OAK RIDGE NATIONAL LABORATORY
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for the
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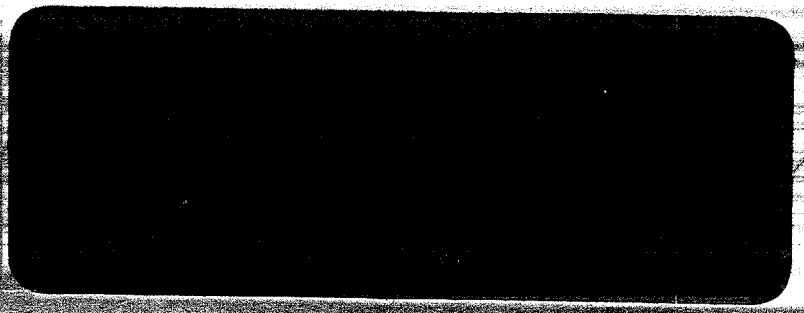
ORNL NUCLEAR SAFETY RESEARCH AND DEVELOPMENT PROGRAM

BIMONTHLY REPORT FOR JULY-AUGUST 1968

Wm. B. Cottrell



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Wm. B. Cottrell

NOVEMBER 1968

OAK RIDGE NATIONAL LABORATORY
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CONTENTS

	Activity No.	Page
ABSTRACT		vii
SUMMARY		ix
1. FUEL FAILURE AND BEHAVIOR OF ACCIDENT-RELEASED FISSION PRODUCTS		1
1.1 Comparison of Real and Simulated Fission-Product Aerosols	04 60 10 01 1	3
1.2 Failure Modes of Zircaloy-Clad Fuel Rods	04 60 10 01 1	7
1.3 LOFT Assistance Program	04 60 50 05 1	11
2. FILTRATION AND ADSORPTION TECHNOLOGY		23
2.1 Removal of Radioiodine by Solid Sorbents	04 60 80 01 1	25
2.2 Removal of Solid Aerosols	04 60 80 01 1	27
2.3 Characterization of Fission Products Under LOFT Conditions	04 60 50 05 1	35
2.4 Ignition of Charcoal Adsorbers by Fission-Product Decay Heat	02 30 02 90	36
2.5 Separation of Noble Gases from Air with a Permselective Membrane	04 60 80 01 1	40
2.6 High-Efficiency Air-Filtration Engineering Manual	04 60 80 01 1	48
3. SPRAY AND POOL PRESSURE-SUPPRESSION TECHNOLOGY		49
3.1 Effect of Additives on Distribution of I_2 and CH_3I Between Air and Water	04 60 80 01 1	51
3.2 Uptake of I_2 and CH_3I by Water Solutions and Drops	04 60 80 01 1	57
3.3 Spray Studies at the Nuclear Safety Pilot Plant	04 60 80 01 1	64
3.4 Radiation Stability of Sprays	04 60 80 01 1	73
3.5 Corrosion Studies	04 60 80 01 1	76
3.6 Pressure-Suppression Experiments	04 60 80 01 1	82
3.7 Scale-Model Tests of Fission-Product Removal in Suppression Pool	04 60 80 01 1	84

		Activity No.	Page
4.	SAFETY STUDIES FOR HTGR		93
	4.1 Program Development	04 60 10 01 1	95
	4.2 Steam-Carbon Reaction and Fission-Product Release and Transport Studies	04 60 10 01 1	97
	4.3 Engineering-Scale Steam-Graphite Reaction Rate Experiment	04 60 10 01 1	100
	4.4 High-Temperature In-Pile Fuel Integrity Tests	04 60 10 01 1	109
	4.5 High-Temperature Behavior of Gas-borne Fission Products	04 60 10 01 1	110
	4.6 In-Pile Studies of Reactions of Fueled Graphite with Steam Under Accident Conditions	04 60 10 01 1	113
5.	HEAVY-SECTION STEEL TECHNOLOGY PROGRAM		115
	5.1 Heavy-Section Steel Technology Program	04 60 80 03 1	117
	5.2 Experimental and Analytical Investigations of Nozzles	04 60 70 01 1	124
	5.3 Design Criteria for Piping, Pumps, and Valves	04 60 80 03 1	129
6.	RDT STANDARDS PROGRAM		133
	6.1 Program Objectives and General Activities	04 60 80 03 1	135
	6.2 Revision of Existing Standards	04 60 80 03 1	138
	6.3 Standards for Components of Water-Cooled Reactor Primary System	04 60 80 03 1	141
	6.4 Electrical, Instrumentation, and Control Standards	04 60 80 03 1	145
	6.5 Quality Assurance	04 60 80 03 1	147
7.	GENERAL NUCLEAR SAFETY STUDIES		149
	7.1 HTGR Safety Program Office	04 60 70 01 1	151
	7.2 Fuel Transport Safety Studies	04 60 80 03 1	153
	7.3 Discussion Papers on Various Aspects of Water-Cooled Reactor Safety	04 60 80 03 1	156
	7.4 Antiseismic Design of Nuclear Facilities	04 60 80 01 1	158

		Activity No.	Page
8.	NUCLEAR SAFETY INFORMATION		161
8.1	Nuclear Safety Information Center	04 60 70 01 1	163
8.2	Computer Handling of Reactor Data – Safety (CHORD–S)	04 60 70 01 1	166
8.3	Technical Progress Review Nuclear Safety	07 13 02 02	169

ABSTRACT

The accomplishments during the months of July and August in the research and development program under way at ORNL as part of the U.S. Atomic Energy Commission's Nuclear Safety Program are summarized. Included in this report are work on various chemical reactions, as well as the release, characterization, and transport of fission products in containment systems under various accident conditions and on problems associated with the removal of these fission products from gas streams. Although most of this work is in general support of water-cooled power reactor technology, including LOFT and CSE programs, the work reflects the current safety problems, such as measurements of the prompt fuel element failure phenomena and the efficacy of containment spray and pool-suppression systems for fission-product removal. Several projects are also conducted in support of the high-temperature gas-cooled reactor (HTGR). Other major projects include fuel-transport safety investigations, a series of discussion papers on various aspects of water-reactor technology, antiseismic design of nuclear facilities, and studies of primary piping and steel pressure-vessel technology. Experimental work relative to pressure-vessel technology includes investigations of the attachment of nozzles to shells and the implementation of joint AEC-PVRC programs on heavy-section steel technology and nuclear piping, pumps, and valves. Several of the projects are directly related to another major undertaking; namely, the AEC's standards program, which entails development of engineering safeguards and the establishment of codes and standards for government-owned or -sponsored reactor facilities. Another task, CHORD-S, is concerned with the establishment of computer programs for the evaluation of reactor design data. The recent activities of the NSIC and the Nuclear Safety journal in behalf of the nuclear community are also discussed. In this issue the reports presented have been regrouped by subject matter; however, for the benefit of program sponsors, the responsible AEC Division and Activity number are also listed in the Table of Contents.

SUMMARY

1. Fuel Failure and Behavior of Accident-Released Fission Products

Comparison of Real and Simulated Fission-Product Aerosols

Both the in-pile and out-of-pile experiments in this program were completed, and analytical results of the last two tests are being compiled and evaluated. Preliminary results of the CRI experiments indicate similar behavior of tellurium and cesium in both real and simulant fission-product aerosols. A comprehensive evaluation of all data is being undertaken in an attempt to define the role and limitations of the use of simulants in the Containment Systems Experiment.

Failure Modes of Zircaloy-Clad Fuel Rods

A program document defining both present understanding and the additional information necessary to evaluate the effects of the behavior of Zircaloy-clad fuel rods during a loss-of-coolant accident was prepared. Additional information is needed in the following four areas: (1) effect of metallurgical variables on high-temperature properties of Zircaloy cladding, (2) cladding behavior in simulated loss-of-coolant accident environment, (3) fission-product release upon burst of irradiated Zircaloy-clad fuel capsules, and (4) effect of transients on Zircaloy-clad fuel rods clusters.

Work is under way only in the last area, with the preparation of a seven-rod cluster. Each rod will be pressurized to induce stress-rupture failure of the cladding during a transient in TREAT.

LOFT Assistance Program

This program is being redirected to define the most useful experimental program compatible with the present funding.

Results are reported on previously initiated experiments concerned with the determination of aerosol particle size and deposition behavior following fuel meltdown in the CRI. The particle size distributions and concentrations of two aerosols differing in initial mass concentration

by a factor of 500 were observed as a function of time. In another test series the behavior of the fuel cladding of a highly irradiated Dresden 1 fuel pin in a bundle of six electrically heated rods will be observed during rapid heatup and spray cooling injections. Screening experiments conducted last spring will be repeated under carefully controlled conditions.

2. Filtration and Adsorption Technology

Removal of Radioiodine by Solid Sorbents (Terminal Report)

Studies of the effect of weathering on the performance of impregnated charcoals in trapping elemental iodine and methyl iodide were extended to include additional types of charcoal. Iodized charcoals decreased in CH_3I trapping efficiency by the order of 1% per month under the conditions tested. Triethylenediamine-impregnated charcoal withstood weathering very well, with the removal efficiency decreasing by less than 0.1% per month in 2-in. depths.

Removal of Solid Aerosols

A theoretical model is being developed to estimate the aerosol particle concentration in a steam-filled containment vessel as a function of time after an accident. A system of equations was organized to treat the initial rapid coagulation and tranquil settling, in addition to the processes of diffusiophoresis, thermophoresis, prolonged coagulation, and stirred settling previously treated. Particle size distribution information, needed as input for these calculations, are being obtained from experimental measurement of spark-generated aerosols of stainless steel oxides.

Characterization of Fission Products Under LOFT Conditions (Terminal Report)

A final report was issued.

Ignition of Charcoal Adsorbers by Fission-Product Decay Heat

Effects of water vapor on ignition temperatures of several types of charcoal were observed to be fairly small up to about 50% water vapor,

where ignition begins to be inhibited by lack of oxygen. A reduction of ignition temperature was observed when air sweep was reestablished after prolonged purging at high temperature by water vapor, presumably because of readsorption of oxygen on active sites.

Two new types of charcoal (Witco OF 42 and OF 31) recently observed to be highly promising for trapping methyl iodide were found to have relatively high ignition temperatures.

A computer model is being developed for calculating temperatures in adsorbers loaded with decay heat.

Separation of Noble Gas from Air with a Permselective Membrane

The membrane used in this study was recently identified by the manufacturer (General Electric Co.) as methylphenyl silicone rubber.

Steady-state experiments made possible by equipment improvements showed no difference between cocurrent and countercurrent flow for $Kr-N_2$ and for $Kr-O_2$ separation. The calculated permeability of krypton decreased with cut (amount of gas permeating the membrane) to an extent dependent on carrier gas. The value extrapolated to zero cut is independent, however, and agrees with data for pure gases.

An improved membrane has been developed by GE by optimization of support screens to avoid compression and to promote turbulence. Laboratory tests at ORNL have confirmed the improved performance of this membrane.

High-Efficiency Air-Filtration Engineering Manual

The filtration engineering manual is scheduled for limited distribution in September for comment and review. It will then be reissued as an ORNL-NSIC report.

3. Spray and Pool Pressure-Suppression Technology

Effect of Additives on Distribution of I_2 and CH_3I Between Air and Water

The search for spray solutions conducted at Furman University was completed during this report period. A final report will be issued in

the near future. Preliminary results on a very promising CH_3I getter are reported. The material, a surfactant, appears to greatly increase the base-borate system affinity for CH_3I.

Uptake of I_2 and CH_3I by Water Solutions and Drops

Theoretical considerations have indicated that the additions of a surfactant and a strong reducing agent to the base–borate–sodium thiosulfate solution should improve the CH_3I removal of the resulting spray solution. Subsequent tests with the surfactant (PEI 1000), both with and without a reducing agent, at several temperatures, confirmed the anticipated behavior of both additives, as well as the enhanced effect of higher temperature (from 25 to 70°C).

Spray Studies at the Nuclear Safety Pilot Plant

Extensive system modification was completed in the Nuclear Safety Pilot Plant to allow experiments to be conducted on aerosol removal by sprays. Three experiments were completed that involved (1) I_2 removal with a spray nozzle different than any previously used, (2) melting of stainless steel-clad UO_2 pellets in a system shakedown run to verify ability for aerosol generation, and (3) CH_3I removal with an additive suggested by single-drop studies.

Radiation Stability of Sprays

The study of sulfur production and pH relationship under gamma irradiation for the $Na_2S_2O_3$ system was completed. Sulfur generation will not be a problem if system pH is properly controlled. The combined effects of radiation and temperature were also examined. The results were those anticipated and indicated that the thiosulfate system can be expected to withstand anticipated accident conditions.

Corrosion Tests

Initial runs with the base-borate solution were completed in the facility for corrosion tests. While the majority of construction materials showed little corrosion, the aluminum alloys, except 5052, had to be

removed after 75 hr because of excessive corrosion rates. The 5052 alloy developed a protective coating that seemed to inhibit further attack. The attack on copper was measurable. A second run was started with the thiosulfate system, but results are not yet available.

Pressure-Suppression Experiments

Pressure-suppression work at ORNL was directed to system modification to accommodate larger injection pipes. Data from runs with larger pipes are expected to confirm initial modeling work.

Scale-Model Tests of Fission-Product Removal in Suppression Pools

The work under subcontract to General Electric Company, San Jose, involved equipment design and procurement of a 1/10,000 scale-model pressure-suppression system. Experiments were also designed for studying the adsorption of simulated fission products under simulated loss-of-coolant conditions.

4. Safety Studies for HTGR

Program Development

A revised version of the recommended program for research relating to graphite-steam reactions and their effects on fission-product release and transport under HTGR accident conditions was completed. Available funding will permit continued effort only on the higher priority tasks; namely, (1) investigation of the thermal and radiolytic reactions between HTGR fuel elements and steam and (2) determination of failure mechanism of coated particles at above normal temperatures.

Steam-Carbon Reaction and Fission-Product Release and Transport Studies

The deposition characteristics of ^{110}Ag in a thermal-gradient tube were being studied as a means for interpreting the release and transport behavior of fission products from graphite surfaces. Reasonable analytical correlation was obtained, but funding limitations will restrict future

activities in this task to the study of steam-graphite reaction rates. Work is being done on an analytical model and gaseous diffusion apparatus to study the transport of steam through the graphite, since this appears to be a critical process.

Engineering-Scale Steam-Graphite Reaction Rate Experiment

This task was cancelled. An analytical model describing the reaction processes in both graphite and steam, and including the effects of recirculation and removal of water vapor from the loop by a cleanup system, was developed and will be used elsewhere in the program.

High-Temperature In-Pile Fuel Integrity Tests

The experiment in the ORR B9 for determining the dependence of failure of fuel coating on temperature is progressing satisfactorily. After attaining the planned burnup at a normal HTGR temperature the temperature was elevated in steps. Only preliminary results are so far available.

High-Temperature Behavior of Gasborne Fission Products

The investigation of the enhancement of the vapor pressure of fission-product species through their conversion to oxides in the presence of water vapor was concerned with the Te, O, H_2O system. Additional work is necessary to clarify the mechanism for the observed vapor pressure enhancement, but the task has been terminated because of lack of funds.

In-Pile Studies of Reactions of Fueled Graphite with Steam Under Accident Conditions

Design and construction of equipment for the in-pile work is continuing.

5. Pressure Vessel and Piping Technology

Heavy-Section Steel Technology Program

Most HSST program activities planned for FY-69 are in progress. Work is under way on five subcontracts and four are being negotiated. Three

or four more subcontracts are to be initiated this fiscal year. Currently the program outlined in the program plan is on schedule.

Numerous size-effects results from the fracture mechanics and transition-temperature studies are expected within a few months, and initial results are currently available. With these results in hand, the final program directions and plans can be made.

Experimental and Analytical Investigations of Nozzles

A comprehensive and up-to-date description of the research work performed on the problems of radial nozzles attached to both spherical and cylindrical shells was published. The report contains comparative discussions of various research projects and recommends new rules for the design of reinforced openings in nuclear pressure vessels. These recommended rules are being considered by the ASME pressure vessel code bodies. Experimental and analytical work is continuing on both single nozzles and cluster in both spheres and cylindrical shells under various loading conditions. Computer programs were recently developed for a single nonradial cylindrical nozzle in a spherical shell and a single radial nozzle in a cylindrical shell.

Design Criteria for Piping, Pumps, and Valves

Both analytical and experimental work are in progress under the ORNL portion of the joint AEC-PVRC program to develop stress indices and flexibility factors for nuclear-service piping, pumps, and valves. The literature survey and technical evaluation report are nearing completion. Some of the analytical work documented in the literature is being used to develop design charts, etc., that can be used directly by the code-writing bodies.

The dimensional analysis work on commercial pipe fittings is providing needed insight for defining analytical stress studies on the effects of geometric deviations. Preliminary results from the finite-element analysis and the test data on the thin-shell tee model are quite encouraging. On the basis of these results it is planned to proceed with an extension of the analysis for B16.9 tees.

6. RDT Standards Program

Program Objectives and General Activities

Planning and development of lists of proposed standards, scopes, work arrangements, and programs continued. Schematic diagrams are being developed to depict typical BWR, PWR, and research reactors, with identification of areas, systems, and components for which specific RDT standards exist or are being planned. Manpower problems in obtaining the necessary expertise are being solved. Organization of the administrative process for preparing standards, which consists of soliciting and handling technical data, disseminating information, enlisting the cooperation of AEC contractors and industrial organizations, and reproducing and distributing documents, has been time-consuming because of the rapid growth of interest and scope of the Standards Program.

Revision of Existing Standards

Seven revised standards have been submitted to RDT and 52 have now been issued as tentative standards. Revision of other existing "standards" continues at a steady pace, while a comprehensive listing of subjects proposed for standardization is being developed. A standard index system is also being developed.

Standards for Components of Water-Cooled Reactor Primary System

Many components of the primary systems of water-cooled reactors are being examined with the intent of developing standards covering critical aspects of these components. This work is being performed by various subcontractors on components such as reactor internals, pumps, valves, piping, and heat exchangers. Specific developments include a preliminary draft standard for reactor internals design and continued work on the development of draft standards for each of pump design, valve design, and piping design. A draft report on the heat exchanger tube vibration survey was completed and distributed for comment.

Electrical, Instrumentation, and Control Standards

An ad hoc committee was established to guide the work on reactor protection system standards, which is intended to supplement the existing drafts of both AEC and IEEE criteria. Some work was also initiated on electric cables for nuclear installation and process instrumentation.

Quality Assurance

The program on quality assurance was approved by the Commission. It will consist initially of the preparation of two documents: (1) a description of quality-assurance programs and (2) a description of quality-assurance systems requirements.

7. General Nuclear Safety Studies

HTGR Safety Program Office

The Program Office is preparing a revised version of the HTGR safety plan and has undertaken special reviews of graphite oxidation research and ACRS comments on the Fort St. Vrain plant.

Fuel Transport Safety Studies

ORNL and the AEC are cosponsoring the Second International Symposium on Packaging and Transportation of Radioactive Materials. This meeting to be held the week of October 14 in Gatlinburg has an agenda of 55 papers from authors throughout the world.

Work is continuing on the draft of the Shipping Cask Criteria Manuals. Most of the writing on Chapter 2, Engineering Standards and Guide to the Design of Spent Fuel Shipping Casks, was completed during the past two months.

Consulting with AEC contractors regarding problems of shipping casks continues to constitute a significant fraction of project effort. Casks for seven different facilities were reviewed during the reporting period.

Discussion Papers on Various Aspects of Water-Cooled Reactor Safety

Six of the eight original discussion papers and one supplemental report were completed and published. The two outstanding papers, Design Principles of Reactor Protection Instrument Systems and Earthquakes and Nuclear Power Plant Design, exist in draft form but are still being reviewed.

Antiseismic Design of Nuclear Facilities

The draft of the plan for an antiseismic research and development program was revised and will be released to the Commission before the end of September.

Shaking tests were conducted at the EGCR and CVTR with two 5000-lb shaking machines by a task group from UCLA. Preliminary results indicate that some (at least) resonant frequencies are predictable. However, some surprisingly large amplification factors were observed.

Results of conceptual studies of designs to provide resistance to seismic effects of up to 1 g horizontal and up to 6 ft of vertical and horizontal slip were completed and distributed. Cost differentials range from 1.2 to 15.3 millions of dollars for a typical 1000-Mw(e) plant, but further development is required in all cases, except the floating concept.

8. Nuclear Safety Information

Nuclear Safety Information Center

The Center's storage file now contains computer retrievable information on over 22,000 documents, and new input is scanned biweekly for over 1300 users who receive SDI.

The remote consoles (IBM 2740) serving NSIC offices are being augmented by the addition of two IBM 2260 (CRT) for which programs are being debugged.

The Management File for the computer storage and retrieval of information on nuclear safety research and development is being shaken down and data storage initiated.

NSIC is preparing a booklet on nuclear power education and training.

Computer Handling of Reactor Data – Safety

The principal effort in the CHORD–S project has been to complete the characteristics listing in the "Summary" category and to include specific data on ten power reactors from the information in the PSAR's and amendments. The work has been completed except for one subsection.

Operation of the remote console (IBM 1050) in Bethesda that is coupled to the Computing Technology Center (IBM 360 combination 50 and 65) in Oak Ridge is being debugged to provide more reliable service, and the computer program is being modified to permit more precise use by DRL.

Studies of hardware capability, compatibility, and costs as related to the needs of CHORD–S are continuing.

Technical Progress Review – Nuclear Safety

Nuclear Safety Vol. 9, No. 5, was issued, and the requisite review, editing, printing, etc. are under way on issues 9(6), 10(1), and 10(2).

1. FUEL FAILURE AND BEHAVIOR OF ACCIDENT-RELEASED FISSION PRODUCTS

1.1 COMPARISON OF REAL AND SIMULATED FISSION-PRODUCT AEROSOLS

(AEC Activity 04 60 10 01 1)

S. H. Freid B. F. Roberts
G. W. Parker

One of the important objectives of the fission-product release program is the determination of the extent to which stable and trace nuclides are capable of simulating the release and transport behavior of high-level fission products. The degree of simulation is of particular importance in the Containment Systems Experiment where the size and nature of the facility preclude the release of actual fission products in realistic accident concentrations. A two-phase test program is being conducted in which (1) the in-pile fission-products meltdown facility will be operated both with and without the simulants in addition to real fission products from a high-burnup Zircaloy-clad UO_2 fuel piece and (2) in out-of-pile experiments, the behavior of fission products released from the melting of irradiated fuel will be compared with the behavior of simulant aerosol. The experiments have been run, and the results are being analyzed. All results of this and previous ORNL simulation experiments are being compiled in a summary report.

In-Pile Release Tests

The first half of a two-experiment program for the verification of a fission-product simulation technique was described in the previous report.[1] The verification program consists of comparing the transport and plateout behavior of stable nuclides and their fission-product analogs under identical environmental conditions. The second experiment was completed in July. Radiochemical analyses of samples are still under way, and the data are being evaluated.

Out-of-Pile Release Tests

The behavior of radioactivity released by the meltdown of a highly irradiated fuel capsule is being compared in the Containment Research Installation (CRI) with the behavior observed when the fission products

are vaporized from separate heater filaments, as in the Containment Systems Experiment. The results of a reference simulant behavior experiment, CRI run 107, are to be compared with the results obtained with a real fission-product source in CRI run 114. Conditions of the release in run 114, given in Table 1.1, were adjusted as closely as possible to match those previously used in run 107. The quantities of fission products released in run 107 were chosen to approximate those expected from the high-burnup capsule that was at that time expected to be used in the comparison run. The departure from this plan was that the burnup level actually found in the capsule was nearer 7000 Mwd/T than the 10,000 Mwd/T calculated for use in run 107.

The results of run 107 were previously published[2] and compared with those of subsequent CRI tests. The complete analysis of run 114 is not available at this time; however, preliminary data are shown in Fig. 1.1 for the comparison of time-space behavior of the aerosols in the two runs. The relative attenuation rates are almost identical and thus justify the conclusion that no significant difference exists between the aerosols of cesium and tellurium generated by the two methods. We do not expect radioiodine behavior to compare quite so favorably, since iodine is much more susceptible to minor environmental differences.

Table 1.1. Conditions for CSE Fission-Product Simulant Validation Tests in CRI

	Run 107 with CSE Simulant	Run 114 with 7000-Mwd/T Burnup Material
Atmosphere	Air-steam	Air-steam
Aging time, hr	20.6	20.2
Pressure, atm abs	3.3	3.6
Temperature, °C	110.5	111.0
Cladding on UO_2 pellets	Zircaloy-2	Zircaloy-2

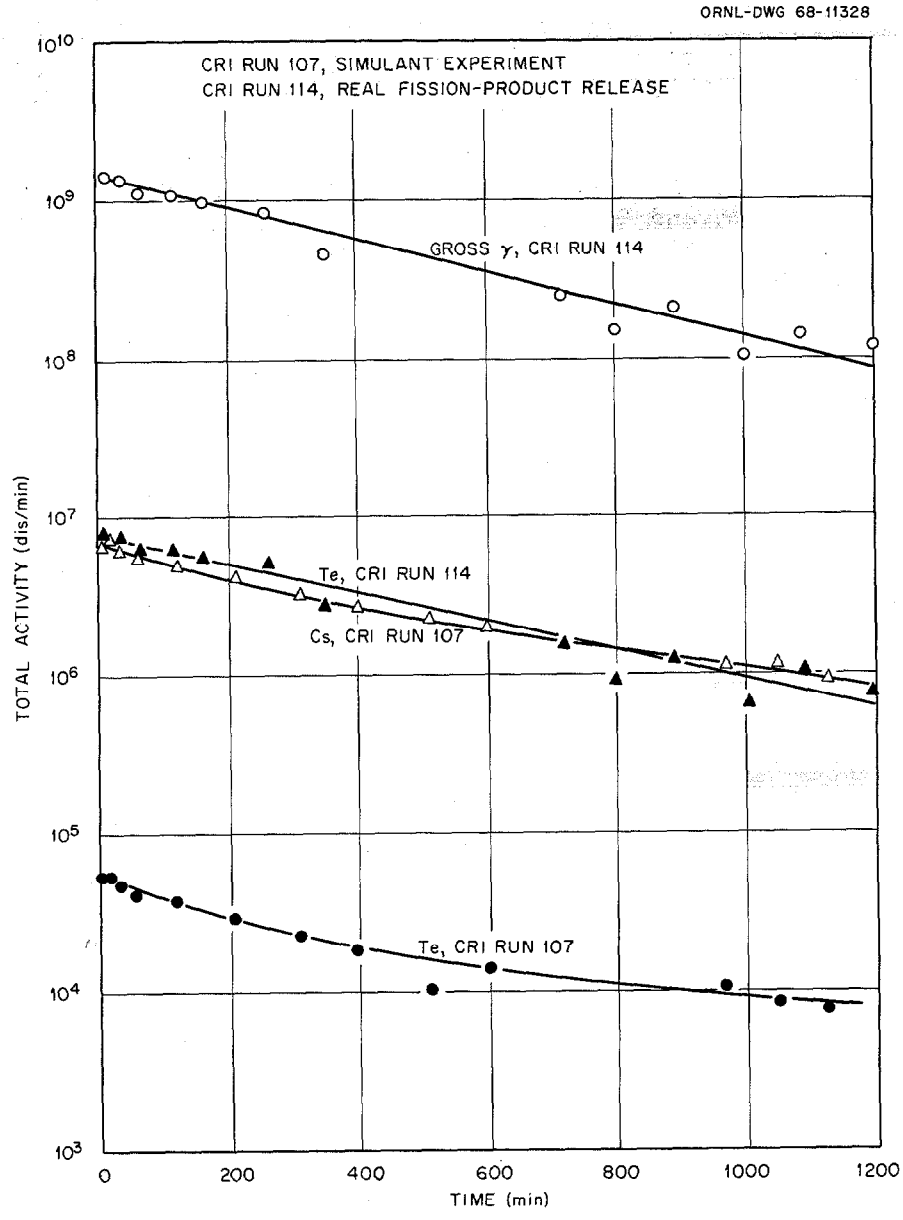

Fig. 1.1. Comparison of Aerosol Behavior in Simulant and Real Fission-Product Releases in CRI.

References

1. S. H. Freid, B. F. Roberts, and O. Sisman, p. 7 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1968, USAEC Report ORNL-TM-2283, Oak Ridge National Laboratory, July 30, 1968.

2. G. W. Parker, G. E. Creek, and W. J. Martin, p. 8 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for September–October 1967, USAEC Report ORNL-TM-2057, Oak Ridge National Laboratory, Nov. 27, 1967.

1.2 FAILURE MODES OF ZIRCALOY-CLAD FUEL RODS

(AEC Activity 04 60 10 01 1)

P. L. Rittenhouse

It is possible that the failure behavior of the fuel cladding during a loss-of-coolant accident in a water-cooled reactor may alter the thermal response of the core and adversely affect the course and consequences of the accident. Our present understanding of the events occurring during the accident and the properties and behavior of the fuel rods subjected to the accident is not sufficient to justifiably support or deny this contention. The information needed to permit a judgment falls into four categories.

1. Effect of Metallurgical Variables on High-Temperature Properties of Zircaloy Cladding. We need to know the high-temperature properties of Zircaloy cladding and how these properties are affected by environmental and material variables.

2. Cladding Behavior in Simulated Loss-of-Coolant Accident Environment. We need to determine the effect of fuel rod dimensional changes on the efficiency of emergency cooling. Included under dimensional changes should be cladding swelling and perforation; fuel rod bundle distortion, including buckling, twisting, etc.; and formation of rubble by fragmentation of fuel rods.

3. Fission-Product Release Upon Burst of Irradiated Zircaloy-Clad Fuel Capsules. We need to know the effect of heating rate, internal pressure, and fuel cladding rupture, etc., on the fission-product fraction released, as well as the chemical and physical states of the nuclides released. Hot-cell burst tests of fuel capsules will be required for these determinations.

4. Effect of Transients on Zircaloy-Clad Fuel Rod Clusters. We need to determine the effects of temperature, environment, and material history on the integrity of the fuel rods during reactor transients. The necessary tests can be performed in TREAT.

Program Development

The direction and extent of the research effort needed in the first three of these areas is as yet not fully determined. AEC and industry response to the program document[1] distributed in September will be needed for decisions. Research on the effect of transients on Zircaloy-clad fuel rod clusters is to be undertaken in TREAT, as described in the following section.

Transient Tests of Zircaloy-Clad Fuel Rod Clusters in TREAT

The TREAT experiments are designed for studying fuel rod behavior during a loss-of-coolant accident in a PWR or BWR. A seven-rod cluster of 27-in.-long Zircaloy-clad UO_2 rods will be used in each experiment. In the first test the center rod will contain fission products from a 900-Mwd/T burnup irradiation in the MTR at 14.2-kw/ft heat rating. This rod was prepressurized to 215 psia with helium (25°C) to simulate the released fission gas expected for a burnup of 10,000 Mwd/T. The six unirradiated rods have internal pressures ranging from 65 to 373 psia. The TREAT reactor will operate at steady power for about 17 sec so that the fission heat in the rods will raise the fuel and cladding temperature 70°C/sec from an initial temperature of 280°C. Table 1.2 lists the fuel rod characteristics and their predicted rupture temperature.

Table 1.3 summarizes the irradiation conditions expected in the MTR. During the loss-of-coolant accident test in TREAT a flowing gas mixture of 11 liters/min of steam plus 2 liters/min of helium will carry fission products released from the center rod into two sequentially operated collection systems. Irradiation of the center rod should begin in September. After about four weeks of decay the rod will be installed in the main experiment package and the TREAT experiment will be performed.

Reference

1. P. L. Rittenhouse, Failure Modes of Zircaloy-Clad Fuel Rods, USAEC Report ORNL-TM-2347, Oak Ridge National Laboratory, Sept. 13, 1968.

Table 1.2. Characteristics of Zircaloy-Clad UO_2 Fuel Rods for Transient Tests in TREAT

Rod No.	Burnup (Mwd/T)	Type[a]	Total Void Volume (cm^3)	Helium Plus Fission Gas Volume [cm^3 (STP)]	Helium Plus Fission Gas Pressure at 25°C (psia)	Predicted Pressure at Rupture (psia)	Predicted Temperature at Rupture	
							°C	°F
Center	900	Sealed	2.63	36.6	224	750	727	1340
1	0	Sealed	3.13	41.9	215	725	732	1350
2	0	Sealed	3.47	24.9	115	425	827	1520
3[b]	0	Sealed	3.90	15.8	65	262	927	1700
4	0	Pressure monitored	7.8	180	373	750	727	1340
5	0	Pressure monitored	7.6	97	207	425	827	1520
6	0	Pressure monitored	7.1	56	127	262	927	1700

[a]All rods are clad with 0.568-in.-OD Zircaloy-2 of 0.033-in. wall thickness.

[b]Four platinum-rhodium thermocouples will be spot-welded to the cladding of this rod.

Table 1.3. Summary of Irradiation Conditions in MTR for Fuel Rods To Be Tested in TREAT

	Conditions		
	Peak	Average	Total
Time (18 days), sec			2.08×10^6
Fission rate			
Fissions/sec			1.062×10^{15}
Fissions/cm^3·sec	1.51×10^{13}	1.30×10^{13}	
Fission density, fissions/cm^3	3.14×10^{19}	2.7×10^{19}	
Total fissions			2.21×10^{21}
Burnup, Mwd/T	1046	900	
Fission power density, w/cm^3	435	374	
Fission linear power			
w/cm	502	432	
kw/ft	16.46	14.16	
Total fission power, kw			30.65
Total experiment linear power, kw/ft	17.10	14.63	
Fuel enrichment, %		1.517	
UO_2 volume, cm^3			82.1
UO_2 density, g/cm^3		10.38	
Oxygen-to-uranium ratio		2.001	
Measured void volume, cm^3			2.63
Helium pressure, psia at 25°C			215
Helium volume, cm^3 (STP)			35.2
Helium purity, % He			99.992
Rare gas diffusion to voids, cm^3 (STP)			1.4
Radiation through 6 in. of lead after 30 days of decay, mr/hr at 1 m			60

1.3 LOFT ASSISTANCE PROGRAM
(AEC Activity 04 60 50 05 1)

G. W. Parker G. E. Creek
W. J. Martin

Numerical data are being obtained to aid in the development and evaluation of analytical models for predicting release and transport behavior of fission products and to demonstrate the reliability and effectiveness of emergency core cooling for limiting release from the core. While the results reported below represent activities in the support program, which was revised to comply with the current objectives of the LOFT program, the work is being further redirected in view of the limited funding available. Discussions are being held with both the AEC and Phillips staffs to define the program for the remainder of this fiscal year.

Determination of Particle Size and Deposition Behavior from Fuel Meltdown in CRI

In order to establish the correlation between fission-product radioactivity and the solid disperse phase in a reactor-accident-generated aerosol, both for the rate of natural attenuation and the expected filter and spray-removal processes, it has been necessary to classify particles and determine concentrations and type of associated radioactivity with more precision than has previously been possible. The ORNL low-pressure cascade impactor has been successfully applied to the problem of unambiguously correlating airborne radioactivity (radioiodine, as well as solid fission products) with the mass and removal rate of the vaporized fuel components.

In the fuel-melting studies carried out in the CRI, in order to compare the effect of an increase in aerosol concentration over the range of the various accident regimes pertinent to LOFT, two cases were analyzed by means of the Mark-II impactor. In the first case (run 112) a low concentration of U_3O_8 (2 mg/m^3) was supplied, and in the second case (run 113) almost 1 g/m^3 was provided. Smaller particle sizes were observed in the lower concentration U_3O_8 aerosol, as indicated by the plots of relative concentration versus impactor stage number in Fig. 1.2. The

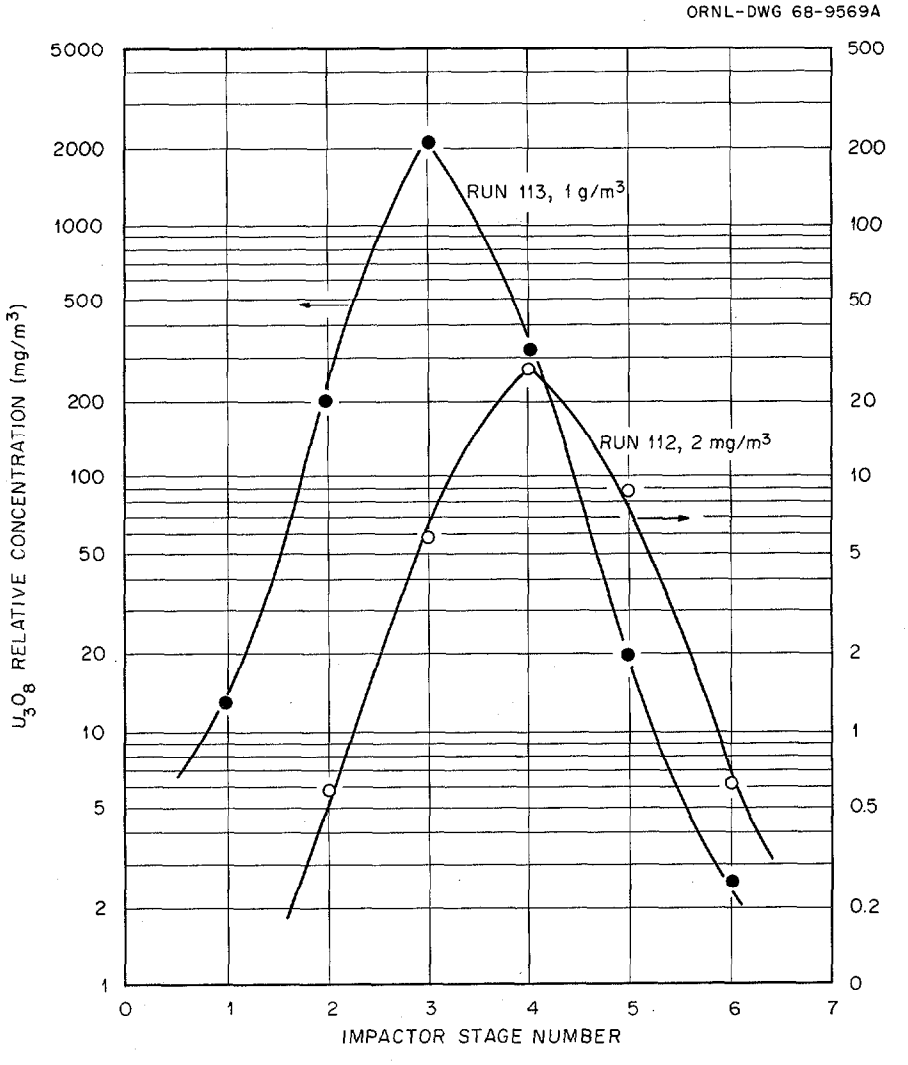

Fig. 1.2. Classification of Uranium Oxide Particles at Two Concentration Levels in CRI Runs 112 and 113.

particle size is in inverse relation to the impaction stage number, with stage number 6 collecting the smallest particles.[1] As a test of the agglomeration processes a total particle number count was made during the same time that the particle concentration and size analyses were made. These results for run 113 are shown in Fig. 1.3. The change in particle diameter with time is shown in Fig. 1.4. In the dilute aerosol, the change in mass median diameter decreased from 0.13 μ to about 0.05 μ after 30 hr. In the higher concentration aerosol, the decrease was from 0.40 to 0.17 μ in the same time.

Analysis of the CRI Aerosol by the Effect of Stirred Settling

In both the SnO_2 and U_3O_8 aerosols, an analysis of the settling behavior was applied to the gas-phase concentration data. Particle sizes (Stokes, d_s) calculated from the rate of aerosol settling are in fair agreement with particle sizes measured with the low-pressure cascade impactor in this study.

The correlation for stirred settling given in the "AEC Handbook on Aerosols" by Sinclair[2] and by Green and Lane[3] was used:

$$-\frac{d}{dt}\log M_t = 1.3 \times 10^5 \frac{\rho}{h} d_s^2 , \qquad (1)$$

where

M_t = mass concentration at time t,
ρ = particle density,
h = settling height.

The data comparisons are given in Tables 1.4 and 1.5. Note that d_s is the diameter calculated from stirred settling rates; d_6 (the mass median diameter) and d_g (the count median diameter) were then calculated from d_s by assuming values of σ and using the equation

$$\log d_m = \log d_g + 2.303 \frac{m}{2} \log^2 \sigma .$$

The differences between d_s, d_6, and d_g depend on σ, and the equations are

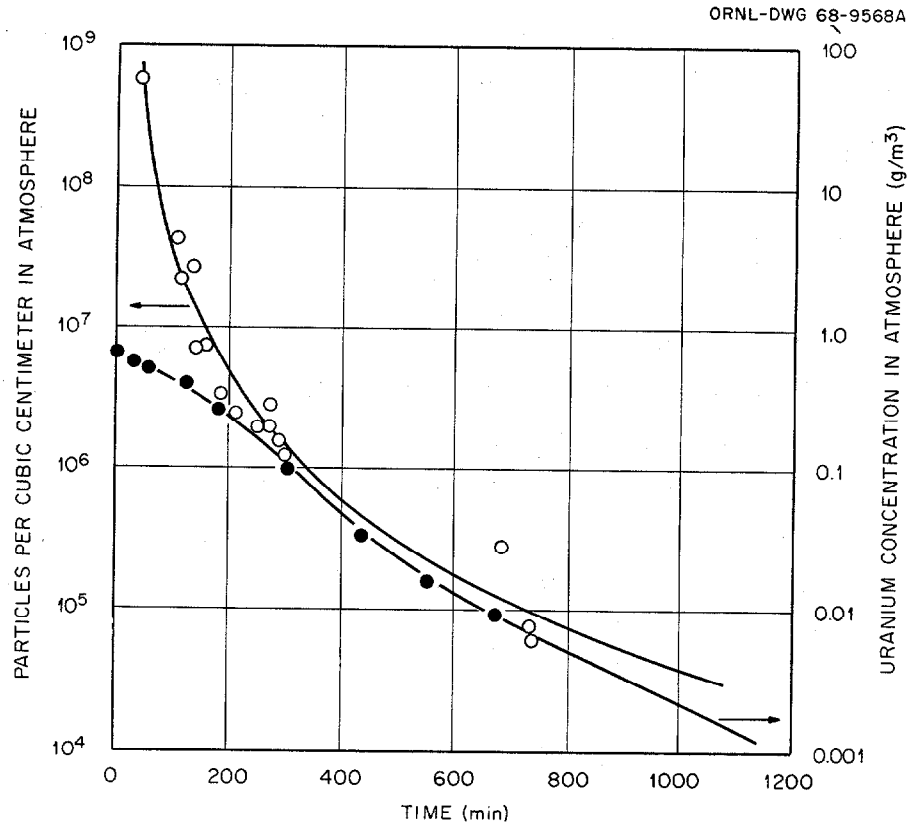

Fig. 1.3. Change in Aerosol Concentration with Time in Run 113.

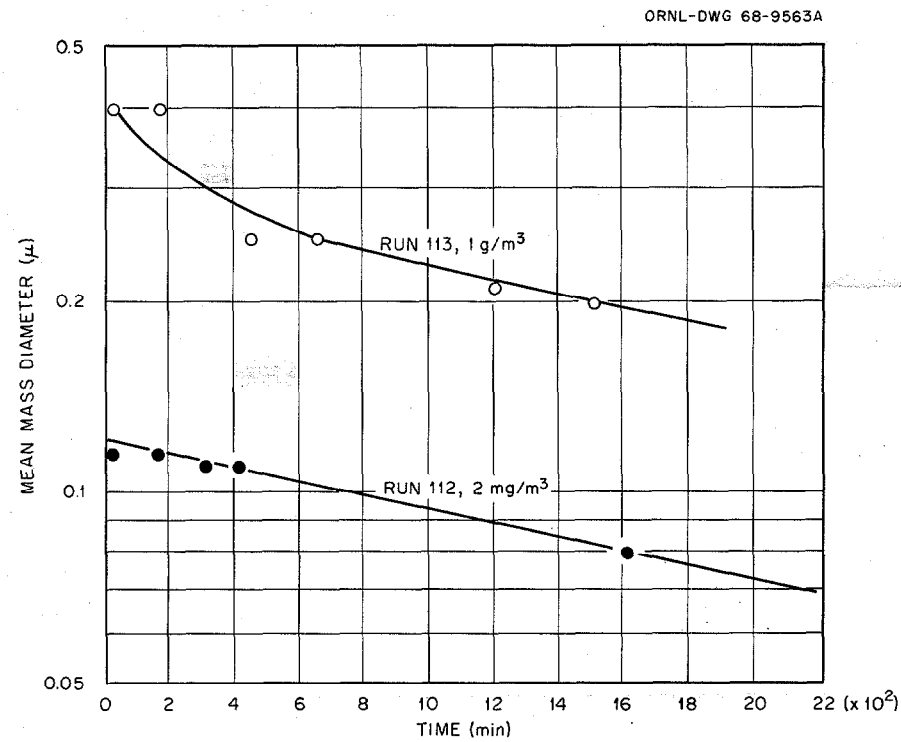

Fig. 1.4. Change in U_3O_8 Particle Diameter with Time in Aerosol in CRI Runs 112 and 113.

Table 1.4. Comparison of Sizes of SnO_2 Particles Calculated from Settling Velocity and Impactor Analysis

Test No.	Calculated Particle Diameter (μ)[a]			Mass Median Diameter Measured with Low-Pressure Impactor, d_6
	d_8 (b)	d_6 (c)	d_g (d)	
CRI run 107, initial slope	0.86	0.61	0.216	0.31 (σ = 1.8)
CRI run 108, constant slope	0.75	0.53	0.188	0.55 (σ = 1.7)

[a]Standard deviation, σ, assumed to be 1.8.

[b]Diameter calculated from stirred settling rates.

[c]Calculated mass median diameter.

[d]Count median diameter from electron microscope.

accurate only for spherical particles that have a log-normal size distribution. The tables also show the values of mass median diameter (d_6) and σ as measured by the low-pressure cascade impactor.

By sampling frequently with the Mark-II impactor a gradual change in the aerosol characteristics, including the standard deviation, is noted. Figure 1.5 shows the results of two analyses over the time intervals of most importance for the two uranium oxide test aerosols (i.e., 12 to 15 min for sample 1 and 180 to 430 min for sample 4).

A similarly useful analysis has been proposed by Adams and his associates[1] based on the concept that agglomeration and other natural processes tend to preserve the natural size distribution of the particles. In noting the change in mean mass diameter over the essential life span of the aerosol (reduction of 10^3), the "self-preserving" size rule proposed by Davis is not seriously in conflict with our observations. The size (Fig. 1.4) appears to remain relatively constant for a period needed to reduce the concentration by about 100 (Fig. 1.3) before the smaller sizes and lower settling rates are noted.

Table 1.5. Comparison of the Size of U_3O_8 Particles Calculated from Settling Velocity and Impactor Analysis

Time Interval (min)	Half Time (min)	Settling Velocity (cm/sec)	Calculated Particle Diameter (μ)			Measured Values[a]	
			d_g, Mean if $\sigma = 2.0$	Mass Median if $\sigma = 2.0$	Mass Median if $\sigma = 2.5$	d_6	σ
0–120	140	1.65×10^{-2}	0.74	0.46	0.32	0.40	1.8
180–430	85	2.72×10^{-2}	0.96	0.60	0.42	0.30	2.3
1200–1550	200	1.15×10^{-2}	0.61	0.38	0.265	0.20	2.2

[a] From impactor analysis.

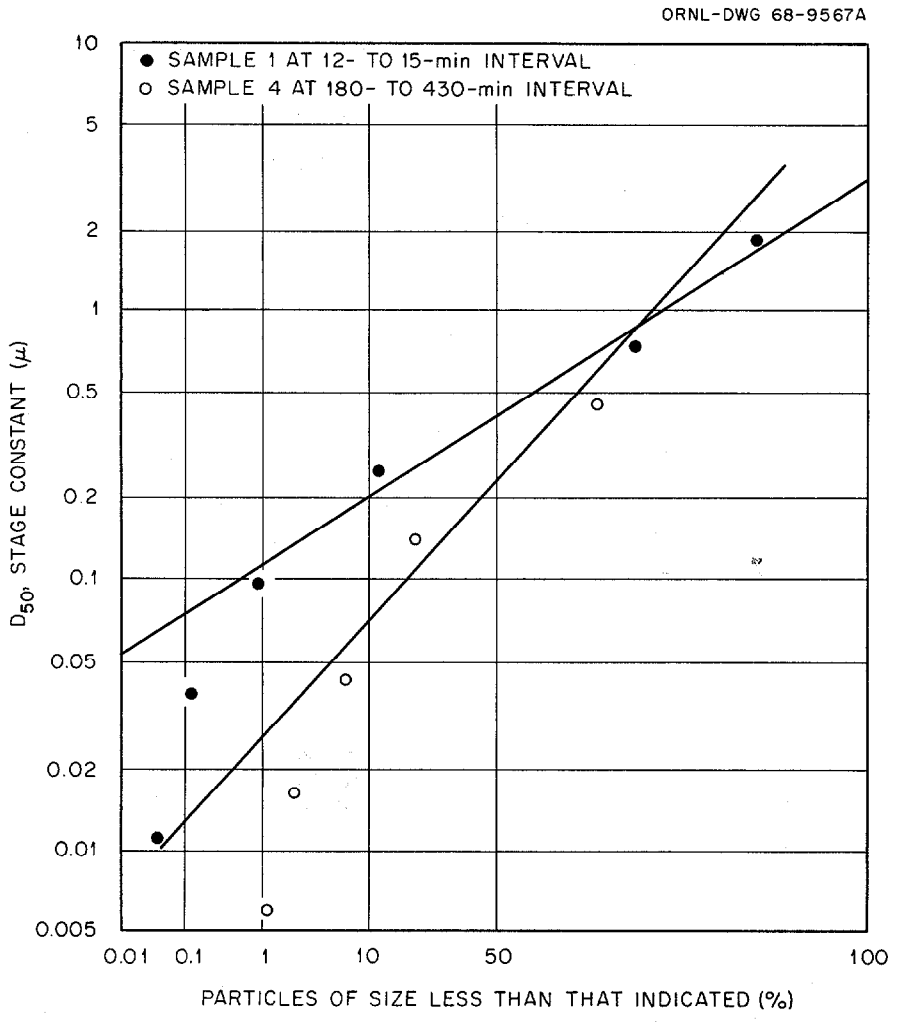

Fig. 1.5. Size Distribution of U_3O_8 Particles in CRI Run 113.

Extrapolation of Results to Higher Aerosol Concentrations

Based on observed deposition behavior in two runs (112 and 113), a three-part rate equation was derived that fits the data. As an additional exercise, a higher concentration was assumed for a third case and the corresponding behavior was calculated. The results for a unit height of containment are shown in Fig. 1.6. The time scale should be multiplied by 2 for the CRI (2-m height). For a full-sized containment vessel, the factor is about 30 (30-m height).

Effect of Emergency Core-Cooling Spray on the Integrity of Reactor Fuel Rods and Resultant Fission-Product Behavior

One of the programs of highest priority in the LOFT project, as well as in the Zircaloy fuel rod failure program (see Sect. 1.2), is the assessment of damage to the reactor core coincident with the injection of emergency cooling. While the fuel failure program is oriented toward the assurance of sufficient integrity of the core to permit effective cooling, the LOFT assistance program is directed toward the evaluation of the effect of physical damage on the fission-product release and transport processes.

The initial design of a system for the failure of a highly irradiated fuel rod of the Dresden I type in a six-member electrically heated bundle was previously reported.[4] Several rapid heatup and spray-cooling injection demonstrations were conducted[5] with a preliminary mockup of the fuel-rod bundle. In order to accomplish the real objective of the fuel-rod failure test, however, a more carefully engineered design for remote loading and handling of a highly irradiated fuel rod in an assembly was required. An artist's concept of the remote system is shown in Fig. 1.7. The six rods are indicated in a shrouded assembly with the BWR reactor design ratio of open-to-closed surface in the tube sheets (the upper plate of the assembly). A similar design is included in the bottom plate. The directed spray is sized to conform to the coolant delivery rate expected in the BWR system. Alternate flooding or a combination of spray and flooding is also permissible in the CRI primary simulator vessel.

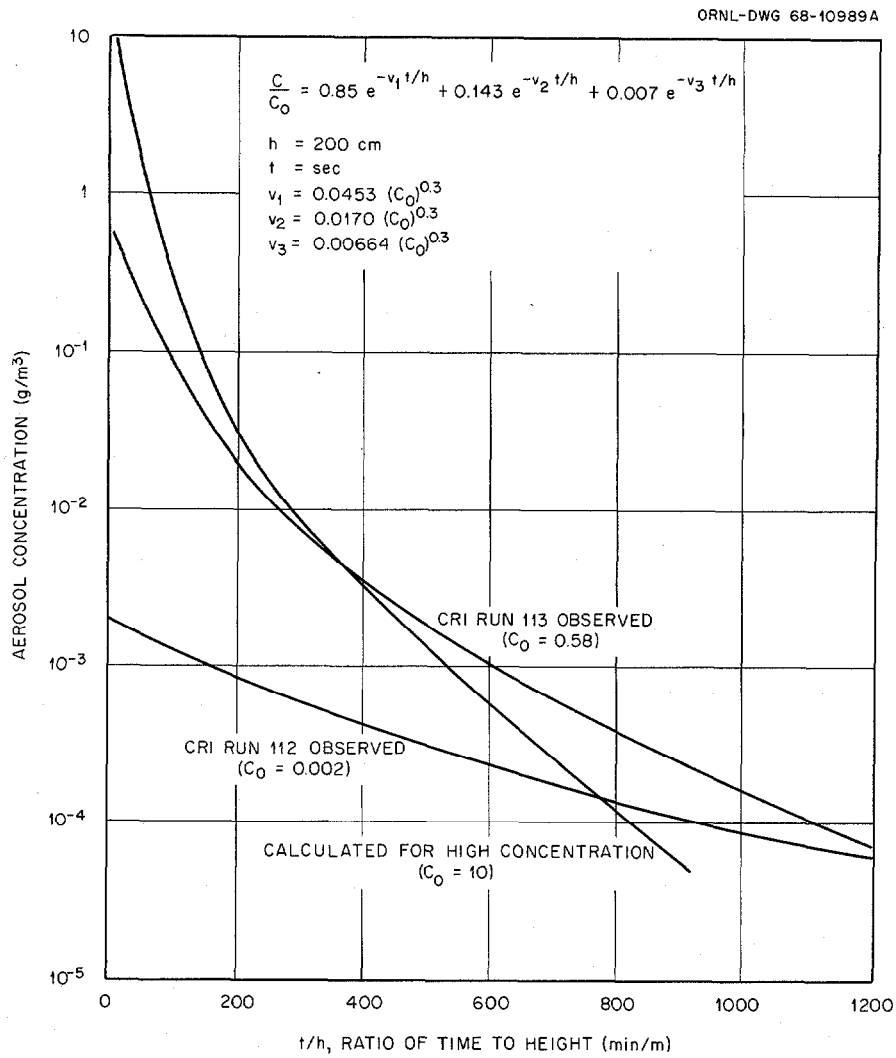

Fig. 1.6. Observed and Calculated Concentrations of U_3O_8 in CRI Aerosol.

ORNL DWG 68-9932R

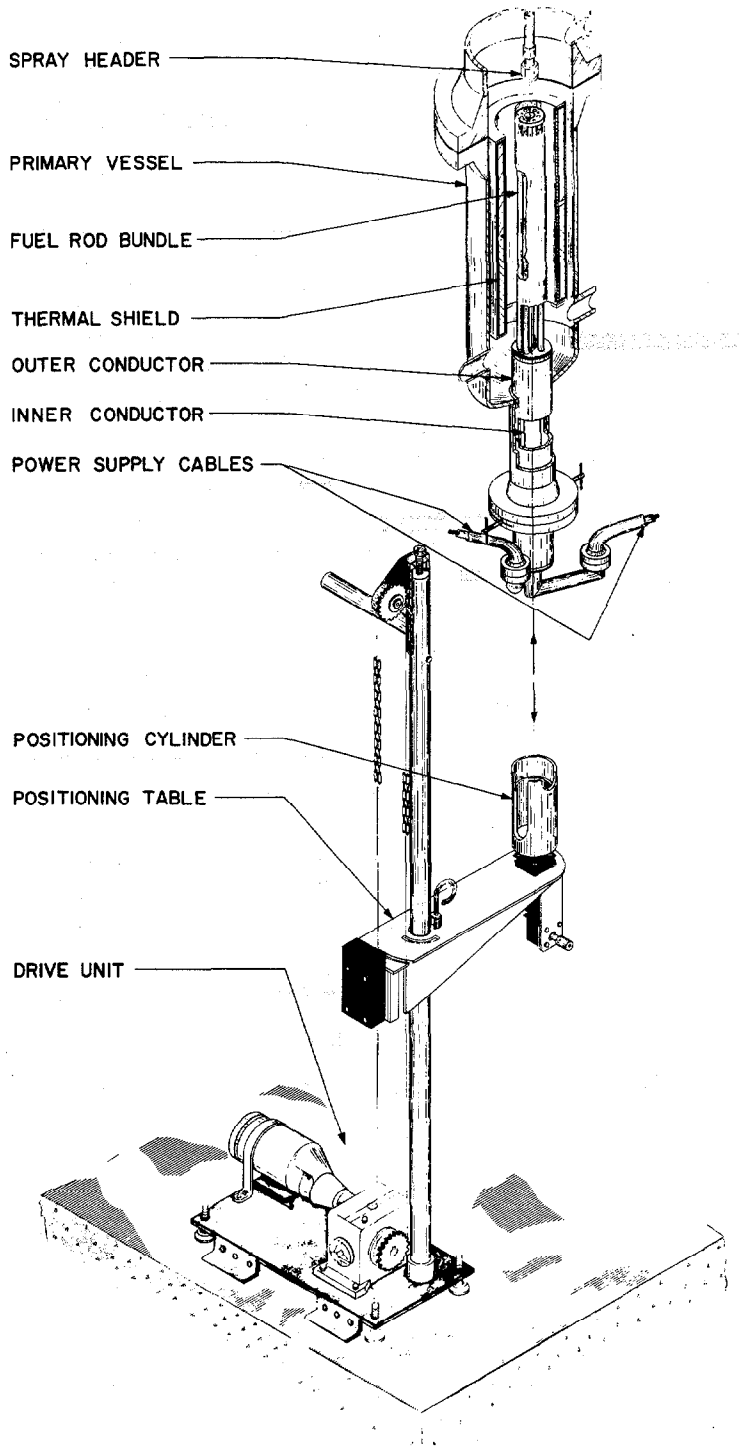

Fig. 1.7. Equipment for CRI Rod Failure Experiment.

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2. D. Sinclair, Stability of Aerosols and Behavior of Aerosol Particles, Chapt. 5, pp. 64–76, AEC Handbook on Aerosols, USAEC Report STR-10-1, H. F. Johnstone (Ed.), 1963.

3. H. L. Green and W. R. Lane, Particulate Clouds: Dusts, Smokes, and Mists, Van Nostrand, Princeton, New Jersey, 1964.

4. G. W. Parker and W. J. Martin, LOFT Assistance Program, pp. 65–66 in Nuclear Safety Program Ann. Progr. Rept. Dec. 31, 1967, USAEC Report ORNL-4228, Oak Ridge National Laboratory.

5. G. W. Parker et al., Fission-Product Release and Transport Under LOFT Conditions, p. 27 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for March–April 1968, USAEC Report ORNL-TM-2230, May 30, 1968.

2. FILTRATION AND ADSORPTION TECHNOLOGY

2.1 REMOVAL OF RADIOIODINE BY SOLID SORBENTS
(AEC Activity 04 60 80 01 1)

R. E. Adams R. D. Ackley Zell Combs

(Information presently available indicates that no funds are allotted for continuing the work on iodine and methyl iodide trapping by impregnated charcoals; therefore, this report summarizes the data available at the termination of the study.)

Fission-product iodine is of concern in reactor accidents because it is readily released and constitutes a rather unique biological hazard. As a consequence, iodine adsorbers are installed to mitigate the effect of an accident, and their expected performance under the conditions that would prevail needs to be critically evaluated.

Investigation of the effect of weathering (or exposure to continued air flow) on the performance of impregnated charcoals used for trapping not only elemental iodine but also the radioiodine occurring as methyl iodide (CH_3I) was continued. In this study, test beds of charcoal were exposed, in one case, to treated process air in a laboratory weathering setup and, in the other case, to building air of the Oak Ridge Research Reactor (ORR) in the ORR building. Procedures and conditions were as previously described,[1] except that the durations of exposure were shorter to permit measurements to be made for a greater variety of charcoal types. The durations employed were 2 and 4 months, and the effect of weathering was monitored by determining $CH_3{}^{131}I$-removal efficiencies (as before) if the charcoal was iodized or CH_3I-removal efficiencies if the charcoal was triethylenediamine impregnated.

The more recent data thus obtained were for two types of iodized charcoal, MSA 85851 and BC-727, and for a triethylenediamine-impregnated charcoal designated 5% TEDA (UK). The MSA 85851 charcoal was used in the earlier study of weathering and was again included to provide a basis for relating the earlier and the more recent test data. The results are summarized in Table 2.1.

The values in the table represent overall rates of decrease in $CH_3{}^{131}I$- or CH_3I-removal efficiency corresponding with that in a standardized removal test in percent per month. The tests for determining

Table 2.1. Rate of Decrease in CH_3I- or $CH_3I^{131}I$-Removal Efficiency During Weathering

Charcoal Type	Rate of Decrease of Removal Efficiency (% per month)			
	Laboratory-Exposed[a]		ORR-Exposed[b]	
	1-in. Depth[c]	2-in. Depth	1-in. Depth	2-in. Depth
MSA 85851				
Older data	0.72	0.11	2.7	0.50
Newer data	2.1	0.45	3.4	0.54
BC-727	2.5	0.32	2.6	0.23
5% TEDA (UK)	0.51	0.001	1.5	0.02

[a]Weathered at 78°F and 50% relative humidity in air flowing at approximately 40 fpm.

[b]Weathered at 78°F and 50% relative humidity in air flowing at approximately 20 fpm.

[c]Depth of charcoal bed.

removal efficiency were conducted at 25°C and 65% relative humidity with air flowing at a face velocity of 40 fpm; 0.5 mg of CH_3I was injected per cm^3 of charcoal. The extent of disagreement between the older and newer data for laboratory-exposed MSA 85851 is somewhat disturbing, and possible reasons for this change in behavior should be investigated. The relatively low rates of decrease observed for the triethylenediamine-impregnated charcoal, 5% TEDA (UK), indicate that it withstands the effects of this type of weathering very well.

Reference

1. R. E. Adams, R. D. Ackley, and Zell Combs, Removal of Radioiodine by Solid Sorbents, pp. 39–42 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for March–April 1968, USAEC Report ORNL-TM-2230, Oak Ridge National Laboratory.

2.2 REMOVAL OF SOLID AEROSOLS

(AEC Activity 04 60 80 01 1)

R. E. Adams J. S. Gill
R. J. Davis J. Truitt

The atmosphere in the containment vessel of a water-cooled reactor after a loss-of-coolant accident will be thoroughly saturated with condensing steam. Such an atmosphere will interact with the released aerosol particles and the filter system. The objective of this study is to determine the effect of those interactions. An effort is being made to provide a method (or model) with which to estimate the aerosol particle concentration in the containment vessel as a function of time after an accident. Toward this end a system of equations has been organized to make such estimates. The input required is particle size distribution as a function of environmental parameters such as initial aerosol concentration, turbulence of the gases and vapors, and humidity. These data are being collected in the laboratory. A brief outline of the method of calculation and some preliminary data concerning particle size distribution are given below.

Much of the complexity of aerosol behavior arises from the fact that coagulation and settling occur together and both depend on and affect the particle size distribution. The interrelationships are very difficult to keep track of in detail, and efforts to do so have not proved useful. The technique used in the following (which effectively separates the calculation of particle removal by the two processes) is first to presume that, at high concentration, collisions between particles are very frequent, that many particles undergo many coagulation events (before they have time to diffuse to a wall or settle by stirred settling), and that some of the particles become quite large and settle quickly (by so-called tranquil settling). The concepts of stirred and tranquil settling have been discussed elsewhere.[1] These assumptions, along with a particle size distribution function, permit a calculation of rate of particle removal. At lower concentrations, coagulation is much slower and settling is presumed to occur according to stirred settling. A technique for making the

calculation of rate of particle removal by this process has also been reported.[2]

In the calculation for high concentration aerosols, the rate of decrease in the number concentration of particles, due to coagulation, n (as discussed in Ref. 3) is approximately

$$\frac{dn}{dt} = -3 \times 10^{-10}\, n^2 \ . \tag{1}$$

The value of the constant (3×10^{-10}) was estimated from diffusion theory. Integration yields

$$\frac{1}{n} = \frac{1}{n_0} + 3 \times 10^{-10}\, t \ , \tag{2}$$

where n is the number concentration of particles at time t, and n_0 the concentration at zero time. If settling proceeds to maintain the so-called "self-preserving" particle size distribution function,[4]

$$\frac{dn}{dr} = 0.05\phi r^{-4} \ , \tag{3}$$

where r is the particle radius and ϕ is the volume fraction of disperse phase (i.e., volume of particle phase per volume of aerosol). Then

$$n = 0.0167\phi \left(\frac{1}{r_{min}^3} - \frac{1}{r_{max}^3} \right) \tag{4}$$

and

$$n_0 = 0.0167\phi_0 \left(\frac{1}{r_{min}^3} - \frac{1}{r_{max}^3} \right) , \tag{5}$$

where ϕ_0 is the aerosol concentration (as volume fraction) at zero time. Commonly r_{max} is much greater than r_{min} (i.e., the maximum particle size is much greater than the minimum particle size); hence, for

$$1/r_{min}^3 \gg 1/r_{max}^3 \ ,$$

$$n = 0.0167 \frac{\phi}{r_{min}^3}, \tag{6}$$

and

$$n_0 = 0.0167 \frac{\phi_0}{r_{min}^3}. \tag{7}$$

Substituting into Eq. (2) and rearranging gives

$$\frac{\phi}{\phi_0} = \frac{1}{1 + 5\phi_0 \frac{t}{r_{min}^3}}, \tag{8}$$

where r (the particle radius) is in microns and t is in seconds.

The foregoing derivation provides a simple estimate of ϕ/ϕ_0 (i.e., the fraction of original particle material that is still airborne , which results from rapid coagulation and tranquil settling of large particles. Calculated values of ϕ versus time (for particular conditions are shown in Fig. 2.1 and labeled "Rapid Initial Coagulation and Settling." It is clear that this process is the dominant removal mechanism at high particle concentration.

A calculation reported previously[2] provides a simple calculation of ϕ/ϕ_0 versus time due to stirred settling. Calculated values (of ϕ vs time) from this mechanism alone are shown in Fig. 2.1 and labeled "Coagulation and Settling Alone." This process becomes important at longer times and lower particle concentrations.

Another pair of removal mechanisms, diffusiophoresis and thermophoresis (i.e., the sweeping action of condensing steam onto spray drops) was described in the previous report in this series.[5] Calculated values of ϕ resulting from these mechanisms and for particular conditions are plotted in Fig. 2.1 and labeled "Diffusiophoresis and Thermophoresis."

There is also a curve on Fig. 2.1 that represents the summation of all the removal phenomena. It is labeled "Sum of All Effects." It was

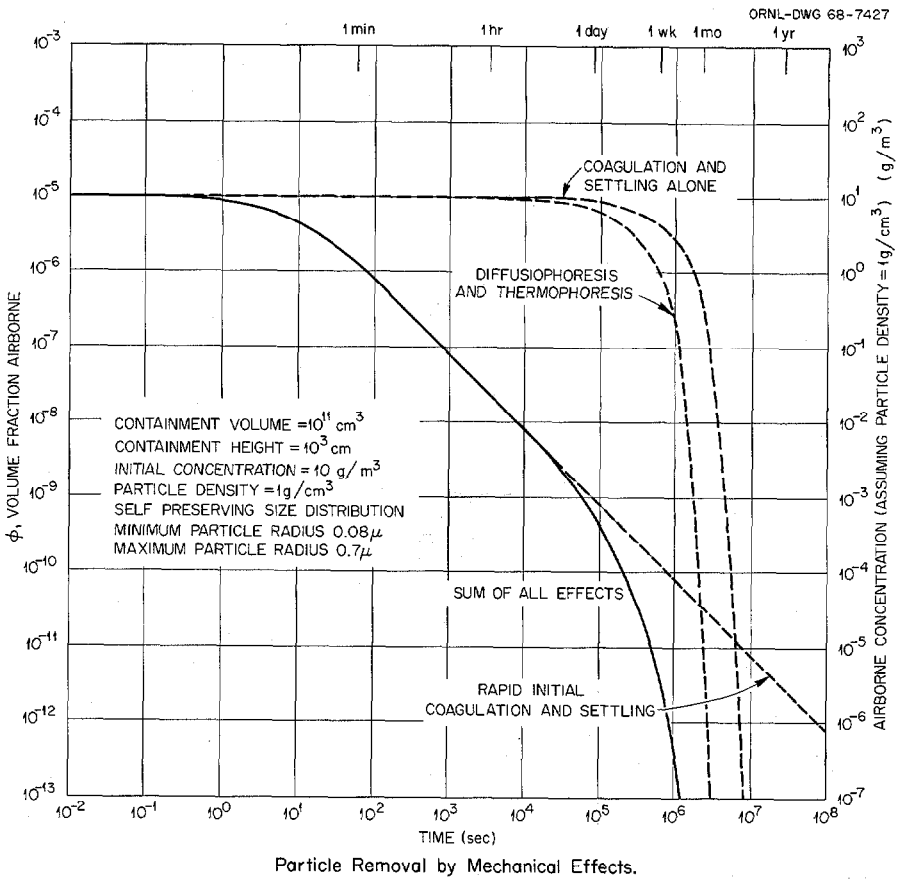

Fig. 2.1. Particle Removal from an Aerosol by Mechanical Effects.

calculated on the basis that each effect functions independently of the other; i.e.,

$$[\phi/\phi_0]_{sum} = [\phi/\phi_0]_C[\phi/\phi_0]_S[\phi/\phi_0]_D \ , \qquad (9)$$

where the subscripts C, S, and D refer to each independent mechanism: rapid initial coagulation, stirred settling, and diffusiophoresis plus thermophoresis.

Intrinsic to most of these calculations is the presumption of a particular particle size distribution function, as in Eq. (3). The experimental program is to provide actual size distribution functions. Some preliminary results have been obtained with very young aerosols of stainless steel oxide produced by an electric spark. In the experiment clean air is swept over two electrodes, one of which is stainless steel. The volume of the spark chamber is about 2.5 liters. Part of the aerosol is diluted with clean air and fed to an electrostatic particle counter (manufactured by Thermal Systems, Inc., St. Paul, Minn.) which gives an indication of the number of particles in each of a succession of particle size ranges. In separate tests, a measured volume of aerosol is pushed through a filter. The increase in weight of the filter provides an indication of the mass concentration (i.e., micrograms of particle phase per liter of aerosol).

Aerosol concentrations corresponding to several flow rates (i.e., flows past the spark) are listed in Table 2.2. Also listed are the residence times in the spark chamber. The products of flow rate and mass concentration are given; these values are in fact the total amounts of aerosol material removed from the electrodes (per second) minus whatever settles or is otherwise removed. The fact that these values are roughly constant indicates that relatively little particle material is lost (by settling, etc.) during these short times.

Figure 2.2 shows some of the data from the particle counts; this is a plot of distribution function dN/dD_p (change in number concentration with particle diameter) versus particle diameter. These results were calculated from particle count and particle size data as obtained from the counting instrument (i.e., a complete calibration of the instrument with the stainless steel oxide aerosol has not yet been done).

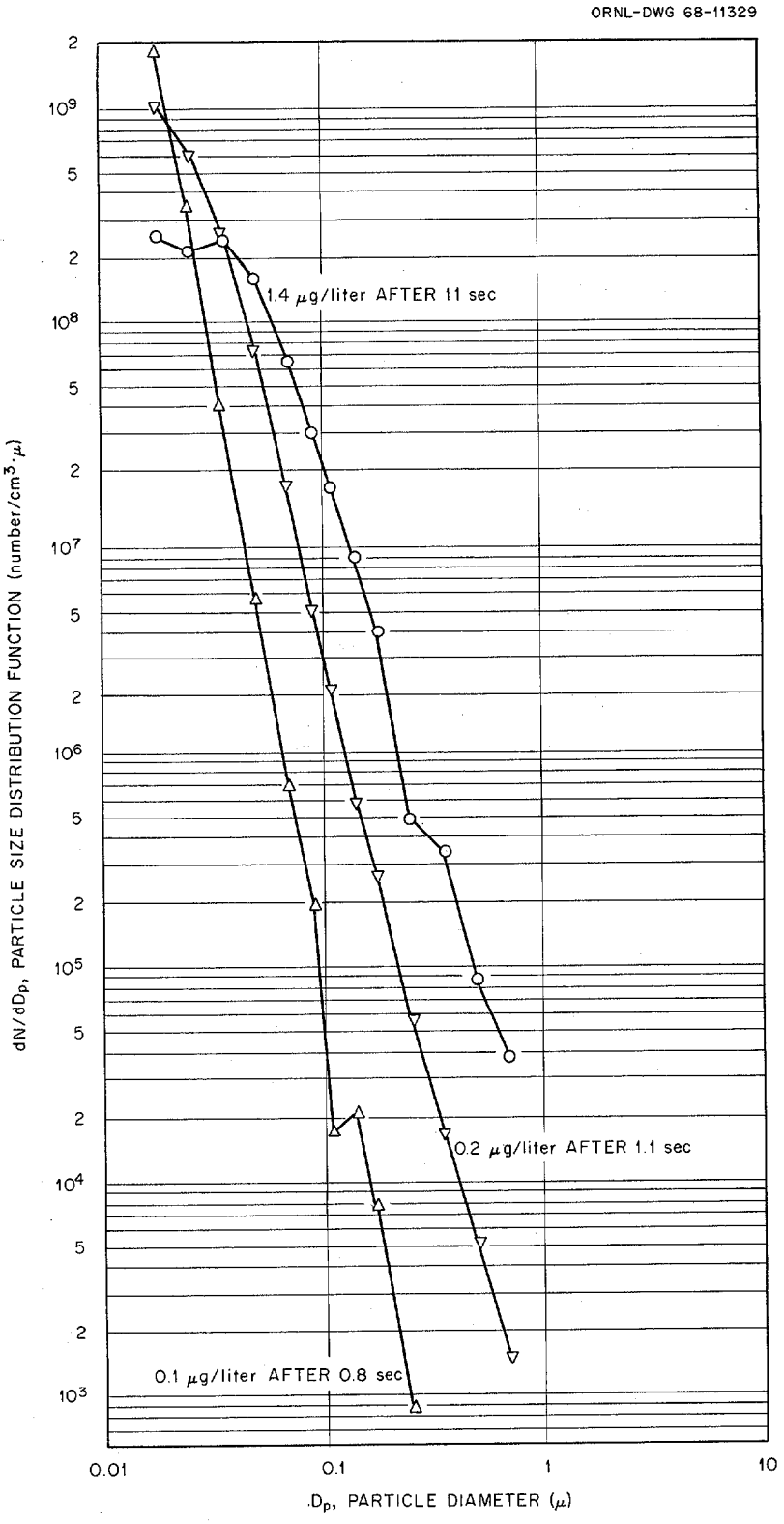

Fig. 2.2. Particle Size Distribution of Electric-Spark-Generated Stainless Steel Oxide Aerosol

Table 2.2. Preliminary Results of Experiments with Aerosols of Stainless Steel Oxide Produced by an Electric Spark

Flow Rate Past the Electrodes (liters/sec)	Aerosol Mass Concentration (μg/liter)	Residence Time Before Dilution (sec)	Flow Rate Times Mass Concentration (μg/sec)
0.24	1.4	11	0.32
0.38	0.93	6.7	0.35
0.72	0.51	3.4	0.37
2.4	0.18	1.1	0.43
3.3	0.09	0.76	0.30

The curves are labeled according to the mass concentration and residence time. The "1.4 μg/liter after 11 sec" and "0.2 μg/liter after 1.1 sec" curves approximately fit the r^{-4} size distribution, as given in Eq. (3). The very youngest aerosol, labeled "0.1 μg/liter after 0.8 sec," does not fit the r^{-4} relation.

It is premature to make any generalizations. The data are presented as examples of the kind of information which we expect to supply. We also expect to provide more data, as in Fig. 2.2, that include maximum and minimum particle sizes to describe aerosols of interest over ranges of environmental parameters.

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2.3 CHARACTERIZATION OF FISSION PRODUCTS UNDER LOFT CONDITIONS

(AEC Activity 04 60 50 05 1)

R. E. Adams R. L. Bennett
W. H. Hinds Ruth Slusher

This activity was not funded for FY 1969. Therefore the study was terminated, and a final report was issued.[1]

Reference

1. R. L. Bennett, W. H. Hinds, and R. E. Adams, Development of Iodine Characterization Sampler for Application in Humid Environments, USAEC Report ORNL-TM-2071, Oak Ridge National Laboratory (in press).

2.4 IGNITION OF CHARCOAL ADSORBERS BY FISSION-PRODUCT DECAY HEAT

(AEC Activity 02 30 02 90)

R. E. Adams J. P. Sanders
R. P. Shields N. Clampitt, Jr.*

Charcoal adsorbers for radioiodine constitute an important part of the safety systems for the removal of accident-released fission products from the reactor containment volume either during recirculation of the atmosphere or as it is being exhausted from the containment vessel. During operation, the charcoal adsorbers would be subjected to loading with large quantities of fission products and, consequently, would be heated by the radiodecay of such materials. A program is under way, through laboratory and in-pile experiments and by heat transfer analysis, to establish the effects of fission products and irradiation on the ignition behavior of charcoal adsorbers.

The effect of water vapor in the sweep gas on the ignition temperature of several charcoals was studied; this atmosphere is representative of the accident atmosphere which would prevail following a loss-of-coolant accident to one of the present-generation power reactors. Three charcoals were used: BC-416 (also known as BC-513), a nonimpregnated (noniodized) coconut charcoal; BC-592, a nonimpregnated coconut charcoal specially treated to have a high ignition temperature; and MSA 85851, an impregnated (iodized) coconut charcoal. The results of measuring the ignition temperature of these three charcoals in sweep air (velocity, 40 fpm) containing up to 70% water vapor are given in Fig. 2.3. The ignition temperature of BC-416 is virtually unaffected by water-vapor concentrations up to about 30%. At the higher water-vapor concentrations the increase in ignition temperature is probably a consequence of reduction in available oxygen (air containing 50% water vapor has its gaseous oxygen content reduced about one-half). This same effect is noted for BC-592 and MSA 85851 after the initial reduction in ignition temperature noted between water vapor concentrations of 0 and 10%. In the case of MSA 85851 this lowering of the ignition temperature may be the result of some loss

*Temporary employee, ORAU Summer Program.

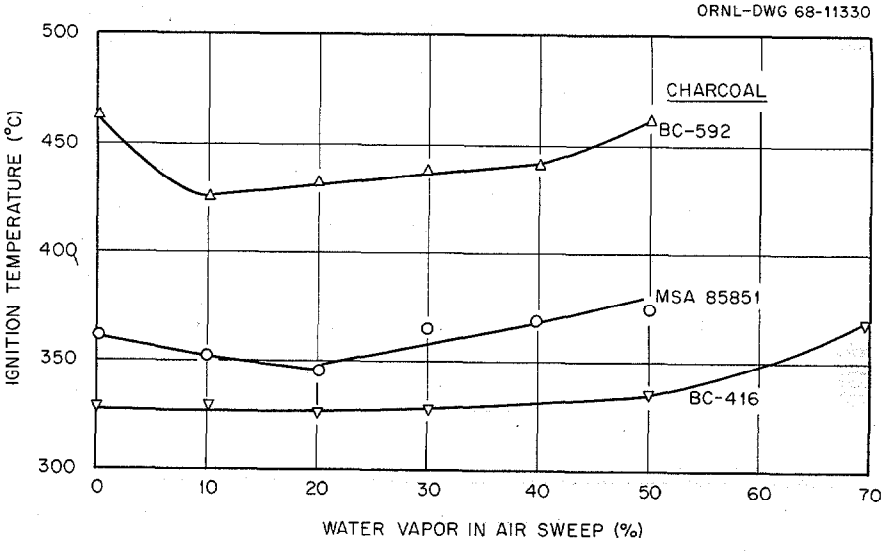

Fig. 2.3. Ignition Temperature of Charcoal as a Function of Water-Vapor Content of Air Sweep.

of iodine impregnant. During one of the tests with BC-592 charcoal, the exhaust gases were passed through a steam condenser, and condensate was collected while the test charcoal was heated over the range 385 to 435°C. This condensate was found to contain a significant quantity of phosphorus. Since phosphates are reported to be effective in increasing the ignition temperature of charcoal it seems reasonable to conclude that the observed lowering of the ignition temperature of BC-592 is due to removal of some of this agent by the water vapor.

It has been postulated that one of the more hazardous situations that might occur during operation of charcoal adsorbers is one wherein the air-moving equipment would fail after large quantities of radioiodines had been adsorbed. As a result the charcoal mass would heat up in an oxygen-deficient humid atmosphere. Reestablishment of air flow (and oxygen supply) might cause ignition rather than a reduction of the threat of ignition by cooling the charcoal. Laboratory tests simulating this situation were performed by heating the charcoals noted above, in the standard ignition tube, to about 100 to 120°C and starting a very slow sweep of water vapor through the charcoal. The temperature was then increased to various levels depending upon the normal ignition temperature of the charcoal and held at this temperature for 1 hr. When an air sweep was established after the holding period a rapid increase in the temperature was noted. In some cases the charcoals, which were being maintained at temperatures as much as 60°C below their ignition temperature, ignited. The increase in temperature was the result of oxygen being readsorbed on the active sites on the charcoal surface. These sites had been freed of oxygen during the heatup in the humid oxygen-deficient atmosphere. Similar cleaning of oxygen complexes from charcoal surfaces can be demonstrated by sweeping a charcoal bed, at elevated temperatures, with any inert gas, such as helium or nitrogen.

Equipment has been assembled with which to measure the thermal conductivity of the newer charcoals at elevated temperature. Preliminary measurements for the purposes of equipment shakedown are being made on BC-416. Additional measurements will be made on this charcoal and on the impregnated charcoals at temperatures up to around 200°C.

Ignition temperatures were determined for two charcoals from Witco Chemical Company. These iodized charcoals were prepared from petroleum residues and had been found to be highly promising for methyl iodide trapping.[1] Both charcoals had high ignition temperatures. Type OF-42 ignited at 442°C and Type OF-31 ignited at 466°C. No measurements were made to determine impregnant loss prior to ignition.

The program to develop a computer model to represent the heat generation and temperature distribution within a full-scale charcoal adsorber after having been exposed to accident-released fission products was continued. Several computer codes available at ORNL were reviewed for applicability to the calculation of the temperature distribution in the charcoal bed. All these were directed toward a specific geometry which differed from that of a charcoal adsorber and were therefore not applicable. The general heat transfer codes at ORNL could not be used to represent accurately the heat source as a function of position and time as represented by the beta and gamma emission of the two chemical forms of iodine (I_2 and CH_3I) and the distributed heat source represented by the transfer to the gas flowing through the adsorber. The exact differential equations representing the exact situation were written, and work is in progress to obtain the analytical solution.

Reference

1. R. E. Adams, R. D. Ackley, and Zell Combs, Removal of Radioiodine by Solid Sorbents, pp. 27–29 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1968, USAEC Report ORNL-TM-2283, Oak Ridge National Laboratory.

2.5 SEPARATION OF NOBLE GASES FROM AIR WITH A PERMSELECTIVE MEMBRANE

(AEC Activity 04 60 80 01 1)

R. H. Rainey W. L. Carter

A process is being developed for removing xenon and krypton from other gases by use of a permselective membrane composed of thin sheets of methylphenyl silicone rubber. This separation is based on the difference in the solubility of the gases in the membrane and their different rates of transport through the membrane. Applications for the process include the removal of noble-gas fission products from (1) the air within a reactor containment building after an accidental release of fission products; (2) the off-gas from a plant for processing spent reactor fuels; and (3) the gas that blankets nuclear reactors, especially reactors such as the molten-salt-fueled and sodium-cooled breeder reactors, which continuously vent volatile fission products.

The following identification of the membrane used in this study was received from the Silicone Products Department of the General Electric Company:

100 parts (by weight) methylphenyl polymer containing 5.3 mole % phenyl groups and 0.1% vinyl groups (specific gravity, 0.98)

44 parts (by weight) fumed silica

0.164 volume fraction of filler (calculated)

The membrane was partially cured with 0.5% bis-2,4-dichlorobenzoyl peroxide and then heated at 150°C for 1 hr in an oven. We had previously understood that the membrane was dimethyl silicone rubber.

Laboratory Measurements of Permeability and Separation Factors

Our laboratory equipment for measuring the permeability and separation factor of gases was modified to provide for the continuous counting of both the extract and raffinate streams and to increase the volume of these gases that may be used in a single experiment. As a result of

these changes we are now able to run the equipment until steady-state conditions are measured. The material balance for each experiment was ±3%.

Data obtained with this improved equipment showed several differences from that previously reported. There was no significant difference due to cocurrent or countercurrent flow in separation of krypton from nitrogen (Fig. 2.4) or krypton from oxygen (Fig. 2.5). The mixtures contained less than 0.5% krypton. These data were extrapolated to the separation factor at zero cut, which was calculated from Blumkin's equation.[1] The calculated permeability of the krypton decreased with cut. This decrease depended upon the carrier gas (Figs. 2.6 and 2.7) and thus explained the dependence of the permeability of dilute gas on the permeability of the carrier gas previously reported.[2] The "true" permeability of the dilute gas may be obtained by extrapolating either set of data to zero cut. This value does not vary with carrier gas, and the ratio of the permeability of krypton to permeability of nitrogen or oxygen is in agreement with Robb's[3] data for pure gases.

Development of Improved Membrane

The General Electric Company Research and Development Center developed the membrane package used in this program under a subcontract from the USAEC. As reported previously, the permeability of the membrane package (i.e., membrane that is supported between two sheets of Dacron mat) was about half that of an unbonded membrane. Investigations by H. P. Briggs at General Electric have shown that about 40% of this loss in permeability may be recovered by bonding the membrane to a properly selected mat on only the low-pressure side and by proper selection of support and turbulence promoting screens.

Briggs and his co-workers tested numerous backing materials and supporting screens in order to select a combination of materials that would give the lowest flow resistance in the directions normal and parallel to the membrane. The tests consisted in measuring the permeation of oxygen under pressure drops ranging from 15 to 150 psi for each support combination. With similar membrane samples, some backing materials exhibited low permeation rates, while other samples of the same material with

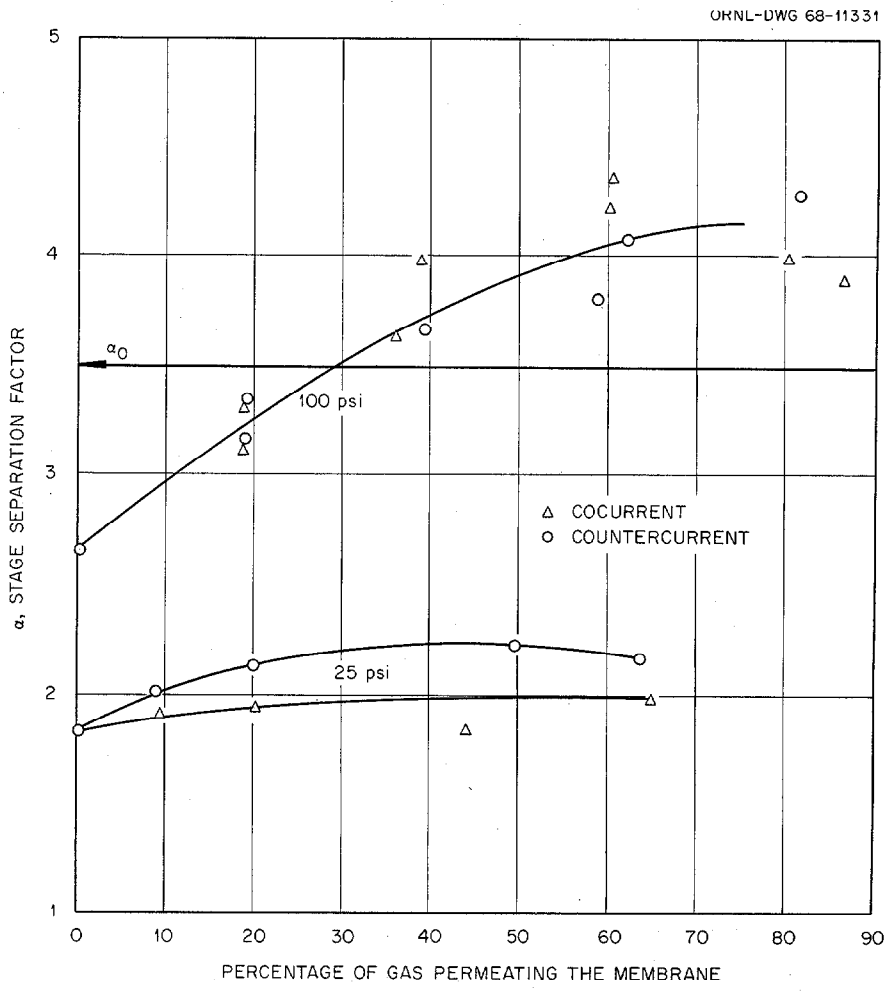

Fig. 2.4. Increase in Separation Factor of Krypton from Nitrogen with Increasing Percentage of Gas Flowing Through the Membrane.

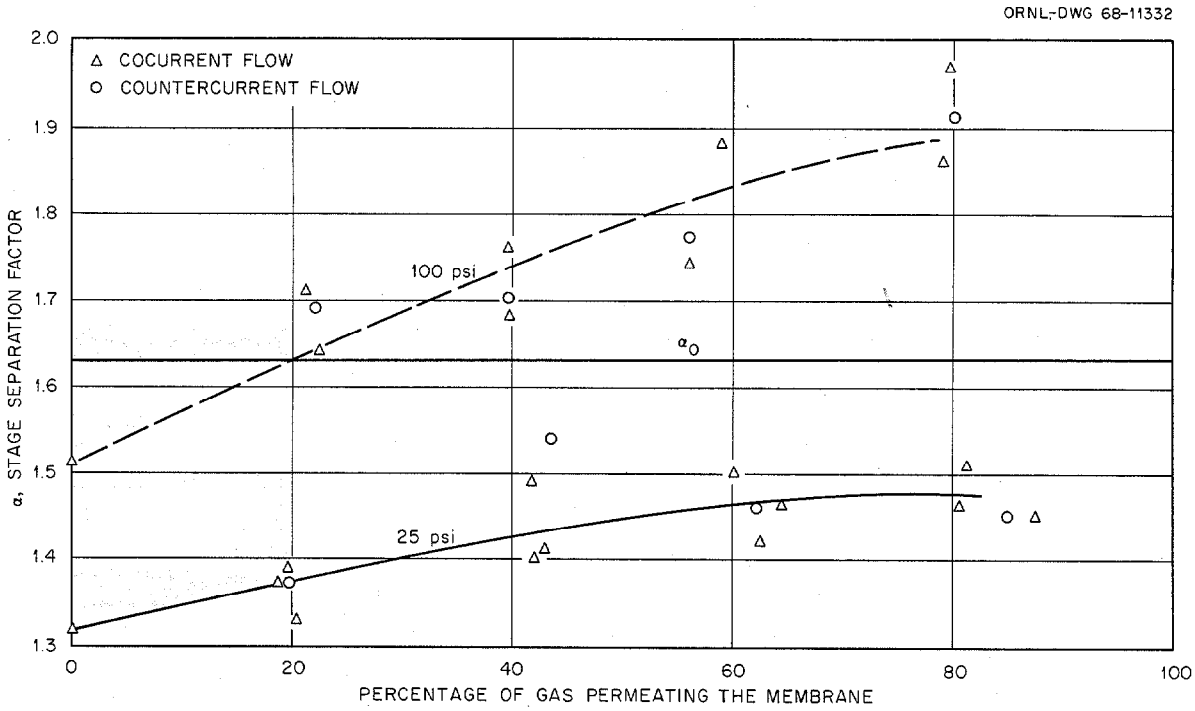

Fig. 2.5. Separation Factors of Krypton from Oxygen as a Function of Cut.

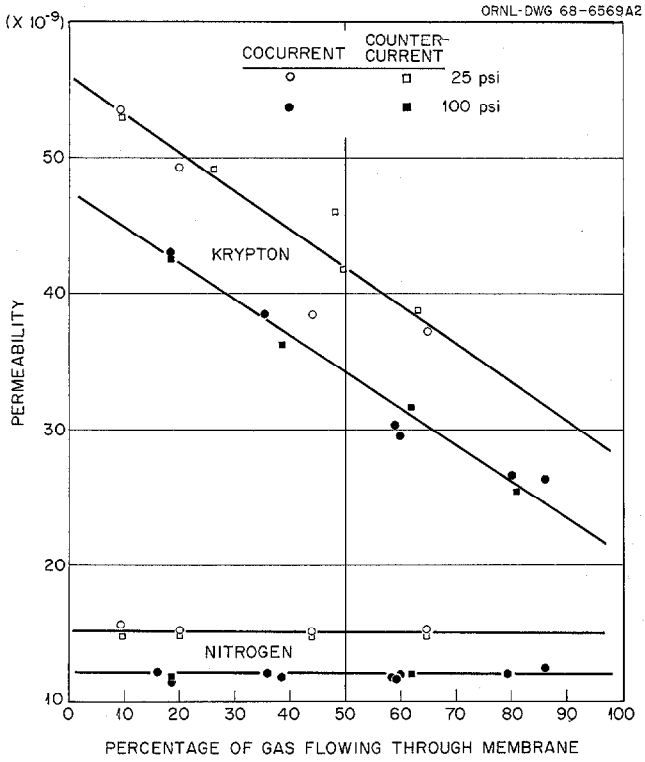

Fig. 2.6. Permeabilities of Krypton and Nitrogen as Functions of Cut and Pressure Drop Across the Membrane.

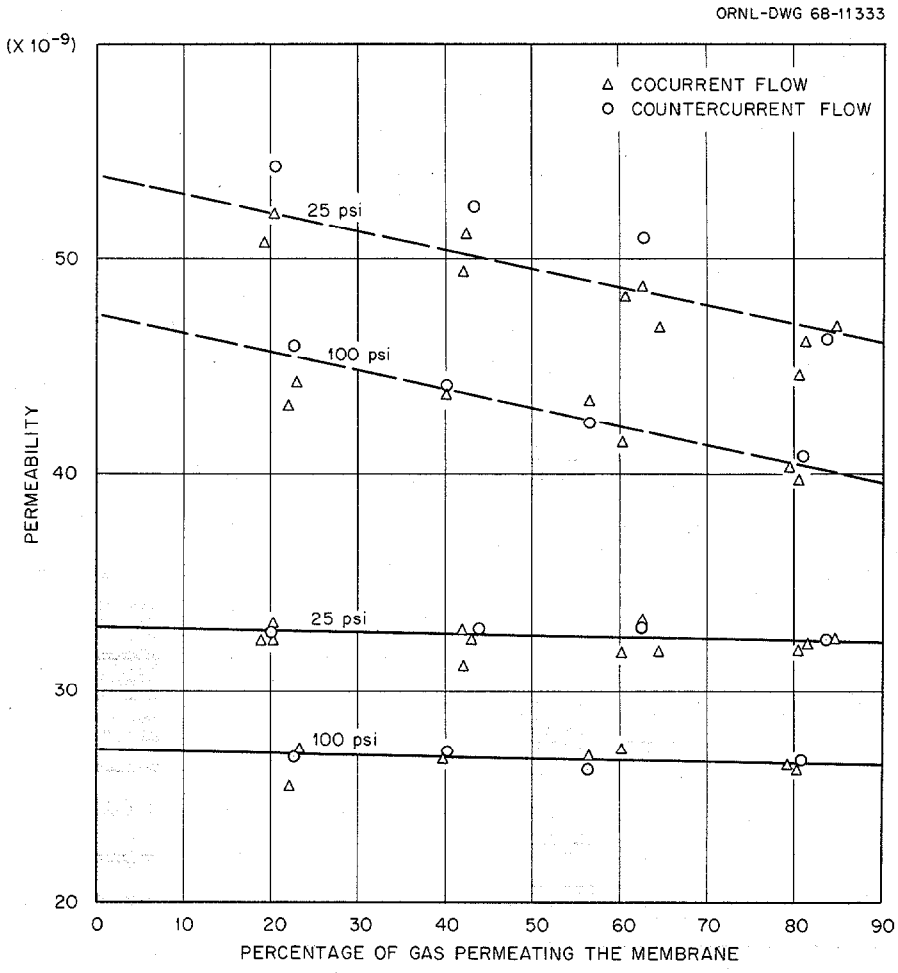

Fig. 2.7. Permeability of Krypton and Oxygen as a Function of Cut (<0.5% Krypton in Oxygen).

different thicknesses and weights exhibited higher permeation rates. Also, some of the samples were found to have satisfactory porosity normal to their surfaces under low pressure but were relatively nonporous (i.e., showed a decrease in the permeation rate) as the pressure drop across the membrane increased to 150 psi.

In many of the tests the membrane showed evidence of being depressed, or deformed, by the openings in the supporting screen. The permeabilities obtained by using these supporting materials were generally lower than those obtained with the other materials, and they decreased as the pressure increased. The deformation of the support mat into the openings in the screen reduced the cross-sectional volume of the screen available for gas flow; it also created many points of high stress. Therefore, the fibers of the support mat must have sufficient strength to span the openings in the screen, as well as exhibit high resistance to compression. However, the fibers must not puncture the membrane.

The most favorable permeability data were obtained with a mat 0.018 cm thick weighing 30 g/m^2 that was composed of polyester fibers and a polyvinyl binder. The oxygen permeability factor for this material was 56.5, as compared with 60 for an unbonded membrane. No decrease in permeability occurred with pressure drops up to 150 psi across the membrane.

Briggs and his co-workers conducted a similar study to choose the best screen support. They investigated screens made of polyester nonfilament or stainless steel wire having various diameters and sizes of mesh openings. They found that two screens on the low-pressure side of the membrane provided the most satisfactory arrangement. A polyester screen (0.028-cm threads, 60 threads per centimeter, 0.081-cm mesh opening, 55% open area) was positioned adjacent to the support mat, and a stainless steel screen (0.019-cm wire, 61 wires per centimeter, 0.087-cm mesh opening, 67.2% open area) was placed behind the plastic screen. These two screens have similar fiber spacing, but the stainless steel wire is smaller than the plastic filament, and thus there is a larger open area. The stainless steel screen could not be used immediately adjacent to the mat because it contained sharp wire points that would puncture the membrane. Also, it had a tendency to buckle under high stress conditions to form sharp ridges, which would also puncture the membrane.

A polyester nonfilament screen (0.114-cm threads, 44.5 threads per centimeter, 0.114-cm mesh opening, 58% open area) was included on the high-pressure side of the membrane to induce turbulent flow. The large, smooth fibers of this screen exhibited a negligible tendency to puncture the unprotected rubber membrane. A laboratory unit containing this improved membrane and support combination was installed in our test facility, and preliminary results confirm improved operating characteristics.

Briggs reports that the construction of the 10-yd^2 engineering scale unit is nearly complete, and he expects to deliver it to us by October. However, due to the cut in our funds, we will not be able to conduct the engineering test program planned for this year with this unit.

References

1. S. Blumkin, A Method for Calculating Cascade Gradients for Multicomponent Systems Involving Large Separation Factors, USAEC Report K-OA-1559, Oak Ridge Gaseous Diffusion Plant, Jan. 15, 1968.

2. R. H. Rainey, Separation of Noble Gas from Air Using a Permselective Membrane, pp. 43–48 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for July–August 1967, USAEC Report ORNL-TM-1986, Oak Ridge National Laboratory, Sept. 21, 1967.

3. W. L. Robb, Thin Silicone Membranes – Their Permselective Properties and Some Applications, Report 65-C-031, General Electric Company, October 1965.

2.6 HIGH-EFFICIENCY AIR-FILTRATION ENGINEERING MANUAL (AEC Activity 04 60 80 01 1)

C. A. Burchsted A. B. Fuller

A manual entitled "Design and Construction of High-Efficiency Air Filtration Systems for Nuclear Applications" was prepared as a guide for processing plants, laboratories, and other installations that produce radioactive exhaust. The manual gives background information concerning the special hazards and problems of radioactively contaminated exhaust, as well as specific design, construction, and testing recommendations for exhaust systems. Problem areas and factors that often compromise operations are identified, and mechanical, structural, and layout requirements are discussed in detail. Reliability and total cost, rather than first cost, are stressed. There is a separate chapter on the special problems of remotely maintained systems and reactor postaccident cleanup systems.

A draft of the manual, Report ORNL-TM-2300, will be sent to a limited distribution for comment review in September and will also be reviewed in detail with the AEC Divisions of Operational Safety and Reactor Development and Technology. Following these reviews the manual will be modified as necessary and issued as an ORNL-NSIC report. Comments received from the field in response to the NSIC issue will be considered in a possible reissue as an AEC-TID handbook. The release date as an NSIC report is still uncertain.

3. SPRAY AND POOL PRESSURE-SUPPRESSION TECHNOLOGY

T. H. Row, Coordinator

3.1 EFFECT OF ADDITIVES ON DISTRIBUTION OF I_2 AND CH_3I BETWEEN AIR AND WATER

(AEC Activity 04 60 80 01 1)

C. S. Patterson* W. T. Humphries*

Work was extended in the following areas: (1) iodine capacities for solutions of six additional compounds, (2) changes in pH of some scrubber solutions on saturation with iodine, and (3) methyl iodide capacity coefficients for some possible additives.

Iodine-Capacity Coefficients of Solutions

Iodine-capacity coefficients, K_d', obtained for solutions of six compounds not previously reported are given in Table 3.1. A number of reducers were found to be good scrubbers for iodine. Two oxidizers ($KMnO_4$ and NaOCl) are included in Table 3.1 that show some affinity for iodine, but neither is good enough to be of any practical value. The high value for NaOCl is due to the more concentrated solution.

Table 3.1. Iodine-Capacity Coefficients, K_d', for Iodine at 25°C

Compound	Concentration	K_d'
Maleic acid	2.51×10^{-3} M	98
Ammonium oxalate	2.50×10^{-3} M	100
Sodium hypochlorite	0.1%	633
Potassium permanganate	2.50×10^{-3} M	125
Hypophosphorous acid	0.1%	122
Calcium chloride	0.1 M	108

Calcium chloride was tested because a question was raised about the possibility of leaching this compound from concrete. Other components of concrete (i.e., $CaSO_4$ and $CaCO_3$) were not studied because of their negligible solubility in water.

*Work performed under subcontract with Furman University.

Changes in pH of Solutions Upon Saturation with Iodine

When I_2 reacts with water the pH is always shifted toward the acid side. This shift can produce adverse effects, so the pH's of the various scrubber solutions as I_2 is added are of interest.

This study was limited to compounds that had previously shown a reasonable affinity for iodine. The pH was determined initially for solutions of the concentration usually tested (2.5×10^{-3} M) and then after saturation with iodine. The results are given in Table 3.2, with the solutions tabulated in the order of increasing acidity of the final solution. Note that the final pH's are applicable only when the spray solution

Table 3.2. Changes in pH of Solutions Upon Saturation with Iodine

Compound	Initial pH	Final pH	Δ(pH) (initial − final)
4-Aminopyridine	9.80	8.74	1.06
Morpholine	9.45	8.39	1.06
Sodium hydroxide	10.75	7.70	3.05
Hexamethylenetetramine	7.42	7.37	0.05
Imidazole	9.12	7.17	1.95
Pyridine	7.20	6.80	0.40
Piperazine	10.20	6.80	3.40
Quinoline	7.55	6.60	0.95
Piperidine	10.75	6.40	4.35
1,4 Diazabicyclo (2,2,2) octane	9.60	6.05	3.55
Triethanolamine	9.25	4.70	4.55
Potassium thiocyanate	6.80	4.05	2.75
Phenol	6.70	3.25	3.45
Sodium thiosulfate + boric acid	5.45	2.55	2.90
Pyrrole	7.02	2.45	4.57
Semicarbazide HCl	3.32	2.35	0.97
Resorcinol	6.18	2.39	3.79
1,3,5 Trihydroxybenzene	6.25	2.25	4.00
Thiourea	7.10	2.12	4.98
Sodium thiosulfate	7.05	2.10	4.95
Aminoguanidine bicarbonate	8.41	1.98	6.43
Carbohydrazide	6.95	1.95	5.00
Hydrazine	9.63	1.87	7.76
Thiosemicarbazide	4.70	1.78	2.92

is saturated with iodine. In the case of an accident the probability of saturation is remote, but this ranking of the solutions should be a useful indication of the relative buffer capacities of the solutions against increases in acidity from any source.

Methyl Iodide-Capacity Coefficients of Solutions

The "direct method" described previously[1] was used to compare the affinities of various aqueous solutions for methyl iodide. The solutions studied were chosen because of their iodine capacities or because of current interest in them as spray additives.

Similar gas-phase samples of CH_3I were prepared in a reaction flask as a starting point for each run. The effect of the scrubber solution was measured as percentage of initial gas-phase CH_3I remaining as a function of time of exposure to the solution under controlled conditions.

Figure 3.1 compares the results obtained for NaOH, $Na_2S_2O_3$, and a mixture of the two. Thiosulfate is far superior to the base (NaOH) in removal of CH_3I, but the single mixture of NaOH and thiosulfate suggests somewhat the same synergistic effect as that observed earlier for iodine. The mixture contains approximately 50% of each solution, but the curve lies much closer to that for thiosulfate. The infinite time limits for thiosulfate and the mixture both seem to be essentially zero (as indicated by reciprocal plots), but the base definitely stops short of quantitative removal. This suggests that the reaction with base may be reversible.

Figure 3.2 gives similar results for hydrazine and a base-hydrazine mixture. Neither is comparable to the same concentration of thiosulfate. On the other hand, $AgNO_3$ is quite comparable to thiosulfate.

Figure 3.3 shows results for three of the organic additives found to be most effective against elemental iodine. Piperidine is the best of these tested to date.

Reference

1. C. S. Patterson and W. T. Humphries, pp. 51–60 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1968, USAEC Report ORNL-TM-2283, Oak Ridge National Laboratory, July 30, 1968.

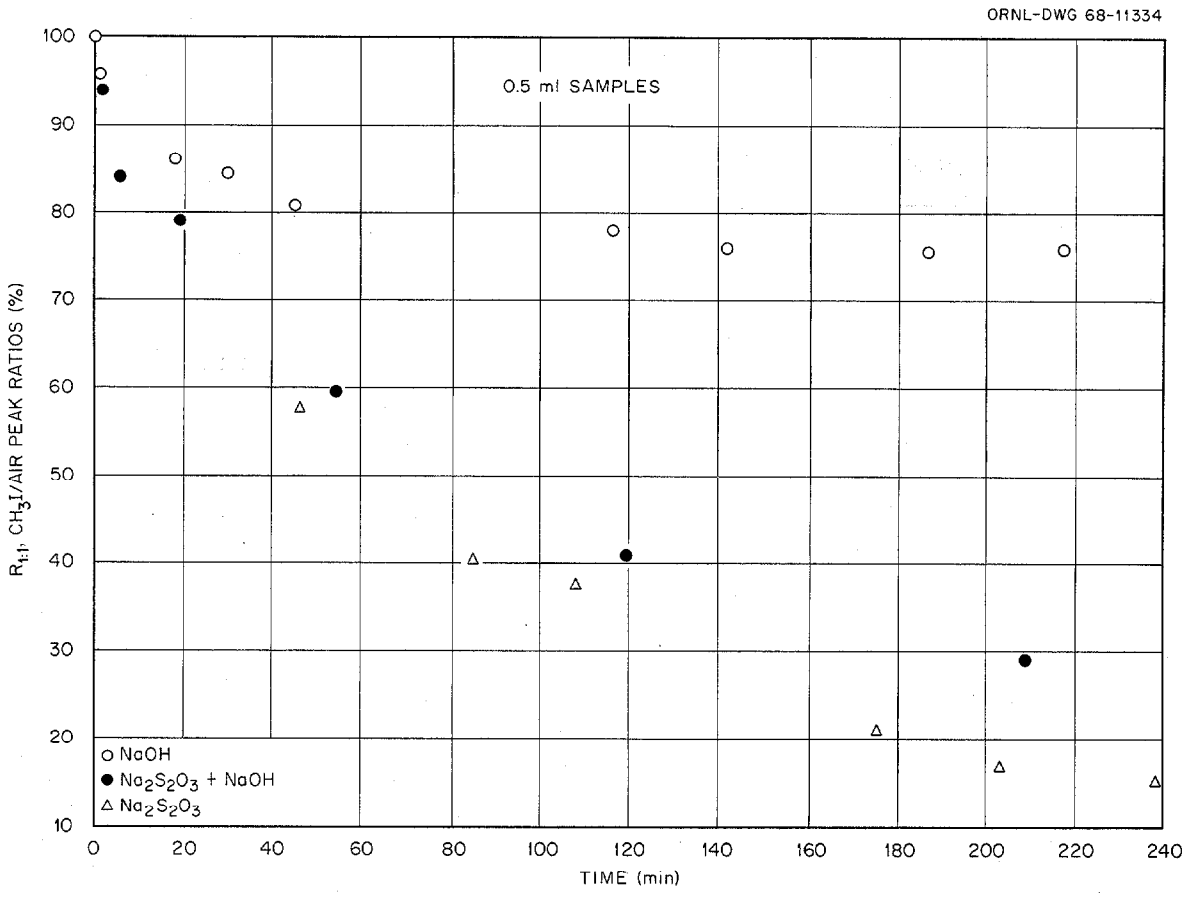

Fig. 3.1. CH_3I Peak Ratios in Air as a Function of Time of Exposure to NaOH, $Na_2S_2O_3$, and $Na_2S_2O_3$ + NaOH.

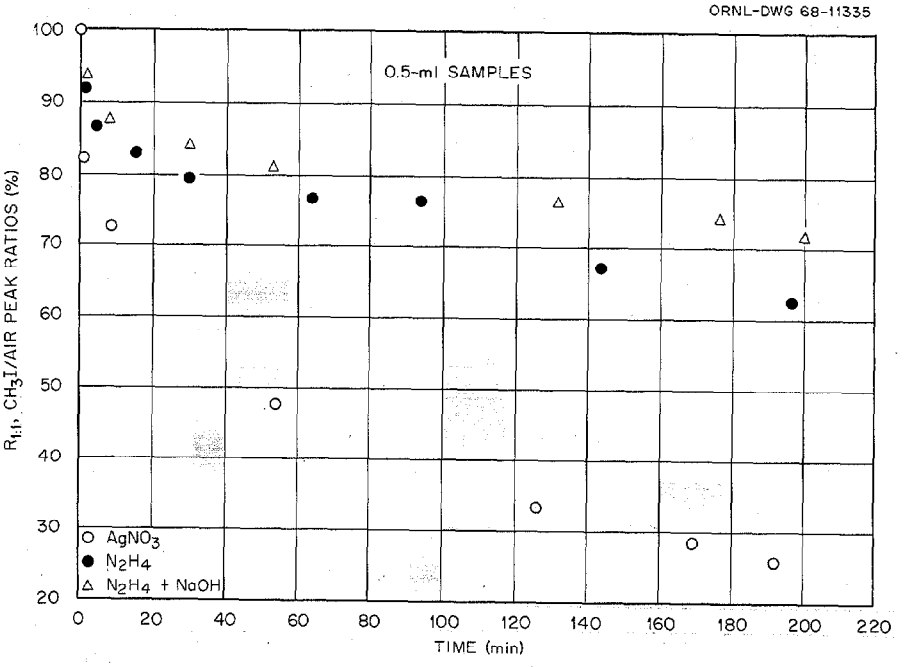

Fig. 3.2. CH_3I Peak Ratios in Air as a Function of Time of Exposure to $AgNO_3$, N_2H_4, and N_2H_4 + NaOH.

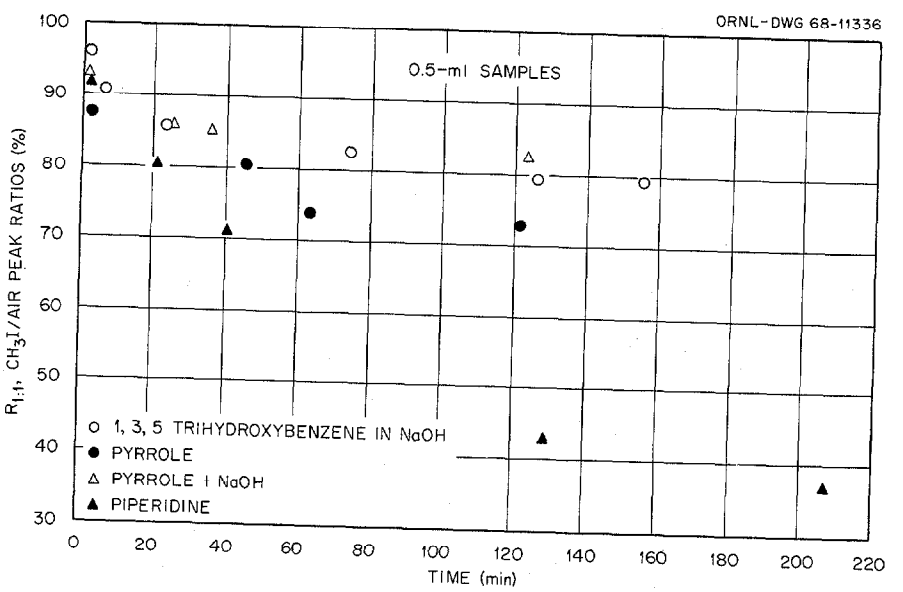

Fig. 3.3. CH_3I Peak Ratios in Air as a Function of Time of Exposure to 1,3,5 Trihydroxybenzene in NaOH, Pyrrole, Pyrrole Plus NaOH, and Piperidine.

3.2 UPTAKE OF I_2 AND CH_3I BY WATER SOLUTIONS AND DROPS

(AEC Activity 04 60 80 01 1)

B. A. Soldano W. T. Ward

Spray studies have shown that either of two spray additive solutions — (1) 0.3 wt % boron in the form of borate, with a pH of 9.1, and (2) 1 wt % $Na_2S_2O_3$ containing 0.3 wt % boron in the form of borate, with a pH of 9.1 — is highly effective in the removal of I_2. As far as the absorption of CH_3I activity is concerned, the solution containing the reducing agent $Na_2S_2O_3$ is superior, since the $Na_2S_2O_3$ alone has an affinity for CH_3I. Nevertheless, from a rate standpoint, the $Na_2S_2O_3$ is not sufficiently effective for CH_3I removal. A search therefore has been in progress for additives other than $Na_2S_2O_3$ to enlarge the number of systems that can effectively remove both CH_3I and I_2 and yet meet rigid standards on rates, radiation and thermal stability, corrosion, etc.

Single-drop wind-tunnel studies of the factors affecting the removal of CH_3I from gas streams suggest that the activated state for CH_3I removal involves initially a surface reaction whereby the iodine of an entrapped CH_3I reacts with water molecules and is oxidized to the I^0 state. This step is coupled with the subsequent removal of I^0 from the drop surface into the interior of the drop by means of the reduction of the I^0 to I^-.

The proposed mechanism suggests that a reagent that can minimize the surface barrier to penetration and increase the CH_3I retention on the surface, as well as enhance the solubility of I^0, should accelerate this step. Such behavior might be expected of surfactants, which not only reduce the surface barrier to penetration but also aid in the solubilization of I^0. Moreover, imino structures $(CH_2\text{-}CH_2\text{-}NH)_n$ built into these surfactants would have a relatively high affinity for the methyl group of CH_3I (with a tendency to form a quaternary ammonium salt) and thereby help to stabilize the activated complex on the drop surface. Finally, the addition of a strong reducing agent should help in the subsequent reduction of the I^0 in the interior of the drop and therefore aid in the removal of I^0 from the drop surface.

With these factors in mind we investigated the effect of an addition of a combination of a surfactant and a reducing agent to solution 1. This

solution was studied both with the addition of a surfactant alone, as well as with a surfactant combined with a strong reducing agent (formaldehyde); the latter was employed in order to speed up the interior reduction step. Our particular selection was predicated on the idea that formaldehyde is not only a strong reducing agent but its oxidation products are relatively innocuous (CO_2 and H_2O) and many troublesome side products are eliminated. These studies were extended to higher temperatures in order to cover the conditions expected to be encountered in an actual accident.

Experiments

A series of bubbler tests were performed with radioactively tagged CH_3I to test the concept at the bench level. Air containing approximately 5×10^{-7} moles of CH_3I (tagged with ^{131}I tracer) per liter was admitted to the bottom of the column of solution to be tested through a porous glass disk that produced fine bubbles. After passing through the 1 1/2-in.-diam 28-in.-deep column of liquid, the effluent gas was passed through two activated charcoal beds in series to collect the ^{131}I not retained by the solution.

The air containing the CH_3I was bubbled through the solution for 1 to 2 hr, and then clean air (containing no CH_3I) was bubbled through for a minimum of 16 hr to make sure that all the iodine not "held" by the solution would end up in the charcoal. The flow rate for both the contaminated air and the clean air purge was maintained at 20 ml/min. The length of time the contaminated air flowed (varied in one series of tests from 80 to 458 min) affected the results only a few percent.

At the end of each run the charcoal and an aliquot of the solution were "counted" to determine the efficiency of the solution. The second charcoal bed never showed more than a trace of activity. The ingredients added to a pH-9.1 0.3-wt % borate system were solution 1 and PEI 1000* surfactant (polyethyleneimine) with a nominal structure $(-CH_2-CH_2-NH-)_n$ in branched form. This surfactant comes in a spectrum of molecular weights ranging from 450 to 750 and all the way up to 100,000; the latter is called

*Trademark of Dow Chemical Co.

PEI 1000. The experiments were conducted with the PEI 1000 solely on the ground of its being readily available for testing. There would be a decided advantage in employing the lower molecular weight forms, such as PEI 6 (molecular weight 450 to 750), since a smaller weight of material would be required. The important point, however, was the need to have the imino structure $(-CH_2-CH_2-NH-)_n$ built into the surfactant. Since the imine structures are basic, they tend to be compatible with the base-borate system required for I_2 removal and chemical shimming.

Experimental Results

The efficiencies of CH_3I pickup by solution 1 as a function of varying amounts of the surfactant and formaldehyde are plotted in Fig. 3.4 and compared with results for solution 2 with various amounts of $Na_2S_2O_3$. On a molar basis the surfactant (PEI 1000) is far superior in efficiency of pickup of CH_3I to $Na_2S_2O_3$. This is not surprising, however, since the use of this surfactant, that is, one having a high molecular weight, involves the addition of a considerable mass of material to the spray liquid. It should be reemphasized, however, that surfactants structurally identical to PEI 1000, with molecular weights as low as 400, are available. Their use would provide one method of reducing the amount of surfactant required.

The gain in the use of surfactant can be shown in two ways. First, if we add surfactant to solution 2, which contains $Na_2S_2O_3$, the efficiency of the resultant system is increased. This indicates that this surfactant acts additively with the reducing agent $Na_2S_2O_3$ in the base-borate system. A striking improvement in the efficiency of CH_3I pickup, however, is shown when a reducing agent, such as formaldehyde, is added to the surfactant (Fig. 3.4 and Table 3.3). It is also suggested that it is, in part, the combination of formaldehyde and surfactant that gives significant improvement in the efficiency of CH_3I pickup, since formaldehyde alone (Table 3.3) has practically no affinity for CH_3I. Moreover, it is worth noting at this point that many candidates for CH_3I removal that appear to be effective when tested alone in water will fail when combined with the base-borate system. For example, tributyl phosphine oxide, which has been reported to

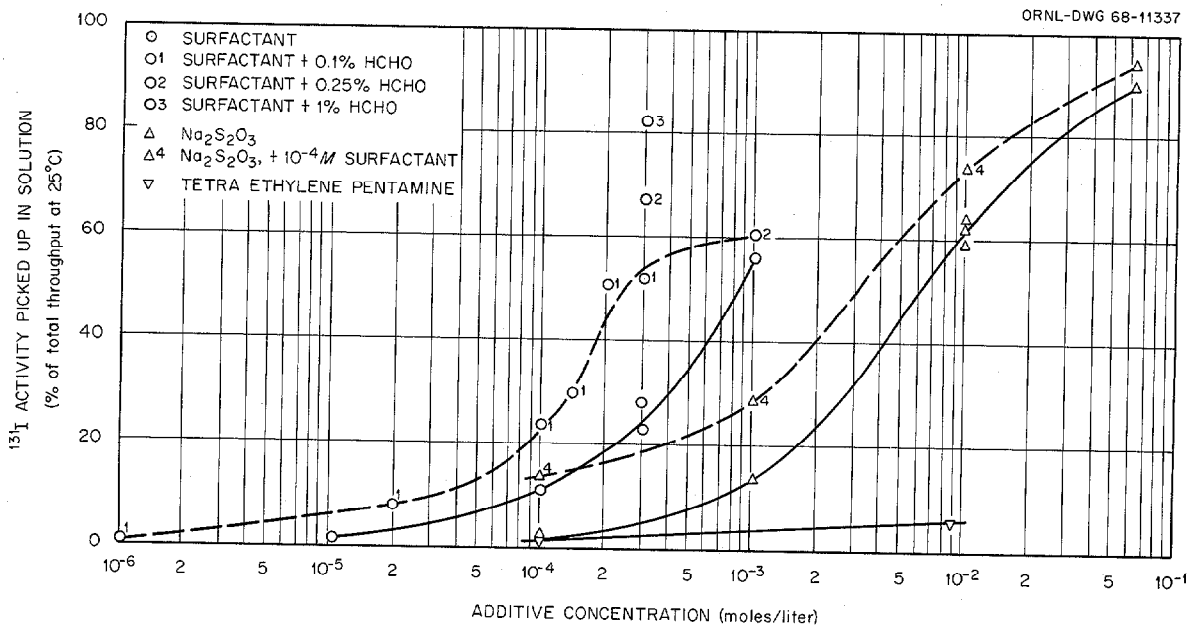

Fig. 3.4. Effect of Additive Concentration and Type on the Efficiency of CH_3I Absorption.

Table 3.3. Results of Tests of Effect of Additives on Efficiency of CH_3I Removal

Additive	Concentration[b]	Solution Temperature (°C)	Iodine Activity Picked up by Solution (%)
	Single Additive[a]		
$Na_2S_2O_3$	0.0001	25	2.0
	0.001	25	13.6
	0.001	70	79
	0.01	25	59
	0.01	25	62
	0.01	25	64
	0.063	25	90
	0.063	50	99+
Tributyl phosphine oxide	0.1%	25	0.5
Tetraethylene pentamine	0.0001	25	1.5
	0.0088	25	5.5
None		25	0
HCHO	0.1%	25	0.2
Surfactant[c]	0.00001	25	1.5
	0.0001	25	11.4
	0.0003	25	28
	0.0003	25	23
	0.0003	50	78
	0.0003	70	98
	0.001	25	56
	Two Additives[a]		
$Na_2S_2O_3$ Surfactant	0.001 0.0001	25	29
$Na_2S_2O_3$ Surfactant	0.01 0.0001	25	74
$Na_2S_2O_3$ Surfactant	0.0001 0.0001	25	14
HCHO Surfactant	0.1% 0.000001	25	0.6
HCHO Surfactant	0.1% 0.0001	25	24
HCHO Surfactant	0.1% 0.001	25	60
HCHO Surfactant	0.1% 0.0003	25	52
HCHO Surfactant	0.25% 0.0003	25	67
HCHO Surfactant	1.0% 0.0003	25	82

[a] In addition to the listed additives, all the solutions contained 0.3% boron and were adjusted to a pH of 9.1 with NaOH.

[b] Concentrations in moles per liter unless otherwise indicated.

[c] The surfactant used was Dow's PEI 1000, which has a molecular weight of 50,000 to 100,000. For the purpose of calculating molarity of the above solutions, a molecular weight of 60,000 was arbitrarily used.

have high affinity for CH_3I, has practically no affinity when tested in the base-borate system (Table 3.3).

The improved efficiency of CH_3I pickup of the surfactant when compared with that of the structurally related pentaethylamine (the latter in a nonsurfactant form) is an indication that the surfactant form is useful beyond the specific nature of its active chemical composition.

Finally, we examined the effect of temperature on the efficiency of CH_3I scavenging (Fig. 3.5). It is to be noted that temperature elevation increases the efficiency of CH_3I removal. This is especially gratifying, since it suggests that any CH_3I removal tests performed at room temperature give the lower limit on removal efficiencies, since higher temperatures are expected in the event of an accident.

In recapitulation, we suggest that each of the following factors increase the removal of methyl iodide in aqueous sprays containing a base-borate system: (1) surfactant and its functional group $(CH_2\text{-}CH_2\text{-}NH)_n$, (2) the presence of a reducing agent, such as formaldehyde, and (3) temperature. Specifically we suggest that the <u>combination</u> of imino-type surfactants of the Dow PEI series and small amounts of formaldehyde should make base-borate solution 1 useful in the removal of CH_3I. Moreover, bench-type results (Figs. 3.4 and 3.5 and Table 3.3) indicate that this combination may significantly increase the efficiency of CH_3I removal in spray solutions that are at present effective only for I_2 removal.

Preliminary pilot plant results for a spray containing standard base-borate solution 1 to which 2×10^{-5} <u>M</u> Dow PEI plus 0.22 wt % of formaldehyde was added indicate an improvement in removal of CH_3I. The results are discussed in Section 3.4 of this report.

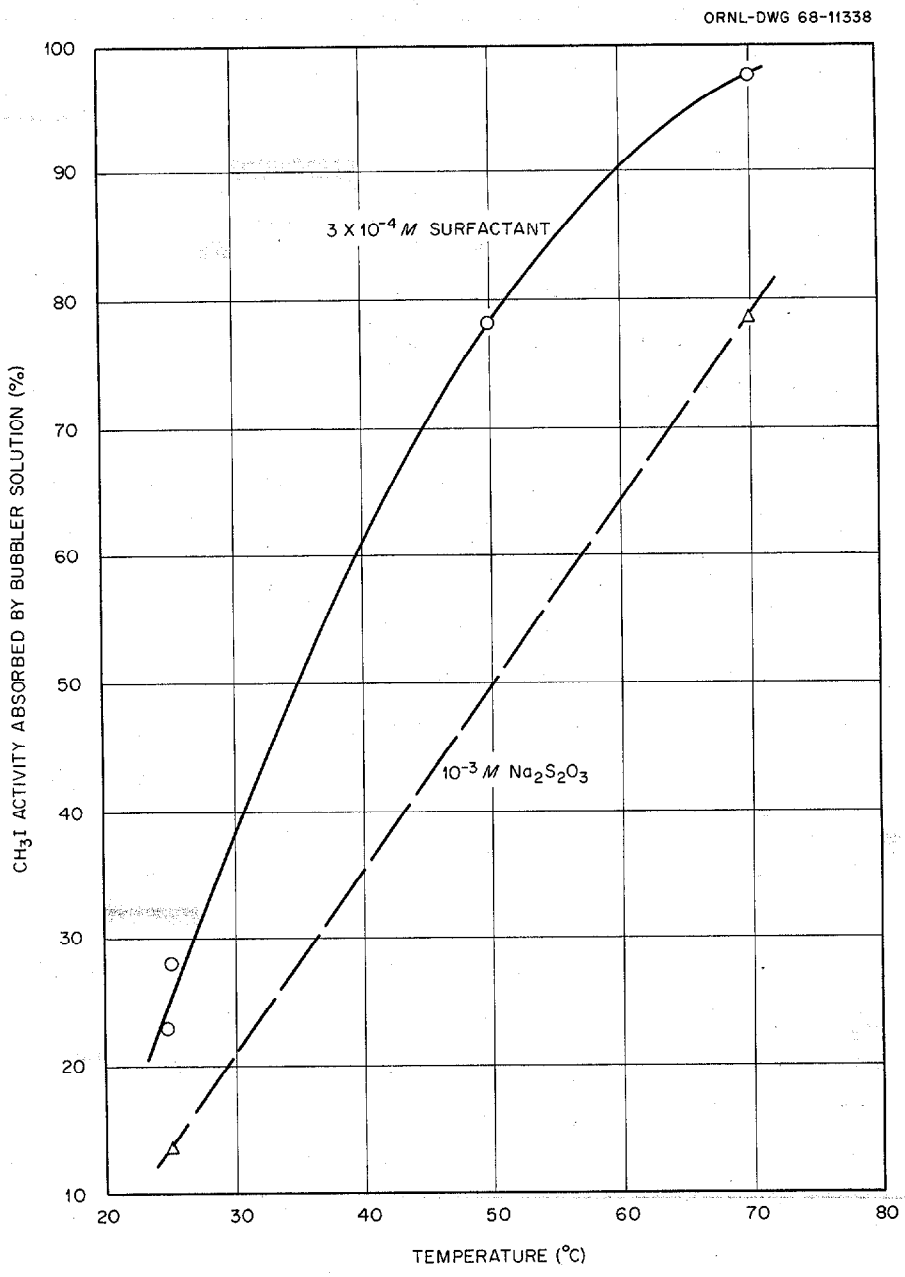

Fig. 3.5. Effect of Temperature on the Efficiency of CH_3I Absorption.

3.3 SPRAY STUDIES AT THE NUCLEAR SAFETY PILOT PLANT
(AEC Activity 04 60 80 01 1)

L. F. Parsly J. L. Wantland

The spray program in the Nuclear Safety Pilot Plant is designed to determine whether sprays can remove fission products from the containment atmosphere rapidly enough to reduce significantly the exposure calculated to result from a design-basis accident to a nuclear power plant. This pilot plant is large enough for tests of full-size nozzles. In view of the difficulty and uncertainty of spray-drop size measurements, empirical performance data probably provide the best basis for design calculations.

The overall problem consists of three main parts: (1) removal of iodine vapor, (2) removal of methyl iodide, (3) removal of particles. During the past year the pilot plant has dealt exclusively with the first two parts. We believe that we have obtained a good understanding of these. We have shown that iodine-vapor removal is gas-film controlled and that both the base–boric acid and base–boric acid–thiosulfate solutions proposed will work. We expect future work with iodine vapor to be limited to testing alternative spray-nozzle designs or to obtaining answers to specific questions raised during the evaluation of industry-designed spray systems. In the case of methyl iodide, we have found that the removal process is reaction-rate controlled and that a better solution than those now proposed is needed. We expect to do a few experiments on methyl iodide during the coming year to check the performance of solutions that look promising in laboratory-scale tests.

This year's program will emphasize particle removal, but we will also do a few iodine and methyl iodide experiments to carry out the objectives discussed above.

During the current reporting period we have been rearranging the sample-vaporizing facilities and have installed a furnace for generating particles by melting UO_2 with a plasma torch. Also, we moved the iodine and methyl iodide vaporizers to a new location. Zircaloy-clad UO_2 samples containing cesium were fabricated. Three experiments were run: one iodine-removal experiment, one methyl iodide-removal experiment, and one UO_2 melting experiment. Supporting data analysis work was continued.

Iodine Removal — Run 51

Run 51 was made to test a spray-nozzle design, the Bete fog nozzle (Type N5-303) being considered by a reactor manufacturer for containment building use. Initial conditions were 45 psig and 130°C, which are the maximum working conditions in the NSPP. Cold base—borate—thiosulfate solution was used.

Gamma-intensity measurements indicated an initial half-life of 35 sec. An overall decontamination factor for elemental iodine of 39,000 was obtained in 5 min 40 sec of spraying. Approximately 0.4% of the iodine in the sample was in organic form.

The results indicate that this nozzle is capable of giving adequate performance in removing iodine vapor.

UO_2 Melting Test — Run 52

Run 52 was made to check the new furnace configuration. The pressure and temperature were significantly higher than used in previous plasma-torch operations, and we found that it was necessary to reduce the torch-to-work spacing in order to achieve effective melting because the denser atmosphere shortened the plasma "flame." Good melting was achieved, and torch operation was stable.

Methyl Iodide-Removal Experiment — Run 53

Run 53 scheduled because solution screening experiments by the single-drop study group (see Sect. 3.2) had identified a promising candidate. During the interim between scheduling and performing the run, the screening program identified a still more attractive candidate and the latter was used. However, only a limited quantity of the critical reagent was available, and we were forced to use a much lower concentration than the laboratory experiments indicated should have been used.

The solution used was

H_3BO_3	0.28 M
NaOH	0.17 M
HCHO	0.074 M
PEI 1000 (polyethyleneimine)	2.7×10^{-5} M

The vessel was initially at 130°C and 45 psig, and the solution was maintained at 120°C. Spraying was continued for a period of 3 hr, but it was interrupted at 1 and 2 hr to take gas samples.

Run 53 was a duplicate of run 48, except for the PEI and formaldehyde added to the solution. The results of runs 48 and 53 are compared below:

	Run 48	Run 53
Overall decontamination factor	1.13	1.6
Half-life for methyl iodide, min	1000	260

These data show that use of the modified solution in run 53 reduced the methyl iodide half-life by a factor of about 4; however, the half-life obtained is still much longer than desirable. The laboratory data indicate that the methyl iodide-removal rate can be increased significantly by using a higher concentration of PEI. An experiment to test this has been scheduled.

Review of NSPP Experiments 39, 40, and 41

In runs 39, 40, and 41 we observed a very short initial half-life, which we concluded was due to the presence of elemental iodine in what were intended to be methyl iodide experiments. Just before this group of experiments, we had set up the feed system shown in Fig. 3.6 to enable experiments to be done in which methyl iodide and iodine vapor would be introduced simultaneously. We thought it would be desirable to heat the iodine vaporizer tee by means of a steam jacket to assure that none of the methyl iodide would condense in it. We used 75-psig steam (160°C) in run 39 and 25 psig steam (130°C) in runs 40 and 41. In later runs, we did not heat the tee, and we then found that we did not get the short initial half-life and also we did not find a significant fraction of the activity in the tee, as we had in runs 39, 40, and 41.

It occurred to us that we might be able to determine how much conversion to iodine was accomplished by the following procedure. We made subsequent runs that duplicated runs 39, 40, and 41, except for heating the tee, and from these we got the decontamination factor for methyl iodide. We had purge data from which we could determine how much methyl

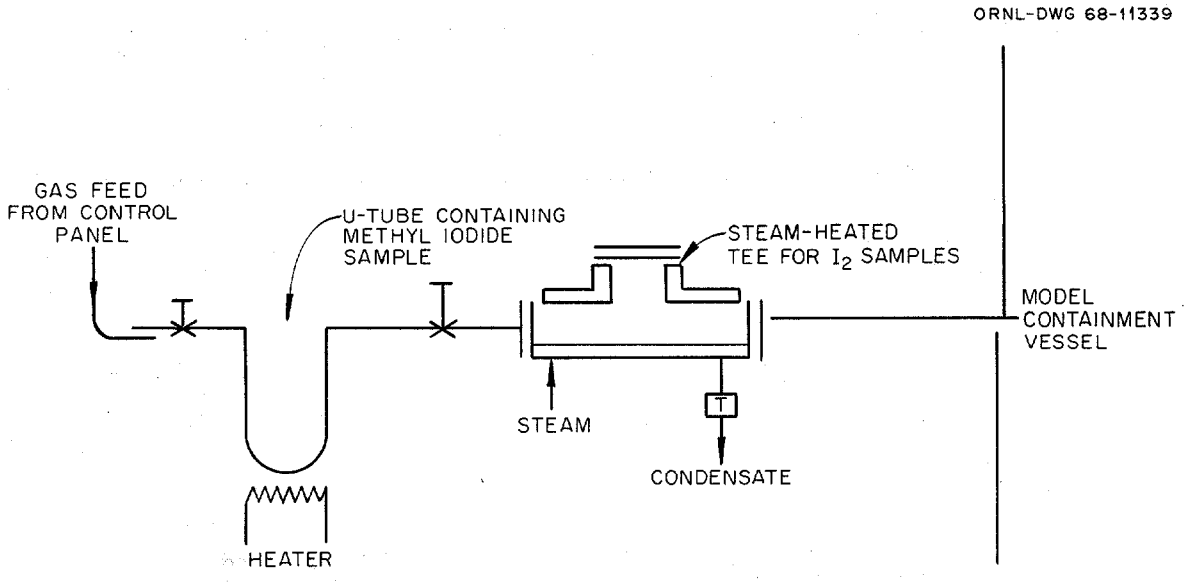

Fig. 3.6. NSPP Iodine Feed System.

iodide was left at the end of the runs and from this and the decontamination factor we could calculate how much was present at the start. All the rest of the iodine found in the MCV was presumed to be iodine vapor. This calculation was made and the results are presented in Table 3.4. The conversion to iodine turned out to be surprisingly high; the table shows that it was over 90% in all three runs.

We believe a unique set of circumstances was necessary to cause the thermal decomposition of methyl iodide. The amount of steam heating applied was sufficient to prevent condensation from the steam-air atmosphere (in runs 39 and 40) onto the tee surface, so the surfaces were dry. We cannot be sure whether the methyl iodide entered the tee as liquid or as vapor, but the concentration in the tee at the start of the experiment, just after the valves on the U-tube had been opened and sweep air flow had been started, was high.

We believe this shows that rapid decomposition of methyl iodide can be expected if a high concentration is brought into contact with a dry stainless steel surface heated to 130 to 160°C. We doubt, however, that use could be made of this mechanism as a means of removing methyl iodide from accident atmospheres.

Partition Coefficient for Iodine Between Water and Air

In making calculations of spray performance it is often necessary to know the partition coefficient of iodine between water and air under the conditions of temperature, pH, and concentration applying at the moment. Eggleton[1] presents a procedure for calculating the partition coefficient. The calculations are quite simple to do, and to get more flexibility we have programmed them for the CDC-1604. The program calculates the partition coefficient as a function of concentration down to 10^{-9} mole total iodine per liter for a given temperature and pH. The running time per case (combination of temperature and pH) is 1 to 2 sec.

Table 3.4. Results of Experiments 39, 40, and 41

	Run 39	Run 40	Run 41
Conditions			
Solution	0.28 M H_3BO_3 + 0.17 M NaOH	0.28 M H_3BO_3 + 0.17 M NaOH + 0.06 M $Na_2S_2O_3$	0.28 M H_3BO_2 + 0.17 M NaOH
Initial pressure, psig	45	45	3
Initial temperature, °C	130	130	30
Solution temperature, °C	120	120	30
Solution flow, gpm	11	11	11
Spraying time, min	180	180	180
Tee steam pressure, psig	75	25	25
Results			
Initial half-life (gamma data), sec	162	30	36
Iodine removal by spray, mc	7.843	15.071	17.862
Iodine in purge, mc	0.003	$<1.5 \times 10^{-4}$	0.029
CH_3I in purge, mc	0.773	0.104	2.378
U-tube iodine residue, mc	0.022	10.226	0.752
Iodine deposited in MCV, mc	3.826	14.562	5.431
Decontamination factor for CH_3I (parallel run)	1.13	8.3	1.03
CH_3I collected by spray	0.089	0.759	0.062
Iodine collected by spray (difference)	7.754	14.312	17.800
Decontamination factor for I_2	2580	>9500	610
Percentage of total iodine converted to I_2	93.3	97.9	91.4

Calculations of Terminal Velocity of Drop

One of the subroutines in our spray performance computer program calculates the terminal velocity of the spray drop by a stepwise routine for which an initial drop velocity is assumed, and it calculates the position and velocity of the drop after a short time interval (the basic time interval is 0.01 sec, but we add a constraint that the vertical velocity may not change by more than 2% per time step). When the vertical velocity change in a time step becomes less than 0.02%, we assume that the terminal velocity has been attained. We use drag coefficients recommended by Dallevalle,[2] primarily because these are given as explicit functions of the Reynolds number and thus were easy to build into the computer program.

In Fig. 3.7 we show a comparison of the terminal velocities obtained by our subroutine with experimental values reported by Mason.[3] The agreement is good, with the worst deviation being about 17% for 1300-μ-diam drops. Since the rate coefficient varies as the square root of velocity, the accuracy is adequate.

Theoretical Treatment of Absorption Plus Fast Chemical Reaction

In the previous report[4] we were able to show that removal of methyl iodide rapidly enough to yield a significant dose reduction can only be accomplished in the fast reaction regime. A reaction is considered "fast" if significant reaction occurs during the lifetime of a spray drop. We have adapted a solution by Danckwerts[5] to calculate the enhancement relative to purely physical absorption resulting from the chemical reaction. We assume negligible gas-film resistance and a rigid-drop model.

The equation derived is

$$E(A,\theta) = \frac{\sum_{n=1}^{\infty} \frac{1}{A + n^2\pi^2}\left[A\theta + \frac{n^2\pi^2(1 - e^{-A\theta}e^{-n^2\pi^2\theta})}{A + n^2\pi^2}\right]}{\sum_{n=1}^{\infty} \frac{1 - e^{-n^2\pi^2\theta}}{n^2\pi^2}},$$

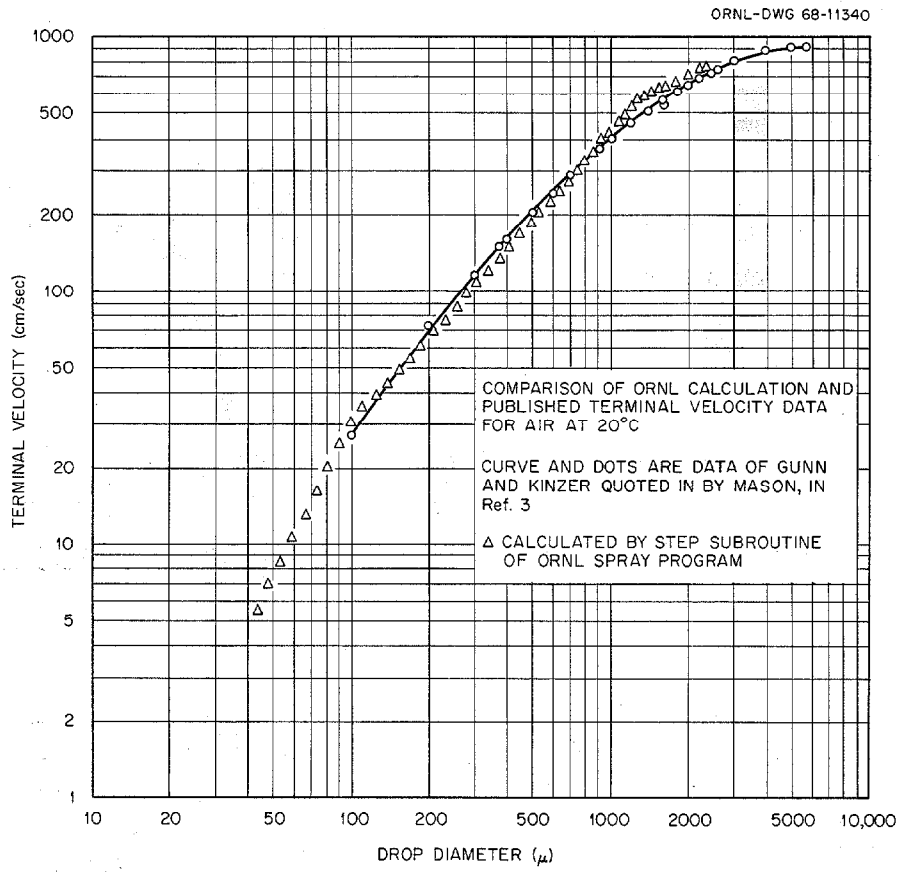

Fig. 3.7. Comparison of ORNL-Calculated and Published Terminal Velocity Data for Drops in Air at 20°C.

where

E = enhancement, dimensionless,

A = dimensionless rate constant = $r_0^2 k/D$,

θ = dimensionless time = Dt/r_0^2,

k = pseudo-first-order reaction rate constant, sec^{-1},

r_0 = radius of drop, cm,

D = diffusivity, cm^2/sec,

t = time, sec,

n = 1, 2, 3, ...

Numerical solutions of this equation are being calculated. They will be reported in the next report in this series.

References

1. A. E. J. Eggleton, A Theoretical Examination of Iodine-Water Partition Coefficient, British Report AERE-R-4887.

2. J. M. Dallavalle, Micromeritics, The Technology of Fine Particles, 2nd ed., Putnam, New York, 1948.

3. B. J. Mason, Cloud Physics, Oxford, 1957.

4. L. F. Parsly and J. K. Franzreb, pp. 64–69 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1968, USAEC Report ORNL-TM-2283, Oak Ridge National Laboratory, July 30, 1968.

5. P. V. Danckwerts, Absorption by Simultaneous Diffusion and Chemical Reaction into Particles of Various Shapes and into Falling Drops, Trans. Faraday Soc., 47: 1014 (1951).

3.4 RADIATION STABILITY OF SPRAYS

(AEC Activity 04 60 80 01 1)

H. E. Zittel

A study of radiolytic sulfur formation in borated $Na_2S_2O_3$ solutions was completed. The study involved the relationship between the dose required to bring about sulfur formation and pH. As indicated in the previous report,[1] the minimum dose at which radiolytic sulfur appears is a function of the pH of the solution being irradiated. The data obtained are given in Table 3.5. As may be seen, the dose required to bring about visible sulfur formation is essentially a log function of the pH of the solution. It should be emphasized that the dose values given are not to be considered highly accurate; they are, essentially, "ball park" figures. The samples were inspected for visible sulfur at approximately 2×10^6-r intervals for the first sample (pH 4.9) and then at appropriate intervals for the samples of higher pH. Therefore the dose figure given for sulfur appearance can be no more accurate than the frequency of inspection

Table 3.5. Effect of pH on Radiolytic Sulfur Formation

Test solutions: 1 wt % $Na_2S_2O_3$, 3000 ppm B, with sufficient 1.0 N NaOH to give desired pH

Solution pH[a]	Sulfur Appearance Dose[b] (r)
	$\times 10^7$
4.9	1
6.0	2
8.0	6
9.0	15
9.6	30

[a] pH as measured before irradiation.

[b] Dose required to bring about visible sulfur formation.

would allow. The results of the study seem to show that if the solution pH is kept fairly alkaline (pH > 9), very little radiolytic sulfur can be expected to exist at any dose anticipated.

All the radiolysis studies have been carried out at ambient source temperatures (40 to 50°C). Because of lack of information on the effect of temperature on the radiolytic reactions involved in the proposed sprays, it is considered highly necessary to repeat the key studies at a range of temperatures consistent with those expected in the accident case. An apparatus has been devised and built to allow this to be accomplished. The first study has been completed, and the results are given in Fig. 3.8. The data indicate that the effects of radiation and temperature on alkaline borated $Na_2S_2O_3$ are essentially additive. The values shown as "estimated values" were obtained by extrapolation of previously reported data. Corrections were made to fit the conditions under which this study was made. The agreement between the experimental results and calculated additive results is surprisingly good in view of the significant errors involved in the extrapolation of the previous data. The results of this study indicate that the proposed alkaline borated $Na_2S_2O_3$ spray will withstand both the thermal and radiation conditions expected.

Reference

1. H. E. Zittel, pp. 70–73 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1968, USAEC Report ORNL-TM-2283, Oak Ridge National Laboratory, July 30, 1968.

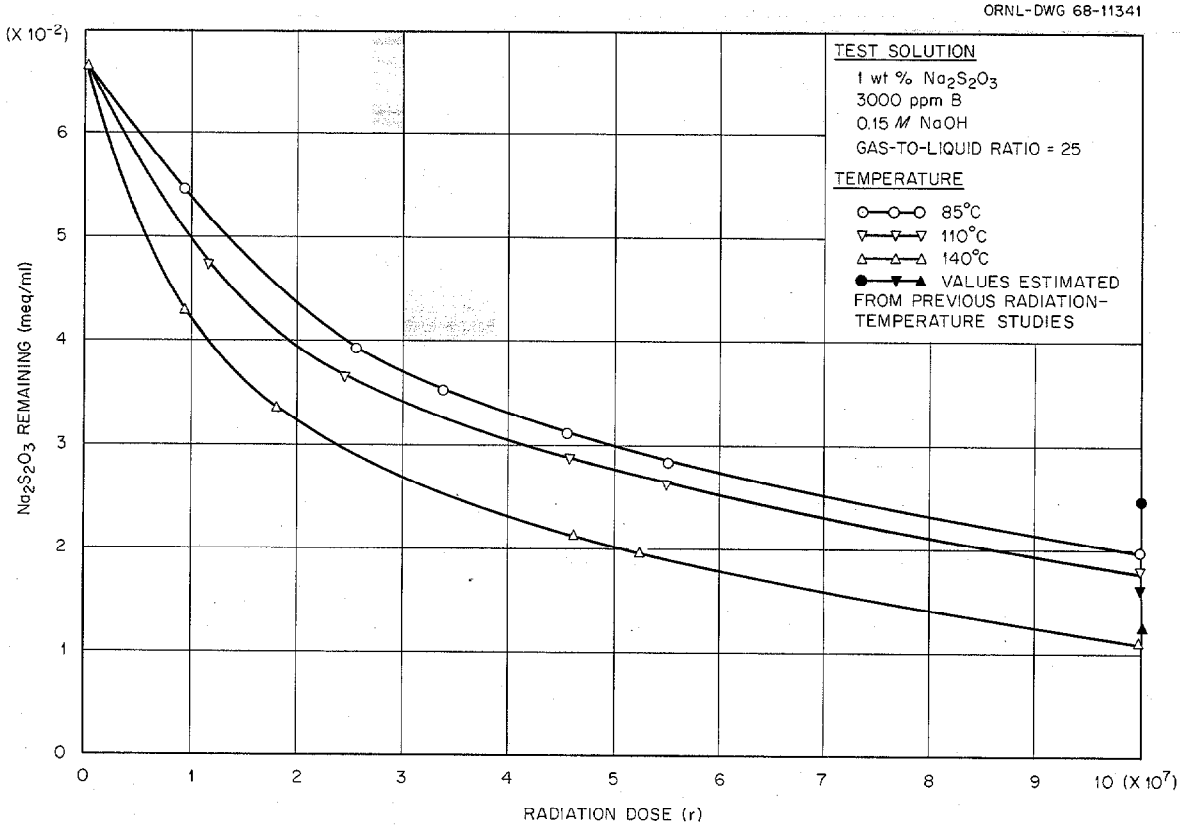

Fig. 3.8. Effect of Temperature and Radiation on $Na_2S_2O_3$ Reactions.

3.5 CORROSION STUDIES

(AEC Activity 04 60 80 01 1)

J. C. Griess H. C. Savage
P. D. Neumann A. L. Bacarella

A facility for testing the corrosion resistance of a variety of materials in iodine-absorbing sprays, as well as in the liquid phase of such solutions, was constructed, and preliminary tests were conducted. Figure 3.9 is a sketch of the apparatus showing the dimensions of the overall facility and the various components in the system. All surfaces of the system that are in contact with the solutions are type 304 stainless steel.

The loop is designed to operate at pressures up to 150 psig and has provisions for removing samples for analyses from the liquid and gas phases. A canned-rotor centrifugal pump recirculates the test solution. There are eight thermocouples located at various positions on the loop. One is located on the line immediately ahead of the spray nozzle, and this thermocouple is used to control the temperature of the spray solution. Two thermocouples are located in wells in the spray chamber; one terminates near the top and is in the spray region and the other extends to within 8 in. of the bottom and will usually be in the liquid phase. As the solution leaves the spray chamber, it is cooled slightly to prevent possible cavitation of the pump and is then heated or cooled to the desired temperature by a heater-cooler in the line to the nozzle. This line also contains a flowmeter to monitor the flow rate of solution passing through the nozzle.

A full-cone center-jet spray nozzle (Spray Engineering Co., Burlington, Mass., Nozzle 3C) has been used at a flow rate of about 0.5 gpm. The orifice in the nozzle is 0.093 in.

A stainless steel rack from which 108 specimens can be suspended on Teflon-insulated hangers fits into the spray chamber and rests on the bottom. In tests to date the depth of liquid in the spray chamber was 24 in., and thus the distance through which the spray fell was 28 in. Similar specimens have been exposed both in the spray and totally submerged in the liquid.

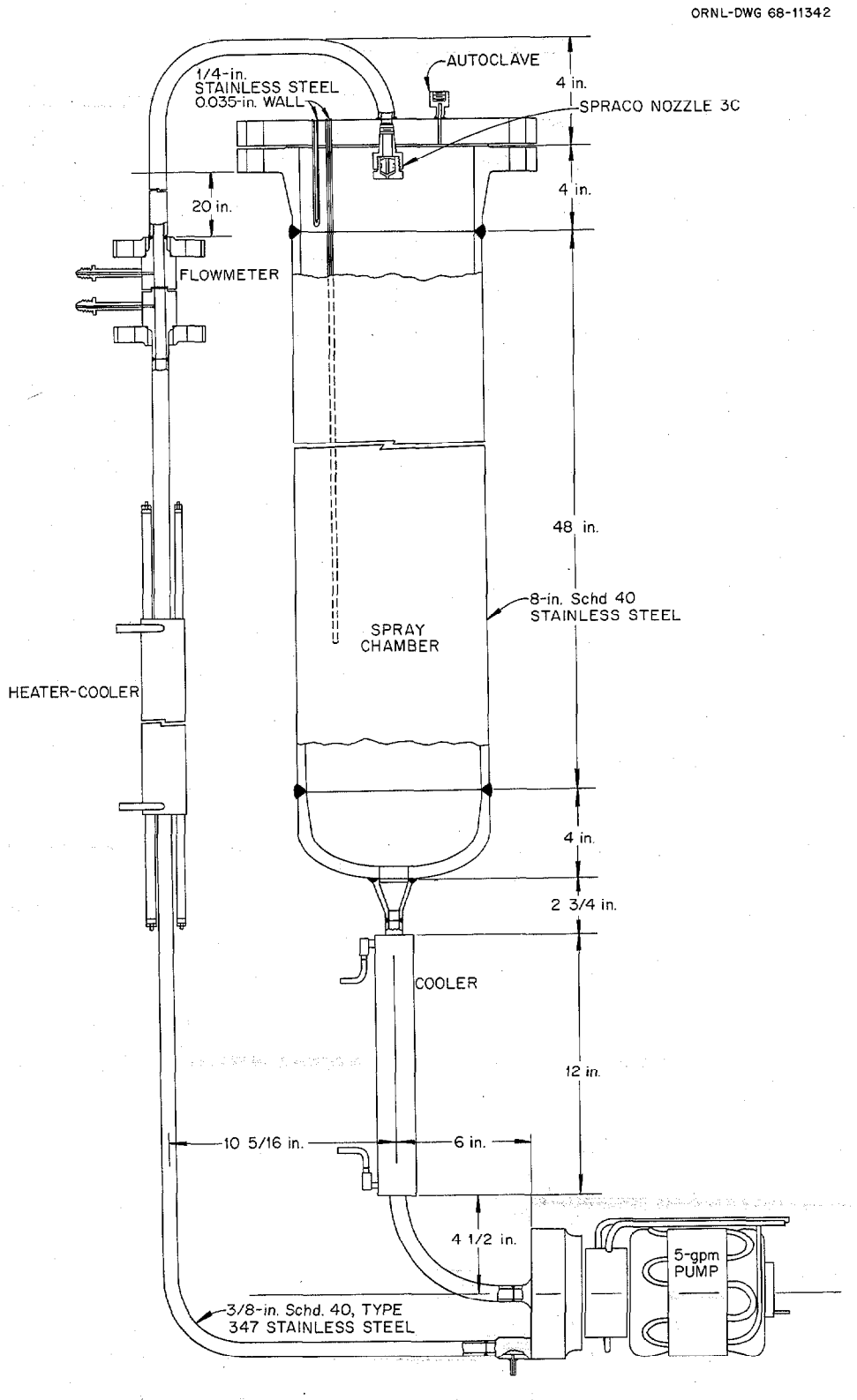

Fig. 3.9. Sketch of Spray Loop for Corrosion Tests.

Figure 3.10 is a photograph of the test facility. The sample rack is shown at the left.

The only test fully completed to date was one in which the spray solution contained 0.15 m NaOH and 0.28 m H_3BO_3 and had a room temperature pH of 9.3. The temperature of the test solution was 100°C, and the atmosphere was air at room temperature. The test lasted 168 hr but was interrupted after cumulative times of 24 and 72 hr in order to determine weight changes of the specimens. At each interruption new solution of the same composition as the original was added.

The specimens were rectangular coupons with a 1/4-in. hole drilled near one end so that they could be hung on the rack. All were lightly abraded, degreased, and weighed before being placed in the test chamber. When the specimens were removed, they were scrubbed with a soft brush, dried, and weighed. Four specimens of each of the following materials were exposed in the spray region and two were totally submerged in the solution: types 304 and 316 stainless steel; A-108 and A-210 carbon steel; 90-10 and 70-30 cupronickel; copper; Monel 400; 1100, 3003, 5052, and 6061 aluminum; Inconel 600 and 718; and Zircaloy-2.

The results indicated that the stainless steels, Inconels, cupronickels, Monel, and Zircaloy-2 experienced negligible changes in appearance and negligible weight loss. Both carbon steel alloys developed occasional rust spots and showed slight weight gains; corrosion damage was insignificant. Copper underwent slight attack, with the weight losses given in Table 3.6. Also included in Table 3.6 are the weight losses of the aluminum specimens. All aluminum alloys, except 5052, corroded at such high rates that they were not included in the final 96-hr test period.

All the aluminum alloys corroded more or less uniformly, and with a few exceptions the agreement among replicate samples was satisfactory. A weight loss of 6.86 mg/cm^2 corresponds to a uniform penetration of 1 mil. Thus all aluminum alloys, except 5052, had intolerably high corrosion rates of several inches penetration per year. The 5052 alloy developed a hard, smooth coating that seemed to greatly minimize attack. Although the results do not allow an accurate determination of the corrosion rate, it appears that after an initial period of relatively high attack, corrosion rates in the range of a few mils per year might be expected.

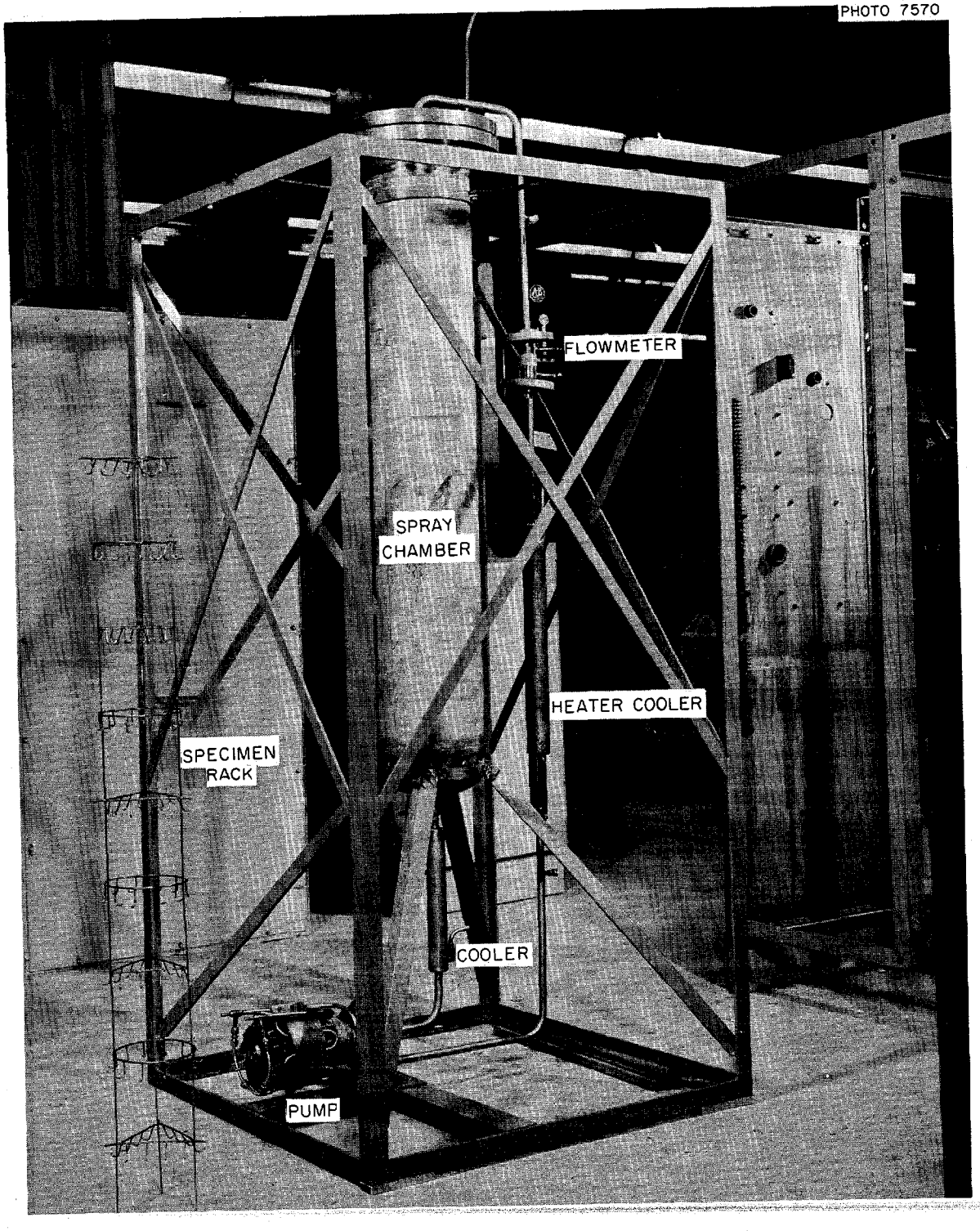

Fig. 3.10. Spray Loop for Corrosion Tests.

Table 3.6. Corrosion of Aluminum Alloys and Copper in a 0.15 m NaOH–0.28 m H_3BO_3 Solution at 100°C

Alloy	Cumulative Weight Loss (mg/cm^2)					
	24 hr		72 hr		168 hr	
	In Spray	In Solution	In Spray	In Solution	In Spray	In Solution
1100 aluminum	69	64	145	168		
	81	65	166	175		
	67		193			
	76		141			
3003 aluminum	61	29	171	65		
	52	25	137	63		
	33		99			
	38		64			
5052 aluminum	11	1.1	11	0.94	12	1.4
	7.3	1.1	7.1	0.95	12	1.5
	1.1		1.2		3.8	
	1.3		1.1		3.3	
6061 aluminum	40	29	113	78		
	33	28	68	79		
	35		96			
	46		86			
Copper	0.16	0.20	0.40	0.17	0.85	0.28
	0.29	0.09	0.81	0.04	1.30	0.16
	0.11		0.14		0.43	
	0.10		0.04		0.36	

In all cases attack was greater on those specimens in the spray than on those totally submerged.

Copper suffered slight, but measurable, corrosion, as indicated in Table 3.6. A weight loss of 22.6 mg/cm^2 corresponds to a uniform penetration of 1 mil. Thus the data indicate that the maximum corrosion rate observed on any of the copper specimens was 3 to 4 mils penetration per year.

After the first two parts of the run, the solution contained large amounts of flocculent aluminum hydroxide, but after settling it was colorless. In no case did the pH of the solution change, and the suspended solids had no effect on flow through the spray nozzle. In those cases where aluminum was in the system the pressure increased during the run due to hydrogen from the corrosion of aluminum; during the 48-hr test interval the pressure increased by 20 psi.

A second run was made under the same initial conditions as for the first, except that the solution contained 0.15 m NaOH, 0.28 m H_3BO_3, and 0.064 m $Na_2S_2O_3$. Corrosion data have not yet been obtained because of plugging of the spray nozzle. Aluminum reduced at least some of the thiosulfate to sulfide and corrosion products (probably sulfides of copper and nickel), which plugged the 0.093-in.-diam orifice in the spray nozzle. In a subsequent attempt to complete the run the aluminum specimens were removed and a 6-in.-diam sintered stainless steel filter was placed in the line to the nozzle. After the system reached temperature, the flow through the filter began to decrease, and within 2 hr the flow had completely stopped. The filter appeared to be plugged with corrosion products present in the solution, which in the absence of aluminum contained no sulfide. It now appears that without aluminum in the system the corrosion products suspended in the solution will not plug the nozzle, and hopefully the test can be completed. Detailed results will be given in the next report in this series.

3.6 PRESSURE-SUPPRESSION EXPERIMENTS

(AEC Activity 04 60 80 01 1)

F. T. Binford L. E. Stanford
C. C. Webster

The experimental setup was revised to permit pressure-suppression experiments with injection pipes up to 2 in. in diameter (the largest pipe previously used was 1 in. in diameter). The larger pipe provides a factor of 4 increase in the flow area, and experiments of this magnitude should considerably reinforce the information already obtained. Experiments will begin in early September.

The General Electric Company has supplied some representative blowdown rates for air, steam, and water following a design-accident rupture of a typical power reactor. These rates, given in Table 3.7, indicate that not only steam-air mixtures but also steam-air-water mixtures must be considered in the evaluation of pressure-suppression systems. There is a need for even qualitative information on the results of injecting such mixtures into water. Although studies involving steam-air mixtures were anticipated in basic investigations at ORNL, no provisions were made for studies involving steam-water or steam-air-water mixtures. Since such investigation now appears basic to a thorough understanding and description of pressure-suppression systems, equipment to permit such experiments

Table 3.7. Blowdown Rates Following Design Accident in Typical BWR

Time After Rupture (sec)	Mass Flow ($lb/hr \cdot ft^2$)		
	Air	Steam	Water
4	68,000	160,000	344,000
8	600	162,000	361,000
12	50	104,000	112,000
16	50	45,000	39,000
20	50	34,000	15,000

was designed and is being fabricated. Initial experiments with steam-air-water mixtures should be possible within a few weeks.

Following the experiments with the 2-in.-diam injection pipe, a series of runs will be conducted with nonradioactive molecular iodine. These experiments will be designed to determine the problems associated with estimating the iodine-removal efficiency of pressure-suppression pools.

3.7 SCALE-MODEL TESTS OF FISSION-PRODUCT REMOVAL IN SUPPRESSION POOL*

(AEC Activity 04 60 80 01 1)

M. Siegler D. P. Siegwarth

Absorption of fission products in pressure-suppression pools under simulated loss-of-coolant accident conditions is being studied in a 1/10,000-scale model. Detailed design of experiments and procurement and fabrication of equipment are under way. Equipment debugging is scheduled to start by early October.

Model and Experiment Design

A number of shakedown and exploratory tests will be needed to check out the experimental apparatus, sample systems, and instrumentation. The detailed plans for these tests were completed.

Preliminary Shakedown Procedures

The entire system will be pressurized (30 to 40 psi) with air, and the pressure-time response will be recorded over a period of several hours to leak-check the containment system. The operation of the pressure transducers and readout equipment will be checked at the same time. Thermocouples and temperature-recording equipment will be checked and calibrated prior to installation in the model containment system.

Since the blowdown transient for the simulated maximum credible accident (reactor coolant-recirculation line rupture) is over in about 2 min, the delay time in the pressure-suppression vessel sample system must be accurately determined. The delay time can be calculated from the known sample system volume and the volumetric flow rates, which will be measured under the simulated blowdown conditions (suppression-vessel pressure of 5 to 40 psi).

Some methyl iodide may be absorbed by or react with the Carbo Zinc No. 11 coating on the interior of the pressure-suppression vessel, and

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*Work performed under subcontract by General Electric Company, San Jose, California.

heating of the drywell walls could possibly cause some methyl iodide decomposition. The time-dependent methyl iodide concentration in the hot drywell and the suppression vessel will be measured to determine whether either of these possible surface effects is significant.

Blowdown Experiments

The blowdown tests required to debug the equipment are summarized in Table 3.8. Plans for the first six tests specify a small break area (approximately 1/5 the maximum break) to make sure that all components of the test equipment are operating properly before tests at larger break areas. Conditions for the tests in Table 3.8 include an initial steam pressure of 1000 psi in the model reactor vessel. Except for the first test, the electric heaters in the reactor vessel will be programmed to simulate fission-product decay heat.

The primary purpose of these blowdown tests is to determine the operating conditions necessary to simulate the computer calculated pressure and temperature response of the prototype and to measure the duration of air flow in the downcomer. The model should approximate the blowdown time of the prototype because the flow areas and vessel volumes of the prototype were scaled down by the same factor.[1] However, due to distortions in the energy storage and piping connections, experimental adjustments of the model flow areas and initial vessel temperatures will be necessary in order to properly simulate the pressure and temperature transients during blowdown. Six of the tests in Table 3.8 will be devoted to determining the areas and initial temperatures that will give the proper time sequence of events. Because the noncondensables in the drywell are ejected into the suppression pool during a blowdown and may transport fission products to the suppression-pool gas phase, it is important to know the duration of air flow in the downcomers. The concentration of noncondensables in the downcomer will be measured in four of the blowdown tests in Table 3.8. The hardware needed to measure the noncondensable concentration in the downcomer is presently being checked out in the laboratory.

As indicated in Table 3.8, provisions have been made in the blowdown experiments to debug the sample systems and observe the pressure-

Table 3.8. Blowdown Experiments for Equipment Shakedown

Run No.	Drywell Heat[a]	Downcomer Submersion (ft)	Break Area[a] (% of total)	Connection of Blowdown Line		Purpose of Run
				Drywell	Pressure Vessel	
BE-1	Off	4	20	Bottom	Bottom	Blowdown behavior test
-2[b]	240°F	4	20	Bottom	Bottom	Transient temperature and pressure sample system shakedown and suppression-pool behavior test
-3[b]	BE-2	4	BE-2	Bottom	Bottom	Transient temperature and pressure sample system shakedown and suppression-pool behavior test
-4[b]	BE-3	1	BE-3	Bottom	Bottom	Transient temperature and pressure suppression-pool behavior test
-5	BE-4	4	BE-4	Bottom	Bottom	Test of duration of air flow in vent
-6	BE-4	4	BE-4	Top	Bottom	Test of duration of air flow in vent
-7	Off	4	100	Bottom	Bottom	Transient temperature and pressure blowdown-behavior test
-8[b]	BE-4	4	100	Bottom	Bottom	Transient temperature and pressure suppression-pool behavior and sample system operation tests
-9[b]	BE-8	1	BE-8	Bottom	Bottom	Transient temperature and pressure suppression-pool behavior tests
-10	BE-9	4	BE-9	Bottom	Bottom	Tests of duration of air flow
-11	BE-9	4	BE-9	Top	Bottom	Tests of duration of air flow

[a]Exact requirements to be determined from indicated experiment.

[b]Degassed water in suppression pool.

suppression pool operating characteristics. The pressure-suppression vessel has provisions for either visual or photographic observations of the water pool during a blowdown transient.

Preliminary Methyl Iodide Experiments

The planned preliminary methyl iodide experiments are summarized in Table 3.9. The purpose of the first test is to measure the spatial variation of the methyl iodide concentration in the pressure-suppression vessel gas space during a blowdown transient. The results of this test will indicate how many sample locations will be needed to obtain a representative measurement of the methyl iodide concentration. The next three tests are designed to investigate the effect of different methyl iodide injection locations and determine which location best simulates the expected fission-product behavior of the prototype.

The model can be used to simulate a break in the reactor-coolant recirculation-line or a steam-line break. The fifth test in Table 3.9 will be a top blowdown to simulate a steam-line break. The capability of the pressure-suppression pool to absorb methyl iodide in the absence of noncondensable gases will be measured in the final test. The results can be compared with those for an ordinary blowdown at the same conditions to determine the effect of the noncondensable bubble transport mechanism on the absorption efficiency of the suppression pool.

Experimental Equipment

The major equipment items (reactor vessel, drywell, and pressure-suppression pool) and their mode of operation were described previously.[1,2] With the completion of the pressure-suppression vessel during the present reporting period, the fabrication of these vessels is now complete. The interior of the drywell has a thin coating of fluorinated ethylene propylene. The interior of the pressure suppression pool is being coated with Carbo Zinc No. 11, which is typical of the paint used in the primary containment system of the prototype. The exteriors of both vessels will be protected with weather coating. A support structure for the facility was fabricated and erected adjacent to the Development

Table 3.9. Planned Preliminary Methyl Iodide Experiments

Run No.	Drywell Heat[a]	Downcomer Submersion (ft)	Break Area[a] (% of total)	Connection of Blowdown Line		CH_3I Injection Location	Purpose of Run
				Drywell	Pressure Vessel		
PMI-1	BE-10	4	BE-10	Bottom	Bottom	Blowdown line	Test of spatial distribution of CH_3I in suppression-chamber gas phase
-2	BE-10	4	BE-10	Bottom	Bottom	Blowdown line	Determine effect of injection location
-3	BE-10	4	BE-10	Bottom	Bottom	Drywell	Determine effect of injection location
-4	BE-10	4	BE-10	Bottom	Bottom	Drywell vent	Determine effect of injection location
-5	BE-10	4	BE-10	Bottom	Top	Blowdown line	Transient temperature and pressure CH_3I absorption test
-6[b]	BE-10	4	BE-10	Bottom	Bottom	Blowdown line	CH_3I absorption test in absence of noncondensables

[a]Exact requirements to be determined from indicated experiment.

[b]Drywell and suppression-pool water degassed before blowdown.

Engineering test building at the General Electric San Jose site. The drywell and pressure-suppression vessel are mounted on the support structure as shown in Fig. 3.11. The drywell and pressure-suppression vessels will be insulated with polyurethane foam and the reactor vessel with asbestos.

A detailed piping and instrument diagram of the model is given in Fig. 3.12. The sampling manifolds for the pressure-suppression pool gas are located in the Chemical Engineering Laboratory, adjacent to the test facility, and connected to the vessels with 1/8-in.-O.D. stainless steel tubing. Samples will be taken for gas chromatographic analysis with Pressure-Lok gastight syringes. The sample port in the drywell vent that is to be used for measuring the concentration of noncondensable gases will be connected to sample traps near the experimental apparatus.

In order to study the absorption of methyl iodide in the pressure suppression pool during a simulated loss-of-coolant accident, it is necessary to start the recording equipment before the initiation of the break, a running-time meter concurrent with the break, and injection of the methyl iodide a few seconds after the break. The timing circuit shown in Fig. 3.13 was designed to actuate the recorder drives, rupture-disk pressurizing system, running-time meter, and methyl iodide injection system. This circuit uses a series of adjustable time delays to automatically control the time sequence of events at the beginning of an absorption test.

References

1. F. T. Binford, L. E. Stanford, and C. C. Webster, pp. 83–85 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for March–April 1968, USAEC Report ORNL-TM-2230, Oak Ridge National Laboratory, May 30, 1968.

2. F. T. Binford, L. E. Stanford, and C. C. Webster, pp. 75–95 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1968, USAEC Report ORNL-TM-2283, Oak Ridge National Laboratory, July 1968.

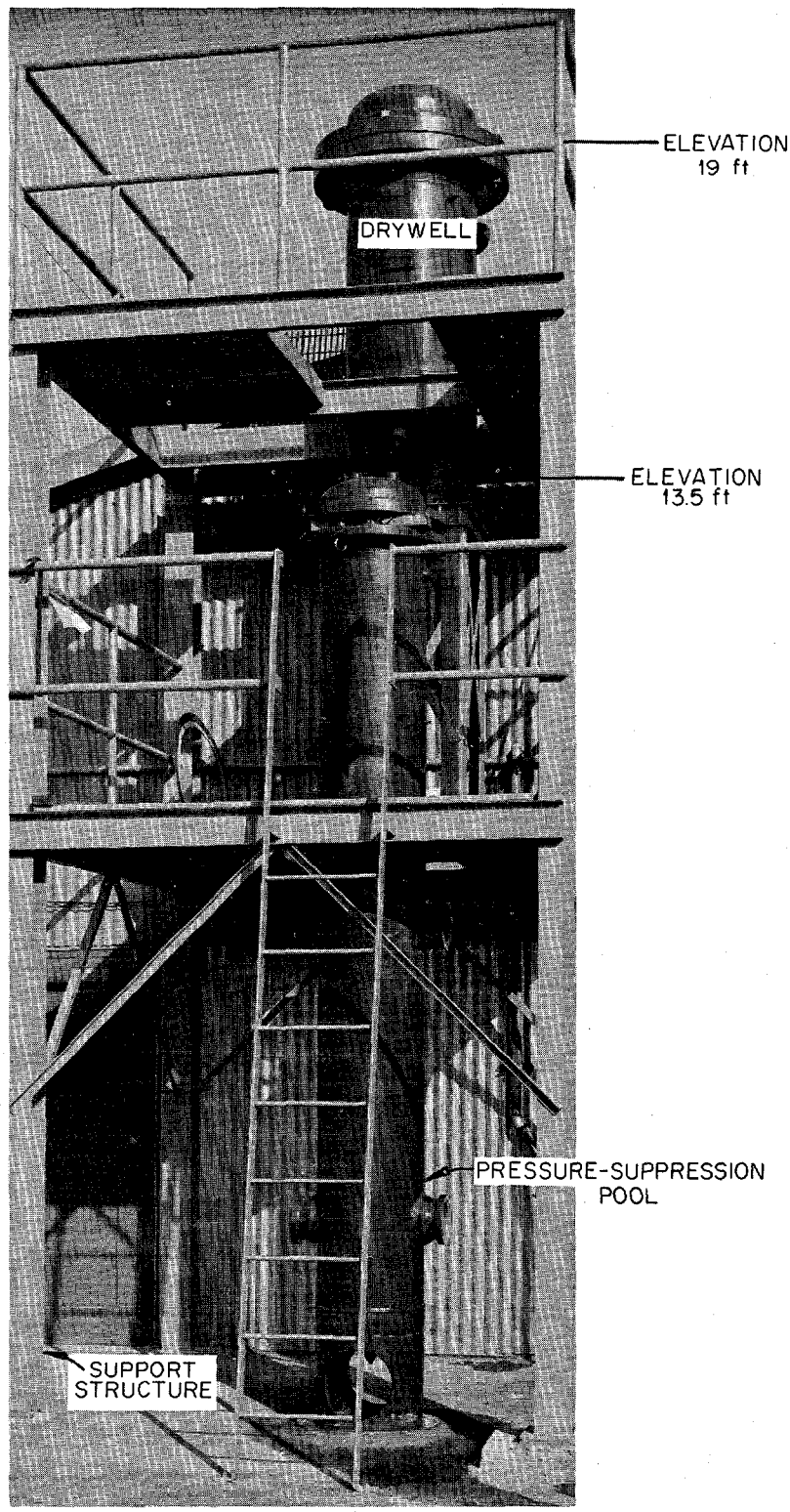

Fig. 3.11. Support Structure and Containment Vessels for Suppression-Pool Tests. (General Electric Company photograph)

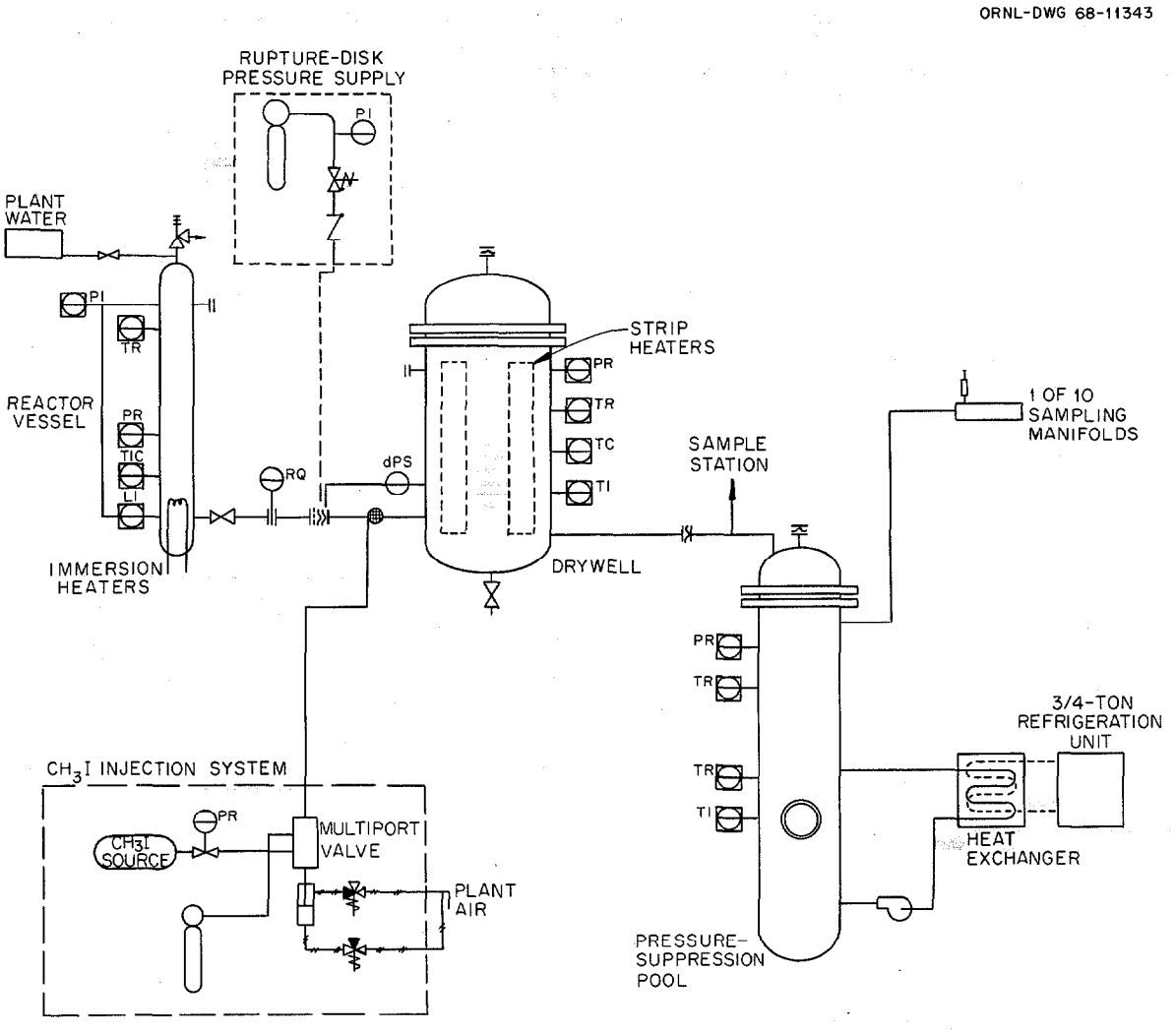

Fig. 3.12. Piping and Instrument Diagram of Equipment for Suppression-Pool Tests.

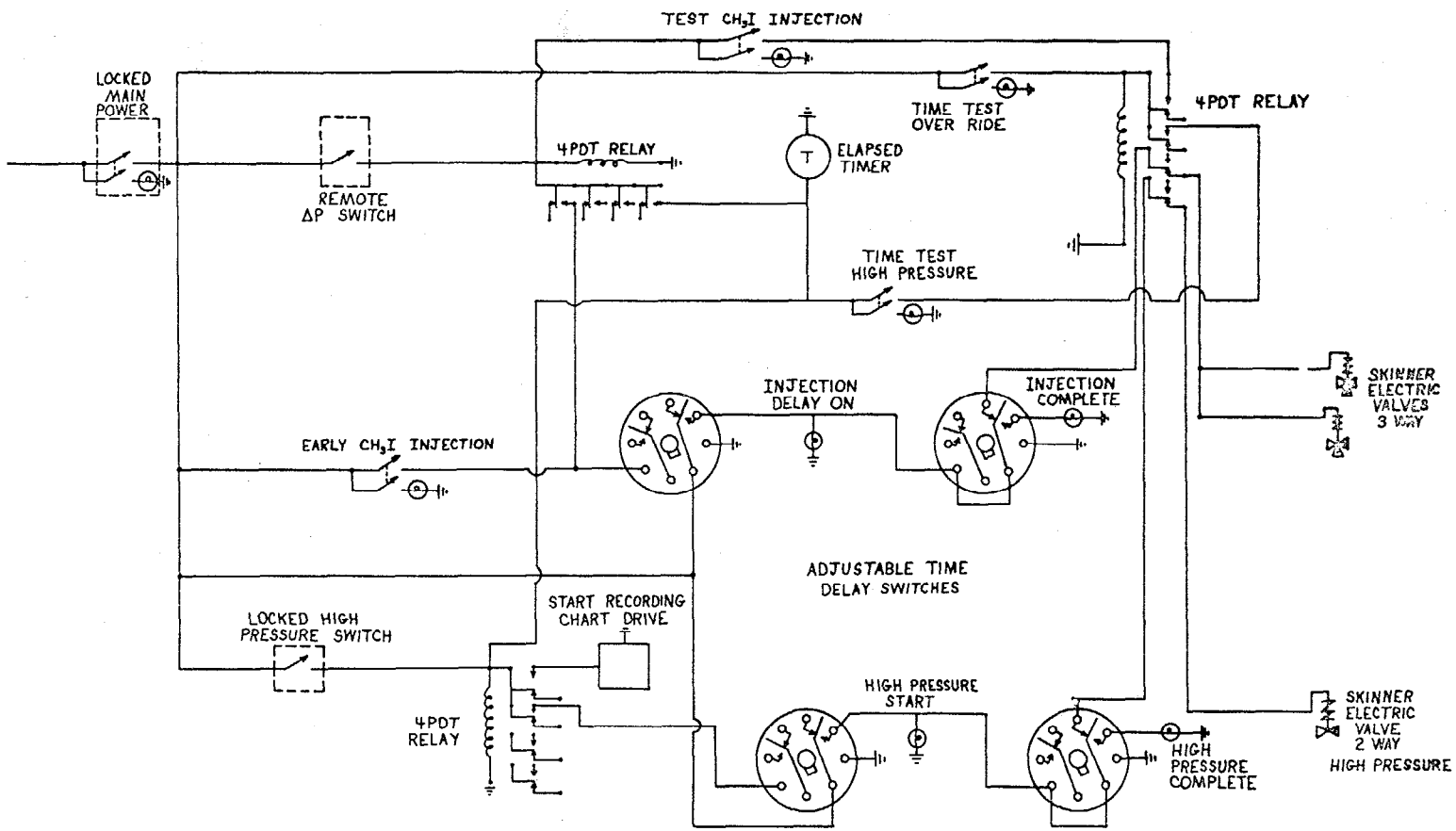

Fig. 3.13. Blowdown Sequence Timing Circuit.

4. SAFETY STUDIES FOR HTGR

4.1 PROGRAM DEVELOPMENT

(AEC Activity 04 60 10 01 1)

H. J. de Nordwall, Coordinator

The completion of a revised version of a draft HTGR Safety Program review by R. H. Bryan, S. I. Kaplan, and H. J. de Nordwall coincided with a large reduction in the funds available for the study of gas-cooled reactor safety. The review attempted to identify areas in which further study is necessary for useful and constructive criticism of licensing proposals to be possible without falling into the trap of achieving nothing more than consumer testing of current components and without subsidizing current reactor design research. This last separation is particularly hard to make for the HTGR, since a knowledge of fission-product distribution at the beginning of any accident is much less amenable to a very simple approximation, such as that all fission products are in the fuel elements, as in reactors with cooler metallic cans. In other words the principal safety problem is being studied as part of the development of the reactor system.

The most important need is for a generally-agreed-upon method of discussing accidents in terms of severity and frequency. Without this framework, planning of a safety program suffers from a certain unreality, not the least because the precision and priority of the data being sought are unknown.

The principal short-term needs relate to the parameters governing strontium distribution. Diffusion and vaporization are covered by other programs,* but the pressing need to improve upon the assumptions made about strontium release during blowdown are not being studied systematically anywhere.

The following medium-term objectives were identified:

1. experimental tests of calculational models of fission-product release from fuel elements,
2. understanding of the mechanisms of iodine and metallic fission-product diffusion,

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*HTGR basic programs at Gulf General Atomic.

3. characterization of gasborne and adsorbed fission-product species in the presence of helium or steam-helium mixtures,
4. a fuller understanding of the thermal and radiolytic reactions between water and HTGR fuel elements so that the performance of future systems containing different graphites in different geometries may be predicted more economically,
5. understanding of the failure mechanisms of coated particles above normal operating temperatures, including a correlation between temperature, operational history, and probability of failure.

Experiments on topic 3 have been discontinued for lack of funds. Only topics 4 and 5 are now funded by the safety program. Subsequent bimonthly reports will cover only these two tasks in the HTGR safety program.

Research during the past two months on steam-graphite reactions through in-pile experiments and supporting experiments out-of-pile continued, albeit sporadically because of budget uncertainties. The objective is to obtain minimum but sufficient information on the dependence of the reaction on graphite internal geometry and reactor environment to permit safety evaluations with confidence for a range of reactor conditions.

4.2 STEAM-CARBON REACTION AND FISSION-PRODUCT RELEASE AND TRANSPORT STUDIES

(AEC Activity 04 60 10 01 1)

A. P. Malinauskas C. M. Blood
J. W. Gooch, Jr.

Reaction Rate Experiments

A detailed survey of studies related to the laboratory-scale investigations of the steam-carbon reaction was continued. As a result of this survey, it was concluded that continuing work should attempt to clarify the geometrical aspects of the problem; that is, more attention should be given to transport of the steam through the graphite. This is of particular importance to the present bonded-bed coated-particle fuel compact design for high-temperature gas-cooled reactors in view of the recent observations that the bonded bed is more susceptible to steam attack at lower temperatures than at higher temperatures.[1]

Accordingly a gaseous-diffusion apparatus was designed to investigate the interdependence of reaction and characteristic internal geometry. Construction of the apparatus is virtually complete, and shakedown tests will be initiated shortly.

Thermal-Gradient Deposition Experiments

Two experiments were performed in our effort to assess the feasibility of utilizing thermal-gradient tubes in studies of the release, characterization, and transport of fission products from graphite surfaces. In each experiment, the deposition characteristics of elemental ^{110}Ag were of interest. Although the initial run yielded data of questionable value, the second run was highly successful. In a helium flow rate of about 1 liter/min, deposition of the silver occurred approximately 20 cm from the ^{110}Ag source at a temperature some 60°K less than the source temperature of 1180°K. (The temperature gradient of 6°K/cm actually started 10 cm from the source, however).

The profile of the deposited silver, expressed in terms of fractional amount deposited per unit length, in arbitrary units, is displayed in

Figure 4.1. The point at which deposition initiated was chosen as the origin. The relatively gradual rise to a maximum is consistent with a diffusion model in which the walls of the tube are regarded as partially reflecting surfaces. A detailed analysis of the data has not been completed, but preliminary calculations give a favorable indication of the use of the thermal-gradient device for obtaining reliable estimates of transport and thermodynamic properties. Further experimental work on this thermal-gradient device must be curtailed, however, because of a lack of financial support.

Reference

1. C. M. Blood, G. M. Hebert, and L. G. Overholser, Oxidation of Bonded Coated-Particle Fuel Compacts by Steam, USAEC Report ORNL-4269, Oak Ridge National Laboratory.

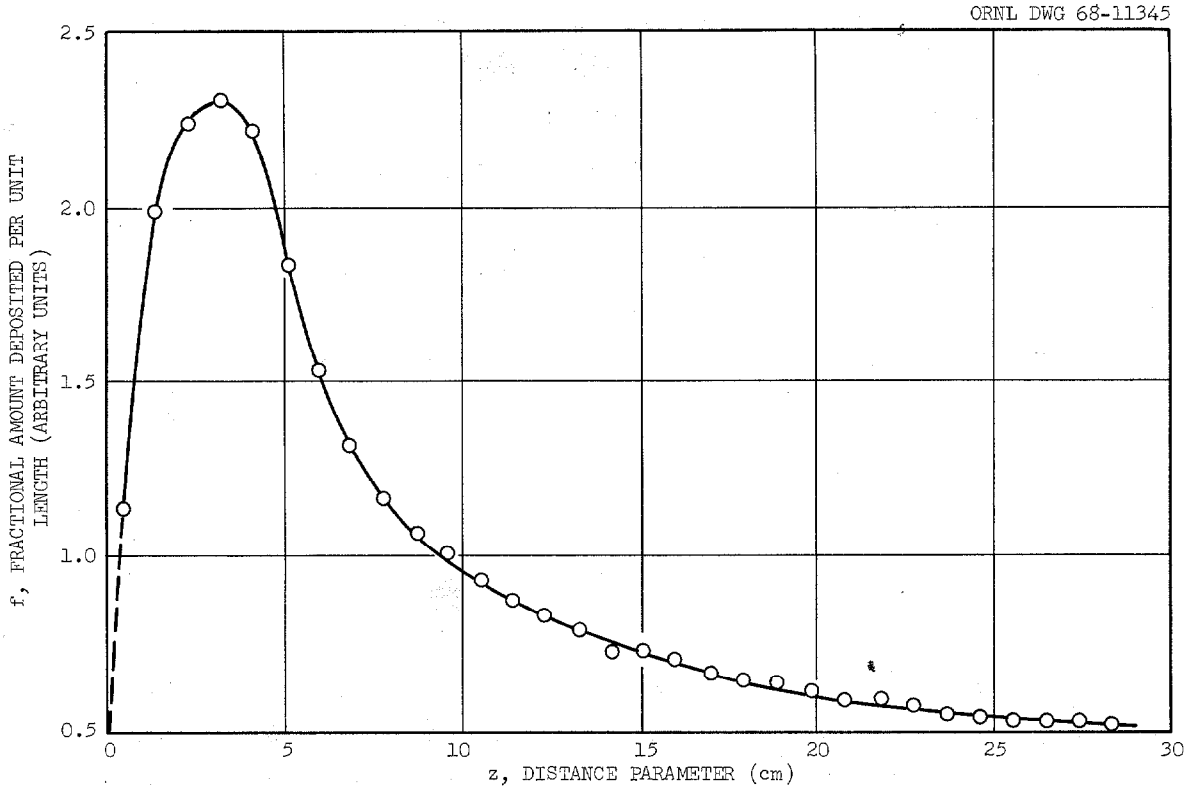

Fig. 4.1. Deposition Profile of ^{110}Ag in a Thermal-Gradient Tube.

4.3 ENGINEERING-SCALE STEAM-GRAPHITE REACTION RATE EXPERIMENT (AEC Activity 04 60 10 01 1)

D. M. Richardson F. H. Neill
M. N. Ozisik*

An engineering-scale facility was being modified for studies of the reaction of high-temperature graphite with pure superheated steam flowing at large Reynolds numbers. The experimental damage profiles of test specimens which could be obtained in the modified facility were intended to provide a basis for extrapolating reaction rate data to actual reactor core geometry and flow conditions and to serve as a test for analytical models. However, in view of funding cuts, this task has been cancelled. An analytical model describing the reaction processes in both the graphite and steam and including the effects of recirculation and removal of the water vapor from the loop by a cleanup system was developed and will be used elsewhere in the program.

Test Facility

The results of an induction heating test on a mockup test stand are shown in Fig. 4.2. The argon purge removed 1700 w and, except for the absence of preheating and the gasification reactions, simulated the steam flow of an actual test. The graphite temperatures were low near the entrance due to argon cooling. The canister temperatures were low near the entrance because the assembly was vertically oriented, with flow directed upward, and there was free convection of room air along the outer wall. Since the water-cooled induction coil is a long solenoid with equally spaced turns, the mid-region of the assembly was heated more strongly, and a range of graphite temperatures was obtained. Thus a number of temperature versus damage-profile measurements can be obtained from a single run in the autoclave.

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*Furman University, Consultant.

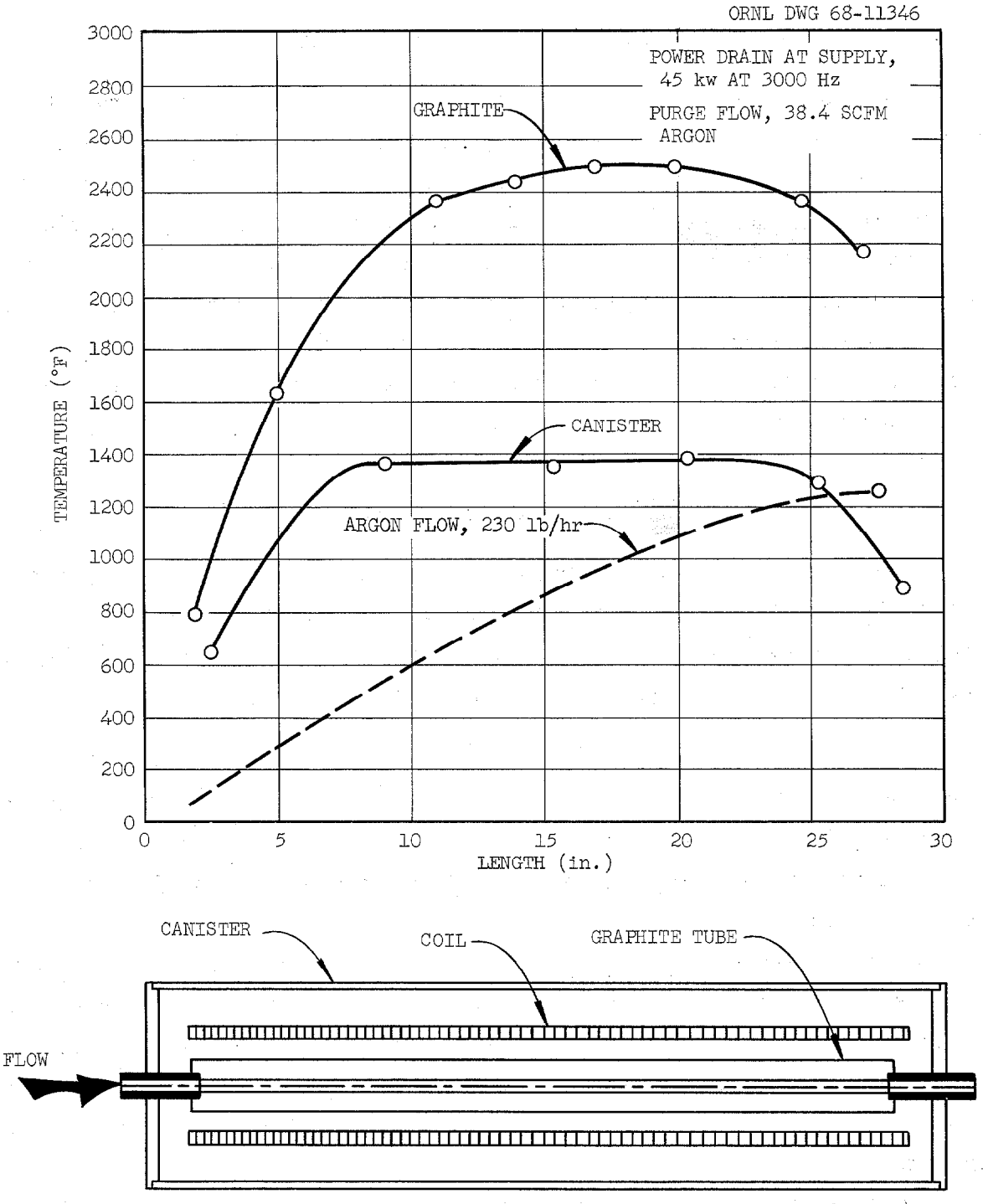

Fig. 4.2. Results of an Induction Heating Test.

Measurement of Density Profile

The central bores of the graphite specimens for autoclave runs have been machined to tolerances of 0.505/0.500 in. on the diameter and better than 0.003 in. for maximum variation in concentricity with the outer cylindrical surface. Following exposure to flowing steam, right sections of 5 to 10 mm thickness can be cut from the numerous temperature positions of the specimen and subjected to examination by low-voltage radiography.[1] If the special care and calibration required by this technique are observed, the resulting x-ray films can be measured to determine accurate and essentially microscopic profiles of density.

A Transient Analysis of Steam-Graphite Reaction

The water vapor that enters the coolant stream of a high-temperature, gas-cooled reactor with unclad fuel elements is transported into the graphite core by means of inpore diffusion. The chemical reaction between the water vapor and the graphite in the form

$$H_2O + C \longrightarrow CO + H_2$$

involves the loss of graphite. Furthermore, there is a reaction between the hydrogen and the graphite. The determination of the weight loss of graphite resulting from such reaction is of interest in the design of gas-cooled reactors.

At temperatures above about 1100°C, the partial pressure of hydrogen diminishes significantly, and hence the reaction between hydrogen and graphite becomes negligible as compared with that between water vapor and graphite. Under such conditions the analysis of the problem of determination of graphite loss can be uncoupled from the effects of reaction with hydrogen.

Consider a cylindrical graphite channel of inner radius r_1, outer radius r_2, and length L through which the coolant gas flows either in turbulent or laminar flow. The flow is assumed to be steady and fully developed. Water vapor is introduced into the gas stream in trace quantities from a source at a rate that may be a function of time. We assume further that the coolant stream is recirculating and that a cleaning

mechanism that is situated in the coolant loop removes part of the water vapor from the stream after the coolant gas passes through the graphite channel.

The governing differential equations for the conservation of water vapor in both graphite and the coolant stream that will include the effects of recirculation, removal of water vapor from the loop with a cleaning mechanism, and the appropriate coupling condition are given in dimensionless form as follows:*

1. for water vapor in graphite,

$$\frac{1}{R}\frac{\partial}{\partial R}\left(R\mathscr{D}\frac{\partial W}{\partial R}\right) - G(W) = \frac{\partial W}{\partial \tau} \qquad \text{with } R_1 \leq R \leq 1,\ \tau > 0\ , \qquad (1a)$$

$$\frac{\partial W}{\partial R} = 0 \qquad \text{at } R = 1,\ \tau > 0\ , \qquad (1b)$$

$$W = 0 \qquad \text{for } \tau = 0\ ; \qquad (1c)$$

2. for water vapor in the coolant stream,

$$\frac{dW_g}{d\tau} + (P_1 + P_2)W_g = P_1\, F(\tau) + P_2\, W\big|_{R_1} \qquad \text{for } \tau > 0\ , \qquad (2a)$$

$$W_g = 0 \qquad \text{for } \tau = 0\ ; \qquad (2b)$$

3. for the coupling condition at $R = R_1$,

$$\left(-\mathscr{D}\frac{\partial W}{\partial R} + H_m W\right)_{R_1} = H_m W_g\ ; \qquad (3)$$

where the dimensionless quantities are defined as

$$\mathscr{D} \equiv \frac{D}{D_0}\ ,$$

*See p. 107 for nomenclature definitions.

$$G(W) \equiv \frac{g(w)\ r_2^2}{D_0\ f(0)} ,$$

$$F(\tau) \equiv \frac{f(\tau)}{f(0)} ,$$

$$H_m \equiv \frac{h_m r_2}{D_0} ,$$

$$P_1 \equiv \frac{2\nu r_2^2}{LD_0} = \frac{Re}{L/r_2}\ \frac{\nu}{D_0} ,$$

$$P_2 \equiv \frac{2h_m r_2^2}{r_1 D_0} ,$$

$$W \equiv \frac{W}{f(0)} ,$$

$$W_g \equiv \frac{W_g}{f(0)} ,$$

$$R \equiv \frac{r}{r_2} ,$$

$$R_1 \equiv \frac{r_1}{r_2} ,$$

$$\tau \equiv \frac{tD_0}{r_2^2} .$$

The reaction term g(w) is represented with the following relation:

$$g(w) = \phi_c M_{H_2O} \mathcal{R}_{H_2O} K_{01} \exp\left(-\frac{E_1}{\mathcal{R}T}\right) TW ,$$

where, when cgs units are used, we have

$$M_{H_2O}\mathcal{R}_{H_2O} = 8.314 \times 10^7 \text{ erg/g-mole}\cdot^\circ\text{K} ,$$

$$\mathcal{R} = 1.986 \text{ cal/g-mole}\cdot^\circ\text{K} .$$

The function $F(\tau)$ is the dimensionless concentration of water vapor in the gas stream at the inlet to the graphite channel. Here, we give a brief discussion of the definition of function $F(\tau)$ to cover the source functions of interest:

1. For uniform concentration of water vapor in the gas stream at the inlet and no recirculation effects,

$$F(\tau) = 1 .$$

2. For a source releasing water vapor into the gas stream at a constant rate for a period of τ_i and then stopping, we assume that the stream is recirculated and that after the stream leaves the channel a cleaning mechanism removes the water vapor from the stream at a rate proportional to the concentration of water vapor in the channel, and we let

τ_i = dimensionless period of injection,

τ_0 = dimensionless time for one complete recirculation,

$\tau_i = \gg \tau_0$.

Then the function $F(\tau)$ should be defined as follows:

$$F(\tau) = 1 \qquad \text{for } \tau < \tau_0 ,$$

$$= 1 + [2W_g(\tau - \tau_0) - F(\tau - \tau_0)]\,(1 - \lambda) \qquad \text{for } \tau_0 \leq \tau < \tau_i ,$$

$$= [2W_g(\tau - \tau_0) - F(\tau - \tau_0)]\,(1 - \lambda) \qquad \text{for } \tau \geq \tau_i ,$$

where λ is the removal coefficient for the water-vapor cleaning mechanism.

The system of Eqs. (1), (2), and (3) has been programmed for solution with a digital computer using finite differences. Provision has been made for the variation of the dimensionless diffusion coefficient as a function of temperature in the form

$$\mathcal{D} = \left(1 + \frac{Q \ln \frac{R}{R_1}}{1 + \frac{Q}{H_h R_1}} \right)^{3/2} ,$$

where

$$H_h = \frac{h_h r_2}{k_c} ,$$

$$Q = \frac{q|_{r=r_2} r_2}{k_c T_g} ,$$

which implies that temperature variation within the graphite tube is according to a prescribed heat flux at the outer surface and convective heat loss to the coolant stream at the inner surface. Subroutines may be added to allow the variation of diffusion coefficient as a prescribed function of the reaction rate in addition to variation with temperature.

From the solution of the above system of equations the following results could be obtained:

$m(r_i,t)$ = fractional rate of loss of graphite at any position r_i at any time t (g/g·sec),

$m(r_i;0\rightarrow t)$ = fractional loss of graphite at any position r_i, from time t = 0 to time t (g/g),

$m(r_i\rightarrow r_2;t)$ = fractional rate of loss of graphite between r_1 and r_2 at any time t (g/g·sec),

$m(r_1\rightarrow r_2;0\rightarrow t)$ = fractional loss of graphite between r_1 and r_2 from time t = 0 to time t (g/g).

Computer studies are now in progress. Presentation of complete analyses and results will be the subject of a later paper. Here we give only the following conclusions on graphite loss based on analysis of preliminary computer calculations.

When the radial thickness of the graphite sample is small, the steady-state conditions are reached very fast, but for actual reactor situations where the graphite thickness is rather large, it will take some time before the steady-state conditions are reached. Therefore in the analysis of graphite loss for reactor accidents of short duration it is important that the reaction rates appropriate for the transient conditions be

included in the analysis, because the steady-state reaction rates may overestimate the graphite loss.

The graphite loss is much larger in the regions near the coolant stream than in regions far away from it. The difference may be of orders of magnitude, depending on the system parameters chosen.

The results of the present analysis are very sensitive to the magnitude of the activation energy. This situation may be advantageously used to correlate the present analysis with steam-graphite reaction experiments to determine the activation energy. The analysis may also be used to estimate graphite loss in reactor systems as a function of time and radial position.

Nomenclature

D_0 = reference diffusion coefficient for graphite, cm^2/sec,

D_g = diffusion coefficient for diffusion of water vapor in coolant gas, cm^2/sec,

E_1 = activation energy, cal/g-mole,

L = tube length, cm,

$f(\tau)$ = concentration of water vapor in the gas stream at the inlet,

$g(w)$ = reaction rate, $g/cm^3 \cdot sec$,

h_h = heat transfer coefficient, $cal/sec \cdot cm^2 \cdot °C$,

h_m = mass transfer coefficient, cm/sec,

k_c = thermal conductivity of graphite, $cal/cm \cdot sec \cdot °C$,

k_g = thermal conductivity of gas, $cal/cm \cdot sec \cdot °C$,

K_{01} = reaction rate constant, $g\text{-mole}/g \cdot sec \cdot atm$,

P_r = Prandtl number,

$q|_{r_2}$ = heat flux at outer radius, $cal/cm^2 \cdot sec$,

r_1 = inner radius, cm,

r_2 = outer radius, cm,

Re = Reynolds number,

T_2 = temperature of the outer surface, °K,

T_g = gas temperature, °K,

ν = kinematic viscosity of gas, cm^2/sec,

t = time, sec.

Reference

1. R. W. McClung, Techniques for Low-Voltage Radiography, Nondestructive Testing, 20(4): 248–253 (July–August 1962).

4.4 HIGH-TEMPERATURE IN-PILE FUEL INTEGRITY TESTS

(AEC Activity 04 60 10 01 1)

J. G. Morgan P. E. Reagan
O. Sisman

Experiment B9-40 in the Oak Ridge Research Reactor B-9 facility is being conducted to determine the rate of breakage for typical HTGR design fuel particles as a function of temperature. The fuel particles, from batch OR-793, are 212-μ-diam uranium carbide spheres with a 43-μ-thick buffer layer, a 22-μ-thick silicon carbide barrier layer, a 69-μ-thick pyrolytic-carbon coating, a thin anisotropic layer, and a thin sacrificial layer. To attain the proper fuel loading, 0.06 g of thorium-uranium carbide particles was added. The particles are cast in a bonded bed.

Capsule B9-40 has been irradiated to a total of 14.9-at % burnup. The irradiation to 13.9-at. % burnup was at a constant temperature of 1350°C. Then the temperature was changed by decreasing the cooling air flow and holding a constant neutron flux. The temperature was increased to 1450, 1550, and 1600°C and held at each temperature for 24 hr. There was a 48-hr period at 1350°C between each increase. The fission-gas release is not yet available, but the overall exit helium activities after 24 hr at constant temperature are listed below:

Temperature (°C)	Exit Helium Activity (mr/hr)
1350	0.9
1450	1.2
1350	1.0
1550	1.5
1350	1.1
1600	2.7
1350	1.7

At 1550 and 1600°C the fission-gas release rate continued to increase with time but returned to constant rates at 1350°C. No bursts of fission gas were released.

4.5 HIGH-TEMPERATURE BEHAVIOR OF GASBORNE FISSION PRODUCTS
(AEC Activity 04 60 10 01 1)

A. P. Malinauskas J. W. Gooch, Jr.
J. D. Redman

An attempt is being made to elucidate the nature of vapor-pressure enhancement of oxides in the presence of water vapor, with particular emphasis on important fission-product oxides. The research commenced with studies on tellurium dioxide, for which the accepted mechanism causing enhancement is believed to be the formation of gaseous $TeO(OH)_2$ in accordance with the reaction

$$TeO_2(s) + H_2O(g) = TeO(OH)_2\ (g)\ . \qquad (1)$$

If this mechanism is correct, the "oxy-hydroxide" species appears to be stable only in the gaseous phase; the reverse reaction readily occurs at lower temperatures.

As a consequence of Eq. (1), the apparent vapor pressure, p*, of TeO_2 should increase linearly with water vapor pressure, p_w, at a given temperature. This has been verified experimentally, as may be seen by the typical data plot shown in Fig. 4.3 for experiments conducted at 925°K. Moreover, an investigation of the temperature dependence of the phenomenon indicates that the heat of reaction corresponding to Eq. (1) is 30.4 ± 0.4 kcal/mole, as compared with a value of 60.6 ± 0.3 kcal/mole for the heat of sublimation of TeO_2.[1]

Although virtually all the data were obtained in an oxygen atmosphere, several experiments conducted in a helium atmosphere yielded identical results. A number of experiments had also been performed with pure tellurium, rather than the dioxide, to ascertain whether enhancement in vapor pressure could be observed. These experiments likewise utilized a helium atmosphere. Water vapor had no effect on the equilibrium vapor pressure.

All the results reported above were obtained with the mass transport method for vapor-pressure determination.[2] To supplement this work, mass spectrometric studies were also made with a Knudsen cell. These investigations, which only commenced a few months prior to this report, were beset with experimental difficulties, particularly when water vapor was

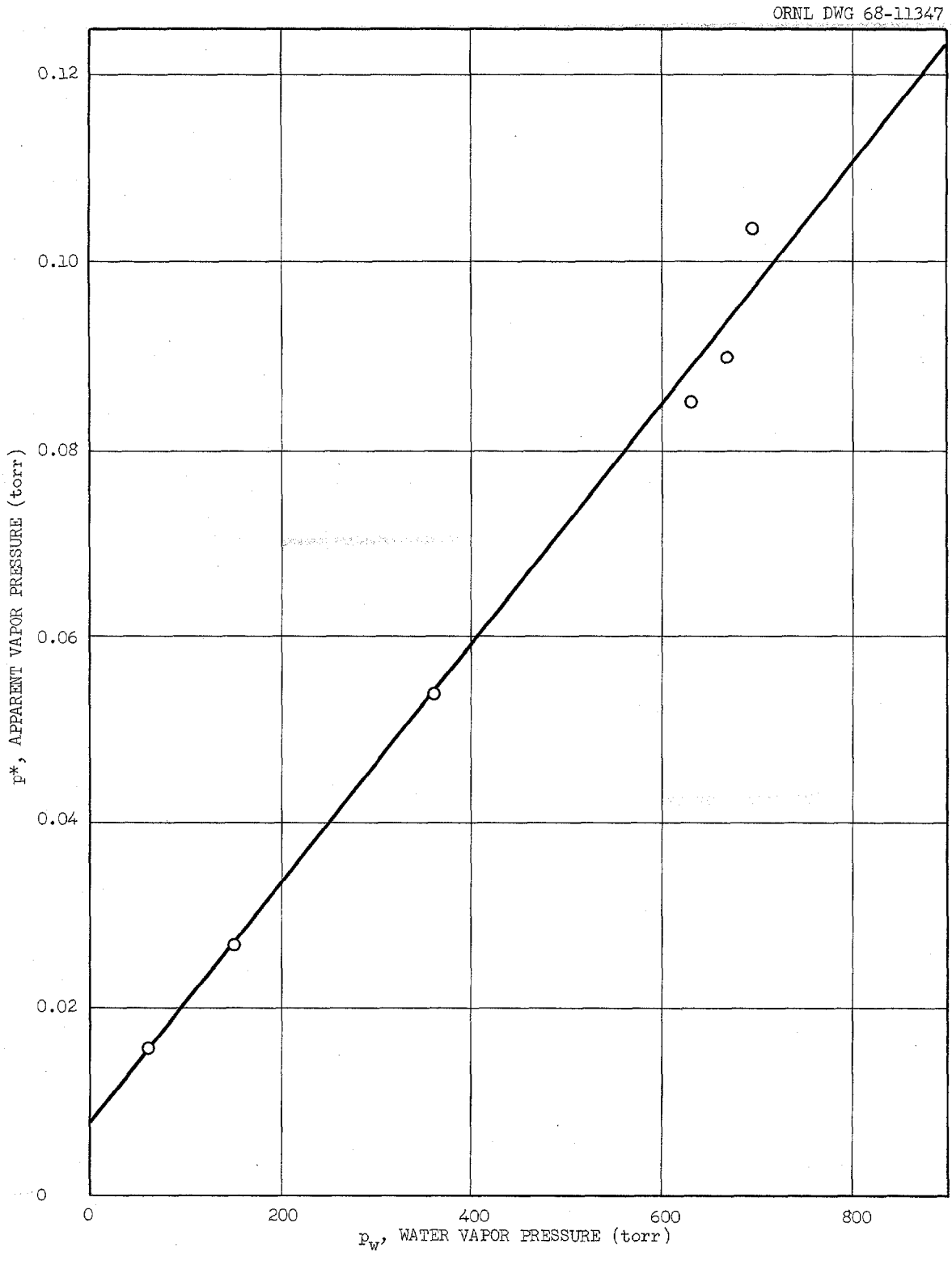

Fig. 4.3. Apparent Vapor Pressure of TeO_2 as a Function of Water Vapor at 925°K.

introduced into the Knudsen cell. Nonetheless, it should have been possible to at least detect ion fragments of the gaseous $TeO(OH)_2$ species; none in fact were observed.

Additional experimental investigations are necessary to clarify unambiguously the mechanism responsible for the observed oxide vapor pressure enhancement. However, the project must be terminated at this stage because of a lack of financial support.

References

1. A. P. Malinauskas, Vapor Pressure of Tellurium Dioxide, USAEC Report ORNL-4300, Oak Ridge National Laboratory, June 6, 1968.

2. A. P. Malinauskas, Thermodynamic and Transport Properties of Fission Products, Nuclear Safety Program Ann. Progr. Rept. Dec. 31, 1967, USAEC Report ORNL-4228, Oak Ridge National Laboratory.

4.6 IN-PILE STUDIES OF REACTIONS OF FUELED GRAPHITE WITH STEAM UNDER ACCIDENT CONDITIONS

(AEC Activity 04 60 10 01 1)

S. H. Freid
H. J. deNordwall
B. F. Roberts
O. Sisman

The release of fission products from HTGR fuel and the consequences of steam leakage into the reactor coolant are being studied on fuel specimens irradiated in the F-9 facility of the ORR. The existing facility is being modified to provide on-stream mass spectrometry of the capsule sweep gas. Uncertainties in the program and funding delayed the installation of this equipment, but this work is again proceeding.

5. HEAVY-SECTION STEEL TECHNOLOGY PROGRAM

5.1 HEAVY-SECTION STEEL TECHNOLOGY PROGRAM (AEC Activity 04 60 80 03 1)

F. J. Witt

The Heavy-Section Steel Technology (HSST) Program is an engineering effort aimed at assessing the degree of structural safety of light-water-cooled reactor pressure vessels that contain flaws, material inhomogeneities, and discontinuities under startup, operating, and accident conditions. Questions of vessel integrity have risen mainly from regulatory and licensing bodies and the Advisory Committee on Reactor Safeguards. These questions will be answered by reevaluating state-of-the-art methods and extending these methods to develop the necessary technologies. First, the information that small standard specimens reveal about thick plate, vessels fabricated from such plate, and actual in-service vessels will be evaluated, with emphasis on transition temperature. Similarly the applicability of fracture mechanics methods for evaluating vessels under in-service conditions will be investigated. Where deficiencies are found, these technologies will be extended to cover the desired temperature and toughness ranges. The program will culminate in a series of simulated service tests aimed at validating the total research effort by comparison with results of tests of pressure vessels up to full size and full thickness. Finally, quantitative engineering methods will be developed that can be described in codes and standards and can be successfully used in average engineering laboratories. The development of the related technologies necessary to make meaningful use of these results and methods will be defined and encouraged.

The HSST program is being carried out with the cooperation of and in coordination with many other research efforts sponsored by the federal government, private institutions, and industry. In particular, industry is sponsoring complementary efforts in nondestructive testing, material properties, and in-service inspection. Thus, with material properties obtained (industry and HSST), a fracture technology established (HSST), and inspection methods available (industry), the safe operating conditions for reactor pressure vessels in water-cooled reactor systems can be established.

The major emphasis thus far in the HSST program has been on material procurement, documentation, and investigations; definition, negotiation, and inception of research contracts; and completion of general long-range program plans. There are ten tasks of the HSST program in addition to the general administration task. The specific activities are discussed in the context of these tasks.

Task A. Program Administration

The material procurement phase of the HSST program is near completion. Negotiations were completed for fabrication by Lukens Steel Company of a 10-in.-thick 40-ton ASTM A 543 plate. Both class 1 and class 2 materials will be obtained. Bettis Atomic Power Laboratory has provided the HSST program with a 36 in. by 72 in. by 8-in.-thick ASTM A 302, grade B (essentially A 533) plate and a 28 in. by 28 in. by 30-in. SA 326 forging in the as-forged condition. Additional materials are currently expected from both Bettis and Knolls Atomic Power Laboratories. Another 10 ft of submerged-arc weldment is being obtained.

Contract negotiations are currently under way for investigating the gross strain concept of fracture behavior, strain rate, and crack-arrest effects and for three-dimensional elastic-plastic analyses of flawed structures. The analyses will initially emphasize the solution for a part-through crack in a finite plate. In addition a contract is being negotiated for a feasibility study on testing the large tensile specimens of the simulated service test effort. Discussions are being held on work related to fracture toughness testing and fatigue. Procedures are being set up for providing HSST materials to system operators for use as surveillance specimens.

Task B. Material Inspection and Control

A topical report documenting the fabrication history of HSST plates 01 and 02 was written and is currently being edited. The year-end inventory of program materials showed that all materials in storage were properly identified and accounted for.

The ultrasonic testing phase of the investigation of a flaw indication in plate 01 is complete, and the data are being evaluated. Preliminary slices through the indication were made, and visible evidence of the flaw was found to exist.

Task C. Material Characterization

The main emphasis in Task C is currently on plates 02 and 03. Plate 02 was heat treated (edge quenched) by Combustion Engineering, Inc., like plate 01, while plate 03 was flat quenched by Lukens Steel Company. The homogeneity of the interior portion of plate 03 is an important factor in determining the availability of the plate material for research programs.

Westinghouse Electric Corporation characterized, to a degree, the plate 02 material.[1] Specimens are being obtained in such a manner as to complement the Westinghouse results. Eight full sets of Charpy V-notch impact specimens from plate 03 were machined and are currently being tested. Metallurgical and chemical analyses are being performed on material from plate 01.

Task D. Variability in Plates, Heat-Affected Zones, and Weld

The results from Task C are a part of this activity. A four-channel thermal-cycling simulator is on order and will be used to simulate the various heat treatment conditions to which reactor vessel materials could be subjected.

Submerged-arc and manual-metal-arc weldments were obtained. A program to investigate the weld metal and heat-affected zone was outlined and shop drawings for machining specimens are being made.

Task E. Transition Temperature Investigations

Brittle welds are being placed on the specimens for the size-effect studies of the drop-weight tests being performed by Martin-Marietta Corporation. The load capacity of the dynamic tear machine at Naval Research Laboratory is being increased to 500,000 ft-lb so that additional

12-in.-thick dynamic tear specimens may be tested at elevated temperatures. In the interim, the Naval Research Laboratory has tested three or four more 6-in. specimens of A 533, grade B, class 2 material. The results indicate very little if any size effect and rapid transition behavior based on energy per unit area. Hopefully the four remaining 12-in.-thick specimens will be tested during the next three or four months.

Task F. Fracture Mechanics Investigations

Valid fracture toughness values were obtained by Westinghouse Electric Corporation under the Euratom program up to −50°F at quarter thickness with a 2T specimen.[1] The values range from 37,500 psi·in.$^{1/2}$ at −200°F to 65,000 psi·in.$^{1/2}$ at −50°F. A preliminary comparison of variation through the thickness at −100°F is

	Fracture Toughness (psi·in.$^{1/2}$)
Top surface	70,000–80,000
Top quarter surface	55,000–60,000
Center	50,000
Bottom quarter surface	51,000
Bottom surface	60,000–80,000

Under the cooperative effort, Westinghouse has tested up to 6T specimens. The data are preliminary and will be reported later.

The elastic-plastic plane strain analysis is in progress at Westinghouse. Contracts are currently being negotiated for research efforts in the gross strain concept, strain rate, and crack-arrest toughness investigations and the three-dimensional elastic-plastic analyses. It is expected that these new efforts will be active by October 1, 1968.

Task G. Fatigue and Crack Propagation

A tentative program was outlined in this area and proposals are being solicited. Portions of this effort are expected to be carried out in conjunction with Task J, Periodic Proof Testing and Warm Prestressing.

Task H. Irradiation Effects

A phase I activity was outlined. The irradiation conditions are 1×10^{19} neutrons/cm^2 at 450 and 550°F and 5×10^{19} neutrons/cm^2 at 550°F. Fracture mechanics, dynamic tear, Charpy V-notch impact, and tensile specimens will be investigated. The work will be a coordinated effort of Oak Ridge National Laboratory, Pacific Northwest Laboratory, and Naval Research Laboratory. Westinghouse Electric Corporation will be a participant under their Euratom Program. A phase II activity is presently under discussion.

Task I. Complex Stress State

The details of Task I are presently being outlined. An initial phase will be an investigation of fracture toughness as related to the complex stress states near discontinuities. Epoxy models will be examined first.

Since many epoxy model specimens will be examined, an inexpensive procedure for fabricating the vessels must be developed. Efforts to mold models have been successful. A molded nozzle, head, and cylinder region may be seen in Fig. 5.1, along with the molds used.

The character of the epoxy material for the desired use is currently being optimized.

Task J. Periodic Proof Testing and Warm Prestressing

A final report and recommendations from General Electric Company are due by September 30. Investigations will be initiated this fiscal year.

Task K. Simulated Service Test

Design and stress analyses continued on the intermediate vessels and are on schedule. A feasibility study on the laboratory test phase of the program was awarded to Southwest Research Institute.

The phase I irradiation effects activity is being vigorously pursued, and a phase II activity is being formulated.

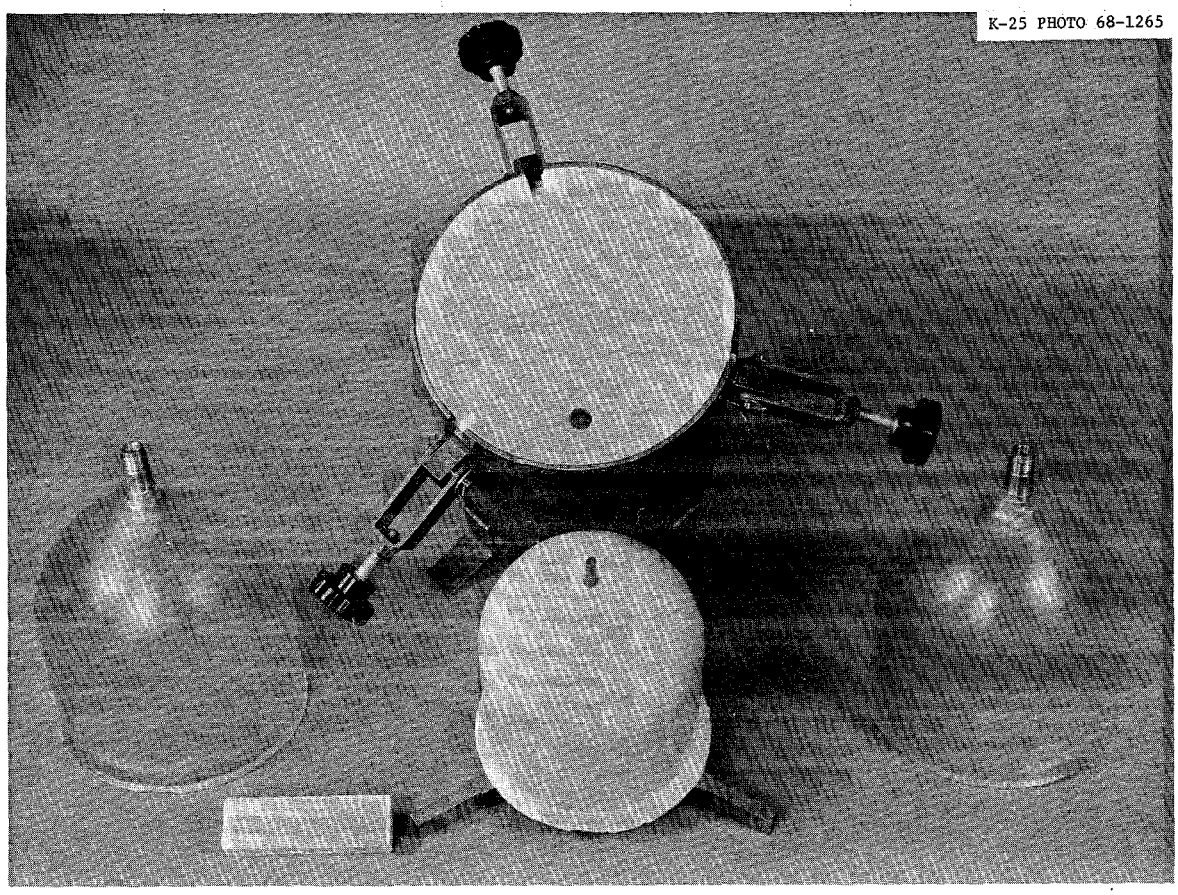

Fig. 5.1. Molds and Molded Epoxy Vessels Being Evaluated for Use in Complex Stress Activity.

Reference

1. L. F. Cochrun, Fracture Mechanics Evaluation of Reactor Vessel Steels, Quarterly Progress Report for the Period Ending June 30, 1968, WCAP-3677-8, Westinghouse Electric Corporation, July 1968.

5.2 EXPERIMENTAL AND ANALYTICAL INVESTIGATIONS OF NOZZLES (AEC Activity 04 60 70 01 1)

B. L. Greenstreet R. C. Gwaltney

The basic purpose of the experimental and analytical investigations of nozzles is to establish procedures that have been thoroughly substantiated by experimental data for analyzing designs for nuclear pressure vessel penetrations and nozzle connections. The work is directed specifically toward the stress and strain analysis of those areas in a pressure vessel where high stress concentrations occur – the intersection of a nozzle and the main pressure vessel – when the vessel is loaded with an internal pressure or moment and thrust forces are applied through the nozzle. The investigations cover nozzles radially and nonradially attached to a spherical shell, nozzles attached in clusters to spherical shells, and nozzles attached to cylindrical shells. They are carried out in close cooperation with subcommittees of the Pressure Vessel Research Committee (PVRC) of the Welding Research Council. Problem areas being studied are (1) the effect of superposition of various loadings on the nozzle, including internal pressure, bending, twisting, and direct thrust, (2) the mutual interaction between adjacent openings or nozzles, and (3) nonradial (or hillside) nozzles in order to establish their potential limitations and the general magnitude of the stresses.

The results obtained from this program to date have been published[1] and are being used to establish rules for both industry and AEC-RDT Standards and to provide analyses and design methods commensurate with meeting reliability and safety requirements. The results are made available as computer programs and design charts and tables based on parametric studies and/or empirical correlations.

For reporting purposes, the program has been divided into four activities. The following are brief accounts of the progress on these activities.

Project Management

As a member of the Pressure Vessel Research Committee, ORNL has been assigned the task of coordination, review, and evaluation of the research program for the PVRC Subcommittee on Reinforced Openings and External Loadings. This management function includes (1) direction of AEC-sponsored work, including that done at ORNL or under subcontract, (2) generation of parameter studies in support of various evaluation and correlation studies being done at Battelle Memorial Institute, (3) reporting to the PVRC subcommittee on non-AEC-sponsored projects and making recommendations concerning those projects, and (4) soliciting comments and recommendations from the PVRC subcommittee concerning AEC-sponsored work.

AEC-sponsored work includes strain-gage studies on steel models at the University of Tennessee and at Auburn University, in addition to analytical studies on intersecting shells at Auburn University and at ORNL.

Single Nozzles in Spheres

Single nozzles radially and nonradially attached to a spherical shell are being experimentally analyzed at the University of Tennessee. One of the two models currently being examined has a radially attached 7 7/8-in.-OD nozzle and the other has two 2 5/8-in.-OD nozzles attached at angles of 22 1/2 and 45 deg from the radius. During the first series of tests on the 22 1/2-deg nonradial nozzle model, a number of strain gages in critical locations were lost. These defective gages were replaced, and the model is now ready for retesting. A second series of tests is being run on the model with the 7 7/8-in.-OD radial nozzle. For these tests the nozzle was bored out so that the wall thickness was reduced from 0.375 to 0.1875 in.

The analytical problem of a single nonradial cylindrical nozzle in a spherical shell is being studied at Auburn University. A solution based on shallow-shell theory was programmed for computer evaluation. Results from the first computer run are presently being studied.

Single Radial Nozzles in Cylindrical Shells

Analytical studies and computer programming for the problem of a single, radial, cylindrical nozzle intersecting a cylindrical shell are being conducted at ORNL. Four loading conditions are being studied: (1) internal pressure loading, (2) an in-plane bending moment applied to the nozzle, (3) an out-of-plane bending moment applied to the nozzle, and (4) axial thrust on the nozzle. The internal pressure loading case was successfully programmed for computer studies.

Work is presently under way on the computer program for the out-of-plane bending case. In order to adapt the mathematical solution of the problem to machine computation, a considerable amount of analytical work was necessary. This work is being done with the assistance of University of Tennessee personnel. Analytical work on the in-plane bending case is being done in preparation for writing the computer programs.

Clusters of Nozzles in Flat Plates

A series of experimental studies of flat plates loaded in biaxial tension with unreinforced circular holes and clusters of nozzles is being conducted at the University of Tennessee. Two different loading frames have been designed. One of these is capable of loading the nozzles with an axial thrust and external moment, whereas the other is only capable of loading the plates in uniaxial or biaxial tension. Figure 5.2 shows an unperforated plate in the biaxial tension loading frame. At present an extensively strain-gaged flat plate with two unreinforced holes is being tested.

Clusters of Nozzles in Spherical Shells

Experimental stress analyses of clusters of nozzles intersecting a hemispherical shell are being conducted at Auburn University. Two steel models extensively instrumented with strain gages are being used. One of the models has two large radial nozzles and the other is to have five separate clusters of small nozzles. Additional strain gages were added to the first model and internal pressure tests are being run. A loading

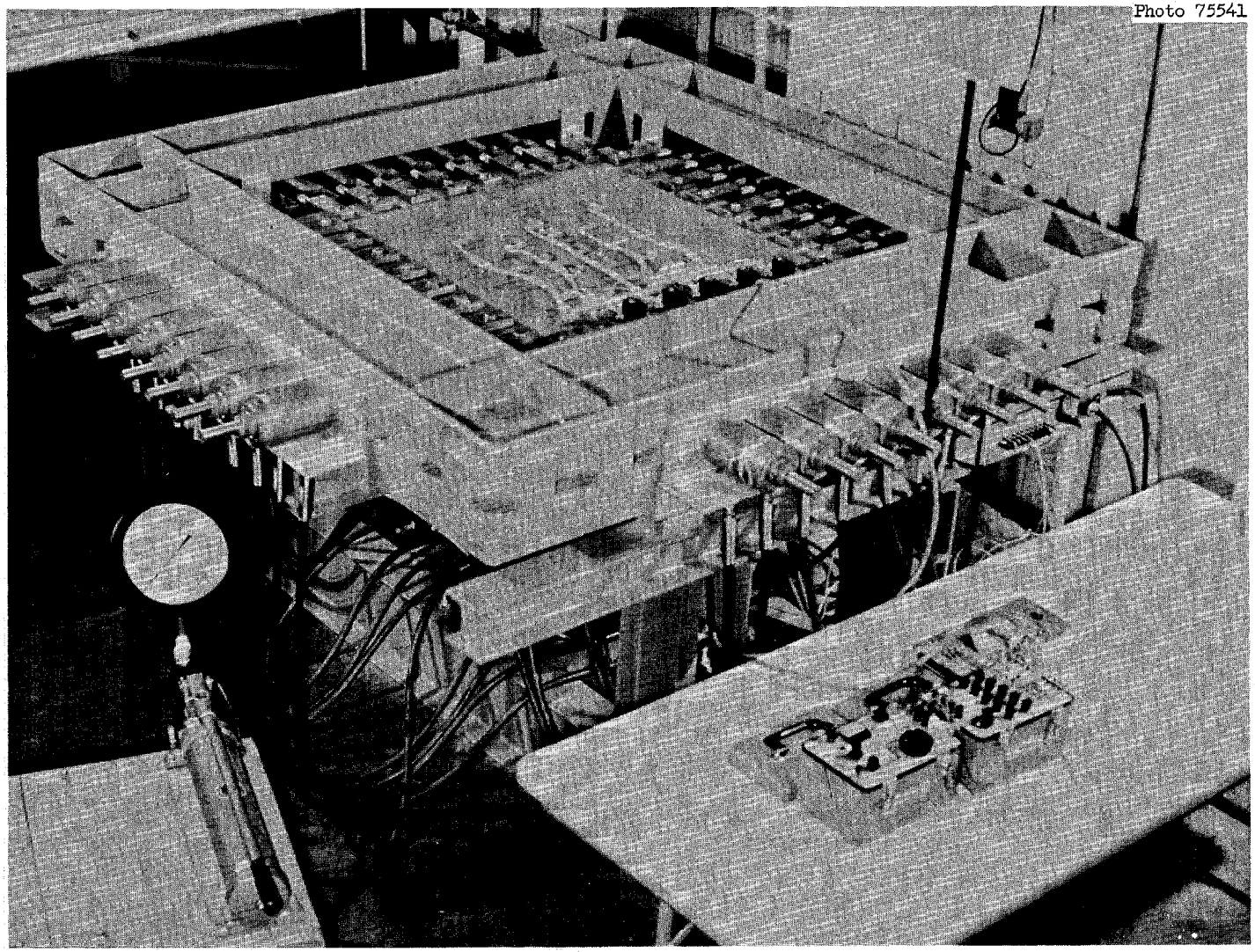

Fig. 5.2. Unperforated Plate Being Tested in the Biaxial Tension Loading Frame at the University of Tennessee.

frame for applying external loads to the nozzles is being designed in preparation for conducting the next series of tests.

Reference

1. E. C. Rodabaugh et al., Evaluation of Experimental and Theoretical Data on Radial Nozzles in Pressure Vessels, USAEC Report TID-24342.

5.3 DESIGN CRITERIA FOR PIPING, PUMPS, AND VALVES (AEC Activity 04 60 80 03 1)

B. L. Greenstreet S. E. Moore

Definitive information and data are being developed for assuring adequate and safe design of piping systems for nuclear service. The ORNL work, which is primarily concerned with piping components is an essential part of a larger AEC-Industry cooperative program for developing piping, pump, and valve design criteria. Through this program, factors urgently needed for use in both industry and AEC-RDT codes and standards will be developed that delineate design practices commensurate with meeting reliability and safety requirements.

Primary tasks to be carried out are accurate and thorough experimental and analytical stress analyses of pipe fittings. The experimental work is confined almost entirely to tees, while analytical studies will be made for all standard pipe fittings. The tee investigations are needed and, at the same time, are so complex as to tax existing technology to the limit. From the analyses and companion studies, data correlations and evaluations will be made; design charts, graphs, and tables will be prepared; and code rules will be drafted for consideration by the various code bodies. Data will be presented in terms of stress indices and flexibility factors for direct use. Also, overall interpretive reports will be written on work sponsored under this program plus that done by others.

For reporting purposes, four activities have been identified. The following are brief accounts of the progress on these activities.

Literature Survey and Review

A literature review of field failures in power plant piping components and of the design of reducers, expansion joints, and other mechanical connections, including patented joints, was completed. Four chapters of the overall survey report being prepared by Battelle Memorial Institute and Southwest Research Institute were drafted. Several chapters remain to be drafted; however, in order to expedite comment and review of the

report, copies of the completed material are to be sent to the AEC and to members of the PVRC subcommittee. This material is presently being prepared for distribution.

Dimensional Study of Standard Pipe Fittings

Dimensional measurements were made on three 12-in. Sched. 80 carbon steel B16.9 straight tees obtained for experimental stress analysis. Each of these tees was obtained from a different manufacturer and is typical of their product design. The most obvious differences occur in the design of the crotch, with the radius of the intersection and the amount of reinforcing material, i.e., wall thickness, being the primary variables. Less obvious differences occur in the main body of the fitting due to differences in the forming technique and size of pipe used as the starting material.

Three 12 × 12 × 6-in. stainless steel reducing tees, also scheduled for extensive strain gaging and test, were received from the manufacturers. Dimensional measurements of these tees will begin shortly.

Dimensional data obtained from 4-in. concentric and eccentric reducers are presently being analyzed. An attempt will be made to define variations in shape in a manner similar to that reported for elbows in the last progress report.

Structural Analysis of ASA B16.9 Tees

The thin-shell cylinder-to-cylinder steel model ($d/D = 0.5$, $d/t = D/T = 100$) was subjected to a preliminary internal pressure test (50 psi). In this first test, there were 22 three-gage delta rosettes placed on the outer surface of the model. Gages were located on both the run and the branch along two lines, one in the principal plane of the tee and the other in a plane perpendicular to the run of the tee. This preliminary test was run in order to obtain some data which could be compared with the finite-element analysis performed by Professor Clough at the University of California. The model is presently being strain gaged with a complete set of three-gage rosettes. Approximately 330 rosettes will be

used to instrument both the inner and outer surfaces. The model should be ready for testing early in 1969.

Professor Clough completed the finite-element analysis of the thin-shell model discussed above for the 13 loadings described in Task 1 of the PVRC program outline. Two different size elements were used in the analysis; they corresponded to a large grid and a smaller grid about half the size of the large one. Preliminary comparisons of the analysis and the pressure test data indicate that the larger grid is unsatisfactory. The smaller grid analyses gave results that compare favorably with the test data, except immediately at the junction. As a consequence, more gages will be placed near the junction for the complete model test. In addition, a finer grid may be needed for the analysis in the region of the junction. Stress plots and a more complete comparison of the finite-element analysis with the test data are presently being made.

Structural Analysis of Other Piping Components

A report is presently being written on analytically derived stress indices and flexibility factors for elbows and curved pipe. The elbow is assumed to be a perfect toroidal segment and unconstrained at the ends, that is, end effects are neglected. Analytical studies to determine the influence of straight pipe welded to the elbow and the influence of minor geometric deviations on the stress intensities are being started.

6. RDT STANDARDS PROGRAM

6.1 PROGRAM OBJECTIVES AND GENERAL ACTIVITIES
(AEC Activity 04 60 80 03 1)

W. F. Ferguson R. A. Schmidt

The initial objectives of the RDT Standards Program are to develop engineering standards covering design, analytical methods, materials, fabrication, construction, quality assurance, inspection, testing, operation, and maintenance of equipment and systems for water-cooled reactors. The AEC Division of Reactor Development and Technology requested the Oak Ridge National Laboratory to assume the responsibility to organize, develop, prepare, distribute, and revise these standards to reflect current developments, information, techniques, and experience in cooperation with other AEC offices and contractors. The Standards Program provides a source of sound engineering information for designers, builders, and users of reactor systems and components that will result in improved quality assurance and an attendant improvement in reactor safety and reliability. Additionally, this program is closely coordinated with that of the Liquid Metals Engineering Center where individual standards can fulfill the requirements of both water-cooled and liquid-metal-cooled reactor technology.

Current objectives include (1) continuation of the preparation, modification, and review of existing standards, specifications, and supplements; preparation of preliminary drafts; and the issuance of tentative standards approved by RDT; (2) development of program plans, schedules, and long-term goals; (3) consolidation of plans for a complete numbering, format, and titling system embodying content, referencing, retrieving information, and administrative control; (4) establishment of uniform definitions and nomenclature; (5) continuation of information exchange with others in reactor technology; (6) consultation with reactor builders and operators and participation in conferences and meetings, including visits to other sites and agencies as required in the Standards Program; (7) cooperation with the liquid-metals standards program; (8) continuation of review and direction of engineering consultant activities and subcontract participation; (9) participation with Idaho Nuclear Corporation in the preparation of quality assurance standards; and (10) continuation of the study of reactor design, testing, operating, and maintenance reports.

An informal conference on the RDT Standards Program is under consideration in order to better acquaint industry representatives, architect-engineering groups, utilities, professional societies, and other potential users of standards with program plans. The proposed conference would be held at ORNL or at some western location, depending on the response received from questionnaires being mailed to the above groups. The timing of such a meeting is uncertain and is subject to further discussion with AEC-RDT but will not be prior to calendar-year 1969.

Procedure for Developing Standards

A preliminary list of standards and brief scopes were developed and circulated to indicate some of the standards needed. Approximately 175 standards were included in the list. Recommendations from RDT, input information from other sources, manpower requirements, and further work on planning and scheduling at ORNL should result in significant extension and modification of the list which will, because of the nature of the nuclear business, be constantly changing. As standards are developed and feedback is received from users, the Standards Program will be redirected accordingly. Priorities and schedules for completion of specific standards will be affected by limited sources of sound engineering information, availability of qualified specialists to prepare standards, interfaces, the existence of special programs sponsored by RDT, and the need for additional reference standards.

Testing

Comments on the Preoperational Testing Standard outline are being collated for transmittal to Burns & Roe. Presently, Burns & Roe is preparing a brief description for each item of the outline to enable reviewers to have a better insight to what is intended. An investigation is under way to determine what effect our funding limitations will have on the overall scope of the work and the schedule.

Maintenance

Arrangements were made for Douglas United Nuclear (DUN) personnel to visit ORNL on August 27. In this meeting, discussions were directed toward the RDT Standards Program and DUN programs. A meeting was held with ORNL Instrumentation and Controls Division personnel to discuss instrument standards and their involvement in the maintenance portion of the RDT Standards Program.

Considerable effort was directed toward identifying individual maintenance standards by title and scope that are expected to be prepared during FY-69 and later. It appears that more than 20 documents is a realistic estimate of the number of such standards needed, including instruments and controls standards.

The merits of organizing a voluntary group oriented toward maintenance in governmental and privately owned reactors are being studied. Such a group should be capable of furnishing considerable input to the maintenance portion of the Standards Program.

Miscellaneous

The preliminary partial draft of the ASME's proposed Pump and Valve Code was reviewed and comments sent to AEC-RDT. Our conclusion was that the objectives of the Code were worthy of continuing AEC support and would fill an important need for standards to establish the structural integrity of pump casings and valve bodies. The design section of this Code is still under development and was not reviewed. The operability requirements of these components are not covered in the proposed Code but will be included in the broader RDT standards.

Work continued on consolidating plans for a complete system embodying title, format, numbering, content, referencing, retrieving information, and administrative control for RDT standards. The resulting document will be a "standard" for preparing RDT standards.

6.2 REVISION OF EXISTING STANDARDS
(AEC Activity 04 60 80 03 1)

R. A. Schmidt

Revision and updating existing standards is proceeding at a steady pace, and several standards under development are nearing completion. RDT Standard for Welding, S-902, was revised; a tentative draft for RDT Standard for Nuclear Fuel Rod Fabrication, S-900, was submitted to RDT; and the following tentative standards were distributed for review, comment, and use:

S-923 RDT Standard for Material and Fabrication Requirements for Ni-Cr-Fe Alloy Seal Membranes

S-924 RDT Standard for Inspection Requirements and Acceptance Standards for Seal Membranes

To date, 77 drafts have been submitted to RDT; 61 of those have been approved by RDT as tentative, and 52 have been issued as tentative standards. Approximately 250 copies of each tentative standard were distributed to 66 addressees, including AEC installations, architect-engineers, consultants, nuclear component manufacturers, and electric utilities. The specialists continued to modify, prepare, and review existing standards, specifications, and supplements; participated in conferences and meetings; consulted with reactor builders and operators; reviewed and directed consultant activities and subcontractors; visited other sites and agencies; reviewed and studied reports and documents relative to the design, construction, operation, testing, and maintenance of water-cooled reactors; and participated in informal sessions in-house.

Considerable effort was directed to the preparation of a comprehensive listing of standards proposed for development and issuance in FY-69 and later. Numerous considerations influenced the selection of subjects proposed in the standards list and the accompanying scopes. It is anticipated that the number of proposed standards will continue to increase as the need for additional standards is realized during the process of developing and preparing standards, as well as in the engineering research, investigation, and study periods. Subsequently, a more complete list will be issued, and a schedule for the FY-69 program will be defined.

Effort has begun in the development of a standards index system, which will be issued periodically to standards' users. The index is being designed to provide the standards' users with a listing of current or latest standards issued or revised. Information provided by the index will include standards title, number, issue date, revision number or date, issuing office, applicability, and status, such as tentative, revised, final, etc. The index will establish a system for assuring the user the latest standard issue.

Meetings were held July 23–26, 1968, with Oremet Corporation, Portland, Oregon, and Wah Chang Corporation, Portland, Oregon, to discuss tentative drafts of Zirconium Alloy Standards, RDT-MB-351, MB-352, MB-353, and MB-356. Representatives of both manufacturers were extremely cooperative, and as a result of the discussion, it was agreed that a standard on Zirconium Alloy Ingots should be written. The ingot standard will be based on ASTM B350-64T.

Progress was made in the area of consolidation of RDT-S-908 and RDT-S-934. All requirements of the two documents are covered by reference and modification of ASME Section III, Appendix IX, except ultrasonic examination of product forms. To cover product forms, tentative drafts of the following documents were generated:

RDT-M-A388, Ultrasonic Examination of Heavy Steel Forgings

RDT-M-A577, Ultrasonic, Shear-Wave Examination of Plates

RDT-M-A578, Longitudinal Wave Ultrasonic Examination of Plain and Clad Steel Plates

RDT-M-E213, Ultrasonic Examination of Metal Pipe and Tubing for Longitudinal Discontinuities

On July 30, 1968, representatives of ORNL and Idaho Nuclear attended a meeting at ASTM Headquarters, Philadelphia, Pennsylvania, to discuss the use of valve castings presently located at Idaho. The data obtained will be used in updating ASTM E71 to provide a more realistic standard for castings up to 2 in. Renewed interest has been shown in updating E71, since some branches of the Navy have stopped using it and are attempting to use E186 with appropriate modifications. Several groups presented radiographs of castings at the July meeting and also at a previous meeting held at Watertown, Massachusetts, in May. As a result, the

number of castings needed from the Idaho group will be limited. The Idaho radiographs indicate that considerable grinding has been done on the castings in preparation for repair; therefore, where a radiograph from another source indicated a comparable quality level the Idaho castings were classed as alternates. Idaho Nuclear has contributed substantially to the ASTM subgroup through technical assistance, as well as in supplying radiographs of castings.

6.3 STANDARDS FOR COMPONENTS OF WATER-COOLED REACTOR PRIMARY SYSTEM

(AEC Activity 04 60 80 03 1)

W. A. Bush R. W. Dehoney
D. L. Gray

Reactor Internals

The ATR Design Manual was received from Idaho Nuclear. Also B&W submitted a report entitled "ATR Internals Vibration Summary," which is a comprehensive discussion of the vibration problems encountered, the analysis and test results, and the design modifications made to alleviate the vibration problems. Both documents are being reviewed for material to be included in the RDT Standards for Reactor Internals Design.

A revised draft of the RDT Standard for Reactor Internals Design was received from United Nuclear Corporation and copies were distributed to ORNL personnel and consultants for review. To date, comments have been received from one reviewer, and discrepancies have been discussed with another reviewer. It is apparent that the draft contains several errors, both in the technical and narrative areas. Upon receipt of other reviewers' comments, discussions will be held with UNC and proper corrections made.

Pumps

The acceptance of the proposal received from Franklin Institute for the preparation of a standard for the design and application of static seals was deferred because of funding limitations. Progress on the RDT Standard for Pump Design may be summarized as follows:

1. Bearings: Bearing characteristic design plots have been prepared. The writeup, which includes illustrative design problems, is being completed.

2. Dynamic Seals: Fluid-film characteristics have been generated and plotted for face seals, inclined surface seals, and hydrostatic seals. The draft, which also includes sample problems and a section covering the thermoelastic effects on seal operations, remains to be finished.

3. Hydraulics: This section is complete.

4. Rotor Dynamics: An analytical model has been established, and computer runs have been started to establish the dynamic response as a function of speed and bearing parameters. Sample problems that use input data from the bearing and rotor dynamics sections will indicate safe design techniques.

5. Structural Analysis: Three sections of the pump casing have been modeled with drawings supplied by the Bingham Pump Company, and the input data have been coded for the Finite Element Analysis of Axisymmetric Solids (FEAAS) computer program. One computer run covering pressure loading at room temperature was completed. Work on the following is planned: complete the runs covering the other two modeled sections for internal pressure loading only, repeat the above but include thermal effects, and investigate the effects of bolt loading on the casing while the casing is loaded by internal pressure.

Information on the strain gage results is expected soon from Bingham Pump Company. Two contacts were made with Byron Jackson Pump Company to obtain information relative to two pumps they tested by using strain gages. Byron Jackson Company has indicated that the results of these tests will be made available to the committee on design of the ASME Ad Hoc Committee on the Code for Pumps and Valves for Nuclear Power. They appear, however, reluctant to reveal what they consider proprietary or commercially sensitive information, despite assurances that such information will remain confidential.

Information was received and reviewed on the ATR primary coolant pumps. This information was transmitted by Idaho Nuclear Corporation to ORNL at the request of R. J. Schirk of the AEC Idaho Operations Office. It includes selected documents and reports summarizing analyses and tests performed to establish the integrity of the ATR pumps. Although the review of the data is incomplete, it is apparent that the method of analysis of the ATR pumps is similar to the method referenced in the RDT standard. The data transmitted are not prepared in a suitable way for direct incorporation in the RDT manual.

Valves and Piping

With the assistance of Teledyne Materials Laboratory, RDT Standard for Piping Design is being prepared. This standard, which is about 20% complete, presents techniques and methods of analysis of piping systems that meet the requirements established by USAS B31.7, Nuclear Power Piping Code. The preliminary drafts of the sections covering pressure design, initial flexibility analysis, and support and vibration control have been prepared.

Work continued on the ORNL effort to formulate a standard simplified method to calculate thermal gradients through pipe walls in water-cooled nuclear power plant piping systems.

Completion of the draft on the RDT Standard for Valve Design is now scheduled for October 1968. All sections, except valve fabrication, are in rough draft form. Sections entitled "Installation and Maintenance," "Reliability," "Operators and Position Indicators," and "Locking Devices" were recently completed.

A visit was made to the offices of NUS in Washington for discussion of the contents of the valve design section on fabrication. The Title page, Table of Contents, Index, and approximately 75 of the 100 illustrations for the standard were submitted for inclusion in the standard.

Velan Engineering Company of Montreal, a manufacturer of nuclear-grade valves for U.S. installations, was visited in the interest of getting a better understanding of current valve manufacturing technology, as well as to acquaint industry with the RDT Standards Program.

Heat Exchangers

A trip was made to Burns & Roe to review Heat Exchanger Transfer Equipment Design progress and to discuss specific modifications and additions. Other discussions centered on a method for discontinuity stress analysis of a tube sheet–shell–channel assembly for steady-state loads. Further discussions were concerned with light-water reactor system descriptions of various heat exchanger applications.

Selected sections of the evaluation guide on steam generator performance are being reviewed. Study and evaluation of comments are in progress on the revised RDT Standard for Steam Generator, E-3.

A meeting was held at ORNL on August 28, 1968, with representatives of RDT and ORNL during which time planning for future investigation of the heat exchanger tube vibration problem was reviewed.

A draft of the Heat Exchanger Tube Vibration Survey was completed and distributed for comments. This report evaluates the results of the survey on excessive tube vibration problems of primary heat exchangers in AEC reactors. Based on the limited data submitted, the results of the study, although not conclusive in establishing absolute control of vibration and related problems, indicate that a safe margin in design of heat exchangers can be established with some confidence if conservative engineering practices are permissible within performance limits.

A member of the Standards Program attended the Tenth Annual National Heat Transfer Conference held August 11–14, 1968, at Philadelphia, Pennsylvania.

6.4 ELECTRICAL, INSTRUMENTATION, AND CONTROL STANDARDS (AEC Activity 04 60 80 03 1)

A. E. G. Bates J. R. Potts

Electrical

Various aspects of electrical cable and cabling, which are peculiar to nuclear installations, were studied, since these will form the basis of the cable standard. An engineering survey revealed that other activities are in progress in this general area (for example, the IEEE "Guide for Classifying Electrical Insulating Materials Exposed to Neutron and Gamma Radiation") that can be referenced in this standard.

Visits were made to Babcock & Wilcox, Lynchburg, Virginia, to discuss their experience in the areas of electrical cable and penetrations; to Duke Power Company to discuss their experiences with cable and penetrations in the design and construction of the Oconee plant; and to the Electrical Design Branch of TVA to discuss cables and penetrations.

At the suggestion of B&W, a visit was arranged with Kenite Company, who have shown great interest in having their cable meet the nuclear requirements.

Instrumentation and Controls

The reactor protection system standards program is now under the guidance of an ad hoc committee, consisting of personnel from AEC DRL, DRT, and DRS; PNL; and ORNL General Engineering and Instrumentation and Controls Divisions. The scope of the program was enlarged to include not only AEC research reactors but also all privately owned power and other reactors. The new goal is to prepare supplementary criteria to supplement and complement the existing AEC "70 criteria," as well as the IEEE Protection System criteria when they are issued. The target date for completion of a draft of the supplementary criteria is mid-January 1969. Some of the material already prepared, or in various stages of completion, will be suitable for inclusion in the supplemental criteria.

The preparation of process instrumentation material is proceeding at a rate commensurate with the support available. In order to determine the prevalence of quality control, quality assurance, and similar programs in instrumentation, visits were made in July 1968 to Burns & Roe and to two old-line instrument manufacturers, Honeywell and Leeds & Northrup.

6.5 QUALITY ASSURANCE
(AEC Activity 04 60 80 03 1)

J. W. Anderson

A proposal was made by Idaho Nuclear Corporation for preparing the RDT Quality Assurance Standards in the form of three documents: (1) a description of a quality-assurance program, (2) a description of quality-assurance systems requirements, and (3) a handbook of guidelines for application and usage. Outlines of the first two documents were prepared by Idaho Nuclear Corporation. These outlines were reviewed by the Quality-Assurance Steering Committee who were generally agreeable to their content and arrangement. During visits to Stone & Webster, Burns & Roe, and Westinghouse Atomic Power Equipment Division to review quality-assurance practices, the outlines were also discussed.

The proposed plan for development of the standards was presented to RDT on August 22. General agreement was reached to proceed with the plan with initial efforts directed toward the preparation of the quality-assurance systems requirements document with additional consideration given to the areas of construction and operation.

In addition to the conferences noted previously, plans are being made to visit Pacific Gas and Electric Company; Southern California Edison Company; and General Electric Company, San Jose Division, to gain further insight into the quality-assurance practices of the nuclear utility industry.

7. GENERAL NUCLEAR SAFETY STUDIES

7.1 HTGR SAFETY PROGRAM OFFICE

(AEC Activity 04 60 70 01 1)

S. I. Kaplan M. D. Silverman
H. F. Bauman W. C. Ulrich

The HTGR Safety Program Office (HTGR-SPO) observes and evaluates research that contributes to the safety of high-temperature gas-cooled reactors and prepares analyses and recommendations on this topic for the guidance of the AEC Division of Reactor Development and Technology (DRDT). It is also responsible for preparing a long-range plan for the HTGR Safety Program.

Information Exchange

A trip report was issued covering HTGR-SPO participation at the Symposium on the Technology of Integrated Primary Circuits for Power Reactors sponsored by the European Nuclear Energy Agency in Paris, May 20–22, 1968, and visits to European gas-cooled reactor research sites.

Research Analysis

Contract research on various aspects of graphite oxidation is sponsored by at least four AEC branches with the general objective of developing a systematic means of predicting graphite attack in a reactor environment. The HTGR-SPO prepared and submitted recommendations on coordinating these activities to maximize the usefulness of the work for reactor programs of current interest.

In approving the preliminary safety analysis report for the Fort St. Vrain reactor, the ACRS specified several distinct areas in which they believed special attention was required to assure safe operation of the Fort St. Vrain plant. The Program Office reviewed the Committee's findings and submitted recommendations to DRDT on research to provide useful information in certain of the areas specified.

Safety Plan

Preparation of a revised HTGR safety plan is continuing with present emphasis on the areas of accident analysis and requirements for standards governing operational surveillance. In advance of the formal submittal of a complete plan, a series of safety-related research projects appropriate for AEC support in the future was discussed with the Research and Development Branch, DRDT, to aid in preparing long-range budget projections.

7.2 FUEL TRANSPORT SAFETY STUDIES
(AEC Activity 04 60 80 03 1)

L. B. Shappert

The fuel transport safety studies have three purposes: (1) the development of shipping cask criteria (the most important purpose), (2) the conduct of experimental work as required to develop the criteria, and (3) provide consultation on shipping cask problems. The criteria are to set forth minimum engineering practice standards for the design, fabrication, testing, inspection, and maintenance of irradiated fuel shipping casks. The experimental investigations include studies of the capabilities of the cask materials to absorb energy and withstand the effects of fires.

ORNL is cosponsoring, with the AEC, the Second International Symposium on Packaging and Transportation of Radioactive Materials, which is to be held at Gatlinburg, Tennessee, October 14–18, 1968. In addition to arrangements for the meeting, the Oak Ridge staff will be presenting six of the 55 papers on the meeting agenda.

Shipping Cask Criteria

Most of the section on Structural Design of Chapter 2 on Engineering Standards and Guide to the Design of Spent Fuel Shipping Casks, was reorganized and rewritten to reflect the latest information available. Information not previously incorporated includes several examples that utilize the equations presented and comparisons with model test results. Review of Chapter 5, Heat Transfer, was also completed.

Experimental and Analytical Studies

Demonstration Uranium-Shielded Cask

The demonstration cask successfully attenuated the radiation from an internal source that simulated the activity of four Indian Point Core A spent fuel elements. When loaded with 90,000 curies of ^{60}Co, the external radiation level at a distance of 1 m from the center of the cask

was less than 10 mr/hr. There was no indication of excessive radiation at any location on the cask surface.

This test demonstrated the adequacy of the basic design criteria and the fabrication procedures in regard to shielding thickness, closure design, and the method used to make cavity penetrations. Furthermore, it showed that the hypothetical accident series the cask was subjected to did not in any way impair its shielding ability.

The test was conducted in a hot cell at Los Alamos Scientific Laboratory with ^{60}Co borrowed from ORNL. The ^{60}Co was positioned along the center line of the cavity as a line source. No difficulty was experienced while inserting or removing the closure plug, even though these operations were conducted remotely.

Other Investigations

A 40-ft-drop-tower was designed to fit on top of the drop pad behind the steam plant. This tower will be useful in making controlled drop tests of specimens up to at least 500 lb.

A proposal for studying energy-absorbing capabilities of toroidal shells was received from the University of Tennessee. Decision on this subcontract awaits clarification of the Laboratory budget for this fiscal year. A summary report of the efforts of the University of Tennessee during the last fiscal year is now in preparation and should be received shortly.

Preparation of two further volumes of the ORNL-TM-1312 series on structural analysis of shipping casks was initiated. The first addition is an extension and rewriting of the draft on lifting devices prepared by L. R. Shobe. The second is on tie-down systems. Primary effort of these two activities is being provided by J. H. Evans.

Shipping Cask Consultant Work

PM-3A Reactor

Two requests for certification of the carriers for the PM-3A unirradiated type I spare core peripheral and center bundles were prepared

and are awaiting only a criticality evaluation prior to submission to the AEC.

Chemical Technology Division Cask Analysis

The applications for certification of the Foamglas shipping container, B of E 2884, and the vermiculite shipping container, B of E 2302, were approved by the AEC and returned to ORNL. The vermiculite shipping container was submitted to the DOT, and the Foamglas shipping container will be submitted to the DOT in the near future.

The modification of the trailer for the 10-ton californium shipping container was started. Application for certification of the TRU 10-ton californium shipping container, commonly called the Wabash Cannon Ball, which has been approved by the AEC, was forwarded to the DOT for approval.

Request for certification of the D-38 uranium-shielded cask was prepared and forwarded to the AEC for approval. A work order was written to modify this carrier, and the parts were made. It will be modified in the near future.

Analysis of the dry-hole charger cask was completed by General Engineering Division personnel. The cask modifications necessary to assure compliance to AEC Manual Chapter 0529 were minor. The formal request for AEC and DOT certification was prepared and will be forwarded to the AEC after internal ORNL approval. A work order was written to modify this cask in accordance with the General Engineering Division's recommendations.

A revised certification for the transuranium curium shipping container, DOT Special Approval No. 5461, was prepared to request permission to ship 425 μg of ^{252}Cf about September 15. This shipment will be made in an exclusive-use closed van, and requests to the AEC and DOT are being expedited.

Funds were approved to start design and evaluation work on the new cask to replace the hot garden carrier. This new cask is to be similar to the dry-hole charger but will have an internal cavity 7 in. in diameter by 24 in. tall and 8 in. of lead shielding.

Work on the summary report that describes all the radioactive and fissile material shipping containers available for use in the Chemical Technology Division is continuing.

7.3 DISCUSSION PAPERS ON VARIOUS ASPECTS OF WATER-COOLED REACTOR SAFETY

(AEC Activity 04 60 80 03 1)

R. H. Bryan

The titles of the eight discussion papers on various aspects of water-cooled reactor safety are listed in Table 7.1, together with the authors and actual (or expected as the case may be) publication dates of the final documents. As of September 20, 1968, seven of the documents had been distributed. Of the remaining two, one is awaiting AEC review and the other will be submitted for such review the first of October.

Table 7.1. Schedule of Publication of "Discussion Papers"

Discussion Papers	Authors	Reproduction and Distribution Completed
1. Missile Generation and Protection in Light-Water Reactor Plants	R. C. Gwaltney	9-12
2. Potential Metal-Water Reactions in Light-Water Power Reactors	H. A. McLain	8-5
3. Emergency In-Core Cooling Systems for Light-Water Power Reactors	C. G. Lawson	9-9
4. Air Cleanup as an Engineered Safety Feature in Light-Water Power Reactors	G. W. Keilholtz, E. C. Guthrie, and G. C. Battle	9-10
5. Testing of Containment Systems Used with Light-Water Power Reactors	F. C. Zapp	8-28
6. Consideration of Additional Nuclear Reactor Plant Safety Features for the Control of Accident Released Activity	NSIC Staff	9-9
7. Earthquakes and Nuclear Power Plant Design	C. G. Bell and T. F. Lomenick	(a)
8. Protection Instrumentation Systems in Light-Water Reactor Plants	C. S. Walker and H. G. O'Brien	(b)
Related Reports		
1. Design Principles of Reactor Protection Instrument Systems	S. H. Hanauer and C. S. Walker	9-19
2. Some Considerations in Earthquake Resistant Designs of Nuclear Power Plants	Holmes & Narver, Inc.	(c)
3. Summary of Current Seismic Design Criteria for Nuclear Reactor Facilities	J. A. Blume & Associates	(c)

[a]Final draft submitted 5-25, AEC review in process.

[b]Final draft to be submitted 10-4.

[c]No approval required; reports completed and distribution to be made by DTIE.

7.4 ANTISEISMIC DESIGN OF NUCLEAR FACILITIES
(AEC Activity 04 60 80 01 1)

R. N. Lyon

All nuclear power plants being built in the United States are designed to resist shaking by at least a moderate earthquake. Where earthquakes are more frequent than average, power plants are designed for very severe earthquakes. Many otherwise suitable sites along the West Coast are close enough to a major fault to support the possibility of a fault slip or secondary slip under the plant.

A program of antiseismic design studies of nuclear facilities is under way in which the initial tasks included both shaking tests of existing plants and conceptual designs of nuclear power plants capable of withstanding faults of up to 6 ft. The draft of the research and development program incorporating these tasks was revised following AEC and internal ORNL review. It will be released to the AEC before the end of September.

Conceptual Antiseismic Design Studies

A review of four conceptual design studies completed in June is summarized in Table 7.2. The concepts were intended to provide resistance of several levels of shaking up to 1 g horizontal and up to 6 ft of vertical and horizontal slip. It was found that containment vessels not designed to resist slip would, in general, be ruptured by only a few inches of differential horizontal slip in rock or reasonably compacted cohesionless soil.

There appear to be no significant development problems associated with the floating concept of Daniel, Mann, Johnson, and Mendenhall.[3] The other three concepts need investigation of ways to avoid rupture of the containment system due to unusual strain of umbilicals. In the case of the Holmes & Narver[1] concept, flexible lines for essential services are required. Other lines might be designed to disconnect or "break away" at strains below those that would endanger the containment vessel. The Holmes & Narver[1] concept needs additional information on passive pressure

Table 7.2. Summary of Studies on Antiearthquake Nuclear Power Plant Designs

Concept	Engineering Firm and Reference	Incremental Cost ($ million)	Remarks and Limitations
Engineered imbedment	Holmes & Narver (Ref. 1)	1.3 to 2.2	No allowance is made for umbilicals; cost includes hardening the containment against shaking, but little provision against shaking of internal systems; report contains an annotated list of important components
Unitized	Kaiser Engineers (Ref. 2)	5 to 5.6	Containment vessel analyzed for shaking, but internals not completely analyzed for shaking; can maintain shutdown for four days without external connections
Floating on barge	Daniel, Mann, Johnson, and Mendenhall (Ref. 3)	Barge cost 14.7 Breakwater 4.3 Total 19.0 About 7 to 9 above unprotected plant on an artificial island	Barge contains entire 1000-Mw package power plant; operating-basis earthquake may be same as design-basis earthquake; no additional provision needed to resist shaking; barge draws about 30 ft of depth
Floating of reactor in a dense solid-liquid suspension	Burns & Roe (Ref. 4)	Equipment for fluid preparation and conditioning and fluid ingredients 5.3 Caisson basin 6.8 Other costs 3.2 Total 15.3	Cost analysis reasonably complete; major problem is gelling and settling properties of chosen slurry; gelling and plastic properties limit fluid loss in case of crack in basin wall; system can be maintained safely in shutdown condition for about 10 hr without external connection

of soils against the containment vessel and on the motion of the containment vessel during a fault slip.

Both the Holmes & Narver and Kaiser[2] concepts would benefit from more information on shock-absorbing or crushable materials. The Burns & Roe[4] concept can be improved by development of a better slurry or suspension and research on the rocking behavior of a floating body in a confined fluid.

Shaking Tests at EGCR and CVTR

In early August the University of California at Los Angeles attached two 5000-lb-capacity shaking machines to the service machine floor of the Experimental Gas-Cooled Reactor and measured the resonant frequencies and amplitudes at a wide variety of locations. By arrangement with the Phillips Petroleum Company, similar tests were made by UCLA on the Carolinas-Virginia Tube Reactor. The results are still being studied, but it has already been determined that some of the resonant frequencies were accurately predicted by Witt, Carver, and Maxwell[5] in 1962.

It was found that vibrations were strongly intensified in some of the piping. An amplification by a factor of 120 was noted in the steam line from the top of the steam generator. Most of the measurements were made below 10 cps, but several sweeps were made up to 50 cps.

References

1. Holmes & Narver, Inc., Antiseismic Design of a PWR Nuclear Power Station, USAEC Report HN-192, April 1968.

2. Kaiser Engineers, Design Study for Earthquake Resistant Nuclear Power Plant, USAEC Report K-68-4, January 1968.

3. Architect-Engineers Daniel, Mann, Johnson, and Mendenhall, A Floating Earthquake-Resistant Nuclear Power Station, December 1967 (undocumented).

4. Burns & Roe, Inc., Conceptual Design of a Boiling Water Reactor Plant to Resist Ground Fault Displacement, May 1968 (undocumented).

5. F. J. Witt, D. R. Carver, and R. L. Maxwell, Effects of Seismic Vibrations on the Experimental Gas-Cooled Reactor, USAEC Report ORNL-3224, Oak Ridge National Laboratory, June 8, 1962.

8. NUCLEAR SAFETY INFORMATION

8.1 NUCLEAR SAFETY INFORMATION CENTER
(AEC Activity 04 60 70 01 1)

J. R. Buchanan Wm. B. Cottrell

The Nuclear Safety Information Center was established in 1963 by the USAEC Division of Reactor Development and Technology to collect, evaluate, and disseminate nuclear safety information to governmental agencies, research and educational institutions, and the nuclear industry. The Center's basic computer reference system now contains 100-word abstracts of 22,000 documents.

Research and Development Management File

Programming and development continued on a system for computer storage and retrieval of administrative and technical achievement information on nuclear safety research and development contracts. The finished system will be quite similar to the present NSIC system as far as output, adaptability to remote consoles, and general use are concerned. However, the length of a given contract description is longer than an NSIC bibliographic accession, since it contains more data and text. The batch printout program has performed successfully on a group of 20 sample contracts. The first group of real contracts, 10 projects, was received from the Water Reactor Safety Program Office, edited, and submitted to the computer facility for keypunching.

A review of the plans for the file was held with AEC on July 30. As a result, a few minor modifications are being made in type of data and text to be stored in the file and in the style of output.

Telecommunications

The IBM-2740 remote typewriter-printer continues to be a useful tool in retrieving information from the computer reference storage file.

Checkout of seven input programs on the IBM-2260 cathode-ray tube station is under way. The two 2260 consoles are to be used initially for putting new references into the computer master storage file. Two programs

necessary to the orderly functioning of this operation are currently being written. One, a background batch file maintenance routine, will transfer the data that is directly input into the IBM-360 "data cell" to magnetic tape on an end-of-day basis. This will provide a backup file so that the input work of only one day would have to be reconstructed in case the "data cell" file should be destroyed. The other program, still being written, will provide an alternate mode of card input to handle any overload of the remote input system. In addition two 2260 conversion programs were written but are yet to be tested. These programs will make the data input remotely available to routine production operations, such as SDI and the preparation of the indexed bibliography.

Selective Dissemination of Information Program

The SDI program continues to grow and now reaches over 1300 users. Profiles of the SDI participants must be revised occasionally. This is usually due to a change in job assignment, but sometimes revision is necessary because the user did not fully understand the nature of the material he requested and consequently he finds himself receiving some references of peripheral interest.

Two aids have been used in profile revision in addition to the feedback cards. The first was a survey to determine the desire of users relative to receiving abstracts of the questions the AEC Division of Reactor Licensing directs to those who have filed license applications and the responses to these questions. While over 70% of the participants found this information very useful, 28% of those responding to the survey said they would prefer to have it deleted from their SDI. Their profiles were revised accordingly.

The second aid is the output of a program that computes the percentages of the references of "much," "some," and "little interest" to the SDI user. Those participants who have the highest percentage of "little interest" references are given priority in profile adjustment.

During the report period, special profiles were prepared for a group with specialized interest in the AEC Division of Reactor Licensing. These profiles sort out all failures and incidents by reactor type. These

profiles are also being sent to several utility staff members. Sufficient feedback has not as yet been received to determine the effectiveness of these profiles.

Reports

The Center's fourteenth indexed bibliography was issued in July. It contains references sorted into the usual 19 subject categories.[1]

Sixteen copies of a draft of the report "Sources of Tritium and Its Behavior Upon Release to the Environment" were distributed in August for a final review. Comments are now available, and it is anticipated that the report will be revised and ready for publication very soon.

Guide to Nuclear Power Education and Training Programs

Tentative agreement was reached with AEC for NSIC to prepare the text for a booklet on nuclear power education and training programs. The booklet will review utility requirements for personnel, individual job qualifications, typical education and training schedules, and availability of education and training programs.

Other Information Activities

During the reporting period the following requests were handled by NSIC personnel:

Information inquiries (letter or phone)	91
Specific NSIC reports	301
Additions to NSIC reports distribution	18
Visits for consultation or use of reference material	11
Additions to SDI program (net)	44

Reference

1. NSIC Staff, Indexed Bibliography of Current Nuclear Safety Literature -14, USAEC Report ORNL-NSIC-45, July 1968.

8.2 COMPUTER HANDLING OF REACTOR DATA – SAFETY (CHORD–S) (AEC Activity 04 60 70 01 1)

T. E. Cole

The Computer Handling of Reactor Data – Safety (CHORD–S) project is developing a computerized system for the clear, concise, and flexible documentation of safety characteristics of nuclear power plants. This task has four distinct subtasks:

1. identification and formatting of safety characteristics,
2. collection of information,
3. computer programming for data retrieval and comparisons,
4. development of system hardware.

Identification and Formatting of Safety Characteristics

All characteristics listings for the summary category except for subsection AI (Auxiliary Systems) are now stored in the computer in revised form to accommodate suggestions from DRL. Characteristics listings for the PWR containment section, except for subsection EBC (Electrical, Instrumentation, and Control), were released by ORNL on July 19 for DRL/MPR review. Recommended changes from DRL/MPR technical personnel were provided on August 9; DRL administrative approval is pending. The characteristics listing for subsection EBC is scheduled for completion by November 1, 1968; however, uncertainties regarding the FY-69 budget may require rescheduling as a result of probable reductions in subcontractor effort.

Collection of Information

Summary section data for ten reactors were compiled from PSAR documents and amendments and then entered into the computer. A rewrite of this data was almost completed in compliance with recommendations from DRL for less truncation, more abbreviation, and the elimination of inaccuracies or multiple interpretations.

At the request of DRL, ORNL is working on a preliminary questionnaire on the site category of the summary section that could be used in the future for soliciting CHORD–S information directly from industry.

Computer Programming for Data Retrieval and Comparisons

During recent months, major CTC efforts in CHORD–S computer programming were concentrated on changes in terminal commands and readout formats for greater convenience and utility to potential DRL users. Operation of the Bethesda console provided experience that led to additional improvements, most of which were specifically requested by DRL. A prime example of these useful improvements is the development and implementation of a new option for $COMPARE whereby users can specify a value or range of values or specify a search for the maximum value for the specified key-code and obtain a readout that lists the reactors and gives the data falling into such designated ranges. Still further improvements are currently being made and are expected to be operational in the next few weeks.

Development of System Hardware

For several weeks a demonstration training program was conducted by ORNL on the Bethesda telecommunications terminal for a large number of DRL staff members. Retrieval of CHORD–S data by telecommunications link to CTC Oak Ridge has been successful for all the various modes programmed thus far. Permanent IBM 2740 terminal equipment for Bethesda will replace the temporary IBM 1050 unit about October 1.

Terminal unit (IBM 2740) for the CHORD–S group in Oak Ridge was installed and placed into operation August 15. This equipment is now being used for further testing and developing of data-retrieval techniques. (Previously, the NSIC station was used for that purpose, when available.) To match the capability of the new IBM 2740 telecommunications typewriters, the line length of the computer feedback was increased to 154 characters. The number of characters for each of three columns (vertical $COMPARE) was also increased to 27 and provision was made to continue data items of greater length on succeeding lines in order to eliminate truncations of data.

A preliminary internal report was prepared by ORNL to describe information collected thus far on the nature of complex factors that will have to be optimized in selecting CRT terminal equipment capable of interfacing

with a telephone system connected to a central time-sharing computer. On July 29, the CHORD-S group sponsored a demonstration in Oak Ridge of one representative CRT telecommunications terminal for the purpose of evaluating the potentials of that medium of displaying CHORD-S data.

8.3 TECHNICAL PROGRESS REVIEW NUCLEAR SAFETY

(AEC Activity 07 13 02 02)

J. P. Blakely Wm. B. Cottrell

Nuclear Safety

Nuclear Safety is a bimonthly Technical Progress Review that is prepared and edited at ORNL under AEC contract. The Sept.–Oct. 1968 issue (Vol. 9, No. 5, 112 pages, distributed Sept. 6) contains a Feature Article on reactor safety standards by Milton Shaw, Director of AEC's Division of Reactor Development and Technology. Mr. Shaw discusses the essential role that standards play in achieving and assessing reactor safety. He states that nuclear industry use of accepted standards is needed to enable users and vendors inexperienced in the new technology to undertake complex power reactor activities with an acceptable degree of success and safety. Other articles deal with waiving defenses: a new approach to protecting the public against financial loss from use of atomic energy; the pool-boiling systems part of a two-part article on burnout in boiling heat transfer; the control and safety of pulse reactors; removal of radioiodine from gases; environmental dose evaluations; ion-exchange resin systems failures in processing actinides; and the regular current events activities (see Fig. 8.1 for contents of Vol. 9, No. 5).

The Nov.–Dec. 1968 issue (Vol. 9, No. 6) was distributed for external review in draft form; the review was completed; and comments are now in the hands of the authors. Final changes are scheduled to be transmitted to DTIE by mid-September, with distribution of the issue scheduled for early November.

The review draft for the Jan.–Feb. 1969 issue of Nuclear Safety (Vol. 10, No. 1) was distributed for comment as report ORNL-TM-2329, with comments due by Oct. 1. Following author response to comments, makeup, and printing, final distribution is scheduled for early January.

Article manuscripts were received for the March–Apr. 1969 issue of Nuclear Safety (Vol. 10, No. 2) and are undergoing in-house review. The draft is scheduled to be distributed for external comment in early November, with publication due in early March.

NUCLEAR SAFETY

Vol. 9, No. 5 Sept.–Oct. 1968

Contents

FEATURE ARTICLE

Standards and Reactor Safety
by Milton Shaw See page 337

General Safety Considerations

343 Waiving Defenses: A New Approach to Protecting the Public Against Financial Loss from Use of Atomic Energy
Leonard M. Trosten and William T. England

Accident Analysis

351 Burnout in Boiling Heat Transfer—Part I: Pool-Boiling Systems
W. R. Gambill

Control and Instrumentation

363 The Control and Safety of Pulse Reactors *L. C. Oakes*

Plant Safety Features

373 Removal of Radioiodine from Gases *R. E. Adams and R. D. Ackley*

Consequences of Activity Release

383 A Survey of Environmental Dose Evaluations *J. F. Honstead*

Operating Experiences

394 Resin Explosions in Processing Actinides *F. W. Miles*

Current Events

407 Progress Summary of Nuclear Safety Research and Development Projects
Compiled and Edited by A. W. Savolainen and H. B. Whetsel
430 General Administrative Activities *Compiled by J. P. Blakely*
432 Action on Reactor and Other Projects Undergoing Regulatory Review or Consideration *Compiled by J. P. Blakely*
441 Table of Proposed Rule Changes as of June 1, 1968

Miscellany

362 *Nuclear Safety* Cumulative Index (Availability)
382 Nuclear Safety Information Center Reports
440 Packaging of Radioactive Materials Meeting at Gatlinburg (Announcement)
442 UKAEA Reactor Safety Course No. 8 (Announcement)

Fig. 8.1

Outlines were developed and authors are in various stages of contact for future issues through Vol. 10, No. 6, Nov.–Dec. 1969.

Nuclear Safety Program Seminars

In another continuing program of information presentation, seminars were held in July and August on Nuclear Safety Program topics. The July seminar was given by R. H. Bryan who discussed AEC policy and procedures derived from the Atomic Energy Act of 1954 for evaluating the safety of reactors. The Act enjoins the use of minimum regulation consistent with the health and safety of the public. He noted that conversion of the legislative requirement into a technological procedure creates a dilemma that may account for some of the often-noted variations in licensing practice.

The August seminar was presented by Meyer Bender (General Engineering Division) who discussed the AEC-RDT campaign to establish high-quality standards in the nuclear industry. He explained the controversy in fixing standards, outlined the positions of various concerned interests, and illustrated the RDT approach to standards for both commercial power and experimental reactor installations.

Future seminars were scheduled for September, October, and November.

Internal Distribution

1. R. D. Ackley
2. R. E. Adams
3. T. D. Anderson
4. J. E. Baker
5. C. J. Barton
6. S. E. Beall
7. C. G. Bell
8. M. Bender
9. R. L. Bennett
10. C. R. Benson
11. R. F. Benson
12. R. G. Berggren
13. D. S. Billington
14. F. T. Binford
15. J. P. Blakely
16. R. E. Blanco
17. C. M. Blood
18. A. L. Boch
19. G. E. Boyd
20. R. B. Briggs
21. W. E. Browning, Jr.
22. R. H. Bryan
23. J. R. Buchanan
24. C. A. Burchsted
25. T. J. Burnett
26. D. A. Canonico
27. D. W. Cardwell
28. T. E. Cole
29. Z. Combs
30. J. A. Conlin
31-35. Wm. B. Cottrell
36. K. E. Cowser
37. E. N. Cramer
38. G. E. Creek
39. D. J. Crouse
40. F. L. Culler, Jr.
41. R. J. Davis
42. W. DeLaguna
43. H. J. deNordwall
44. W. K. Ergen
45. D. E. Ferguson
46. W. F. Ferguson
47. H. F. E. Feuerstein
48. M. H. Fontana
49. A. P. Fraas
50. J. K. Franzreb
51. S. H. Freid
52. J. H. Frye, Jr.
53. A. B. Fuller
54. W. R. Gall
55. J. S. Gill
56. D. L. Gray
57. B. L. Greenstreet
58. J. C. Griess
59. W. R. Grimes
60. R. C. Gwaltney
61. R. P. Hammond
62. P. N. Haubenreich
63. F. A. Heddleson
64. R. E. Helms
65. W. H. Hinds
66. J. M. Holmes
67. D. G. Jacobs
68. W. H. Jordan
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537-538. General Atomic Library, P. O. Box 608, San Diego, California (Attention: R. M. Fryar)
539. Librarian, Bechtel Corporation, Gaithersburg, Maryland
540. Laboratory and University Division, ORO
541-555. Division of Technical Information Extension (DTIE)