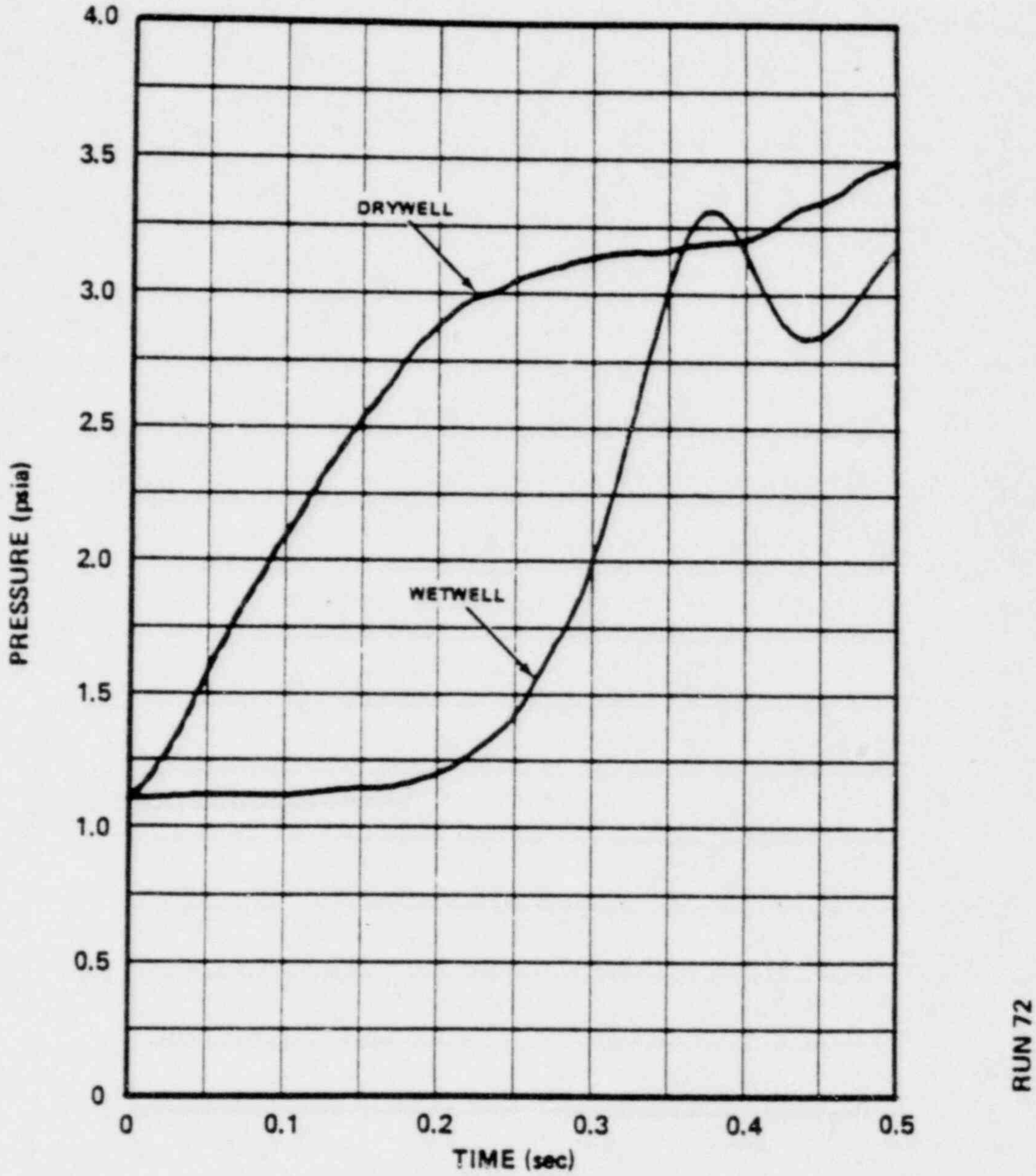


FIGURE 14 1/13.3-SCALE MARK II MODEL



PRESSURE HISTORIES

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FIGURE 15 RESULTS FROM TEST 72

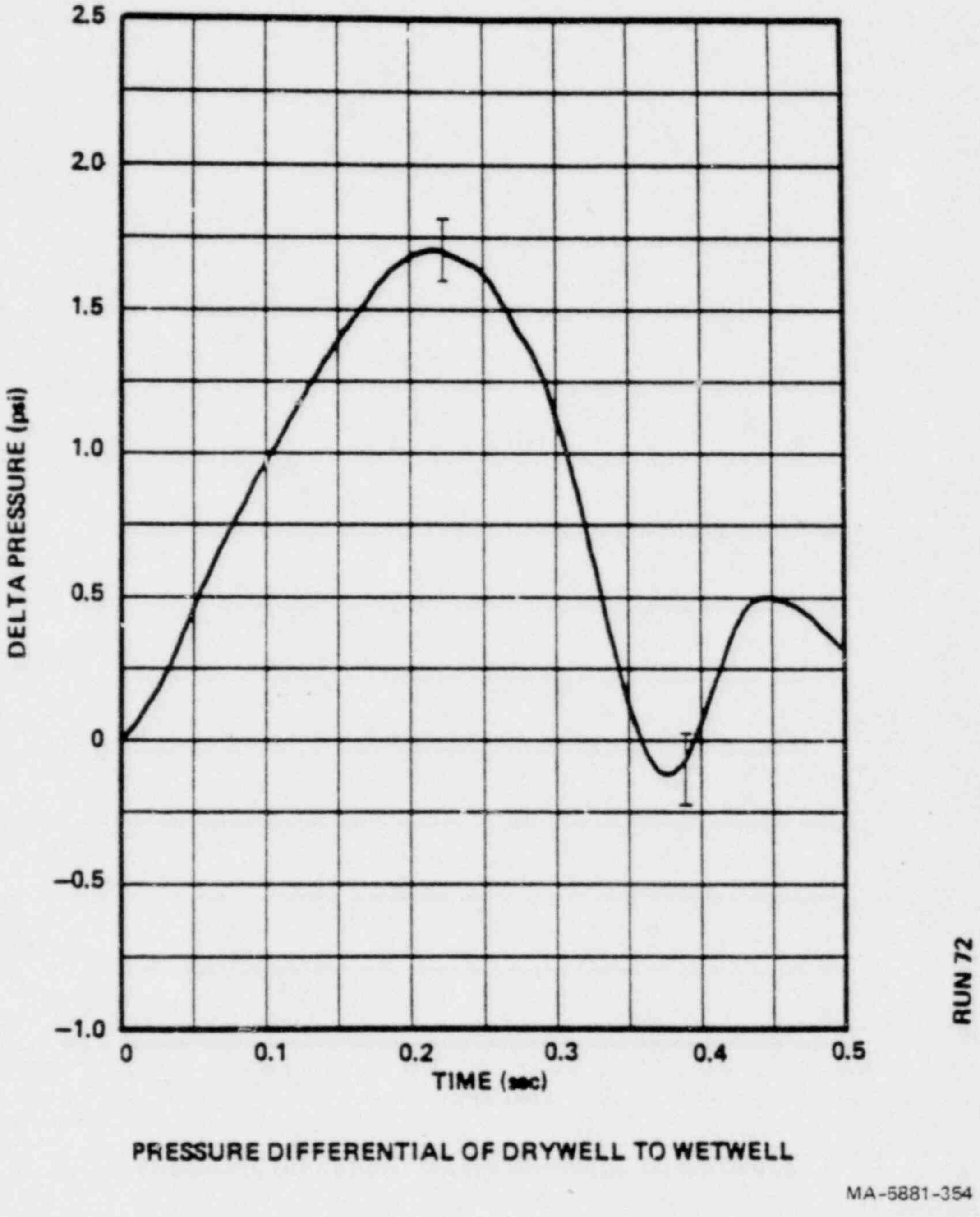

FIGURE 16 RESULTS FROM TEST 72

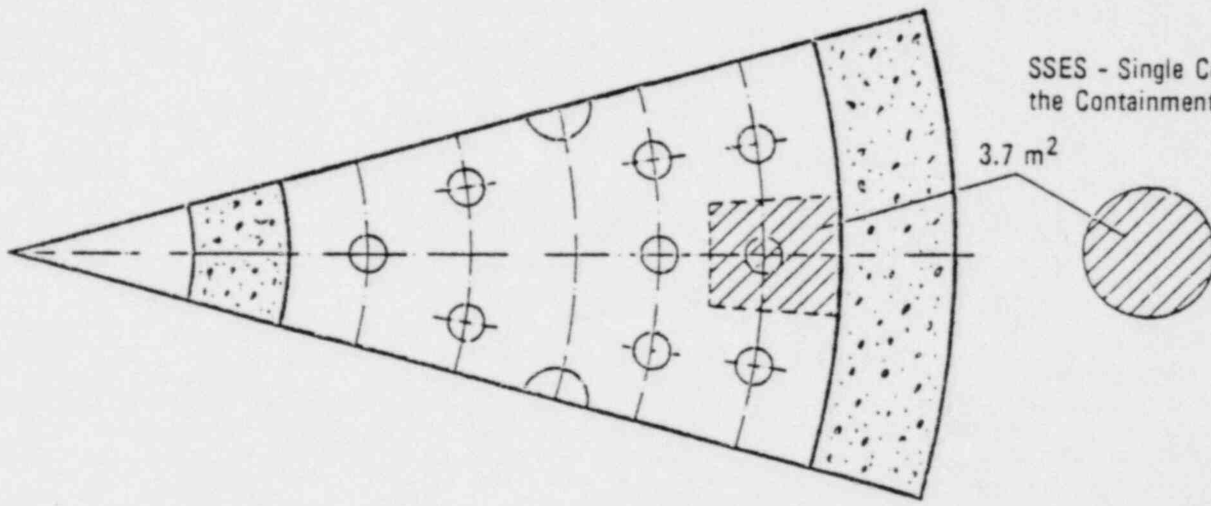

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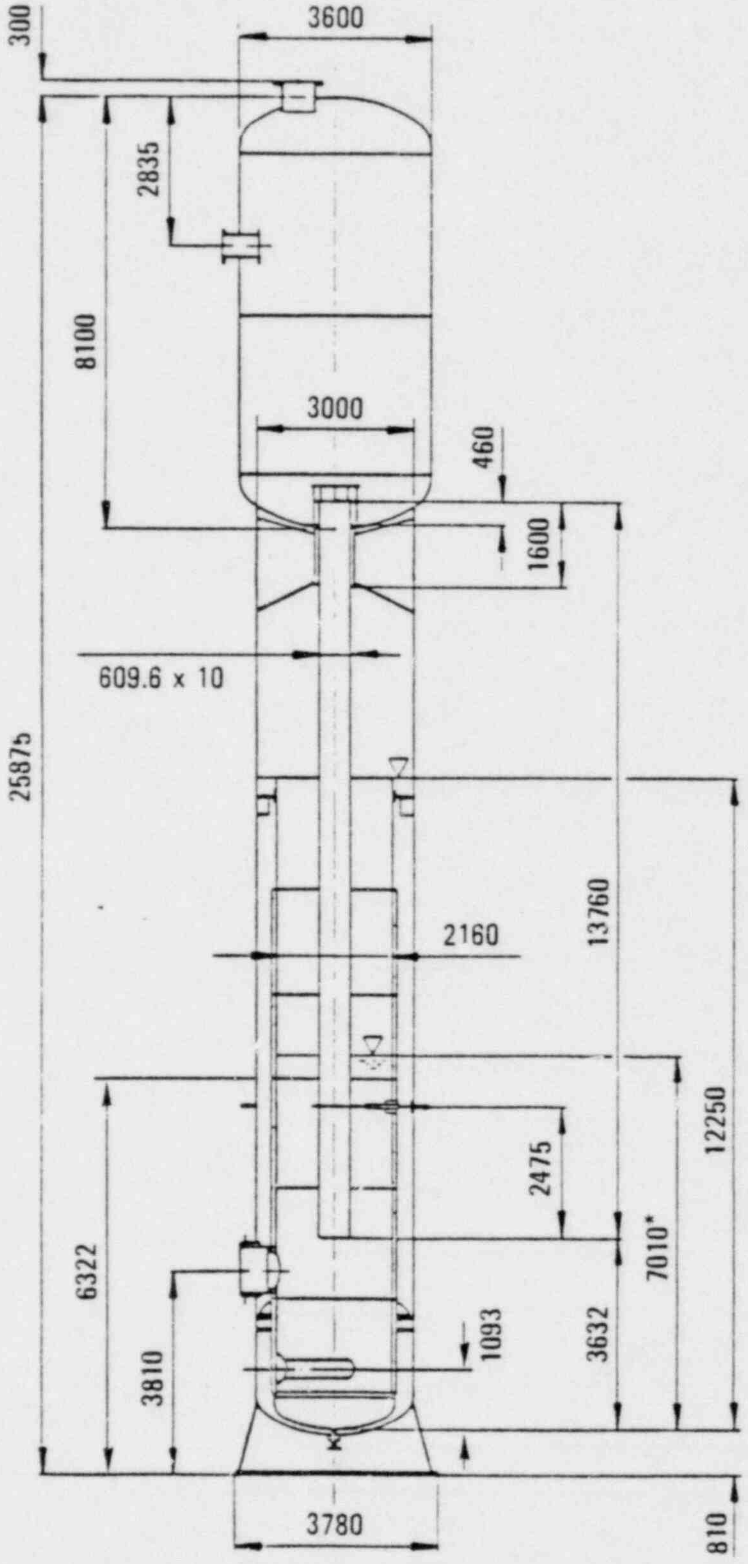
FIGURE 17 ALLOCATION OF POOL AREA TO DOWNCOMERS



NOTE: Dimensions in mm.

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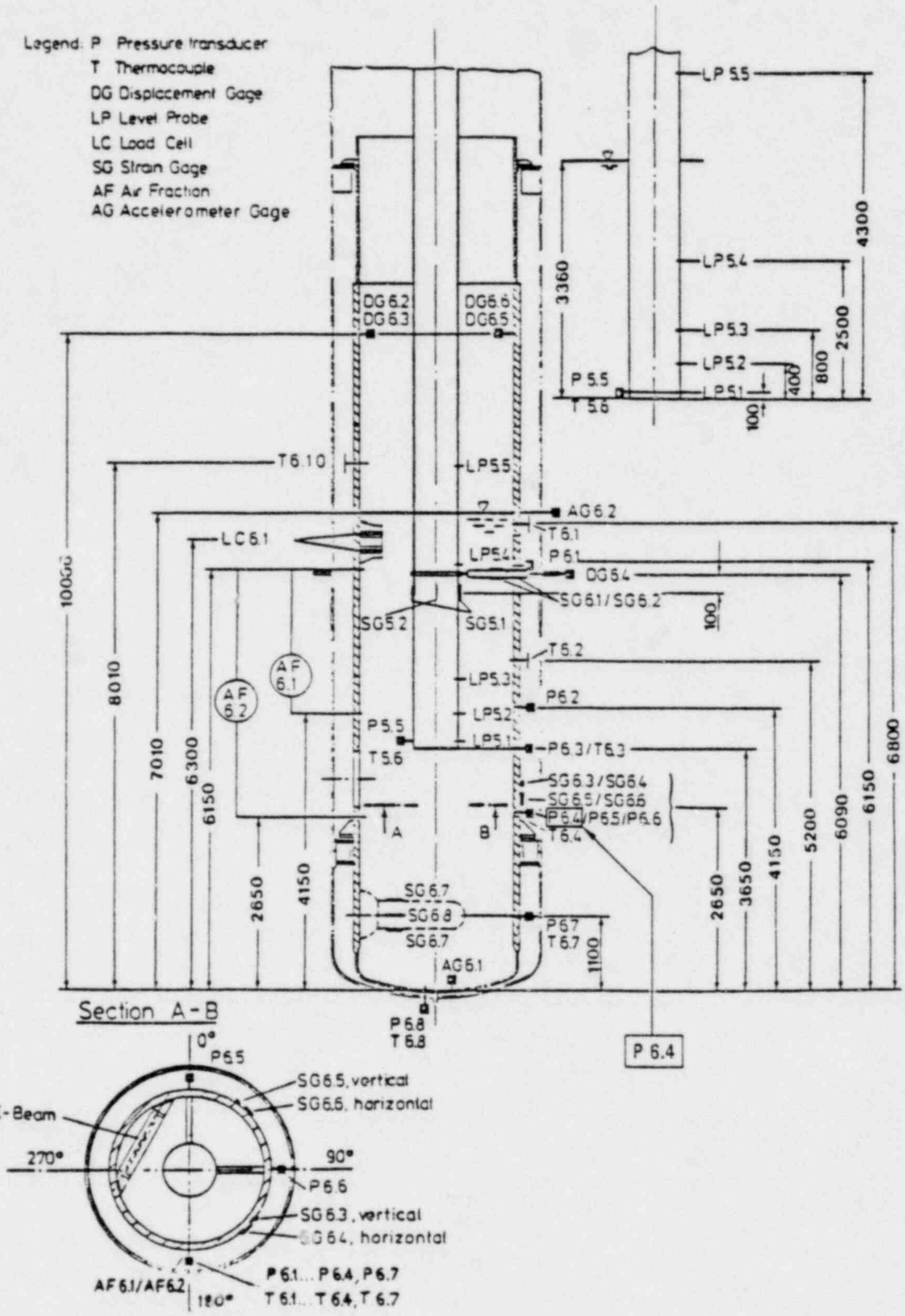
FIGURE 18a SINGLE CELL TEST FACILITY



NOTE: Dimensions in mm.

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FIGURE 18b INSTRUMENTATION

Test Number		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	33	34
Break Size (mm)	Ø 210 (RCL)	•	•																			•	•
	Ø 190 (MSL)			•	•	•	•	•	•	•	•												
	Ø 110 (1/3 MSL)											•	•										
	Ø 80 (1/6 MSL)													•	•	•	•	•	•	•	•		
Pool Temperature	24°C (75 F)			•	•									•	•								
	32°C (90 F)	•	•			•	•	•	•			•	•			•	•	•	•				
	55°C (130 F)									•	•									•	•	•	•
Drywell Air Content	100 %	•	•	•	•	•	•			•	•	•	•	•	•	•	•			•	•	•	•
	85 % (approx.)							•	•									•	•				
Repeat Test			•		•		•		•		•		•		•		•		•		•		•

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FIGURE 19 LOCA TEST MATRIX

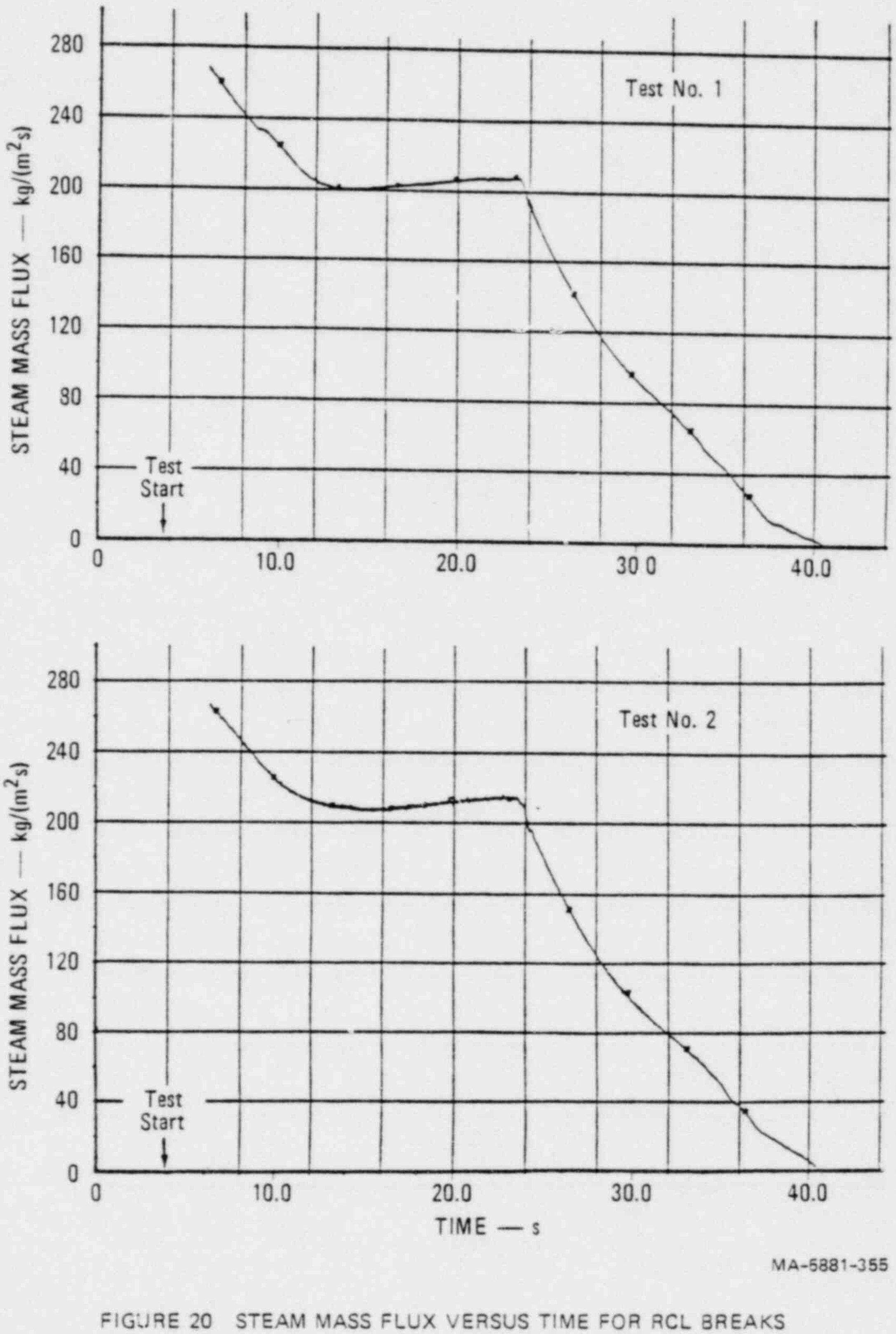


FIGURE 20 STEAM MASS FLUX VERSUS TIME FOR RCL BREAKS

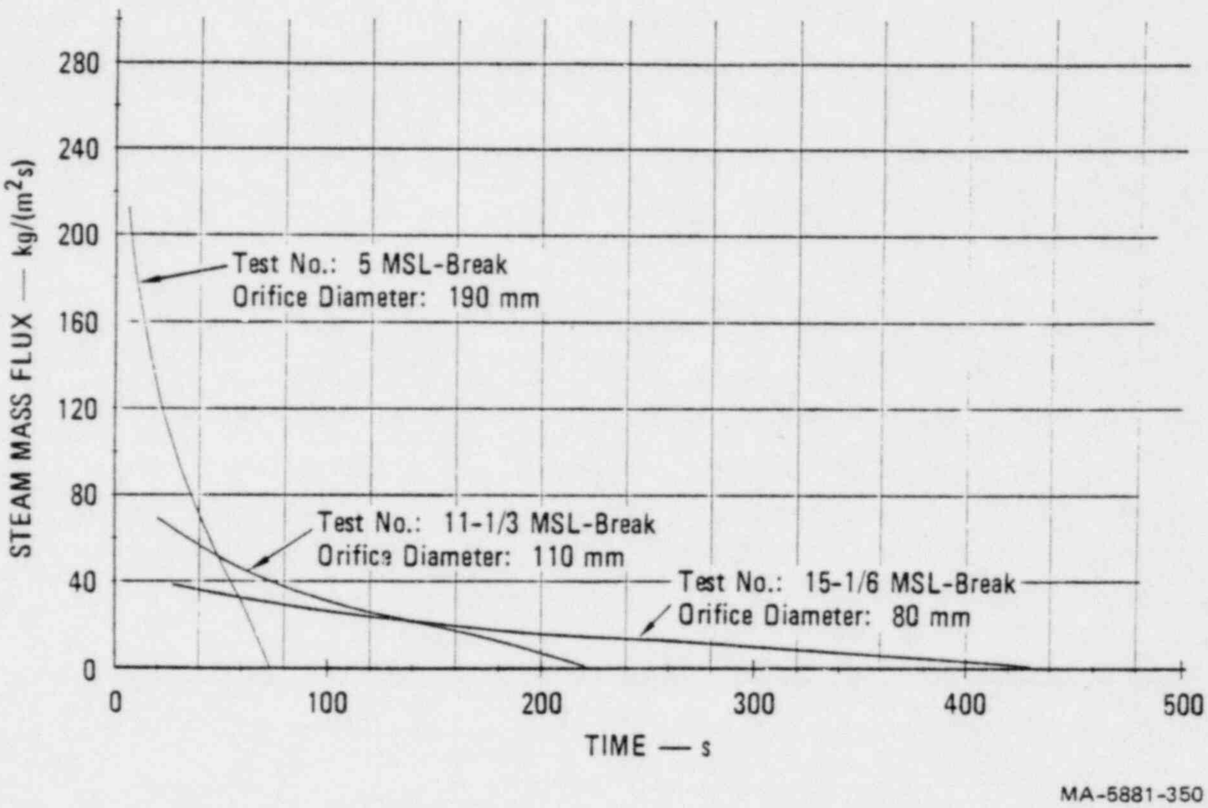
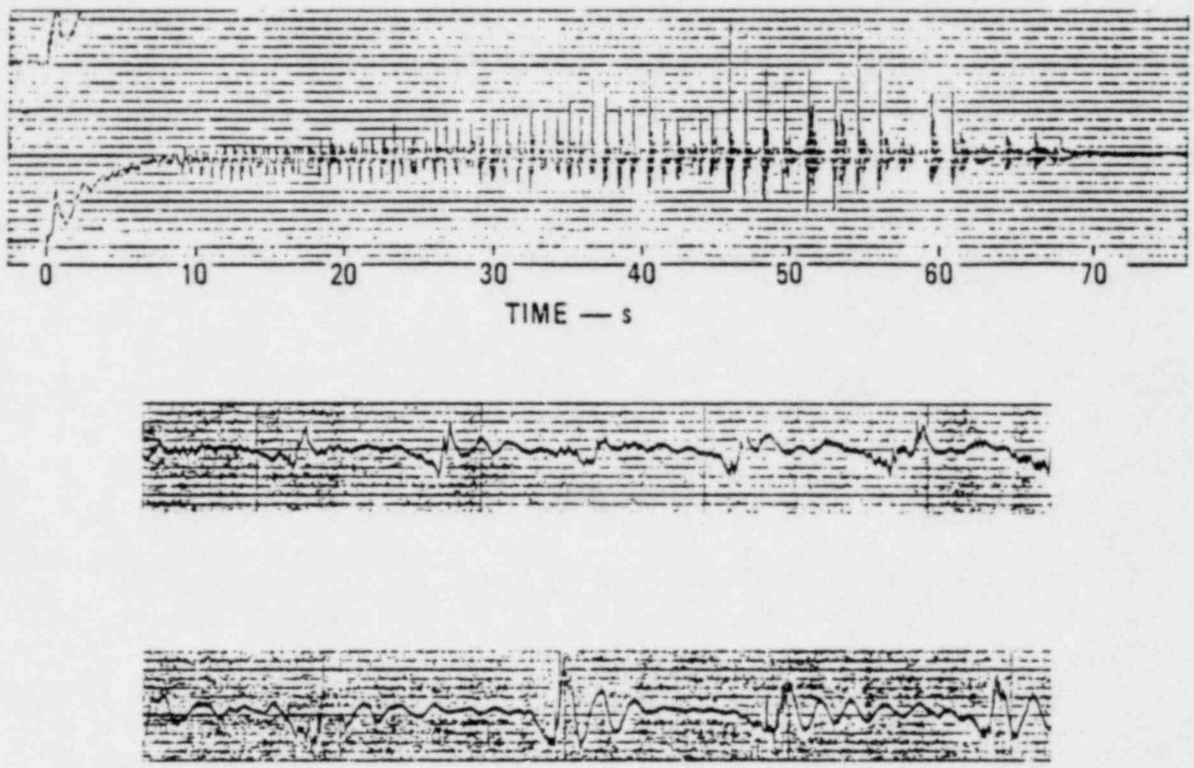


FIGURE 21 STEAM MASS FLUX VERSUS TIME FOR MSL BREAKS



MA-5881-351A

FIGURE 22 PRESSURE TRACES FROM GAGE P6.4 FOR MSL TEST 5

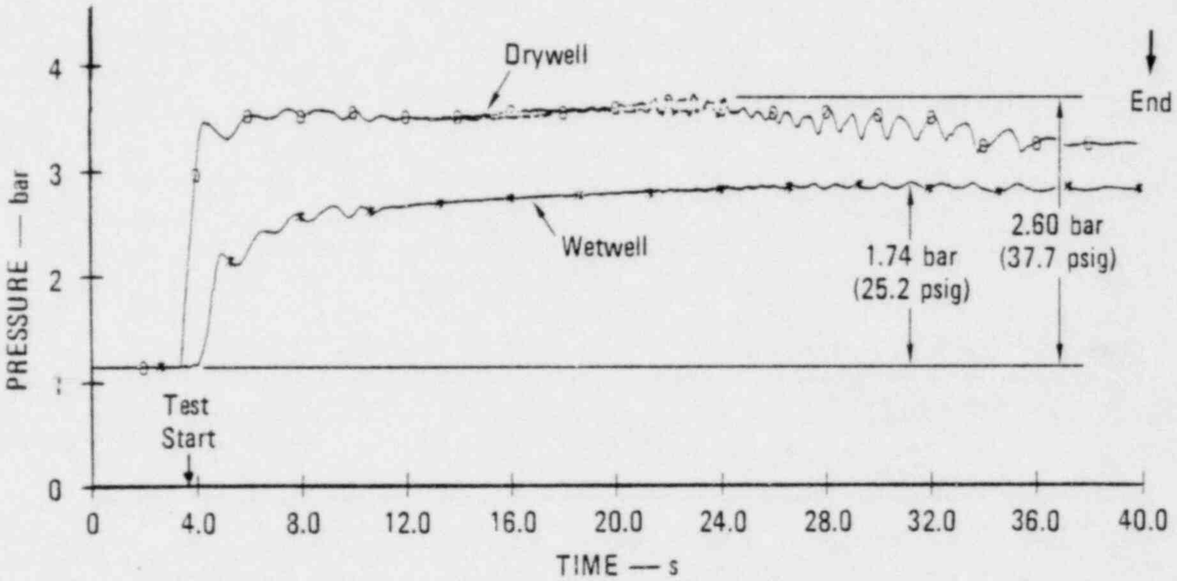


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FIGURE 23 DRYWELL AND WETWELL AIR SPACE PRESSURES FOR RCL TEST 1

EXHIBIT A

GEORGE R. ABRAHAMSON

Vice President
Physical Sciences Division

SPECIALIZED PROFESSIONAL COMPETENCE

Dynamics of structures under pulse loads; nuclear reactor safety; nuclear weapons effects; simulation of pulse loads with explosives; properties of explosives; mechanics of penetration; precision explosives devices

REPRESENTATIVE ASSIGNMENTS AT SRI (since 1953)

Hydrodynamic loads in boiling water reactors
Response of nuclear reactors to sudden pressurization
Assessment of vulnerability and lethality of ICBM reentry vehicles to nuclear effects
Development of techniques for simulating nuclear weapons effects and nuclear reactor excursions with explosives
Experimental and theoretical investigations of the response of structures to pulse loads
Development of shaped charges for oil well perforation
Development of precision explosive timer
Special assignments;
- Panel chairman for Olen Nance ad hoc acommittee for the Materials Advisory Board, National Academy of Science
- Staff development leave from SRI, spent at the Norwegian Research Establishment, Norway (1968-1969)
- Panel chairman for DASA summer study on nuclear weapons effects
- Participant in ARPA summer study on Casaba-Howitzer and laser effects

Manager, Engineering Mechanics Group, Poulter Laboratory (1958-1969)
Director of Poulter Laboratory (1969-present)

ACADEMIC BACKGROUND

A.A., Compton College; B.S., M.S., and Ph.D. (1958) in engineering mechanics, Stanford University

PUBLICATIONS

Fifteen publications in professional technical journals and numerous technical reports

PROFESSIONAL ASSOCIATIONS AND HONORS

American Nuclear Society; American Society of Mechanical Engineers (Western Committee of Applied Mechanics Division, 1970-1974; Chairman, 1974); Phi Beta Kappa; Sigma Xi; Tau Beta Pi; NSF fellowship for advanced study, Stanford University (1955)