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AEC Research and Development Report

CONTAINMENT SYSTEMS EXPERIMENT
FINAL PROGRAM SUMMARY

July 1971

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CONTAINMENT SYSTEMS EXPERIMENT
FINAL PROGRAM SUMMARY

By

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Mechanical Engineering Department

July 1971

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PACIFIC NORTHWEST LABORATORIES
RICHLAND, WASHINGTON 99352

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A. K. Postma and B. M. Johnson

ABSTRACT

A technical summary of the Containment Systems Experiment (CSE) Program is presented. This effort, which was part of the AEC Nuclear Safety program, was initiated in FY-1964 and phased out approximately 6 1/2 years later at the end of FY-1970. It was directed toward obtaining data in support of safety analyses on the transient response of several aspects of the reactor containment system during and following a loss of coolant accident. Emphasis was placed on the study of fission product transport and removal in the containment structure, and on fluid behavior in the reactor vessel during the blowdown event. The scale of these experiments was approximately a one fifth linear scale of a 1000 MWE nuclear reactor, making them the largest scale experiments within a group of AEC sponsored programs carried on at several locations in the country.

The program achieved its objective of providing data from large scale systems against which calculational methods could be compared and consequently improved upon. Data was obtained on methods of detecting and measuring leakage from large containment systems; on fission product removal (following a simulated major release resulting from a loss of coolant accident) via natural deposition, spray action, and recirculation of the containment atmosphere through a cleanup system; and on fluid behavior in the reactor vessel during blowdown. The applicabilities of several mathematical models for calculating the consequences of a loss-of-coolant accident were demonstrated while the weaknesses of several others were pointed out.

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CONTAINMENT SYSTEMS EXPERIMENT
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INTRODUCTION

This report is the final summary report of the Containment Systems Experiment (CSE) program. The intent of the report is to give a concise technical overview of the significant results of the program. For detailed treatment of these results and the possible impact they have on the licensing of nuclear reactors, one should read the applicable topical reports as referenced in the document.

The safety assessment of nuclear power generating stations is unique from that of conventional plants because of the radioactive nature of the fuel, which necessitates thorough analyses (based on hopefully very conservative assumptions) of the short and long term containment of the radioactivity. The allowable limits of escape of radioactivity from the plant site boundary has been set by the AEC Division of Reactor Licensing (DRL). The utility company desiring to build and/or operate a reactor must submit a preliminary safety analysis to DRL which reviews the report and requires the submission of appropriate information concerning the public safety of the reactor. Following this, there are reviews by the Advisory Committee on Reactor Safeguards (ACRS) and the Atomic Safety Licensing Board, who have the power to grant a construction permit.

The CSE program was directed toward obtaining data in support of these safety analyses on the transient response of several aspects of the overall containment system during the postulated accident, in equipment of sufficiently large size so as to make extrapolation to full scale reasonably valid. Initially the postulated maximum credible accident involved

loss of coolant to the reactor, meltdown of a significant portion of the core, release of an assumed fraction of the fission product inventory to the containment system, and removal of these fission products from the containment atmosphere via natural processes only. With the advent of the very large reactor units presently being built or planned, it has been shown that reliance must be placed on engineered safeguards to satisfy safety criteria. Consequently, the emphasis in the CSE program, and other programs within the AEC water reactor safety program, was shifted to the evaluation of the ability of calculational methods to predict the behavior of engineered safety systems. This resulted in a change in priorities of the program. Initially the CSE program was directed primarily at evaluating fission product behavior under near prototypic accident conditions including blowdown of a simulated reactor vessel into containment. Subsequent redirection led to the program being structured around parametric studies of fission product behavior under condition which were basically similar to accident conditions, but somewhat simpler to mathematically model so as to allow more precise evaluation and upgrading of mathematical approaches. For example, the reactor simulator vessel was used to study extensively fluid behavior in the reactor system during blowdown in support of setting emergency core cooling requirements, with no initial emphasis on the effect of blowdown on the pressure-temperature transient within the containment system. Fission product transport in the containment vessel was studied under steady state conditions in many runs, rather than under the postulated pressure-temperature transients, to simplify mathematical modeling and permit isolation of possible weaknesses in the assumptions used in the models.

The work in CSE was complimentary to other AEC nuclear safety programs at other sites. For example, the work at the

nuclear safety pilot plant (NSPP) at Oak Ridge, a smaller shielded vessel with approximately 1/22 the volume of the CSE vessel, has emphasized studies of fission product removal by sprays containing reactive additives and the development of mathematical models which predict the fission product behavior. Basic information on the behavior of iodine in containment atmospheres in a variety of chemical forms has been carried on at the Harvard Air Cleaning Laboratory. The Battelle Columbus Laboratories studied the removal of fission products by reactive paint coatings and evaluated the applicability of a wide variety of mathematical models to reactor accident situations. Finally, the Loss of Fluids Test (LOFT) program at the National Reactor Testing Station at Idaho Falls is directed toward final integrated testing of an actual reactor during a loss of coolant accident. In support of these integrated tests they have also carried on extensive programs on a smaller scale directed toward fluid behavior in the reactor system during the accident event. In this overall AEC water reactor safety program, the CSE provided the largest facility, short of the LOFT integrated test test facility, for evaluating loss of coolant behavior, and provided the information many years before the LOFT tests are to be carried out.

The scope of the CSE program included evaluating the effectiveness of selected engineered safeguard systems in limiting fission product release to the environs under conditions representative of maximum credible accidents for pressurized and boiling water reactors. These evaluations were directed toward the following major objectives.

- Determination of the effects of the natural processes of agglomeration, deposition, and condensation of steam in attenuating airborne fission product concentration in the containment vessel.

- Evaluation of the effectiveness of active and passive safeguards including spray systems, and air filter systems, for fission product removal from the containment atmosphere as well as for reduction of postaccident containment pressure.
- Investigation of the dynamic effects of large ruptures in high temperature water piping systems as they relate to containment system design and performance and to evaluation of the differential pressures and mechanical loads exerted on reactor vessel internal structures during coolant blowdown.
- Determination of the relations between air leakage rate tests on containment vessels and the escape of fission products under post-accident conditions for several types of representative leakage paths.

The experimental program was divided into the four major tasks.

Task A. Containment Leakage Studies

Performance of leakage rate experiments with the CSE containment vessel using air, steam-air mixtures, and steam-air fission product mixtures at various pressures and temperatures.

Task B. Fission Product Transport Studies

Performance of fission product transport studies by releasing test aerosols directly into containment vessels to permit studies of the effectiveness of steam condensation, spray systems, air filter systems, and pool scrubbing systems under a range of possible accident conditions.

Task C. Coolant Blowdown Studies

Performance of coolant blowdown experiments using a large reactor simulator vessel to study the effects of coolant conditions, break size, and location on the blowdown rates and to determine hydraulic effects and forces within representative core components.

Task D. Combined Systems Experiments

System performance tests of containment was to have been conducted combining blowdown, fission product transport, and leakage. These tests would have started with the sudden release of high-pressure, high-temperature water followed by continued heating to simulate decay heat and chemical reactions in the reactor core. Aerosols representing the fission products evolved from a core during an accident would have been injected into the system and their behavior and ultimate locations determined by analyses of appropriate samples. Stresses, pressures, and temperatures which occur throughout the containment vessel complex would have been measured by appropriate fast response instruments. Comparison of the experimental results with mathematical models, to allow refinement of analytical descriptions of the consequences of reactor accidents would have been made.

The CSE program was initially conceived at the beginning of FY-64 and the experimental program was cut off and the facilities shut down at the end of FY-70. As a consequence of this premature termination, Task D was not conducted and portions of the other tasks were not completed. Specifically, no studies of pool scrubbing were conducted in the CSE vessel as part of Task B and no blowdown studies were made in the reactor simulator vessel with BWR internal configurations.

SUMMARY

Although the program was closed out earlier than initially anticipated, the main objective of the CSE program was accomplished, namely, the obtaining of experimental information against which calculational models can be tested and from which improved models can be developed, thereby permitting improved methods of predicting events following a reactor incident. The results are being used to furnish better technical basis for reactor site evaluation and for those parts of the reactor plant design which involve public safety.

Specific conclusions that are supported by the results of the CSE tests are the following.

- Measurement errors limit the detection of leak rates by either the "absolute" or "reference vessel" methods to greater than 0.1 percent per day. Both of these methods give equivalent results. None of the standard techniques can measure leakage of steam-air atmospheres.
- Leakage of atmosphere from containment is accompanied by appreciable decontamination of reactor fission product gases and aerosol particles (particularly elemental iodine) as a result of condensation and deposition in the leakage paths.
- Removal of inorganic iodine by natural convection will result in a 2-hour dose reduction factor (DRF) for inorganic iodine in a typical large PWR of about 2.6. Methyl iodide is removed much more slowly, equivalent to an overall decontamination factor of 1.7 after one day.

- Particulate cesium and uranium are removed from the gas space primarily by gravitational settling. The resulting 2-hour DRF for all particulate aerosols in a typical large PWR is estimated to be about 1.5. After 1 day, it would increase to about 3.
- Mixing within the containment after an accident is probably such that the concentrations of all fission product will be essentially uniform throughout the gas space of a single compartment but large differences between compartments will persist for several hours where no forced convection is provided.
- The removal of elemental iodine by sprays can be predicted via a mathematical model which assumes the limiting resistance to mass transfer is the gas film surrounding the liquid droplet. The iodine in the liquid is held in a non-volatile form and is not re-evolved when the water is recirculated for 20 hr. at 250 °F. The 2-hr DRF for elemental iodine for a large PWR is about 50 using a basic borate spray with $\text{PH} \sim 10$.
- Removal of methyl iodide by a hot basic borate-sodium thiosulfate spray agreed well with prediction by a model based on chemical reaction rate limiting process. The removal rate is temperature sensitive and the wall film is more effective than the drops. The 2-hour DRF for methyl iodide would be about 1.5 in a large PWR.
- Aerosol particles are effectively removed by sprays because of particle agglomeration and water adsorption. A conservative estimate of the 2-hour DRF is 12 for a large PWR.

- Removal of iodine by air recirculating systems is closely approximated by a simple mathematical model which assumes perfect mixing in the containment vessel. Mechanical problems of operating an air cleaning loop at accident conditions (with respect to pressure and temperature) are not of major importance.
- The FLASH2 code predicts the pressure transient observed in the blowdown event if the proper values of three adjustable parameters (the discharge coefficient, the bubble rise velocity, and the initial quality gradient) are chosen. No systematic way of predicting these parameters based on the initial conditions of the blowdown could be developed. The discharge coefficient for the simple vessel tests could be empirically correlated with the ratio of the break size to the vessel cross sectional area but this correlation did not hold for blowdowns from the vessel with internal restrictions. Fortunately the deviations from theory are normally in a conservative direction, i.e., the blowdowns are slower than theoretically predicted.
- Liquid level during the blowdown again could be predicted by the FLASH2 code provided the proper adjustable parameters were used. However, there was some discrepancy between the best set of parameters to fit the pressure history and those needed to fit the liquid level data. The predicted level of residual water remaining after the blowdown was always higher than that observed experimentally.

- The ability to predict the pressure history at various vessel locations during the subcooled portion of the blowdown is dependent on first matching the pressure history at the outlet pipe. Once this has been done, the WHAM code approximates the pressure wave behavior in the rest of the system reasonably well. Multidimensional effects were not prevalent and the one dimension WHAM code appeared to be adequate once the proper initial conditions were obtained.
- The existence of a slightly soluble gas in the blowdown fluid can markedly reduce the violent decompression wave.
- Forces on the internal structures within the reactor simulator vessel were not predicted closely by the WHAM. The calculated stress values were from 2 to 5 times higher than those observed.
- The thrust forces generated by the blowdown could be approximated from a conservation of momentum equation. The coefficient ranged from 0.5 to 1.0 and were generally higher for smaller breaks.

DESCRIPTION OF THE CSE PROGRAM

OBJECTIVES OF THE CSE PROGRAM

The objective of the CSE program was to investigate the behavior of containment systems and related engineered safety systems under maximum credible accidents for pressurized and boiling water reactors. Specific objectives defined at the time the program was formulated are as follows:

- Determine the rate at which airborne fission products are removed from containment atmospheres by natural processes.
- Measure the effectiveness of sprays and filter systems in removing airborne fission products from containment atmosphere.
- Measure the leakage rate of fission products and compare with that calculated from low temperature leak tests.
- Evaluate pressure suppression by water pools and cold sprays.
- Measure pressure, temperature, and force transients resulting from the sudden rupture of a high temperature water system and compare with predictions.

At the time the CSE program was in the formulation process (1963-1964), the maximum credible accidents assumed for accident assessment involved gross loss of cooling with subsequent melting of a large part of the core. Under these ground rules, the worst accident involves gross release⁽¹⁾ of fission products to the containment atmosphere with only natural removal processes operating. Thus natural removal processes which reduce the concentration of fission products suspended in a leakable state was highly important in the program initially formulated.

The total failure of cooling systems has since been shown to lead to conditions which are not amenable to reliable analysis. Hence major emphasis has been placed on design of emergency cooling systems which contain enough redundant features to insure that a large part of the core will not melt. These emergency systems include sprays or heat exchangers for removing heat from the containment vessel atmosphere, hence no reactors are being built in which natural transport would be the only mechanism for removing airborne fission products. For this reason, emphasis in the CSE program was shifted from the natural deposition of fission products to their removal by sprays and filters.

As initially designed, the reactor simulator (primary) vessel was to have been installed within the containment vessel before experiments were begun. However, in the interest of accelerating the program schedule, the decision was made to proceed with experiments prior to installation of the primary vessel and to carry out a detailed independent program of studies of the blowdown event with the primary vessel. Fission product transport and leakage experiments were carried out using boiler house steam to pressurize the containment vessel. Simultaneously, blowdown experiments were carried out using the primary vessel housed in a separate facility. A final task of the program involved a combined systems study in which the primary vessel was to be installed within the containment vessel. Experiments carried out under this final task were to provide information on transient heat transfer and mass transfer, along with mechanical force transients which would accompany the blowdown. Funding limitations forced cancellation of these combined system tests and of tests involving fission product transport in a pressure suppression pool modelled after a BWR containment system.

SIZE SCALE OF CSE FACILITY

The CSE facility was sized to represent a scale factor of approximately one-fifth compared to a 1000 MW_e water-cooled civilian power plant. This size scale was selected by the Division of Reactor Development and Technology. The CSE size scale is intermediate compared to the smaller Nuclear Safety Pilot Plant at Oak Ridge and the larger Loss-of-Fluids-Test (LOFT) Facility at the National Reactor Test Station.

Size scale comparisons for the several experimental containment vessels and a containment vessel for a full scale PWR are shown in Figure 1. The primary vessel sizes for several experimental facilities are compared with the primary vessel used in a 1000 MW_e PWR plant in Figure 2.

FIGURE 1. Size Comparison of Experimental Containment Vessels with Typical Power Reactors

FIGURE 2. Comparison of Primary Vessel Size in Experimental Facilities with that for a PWR

These figures show that the CSE system represents the largest system for which extensive experimental data have been obtained. The LOFT facility is considerably larger, but will not provide the parametric data obtained in the CSE and smaller scale facilities.

TECHNICAL APPROACH

Program Requirements

The CSE program was designed to provide technical backup for analytical methods used in assessing containment capabilities of power reactors. In order to provide the information on a timely basis, most information was needed in a time span about 4 years. In addition to this time limitation, the facility was to cost not more than about 1 million dollars. Operating expenses were limited to approximately 1 million dollars per year.

Information on containment centered on the following:

- (1) Leakage of fission products under accident conditions as compared to leakage data obtained under ambient temperature;
- (2) Removal of airborne fission products by natural deposition and by filters and sprays;
- (3) Temperature, pressure, force, and flow transients associated with sudden rupture of the primary system.

These topics cover a wide range of phenomena; hence the development of significant information requires performance of many experiments with each experiment being conducted under carefully controlled conditions.

Leakage

Currently, the leakage rate of fission products is predicted on the basis of leak rates measured at ambient temperature. At accident conditions, the gas phase is assumed to leak through the path at a rate which would be calculated to account for temperature and pressure effects, but assuming that the leak

path geometry did not change. Decontamination of gas within the leak path has been ignored. Following are the several pertinent questions which pertain to leakage and a brief description of the program designed to answer the questions.

First, what is the leakage under ambient conditions? Of the methods available for doing this, the absolute method and the reference vessel technique appeared most practical. Hence a series of experiments was planned to test both of these techniques. These measurement techniques were to be compared to each other, and to a known leak formed by carefully measuring the flow from a controlled path.

Second, what is the influence of temperature, pressure, and water vapor on the leak rate? Experiments were designed to measure changes in the leak rate as air temperature and pressure were changed. Also, the influence of steam on the methods for measuring leak rate, and the rate of leakage from known paths was to be measured.

Finally, to what degree do the fission products leak with air-steam mixtures? Experiments were designed to collect fission products leaked from known points, and to compare the fission product concentration of the leaked gas and liquid with the concentration predicted on the basis of the concentration inside the containment vessel.

Fission Product Transport

Currently, mathematical models are used to predict the removal of fission products from postaccident containment atmospheres. The CSE tests were designed to provide a proof test for these models. Since the number of physical parameters which are important is large, a relatively large number of experiments would be required. This need for a large number of experiments to study the influence the many important parameters had to be balanced against the relatively long time required to prepare for and carry out each large experiment.

Ideally, one would generate fission products in exactly the same way they would become airborne in an accident. This is impractical when one considers the very high levels of radioactivity which would result from heating a multi-kilogram quantity of highly irradiated fuel. For this reason, fission product simulants were selected. Elemental iodine, made airborne by heating pure crystals, was selected as the simulant for most iodine released from fuel. Methyl iodide is also produced to a small extent, hence methyl iodide was also added. Soluble volatile solids were simulated by use of cesium vaporized from cesium carbonate. Insoluble, refractory particles were formed by heating inductively a zirconium-clad, normal-uranium fuel element. The elemental iodine and cesium aerosol streams were passed over the molten uranium to permit intimate contact as would happen in an accident. Details of the generation techniques are described by Coleman.⁽²⁾ This use of simulants was not accepted without verification. Hilliard, et al⁽³⁾ and Roberts, et al⁽⁴⁾ verified that iodine and particles released from irradiated fuel behaved substantially the same as simulants generated as described above.

Sampling of the containment atmosphere was done by using a Maypack sampling train.⁽⁵⁾ This sampling train, shown in Figure 3, permits materials to be characterized as particulate (cesium, uranium, and iodine), as highly reactive gases (elemental iodine), and as nonreactive gases (methyl iodide and other organic iodides).

Clusters containing 12 of these Maypack samplers were suspended within the containment atmosphere at widely dispersed locations. Use of 12 samplers permitted sampling at 11 discrete times, retaining one of the samplers at each location to assess blank contamination. Installation of the samplers within the containment vessel was found to be necessary to minimize surface deposition and water condensation. The spatial arrangement

1 Inlet	6 Six Silver Plated Screens	11 Screen
2 Teflon Ball	7 Silver Membrane Filter	12 Spring
3 Nose Cone	8 Charcoal Loaded Filter	13 Body
4 Teflon Gasket	9 Stop Ring	14 End Cap
5 Two Particle Filters	10 Charcoal Bed	15 Outlet

FIGURE 3. Cross Section of CSE Maypack Sampling Train

of the sample clusters permitted mixing to be measured with single room and between the several rooms which make up a containment vessel.

Natural cleanup of containment atmospheres occurs as a result of deposition on wall and floor surfaces. For the wall surfaces, transfer occurs across a flow boundary layer. The flow velocity, and hence the mass transfer coefficient, depends on the magnitude of the thermal gradients at the wall. The thermal gradient and flow regime (laminar or turbulent) depend upon: (1) the thermal insulation of the wall, (2) the length of the wall, (3) the temperature differences across the vessel walls, and (4) the air-steam ratio. For aerosol particles, gravity settling on the vessel floor is also possible and the rate depends on the particle settling velocity.

Six natural process experiments were carried out under conditions designed to allow the influence of the important parameters to be measured. These tests are described in Table 1.

TABLE 1. Conditions for Natural Transport Experiments

Test	Vessel	Temperature, °F	Wall Heat Transfer	Grams I ₂ Injected
			Coefficient, Btu/hr ft °F	
D-1	Drywell	252	23	0.43
D-2	Drywell	250	23	0.60
A-1	Containment	181	37	0.70
A-2	Containment	185	37	56
A-5	Containment	253	0.31	84
A-11	Containment	252*	0.31	98

* Initial condition. Steam flow was terminated and vessel allowed to cool.

The rate of removal of fission products by sprays depends on the spray flow rate and on the effectiveness of each drop in collecting airborne contaminants. The drop effectiveness depends on a number of factors. These include: (1) drop size, (2) solubility of gas, (3) approach to saturation, and (4) physical properties of gas phase. For aerosol particles, the drop capture effectiveness depends primarily on particle size.

Eight spray experiments were carried out to assess the removal of airborne contaminants from containment atmospheres. The most important fission product specie is elemental iodine, and the experiments were designed primarily to test models of iodine washout. The condition selected for the eight spray tests are noted in Table 2. Additional details are provided by Hilliard⁽⁶⁾ and Postma.⁽⁷⁾

TABLE 2. Conditions for CSE Spray Tests

Test	Initial Temperature, °F	Spray Flow, gpm/ft ²	Drop Diameter, mmd* microns	Solution Makeup	I Release Conc, mg/m ³
A-3	77	0.026	1200	Caustic Borate	160
A-4	77	0.10	1200	Caustic Borate	160
A-6	255	0.10	1200	Caustic Borate	160
A-7	249	0.10	1200	Boric Acid	160
A-8	250	0.10	770	Caustic Borate	160
A-9	250	0.30	1200	Caustic Borate	160
A-10	250	0.10	1200	Caustic Borate	160
A-12	250	0.10	1200	Caustic Borate with Thiosulfate	160

* From manufacturer of spray nozzles.

The rate of cleanup of containment atmospheres by a recirculation system containing a filter depends primarily on the fraction of the containment volume which passes through the

system per unit time. Component efficiency enters, but is relatively unimportant provided it is above about 90%. The CSE filter cleaning tests were designed to (1) measure the effect of loop operation on the containment atmosphere, (2) determine the loop component efficiencies, and (3) observe the mechanical behavior of the loop. The five air cleaning tests carried out in CSE are described in Table 3.

TABLE 3. Conditions for CSE Filter Loop Tests

<u>Test</u>	<u>Initial Temperature, °F</u>	<u>Rate, cfm</u>	<u>Loop Components in Order</u>
A-13	96	1000	Prefilter, HEPA, 3 Char Beds
A-14	250	1000	Ht. Exch. demister, prefilter, HEPA, 3 Char Beds
A-15	250	1000	Prefilter, HEPA, 3 Char Beds
A-16	245	1000	Ht. Exch. demister, prefilter, HEPA, 3 Char Beds
A-17	246	1820	Ht. Exch. demister, HEPA, 3 Char Beds

Additional details on the CSE Filter loop tests are given by McCormack.⁽⁸⁾

Coolant Blowdown

At present, as part of the licensing procedure for a nuclear reactor, calculations are made of the behavior of the reactor and associated heat transport systems during the loss of coolant accident. Computer codes of varying degrees of sophistication are used to calculate flow rates, core temperatures, water remaining, forces on the core barrel and fuel assemblies, and containment pressure transients, etc., to assess the safety of the reactor during and after blowdown. The models used in these computer codes have

evolved generally from small scale laboratory tests. One problem to date has been the lack of data from well-instrumented large-scale simulations capable of providing measurements of fluid behavior in, and forces on reactor internal parts. The design bases of the CSE simulator vessel and internal structures, and the test sequences carried out in the facility, represent a compromise between, on the one hand, the desire for realistic modeling of reactor systems and accident events and, on the other hand, the need for a parametric study of systems that are sufficiently simple that comparison of the data with computer codes will give insight into the validity of the mathematical models which were used in formulating the codes.

To provide a sequence of parametric studies of blowdown phenomena, four test programs were planned. These were to be followed by a combined test program during which the blowdown of the reactor simulator vessel into the containment structure was accompanied by simultaneous release of fission products. The four blowdown test series were:

1. Blowdown from a simple vessel with no internal core structure: From these tests the validity of computer codes to model the subcooled, transient, and saturated phases of the blowdown event (so as to predict, transient pressures, flow rates, and liquid levels) would be assessed and needed changes suggested.
2. Blowdown from a vessel fitted with a simple perforated plate divider: These tests would serve to evaluate the capability of codes to model hydrodynamic forces within a simple system which bears a resemblance to the configuration within the reactor, yet is relatively simple to represent analytically. Transient forces on the plate as well as fluid behavior in a two region system would be measured.

3. Blowdown from a vessel fitted with a dummy core resembling the gross internal structure of a PWR: The ability of computer codes to model the added complexity of the three dimensional flow behavior in the interior of a reactor vessel would be evaluated in this series. Measurements essentially similar to those made in Series II would be made to evaluate transient forces on the core and the fluid behavior in the more complicated configuration.
4. Blowdown from a vessel fitted with a heated dummy core providing a modeling of the internal structure of a BWR: The existence of a quality gradient through the core region in a BWR and the presence of jet recirculating pumps within the vessel which provide alternate paths for fluid movement was believed to materially affect the course of the blowdown event. The forces on the core and fluid behavior in a configuration unique to BWR's would be evaluated.

As the result of early termination of the program, the first two of these series were completed and the third series was partially completed. Work on the fourth series was limited to design effort. The final combined tests, involving vessel blowdown into the containment structure and simultaneous release of fission products, was also cancelled.

Table 4 is a complete listing of all the blowdown runs performed and the initial conditions. Essentially all of the reduced data from these runs are available in the appendices of the CSE topical reports. One of the objectives of the CSE program was the generation of reliable data which could be subsequently used by other workers to check out future codes developed for analyses of blowdown and other related events.

TABLE 4. Summary of Blowdown Run Data

Run No.	Nominal Orifice Size, (c) in.	Orifice Location	Pressure psig	Temp. (a) °F	Wt. of Water, lb	Transition Time, sec
Series I - Sample Vessel						
B-3	3	C	600	Ambient	6250	21.0
B-4	4	C	600	Ambient	6250	11.8
B-5	4	C	1550	Ambient	6250	7.0
B-6A	6	C	1550	Ambient	6250	3.1
B-6B	8	C		Ambient	6250	2.5
B-7(b)	4	C	656	490	7300	27.0
B-8(b)	8	C	1565	495	7100	8.0
B-9(b)	8	C	1610	570	6750	5.8
B-10(b)	2	C	1400	575	6750	54.0
B-11(b)	4	C	1435	575	6700	16.0
B-12A(b)	4	C	1565	400	6600	12.0
B-12B(b)	4	C	1650	600	6500	14.0
B-13(b)	8	C	1600	590	5800	3.9
B-14A	4	C	2160	500	7400	21.0
B-14B	4	C	2160	492	7275	20.3
B-15	4	C	2185	560	6500	14.0
B-16A	4	C	2384	583	6000	13.7
B-16B	4	C	2385	577	6330	15.0
B-17	2	C	2375	589	6450	49.6
B-18	8	C	2400	600	5350	
B-80	8	B	985	509	7400	6.9-7.2
B-85	2	B	983	487	7400	74
B-84	4	B	984	495	7400	22.8-23.0
B-83	6	B	980	506	7400	10.5
B-82	6	B	1086	530	4700	4-4.5
B-81	8	B	984	542	4400	2.7-3.1
B-86	4	B	665	498	4700	10.3-10.8
Series II - Sieve Plate Separator						
19A	2	C	2150	602	6400	~52
19B	2	C	2100	602	6400	58
20	4	C	990	500	7400	29.6
20B	4	C	968	504	7300	27-28
21	6	C	999	502	7300	13.5
22B	2	B	1286	496	7400	74
23	1	B	2137	498	7400	233
24	4	B	810	504	4100	18
25	4	B	1285	503	7400	25.8
26	2	B	976	542	5500	46-47
27B	4	B	1075	549	5600	13.3
28	6	B	1032	545	5250	6.4
29	4	B	1302	515	7215	21.7
40	4	B	854	504	7300	26.5
50	2	A	2082	504	7400	14.5
51	4	A	1682	499	7400	10 sec
52	6	A	1289	499	7400	5.6
53B	2	A	980	540	3750	0
Series III - Simulated Core						
B-60	2	A(d)	2086	75	8700	
B-73	2	A(d)	1285	75	8700	
B-63	2	A(d)	2081	500	full	
B-74	4	A(d)	1283	500	full	
B-66	4	A(d)	952	535	steam line	
B-75	6	A(d)	1000	500	full	

a. Temperature gradient in vessel as much as 20 to 30 °F from bottom to top and, 100 to 125 °F in bottom outlet pipe.

b. Run was made with initial conditions in vessel as close as possible to saturation values.

c. Nominal Orifice Size, in. ID Area, ft²

2	1.689	0.0156
4	3.438	0.0643
6	5.189	0.146
8	6.813	0.253

d. Water exits from region below the core via an annular flow path surrounding the core and out a top nozzle.

PHYSICAL FACILITIES

Containment Vessel

The containment vessel was located within a large building on the Hanford reservation. The vessel arrangement and size scale are described in Figure 4. As noted in this figure, the outer containment vessel encloses several compartments. The largest of these is the drywell vessel designed to accommodate a reactor simulator vessel. Surrounding the drywell vessel were 4 wetwells designed to contain cold water for pressure suppression experiments. During the experiments performed, the lid of the drywell vessel was retracted so that the drywell volume became common to the main room volume. The wetwell volumes were sealed and hence their volumes played no role in the fission product transport processes.

Figure 5 is a photograph of the upper half of the containment vessel. The reflective outer coating is the aluminum foil-backed fiber glass insulation.

For the spray tests, the containment vessel was equipped to permit sampling of the spray liquid. About 75% of the spray liquid fell onto the main deck of the vessel, giving a drop fall height of 30.8 ft. The remaining 25% fell into the drywell. For this fraction, the fall height was 50.5 ft. The vessel arrangement used in the spray tests is shown schematically in Figure 6.

The air cleaning tests were carried out with a vessel arrangement similar to that used in the spray tests. The air cleaning loop was installed in the main room of the containment vessel. The intake point was located near the floor of the main room and the air was exhausted near the top of the vessel. The vessel arrangement used in the recirculating loop tests is shown in Figure 7.

FIGURE 4. Containment Vessel

Neg 43781-57

FIGURE 5. View of Upper Half of Containment Vessel

FIGURE 6. Schematic Diagram of Containment Arrangement Used in CSE Spray Tests

FIGURE 7. Air Cleaning Loop Installation

Important physical characteristics of the containment vessel are listed in Table 5.

TABLE 5. Physical Characteristics of CSE Containment Vessel

<u>Component</u>	<u>Physical Quantity</u>	<u>Units</u>	<u>Magnitude</u>
Containment Vessel	Overall height	ft	67.5
Containment Vessel	Diameter	ft	25
Containment Vessel	Working pressure	psig	75
Drywell Vessel	Height	ft	25
Drywell Vessel	Diameter	ft	11
Drywell Vessel	Working pressure	psig	150
Wetwell	Height	ft	16

Reactor Simulator Vessel

The reactor simulator vessel was designed and constructed for 2750 psig and 600 °F service in accordance with Section III of the ASME code for nuclear vessels. It was sized to represent about a one-fifth linear scale of a 1000 MW_e power reactor proposed at that time (1964).

The simulator vessel is diagrammed in Figure 8. A number of nozzles were installed on the vessel to permit blowdown simulation pipe breaks near the top, bottom, and intermediate elevations. All the wetted surfaces are clad with metallurgically bonded 304 stainless steel. Minimum clad thickness is 1/16 inch.

Figure 9 shows the vessel prior to being installed in the support cage.

Most blowdown experiments were carried out with an empty vessel. Some experiments were done using core simulators in the form of simple perforated plates and 3-dimensional simulated core to provide information on the forces exerted on internal structures. This is important in assessing the influence of a blowdown on a reactor core. The core simulators internal structures used in the CSE blowdown tests are pictured in Figure 10.

NOZZLE IDENTIFICATION

MARK	SIZE	FUNCTION
A	8"	BLOWDOWN
B	8"	BLOWDOWN
C	8"	BLOWDOWN
D	8"	BLOWDOWN
E	8"	BLOWDOWN
F	6"	WATER SPRAY SYSTEM AND HEATING ELEMENT INSERTION
G	4"	FISSION PRODUCT INSERTION
H	4"	AUXILIARY
J	4"	FISSION PRODUCT INSERTION
K	2"	PRESSURIZED AIR INLET AND PRESSURE GAGE PORT
L	2"	LOOP OUTLET
M	2"	SAFETY AND RELIEF VALVE PORT
N	2"	LOOP INLET
O	4"	FISSION PRODUCT INSERTION
P	2"	TEMPERATURE ELEMENT
Q	4"	FISSION PRODUCT INSERTION
R	4"	FISSION PRODUCT INSERTION
S	4"	FISSION PRODUCT INSERTION

DETAIL I

LOCATE A PAIR OF THERMOCOUPLE HOLES AS SHOWN ABOVE AT EACH LOCATION DESIGNATED BY TC.

GENERAL NOTES

1. VESSEL SHALL BE DESIGNED, FABRICATED, TESTED, AND STAMPED IN ACCORDANCE WITH ASME BOILER AND PRESSURE VESSEL CODE SECTION III, NUCLEAR VESSELS.
2. MAXIMUM ALLOWABLE WORKING PRESSURE TO BE 2750 PSIG AT 600 F.
3. VESSEL WILL BE COVERED WITH 3" OF INSULATION.
4. THE INSTRUMENT AND ELECTRICAL FEED THRU COLLAR WILL NOT ALWAYS BE USED DURING VESSEL OPERATING CONDITIONS, THEREFORE VESSEL SHALL BE DESIGNED FOR USE WITH OR WITHOUT COLLAR.
5. THERMOCOUPLES IN FLANGE, (TC9), ONE AT MID-DEPTH
5. THERMOCOUPLES IN FLANGE, (TC9), ONE AT MID-DEPTH

FIGURE 8. Reactor Simulator Vessel Diagram

FIGURE 9. Reactor Simulator Vessel
and Support Cage

FIGURE 10. Core Simulator Structures Used in Blow Tests

FIGURE 11. Time Schedule of CSE Program

TECHNICAL RESULTS OF CSE PROGRAM

LEAKAGE FROM CONTAINMENT VESSELS

One of the objectives of the CSE program was to investigate on an experimental basis leakage from containment vessels. Leakage is important because airborne contaminants released as a result of an accident can reach the environment only through leakage from the containment vessel. Both the rate of leakage of gases and decontamination of fission products within leak paths was studied.

Leakage of Air

Two techniques are commonly used to evaluate the leak rate of gas from a large contained gas volume. In the absolute method, the average temperature within an enclosed volume is measured, and this information, along with precise pressure measurements, is used to calculate changes in the total gas inventory. Changes in temperature and pressure with time are used in an equation of state to calculate changes in the gas inventory which is the leak rate. In the reference vessel technique, the pressure difference between a leak-tight enclosure and the containment vessel atmosphere is measured. The leak rate is then calculated from the loss in air pressure in the containment vessel. Both of these techniques were used in the CSE tests, and results were found to be equivalent. These methods gave meaningful data for leak rates down to about 0.1% per day. For lower leak rates, these techniques were not sensitive enough to give reliable results.

The change in leak rate with pressure depends on the nature of the flow which occurs within the leak path. Results typical of those obtained in CSE tests⁽⁹⁾ are shown in Figure 12. As shown in Figure 12, the measured leak rate fell between limits predicted from flow models based on turbulent smooth flow and laminar flow.

FIGURE 12. Effect of Pressure on Leak Rate of Air

Several tests were carried out in which the containment vessel atmosphere was heated, permitting the effect of temperature on leak rate to be measured. These tests⁽¹⁰⁾ showed that the leak rate decreased markedly as the temperature was raised from room temperature to 225 °F. Decreases in leak rates of the magnitude observed were not expected for cases where the flow geometry remains constant. Hence it was deduced that expansion of the vessel with temperature closed off the leakage paths.

Leakage of Air-Steam Mixtures

The presence of a condensable vapor may lead to formation of liquid in the leakage path. The liquid viscosity is much higher than for the gas, causing a marked reduction in the leak rate compared to the noncondensable gas. This is a factor of conservatism in estimating leak rates in steam-air mixtures on the basis of leak rates measured for air alone.

The presence of a liquid-vapor equilibrium also complicates the measurement of the leak rate using the standard reference or absolute method. These methods rely on accurate and representative temperature measurements for use in an equation of state. The very rapid increase in vapor pressure with temperature demands extremely precise temperature measurements if one is to accurately know the gas inventory. If temperature gradients exist, then there is necessarily a departure from perfect equilibrium between liquid water and steam. This prevents one from accurately assessing the water partial pressure, and introduces appreciable error into the leak rate measurement.

Measurements of leak rate using the absolute and reference vessel techniques proved to be worthless in CSE tests⁽¹⁰⁾ carried out in steam-air atmospheres. Leakage rate measured

at a known leak point fell drastically when steam was introduced into the vessel. For all the known leak points, the leakage rate fell to immeasurably low values when steam was turned into the vessel.

From the measurements made, it was concluded that leak rates under accident conditions will be less than those predicted on the basis of tests at ambient temperature.

Leakage of Airborne Fission Products

The leakage of fission products from artificial leak points was measured during several of the natural transport experiments. Typical results are shown in Table 6. Two important conclusions from these measurements are: (1) there is a chromatographic holdup of fission product activity in the leak path, and (2) most of the leaked materials is in water solution and thus unavailable for airborne transport. These factors introduce appreciable conservatism in dose calculations which neglect leakage path deposition.

TABLE 6. Leakage of Iodine and Cesium from a Steam-Air Atmosphere

Time After Fission Product Release, min.	Fission Product	Overall Decontamination Factor (Gas and Liquid)	Decontamination Factor for Leaked Gas
23-28	Iodine	1.65	33
48-63	Iodine	1.19	12
73-103	Iodine	0.41	1.7
168-308	Iodine	<u>0.26</u>	<u>0.53</u>
Overall		1.17	8.1
23-38	Cesium	1.10	<99
48-63	Cesium	1.24	<99
73-103	Cesium	1.50	<99
168-308	Cesium	<u>0.37</u>	<u><99</u>
Overall		1.12	99

Conclusions from Leak Tests

The following conclusions are supported by the CSE leak tests.

- The "absolute" and "reference vessel" leak rate measuring methods give equivalent results in air atmospheres for leak rates above 0.1% per day.
- Measurement errors limit these techniques to leak rates greater than about 0.1% per day.
- The standard techniques do not work in steam-air atmospheres.
- Air leakage rate through a given path changes with pressure at a rate intermediate between that predicted by models based on smooth turbulent flow and smooth laminar flow.
- Appreciable decontamination of reactive fission product gases and aerosol particles occurs as a result of condensation and deposition. This is especially important for elemental iodine.

FISSION PRODUCT TRANSPORT IN CONTAINMENT VESSEL ATMOSPHERES

Study of fission product removal from containment vessel atmospheres was a major objective of the CSE program. As noted earlier in this report, the emphasis changed from natural deposition to removal by filters and sprays as more reliance was placed on engineered safety systems.

Depletion of Airborne Concentration by Natural Processes

Two mechanisms play important roles in depleting fission products from postaccident containment atmospheres. One is surface deposition of gases and particles. Studies of natural transport processes have shown that the bulk of the gas phase

in a given compartment is well mixed. A boundary layer, driven by thermal and concentration buoyancy at the wall, is the locus of the mass transfer resistance. The second mechanism is gravity settling of fission products present as particles. While gravity settling is relatively unimportant for submicron particles produced by fuel overheating, agglomeration and water absorption cause growth to much larger sizes. For the larger particles, gravity settling can give appreciable cleanup rates.

Natural transport tests were carried out to explore the influence of the following parameters on the removal rate: iodine concentration, vessel size, heat transfer flux, and continued steam addition. The condition for the natural transport tests were summarized earlier in Table 1.

The depletion of inorganic iodine from the containment vessel atmosphere during a typical run is shown in Figure 13.

FIGURE 13. Concentration of Total Inorganic Iodine in Gas Space, Run A-11.

The data shown are for Run A-11 in which the steam feed was terminated two minutes after iodine injection was complete. Results for the first three hours are typical of those obtained in all other natural transport tests. In other tests, where steam addition continued, the concentration decreased to lower levels before reaching the slower removal rate period.

The initial iodine removal half-lives measured in the natural transport tests, along with the average gas-liquid distribution coefficients, are given in Table 7. Since the

TABLE 7. Iodine Removal by Natural Deposition

Test	Temperature, °F	Initial Half-Life Organic Iodine	Liquid/Gas Concentration Ratio
D-1	252	8.0 ± 1.0	6.1 × 10 ⁴
D-2	250	9.5 ± 0.5	6.3 × 10 ⁴
A-1	181	9.0 ± 4.0	2.6 × 10 ³
A-2	185	10.8 ± 2.0	2.0 × 10 ³
A-5	253	15.2 ± 1.5	5.8 × 10 ³
A-11	252	21 ± 4	7.9 × 10 ³

two hour dose reduction factor is determined by the initial iodine removal half-life, this initial removal rate is of prime importance. The liquid/gas distribution ratio is important in establishing the final equilibrium gas concentration.

The initial iodine removal half-times have been predicted on the basis of a mass transfer model in which one assumes that the transfer rate is gas phase limited, with zero back-pressure at the wall. Laminar flow was assumed to persist for 10 ft, and then the flow was assumed to become turbulent. The mass transfer rate was predicted from an engineering analogy between heat transfer and mass transfer. The predicted and measured transfer rates are compared in Table 8. For the

total number of experiments, the predicted half-time is 0.95 times the measured value. This good agreement verifies the correctness of the theory.

Cesium and uranium exist as suspended particles associated with condensed water. These particles may deposit by mass transfer to the vertical walls, or may deposit by gravity onto horizontal surfaces.

TABLE 8. Comparison of Predicted Initial Removal of Organic Iodine with CSE Experiments

<u>Run Number</u>	<u>Measured Initial t 1/2</u>	<u>Predicted t 1/2</u>	<u>Predicted Measured</u>
D-1	8.0 ± 1.0	7.7	0.96
D-2	9.5 ± 0.5	7.8	0.82
A-1	9.0 ± 4.0	10.0	1.11
A-2	10.8 ± 2.0	9.9	0.92
A-5	15.2 ± 1.5	16.1	1.06
A-11	21 ± 4	16.8	<u>0.80</u>
		Average	0.95

The airborne concentration of cesium in Run A-5 is shown in Figure 14. The initial half life is 38 minutes. The removal rate decreases with time. These results are typical of those obtained for the other tests. Data for uranium particles follow the same pattern, with half-lives in reasonable agreement with those measured for cesium. The mechanism of particle depletion was explored by estimating the fraction of the cesium which deposited on the vertical wall surfaces in the main room. This was done by analyzing the condensate collected by the wall trough. The wall captured cesium was compared to the total cesium lost from the gas phase, permitting that collected on the horizontal floor to be calculated. Results of these analyses are shown in Table 9.

FIGURE 14. Airborne Cesium Concentration in Typical Natural Transport Test

TABLE 9. Cesium Distribution Between Walls and Floor

<u>CSE Test</u>	<u>Fraction to Walls</u>	<u>Fraction to Floor</u>
A-1	0.13	0.87
A-2	0.15	0.85
A-5	0.08	0.92
A-11	0.26	0.74

The results presented in Table 9 show that most of the cesium deposits on the horizontal floor surface. Since the floor surface is small compared to wall surface, gravitational settling must be a controlling factor.

The observed depletion rate can be explained in terms of a well mixed gravity settling model. The model can be used to estimate the particle settling velocities and hence particle sizes. Results of such a calculation are shown in Table 10.

TABLE 10. Effective Cesium Particle Diameters in Natural Transport Tests

<u>CSE Run</u>	<u>Aerodynamic diameter,* microns</u>	
	<u>Early</u>	<u>Late</u>
A-1	18.5	8.6
A-2	16.0	10.3
A-5	11.2	6.7
A-11	8.6	2.7

* *Settling velocity predicted on assumption of unit density spheres.*

These calculations imply that particles in containment are much larger than the size generated, as would be expected.

The following conclusions are supported by the natural transport experiments.

- The removal rates measured at early times were in agreement with values predicted from a simple natural convection mass transfer model.
- The initial removal rate persisted until the concentration decreased to about 1% of the initial value.
- The natural convection model predicts the 2-hour dose reduction factor for inorganic iodine in a typical large PWR to be about 2.6.
- Methyl iodide was removed slowly from the atmosphere in all tests, with an average overall decontamination factor of 1.7 after 1 day.
- The cesium and uranium, representing both water soluble and insoluble fission product particles were removed from the gas space primarily by gravitational settling.
- Based on these experiments, the 2-hour DRF for all particulate aerosols in a typical large PWR is estimated to be 1.5. After 1 day, it would increase to about 3.
- The concentration of all fission product simulants were essentially uniform throughout the gas space of a single compartment. Large concentration differences between compartments persisted for several hours in these tests where no forced convection was provided.

Fission Products Removal by Sprays

Sprays are used in most PWR containment buildings. Their most important function is to condense steam and thereby limit pressure buildup. This protects the integrity of the containment vessel. A second important function is to remove fission products from the gas space. This reduces the potential off-site radiation dose.

The CSE spray tests were designed to test existing models of spray washout on a significant engineering scale. Most of the tests were parameter study experiments in which the influences of the most important parameters was studied. Two of the experiments were demonstration type experiments in which the sprays operated continuously as they would under postulated accident conditions.

The parameter study experiments are described in Table 11. In each of these experiments several brief spray periods were used. Measurement of the concentration before and after a spray period permitted the spray washout rate to be calculated.

TABLE 11. Conditions for Parameter Study Experiments

<u>Run</u>	<u>Initial Temperature, °F</u>	<u>Spray Rate, gpm</u>	<u>Drop Diameter, MMD</u>	<u>Solution,* pH</u>
A-3	77	12.8	12 0	9.5
A-4	77	49	1210	9.5
A-6	255	49	1210	9.5
A-7	250	49	1210	5
A-8	250	50	770	9.5
A-9	250	145	1220	9.4

* *Solution of boric acid, 3000 ppm.*

The thermal and flow conditions for the two demonstration tests are listed in Table 12.

TABLE 12. Conditions for Continuous Spray Tests

<u>Run</u>	<u>Initial Temperature, °F</u>	<u>Spray Rate, gpm</u>	<u>Drop Diameter, μ</u>	<u>Spray Duration, min</u>	<u>Average Temperature, °F</u>
A-10	249	49	1210	1100	250
A-12	253	50	1210	1300	250

The spray removal of elemental iodine in a parameter study test is shown in Figure 15. The iodine was allowed to mix in the main room prior to spray operation. The initial spray period produced a large reduction in iodine concentration. Subsequent spray periods give slower washout rates. This pattern was observed in all spray tests. Initially, the concentration is governed by the fresh spray washout of elemental iodine. For this period, the drops act as perfect sinks for iodine. After the concentration has been reduced several orders of magnitude, further reduction in gas phase concentration is slowed by two factors. First source terms, such as desorption of iodine from concentrated pools and surfaces, will become appreciable. Second, other forms of iodine, such as hypoiodous acid and methyl iodide adsorb to a low level on the silver components of the Maypack. Hence the presence of other forms of iodine, even though initially of low concentration, will finally limit the elemental iodine washout as indicated by samples which are less than perfectly discriminating.

The initial spray washout rates for elemental iodine (defined as that fraction captured by silver components of Maypack) in all of the parameter study experiments are compared to a perfect sink drop absorption model in Figure 16. The data appear to fall close to 80% of the predicted value, using the surface mean spray drop diameter. This agreement is considered good. The small discrepancy is considered to be due to inaccuracies in evaluating the theoretical prediction. Drop size is a very important parameter in the theory, and a slight error could produce the magnitude of disagreement observed here. It should be noted that the spray washout rate shown in Figure 16 is that due to spray alone. It was evaluated by subtracting from the observed removal rate, the removal rate due to natural processes.

FIGURE 15. Concentration of Elemental Iodine in the Main Room, Run A6

FIGURE 16. Comparison of Measured Initial Spray Washout Rates with Theory

Chemical reactions within the liquid phase cause the equilibrium iodine partition coefficient to increase with time. The equilibrium concentration ratios observed for elemental iodine are shown in Table 13. Runs A-4, A-6, A-8, and A-10 employed the caustic-borate solution. In Run A-7, boric acid was used (pH = 5) and in Run A-12, the caustic-borate was augmented by 1% sodium thiosulfate.

TABLE 13. Long Term Equilibrium of Elemental Iodine in Spray Solution

<u>Test</u>	<u>Time of Sampling, min</u>	<u>C_l/C_g</u>
A-4	1300	2×10^5
A-6	1630	8×10^4
A-7	1500	4×10^4
A-8	260	1×10^4
A-10	1100	1×10^5
A-12	500	$>3 \times 10^6$

Methyl iodide was added to the gas space in each test. The spray removal rate was slow (half-life of about 12 hr) of the caustic-borate solution. Addition of sodium thiosulfate increased the methyl iodide washout rate by about an order of magnitude. For methyl iodide, absorption at wall surfaces is relatively more important than for elemental iodine. This is because liquid phase mass transfer resistance limits the absorption rate. Results of several of the tests using sodium thiosulfate are listed in Table 14. Agreement between experiment and theory is good except for Test A-12(2) where all liquid was flooded onto the wall of the CSE vessel. For high wall flow rates, turbulent mixing would be appreciable. This explains the faster removal compared to the laminar flow theory reported in Table 14.

TABLE 14. Methyl Iodide Removal
by Thiosulfate Sprays

<u>CSE Run</u>	<u>Spray Duration, min</u>	<u>Observed Removal Half Time</u>	<u>Predicted Removal Half Time</u>
A-7*	50	130 min	185
A-8*	50	130	150
A-10(2)*	240	85	80
A-12(1)	1270	80	80
A-12(2)*	120	100	382

* Test was an addendum to the listed run.

Particles are captured by drops by a number of mechanisms. These include impaction, interception, brownian diffusion, thermophoresis, diffusioophoris, and electrical effects. For particles in the 0.1 - 0.5 size range, none of these mechanisms are highly effective. Hence particle washout has been predicted to be ineffective for sprays.

The particle washout measurements in CSE have shown that both uranium and cesium are effectively scrubbed from containment atmospheres. The reason is simply that for accident conditions, aerosol agglomeration and water vapor condensation lead to particles larger than the size associated with the minimum in removal efficiency.

Typical results are shown in Figure 17. The removal rate decreases with spray time. This is as expected. The larger, more easily removed particles are scrubbed out, leaving behind a preponderance of smaller particles.

The initial spray washout rate corresponds to theory if the particle size were 3 to 4 microns. The sizes measured with a cascade impactor were generally smaller than 1μ . This apparent anomaly is explainable in terms of particle evaporation during the course of the cascade impactor sampling. The in-situ size is larger than the size indicated in a cascade impactor.

FIGURE 17. Cesium Concentration in the Main Room, Run A-6.

The CSE data on particle washout were applied to a large PWR containment vessel. In applying the CSE results, it was assumed that the drops in a PWR containment vessel would collect particles as effectively as the drops in CSE. A correction was applied to account for the fact the diffusiophoresis would be relatively less effective in a PWR because of the greater drop fall height. Two estimates were made. In the first, a best estimate was made from the "best fit" of a correlation between efficiency and dimensionless flow, ft/V . In the second estimate, a conservative application of data was made by defining a drop collection efficiency which was lower than all observed CSE results. This lower estimate of drop collection efficiency was then used to predict washout in a PWR containment vessel. Results of this calculation are shown in Figure 18.

FIGURE 18. Spray Removal of Particles in a PWR Containment Vessel

is 28.5. The conservative estimate is 12.2. No lower estimate can be made from the CSE washout data. These estimates of dose reduction factor are based on the assumption that particle size under accident conditions will be similar to that in CSE. This is surely a conservative assumption because the initial concentration of particles in the CSE experiments was less than 1% of that predicted on the basis of TID-14844 assumptions.

The following conclusions are supported by the CSE spray experiments.

- Good agreement was obtained between experimental values of initial removal rate for elemental iodine and predictions based on gas phase limited transfer and on mean drop size. The measured concentration half lives of 0.6 to 2.0 min are in the range expected for large power reactor systems.
- Iodine was not reevolved from spray liquid which was recirculated for 20 hours at 250°F. Continuing chemical reactions in the liquid phase bind iodine in a non-volatile form.
- The 2 hour time-integrated dose reduction factor (DRF) for a large PWR is estimated to be about 50 for elemental iodine. The DRF continues to increase with time.
- Removal of methyl iodide by hot base-borate-sodium thio-sulfate sprays agreed well with prediction by a model based on chemical reaction rate limiting process. The removal rate is temperature sensitive and the wall film is more effective than the drops. The 2-hour DRF for methyl iodide would be about 1.5 for a large PWR.
- Aerosol particles are effectively removed by strays. Particle agglomeration and water adsorption insure that particles generated under the maximum hypothetical accident case will be larger than generated in the CSE vessel. Conservative use of CSE data leads to a 2-hour RDF of 12.2 for a large PWR containment vessel.

Fission Product Cleanup by Filter-Adsorbent Systems.

An alternate means of reducing the airborne concentration of fission products is by using filter-adsorbent air recirculating equipment. An air cleaning system was operated internally in the CSE system. This system is pictured schematically in Figure 19 as used in one test, A-15.

FIGURE 19. Air Cleaning Loop Installation

For an air cleaning system, a simple model of cleanup may be derived on the basis of good mixing in the gas phase. This model gives

$$\frac{C_g}{C_{go}} = e^{-\frac{FEt}{V}}$$

where F = flow

V = volume

E = removal eff

This highly simple equation doesn't account for factors such as deposition on surfaces, absorption and desorption by water pools, and changes in removal efficiency. The purposes of the air cleaning tests were to (1) test out the above equation, and (2) assess the behavior of commercially available components in a containment atmosphere. The tests completed are described in Table 15.

Results typical of the air cleaning tests are shown in Figure 20. A number of important features are shown by this figure. First, all iodine forms are rapidly removed from the containment atmosphere initially. For elemental iodine, the removal rate is higher than predicted by the simple recirculation theory. Methyl iodide is initially removed at a rate which corresponds to a well mixed model, with a removal efficiency of 80%. The airborne concentration levels out at a low value. The levelling effect is primarily the result of desorption from pools and surfaces in the containment vessel.

For the run shown a second injection of methyl iodide was used to determine its removal when the charcoal was saturated with water vapor. The result showed that the bed efficiency was not greatly impaired by being saturated with water vapor. Initially the bed adsorbs water vapor, and the heat of adsorption heats

TABLE 15. Summary of CSE Air Cleaning Results

Early Concentration Half-Times, $t_{1/2}$, minutes															
Run	13			14			15			16			17		
	N	N+L	L	N	N+L	L	N	N+L	L	N ^(a)	N+L	L	N	N+L	L
Elemental I	26	7.3	10.1	16	4.9	7.1	27	6.0	7.7	15	7.3	14.3	13	3.8	5.4
"Particulate" I	50	11.0	14.1	27	9.0	13.5	27	10.3	16.7	36	15.0	25.7	25	6.3	8.7
AC Paper	~50	16.0	21.0	39 ^(a)	9.0	11.8	58	10.0	12.5	39	13.7	14.2	20	7.7	12.5
Char Bed	--	15.0	15 ^(c)	--	14.3	14.3 ^(c)	--	14.7	14.7 ^(c) 19.0 ^(b)	--	16.0	16.0 ^(b)	--	11.0	11 ^(b)
Cesium	62	11.0	13.7	42	12.3	17.4	70	13.0	16.0	47	9.3	11.5	33	7.7	10.0
Uranium	80	14.0	17.0	85	13.0	15.4	70	12.0	14.5	31	16.0	33.0	41	8.3	10.4
Average			15.15			13.25			14.4			19.1			9.7
Expected ^(e) , min			14.8			13.5			14.6 ^(d)			14.2			7.9
Ratio: Exp./Calc.			0.98			1.02			1.01			0.743			0.81

N = Natural L = Loop Only

(a) Natural $t_{1/2}$, not measured this run. Ave. of 8 CSE runs under similar conditions used.

(b) Bed moisture equilibrated.

(c) "Dry" Bed.

(d) Excludes (c) above.

(e) $t_{1/2} = \frac{0.693 V}{F}$, F includes steam condensed. Assumed loop efficiency = 100%.

FIGURE 20. Main Room Gas Phase Iodine Concentration, Run A-15

the bed well above the water saturation temperature. After many hours of operation, the charcoal beds become saturated with water and the temperature of the air does not change as it passes through the bed. This is important because earlier work on small scale test beds showed that removal efficiency fell off markedly as the relative humidity approached 100%.

The overall effectiveness of the air cleaning systems for the four puff release tests is shown in Figure 21 for iodine. This figure shows that the simple well-mixed model gives a conservative prediction early in the post accident period when concentrations are high. For long times after the accident, when the concentrations are very low, source terms and equilibrium effects limit the removal.

For cesium, similar results were obtained as shown in Figure 22.

In summary, the air cleaning tests have verified behavior expected for cleanup and have verified that mechanical problems are not of major importance.

FIGURE 21. Comparison of Iodine Removal Rate with Theory

FIGURE 22. Comparison of Cesium Removal Rate with Theory

BLOWDOWN PHENOMENA

The relatively large scale of the CSE reactor simulator vessel and the extensive instrumentation which was capable of recording very rapid transients [described in Reference (11)] provided the opportunity to gain detailed information on the pressure, mass velocities, and liquid behavior under carefully controlled conditions during the blowdown which were then compared with such computer codes as WATER⁽¹²⁾ HAMMER⁽¹³⁾ FLASH 2,⁽¹⁴⁾ WHAM,⁽¹⁵⁾ and RELAPSE to evaluate their ability to model the blowdown event. Details of these results are presented by Allemann et al. in BNWL-1411⁽¹⁶⁾ and BNWL-1470⁽¹⁷⁾ for a simple vessel with no internal core structure, and in BNWL-1463⁽¹⁸⁾ and BNWL-1425⁽¹⁹⁾ for the vessel with a perforated sieve plate separator and a simulated reactor core respectively.

Saturated Blowdown

In a boiling water reactor the blowdown would occur from an initially saturated condition, while in a pressurized water reactor the initial condition of the coolant at the instant of rupture is that of subcooled water. The portion of the blowdown event during which the system is depressurizing to the saturated condition (i.e., subcooled blowdown) has characteristics that differ markedly from the saturated blowdown situation and will be discussed subsequently.

Pressure History: The transient pressures existing in a simple vessel during the saturated portion of the blowdown event are correctly predicted within 50 psi by the FLASH 2 code if appropriate values of several adjustable parameters are used. These parameters are the discharge coefficient (or break area), the bubble rise velocity, and the coefficient of quality gradient in the vessel. Figure 23 shows typically the effect of these parameters upon the fitting of the FLASH 2 code to the experimental data. The discharge coefficient C_D is used to

FIGURE 23. Effect of FLASH 2 Code Parameters on the Fitting of Code Calculations to Experimental Data Run B-13

adjust the maximum theoretical blowdown rate (obtained in the code from the local outlet condition by using Moody's⁽²⁰⁾ theoretical maximum flow rate results) to the flow rates actually observed in the experiment. The value of C_D required for a fit of the data varied with the break size but was essentially independent of the location of the break along the height of the vessel for tests run in the vessel with no internal structures. However, the values of C_D required to fit the data obtained with the sieve plate or the simulated core were somewhat larger and were not related in an obvious manner to the values needed to fit the data obtained from the simple vessel under comparable initial conditions.

The bubble rise velocity, V_b , determines the rate of disengagement of the rising steam bubbles and hence the water-steam mixture level in the vessel. While this parameter does not have a very sensitive effect on the pressure history, it does have a significant effect on the fit of the code and the data with respect to the liquid level. A value of $V_b = 2$ (feet/second) was generally adequate to provide a fit of the transient pressure data but did not consistently provide an adequate fit of the liquid-level data.

The coefficient of quality gradient, C_x , is a parameter in the code which gives a range of quality in the vertical direction in the vessel and relates the quality of any level to the average quality in the vessel. No consistent rationale for the selection of the parameter to obtain a best fit of the data was arrived at. As in the case of V_b , the parameter C_x has a relatively minor effect on the calculated pressure but greater effect on the liquid level calculated by the code.

Mass Velocity: The rate of flux of fluid from the vessel was obtained from measurements of the weight loss during the blowdown event. As previously mentioned, in comparing this to the calculated mass velocity of flow using the FLASH 2 code,

which in turn uses the Moody model for two phase critical flow, it was necessary to apply an arbitrary flow coefficient. Figure 24 presents a summary of available data on the variation of the average mass velocity with break size. The CSE results from the simple vessel tests with no internal core structures could be empirically correlated by the expression

$$C_{\text{CSEM}} = 0.287 R^{-0.165}$$

However, the data from the CSE tests with the sieve plate separator indicated the discharge coefficient was roughly 45 percent higher and varied as the -0.19 power of the ratio of the break area to the vessel cross sectional area. A mechanism proposed to explain the general behavior is that with the increase in break area the blowdown is more rapid and as a consequence more of the bubbles are swept into the break making the local quality in the break region higher. This in turn results in a lower critical velocity. Attempts to formulate a model⁽²¹⁾ which more completely described the local conditions throughout the vessel, and in particular in the region of the break, did not succeed in eliminating the need of an empirical discharge coefficient and it was concluded that the present state of knowledge as to the correct value to be used in conjunction with the Moody model is inadequate. Either a means of better determining the local break situation in the model should be developed or more tests must be run to obtain a rationale for empirical selection of this coefficient for all the cases of potential interest in reactor safety analysis. As a further example of this conclusion, the effect of outlet nozzle length on the blowdown rate was found to be much less pronounced than predicted by the Moody model. The addition of a 45.5 inch extension on the blowdown nozzle resulted in only a 3 percent increase in time to transition (the change from

FIGURE 24. Average Mass Velocity Versus Orifice-to-Vessel Area Ratio

two phase to single phase flow through the break) as compared with a predicted increase of 75 percent using the Moody model. This result suggests caution in assuming that pipe friction will markedly decrease the rate of escape of fluid when applying the calculational models to a reactor safety analysis. However, with this one exception, all of the deviations between theory and these experiments are in a normally conservative direction, at least for the pipe sizes usually found on reactor heat transport systems, since lower than predicted flow rates will mean slower and less rapid containment pressurization due to coolant loss from the reactor.

Liquid Level: The liquid level transient is of primary concern in the analysis of the blowdown event because of its effect on the length of time that the core may be uncovered and without coolant between the time the break occurs and the time the emergency core coolant system is able to provide backup cooling. In the CSE tests a time-domain reflectometer (TDR) was used to gather data on the liquid level during the expulsion of fluid. Figure 25 shows a typical plot of this data. The liquid level shows a relative swell compared to the level calculated from the fluid weight and the specific volume. In the simple vessel with no interior structures, the data for blowdowns from either the bottom or the middle nozzles was adequately fit by the FLASH 2 code by appropriate use of the three parameters previously mentioned. However, the data on blowdowns from a nozzle at the top of the vessel could not be fitted satisfactorily by the FLASH code. A more detailed model of the liquid-vapor action was developed⁽²¹⁾ which correctly predicts the liquid level in top-vessel breaks. Neither code can adequately predict the liquid level behavior in the vicinity of the break because the code cannot treat the break as a region of finite thickness.

FIGURE 25. Liquid Level Versus Time

The presence of a constriction in the vessel, such as the core, introduces the possibility of the formation of a steam dome. Figure 26 illustrates the data and the code prediction from such a situation. The probe, which senses the distance from the top of the vessel to an electronic impedance discontinuity caused by the upper surface of the liquid or foam, receives no information concerning the liquid level in the lower volume until the liquid has disappeared from the upper volume. When the upper liquid is gone, the signal suddenly collapses to the liquid level reading in the lower volume. The FLASH code can simulate the situation by considering the vessel as two separated volumes. However, it is not able to take into account the possible passage of steam up through the sieve plate. Hence, if the blowdown is relatively slow, the FLASH code more nearly depicts the situation by considering a one regional model, while with a rapid blowdown the use of the two volume approach gives a more satisfactory fit. Table 16 lists the parameters used in fitting the code calculations to some of the CSE results. Note the consistency of parametric values to fit data from runs at essentially the same initial conditions, but the need for different parametric values to fit data from differing initial conditions.

The formation of the steam dome beneath the sieve plate provided a means of obtaining an approximate value of the bubble rise velocity. By comparing the results of a series of blowdowns through different size breaks (hence, differing flow velocities through the sieve plate) it was observed that if the velocity through the plate was greater than about 4.6 ft/sec., a steam dome was formed because steam bubbles could not escape countercurrent to the flow of liquid through the plate and the recorded liquid level collapsed to a low value as soon as the fluid in the upper region had been

FIGURE 26. Liquid Level Versus Time for a Blowdown in Which a Seive Plate Was Used

TABLE 16. FLASH2 Parameters

Run Number	No. of Control Volume	Nominal Size, in.	Break Diam, in.	Initial Vol, ft ³	Discharge Orifice Coefficients		Bubble ^(a) Rise Velocity, ft/sec	Quality Gradient Factor
					Low-Quality	High-Quality		
B-22B	1	2	1.689	149	0.80	1.00	2.0 ^(b)	0.8
B-23	1	1	0.815	149	1.049	1.049	3.0	0.8
B-25	2	4	3.438	149	0.664	0.89	2.0, 2.0 ^(b)	0.8, 0.4 ^(a)
B-26	1	2	1.689	113.6	0.80	1.00	2.0 ^(b)	0.8
B-26	2	2	1.689	113.6	0.80	1.00	3.0, 3.0	0.8, 0.8 ^(a)
B-27B	2	4	3.438	113.5	0.664	0.89	2.0, 2.0 ^(b)	0.8, 0.4 ^(a)

a. For two-volume cases, the first number designates the upper volume parameter, the second number the lower volume parameter.

b. Variable bubble rise model from Redfield, J. A., and Murphy, J. H., Void Fraction and Residual Water Predictions During Loss-of-Coolant, WAPD-T-2155, Bettis Atomic Power Laboratory, Pittsburgh, Pennsylvania, September 1968.

exhausted. At lower velocities of flow through the sieve plate, no collapse was noted. Hence, it was concluded that the rate of rise of steam bubbles must be approximately 4.6 ft/sec.

Liquid Remaining After Blowdown: This quantity is also of interest as it is indicative of the liquid swelling that has existed during the blowdown event. Increasing the break size decreases the amount of liquid remaining after blowdown when the break is not at the very bottom. Figure 27 illustrates this effect. It also shows the effect of break location on the difference between the predicted value, calculated by the FLASH 2 code, and the data. The differences may be due to entrained fluid discharged from the vessel during blowdown. Entrainment is not considered in FLASH 2 although it is believed to be significant following the transition to steam "only" blowdown. The presence of the core also has the effect of reducing the amount of liquid remaining after blowdown by increasing the entrainment due to restricted flow passageways and the more violent interaction of the steam and liquid as the former sweeps over the liquid enroute to the break. Analysis should be improved to properly account for these effects for scaling to full size reactor rather than using the ratios calculated from the CSE data for direct application.

Fluid Quality at the Break: In an effort to gain understanding of the flow phenomena at the break location, measurements were made of the void fraction in the outlet duct using a neutron densitometer technique. The densitometer utilizes moderation and/or absorption of fast neutrons (from a neutron generator) by the steam-water mixture in the pipe. These data were used to calculate the fluid quality through the assumption of a relative slip velocity between the steam and water. Generally homogeneous flow (zero slip velocity) was assumed. During the saturated blowdown portion of the event the observed

FIGURE 27. Residual Water in Vessel After Blowdown from Full Vessel

quality remained essentially constant at a low value between 0.01 and 1 percent, depending on the break size, and then abruptly increased to approach 100 percent near the final portion of the blowdown when essentially pure steam was being expelled. The data indicate that during the period of near-constant quality, quality at the orifice increases as break size and initial temperature increases. These results are consistent with other studies of flashing and cavitation in jet flow. For example, Lienhard and Stephenson⁽²²⁾ found that the critical cavitation number in flashing jet flow,

$$\sigma_i = 2 \rho (P - P_v) / G_{crit}^2,$$

is proportional to (orifice area)^{0.33}. Thus G_{crit} should be proportional to (orifice area)^{-0.166} which is close to that which was observed in the CSE results. One can infer that the larger break allows higher turbulence. Consequently, more bubble nuclei are triggered to form bubbles resulting in higher quality which in turn limits the mass velocity to a lower value than with smaller breaks. Photographs taken of the fluid in the region of the break generally substantiate the inference. With small breaks, bubbles in the size range of 1/16 to 1/32 inch were observed moving at essentially the velocity of the water. Larger breaks produced a multitude of very fine bubbles (a cloud) in which the individual bubbles could not be resolved. In actual reactor system ruptures, even small breaks may have sufficiently turbulent flows (because of the normal operating fluid velocity) to initiate rapid nucleation in the event of a rupture, with the consequence that the resulting quality at the break may be higher, and the mass velocity lower than was observed in the CSE tests. However, conversely, the mixing of the cool ECCS flow with the blowdown flow may

reduce the quality at the break and cause more rapid depressurization than predicted. Conceivably this could result in a more rapid loss of fluid from the vessel than without initiation of the ECCS flow.

Subcooled Blowdown

During the initial portion of the blowdown of a pressurized water reactor the depressurization is very rapid because the low compressibility of water requires very little volume change to reduce the pressure. As a consequence, the propagation of the depressurization wave is limited by acoustic phenomena and temporal and spatial imbalances of pressure are set up in the fluid system. The CSE vessel was equipped to record up to 24 points of data sampled over a cycle time of 0.1 milliseconds during this period which lasted up to approximately 100 milliseconds. Consequently, the data are particularly applicable to the evaluation of computer codes such as WATER-HAMMER and WHAM which calculate, in one dimension, the acoustic decompression wave and the interaction with the vessel and piping structures.

Pressure History: Figure 28 is a typical trace of the pressure at a particular location within the vessel during the subcooled portion of the blowdown. The behavior during this period is strongly affected by the back pressure developed at the orifice immediately following the break. This in turn is dependent on the geometric configuration in the orifice region. In the CSE tests the pressure in the orifice generally dropped well below the saturation pressure. Moreover, the method of initiating the break, namely the double-disc rupture system, caused a characteristic two-step decompression. These conditions had to be taken into account in the fitting of the WHAM code calculations to the data. In order to obtain a satisfactory fit of the data by the WHAM code, it was necessary

FIGURE 28. Measured and Revised WHAM-Calculated Pressure Histories at the Instrument Ring for Run B-51

to model the orifice and inlet pipe pressure histories from previous experimental data, rather than to use the standard area-increase ramp of an arbitrary duration as originally prescribed by the program. With such a modeling procedure, the WHAM code was found to be reasonably successful for modeling simple pipe-vessel systems. The fit illustrated in Figure 28 is typical. Depressurization rates in the vessel were predicted approximately correct for a majority of the blowdowns, although a few exhibited slower rates than predicted by the code, indicating either an effective back-pressure higher than atmospheric or a lower than normal exit coefficient. Although some of the pressure data from the blowdowns did indicate the possibility of multi-dimensional effects, the evidence was inconclusive, and most of the data were consistent with the assumption of one-dimensional behavior in the simple vessel configurations. The fit between code prediction and data was somewhat less satisfactory for the blowdowns with the simulated core in the vessel. Prediction of the vessel pressure showed too large an amplitude of fluctuation and some phasing error late in the calculations. It is likely that this is due to the greater complexity of the configuration and the less perfectly defined geometry of the core-in-vessel cases.

The method of pressurizing the vessel with nitrogen in order to obtain subcooled conditions was found to have a large effect on the decompression history. This was due to the amount of nitrogen which could be desolved in the water during heatup. The effect was to reduce the rate of subcooled decompression as the result of the dissolution of the nitrogen from the water as the pressure was reduced. Thus the decompression pattern was less violent and the strain on vessel internal structures (discussed below) was markedly reduced. By changing the configuration of the system so the pressurizing nitrogen interfaced with the water in a cool region, this influence of desolved nitrogen could be avoided.

Mass Velocity: The expression for subcooled flow through an orifice suggested by Zaloudek⁽²³⁾ is

$$G = c \sqrt{\frac{2g_c}{v_f}} (P - P_{sat})$$

The expression assumes that a back-pressure equal to the saturation pressure of the liquid is formed at the orifice. As previously mentioned, in the CSE experiments the noted orifice pressure was generally much below the saturation pressure and more typical results were obtained by assuming ambient pressure at the orifice and a standard value of the orifice coefficient.

Several runs were made to observe subcooled decompression at ambient temperatures. The results were nearly as comparable with WHAM calculations as were the high temperature blowdowns. The success of this method suggests that simple checks of the analytical treatments and the inputs required from the particular configuration can be made for the full scale reactor system by rupture-disc testing of the pressurized system when filled with cold water.

Blowdown Forces: One of the primary concerns in the consideration of the subcooled blowdown is the forces generated on the core structures. These are generally considered to be greater during the subcooled portion of the blowdown event than during the saturated portion. The principal purpose for the development of the WHAM code was the prediction of these forces. In the CSE program the core structure was first simulated by a simple sieve plate in order to work with a configuration which could be analyzed relatively easily and thus provide a firmer basis of comparison between data and analytical calculation. Following these tests, a somewhat more elaborate simulation of the core was provided by the dummy core barrel insert. This configuration provided a more realistic

simulation of the fluid flow paths and the possible three-dimensional effects that would exist in a prototypic configuration.

Forces on the internal structures were measured through the use of strain gages mounted on the mounting lugs within the vessel and on the bolts that connected the sieve plate or dummy barrel to the lugs. The measurement system was calibrated in place at ambient conditions by a hydraulic jack arrangement. Signals generated by the strain bolt configurations were fed to a small digital computer which logged and processed them and turned out a plot for each signal such as shown in Figure 29. Direct comparisons of data with predictions from WHAM calculations of the upper and lower plenum pressure differences were made by calculating the anticipated strain in the system if subjected to the calculated pressure differences. Such a comparison is shown in Figure 30. In general the predicted peak force were from two to five times the peak obtained by averaging the results from all the bolts of a given run. The peak loads observed and those calculated moreover often did not agree in time or direction. The discrepancy may come from several sources: The pressure waves directly measured in the vessel were lower than the calculated values. This could result from the equations themselves or the effect of two-dimensional effects which in reality may have dissipated the decompression wave structure. Another possibility is that the vessel and core plate did not form the ideal attachment that was assumed in the code which converted the WHAM-calculated pressure differences to expected wall-side stresses. However, in this case, the error is in the direction of making a conservative prediction insofar as nuclear reactor safety is concerned.

FIGURE 29. CSE Blowdown (6 inch 1000, 500, 214) High-Speed Strain on Support Bolt 8 inner (Run B-21)

FIGURE 30. Blowdown B-22 - Middle Nozzle
(2 inch, 1330 psia, 500 °F)

The thrust force generated by the blowdown was found to be approximately $T = C(2PA_b)$ where P is the vessel pressure and A is the break area. The values of the coefficient ranged from 0.5 to 1.0 and were generally higher for the smaller breaks. However, the shape of the pipe at the break may have a more pronounced effect on the thrust and result in either higher or lower values. For example, if the pipe bulges and forms a flared end upon breaking it would give more thrust than the simple sharp ended break.

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