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ORNL NUCLEAR SAFETY RESEARCH AND DEVELOPMENT PROGRAM

BIMONTHLY REPORT FOR SEPTEMBER-OCTOBER 1968

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ORNL NUCLEAR SAFETY RESEARCH AND DEVELOPMENT PROGRAM BIMONTHLY REPORT FOR SEPTEMBER—OCTOBER 1968

Wm. B. Cottrell

JANUARY 1969

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

HOSMER TO ADDRESS PARTICIPANTS AT ORNL NUCLEAR SAFETY PROGRAM ANNUAL INFORMATION MEETING

The next annual information meeting on the Nuclear Safety Program at ORNL will be held on February 17, 18, and 19, 1969 in the Laboratory's main auditorium in Building 4500. The primary purpose of the information meeting is to keep ORNL staff members informed regarding the status of the multiple safety-oriented problems under study.

So much outside interest was generated at the February 1968 meeting, however, that it was decided to announce the upcoming meeting in this program bimonthly report so that interested persons could plan to attend. The Honorable Craig Hosmer, Representative to Congress and JCAE member, will speak at a dinner meeting on Monday evening February 17, at the Oak Ridge Country Club, and tours of ORNL will be conducted on Tuesday afternoon, February 18. Four half-day sessions are planned on Behavior of Accident-Released Fission Products, Spray and Pool Pressure-Suppression Program, Failure Modes of Zircaloy-Clad Fuel Rods, Filtration and Absorption Technology, Gas-Cooled Reactor Safety Program, General Nuclear Safety Studies, and Pressure Vessel and Piping Technology. Questions and discussion will be invited from the audience.

Persons interested in attending the meeting may write to Wm. B. Cottrell, Oak Ridge National Laboratory, P. O. Box Y, Oak Ridge, Tennessee, so that prior clearance into the Laboratory may be arranged. There will be no registration fee.

CONTENTS

ABSTRACT

The accomplishments during the months of September and October 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.

SUMMARY

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

Comparison of Real and Simulated Fission-Product Aerosols

The last of a series of tests to compare real and simulated fission-product aerosols was completed during the preceding reporting period. Some analytical results are still outstanding, but a comprehensive evaluation of all tests in the series was initiated and will be completed during the next reporting period.

Failure Modes of Zircaloy-Clad Fuel Rods

Coordination for the study of failure modes of Zircaloy-clad fuel rods was set up, and a program plan was prepared. MTR irradiation of the center fuel rods to be tested to failure in TREAT was begun, and a duplicate instrumented rod is being constructed so that internal temperature histories in the fuel rods can be inferred.

LOFT Assistance Program

The scope of the LOFT assistance program at ORNL was modified to provide support to Phillips Petroleum Company in three task areas:

A. Fission-Product Release from Unclad Fuel,

B. Fission-Product Release from Long Rods Coincident with Emergency Core Cooling,

C. Fission-Product Transport and Containment Behavior and Removal.

However, because of limited funds, only task B and part of task C can be attempted in FY-69.

LMFBR Safety

Preliminary planning and feasibility studies were started to determine areas of research and development on LMFBR safety to which ORNL's facilities and capabilities can best be applied.

2. Filtration and Adsorption Technology

Removal of Solid Aerosols

Filtration efficiencies were determined under reactor accident conditions for wet and dry stainless steel oxide—UO_2 aerosols as a function of face velocity, humidity, and filter media. Penetrations ranged from 0.03 to 0.06%. For dry aerosol, penetration was slightly higher at 60°C than at 25°C. At 60°C the wet aerosol was more penetrating than the dry.

Ignition of Charcoal Adsorbers by Fission-Product Decay Heat

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 used in trapping fission products released in a reactor accident. A laboratory apparatus was constructed and tested on known materials to determine the static and dynamic heat transfer behavior of charcoal, and the results were successfully correlated by theoretical analysis. Experimental tests are being carried out on moist charcoal under accident conditions and will be similarly correlated.

Separation of Noble Gases from Air with Permselective Membrane

Previous capital cost estimates of permselective membrane plants for removing noble gases from the cover gas of a nuclear reactor were used to estimate the costs of similar plants for removing noble gases from the off-gas of a nuclear fuel processing plant. Costs as a function of processing rate were obtained for a noble gas concentration factor of 100 and decontamination factors of 10, 100, and about 1000. For a case of practical interest, the processing of off-gases from a 5-MT/day chemical processing plant, it was estimated that for the three decontamination factors, capital costs would be $260,000, $390,000, and $560,000, respectively. The $390,000 cost may be compared with about $420,000 for a corresponding plant in which the noble gases would be absorbed in fluorocarbon solvents.

High-Efficiency Air-Filtration Engineering Manual

The first draft of the manual entitled "Design and Construction of High-Efficiency Air-Filtration Systems for Nuclear Applications" was completed and issued to a limited distribution for review. When review has been completed, the manual will be revised and issued for general distribution.

3. Spray and Pool Pressure-Suppression Technology

Effect of Additives on Distribution of CH_3I Between Air and Water

The distribution coefficient for CH_3I between water and air was measured. Experimental apparatus was fabricated for measurement of the reaction rate of CH_3I in aqueous spray solutions.

Uptake of I_2 and CH_3I by Water Solutions and Drops

Tests with a surfactant mixed with a reducing agent such as formaldehyde or $Na_2S_2O_3$ indicated significant CH_3I removal rates that will be beneficial to reactor siting. The mass transfer and distribution coefficients associated with the basic borate—thiosulfate system containing surfactant are some of the largest ever measured for CH_3I.

Spray Studies at the Nuclear Safety Pilot Plant

A CH_3I spray removal experiment was completed with a basic borate—thiosulfate solution containing 5 wt % $Na_2S_2O_3$, which compares with the standard mixture of 1 wt %. The removal half-life was 26.6 min, as predicted by the theoretical model.

Radiation and Thermal Stability of Spray Solutions

Basic borate solutions containing surfactant additive were exposed to ^{60}Co radiation to obtain a dose of approximately 10^7 r. While the results indicate good radiation stability, it will be necessary to reevaluate CH_3I-removal ability after irradiation. Radiation-coupled corrosion studies are currently under way.

Spray Solution Corrosion Studies

Samples of construction materials were tested in both basic borate and basic borate—thiosulfate solutions in the corrosion loop. All materials tested with the exception of aluminum and copper exhibited negligible corrosion at 100°C for 168 hr and 140°C for 24 hr. Solution-stability measurements confirmed earlier results obtained by Zittel. Corrosion of samples exposed to the basic borate solution at 140°C for 24 hr was acceptable, except for aluminum.

Pressure-Suppression Experiments

Experiments to obtain information to permit analytical modeling of energy dissipation and fission-product transport in power reactor pressure-suppression pools continued. An initial series of iodine-retention experiments was planned.

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

Computer program development for modeling of fission-product trapping continued. The experimental equipment is nearing completion. Tests of Carbo-Zinc No. 11 protective coating stability under pressure-suppression pool conditions were started.

4. Safety Studies for HTGR

In-Pile Studies of Reactions of Graphite with Steam

During this period the in-pile part of the first experiment to investigate the stability of irradiated coated particles in a bonded bed following blockage of a coolant channel was completed. Data from off-gas sampling during the in-pile phase were compiled. Evaluation of results awaits postirradiation examination.

Fission-Gas Release from Coated Particles Under Accident Conditions

Detailed design of the apparatus for in-pile studies of steam-graphite reactions is almost complete. Current laboratory investigations of the reaction are concerned with a quantitative determination of the effect of changes in the internal geometry of graphite during the reaction with steam. The apparatus and techniques for doing this are being tested with inert gases. Experiments are also being designed to determine the importance of initial transient conditions in the interpretation of experimental results.

Experimental Test of FREVAP-8 Code for Calculating Metal Fission-Product Release from HTGR Fuel Elements

While construction of the in-pile equipment proceeds, some time was used to compare observed release of metallic fission products with predictions. Mathematical models used by GGA in the analysis of the Fort St. Vrain plant were used for this purpose. It was concluded that the model is satisfactory when applied to a steady-state situation, but there is evidence that estimates of release made for this reaction will be high because a steady state will not be reached for a long time. The exact margin of error is not likely to be known until data on strontium diffusion compiled in collaboration with GGA can be evaluated.

5. Pressure Vessel and Piping Technology

Heavy-Section Steel Technology Program

Four contracts were finalized in support of the fracture mechanics and simulated service test activities. The second semiannual program report was issued. Procurement efforts for the simulated-service tests of intermediate-size vessels are proceeding. The ultrasonic indications in a section of the first program plate were shown to define an actual material discontinuity. Properties of program plates 01 and 02 were shown to be almost identical. Investigations of a submerged-arc weld were undertaken. Preliminary results from dynamic tear tests indicate a rapid fracture mode transition for all specimen sizes (up to 12 in. thick). This transition is supplemented by fracture toughness (K_{Ic}) measurements up to 50°F. Irradiation programs are currently under way at Pacific Northwest Laboratory and at Oak Ridge National Laboratory on program plates and weldments.

Experimental and Analytical Investigations of Nozzles

Experimental and analytical work is continuing on both single nozzles and clusters of nozzles in spherical and cylindrical shells and in flat plates. The series of tests on the steel spherical shell model with a single radial nozzle (7 7/8 in. OD by 0.1875 in. in wall thickness) was completed. Loadings included (1) internal pressure, (2) pure bending moment applied to the nozzle, (3) torsional moment applied to the nozzle, (4) axial thrust on the nozzle, and (5) direct shear thrust applied at the nozzle-shell junction. The steel spherical shell model with two large radial nozzles (7.00 in. OD by 0.333 in. in wall thickness) was retested with internal pressure loading. Two computer programs are presently being evaluated and debugged: (1) a solution for a nonradial nozzle attached to a spherical shell loaded with an internal pressure and (2) a solution for a single radial shell attached to a cylindrical shell loaded with an out-of-plane bending moment on the nozzle.

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 preliminary draft of the literature survey and technical evaluation report was completed and distributed for review. Contract negotiations for the finite-element analysis and for the experimental stress analysis and fatigue of two ASA B16.9 tees are under way. Plans are being formulated for further analytical study of elbows, concentric reducers, tapered transition joints, lugs, and bolted-flanged joints.

6. RDT Standards Program

RDT Standards Program Objectives and Activities

The Oak Ridge National Laboratory, at the request of the AEC Division of Reactor Development and Technology (DRDT), assumed prime responsibility for preparing engineering standards for RDT reactor and related programs. Present efforts are directed toward continuation of the review and updating of existing standards and the issuance of tentative standards; preparations of scopes and outlines and collection and review of technical data for standards; and finalizing guidelines necessary for clarity, uniformity, and continuity in writing standards. To date 86 drafts have been submitted to RDT, 61 have been given tentative approval, and 56 were issued as tentative standards. A draft of an RDT Standards Index was developed.

Reactor Coolant Systems and Equipment Standards

Development of standards for reactor coolant systems and equipment continued. The standard for piping and valves is about 37% complete. Preliminary drafts of standards for reactor internal design and for selection of reactor coolant pump functional design requirements are being revised. The standard for pump design is 70% complete. A preliminary draft of a heat transfer design standard for heat exchangers is nearly complete. A survey of primary heat exchangers revealed that those in 9 out of 17 plants had sustained damage, apparently due to tube vibration. Vessel Standards RDT-S-918 and RDT-E-4 are under review for possible reissue.

Instrumentation, Controls, and Electrical Standards

An outline is being developed for documents on reactor protection instrumentation systems to serve as supplements to the AEC 10CFR50 and IEEE No. 279 criteria. A first draft of a standard for thermocouples was prepared and is undergoing internal review. Work was started on an electrical design standard.

Programmatic and Procedural Standards

Work continued on programmatic and procedural standards. A survey was continued of quality-assurance practices in industrial and AEC reactor projects. An initial draft of "Quality-Assurance Program Requirements" is being reviewed, and "Quality-Assurance System Requirements for Construction" is near the end of the review process. At the request of RDT, maintenance standards effort is being shifted from components standards to comprehensive maintenance program standards. A survey is being made of maintenance practices in government and industry nuclear projects. Review is under way of the preoperational testing standard; a first draft is being prepared on general testing requirements; and work was started on the containment testing standard. The document on reliability is being prepared for publication. Work was initiated on water chemistry standards by surveying coolant chemical control methods and experience and by preparation of outlines for standards for control of primary coolant composition. Standardized paragraphs on cleaning of mill products are being prepared for inclusion in specifications and standards. Planning is under way on standards for cleaning, cleanliness control, and preoperational cleaning of critical components, piping, and systems. A style manual for standards was issued for comment and internal use. The standards program participated in the issuance of American Association for Contamination Control Standards on HEPA filters and on laminar-flow clean-air devices.

Materials and Fabrication

Materials and fabrication standards were developed further. A tentative standard on ultrasonic examination was distributed. Tentative standards on zirconium and zirconium alloys for reactor internals were submitted to RDT; aluminum and aluminum alloy standards are 60% complete. The welding standard for fuel elements is being revised to incorporate RDT comments. A tentative draft for nickel-molybdenum-chromium alloy sheet and plate was rewritten. An outline was prepared for a standard on protective coatings.

7. General Nuclear Safety Studies

HTGR Safety Program Office

As a result of discussions between representatives of RDT and the HTGR Safety Program Office on coordinating AEC-sponsored graphite oxidation research, a Program Office member will serve on the AEC Graphite Coordination Working Group. In October, Program Office members presented two papers on helium uses in atomic energy at the Helium Applications Conference in Washington.

Fuel Transport Safety Studies

All writing for the "Engineering Standards and Guide to the Design of Spent Fuel Shipping Casks" was finished. The Guide is to be published during the next reporting period. The recently completed drop-tower facility was used to test a scale-model buffered cask. Plans are being made to use the tower for cask-component testing. All destructive tests on Paducah's uranium-shielded cask were completed. The cask passed all tests satisfactorily.

Discussion Papers on Various Aspects of Water-Cooled Reactor Safety

Six of the eight discussion papers on various aspects of water-cooled reactor safety have now been published. The remaining two were completed and are ready for publication pending final AEC review.

Antiseismic Design of Nuclear Facilities

A proposal for antiseismic studies was outlined for the AEC and is now being slightly revised. Vibrations from some of the scheduled nuclear test shots in Nevada may be used in these studies. Experts will be canvassed for proposals to develop soil-test methods. The pros and cons of seismic scram are being studied.

8. Nuclear Safety Information

Nuclear Safety Information Center

The Nuclear Safety Information Center's storage file now contains computer-retrievable information on over 23,000 nuclear safety documents. The IBM-2260 (CRT) remote consoles serving the NSIC offices are being tested as input devices by adding references to a special storage file via one of the stations. Revisions were made in the programs for the research and development contract management file. A draft of the booklet "Guide to Nuclear Power Plant Staffing: Requirements, Training, and Education Programs" was completed and submitted to the Commission for review. A set of tables compiling reactor design features for comparative purposes were prepared for ACRS use.

Computer Handling of Reactor Data — Safety (CHORD-S)

The principal efforts in the CHORD-S project were to complete the editing and revisions of the data for the Summary section for the ten reactors for which data were collected. A printout of this information was reproduced and copies were forwarded to the USAEC Division of Reactor Licensing (DRL). General facility information on names, addresses, significant dates, and power levels was added to the data bank for 45 additional reactors specified by DRL. The IBM 1050 remote teletypewriter console located in Bethesda and coupled to the IBM 360, 50/65 at the Computing Technology Center in Oak Ridge was replaced by the IBM 2740 console originally specified for this service. The computer programming effort necessary to provide the basic information storage and retrieval features requested by DRL was completed, and the changes are now operational on the remote consoles.

Technical Progress Review Nuclear Safety

The November–December 1968 issue of Nuclear Safety, a bimonthly technical progress review, was distributed on November 25, 1968. It carried a feature article on the Otto Hahn by authors from GKSS and other articles on safety aspects of accelerator shielding, forced-convection systems, component reliability, LMFBR fission-product release, luminescent dosimetry, etc. Other issues of Nuclear Safety through November–December 1969 are in various stages of preparation.

Two seminars were arranged and presented for Nuclear Safety Program members. The September seminar was given by Wilson R. Cooper, TVA-Chattanooga, on TVA's experience in obtaining construction permits for the Browns Ferry reactors. The October presentation was by Earl M. Shank, U.S. technical advisor to the Eurochemic Company, Mol, Belgium, who discussed special considerations necessitated by construction and operation of a multipurpose radiochemical processing plant in a highly populated multinational area.

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   O. Sisman

The last of a series of tests to compare real and simulated fission-product aerosols was completed during the previous report period.[1] Some analytical results are still outstanding, but a comprehensive evaluation of all tests in the series was initiated and will be completed during the next reporting period.

Reference

1. S. H. Freid, B. F. Roberts, and G. W. Parker, pp. 3–6 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for July–August 1968, USAEC Report ORNL-TM-2368, November 1968.

1.2 FAILURE MODES OF ZIRCALOY-CLAD FUEL RODS (AEC Activity 04 60 10 01 1)

P. L. Rittenhouse G. W. Parker
R. A. Lorenz M. F. Osborne

The behavior of the fuel rods in BWR's and PWR's during the design-basis loss-of-coolant accident is a matter of immediate concern. This concern is centered about the possibility that the failure behavior of the fuel cladding may alter the thermal response of the core and thereby adversely affect the accident outcome. Information needed to make judgments as to the magnitude and seriousness of the failure-mode effects is not available at present. Some work has been initiated at ORNL on the investigation of failure behavior during thermal transients. In addition, P. L. Rittenhouse of ORNL, who has been designated Coordinator of this program for the AEC, has submitted a program plan to the AEC for approval. At the present time contacts are being made with reactor vendors in order to coordinate the work called for in this program, significant portions of which are already under way in private industry.

Transient Tests of Zircaloy-Clad Fuel-Rod Clusters in TREAT

Thr TREAT experiments simulate fuel-rod failure following a loss-of-coolant accident in a PWR or a BWR. In the tests of Zircaloy-clad UO_2 fuel rods a seven-rod cluster of 27-in.-long rods will be used. The center rod will contain fission products from a 900-Mwd/MT burnup irradiation in the MTR at 14.2-kw/ft heat rating. Prepressurization with 215-psia helium (at 25°C) will simulate the released fission gas expected for a burnup of 10,000 Mwd/MT. The TREAT reactor will operate at steady power for about 17 sec so that fission heat in the seven-rod cluster will raise the UO_2 and cladding temperatures 70°C/sec from a preheated temperature of 280°C. Prepressurizing to different levels with helium will cause the rods to rupture at different temperatures.

During the tests a flowing gas mixture of 11 liters/min of steam plus 2 liters/min helium will carry fission products released from the center rod into two sequentially operated fission-product collection systems. The safety analysis for the MTR irradiation has been approved, and the experimental assembly is scheduled for insertion in the MTR on November 4. After a few weeks decay time the center rod will be gamma scanned and installed in the main experiment package for the loss-of-coolant fuel-rod rupture test in TREAT.

Before the experiment can be run in TREAT a calibration transient is required to establish the exact equivalence of reactor power and fuel-rod heatup rate. Consideration is being given to the use of a mockup experiment run at full power rather than an extra low-power transient with the full experiment. Table 1 shows a tentative arrangement of rods for a mockup experiment, which would be performed in inert atmosphere. The use of rods with two different void volumes should contribute to knowledge of any plenum volume effect on swelling.

MTR Irradiation of Instrumented UO_2 Capsules

The most important variable contributing to the volatile fission-product inventory in the void space of an operating fuel rod is the volumetric temperature, or thermal profile. For reactors such as the most recent BWR's, the peak rod may be operated at a heat rating near 18 kw/ft. This produces a center-line temperature above 2400°C.

In designing the TREAT experiment, a center-line maximum temperature of 2300°C was chosen in order to provide nearly 10% of the ^{131}I in the void space for an advantage in measuring release and distribution in the seven-rod assembly. Although the proposed A-7-SW position in the MTR was selected in the hope of providing this temperature, no direct observation of temperature is possible in the uninstrumented rod. Postfailure examination is unlikely to provide this information because of the high temperature expected from the later transient in TREAT. Therefore it will be necessary to determine the TREAT-rod center-line temperatures and axial profile indirectly by means of a duplicate irradiation with a fuel piece containing three temperature-monitored fuel capsules, such as shown in Fig. 1.1. The uninstrumented TREAT rod (experiment 1) is shown at the right and the instrumented one (experiment 2) is shown on the left.

Table 1.1. Characteristics of Fuel Rods for Mockup Experiment

Rod[a] No.	Number of Pt-Rh Thermocouples	Total Void Volume (cm^3)	Helium Gas Volume [cm^3 (STP)]	Helium Gas Pressure at 25°C (psia)	Predicted Pressure Differential at Rupture (psi)	Predicted Temperature at Rupture	
						°C	°F
Center	3	3.5	9	40	160	1031	1890
1	2	3.5	14	65	262	927	1700
2	0	7.0	28	65	262	927	1700
3	2	3.5	25	115	425	827	1520
4	0	7.0	50	115	425	827	1520
5	0	3.5	47	215	725	732	1350
6	0	7.0	94	215	725	732	1350

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

ORNL-DWG 68-1250

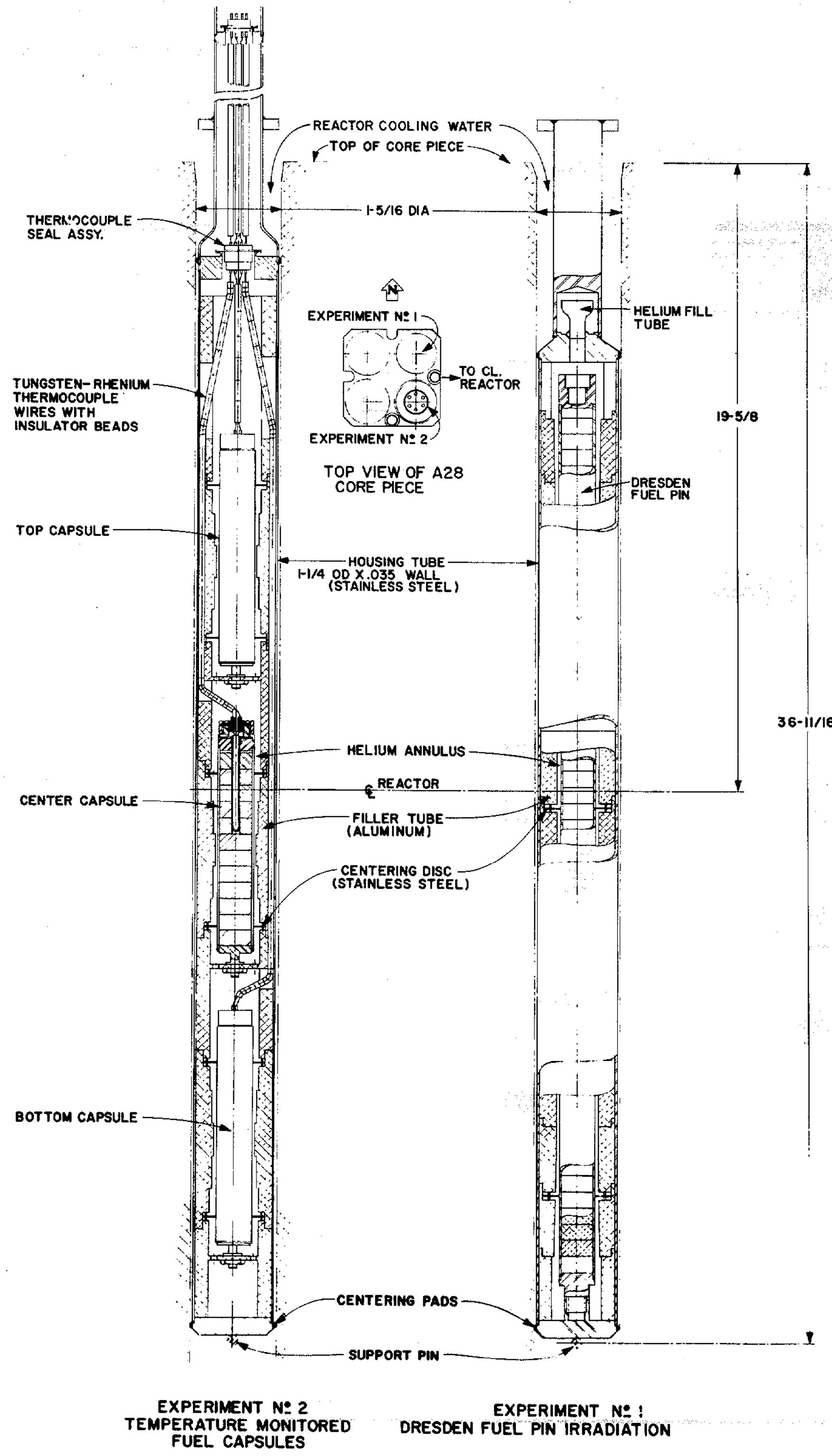

Fig. 1.1. Uninstrumented TREAT Rod (Right) and Temperature-Monitored Fuel Capsules (Left).

A novel design was required to provide direct temperature monitoring at 2300°C (center line). This included (1) a Zircaloy-2—tungsten fusion weld of the tungsten thimble to the capsule body and (2) the development of a secondary mechanical seal with a gold washer on a threaded nut and a silver-soldered ceramic lead-through for the thermocouple wires. The tungsten thermowell is made commercially by vapor-depositing tungsten on a mandrel. A third backup closing seal on the tungsten thimble is made by brazing with Ti-Be-Zr alloy to the Zircaloy body. The welded assembly and UO_2 pellets are shown in Fig. 1.2.

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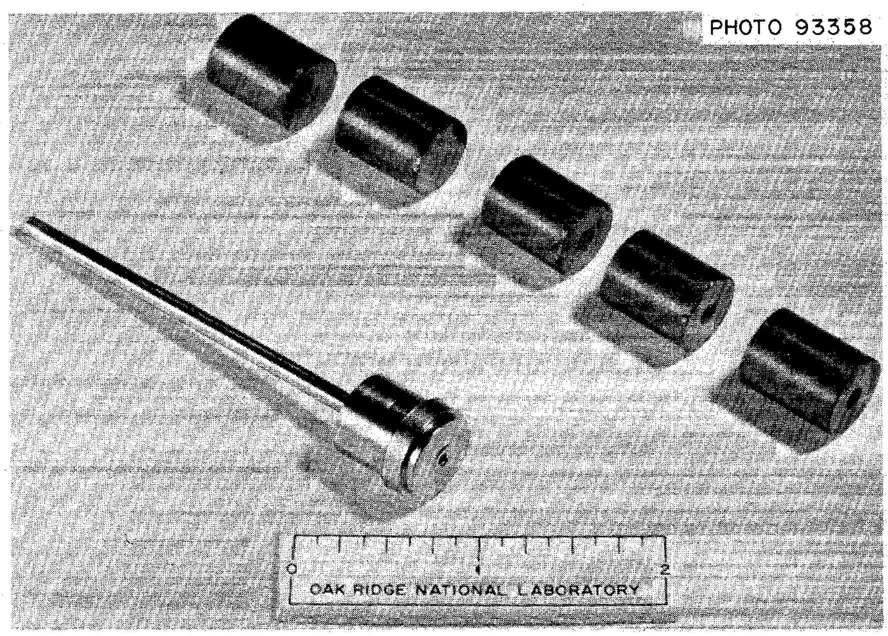

Fig. 1.2. Tungsten Thermowell Electron-Beam Welded to Zircaloy Capsule Plug and Typical UO_2 Pellets for the Instrumented Assembly.

1.3 LOFT ASSISTANCE PROGRAM
(AEC Activity 04 60 60 03 1)

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

Numerical data are being obtained that are needed 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. The several tasks in this program will culminate in evaluations in the Containment Research Installation of both the fission-product behavior and the effectiveness of engineered safety features common to LOFT and typical of currently operating reactors.

The initial scope of the LOFT assistance program at ORNL was modified as the result of recent discussions between ORNL and Phillips Petroleum Company. The range of direct support effort has been defined by three task areas:

Task A: Fission-Product Release from Unclad Fuel (GAP Inventory Study)

Task B: Fission-Product Release from Long Rods Coincident with Emergency Core Cooling

Task C: Fission-Product Transport and Containment Behavior and Removal

However, because of limited funding, not all tasks will be implemented in FY-69.

The FY-69 LOFT Support Program

In FY-69 the program will advance as follows:

Task A: No work in this area is planned for FY-69.

Task B: The CRI Long-Rod Failure Test Facility will be installed for operation inside the primary vessel. The remote facility will be made operational for FY-70. A few iodine-implant releases will be attempted as progress and findings permit.

Task C: The major portion of the funding for FY-69 is directed toward a continuation of this task; however, no direct fission-product release work is possible, since the irradiated fuel rods were to have been provided by Task B. The initial work will therefore be limited to radioiodine behavior studies.

In preparation for the studies under Task C, the CRI vessel was opened and the stainless liner removed. Reassembly with the painted liner was started. Initiation of the new radioiodine behavior studies should be accomplished by November.

Tests in the Containment Research Installation (Task C)

After the CRI containment liner is changed from stainless steel to Amercoat 66, the following experiments will be conducted:

1. Radioiodine behavior in a painted-tank environment will be studied at 230°F in an unperturbed atmosphere. A release with a relative low radioiodine concentration of 1 to 2 mg/m^3 will be conducted in CRI, without using suppression devices, and the iodine will be permitted to adsorb onto the paint during the steam-pressure decay cycle. The external recycle filters will then be operated for several hours, and the amount and form of radioiodine collected will be determined. This information will indicate the desorption processes to be encountered and the minimum reentry period for the LOFT containment vessel. Two experiments will be performed.

2. Radioiodine behavior in a painted-tank environment will be studied at 230°F after chemical-spray reduction. Immediately after radioiodine release, chemical iodine-removal pressure-suppression sprays will be operated for a practical interval, and then the recycle filter will be operated to determine the relative amount of iodine desorbed according to form and quantity. The fixing effect of the chemical spray, which will limit desorption, will thus be learned. Two experiments will be performed.

3. Plans will be made for confirmation of a new volatile radioiodine species in the CRI containment vessel. Cartan and his associates[1] at Idaho Nuclear Corporation have evidence of a new volatile species of radioiodine, hypoiodous acid (HOI), which results from the hydrolysis of elemental iodine in slightly alkaline solutions. The hydrolysis of iodine

in alkali was observed previously, but no evidence of volatility of a stable form of hypoiodous acid has been reported. The evidence given by Cartan is largely based on differential adsorption behavior in or on specific materials; for example, iodine-loaded hydrous zirconium oxide and silver surfaces. As part of the LOFT support effort, the reported adsorber-pack distribution analysis will be duplicated, and then further tests will be applied for the positive identification of the new species in the airborne radioiodine in CRI.

Reference

1. F. O. Cartan et al., Evidence for the Existence of Hypoiodous Acid as a Volatile Iodine Species Produced in Water-Air Mixtures, paper presented at the Tenth Air Cleaning Conference, New York City, Aug. 28, 1968.

1.4 LMFBR SAFETY

(AEC Activity 04 60 60 03 1)

M. H. Fontana

Preliminary planning and feasibility studies were started to determine areas of LMFBR safety research and development to which ORNL's facilities and capabilities can best be applied. This effort is being coordinated with the LMFBR program office and adheres to the program plan.

2. FILTRATION AND ADSORPTION TECHNOLOGY

2.1 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. The released aerosol particles will interact with such an atmosphere, as well as the filter system. A study is under way to determine the effect of such interactions on the aerosols.

Recent data from filtration studies of stainless steel oxide—UO_2 aerosol particles involved the determination of filtration efficiencies. In order to measure filtration efficiencies, a slightly radioactive stainless steel oxide—UO_2 smoke is generated in an electric arc, and air is circulated past the arc through a 100-liter tank. After a short period the arc loop is valved off, and the aerosol is circulated through three 1 1/4-in. disks of test filter media, in series. The filter efficiency is taken to be the ratio of the radioactivity on the first filter disk to the activity on all three disks. Each filter-efficiency value is measured four times and the average value is taken.

The data reported, Figs. 2.1 and 2.2, were taken with the aerosol at 60°C and the filter pack at a slightly higher temperature. Measurements were made with dry aerosol (Fig. 2.1) and with water-saturated aerosol (Fig. 2.2). The variable parameters were the velocity of the aerosol at the face of the filter (5 and 10 ft/min) and the source of the filter media (i.e., the commercial vendor).

There are two notable points in regard to the effects of temperature and steam (under these noncondensing conditions). One is that the penetration of dry aerosol at 60°C was slightly higher than at 25°C; that is, about 0.045 versus 0.03% at 5 ft/min (the 25°C data were reported previously[1]). The second point is that moisture (under noncondensing conditions) at 60°C increases the aerosol penetration slightly (about 0.06 versus 0.045% at 5 ft/min).

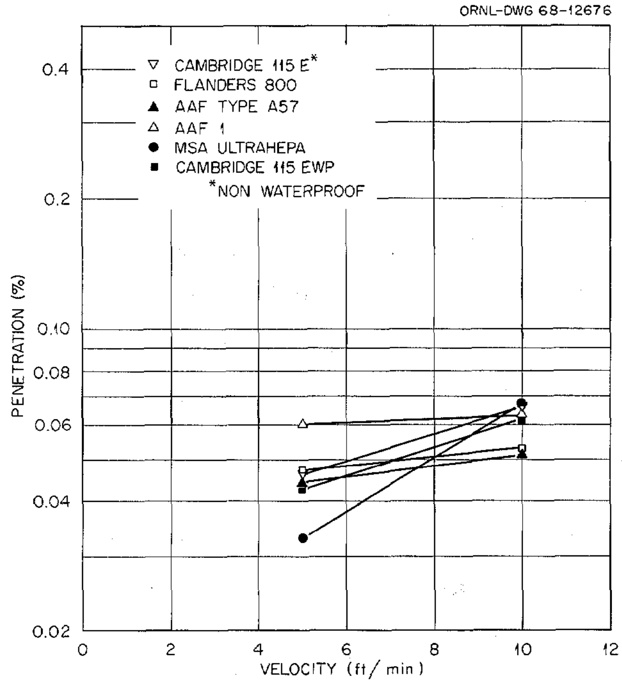

Fig. 2.1. Filter Efficiencies for Removal of Dry Stainless Steel Oxide—UO_2 Aerosol at 60°C.

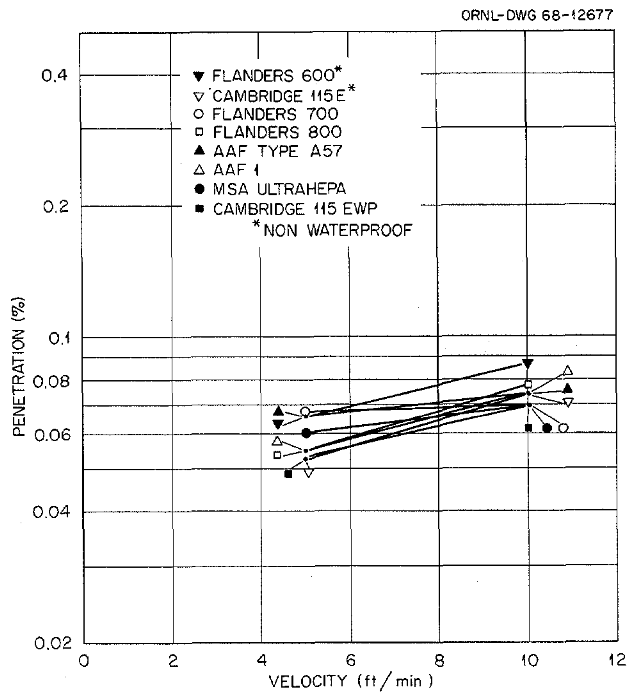

Fig. 2.2. Filter Efficiencies for Removal of Wet Stainless Steel Oxide–UO_2 Aerosol at 60°C.

Reference

1. R. E. Adams et al., p. 31 in ORNL Nuclear Safety Research and Development Bimonthly Report for May–June 1968, USAEC Report ORNL-TM-2283, Oak Ridge National Laboratory, July 30, 1968.

2.2 IGNITION OF CHARCOAL ADSORBERS BY FISSION-PRODUCT DECAY HEAT (AEC Activity 02 30 20 90)

R. E. Adams R. L. Bennett
R. P. Shields

Charcoal adsorbers for radioiodine constitute an important part of safety systems for the removal of accident-released fission products from containment volumes either during recirculation of the atmosphere or as it is being exhausted from the containment shell. During reactor operation, charcoal adsorbers are subject to loading with large quantities of fission products and, consequently, heating by decay of the trapped fission products. A program is therefore 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.

Work on the development of 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 has continued. Two important physical properties of the adsorber needed for such calculations are (1) the rate at which heat will travel through the charcoal and (2) the specific heat; both parameters are required as a function of changing temperature. Under the accident conditions being considered, the charcoal will contain a large amount of adsorbed water. It is under these conditions that values for thermal conductivity and specific heat must be assessed.

A laboratory apparatus with which to develop the desired information under the stated conditions was constructed, and initial tests were run with a material of known thermal conductivity. Packed beds of small glass beads (540 to 580 μ) were tested under static conditions at several temperatures over the range 30 to 130°C. Also, measurements of the temperature differential across a layer of the beads were made while the system temperature was being increased uniformly, at a rate of about 0.3°C/min, from room temperature to greater than 100°C. These temperature measurements correlated well with the static measurements and reflected a small change in the thermal conductivity of the beads as the temperature increased.

Such measurements are under way for activated charcoal, and a similar correlation should permit a close estimation of the more complex changes expected in the thermal conductivity of a moisture-laden charcoal adsorber as it is heated from ambient to higher temperatures by the decay heat of the adsorbed fission products.

2.3 SEPARATION OF NOBLE GAS FROM AIR WITH PERMSELECTIVE MEMBRANE (AEC Activity 04 60 80 01 1)

R. H. Rainey W. L. Carter

A process for separating radioactive xenon and krypton from other inert gases or from air by using a dimethyl silicone rubber membrane is being developed. This method of separation is based on the difference in the solubility of the gases in the membrane and the difference in the rate of transport through it. A previous bimonthly report[1] gave the results of calculations for capital cost versus concentration factor for permselective membrane plants to treat 10 scfm of an Ar-Kr-Xe (99.86-0.004-0.136 at. %) mixture. The withdrawal rates for the concentrated gas were chosen so that the ratios of product to feed concentration factors were 10, 100, and 500; in each case the decontamination factor, ratio of activity in feed stream to activity in decontaminated stream, was about 1000. Since in some practical applications of the membrane process, decontamination factors of only 10 to 100 are required, data from the above calculations and an earlier calculation[2] for a plant that processed 1300 scfm of air were used to estimate capital costs as a function of decontamination factor and processing rate (Fig. 2.3). All results are for concentrating krypton and xenon by a factor of 100 in the exit gas. Also shown on Fig. 2.3 are the estimated capital costs of a plant for absorbing the noble gases in fluorocarbon refrigerant.[3]

A situation of practical interest is that of processing the dissolver off-gases from a 5-MT/day chemical processing plant. The off-gas flow rate is 75 scfm, and the gas is nitrogen oxides (N_2—O_2) containing very small concentrations of krypton and xenon. For either the membrane or absorption plant the nitrogen oxides and iodine must be removed before the gas enters the noble-gas removal plant. For this processing rate the estimated capital costs of a permselective membrane plant are \$260,000, \$390,000, and \$560,000 for decontamination factors of 10, 100, and 816, respectively; whereas the indicated cost for the adsorption plant at a decontamination factor of 100 is about \$420,000.

In each of the above cases the capital cost of a head-end treatment for removing nitrogen oxides and iodine must be added. It is estimated

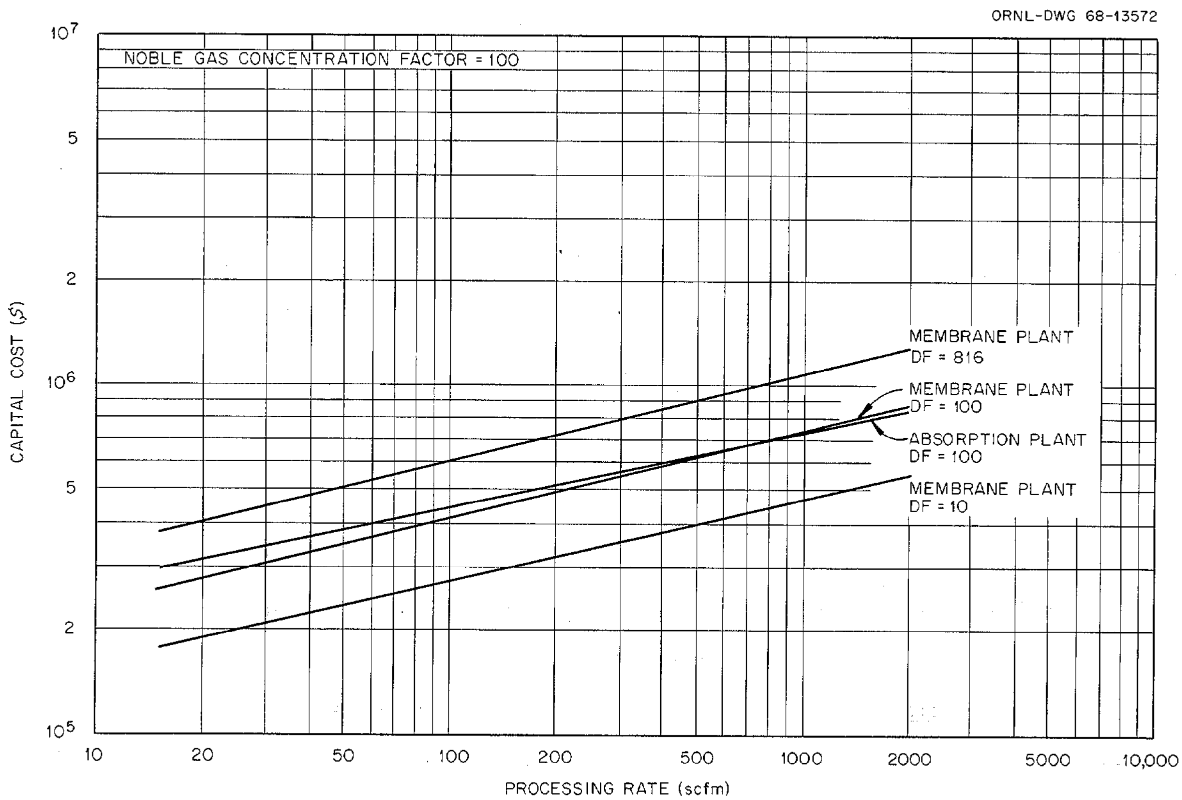

Fig. 2.3. Estimated Capital Cost for Removing Noble Gases From N_2–O_2 System.

that the cost for the facility for head-end treatment is about $400,000. Therefore, the capital cost of treating the off-gases from a 5-MT/day chemical processing plant is in the range $800,000 to $1,000,000.

An additional fact displayed by Fig. 2.3 is that the membrane and adsorption processes appear to be economically competitive. At the lower feed rates or lower decontamination factors, the membrane plant might have a lower capital cost. At higher feed rates or higher decontamination factors the adsorption plant may cost less. However, the differences in costs are within the uncertainties of the estimates.

References

1. W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for March—April 1968, Oak Ridge National Laboratory, May 30, 1968.

2. S. Blumkin et al., Preliminary Results of Diffusion Membrane Studies for the Separation of Noble Gases from Reactor Accident Atmospheres, paper presented at the 9th AEC Air Cleaning Conference, Boston, Mass., September 13—16, 1966.

3. J. R. Merriman, J. H. Pashley, and S. H. Smiley, Engineering Development of an Absorption Process for the Concentration and Collection of Krypton and Xenon, Summary of Progress Through July 1, 1967, USAEC Report K-1725, Oak Ridge Gaseous Diffusion Plant, Dec. 19, 1967.

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

C. A. Burchsted A. B. Fuller

The first draft of the manual entitled "Design and Construction of High Efficiency Air Filtration Systems for Nuclear Applications" was completed and issued to a limited distribution for review. This manual is intended to be a guide for designers and users of air-cleaning systems for nuclear reactors, radiochemical processing operations, and laboratories that require very high-efficiency air cleaning. The manual was written primarily with operating, maintenance, and hazards-control requirements in mind. Problem areas and features that often compromise air-cleaning systems are pointed out, and mechanical, structural, and layout requirements, as well as performance and limitations of major system components, are discussed in detail. Reliability and total cost relative to initial cost are stressed. Requirements for single-filter, multiple-filter, and glove-box installations are described.

A separate chapter is devoted to special problems of remotely maintained and nuclear reactor postaccident air-cleaning systems. Following final review by the AEC Division of Operational Safety and Reactor Development and Technology, the manual will be issued for general distribution as an NSIC Report. The release date of the report is uncertain at this time.

3. SPRAY AND POOL PRESSURE-SUPPRESSION TECHNOLOGY

T. H. Row, Coordinator

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

(AEC Activity 04 60 80 01 1)

R. L. Bennett   Ruth Slusher

A further examination was made of the distribution of CH_3I between water and air, and kinetic measurements of the reaction of CH_3I with potential spray solutions were initiated.

Distribution of CH_3I Between Water and Air

The distribution coefficient of CH_3I between water and air is defined by the equation

$$K_d = \frac{\text{Concentration of } CH_3I \text{ in the liquid phase}}{\text{Concentration of } CH_3I \text{ in the vapor phase}} .$$

The discrepancy between the values of K_d reported previously[1] and other values in the literature prompted a further evaluation of this distribution coefficient. Table 3.1 contains a summary of the distribution coefficients that have been reported by several workers. The large value of 8.7 reported by Patterson and Humphries[1] was obtained by the "reverse" method, which involves use of a saturated solution. This method requires that the solution be saturated and further that the solubility be known, since the total amount of CH_3I is based on this value. The values of K_d appear to decrease with smaller CH_3I concentration. For spray applications the CH_3I concentration range examined by Hasty[3] is of most interest. Accordingly, the direct technique was used to repeat the K_d measurement at the 10^{-8} moles/liter range. The concentration of CH_3I in glass bulbs was measured, water was added, the bulb was agitated sufficiently to attain equilibrium, and the concentration of CH_3I in the vapor was determined. A gas chromatograph with an electron-capture detector was used for CH_3I analysis. The K_d value of 2.98 at 25°C with a 1.5×10^{-8} *M* solution concentration agrees well with Hasty's measurements.

Table 3.1. Summary of Reported Distribution Coefficients of CH_3I Between Water and Air

Investigator	Reference	Method	CH_3I Concentration (moles/liter)	Temperature (°C)	K_d
Patterson and	1	Calculation[a]	9.9×10^{-2}	25	4.58
Humphries	1	Reverse	9.9×10^{-2}	25	8.7
Glew and	2	Direct	$2\text{–}4 \times 10^{-2}$	25	4.47–4.64
Moelwyn-Hughes	2	Direct	2×10^{-2}	29.90	3.95
Hasty	3	Direct	$0.04\text{–}12.8 \times 10^{-8}$	29.90	2.75 ± 0.15
ORNL		Direct	1.5×10^{-8}	25	2.98

[a]Calculated from vapor-pressure and solubility data.

Reaction of CH_3I with Additives in Spray Solutions

The slow reaction rate of CH_3I in aqueous spray solutions seems to be the limiting step in the overall spray removal process. Therefore a study of the reaction kinetics of CH_3I with potential spray additives was started. The technique being used is similar to that developed and used by Hasty[4] for determination of the rate constants for the reaction of CH_3I with hydrazine solutions. The procedure consists of adding CH_3I to a spray solution (contained in a reaction cell thermostated to a known temperature) and then sparging the solution with nitrogen gas. The concentration of CH_3I emerging from the cell is monitored by gas chromatography. Removal of CH_3I is dependent on two processes: the reaction rate in the solution and the sparging rate. By plotting the observed total rate constant versus the amount of CH_3I sparged and extrapolating to zero the amount of CH_3I sparged, a value of the pseudo first-order rate constant is obtained. Apparatus similar to that employed by Hasty was constructed and preliminary rate determinations were made with hydrazine solutions to check the procedure. The determination of pseudo first-order rate constants of the reaction of CH_3I with basic borate and basic borate–thiosulfate solutions will be made in the initial investigations as a function of temperature.

References

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.

2. D. N. Glew and E. A. Moelwyn-Hughes, Chemical Statics of the Methyl Halides in Water, Discussions Faraday Soc., 15: 150 (1953).

3. R. A. Hasty, Partition Coefficient of Methyl Iodide Between Vapor and Water, Can. J. Chem., 46: 1643 (1968).

4. R. A. Hasty, personal communication.

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

The investigation of mixtures of iminic structure plus a reducing agent was broadened in order to ascertain what effect such chemical composition would have in a nonsurfactant, monomeric form and to determine whether variations in molecular weight of the surfactant would introduce any advantages. In addition, the effect of Dow PEI 1000* surfactant combined with $Na_2S_2O_3$ on the removal efficiency of CH_3I was examined.

In a preceding report[1] it was suggested that a surfactant containing an iminic functional group when combined with a reducing agent, such as formaldehyde, materially improved the efficiency of CH_3I removal in a basic solution (pH 9) containing 0.3% boron in the form of borate. This surfactant (PEI 1000) also improved the CH_3I pickup in the presence of another reducing agent, $Na_2S_2O_3$. It is one of few additives stable to radiation and noncorrosive that materially increases the efficiency of 1% $Na_2S_2O_3$ in the removal of CH_3I. At the present time the performance of a 1% $Na_2S_2O_3$ solution alone is approximately a factor of 5 too slow to be considered adequate for CH_3I removal. However, it is the only additive studied so far that is capable of gettering and fixing I_2 and also has some affinity for CH_3I.

Preliminary wind-tunnel runs were made on representative mixtures in order to ascertain whether kinetic factors were present that could account for the remarkably good results obtained in the bubbler tests described in the preceding bimonthly report.

Results

As shown in Tables 3.2, 3.3, and 3.4 the addition of a reducing agent, such as formaldehyde or $Na_2S_2O_3$, leads to increased efficiency of adsorption of methyl iodide by any compound containing the iminic structure, regardless of whether or not the latter is in a monomeric form, such as

*Trademark of Dow Chemical Company.

Table 3.2. Comparison of Effects of Diethylene Triamine and Pentaethylene Hexamine on Efficiency of CH_3I Pickup

Diethylene Triamine (moles/liter)	HCHO (wt %)	Total Activity Retained in Liquid (%)	Pentaethylene Hexamine (moles/liter)	HCHO (wt %)	Total Activity Retained in Liquid (%)
0.0003	0	<1			
0.0003	0.2	1–3			
0.03	0	~3	0.03	0	13.6
0.03	0	4			
0.03	0.2	57	0.03	0.2	31.6
0.03	1.0	54			
0.3[a]	0	73 ± 5			

[a] pH = 10.15 (pH all other solutions = 9.0).

Table 3.3. Comparison of Effects of Dow PEI 6 and Dow PEI 1000 on Efficiency of CH_3I Pickup

Dow PEI 6		HCHO (wt %)	Total Activity Retained in Liquid (%)	Dow PEI 1000		HCHO (wt %)	Total Activity Retained in Liquid (%)
Moles/liter	Grams/liter			Moles/liter	Grams/liter		
0.0003	0.153	0.1	1.5	0.0003	18	0	23
0.0003	0.153	0.3	1.5	0.0003	18	0.1	52
0.03	15.3	0.3	53	0.0003	18	0.25	67
				0.001	60	0	57

Table 3.4. Effect of $Na_2S_2O_3$ on Effectiveness of Dow PEI 1000 in CH_3I Pickup

$Na_2S_2O_3$ (moles/liter)	Dow PEI 1000 (moles/liter)	HCHO (wt %)	Total Activity Retained in Liquid (%)
10^{-4}	0		~2
10^{-4}	10^{-4}		13.7
10^{-3}	0		13.6
10^{-3}	10^{-4}		28.7
10^{-2}	0		62
10^{-2}	0		64
10^{-2}	0		59
10^{-2}	10^{-4}		74
0.063 (1%)	0		90
0.063 (1%)	10^{-4}		94.3
0.063 (1%)	3×10^{-4}		95.9
0.063 (1%)	3×10^{-4}	0.25	95.9

diethylene imine (or Dow PEI 6), which is a lower molecular weight iminic surfactant form.

As for the monomeric forms, that is, $(CH_3CH_2NH)_x$, it does appear (Table 3.2) that diethylene triamine is superior in CH_3I retention efficiency to pentaethylene hexamine. The data of Table 3.3 suggest that PEI 1000 (mol. wt ≅ 60,000) at a concentration of 3×10^{-4} m with 0.25% formaldehyde is superior to Dow PEI 6 (mol. wt ≅ 500) at a concentration of 0.03 m with 0.3% formaldehyde. Both solutions contain comparable weights of surfactant. This increased efficiency of CH_3I retention is maintained in spite of the fact that the PEI 6 system actually contains a higher mass of imine surfactant, as well as a higher concentration of formaldehyde. It should be noted also that the surfactant works equally well when $Na_2S_2O_3$ is employed as the reducing agent (Table 3.4). The addition of the surfactant leads to an improvement in the efficiency of CH_3I retention, even in the case where a large concentration of $Na_2S_2O_3$ is used. This suggests that the addition of the PEI 1000 to a 1% solution of $Na_2S_2O_3$, originally proposed for I_2 gettering, may significantly improve the performance of this system with respect to CH_3I retention.

A 1% $Na_2S_2O_3$ in basic borate is about a factor of 7 too slow in CH_3I removal to make it useful in accident analysis to obtain credit for CH_3I removal.

Discussion

The bubbling tests used to obtain these data provide only a crude measurement of the performance to be expected under pilot-plant spray conditions. In the bubbler experiment there is a small bubble of gas in contact with a large volume of liquid, whereas in sprays the liquid drop is surrounded by a large volume of gas. For example, as mentioned in the preceding report,[1] bubbler tests suggested that 2×10^{-5} m of Dow PEI 1000 plus 0.25% formaldehyde would only be about 2% efficient in CH_3I pickup. Under actual pilot-plant spray conditions, however, it was found that the half-time for CH_3I pickup with this dilute solution of surface active agent was within a factor of 3 of the best results ever found with a much higher concentration of $Na_2S_2O_3$ (1 wt %).

There is, therefore, the suggestion that for a prediction of spray performance the pertinent variables are specifically the mass transfer coefficient (v_t) and the distribution coefficient (K_d). The bubbler tests provided a less-sensitive screening probe. Therefore, via the wind-tunnel single-drop studies, the K_d and v_t values were measured (Table 3.5) for the iminic monomer, formaldehyde alone, a mixture of PEI 1000 and 1% $Na_2S_2O_3$, and $(NH_4)_2S$; the latter system gives an extremely high efficiency for CH_3I pickup under bubbler conditions.

Conclusions

Table 3.5 shows that the surfactant Dow PEI 1000 combined with 1% $Na_2S_2O_3$, a reducing agent, clearly gives the highest K_d value, as well as the largest mass transfer coefficient. These two quantities appear to be indicative of the type of performance to be expected under actual spray conditions.

In summary, a mixture of reducing agent, such as $Na_2S_2O_3$ or formaldehyde, and Dow PEI 1000 added to a basic borate system offers promise of removal of CH_3I by sprays at a meaningful, practical rate.

Table 3.5. Comparison of CH_3I Mass Transfer (v_t) and Distribution (K_d) Coefficients for Representative Additive Systems at Approximately 26°C

Solution Composition	v_t (cm/sec)	K_d
4 wt % tetrapentamine	4.3×10^{-2}	4.25
1 wt % $Na_2S_2O_3$ + 0.3 wt % B (pH 9) with 1.2×10^{-5} *m* Dow PEI 1000	15.0×10^{-2}	55
1 wt % $(NH_4)_2S$	4.22×10^{-2}	8.5
0.5 wt % formaldehyde	2.95×10^{-2}	~3.0
1 wt % $Na_2S_2O_3$ at pH 9	$\sim 2 \times 10^{-2}$	~3.5

Reference

1. B. A. Soldano and W. T. Ward, pp. 57–63 in ORNL Nuclear Safety Research and Development Program Bimonthly Report for July–August 1968, USAEC Report ORNL-TM-2368, Oak Ridge National Laboratory, November 1968.

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

L. F. Parsly J. L. Wantland

The function of the Nuclear Safety Pilot Plant is to conduct engineering-scale experiments on fission-product transport and removal. In the spray-technology program four tasks have been undertaken: experimental studies of the removal of (1) iodine vapor, (2) methyl iodide, and (3) particulate aerosols and (4) theoretical work to explain the experimental results achieved in the experimental studies. An experiment (run 54) on methyl iodide removal, preparations for particle removal runs, an analytical model for spray performance, and questions that have arisen in regulatory evaluation of spray systems are discussed in this report.

NSPP Run 54 — A Methyl Iodide Removal Experiment

Run 54 was originally scheduled for evaluating the performance of a spray containing a basic borate solution with a surfactant (PEI 1000) and formaldehyde for CH_3I removal. However, the PEI had not been received by the scheduled run date. Therefore the run was used to get some data on the effect of reaction kinetics on spray performance.

Basic borate and sodium thiosulfate solution was used, but the sodium thiosulfate concentration was increased from 1 to 5%. Based on the theory described below, this increased concentration should increase the removal rate constant by a factor of 2.1. A half-life of 26.6 min was observed, and 2.1 times this is 56 min, which is to be compared with the half-lives of 53, 57, and 61 min observed in runs 44, 46, and 47 with basic borate and thiosulfate solution containing 1% sodium thiosulfate at the same temperature. The theory appears to account adequately for the effect of reaction rate.

Preparations for Particle-Removal Runs

A major part of the effort during FY-69 will be concerned with the removal of particles from simulated accident atmospheres by sprays. Accordingly, preparations are being made for particle-removal runs. For these, Zircaloy-2-clad UO_2 specimens containing cesium with ^{137}Cs tracer will be used. Fifteen samples were prepared by the ORNL Isotopes Division and delivered during the present reporting period. The ^{137}Cs gamma activity is to be followed during the run and gas samples collected on filter papers will be analyzed for uranium and cesium. Low-pressure cascade impactors[1] will be used to characterize the aerosol before and after spraying. One impactor is on hand, and a second is scheduled for delivery in November.

Analytical Model for Spray Performance

Over the past year, efforts have been made to develop an analytical model to describe spray performance. In the model, it is postulated that the spray will contain a range of drop sizes, and the approximation of a large number of size groups is used on the assumption that average properties within a group will not introduce significant error.

Log-normal size distribution is also assumed. While other size distributions, particularly the "upper limit" distribution of Mugele and Evans[2], fit better the log-normal distribution gives a reasonably good fit to the experimental data, particularly for the drops of less than mass mean diameter, which turn out to be most important. The log-normal distribution has the advantage that the same geometric standard deviation applies no matter which mean diameter is considered. (Marshall[3] cites seven possible definitions for "mean diameter.")

The drops are assumed to originate at the nozzle with a velocity having a magnitude calculated from the volumetric flow rate divided by the nozzle area occupied by liquid (which is taken to be 36% of the total area for hollow-cone nozzles and 100% of the total area for solid-cone nozzles). An initial angle with the vertical equal to half the included angle of the spray cone is assumed, and based on the initial velocity a Reynolds number, a drag coefficient, and horizontal and vertical accelerations are calculated. The initial conditions are assumed to apply for a time interval sufficient for the vertical component of velocity to change 1%. The position and velocity of the drop are calculated for the end of this time interval, as well as the gas-film mass transfer coefficient and the product of the transfer coefficient and time interval. The calculation is then repeated with the new conditions. Iteration is continued in this way until the drop hits the wall or floor of the containment vessel or attains terminal velocity. The terminal velocity is identified by a change in sign of the vertical acceleration, so this technique does not give the "true" terminal velocity, but a value which approximates it within 1%. Since the drag coefficients possibly have greater uncertainty, this approximation is probably justified.

If the drops attain terminal velocity, they are assumed to complete their fall at constant velocity. The average mass transfer coefficient, fall time, and terminal velocity (if attained) are calculated and stored.

The iteration technique with a fixed percentage velocity change is a new variation in the program. Originally a fixed time interval was used, but an elaborate procedure was required for shortening it during the first part of the trajectory. The present technique achieves convergence in a straightforward manner and seems to give a very efficient program.

The solution for simultaneous absorption and chemical reaction described in the previous bimonthly report was also incorporated into the overall spray analysis program.[4] This program enables us to calculate the fraction of saturation for each size group and for the overall spray. With fast reaction, the fraction of saturation can be more than 1.0. The material balance equation, which gives the fraction of the initial CH_3I concentration in the sprayed volume at the end of the spray period as

$$\frac{C}{C_0} = \exp\left(-\frac{EHL}{V}t\right) ,$$

where

E = fraction of drop saturation,

H = gas-liquid distribution coefficient,

L = spray rate, liters/hr,

V = sprayed volume, liters,

t = total spraying time, hr,

can then be used to calculate the removal of methyl iodide by the spray.

When this is applied to removal of methyl iodide in the NSPP, it is found that for $k = 1.0\ \text{sec}^{-1}$ (typical reaction rate constant of basic borate and thiosulfate system at 120°C), $T = 120°\text{C}$, $H = 0.25$, $L = 2498$ liters/hr, and $V = 38{,}300$ liters, the program gives

$$E = 1.77\ ,$$
$$EHL/V = 0.0288\ ,$$
$$t_{1/2} = 24\ \text{hr}\ .$$

A half-life of approximately 0.75 hr is indicated, so the calculated removal rate is low by a factor of 32. The probable explanation is that the effect of the wall film has not been considered. The calculations show that the combined surface area of all the spray drops is 6.7×10^4 cm^2. The wetted surface of the vessel wall and floor, starting from the point where it was calculated that the largest drops hit the wall, is $2.1 \times 10^5\ \text{cm}^2$, so there is three times as much surface available on the wall as in the drops. The residence time for the solution on the wall is significantly longer than that of the spray drops, and the combined effect could well account for the discrepancy.

The previously reported[5] sensitivity analysis on the effect of σ_G was repeated by using, however, the mass-mean and Sauter mean diameters rather than the number mean. The results are given in Tables 3.6 and 3.7. They are quite opposite to those previously obtained for the number mean

Table 3.6. Effect of Varying σ_G for Drops Having a Mass-Mean Diameter of 1200 μ

σ_G	Half-Life (sec)
1.0	84.4
1.5	56.4
1.8	34.3
2.0	23.3

Table 3.7. Effect of Varying σ_G for Drops Having a Sauter Mean Diameter of 1150 μ

σ_G	Half-Life (sec)
1.0	62.7
1.2	60.8
1.5	51.2
1.8	40.1
2.0	33.5

diameter. For the number mean, the half-life increases with increasing σ_G. This is reasonable if it is remembered that the means are related by

$$\ln d_v = \ln d_{ng} + 3 \ln^2 \sigma_G$$

and

$$\ln d_{32} = \ln d_{ng} + 2.5 \ln^2 \sigma_G ,$$

where

d_v = value mean of mass mean diameter,
d_{ng} = number mean of mass mean diameter,
σ_G = geometric standard deviation of number mean diameter G,
d_{32} = Sauter mean diameter.

Tables 3.6 and 3.7 show that for a given mass mean or Sauter mean, a larger σ_G implies a smaller number mean.

Questions Arising in Regulatory Reviews

Concern has been expressed by members of Atomic Safety and Licensing Boards and others regarding spray systems that can be relieved on the basis of information now available. The first of these is the question of why sprays are preferred to filters. The answer is that the advantage of sprays is that they remove iodine from the containment atmosphere faster than filter-adsorber systems. A second concern is the choice of the spray solution, and it has been determined that several different reagents are effective in removing iodine and that the choice of any one represents the applicant's judgment as to which is best.

A third is the point that spray towers have only limited capability and are therefore seldom used by the chemical industry. This is quite true, but the applications are entirely different. Typically the chemical industry uses scrubbers to remove a constituent from a flowing gas stream, and handbooks indicate a maximum mass flow of 500 $lb/ft^2 \cdot hr$, which corresponds to a linear velocity of about 2 ft/sec. If a 60-ft-high tower and a 1-ft/sec linear gas velocity are postulated, the contact time is 1 min. If such a tower removed 50% of the constituent of interest, it would be considered relatively inefficient. In the containment building the gas is held in the building, and the contact time is unlimited. If the same half-time of 1 min were achieved, the dose reduction factor would be given by

$$DRF = \frac{\lambda t}{1 - e^{-\lambda t}} ,$$

where $\lambda = 0.693 \times 60 = 41.6\ hr^{-1}$. Thus the dose reduction factor would be 83.2. In other words the long contact time would compensate for the low efficiency. The containment building application of sprays is analogous to a recirculating filter system, whereas the chemical plant application is analogous to a once-through system.

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3.4 RADIATION AND THERMAL STABILITY OF SPRAY SOLUTIONS (AEC Activity 04 60 80 01 1)

H. E. Zittel

The possible use of Dow PEI 1000 in conjunction with the proposed basic spray solutions to scavenge CH_3I more efficiently has been under study by others on this project.* Since there exists a reasonable probability that the combination may prove to be useful, a short study of the radiation stability of this compound and its sister compounds (PEI 6 and PEI 18) was carried out. In all cases the polyethylenimine was dissolved in 0.1 *N* NaOH—3000 ppm B solution to give the approximate concentration of PEI desired. Since the exact nature of the radiolysis products was not known and no exact method for following the radiolytic degradation was available, an indirect method was used.

A preliminary investigation showed that none of the PEI series exhibit spectral absorbance either in the visible or ultraviolet portion of the spectrum from 700 to 230 nanometers (nm). At approximately 230 nm the solution becomes optically opaque and limits the ultraviolet range to wave lengths above that figure. It was found that upon irradiation all members of the series produced a first degradation product with maximum absorbance at approximately 330 nm. The height of the absorbance peak at approximately 330 nm proved to be a linear function of total radiation dose. Therefore, the growth of this peak was followed as the radiation dose was increased, and the radiation dose required to bring about a change in the slope of the wave height versus radiation dose plot was considered the endpoint. While it is acknowledged that the PEI undoubtedly undergoes further degradation after this point, it was felt that this indirect measurement would give a positive answer to whether the PEI series was usable from a radiation stability viewpoint. No effort was made to identify the degradation products other than to test for radiolytic NH_3, which was not found. Table 3.8 gives the results obtained.

*See Section 3.2 for a discussion of this surfactant.

Table 3.8. Results of Tests of PEI Radiation Stability

PEI Type	Quantity[a] (mg/ml)	Dose to Endpoint (r)	Apparent G
1000 (mol. wt, ≅60,000)	0.1	1×10^5	<0.01
	0.6	4×10^5	<0.01
	1.0	7×10^5	<0.01
	6.0	6×10^6	<0.01
18 (mol. wt, ≅1700)	2.0	1×10^6	~1
6 (mol. wt, ≅500)	1.6	8×10^5	~3

[a]In all cases the PEI was dissolved in 0.1 *N* NaOH—3000 ppm B; all solutions were open to air.

The data indicate that a 1% PEI 1000 solution would require approximately 10^7 r for the first radiolytic degradation to be completed. This information is given only to show that the solution is stable enough to be used under the radiation conditions expected. No statement is made as to the effect of the radiolytic degradation on the ability of the additive to scavenge CH_3I. This can be ascertained only by actual test, since there is a finite probability that the radiation may enhance the reactivity toward CH_3I.

Several other studies are being carried out to determine the effect of radiation on other aspects of the spray program. A series of metal coupons, representative of the various metals to be found in the reactor environs, are under study in irradiated solutions of both alkaline-borate-thiosulfate and alkaline-borate solutions. The effect of the radiation on both the metals and solutions is to be determined in order to decide whether any radiation effect is present when the metals are exposed to the solution as opposed to when they are irradiated separately.

The effect of temperature on radiolytic H_2 generation is being studied on both the above solutions. A more complete gas-phase analysis is being carried out than was performed previously.

3.5 SPRAY SOLUTION CORROSION STUDIES
(AEC Activity 04 60 80 01 1)

A. L. Bacarella J. C. Griess

In a previous report[1] the equipment for testing the corrosion resistance of a variety of materials in iodine-absorbing sprays and totally immersed in the same solution was described and the results of the first test run were given. Additional tests have now been conducted and the results are presented below.

Test in Basic Borate—Thiosulfate Solution at 100°C

The first attempt to test a variety of materials, including aluminum alloys, in a solution containing 0.064 m $Na_2S_2O_3$, 0.15 m NaOH, and 0.28 m H_3BO_3 at 100°C resulted in plugging the spray nozzle after one day. The aluminum alloys caused reduction of a part of the thiosulfate to sulfide, which reacted with the copper corrosion products to form coarse particles of presumably copper sulfide. These particles lodged in the nozzle and completely stopped the flow. The spray nozzle is a full-cone center jet, Model No. 3C, manufactured by Spray Engineering Co. The orifice in the nozzle is 0.093 in. In the absence of aluminum specimens, sulfide ions were not formed, and the copper-containing corrosion products had a gelatinous character. They passed through the nozzle without reducing the flow rate. Therefore, all aluminum specimens were removed from the sample array after the first day, and the test was continued at 100°C for an additional 144 hr. After cumulative times of 24 and 72 hr, the specimens were removed, scrubbed with a soft brush, weighed, and returned to the system. At each interruption, new solution of the same composition as the original was used and an atmosphere of air (room temperature) filled the gas phase of the system. At the conclusion of the test, specimens that had corrosion products on their surfaces were cathodically descaled.

The weight losses of the aluminum alloy specimens that were in the system for only the first 24 hr are listed in Table 3.9. The attack of the aluminum alloys was very similar to that observed in the same solution in the absence of thiosulfate[1] and indicates that the thiosulfate had no

Table 3.9. Corrosion of Aluminum Alloys in a Solution Containing 0.064 *m* $Na_2S_2O_3$, 0.15 *m* NaOH, and 0.28 *m* H_3BO_3 at 100°C During 24 hr Exposure

Aluminum Alloy	Average Weight Loss[a] (mg/cm^2)	
	In Spray	In Solution
1100	58	40
3003	51	35
5052	1[b]	1
6061	37	34

[a]A weight loss of 6.86 mg/cm^2 corresponds to a penetration of 1 mil.

[b]One of four specimens lost 30 mg/cm^2 and is not included in the average.

major effect on the corrosion of these alloys. With the exception of one specimen, the 5052 alloy corroded at a much lower rate than the other aluminum alloys, as was also observed in the absence of thiosulfate. The data indicate that aluminum alloys generally have unacceptably high corrosion rates in the alkaline borate solutions at 100°C.

The average weight losses of other materials in the same solution are given in Table 3.10. In addition to those materials listed, specimens of Inconel 600 and 718, Monel 400, Zircaloy-2, and types 304 and 316 stainless steel were also exposed. Visual inspection, as well as the weight changes of these specimens, showed attack to be nil both in the spray and solution regions.

As indicated previously, aluminum specimens were in the system during the first 24-hr period and some sulfide was produced. However comparison of the data for the rest of the run generally indicates no particularly significant effect due to the presence of sulfide. Copper and the cupronickels corroded uniformly, with copper undergoing the greatest attack. The density of copper and its alloys is 8.9 g/cm^3, so a uniform weight loss of 22.5 mg/cm^2 corresponds to a penetration of 1 mil. Thus copper corroded at a rate of 210 mpy in the spray and only 11 mpy in the solution.

Table 3.10. Corrosion of Materials in a Solution Containing $0.064\ m\ Na_2S_2O_3$, $0.15\ m$ NaOH, and $0.28\ m\ H_3BO_3$ at 100°C

Material	Cumulative-Average Weight Loss (mg/cm^2)					
	In 24 hr		In 72 hr		In 168 hr	
	In Spray	In Solution	In Spray	In Solution	In spray	In Solution
Copper	19	0.1	52	3.3	91	4.7
90-10 cupronickel	0.02	0.06	0.4	0.2	0.8	0.5
70-30 cupronickel	+0.01	0.06	0	0	0.08	0.4
Steel, A 210	+0.3	0.09	+1.1	+0.02	3.7	2.7
Steel, A 108	+0.2	0.04	+1.0	+0.04	4.2	2.9

The cupronickel alloys corroded at rates of less than 2 mpy both in the spray and solution. Both steel alloys developed random patches of rust under which localized attack occurred; pits up to 5 mils deep were present on all specimens.

Test in Basic Borate-Thiosulfate Solution at 140°C

A second one-day run was also completed with the thiosulfate solution. The composition of the solution was the same as in the previous test, but in this case the temperature was 140°C. The alloys exposed were the same, except that all aluminum specimens were excluded.

The only materials that showed attack were those containing copper. The weight and appearance changes of the other specimens were insignificant. The weight losses for the copper-containing alloys are shown in Table 3.11.

Corrosion in all cases was uniform and was greatest for pure copper and least for Monel (66% Ni, 32% Cu). The data clearly show that the more nickel alloyed with copper, the more resistant is the alloy to the thiosulfate solution. As with the 100°C test the attack was much greater in the spray than in the solution. During the 24-hr exposure the average penetrations in the spray were 3.4, 0.53, and 0.036 mils for copper, 90-10 cupronickel, and 70-30 cupronickel, respectively; on the solution-exposed

Table 3.11. Corrosion of Materials in a Solution Containing 0.064 *m* $Na_2S_2O_3$, 0.15 *m* NaOH, and 0.28 *m* H_3BO_3 at 140°C During 24 hr Exposure

Material	Average Weight Loss (mg/cm^2)	
	In Spray	In Solution
Copper	76	1.1
90-10 cupronickel	12	0.9
70-30 cupronickel	0.8	0.2
Monel 400	0.01	0.01

specimens the average penetrations were 0.05, 0.04, and 0.009, respectively. In contrast to the 100°C test specimens, the carbon steel specimens developed only a few areas of light rust and showed no evidence of localized attack.

Periodic sampling of the solution from the spray loop was done to determine the stability of the thiosulfate at 100 and 140°C. The concentrations of thiosulfate in solution at different times during the runs are listed in Table 3.12.

Table 3.12. Concentration of Sodium Thiosulfate in the Spray Solution at 100°C and at 140°C

At 100°C		At 140°C	
Time (hr)	Concentration (*M*)	Time (hr)	Concentration (*M*)
0	0.064	0	0.063
1.6	0.062	0.6	0.062
18.3	0.056	3.8	0.057
26	0.055	19.8	0.053
90	0.054	24.0	0.053
96	0.054		

The results are in good agreement with those obtained by Zittel[2] in an all-glass system under similar conditions and indicate that metallic surfaces or insoluble copper corrosion products suspended in the solution have no appreciable effect on thiosulfate stability. Both at 100 and 140°C there was no decrease in the boron content of the solutions during the test periods.

Test in Alkaline-Borate Solution at 140°C

A run at 140°C was also made in the alkaline-borate solution in the absence of thiosulfate. The solution contained 0.15 *m* NaOH and 0.28 *m* H_3BO_3, and an atmosphere of air filled the gas phase. The materials exposed were the same as in the 140°C run with thiosulfate, except that two specimens of 5052 aluminum were included. The weight losses experienced by the specimens are listed in Table 3.13. Those not listed were unchanged as a result of the exposure.

Table 3.13. Corrosion of Materials in a Solution Containing 0.15 *m* NaOH and 0.28 *m* H_3BO_3 at 140°C During 24 hr Exposure

Material	Average Weight Loss (mg/cm^2)	
	In Spray	In Solution
5052 aluminum	13	7.6
Copper	0.44	0.11
90-10 cupronickel	0.08	0.3
70-30 cupronickel	0.02	0
Type 304 stainless steel	0.05	0.03
Type 316 stainless steel	0.04	0.02

The data indicate no serious attack of any of the materials, except 5052 aluminum. With this alloy, the attack is relatively low considering the alkaline nature of the solution and the temperature. All the other materials underwent corrosion penetration of less than 0.02 mil, which is insignificant in a practical spray system that would be at this high temperature for only a short time. No evidence of localized attack was found on any alloy, and both the pH of the solution and the boron content remained unchanged during the test.

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3.6 PRESSURE-SUPPRESSION EXPERIMENTS
(AEC Activity 04 60 80 01 1)

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

Analytical Modeling of Pressure-Suppression Systems

Experiments to obtain information to permit analytical modeling of energy dissipation and fission-product transport in power reactor pressure-suppression pools continued. A relatively large quantity of data was obtained from experiments involving the injection of dry steam into both degassed water and water saturated with air. These experiments extend the previously reported data to include steam-injection tube diameters to slightly greater than 2 in. and steam mass flow rates up to 4.6×10^5 lb/hr·ft^2. (Previously reported upper limits were: 1-in.-diam injection tube and $\sim 3 \times 10^5$ lb/hr·ft^2 steam mass flow rate.) Quantitative analysis of the data is largely incomplete, but semiquantitative evaluation indicates that the previously reported correlations are valid for the larger injection tube sizes in the range of steam flow rates previously studied.

These results are encouraging, since if the quantitative evaluation confirms this, the validity of the correlation will have been established for a range of injection-tube diameters varying by a factor of 5 and a range of flow areas varying by a factor of 25. This will considerably reinforce the possibility that the correlations will also be valid for full-size steam-injection systems.

Experiments with Elemental Iodine to Determine Iodine Retention in Pressure-Suppression Pools

An investigation of iodine retention in simulated pressure-suppression pools was initiated. It appears that some meaningful tests involving iodine concentrations of the magnitude postulated for design-basis accidents can be performed satisfactorily with the use of elemental iodine. However, only a few of the many possible conditions to be investigated can be studied at this time. It is therefore desirable to perform those tests that will provide the most needed and useful information.

Conditions resulting from a design-basis accident encompass at least the following:

1. Mixtures of steam, saturated water, and air in mass flow quantities ranging from approximately 2×10^5, 4×10^5, and 6×10^3 lb/hr·ft^2, respectively, during the initial blowdown (first 2 to 10 sec) to very low flow rates of almost pure steam after several minutes.

2. Item 1 implies varied flow rates, as well as varied mixture ratios of the three components, steam, water, and air.

3. Fission products resulting from fuel ruptures can be introduced into the blowdown mixture at any time, and thus a broad spectrum of fission-product source conditions is possible. Since the probability distribution (time dependent) for fuel ruptures is unknown, it is necessary to establish the disposition of fission products as a function of source conditions.

4. The pressure-suppression pool represents another spectrum of conditions. The pool water will initially be cool and saturated with air and will presumably contain no fission products. During blowdown, the water temperature will increase to a presently uncertain value. In addition, the fission-product concentration will increase to a maximum value, which in the case of iodine is postulated to be in the range 2×10^{-2} to 10^{-3} g/m^3 of water.

5. With such varied source and receiver conditions (items 3 and 4 above), fission products can be deposited in the water from the blowdown mixture in some cases or removed from the suppression-pool water by the noncondensibles in the blowdown mixture (or air bubbles generated by the steam and saturated liquid, since the water is saturated with air) in other cases. This obviously implies a wide range of deposition rates, as well as the direction of the deposition.

Although this is not a complete statement of the problem, it is evident from the above statements that a few scattered experiments will not provide sufficient information to permit confident extrapolation or interpolation to real and varied systems. It is therefore planned to perform a series of tests that can be evaluated for a particular set of conditions. These conditions will be selected to conform as closely as possible to the design-basis accident postulates and still permit evaluation of data. Initial tests will be the following:

1. Saturated steam containing elemental iodine injected into water saturated with air but initially containing no iodine will be tested at several blowdown rates.

2. Saturated steam with approximately 2 wt % air and containing elemental iodine injected into water saturated with air but initially containing no iodine will be tested at the same blowdown rates as above.

3. Tests 1 and 2 will be repeated, but the suppression-pool water will have an initial concentration of elemental iodine.

4. Iodine concentrations will be varied in both the blowdown mixtures and pool water.

Sampling and analysis will be performed to determine the resulting iodine concentrations in the pool water and in the atmosphere above the pool water for each of the above conditions. This should provide sufficient information to permit evaluation of at least the conditions stated in 1 and 2 above. These experiments are in progress.

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

(AEC Activity 04 60 80 01 1)

M. Siegler D. P. Siegwarth

The purpose of the 1/10,000-scale model in the pool-suppression technology program is to study the absorption of methyl iodide in pressure-suppression pools under simulated loss-of-coolant accident conditions. The facility is not designed to handle radioactivity. Hence methyl iodide will be used, since its concentration may be followed sufficiently accurately with a gas chromatograph. Behavior of molecular iodine will be deduced from the tests described in the previous section. During the present reporting period the major effort was devoted to studying the capability of the existing General Electric computer codes as applied to the fission-product trapping experiment, the development of a computer code to predict the suppression-pool circulation, and completing the fabrication and assembly of the experimental equipment. Debugging of some of the auxiliary equipment was initiated, and the previously discussed shakedown experiments should begin in November.[1]

Model and Experiment Design

The pressure and temperature response of the reactor vessel, drywell, and pressure-suppression pool as a function of different sets of input parameters is being calculated with existing GE computer codes in order to obtain a better understanding of how these parameters affect blowdown behavior. The measured mass flow rates in recent coolant blowdown studies conducted at Battelle-Northwest were lower than those predicted by an analytical model based on F. J. Moody's two phase flow model.[2] To correct for this, a nozzle coefficient of 0.648 was used to reduce the effective area of the nozzle. Figure 3.1 shows a comparison between the measured pressure transient for a pressure vessel with initially saturated water and the pressure predicted by the GE computer code (also based on

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

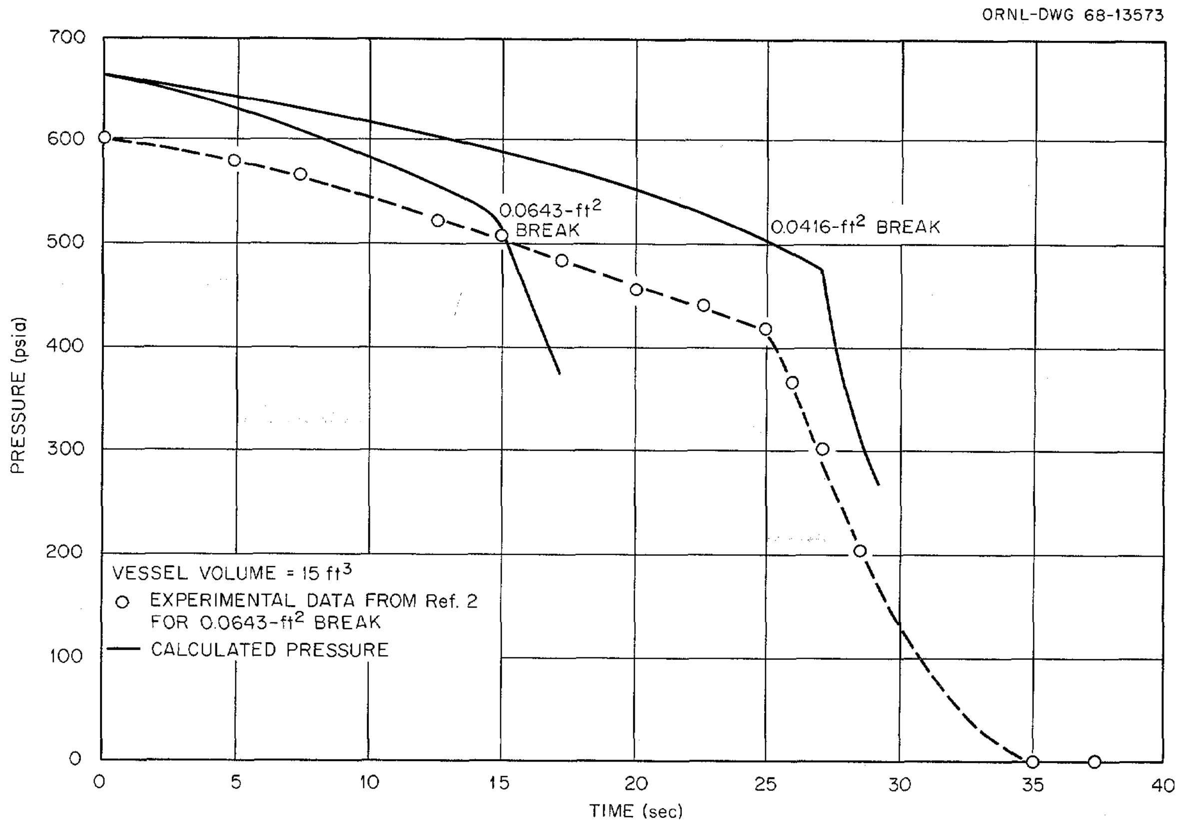

Fig. 3.1. Pressure Decay of Simulated Reactor Vessel.

Moody's model). The conditions for the computed and experimental results are similar, except for the initial pressure. Since the GE code prediction of a shorter blowdown time affects the steam flow into the pressure-suppression pool, plans to investigate this apparent discrepancy in blowdown times are being formulated for the shakedown experiments.

Circulation of the water in a pressure-suppression pool could have an effect on fission-product absorption during a loss-of-coolant accident. Work was started to develop a model to estimate the circulation intensity caused in an initially stagnant cylindrical liquid pool by a jet issuing from a downwardly directed pipe at the center of the pool. If the eddies are considered laminar and axisymmetric, the vorticity transport equation given by Macagno and Hung[3] can be used to describe the circulation. Substitution of the Stokes stream function ψ and dimensionless parameters of Bye[4] into the vorticity equation gives

$$\frac{\partial \zeta}{\partial t} + \frac{1}{r}\left(\frac{\partial \psi}{\partial z}\frac{\partial \zeta}{\partial r} - \frac{\partial \psi}{\partial r}\frac{\partial \zeta}{\partial z} - \frac{\zeta}{r}\frac{\partial \zeta}{\partial z}\right) = B\left(\frac{\partial^2 \zeta}{\partial r^2} + \frac{1}{r}\frac{\partial \zeta}{\partial r} + \frac{\partial^2 \zeta}{\partial z^2} - \frac{\partial \zeta}{r^2}\right), \qquad (1)$$

where r, z, t, and B denote the dimensionless radial distance from the axis, the dimensionless distance from the surface of the pool, the dimensionless time, and the parameter $\rho\nu^2/a^2\tau$, respectively. The liquid density, ρ; kinematic viscosity, ν; pool radius, a; and shear stress, τ, which arises from the assumption that the interaction between the jet and the pool liquid is a constant shear stress at the jet-pool interface, are all properties of the system. The vorticity is defined as

$$\zeta = \frac{1}{r}\left(\frac{\partial^2 \psi}{\partial z^2} - \frac{1}{r}\frac{\partial \psi}{\partial r} + \frac{\partial^2 \psi}{\partial r^2}\right). \qquad (2)$$

The boundary conditions are obtained by assuming that (1) the jet is a cylinder of the same radius as the pipe and of finite length, (2) the no-slip condition applies at the solid surfaces, and (3) there is zero shear at the pool surface. The boundary-condition parameters are r_1, z_1, z_2-z_1, and L/a, which are the dimensionless pipe radius, depth of the pipe below the liquid surface, jet length, and liquid depth, respectively, and are represented by

$$
\begin{aligned}
&z = 0 && \psi = \zeta = 0 \\
&z = L/a && \psi = 0,\ \zeta = \frac{1}{r}\frac{\partial^2\psi}{\partial z^2} \\
&r = r_1,\ z \leq z_1 && \psi = 0,\ \zeta = \frac{1}{r}\frac{\partial^2\psi}{\partial r^2} \\
&\qquad z_1 \leq z \leq z_2 && \psi = 0,\ \zeta = 1 \\
&\qquad z_2 \leq z \leq L/a && \psi = \zeta = 0 \\
&r = 1 && \psi = 0,\ \zeta = \frac{1}{r}\frac{\partial^2\psi}{\partial r^2}
\end{aligned}
\tag{3}
$$

A computer code that uses a finite-difference method was written to solve Eqs. (1) and (2) with boundary conditions (3) and is now being debugged. The use of finite-difference methods requires the calculation of ψ and ζ at the intersection of mesh lines, which are superimposed on the liquid pool, at each discrete time step. A finite-difference analog of Eq. (1) is used to compute the new values of ζ from the known values ψ and ζ at the old time step. The values of ψ may then be obtained from the finite-difference analog of Eq. (2). The finite-difference equations were derived by substituting the ordinary central difference approximations for the derivatives.[5] The computer code uses an extension of the "alternating-direction-implicit" (A.D.I.) method proposed by Peaceman and Rachford[6] to solve Eq. (1). The A.D.I. computational scheme as applied to the solution of Eq. (1) has been discussed in the literature.[7,8] Equation (2) is solved numerically by "successive over-relaxation," which is also known as Liebmann's extrapolated method.[9,10]

After computing the new values of ψ and ζ at the interior mesh points, the new values of ζ on the boundary are calculated from the boundary conditions. Taylor's series expansions for ψ in the vicinity of the wall are used to obtain an approximation of the second derivative.[8] To avoid instabilities, a smoothing factor is used so that the new values of ζ on the boundary are calculated as weighted averages of the old values and values obtained by applying the boundary conditions.[7]

If the values of ψ and ζ are known at a given time, the steps in the computational procedure required to advance one time step are as follows:

1. Solve the finite-difference approximation of Eq. (1) to give the new values of ζ at the interior mesh points.

2. Use the new values of ζ to solve the finite-difference approximation of Eq. (2) and obtain the new values of ψ at the interior mesh points.

3. Calculate the new values of ζ on the solid boundaries from the weighted average of the old values and the values given by the Taylor's series approximation of the boundary conditions.

In order to advance more than one time step the whole process is repeated.

Experimental Equipment

The heaters, which will be used to counteract heat losses so that the model will simulate the prototype blowdown behavior, were bonded to the exterior of the drywell vessel with a silicone adhesive. Several individual heater strips were utilized to provide a total heating capacity of 10 kw. The drywell temperature will be monitored with 15 Chromel-Alumel thermocouples attached to the vessel with thermal-conducting cement.

Two inches of polyurethane foam was applied to the outside of the drywell and pressure-suppression vessels to serve as thermal insulation. A protective coating of PPG Industries No. 6801 paint and a finish coating of No. 6802 Hypalon were applied over the foam.

The connecting piping is being field fabricated with the vessels in place. The piping layout design was submitted for bid. Installation and welding are scheduled to be completed by October 28.

A test program was set up to determine the effect of possible pressure-suppression pool additives on the Carbo-Zinc No. 11 coating. Carbon steel coupons coated with Carbo-Zinc No. 11 were exposed to air-saturated water, an aqueous solution with 550 ppb boron as sodium tetraborate, and three sodium hydroxide solutions (pH = 9.4, 10, and 12). After 45 days of continuous exposure, the coupons in the sodium hydroxide solutions and air-saturated water showed signs of zinc hydroxide formation, while the borate solution caused no visible degradation of the coating. There were no indications of steel corrosion.

Sample Analysis

The Varian model 204-2C gas chromatograph was reinstalled. The methyl iodide calibration work will be resumed as soon as the chromatograph has been functionally checked out.

Literature Survey

The literature survey is being continued. Documents pertinent to fission-product trapping in suppression pools were abstracted for inclusion in the Nuclear Safety Information Center abstracts.

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4. SAFETY STUDIES FOR HTGR

4.1 IN-PILE STUDIES OF REACTIONS OF GRAPHITE WITH STEAM (AEC Activity 04 60 10 01 1)

C. M. Blood
J. W. Gooch, Jr.
H. J. deNordwall
B. F. Roberts
S. H. Freid
O. Sisman
B. K. Annis
A. P. Malinauskas

Steam-graphite reactions in HTGR Accidents are being studied through in-pile experiments supported by small-scale laboratory investigations.

In-Pile Experiments

A series of in-pile experiments is planned to confirm predictions made from quantitative out-of-pile experiments and experiments concerned with the radiolytic reaction. The principal mode and the rate of oxidation of a model fuel element in steam will be determined. The model element will be similar to that used by Blood and Overholser[1] in their laboratory studies, except that the binder of the fuel stick will be labeled with ^{14}C and thermocouple pockets will be provided in the graphite wall. The principal differences between laboratory experiments and this series are that a steep radial temperature gradient can be generated in the fuel stick and a high radiation field will be present.

The in-pile reaction furnace to be used is shown in Fig. 4.1. Helium containing any desired fraction of steam and other gases can be passed over the fuel element. Reaction is confined to the cylindrical faces by alumina end caps. After reaction, the reactants will be analyzed almost continuously for H_2, CO, CO_2, CH_4, and H_2O with a mass spectrometer. The rate of fuel breakage, or more properly the rate of ^{235}U exposure, will be monitored with a single-channel analyzer measuring ^{88}Kr release.

The experimental conditions chosen will bracket the operating conditions of all presently conceived high-temperature gas-cooled reactors. Design is almost complete and construction of the gas-handling system will begin soon.

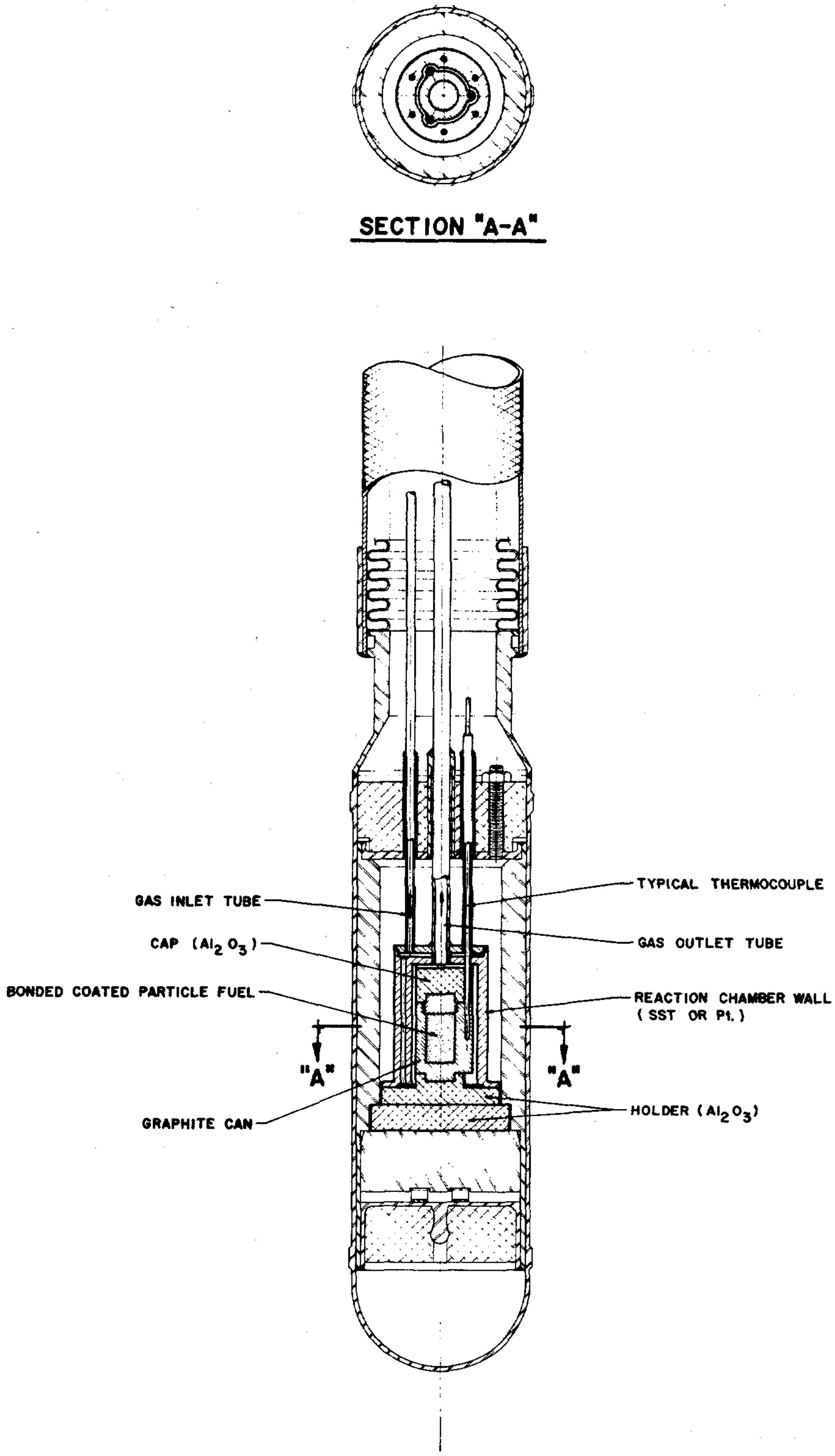

Fig. 4.1. Reaction Furnace for In-Pile HTGR Nuclear Safety Experiments.

Supporting Studies

Laboratory-scale investigations of the steam-carbon reaction comprise a single man-year effort having a dual purpose: (1) to supply appropriate information, as required, for the interpretation of data of the same nature to be obtained from in-pile experiments[2] and (2) to clarify the mechanisms responsible for the observed rate of attack of graphite by steam. A well-known but poorly understood aspect of steam-graphite reaction studies is the variability of the reaction rate with time (or extent of reaction; i.e., "burnoff"). As an example, Fig. 4.2 is typical of results obtained in this connection. The amount of graphite removed from a tubular specimen as the result of steam attack is plotted as a function of time. These data were taken at 1100°C, and the helium-steam mixture used contained 0.033 mole fraction steam. The last datum point represents 5.6% removal of the original amount of graphite. The noteworthy feature of the plot is the increase in slope (a measure of reaction rate) with time. This type of variation generally continues until after about 5 to 10% of the sample has been removed, after which the reaction rate is reported to be reasonably constant.

A number of factors have been proposed to account for this behavior. The most prominent of these is that the apparent acceleration of the reaction is due to significant modifications in internal geometry; that is, the equivalent radii of the pores in the graphite change markedly in the early stages of the reaction experiment. To investigate this proposition and the attendant possibility of correlating reaction rate data on this basis, a method has been sought whereby parameters characteristic of the internal geometry that affect gas transport can be monitored over the course of the reaction.

The logical method of characterization involves nonreactive gas migration, particularly since the associated theoretical aspects are well established.[3,4] Accordingly, a gaseous permeability and counterdiffusion apparatus was designed and constructed and is currently in operation. The apparatus will be employed for purposes of monitoring, by gas transport, changes in internal geometry as a result of reaction. Fortunately, at present, a rather large number of specimens is on hand

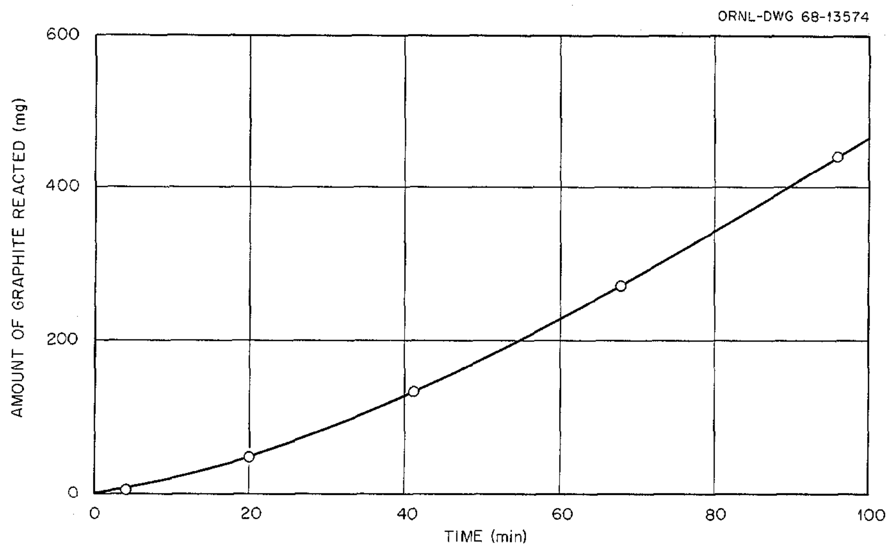

Fig. 4.2. Typical Results of Steam-Graphite Reaction Studies Showing Amount of Graphite Reacted as a Function of Time. These data were obtained with a needle-coke material (EGCR moderator graphite) at 1100°C and 1 atm pressure in a flowing stream of helium that initially contained 0.033 mole fraction steam. The final point corresponds to removal of 5.6% of the original amount of graphite.

for which reaction data like those shown in Fig. 4.2 are available. Moreover, unreacted portions of these samples have also been preserved, and thus the initial and final transport characteristics of these samples can be investigated. This work is currently in progress.

The testing phase of the apparatus included an examination of the possibility of sealing a small notch in one end of a tubular specimen without altering the overall transport characteristics of the material. This became necessary because all available samples were notched to designate influent and effluent ends in the reaction studies. The notches were successfully and simply sealed with epoxy resin, as indicated in Table 4.1. Helium permeability and helium-nitrogen counterdiffusion experiments were conducted with an ATJ graphite whose dimensions nominally duplicate those of the samples of interest. The sample was then removed from the apparatus, notched at one end, and the notch filled with resin. After the resin had cured, the flow experiments were repeated. The data in Table 4.1 likewise demonstrate that the measurements are of good reproducibility. (A discussion of the relevance of the parameters cited to the internal geometry of the system is given in Refs. 3 and 4.)

The effect of steam-graphite reaction as reflected in the transport parameters is presented in Table 4.2. These data compare the characteristic parameters of a needle-coke material (EGCR moderator graphite) essentially before and after reaction with steam. (The reaction data are in fact those displayed in Fig. 4.2.) The increase in the transport characteristics, as anticipated, is unmistakable, but in view of the scant amount of data, only qualitative observation is possible.

Another cause for the apparent increase in reaction rate with time involves a transient condition (in reality the entire process is transient in the sense that the boundary conditions are time dependent). In a simplified way this refers to the condition that prevails at the early stages of the experiment when the "foreign" gas in the graphite pores is being replaced by the helium-steam (and reaction products) mixture. Preliminary estimates indicate that this type of transient passes fairly rapidly; but to test the assumption and the model from which the result was derived requires a more closely controlled reaction study than those performed to date. For this purpose the reaction apparatus is being

Table 4.1. Reproducibility of Method for Determination of Gas Transport Characteristics of Graphite Specimens Employed in Studies of Steam-Graphite Reaction

Parameter	Original ATJ Specimen[a]	Notched Specimen[a]
D_{HeK}, Knudsen diffusion coefficient, cm^2/sec	0.141 ± 0.003	0.144 ± 0.004
B_0/η_{He}, modified viscous flow coefficient, $cm^2/atm \cdot sec$	0.274 ± 0.002	0.275 ± 0.002
(pD_{HeN_2}),[b] effective ordinary diffusion coefficient, $atm \cdot cm^2/sec$	$(2.76 \pm 0.12) \times 10^{-3}$	$(2.87 \pm 0.06) \times 10^{-3}$

[a]All data refer to a temperature of 25°C.

[b]The porosity-tortuosity factor (ϵ/q), which is a measure of the internal geometry from the standpoint of gas transport, is evaluated through the relation $(pD_{12}) = (\epsilon/q)\,(p\mathcal{D}_{12})$, where p is the pressure and the quantity written as $\mathcal{D}_{12}$ is the diffusion coefficient in free space. The appropriate value of $\mathcal{D}_{12}$ can be determined as outlined in Ref. 5.

Table 4.2. Effect of Steam-Carbon Reaction on Transport Characteristics of EGCR Moderator Graphite

Parameter	Unreacted[a] Graphite	Reacted[a,b] Graphite
D_{HeK}, cm^2/sec	1.02 ± 0.15	1.77 ± 0.37
B_0/η_{He}, $cm^2/atm \cdot sec$	8.53 ± 0.09	10.47 ± 0.23
(pD_{HeN_2}), $atm \cdot cm^2/sec$	$(4.41 \pm 0.17) \times 10^{-3}$	$(7.42 \pm 0.22) \times 10^{-3}$

[a]All data refer to a temperature of 25°C.

[b]After 5.6% of the graphite had undergone reaction (see text).

modified, and a series of reaction and characterization experiments is being designed. This activity is being performed in collaboration with Professor M. N. Ozisik.

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4.2 FISSION-GAS RELEASE FROM COATED PARTICLES UNDER ACCIDENT CONDITIONS

(AEC Activity 04 60 10 01 1)

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

An experiment (B9-40) was performed in the B9 facility of the ORR to determine the rate of failure of coated particles in a typical HTGR-design bonded bed as a function of temperatures that would be expected in a blocked-channel incident in an operating HTGR. The fuel particles in this test were 212-μ-diam uranium carbide with a 43-μ buffer layer, a 22-μ silicon carbide barrier, a 69-μ isotropic layer, and a very thin outer anisotropic layer. A typical unirradiated particle is shown in Fig. 4.3.

These particles were irradiated for 2526 hr (13.3% burnup) at 1350°C, and then the temperature was increased in 100°C steps to 1650°C, then to 1700°C, and returned to 1350°C between steps. The fission-gas release remained constant during the 2526 hr at 1350°C. The time at temperature and the fission-gas release rate are shown in Fig. 4.4. Except for the 1450°C period, equilibrium was not attained at any temperature except the final 1700°C period. The 1350°C period between each high-temperature period did give an equilibrium value that corresponds to a fraction of failed coated particles. The mode of failure has not yet been determined.

Based on the fission gas released from bare uranium carbide particles irradiated in capsule B9-36, the fission gas released from capsule B9-40 at 1350°C, after being irradiated at 1700°C, represented about 500 failed coatings, or 10% of the particles in the bonded bed. Analyses of the experiment are still in progress.

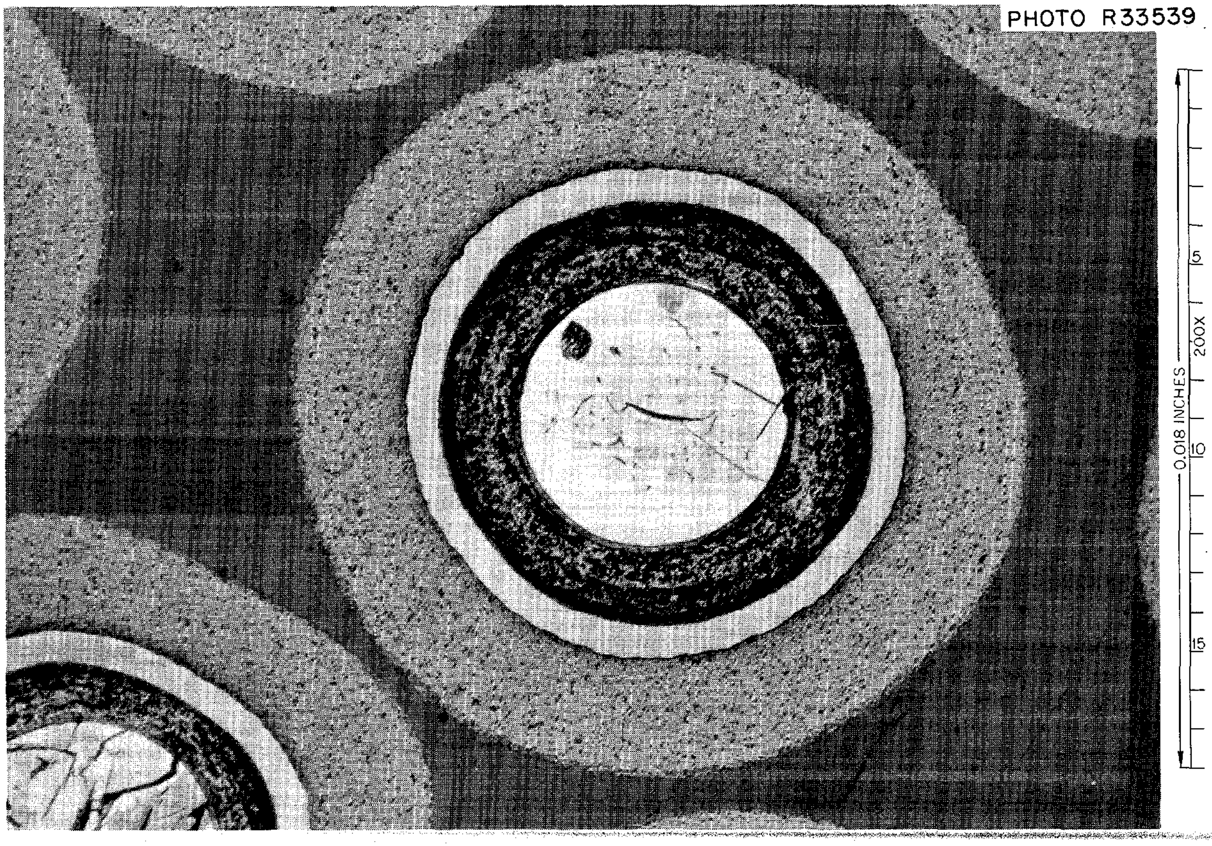

Fig. 4.3. Unirradiated Pyrolytic-Carbon and Silicon-Carbide-Coated Thorium-Uranium Carbide Particles from Batch OR-793. 200×

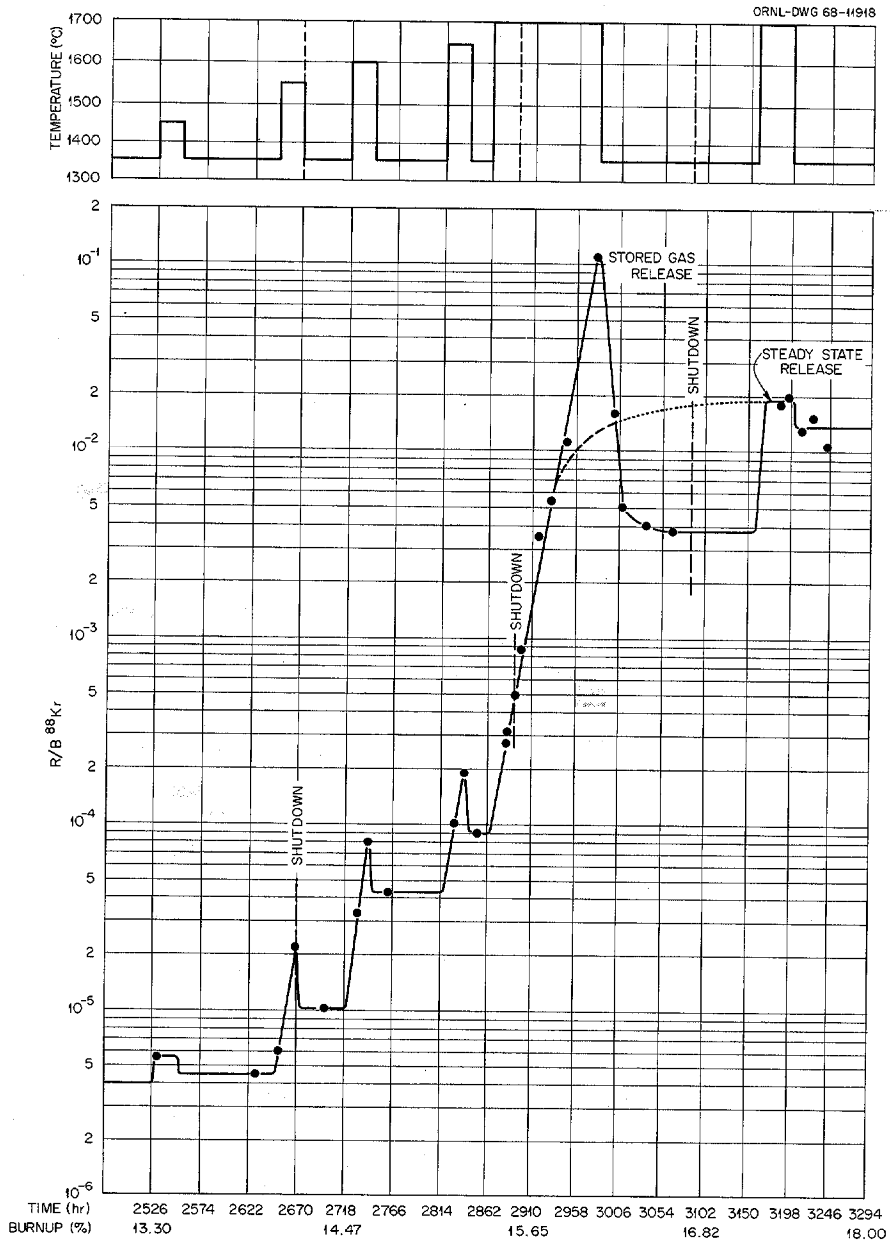

Fig. 4.4. Fission-Gas Release During Irradiation of a Bonded Bed Containing Pyrolytic-Carbon-Coated Uranium Carbide Particles; Experiment B9-40.

4.3 EXPERIMENTAL TEST OF THE FREVAP-8 CODE FOR CALCULATING METAL FISSION-PRODUCT RELEASE FROM HTGR FUEL ELEMENTS

(AEC Activity 04 60 10 01 1)

H. J. deNordwall   V. H. Pierce*   L. R. Zumwalt†

Analyses of experiments performed in the HTGR development program were made jointly by GGA, ORNL, and a consultant.† Some results are reported here, since they are important to HTGR safety.

In the FREVAP-8 calculation[1] it is assumed for simplicity that the migration of fission-product metals, such as strontium and barium, through the fuel-free graphite that separates HTGR fuel from its helium coolant can be represented by steady-state diffusion equations. The external boundary condition contains the metal adsorption isotherms and mass transfer coefficients derived from known heat transfer correlations. These and other more detailed assumptions are tested by comparing observed and calculated releases from experimental fuel elements irradiated in the Pluto loop at Harwell for the Dragon Project[2,3] and in the General Atomic (GAIL) loop in GETR at Vallecitos.[4]

Evidence that a steady state had been reached in the two Pluto experiments is shown by the forms of observed radial concentration gradients in their 1.3-mm fuel-free zones (Fig. 4.5). The configurations of the Pluto and GAIL elements are shown in Fig. 4.6. Similar examination of the GAIL element (Fig. 4.7) indicated clearly that steady state had not been achieved for the longer lived strontium isotopes in the thicker graphite between fuel and coolant.[4]

Release rates for the fuels and diffusion coefficients used in the calculations were derived from unpublished measurements made on components of these and other experiments, both in and out of pile. The adsorption isotherms used were those determined at Gulf General Atomic for a nuclear-grade (TS-688) graphite.[5]

Table 4.3 describes the experiments, and Table 4.4 gives the results of comparisons between experimental and calculated data. Analyses of the

*Gulf General Atomic, Inc.

†L. R. Zumwalt, North Carolina State University, Consultant.

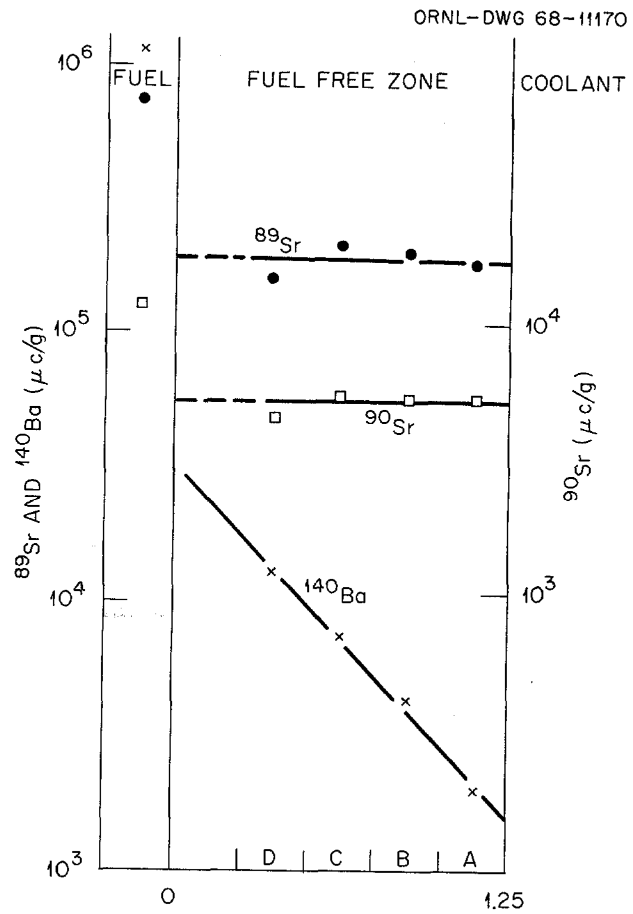

Fig. 4.5. Strontium and Barium Concentration Profiles in Pluto 15 Fuel-Free Zone at 1120°C.

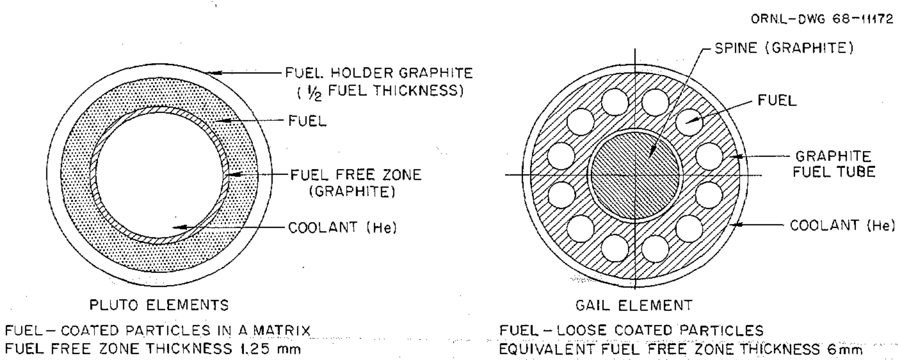

Fig. 4.6. Experimental Fuel Element Configurations.

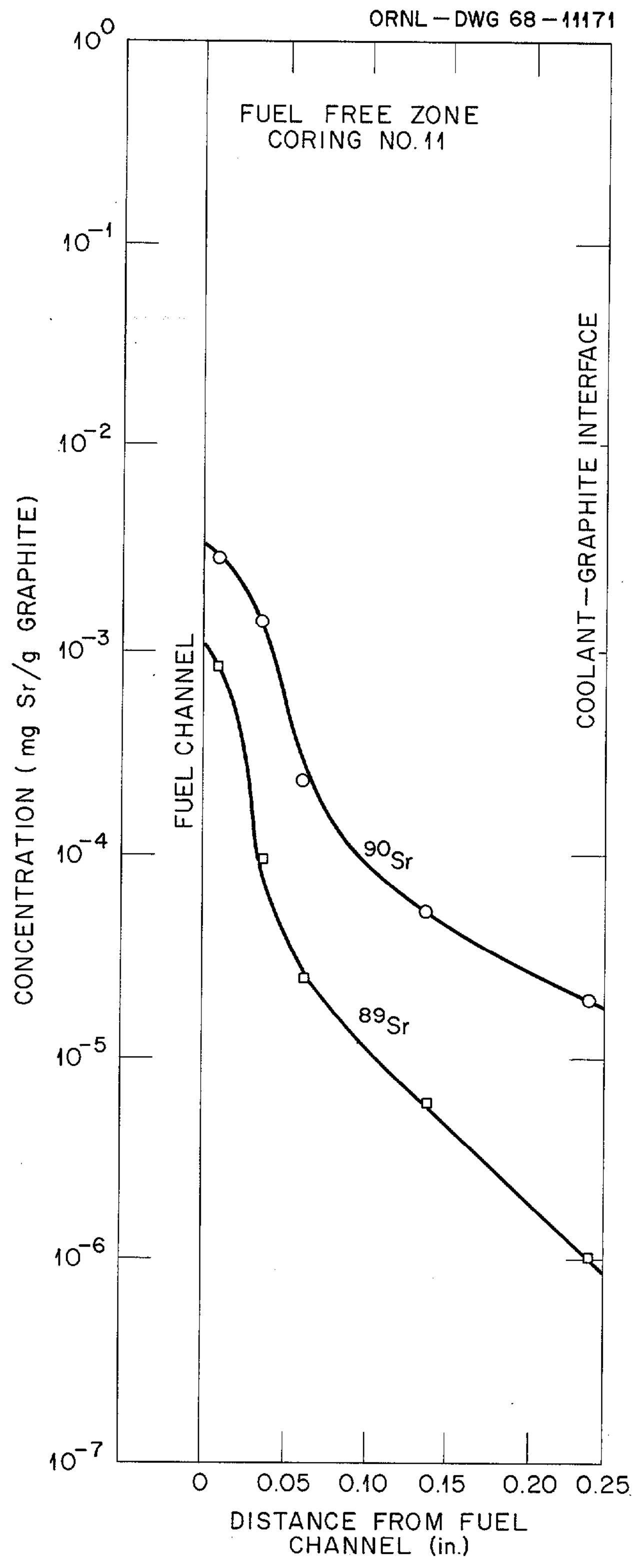

Fig. 4.7. Strontium Concentration Profiles in GAIL 4 Graphite at 950–900°C.

Table 4.3. Description of In-Pile Loop Experiments[a]

Experiment	Fuel Coating	Irradiation Time (days)	Peak Fuel Temperature (°C)	Peak Surface Temperature (°C)	Fuel-Free Zone Thickness (mm)	Coolant Flow Rate (g/sec)
Pluto-8	PyC[b]/SiC/PyC	173	1400	1350	1.25	18
Pluto-15	PyC/PyC	162	1370	1320	1.25	18
GAIL-4	PyC/PyC	400	1450	1100	≤6.35	35

[a] L^2/D for a Pluto element is more than 1/23 that for GAIL 4; hence a Pluto element is that much closer to steady state. Alternatively, a Pluto element represents the behavior of an element with a 6-mm fuel-free zone after more than 10 years irradiation, other things being equal.

[b] PyC indicates pyrolytic carbon.

Table 4.4. Comparison of Observed and FREVAP Calculated Strontium and Barium Activity Release for Pluto-8, Pluto-15, and GAIL-4 Experiments

Experiment	Nuclide and Release Location	Activity Release (curies)	
		Experimental Release	Release Based on FREVAP
Pluto-8	^{90}Sr — to FFZ[a]	2.7×10^{-2}	8.4×10^{-3}
	^{90}Sr — to coolant	7.3×10^{-3}	6.9×10^{-3}
	^{89}Sr — to FFZ	1.2	3.4×10^{-1}
	^{89}Sr — to coolant	3.2×10^{-1}	2.7×10^{-1}
	^{140}Ba — to FFZ	5.2×10^{-2}	3.4×10^{-2}
	^{140}Ba — to coolant	2.2×10^{-3}	1.8×10^{-5}
Pluto-15	^{90}Sr — to FFZ	Not determined	7.0×10^{-1}
	^{90}Sr — to coolant	Not determined	6.5×10^{-1}
	^{89}Sr — to FFZ	4.4×10^{1}	3.6×10^{1}
	^{89}Sr — to coolant	3.5×10^{1}	3.3×10^{1}
	^{140}Ba — to FFZ	4.8	9.2
	^{140}Ba — to coolant	5×10^{-2}	5.0
GAIL-4	^{90}Sr — to FFZ	4.7	6.4×10^{1}[b]
	^{90}Sr — to coolant	2.8×10^{-2}	5.2
	^{140}Ba — to coolant	$(3 \text{ to } 12) \times 10^{-5}$	5.8×10^{-11}

[a]FFZ is fuel-free zone (to FFZ signifies the total quantity entering zone).

[b]Based on a conservative (high) estimate of "release constants." A calculated release of 2.4 curies is obtained with release data from in-pile experiments.

coolant gas showed that the rates of release of ^{89}Kr, ^{90}Kr, and ^{140}Xe were insufficient to account for the ^{89}Sr, ^{90}Sr, and ^{140}Ba found in the Pluto coolant circuits. The large differences between observed and calculated releases of ^{140}Ba are ascribed to uncertainties in the barium adsorption isotherm.

The FREVAP-8 calculation for the GAIL element yielded strontium releases very much higher than those observed. Here, neglect of transient diffusion for strontium resulted in a calculated strontium release more than 100 times that observed. For the short-lived ^{140}Ba the situation is reversed because the calculation predicts that effectively all the

^{140}Ba will decay during its passage through the thick fuel tube, as is in fact found experimentally, whereas in reality there is a small release of ^{140}Ba as ^{140}Xe. Confirmation that ^{140}Xe release can account for observed ^{140}Ba has been obtained from GAIL and other experiments with similar thick fuel tubes.

The estimates of release from HTGR fuel elements were in general quite conservative. Methods of calculation that take into account transient behavior are recommended so that economic penalties from overdesign or the overestimation of hazards associated with ^{90}Sr in the coolant will be avoided.

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5. PRESSURE VESSEL AND PIPING TECHNOLOGY

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 effort for investigating the effects of flaws, material inhomogeneities, and discontinuities on the structural safety of the pressure vessels of water-cooled nuclear reactor electric power plants. The investigation is motivated by the increased thickness (up to 12 in.) of the newest vessels and the economic incentive to place the large power plants, up to 1300 Mw(e), that require such vessels near densely populated and industrial areas.

The effect of section thickness on the fracture behavior of large reactor vessels is being explored by using a systematic progression of specimen sizes, starting with standard laboratory specimens and going up to full plate thickness. The applicability of fracture mechanics with respect to section thickness and temperature level will be investigated. The HSST program will culminate in a series of simulated service tests, including fracture tests of vessels of up to full size and thickness. Also, quantitative engineering codes and standards will be developed for nuclear pressure vessels.

Program Administration

Final contractual arrangements were made with Southwest Research Institute to perform a feasibility study on crack preparation and testing of the large tensile specimens of the simulated-service test effort, with TRW Systems to investigate the gross strain concept of fracture behavior, with Materials Research Laboratory to investigate strain rate and crack-arrest effects on fracture toughness, and with Brown University for a three-dimensional elastic-plastic analysis of flawed structures. Where required, program plates were machined and provided to the investigators. A 40-ton plate of ASTM A 543 material was rolled, and ultrasonic examination is being conducted prior to heat treatment. A 6-in.-thick 30-in.-long electroslag weld was obtained.

The second semiannual program report was published,[1] and contributions to the third report were solicited and are being edited. Specifications for the first intermediate vessel of the simulated-service test series are being finalized. Technical discussions are being held in several fields prior to formal contractual negotiations.

Material Inspection and Control

Work is continuing on the investigation of a large ultrasonic indication in the first program plate. The indication is identified by the arrows on the upper right insert of Fig. 5.1, which shows the bottom ingot end of the plate. A section (approximately a 1-ft cube) was removed from the plate, and after careful ultrasonic examination the top and bottom 4 in. were removed and examined again ultrasonically. The 4-in.-thick section is currently being sliced perpendicular to the surface. The first slice, approximately 2 in. thick, was taken from the end portion of the section, as indicated by the arrow. A picture of the first slice is shown in Fig. 5.1, together with a 6 × insert of the most significant flaw observed. The surface seen is that of the cut made by the first slice. The observed flaw (hole) is about 1/4 in. deep, with facets not completely laminar. Additional cuts are being made through larger portions of the indication. Metallographic and chemical analysis studies have been undertaken.

Plans are currently being made for storing all program test data both in ledgers and on computers. A computer program for data storage and retrieval is available.

Material Characterization

Test data from Program plates 01 and 02 (12-in.-thick ASTM A 533, grade B, class 1) have been compared and found to be almost identical. Plate 03 also compares quite well, except for surface properties. Preliminary data from plate 04 (ASTM A 533, grade B, class 2) indicate a degrading of properties at the higher yield level (approximately 80,000 psi).

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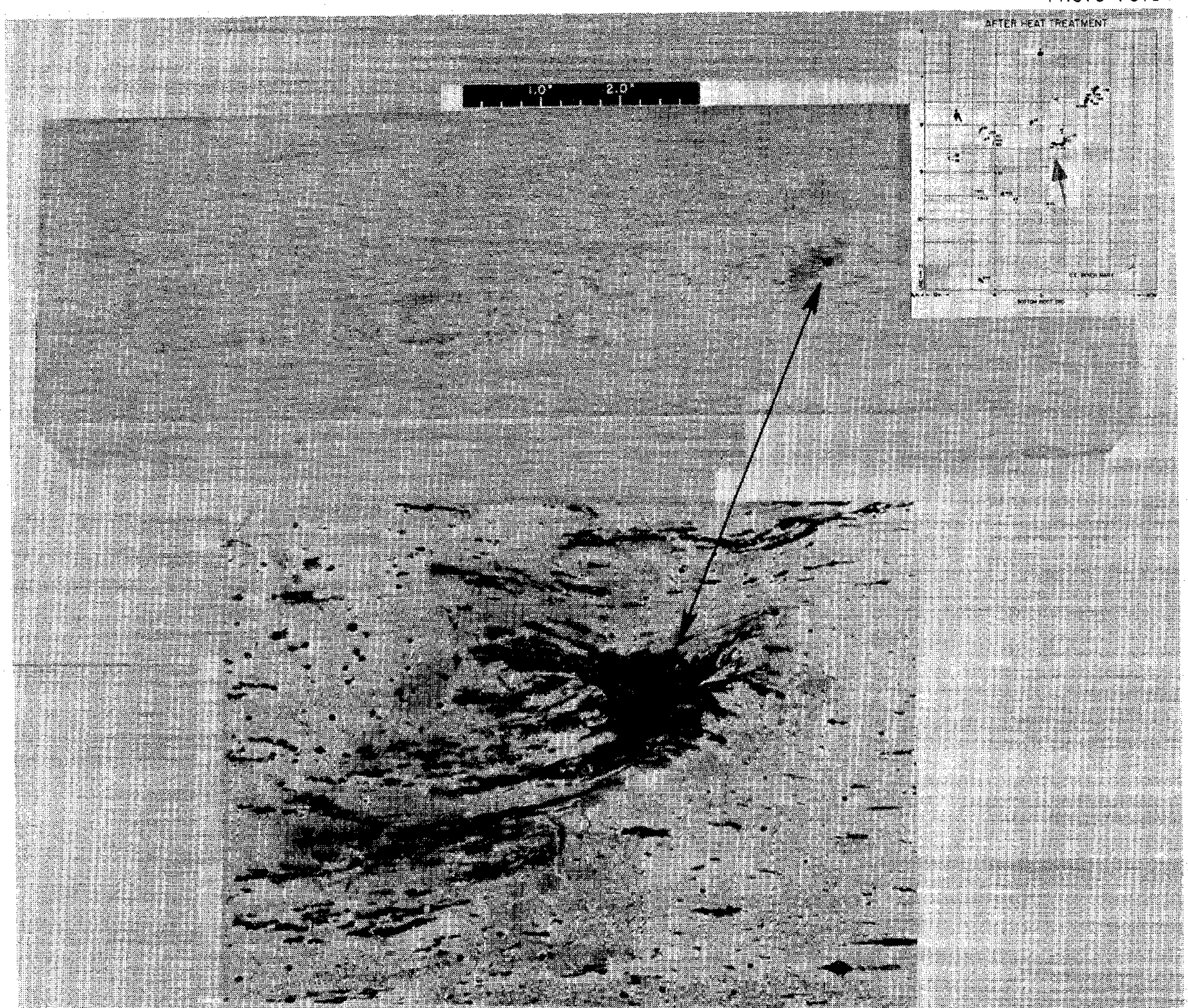

Fig. 5.2. Macrograph of a Material Discontinuity Observed in HSST Program Plate 01.

Variability in Plate, Heat-Affected-Zones, and Weld

A comprehensive investigation is being made of a submerged-arc weld for 12-in.-thick program plate. Initially Charpy V-notch and tensile tests will be made to determine the homogeneity through the weld. Following this the heat-affected zone will be identified metallographically and will be carefully examined for low-toughness material. Specimens are currently being machined.

Transition-Temperature Investigations

At Martin Marietta Company all brittle welds have been placed in the four sizes of P2 drop-weight specimens. Initial slow bend tests will be followed by the drop-weight tests.

Size-effect studies of dynamic-tear (DT) specimens are proceeding at the U.S. Naval Research Laboratory. A summary of data obtained thus far is given in Fig. 5.2, in which the 5/8-in. data are plotted from surface (curve A) to center (curves B through G) in sequence. The data from the three sizes of specimens indicate a significant transition over the same narrow temperature range (ignoring curve A).

Fracture Mechanics Investigations

Several new fracture mechanics activities have been initiated, as reported above under Program Administration. Under the cooperative program at Westinghouse Electric Corporation, valid fracture toughness data were obtained up to 50°F for a 10 T compact specimen. Values for K_{Ic} to over 125,000 psi$\cdot$in.$^{\frac{1}{2}}$ were obtained. These data support rapid transition behavior in the temperature range, as indicated in Fig. 5.2.

Fatigue and Crack Propagation

The program description of the fatigue and crack-propagation task was reviewed, and priorities were established prior to work initiation.

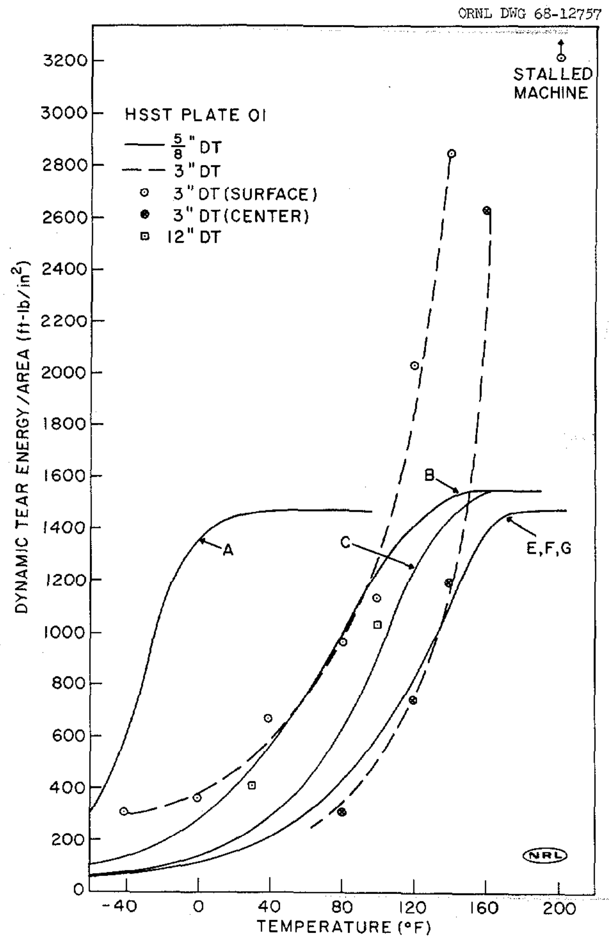

Fig. 5.2. A Comparison of 5/8-, 3-, and 12-in. Dynamic Tear Fracture Energies from HSST Plate 01 in the Transition Region.

Irradiation Effects

The phase I activity of the irradiation effects studies was initiated at Pacific Northwest Laboratory and Oak Ridge National Laboratory. The upper limit of exposure was reset to 8×10^{19} neutrons/cm^2 at 550°F. Additional research in this area was outlined and is currently being reviewed.

Complex Stress State

The activities under the complex stress state task were initiated (1) to determine the applicability of plane-strain fracture-toughness K_{Ic} values to more complex situations, such as near nozzle penetrations, (2) to determine the behavior of naturally occurring flaws as contrasted to fabricated flaws, (3) to evaluate the various test specimens proposed for use in the simulated-service task, (4) to develop methods for determining residual stresses and strains in complex, perhaps welded, structures.

Periodic Prooftesting and Warm Prestressing

A preliminary draft of conclusions and recommendations relating to the periodic prooftesting and warm prestressing task were received from General Electric Company. The draft is being reviewed.

Simulated-Service Task

Specifications for procurement of an intermediate-size vessel (6 in. thick, 39 in. in outside diameter) were written and reviewed internally and by consultants. Requests for bids should be tendered by late November.

At Southwest Research Laboratory, the investigation of sharpening machined cracks emphasized thermal cycling. Preliminary indications are that this procedure could be successful. Corrosion cracking studies will be undertaken shortly.

Reference

1. F. J. Witt (Program Director), Heavy Section Steel Technology Program Semiannual Progress Report for Period Ending February 29, 1968, USAEC Report ORNL-4315, Oak Ridge National Laboratory.

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 analytical design procedures for nuclear pressure vessel penetrations and nozzle connections thoroughly substantiated by experimental data. 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 to 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. These investigations are carried out in close cooperation with subcommittees of the Pressure Vessel Research Committee (PVRC) of the Welding Research Council. 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 on nozzles,
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 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, 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 of these activities.

Program 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.

During the past bimonthly period, three out-of-town meetings were attended in connection with this responsibility: (1) a PVRC Subcommittee Task Group Meeting in New York City, September 18, 1968, (2) a meeting of the PVRC Design Division in New York, October 1, 1968, and (3) a meeting with a subcontractor at the University of Waterloo, Canada, October 24, 1968. Minutes of the meetings and progress reports from the various subcontractors were prepared and distributed. One report was published.[1]

Single Nozzles in Spherical Shells

Single nozzles radially and nonradially attached to a spherical shell are being experimentally analyzed at The University of Tennessee. Three models are being examined. The first has a single radially attached 7 7/8-in.-OD nozzle, and the second has two 2 5/8-in.-OD nozzles attached at angles of 22 1/2 and 45 deg from the apex. The third model, obtained from Southwest Research Institute (SWRI), is a large thin-walled tee, which will be used for plastic limit analysis tests. The second series of tests on the model with the 7 7/8-in.-OD nozzle was finished, and the model was tested for internal pressure, axial thrust, external moments applied to the nozzle, torsion applied to the nozzle, and shear applied at the juncture. For these tests the nozzle was bored out so that the wall thickness was reduced from 0.375 to 0.1875 in. A test rig is being designed and built for the limit test on the SWRI model.

The analytical problem of a single nonradial cylindrical nozzle in a spherical shell is being studied at Auburn University. The shallow-shell theory solution for the case of internal pressure loading has been programmed for computer studies. Computer results are presently being studied and compared with published photoelastic experimental results.

Single Radial Nozzles in Cylindrical Shells

Analytical studies and computer programming for the problem of a single radial 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 programmed for computer studies. Presently the computer program for the out-of-plane bending problem is being debugged.

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. A computer program to reduce and print the data on mats was written and is now being debugged. The strain-gage instrumentation of the plate with the two holes is continuing.

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 that are 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 it was tested with internal pressure loading. The data are now being reduced. The design for the loading frame to apply external moment and thrust loads to the nozzles was completed. The loading frame is ready for fabrication.

Reference

1. R. K. Penny, Small Deflection Behavior of Plates and Shells During Creep, USAEC Report ORNL-TM-2292, Oak Ridge National Laboratory, October 1968.

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 will 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 and 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 thermal stresses and dynamic effects in piping systems and of the design of girth transition joints was completed. This material was added to the preliminary draft of the overall survey report being prepared by Battelle Memorial Institute and Southwest Research Institute. Copies of the preliminary draft, including a short summary discussion of the information most pertinent to each of the 12 tasks in the PVRC program outline, were sent to PVRC headquarters for distribution to the Subcommittee to Develop Stress Indices for Piping, Pumps, and Valves. Copies were also sent to AEC headquarters and to the RDT Standards Program Office at ORNL. A chapter is to be added on pump bodies, and additional material received through the review process before the report is issued in final form will be incorporated.

Dimensional Study of Standard Pipe Fittings

The dimensional data obtained from 4-in. concentric and eccentric reducers were analyzed. Although the statistical sample is small and perhaps not representative of larger sizes, the data indicate that rather wide variations in dimensions and shape exist for these types of fittings, with the eccentric reducers exhibiting the most variation. It appears from these data that a single or universal shape characterization for eccentric reducers would not be meaningful for stress analysis purposes. Further study is needed in order to determine a more reasonable approach.

Several reactor designers were requested to furnish information on piping components currently being used in class I piping systems. These data are being compiled into a brief phase report.

Epoxy models of the 12-in. ASA B16.9 tees to be experimentally stress analyzed and fatigue tested are being made. These models will provide a permanent record of the shape and dimensions of the tee for later reference and a working model for developing the finite-element analysis.

Structural Analysis of ASA B16.9 Tees

Negotiations are in progress with the University of California (Professor R. W. Clough) for the finite-element stress analysis of two 12-in. ASA B16.9 tees. These tees will be analyzed for the 13 loading cases described under Task 1, Phase A, of the PVRC program outline.[1] A certain amount of computer program development work will be done to determine which of several node element descriptions will give the best results in comparison with the experimental data. Thermal stress analysis capability and automatic input-data generation features will be added to the computer program later.

Currently, several proposals for the experimental stress analysis and fatigue test of the 12-in. tees are being evaluated. Subcontracts for this work will be awarded when the evaluations are completed. The following seven additional tees have been ordered from Ladish, Tube Turns, and Taylor Forge to complete acquisition of test specimens for the experimental stress analyses specified under Task 1 of the PVRC outline.

Material	Size (in.)	Schedule
A 106, grade B, carbon steel	6 × 6 × 6	160
	24 × 24 × 24	40
	24 × 24 × 24	160
	24 × 24 × 10	40
	24 × 24 × 10	160
Type 304 stainless steel	12 × 12 × 12	160
	12 × 12 × 6	160

Structural Analysis of Other Piping Components

A computer program was purchased from A. Kalnins, Consulting Engineer, for the analysis of elbows and curved pipe with attached lengths of straight pipe. This computer program will permit studies of "end effects" under various combinations of applied forces and moments and internal pressure. The analyses will also provide additional checks on the approximate theories of von Karman, Rodabaugh and George, Gross, and others.

Plans are being made for implementing the experimental analysis of oval and thin elbows called for in Task 5 of the PVRC outline. Theoretical studies of these problems are also being started. The present plan is to attempt to modify the solutions for the geometrically perfect elbow to account for these additional variables.

Plans are also being formulated for developing stress indices by using established analytical solutions for lugs, concentric reducers, tapered transition joints, and bolted-flanged joints.

Reference

1. Program and Request for Proposals for Development of Stress Indices and Methods of Analysis for Piping, Valves, and Pumps, unnumbered PVRC report dated July 1, 1967.

6. RDT STANDARDS PROGRAM

M. Bender, Coordinator

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

W. F. Ferguson W. W. Goolsby R. A. Schmidt

The Oak Ridge National Laboratory, at the request of the AEC Division of Reactor Development and Technology, assumed the prime responsibility in the organization, development, preparation, and distribution of engineering standards to provide a sound base for RDT reactor and related programs necessary to the planning, building, and operation of water-cooled nuclear reactors. Other standards, related to both water and liquid-metal reactors, are being developed in cooperation with the Liquid Metals Engineering Center. The Standards Program is designed to furnish a source of sound engineering data for designers, builders, and operators of reactor systems and components that result in improved quality assurance and reactor safety and reliability.

Present efforts are directed toward (1) continuation of the review and updating of existing standards and the issuance of tentative standards; (2) preparation of scopes and outlines and collection and review of technical data for standards; (3) finalizing guidelines for format, definitions, titling, paragraphing, numbering, and other features necessary for clarity, uniformity, and continuity in writing standards; (4) direction and review of subcontractor participation and engineering consultant activities; (5) participation in conferences and meetings; consultation with reactor designers, builders, and operators; visits to AEC sites and agencies; and exchange of information with others in the field in reactor technology; (6) continuation of the planning and development of schedules, goals, and program details, including RDT Standards Index and monthly progress résumés and reviews; (7) coordination and cooperation with the liquid-metals standards programs; and (8) participation with Idaho Nuclear Corporation in the development and preparation of quality-assurance standards.

Organization, Scheduling, and Manpower Requirements

The effective prosecution of the Standards Program is proceeding as scheduled with the equivalent of 30 engineers. Subcontract work continued at a steady pace with the involvement of Burns and Roe, Inc., Franklin Institute, MPR, NUS, Teledyne, and United Nuclear, primarily in areas of work assigned in FY-68.

Planning, development, and review of work plans, scopes, proposed standards, schedules, and program are continuing. Numerous work sessions, meetings, and discussions were held, both within ORNL and with outside contacts, for the development, review, and discussion of scopes, outlines, and draft documents. A schedule for the FY-69 program is being defined that reflects the standards being written and issued during this period. The sequence and number of proposed standards will continue to be revised as the need for other standards is realized more fully according to the needs of the AEC and its contractors. Each specialist is applying considerable effort in preparing and obtaining the necessary input data required for the completion of tentative documents within the proposed schedule.

Standards Activities

The specialists continued to develop, prepare, modify, edit, and review scopes, outlines, and standards; review and study pertinent documents, materials, and reports pertaining to the design, testing, construction, fabrication, operation, and maintenance of water-cooled reactors; visit and consult at other sites and agencies; and review and direct engineering consultant activities and subcontract endeavors. Several tentative standards are under development and nearing completion. Also, updating and revising of existing standards is proceeding steadily. The following draft standards were submitted to RDT for review, comment, and approval:

MB-350 Zirconium and Zirconium Alloy Ingots for Nuclear Application

MB-351 Zirconium and Zirconium Alloy Bars, Wire and Rod for Nuclear Application

MB-352 Zirconium and Zirconium Alloy Sheet, Strip, and Plate for Nuclear Application

MB-353 Zirconium and Zirconium Alloy Seamless Tubes for Nuclear Service

MB-356 Zirconium and Zirconium Alloy Forgings and Extrusions for Nuclear Service

To date, 86 drafts have been submitted to RDT; 61 of these have been approved by RDT as tentative, and 56 have been issued as tentative standards. Approximately 250 copies of each tentative standard were distributed to 66 addresses, including AEC installations, codes and standards organizations, architect-engineers, consultants, nuclear component manufacturers, and electric utilities.

Inquiries soliciting interest in an RDT Standards conference were distributed to 177 representatives from codes and standards organizations, industry, and universities. Early replies indicate appreciable interest from those contacted. Plans are being developed to orient the meetings toward the areas of interest expressed in the questionnaires returned. AEC organizations will be contacted as available manpower can handle the correspondence.

RDT Standards personnel have established and continued correspondence and participation with several standards organizations. By contributing to the development of adequate voluntary standards for nuclear requirements the need for supplementary RDT Standards may, in many cases, be forestalled. Recent communications include ASTM A-10 Task Group on cleaning, ASTM E-7 subcommittee on radiography of stainless steel castings, ASTM E-10 covering radioisotopes and radiography, AACC CS-6 on clean rooms, ASME Nuclear Pump and Valve Code, and ANS-7 Reactor Components.

Comments on a number of RDT Standards are being received from several sources outside AEC and its contractors for evaluation and incorporation in subsequent revisions. Comprehensive reviews and many worthwhile recommendations on a number of standards were made by Babcock & Wilcox. Requests for copies of RDT Standards are being received in increasing numbers from electric utilities, architect-engineers, and constructors.

The extension and reorientation of some subcontracts proceeded during this period. A reduction in available funds has necessitated rescheduling and postponement of certain activities. No new subcontracts are planned for FY-69. Work already under way on standards continued as previously scheduled.

A draft of the proposed RDT Standards Index was developed. This Index, to be issued quarterly to all RDT Standards recipients, is designed to provide a current listing and status of standards issued. Information provided by the Index includes standards number, title, issue date, revision number or date, issuing office, applicability, and status, such as tentative, revised, or final. The Index will establish a system for assuring the recipient the latest standard issued.

Lists of standards and brief scopes of each continue under development. The list denotes the standards to be written and issued in FY-69 and subsequent years. Manpower availability, consultation with others, recommendations from RDT, further work on planning and scheduling, input data from other sources, and discussions and visits to other sites and agencies will result in additions and modifications to the list.

Other areas of significant effort include development of work plans and scheduling of standards activities, as well as preparation of slides, charts, and other material for discussion programs and informational meetings. A preliminary Draft Document Schedule was prepared that depicts the schedule, current status, percentage completion, and progress of each document in process or planned for development in FY-69. It is anticipated that the progress will be reviewed and the schedule updated on a monthly basis.

ORNL has been requested to participate in the review of pressure-vessel regulatory criteria, nuclear piping system criteria, boundary charts, and a document on aseismic design for the Division of Reactor Standards.

6.2 REACTOR COOLANT SYSTEMS AND EQUIPMENT STANDARDS
(AEC Activity 04 60 80 03 1)

H. G. Arnold
J. W. Anderson
W. A. Bush
D. D. Cannon
R. W. Dehoney
D. L. Gray
H. A. Nelms
C. L. Segaser

Piping and Valves

With the assistance of Teledyne Materials Research Laboratory, a standard for piping design is being prepared. This standard, about 37% complete, will present techniques and methods of analysis of piping systems that meet the requirements established by USAS B31.7, "Nuclear Power Piping Code." Preliminary drafts of the sections covering pressure design, initial flexibility analysis, support and vibration control, and earthquake analysis are complete.

Work continues toward formulating a simplified method for calculating thermal gradients through pipe walls in water-cooled nuclear power plant piping systems. All illustrations and sections of the draft of the standard for valve design were completed. The overall standard will be available in mid-November.

A meeting was attended with the Pipe Fabricators Institute and DRS in which there was discussion on quality assurance of nuclear power piping as required by AEC and the new Nuclear Power Piping Code. ORNL comments on the ASME Nuclear Pump and Valve Code were transmitted to the RDT Components Branch for assessment and presentation to the Code Committee. Other meetings included the ASME Pump and Valve Code Committee on design. Agreement was reached on the final draft of the section for design of valve bodies. This section of the Code will probably be referenced as a mandatory requirement in the RDT valve standards.

Reactor Internals

The preliminary draft for phase I of the Reactor Internal Design Standard prepared by United Nuclear Corporation was reviewed, and comments for revision of the draft were consolidated.

Work is continuing on a standard covering the design considerations for reactor internals. The scope of this document was rewritten and an outline prepared.

Arrangements are being made with Argonne National Laboratory for the reproduction of AARR documents. These documents contain information similar to that obtained on HFIR and ATR and should provide additional technology for several activities in the Standards Program.

Pumps

The first draft of the RDT Standard for Selection of Reactor Coolant Pump Functional and Design Requirements is being rewritten to reflect the comments received from an internal review.

The overall effort on the first draft of the RDT Standard for Pump Design is approximately 70% complete, with completion anticipated about January 1, 1969. A Franklin Institute stress analyst attended the meeting of the ASME Ad-Hoc Committee on Pumps and Valves for Nuclear Power's Special Committee on Design at the United Engineering Center. Neither Bingham Pump Company nor Byron Jackson Company submitted results of the pressure tests on pump castings at that meeting.

A visit was made to the Oyster Creek Facility to become familiar with the physical arrangement of the components of a boiling-water reactor.

Heat Exchangers

A visit was made to Burns and Roe, Inc., to review the status of the Heat Transfer Design Standard. Substantial progress was achieved on this document. The following sections were received: (1) Scope, (2) Material Selection, (3) Discontinuity Stress Analysis of a Welded Hemispherical Head-to-Shell Joint, (4) Discontinuity Stress Analysis of a Welded Channel Cover Tube Sheet Shell Joint. Other portions of the documents that are in process, partially completed, or being reviewed are (1) heat-transfer design of a water-to-water unit, which is approximately 15% complete, and (2) systems descriptions of the various portions of nuclear power plants employing heat exchangers. The latter comprehensive section is 95% complete and is undergoing internal review before release.

A recently completed survey of tube failures in various primary heat exchangers for RDT revealed that 9 out of 17 of the plants have sustained some degree of damage to the primary heat exchangers, apparently attributable to excessive tube vibration. It is doubtful that 8 out of 17, or 47%, successful operations would be considered reliable performance. This study has also shown that if the natural frequency of vibration of the tubing coincides with the exciting frequency, a condition of resonance is induced, and the probability of failure of one or more tubes is virtually 100%.

The LMEC draft on a sodium-to-air cooler and the LMEC standard RDT-E-76, Circulating Cold Trap for LMFBR and Test System Service, were reviewed and comments returned to LMEC.

Vessels

Extensive comments were received from Babcock & Wilcox relative to Standards RDT-S-918 and RDT-E-4. These comments were reviewed for applicability in preparation for reissuance of these documents. Replies were prepared and sent to B&W indicating the disposition of their comments.

6.3 INSTRUMENTATION, CONTROLS, AND ELECTRICAL STANDARDS (AEC Activity 04 60 80 03 1)

A. E. G. Bates C. S. Walker
W. F. Ferguson T. G. Robinson
J. F. Potts F. W. Symonds
O. M. Thomas

Reactor Protection Instrumentation Systems

Efforts are continuing in the development of an outline for preparing documents for Reactor Protection Instrumentation Systems. These documents are being prepared in consultation with DRDT and DRS and will be used as criteria for application by AEC, rather than standards. The criteria will be written in the form of supplements to the AEC 10CFR50 and IEEE No. 279 criteria. Tentatively, they will present the supplementary criteria and be followed by the pertinent IEEE and AEC criteria. In those instances where the supplementary criteria differ from the others, the reasons for the differences will be explained.

At the second meeting of the Ad-Hoc Instrumentation and Control Review Committee at Bethesda, Md., on October 9 and 10, the outline, introduction, scope, and samples of the supplementary criteria were discussed. A tentative introduction was prepared, and various criteria were rewritten as a result of these discussions.

Instruments

The first draft of a standard for thermocouples was prepared and is undergoing review within the Instrumentation and Controls Division.

Electrical

Specifications, design practices, regulatory requirements, functional requirements, fabrication techniques, and commercial practices for containment electrical penetrations were obtained from a variety of sources. A scope and outline for an electrical penetration standard were begun. Reactor power systems for several power and test reactor installations are being analyzed to form the basis for an electrical design standard.

6.4 PROGRAMMATIC AND PROCEDURAL STANDARDS

(AEC Activity 04 60 80 03 1)

J. W. Anderson N. E. Dunwoody
H. G. Arnold L. F. Lieber
C. A. Burchsted C. L. Segaser
H. C. Savage

Quality Assurance

Visits were made to Southern California Edison Company and Pacific Gas and Electric Company for the purpose of becoming familiar with the quality-assurance practices employed by utilities in their nuclear power projects. Information gained from these visits, along with previous visits to Government laboratories, architect-engineering firms, and equipment suppliers, is being used by Idaho Nuclear Corporation in the development of the quality-assurance standards.

The first draft of RDT-S-945, "Quality-Assurance Program Requirements," was completed by Idaho Nuclear Corporation and transmitted to ORNL for review. INC has completed the review of quality-assurance practices related to the procurement of valves for the Advanced Test Reactor (ATR), and a report will be issued soon. Compilation of the procurement quality-assurance data for heat exchangers for the ATR continued.

Comments were received from most ORNL reviewers on RDT-S-943, "Quality-Assurance System Requirements for Construction." The standard is being reviewed by AEC-ORO, and it is planned to issue this standard and RDT-S-945 to the Quality-Assurance Ad-Hoc Committee for review.

The study of quality-assurance practices used in procurement of the HFIR pressure vessel, heat exchangers, and coolant pumps was completed and distributed for information and guidance.

Maintenance

Douglas United Nuclear has indicated a desire to become actively involved in preparation of maintenance standards. Visits were made to Browns Ferry to discuss maintenance, testing, and quality-assurance programs.

As requested by RDT, the primary maintenance standards preparation effort will shift from components standards to comprehensive maintenance program standards. A visit was made to the Savannah River Plant to review the recently installed computerized maintenance system (Maintenance Information and Control System). The system provides a computerized method of requesting, authorizing, scheduling, and controlling cost of maintenance activities, as well as budget forecasting, historical data, and manpower activities.

Testing

The Preoperational Testing Standard was reviewed with Burns & Roe, Inc., and the outline was rearranged to a more logical sequence. The first draft of the section of General Requirements is progressing well, and a list of terms to be defined is being developed. A scope and an outline for the Containment Testing Standard were prepared.

Reliability

The final draft of the NUS reliability document was received and is being prepared for publication. No further subcontract work on reliability is scheduled under the present funding level. Effort is also being directed to identifying the ways in which reliability analysis techniques are applicable to systems other than engineered safety systems.

Water Chemistry

Work on the preparation of RDT standards covering water chemistry requirements for reactor coolant systems was initiated. The objective is to prepare RDT standards covering the control of coolant composition in the primary system of research and test reactors, boiling-water reactors, and pressurized-water reactors. Progress included the review of literature, reports, and documents relating to the theory and current practice used in water-reactor coolant technology. Letters were sent to selected manufacturers and operators of commercial reactors advising them of the planned preparation of the subject standards and requesting available operating procedures, technical manuals, specifications, and other data relating to water chemistry control. Arrangements are being made to meet with representatives of these organizations to discuss coolant chemical control methods, experience, and problems in successfully operating reactors.

Preliminary draft outlines were written for three proposed standards for control of primary coolant composition in (1) pressurized-water reactors, (2) boiling-water reactors, and (3) research and test reactors.

Visits were made to Dresden Nuclear Power Station at Morris, Illinois, to discuss water chemistry control standards used in boiling-water reactors; to Babcock & Wilcox in Lynchburg, Virginia, to discuss coolant water technology of pressurized-water reactors; and to Yankee Atomic Electric Company, Rowe, Massachusetts, and Connecticut Yankee, Haddam Neck, Connecticut, to discuss details of coolant composition control and specifications.

Cleaning Standards

Preparation of a series of standards covering cleaning of critical materials and equipment and cleanliness control in fabrication and construction areas is in progress. Cleaning of mill products and other materials will be covered by standardized paragraphs that will be added to the specifications for those materials. Meetings were held with personnel of Burns & Roe, Inc., Oyster Creek Reactor, Richardson Chemical Cleaning Service Company, and the Tennessee Valley Authority to review current cleaning and cleanliness control practices. Contracts were also made with Knolls Atomic Power Laboratory, the Sandia Corporation, Babcock & Wilcox, and NASA. Preparation of the standardized paragraphs to be added to mill-product standards is expected to be complete by the end of November. Planning and organization of the following proposed standards is in progress: (1) Cleaning of Critical Components and Piping for Nuclear Reactors, (2) Cleanliness Control in Fabrication and Construction Areas for Critical Components and Piping, (3) Preoperational Cleaning of Critical Systems in Nuclear Reactors.

Postoperational cleaning and decontamination will be covered in another series of standards. The RDT Standards Program was represented at a meeting of the Clean Room Committee (CS-6) of the American Association for Contamination Control and is expected to be represented on a new ASTM committee (A-10) for cleaning of materials for nuclear applications; the latter committee will hold its first meeting in November.

Style Manual

The draft of a proposed RDT standard entitled Preparation of RDT Standards was completed and issued for comment and internal use. Upon approval, this standard will be issued to all participants in the RDT and LMEC standards program as a guide for organizing and writing standards that will be issued under these programs.

Air Cleaning

A meeting of the AEC Air Filter Committee will be held in December in conjunction with the mid-year work meeting of the American Association for Contamination Control (AACC). AACC standards CS-1-68T, HEPA Filters, and CS-2-68T, Laminar Flow-Clean Air Devices, were issued during October. The RDT Standards Program was represented as secretary of the committees for both of these standards.

6.5 MATERIALS AND FABRICATION
(AEC Activity 04 60 80 03 1)

R. M. Fuller H. F. Jackson

The consolidation of RDT-S-908 and RDT-S-934 was completed, and tentative standard RDT-S-934 was distributed. The RDT-S-934, Ultrasonic Examination Standards, and ASME Boiler and Pressure Vessel Code, Section III, Appendix IX, provide the necessary nondestructive examination methods covering welding, materials, cladding, and hard surfacing. Tentative standards for zirconium and zirconium alloys were forwarded to RDT. These standards were written primarily for use with reactor internals. Aluminum and aluminum alloy standards are being written for similar application and are about 60% complete.

A visit was made to RDT to discuss the welding standard for fuel element fabrication, RDT-S-900, which is presently being rewritten to incorporate RDT comments. Other meetings included visits to Airco Rods to discuss weld-wire standards; to LMEC to discuss standards for ultrasonic examination and the use of nondestructive examination standards with the material standards; and a meeting of ASTM E7 Sub. 2, the subgroup on shrinkage, to review radiographic film and discuss ASTM E71 upgrading.

A tentative draft for nickel-molybdenum-chromium alloy sheet and plate (Hastelloy N with modified chemistry) was rewritten. An outline was prepared for a standard covering Protective Coatings for Reactor Components and Facilities. This standard will utilize much of the information on material selection that is covered in the proposed ANS standard and will expand in the areas of designing for use of coatings and preparation and utilization of job procedures for both shop and field conditions.

7. GENERAL NUCLEAR SAFETY STUDIES

7.1 HTGR SAFETY PROGRAM OFFICE

(AEC Activity 04 60 70 01 1)

S. I. Kaplan   M. D. Silverman
E. R. Taylor   W. C. Ulrich

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

Information Exchange

On October 23 and 24, 1968 two HTGR-SPO staff members participated in the Helium Applications Symposium in Washington, D.C. Silverman presented a paper entitled "The Spectrum of Uses of Helium in the Atomic Energy Program," and S. I. Kaplan presented a paper prepared by D. B. Trauger on "Helium-Cooled Nuclear Reactors."

Staff Changes

W. C. Ulrich of the Program Office staff (half time) was granted a year's leave of absence from ORNL as of September 15 to pursue graduate education. He was replaced by E. R. Taylor, who has joined the Program Office on a full-time basis. Taylor's work will be concerned with PCRV and component testing and mechanical design criteria.

Research Analysis

In support of its recommendations issued in the preceding report on steam-graphite research coordination, the HTGR-SPO discussed review procedures with DRDT representatives and also arranged a telephone conference between ORNL and BNWL research workers on graphite-oxidation projects of mutual interest. As an outgrowth of the Program Office recommendations, a Program Office staff member will in the future serve on the AEC Graphite Coordination Working Group.

Supplemental Activities

A Program Office staff member attended a meeting between Gulf General Atomic and representatives of the USAEC Division of Reactor Licensing (DRL) in October at which GGA reported research progress on topics recommended by the Advisory Committee on Reactor Safeguards in connection with the Fort St. Vrain HTGR Plant being erected in Colorado.

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) provision of 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 the effect of fires.

Shipping Cask Criteria

All writing for the "Engineering Standards and Guide to the Design of Spent Fuel Shipping Casks" was finished. The technical chapters were prepared for reproduction and are now being checked prior to submission to the Publications Department. The only nontechnical chapter, the introduction, was submitted to the Division editors. It is expected that the guide will be published during the next reporting period.

Writing was almost completed on two new volumes in the ORNL-TM-1312 series: (1) Lifting Devices and (2) Tie-Down Devices. This information will be used in future revisions of the Guide.

Experimental Investigations

Model Tests at the Drop-Tower Facility

The 30-ft drop test of a model of a buffered all-steel shipping cask was completed at the new drop-tower facility. The information obtained, when analyzed, will be of interest in the fast reactor shipping cask program.

Additional work is being planned for this facility. Specifically, weld configurations, backed with lead, will be tested to determine which behave the best under impact loading. Work to determine the energy-absorption properties of typical closure bolts is planned.

Demonstration Uranium-Shielded Cask Tests

An underwater loading test was conducted which indicated that some lid modifications should be made if the uranium-shielded cask is to be loaded under water. This problem is not a function of the specific shielding material, and no effort to make underwater loading easier is planned unless the cask will actually be placed in service where an underwater loading capability is required. Regarding this matter, AEC Headquarters has indicated that consideration is being given to using this cask to ship radioactive material, and a cost estimate to repair the damage sustained during the test and demonstration program was requested. An analysis indicated that the general appearance of the cask could be improved by mechanically straightening some bent fins; however, to return the cask to a "like new" condition would cost $12,000.

Analysis of Chemical Technology Division Casks

The application for certification of the Foamglas Shipping Container was approved and Special Permit No. 5795 was assigned by the DOT. However, the approval indicated that this container met the 6M specification, but the container, in fact, does not, and a request is being made to DOT to eliminate this reference from the approval.

The Vermiculite Shipping Container was certified and approved with DOT Special Permit No. 5765. Nameplates bearing the DOT number are being made for each shipping container.

The D-38 Uranium-Shielded Cask was certified and approved with DOT Special Permit No. 5787. Modifications to the cask, which were specified in the request for approval, will be made shortly.

The modification of the trailer for the 10-ton Californium Shipping Container, DOT Special Permit No. 5740, was completed. This entire shipping system will give a degree of flexibility heretofore not available in shipping large quantities of transuranium elements from ORNL.

The request for certification of the Dry Hole Charger was prepared and forwarded to the AEC for approval. The modifications to the Dry Hole Charger necessary to bring it up to the state of compliance with the Appendix of AEC Manual Chapter 0529 were completed.

A work order was written to start the design of the new shipping cask to replace the Hot Garden Carrier. The new cask will have a 7-in. inside diameter and 24-in.-long cavity.

Work on the summary report that describes all 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 a statement as to the status of the report. It will be noted that six of the eight reports have now been published and the remaining two submitted to the AEC (one on May 25 and the other on October 4) for review. Publication of these two awaits receipt of these (and other comments), as well as revisions as required.

Table 7.1. Status of the Discussion Papers on Various Aspects of Water-Cooled Reactor Safety

Papers	Authors	Status
Missile Generation and Protection in Light-Water-Cooled Reactor Plants	R. C. Gwaltney	September 1968
Potential Metal-Water Reactions in Light-Water-Cooled Power Reactors	H. A. McLain	August 1968
Emergency Core-Cooling Systems for Light-Water-Cooled Power Reactors	C. G. Lawson	October 1968
Air Cleaning as an Engineered Safety Feature in Light-Water-Cooled Power Reactors	G. W. Keilholtz, C. E. Guthrie, and G. C. Battle, Jr.	September 1968
Testing of Containment Systems Used with Light-Water-Cooled Power Reactors	F. C. Zapp	August 1968
Review of Methods of Mitigating Spread of Radioactivity from a Failed Containment System	R. C. Robertson	September 1968
Earthquakes and Nuclear Power Plant Design	C. G. Bell and T. F. Lomenick	Revised draft submitted to AEC for review 5-25-68
Protection Instrumentation Systems on Light-Water-Cooled Reactor Plants	C. S. Walker	Revised draft submitted to AEC for review 10-4-68

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. Problems of antiseismic design of nuclear facilities can be separated into prediction of the seismic conditions that can occur at a site and methods for meeting those conditions.

A program of antiseismic studies for nuclear facilities was outlined in a draft document and submitted to and discussed with the Nuclear Safety Branch, RDT. Some further revisions are being made. Among the problems of immediate interest are the development of better site investigation methods, a better understanding of the component and system malfunctions that may be caused by earthquake shaking of a nuclear power plant, and an investigation of the pros and cons of a seismic scram.

The primary activity during this reporting period was the development of specific task definitions to carry out the broad objectives outlined.

Some siting problems may be attacked by using vibrations from the underground nuclear tests in Nevada. The conditions which are conducive to soil resonance and other amplifying or attenuating effects might be investigated there. The possibility of quantitatively testing theories is being evaluated.

Soil failure and determination of soil properties essential for safe plant design were identified by experts in the field of earthquake engineering as areas needing study. The definition of specific problems on which work must be done is in progress. A number of soil experts will be canvassed to obtain their opinions on needed development in that field. Deficiencies in soil-testing methods will be used to define tasks for the study of properties of soils and for development of testing methods.

Effects of a seismic scram are under study. An illustration of an advantage of scram is the effect on fuel ruptures that would follow a major loss of coolant. A major rupture of the primary coolant system while the reactor was at power could cause the cladding on as much as 70% of the fuel rods to be "perforated" due to weakness from high temperature (>1200°F) and fission-gas pressure on the inside being greater than the cooling-water pressure. If, however, a reactor shutdown were initiated 1.5 sec before the pipe rupture, no "perforation" of the cladding would be expected.

UCLA has not yet completed analysis of their data from shaking the EGCR, which was reported in the previous progress report.

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 23,000 documents. The several activities and services performed by the Center during this reporting period are discussed below.

Research and Development Contract 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. As a result of a review of the first printout of a sample contract, some changes were made in the background-batch program. Basically, the changes (1) altered the output format by reorganizing and refining some of the entries, (2) identified an abbreviated output format for use by the AEC Regulatory Staff, and (3) provided for lengthening record space for some of the entries.

Programming was completed that will provide extended search capability on the seven-digit project code number. This extended the query capability from the first two digits of the code to a number of different breakdowns. Searches may be done in this manner, as well as by keywords, via the remote consoles.

To date 15 contracts from the Water Reactor Safety Program Office have been placed in the file. Information on a number of other contracts supported by the Office was assembled and will be ready to input as soon as Commission and contractor reviews, now under way, are completed.

Telecommunications

The IBM-2260 cathode-ray tube station was tested as an input device by adding 50 references to a special storage file via the station. No problems were encountered. Background-batch backup and alternate card input programs were tested, and no significant errors were found. Programs for conversion to routine production operations, such as SDI and the preparation of the indexed bibliography, were also tested, and no known problems arose. Parallel test runs on input via the console and by keypunch cards are being initiated.

Routine use of the IBM-2740 remote typewriter-printer for information retrieval continues.

Selective Dissemination of Information Program

Over 50 SDI profiles were revised on the basis of feedback cards and telephone conversations. Some profiles were made broader and some narrower, whichever best suited the needs of the user. During the period there were 32 additions to the SDI program and 29 deletions for a net gain of three new participants.

Special sets of profiles were designed to provide specific coverage to the responsible individuals in a nuclear power plant technical staff group. The idea is to avoid arbitrary choices by providing the manager of technical services with an organization chart (with titles like reactor safety analyst, heat transfer specialist, etc.) so that he can fill in appropriate peoples names. The chart will then be interpreted in the form of a profile for the individuals named. This approach was just used with the engineer in charge of a utility engineering group and the plant superintendent of a midwestern power reactor. Since these profiles should minimize excess cards and yet provide broad and thorough coverage, they will be made available to other utilities as appropriate.

Guide to Nuclear Power Plant Staffing

NSIC prepared and submitted to the AEC for its approval a draft of the booklet "Guide to Nuclear Power Plant Staffing: Requirements, Training, and Education Programs" by E. N. Cramer and H. B. Whetsel. The guide reviews the utility requirements in staffing nuclear power stations with oriented personnel and gives both projected numbers and individual job qualifications; outlines typical training, education, and experience schedules; presents a compilation of 34 current training, education, and experience programs; and estimates the manpower available as graduates from appropriately oriented curricula in 91 universities and 38 technical schools. The intended audience is both the management of utilities who have not already contracted for reactors and those who wish to investigate the possibility of providing special training or revising current educational programs. The text is arranged to describe the various problems and solutions in a concise, readable fashion, with the bulk of the data restricted to appendices.

The guide, which bears the internal number ORNL-NSIC-56, will be printed and disseminated widely by the AEC Division of Technical Information Extension with a "TID" designation.

Compilation of Reactor Design Features for ACRS

NSIC prepared, at the request of ACRS, several sets of tables compiling power reactor design data for comparative purposes. If the tables prove useful, NSIC will plan to maintain them as new information is accumulated. The tables were furnished to ACRS in loose-leaf notebooks to make revision easy.

Reports

The Center's fifteenth bibliography was issued in October.[1] It contained references sorted into the usual 19 subject categories. Beginning with this issue, copies of the bibliography were no longer sent to individuals. It continues to be distributed to libraries and technical groups on NSIC's mailing list and is disseminated on AEC's "Reactor Technology" category distribution. Consideration of the policy change was initiated by economic factors, but the decision was based to a large degree on the fact that the needs of individuals can be met more directly by our SDI program.

A draft of the report "Plume Rise: A Critical Survey" was completed and is being edited following an internal review.

Other Information Activities

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

Information inquiries (letter or phone)	88
Specific NSIC reports	336
Additions to NSIC reports distribution	22
Visits for consultation or use of reference material	25
Additions to SDI program (net)	3

Reference

1. NSIC Staff, Indexed Bibliography of Current Nuclear Safety Literature-15, USAEC Report ORNL-NSIC-52, Oak Ridge National Laboratory, October 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 computer-based information storage and retrieval system for the documentation of design characteristics relevant to the safety evaluation of nuclear power plants. This task has four separate 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

Review by ORNL and DRL of the summary section characteristics for Core Emergency Cooling Systems resulted in a complete revision of the listing for this subsection. Close cooperation between DRL specialists and the CHORD-S group resulted in minimum effort to effect the revisions.

Administrative approval of the list of characteristics covering PWR-type containment by DRL is still pending as a result of an overall program review being made by the Director of Regulation and DRL. A preliminary draft of characteristics covering BWR-type containment was completed; however, this list is being held awaiting the results of the overall program review mentioned above.

The development of the containment subsections covering Electrical, Instrumentation, and Control has been held in abeyance pending the results of the program review and more definitive information regarding the FY-69 budget.

Collection of Information

The summary section data for the ten reactors selected by DRL were reviewed and edited to comply with recommendations from DRL to attempt to abbreviate the data, to make the entries more consistent in format, and to try to eliminate as many inaccuracies as practical. The revised data were entered into computer storage, a printout of the data was reproduced, and copies of the printout were sent to DRL for review and distribution. A draft of a preliminary questionnaire on the "Site" subsection of the summary section was completed and is now being reviewed prior to forwarding to DRL. While no decision has been made regarding the manner in which the AEC may solicit information from applicants, some form of questionnaire appears to be needed.

Computer Programming for Data Retrieval and Comparisons

The computer programming effort required to provide the basic features requested by DRL as a result of their initial experience with retrieving information from the remote console at Bethesda was completed. The changes made operational during this reporting period are listed below.

Program Improvements

1. The line length for remote-console output is now 154 characters to correspond with the line length of the new IBM 2740 teletypewriter consoles used by CHORD—S. In the columnar comparison output the characteristics will occupy no more than 55 characters, and each of the three data columns is provided with a maximum of 27 characters.

2. Truncation of data in the columnar output was eliminated by making provision for continuation of data on succeeding lines.

3. The reactor set number designation is now eliminated unless it is different from "Set 01," the basic set.

4. In columnar output the request for Source (of data) is no longer valid, since the space previously set aside for this information is being used for data. The source reference may still be obtained from the vertical compare or display programs.

5. The reactor name abbreviations are printed out in the vertical compare program rather than the two-letter reactor identification.

6. The TRIV and MATCH options now apply to the vertical, as well as the columnar, format comparisons.

7. Vertical format output for the compare program is now automatic. If columnar output is desired the optional command, "col," must be specified.

8. Truncation was eliminated in the vertical output format by providing for continuation of data on succeeding lines.

9. The numeric search option (Survey) whereby the user can specify a value or range of values or specify a search for the maximum value was changed. In addition to being able to specify that the search be made over all reactors in the data bank, the user can now restrict the search to PWR- or BWR-type reactors, if desired.

General

1. The two-letter reactor code identification used in specifying the reactors of interest in the queries was changed to a hybrid alphabetical—Docket number system, as agreed with DRL.

2. The two-letter source code identification used to identify sources for data items on data input and internally in the computer was changed to simplify the addition of new sources.

3. Characteristics, data, and sources other than up-to-date summary section information were removed from the data bank to minimize data-storage requirements.

4. The CHORD—S abbreviation listing in Section Z was rearranged to facilitate retrieval and additions.

5. The capability of accepting data designated to be printed in the form of a table is now available. An example where this is useful is a list of scram signal names, number of input signals, and setpoints.

6. Test operation and debugging of the display and lead-in programs for the IBM 2260 CRT terminal was started and these programs are now operational.

Development of System Hardware

The temporary IBM 1050 teletypewriter installed at DRL, Bethesda, in June 1968 was replaced with the IBM 2740 teletypewriter console originally specified for this service. The new unit is connected to the Computing

Technology Center in Oak Ridge via a telephone line. This unit is now operating satisfactorily, although occasionally minor difficulties are still experienced as a result of computer difficulties.

Test and debugging operations were conducted on the IBM 2260 CRT terminal located at NSIC. The display and lead-in programs are now operational, and exploratory work to determine the advantages and limitations is in progress.

8.3 TECHNICAL PROGRESS REVIEW NUCLEAR SAFETY
(AEC Activity 07 13 02 02)

J. P. Blakely Wm. B. Cottrell

Nuclear Safety is a bimonthly Technical Progress Review that is prepared and edited at ORNL under AEC contract. The November–December 1968 issue (Vol. 9, No. 6, 120 pages, distributed November 25, 1968), contains a Feature Article on the Safety Design Aspects of the Nuclear-Powered Ship Otto Hahn, by Wiebe, Ulken, Salander, and Henning of GKSS. The authors discuss the main safety features of the ship and her reactor plant, including design, testing, acceptance, and training procedures. This is the first PWR that employs a self-pressurizing principle. Other articles deal with the shielding problems of low-energy particle accelerators (by E. A. Burrill, HVEC), subcooled forced-convection systems (by W. R. Gambill, RD), reliability of reactor components (by P. Rubel, I&C), wide-range nuclear channels (by D. P. Roux, I&C), fission-product release and transport in liquid-metal-cooled fast breeder reactors (by G. W. Keilholtz and G. C. Battle, Jr., RCD), luminescent dosimetry for personnel monitoring (by C. N. Wright, SRL), the proposed definition of radioactive waste categories (by R. E. Blanco, CTD), a review by J. E. Turner (HPD) of a book on radiation protection (by D. J. Rees), items of interest in the Operating Experiences section, and the regular current events articles (see Fig. 1 for contents of Vol. 9, No. 6).

The January–February 1969 issue (Vol. 10, No. 1) was distributed in draft form for external review, and the resulting comments are now in the hands of the authors for response. Final changes in text are scheduled to be submitted to DTIE by early November, with distribution of the issue scheduled for early January. The issue is to contain separate signed editorial comments from G. T. Seaborg, A. M. Weinberg, and Sidney Siegel in recognition of Nuclear Safety's entering its tenth year of publication.

The review draft of the March–April 1969 issue of Nuclear Safety (Vol. 10, No. 2) was distributed for comment as report ORNL-TM-2370;

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Fig. 8.1.

comments are due by December 1. Distribution is scheduled for early March.

Article manuscripts were received for the May–June 1969 issue of Nuclear Safety (Vol. 10, No. 3) and are undergoing internal review. The draft will be distributed for external comment in early January, with publication following in early May.

Outlines were developed and proposed authors are being contacted for future issues through November–December 1969 (Vol. 10, No. 6).

Nuclear Safety Program Seminars

Seminars were held in September and October for members of the Nuclear Safety Program staff. The September seminar was presented by Wilson R. Cooper, TVA-Chattanooga, who discussed TVA's experience in obtaining construction permits for the Browns Ferry reactors. He placed particular emphasis on those areas that were noted to be of special concern in the regulatory reviews, including the effect of sharing equipment among units, emergency power supply requirements, performance ability of the ECCS, and quality assurance.

The October seminar was given by Earl M. Shank, U.S. technical advisor to the Eurochemic Company, Mol, Belgium. He discussed several special considerations necessitated by the construction and operation of a multipurpose radiochemical processing plant in a highly populated area. He also mentioned several unique problems resulting from the high groundwater level, legal limitations on activity disposal, and the international ramifications of the problems.

Future seminars have been scheduled through April 1969.

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272. A. J. Goodjohn, Gulf General Atomic, P. O. Box 608, San Diego, California 92112
273. J. J. Gough, Code 706, U.S. Navy Marine Engineering Laboratory, Annapolis, Md. 21402
274. R. H. Graham, Gulf General Atomic, P. O. Box 608, San Diego, California 92112
275. J. G. Gratton, DTI, AEC, Washington
276. H. Graves, Westinghouse Electric Corporation, Atomic Power Division, P. O. Box 355, Pittsburgh, Pa. 15230
277. D. K. Greenwald, Ladish Company, Cudahy, Wis.
278. R. F. Griffin, Bechtel Corp., Box 3965, San Francisco, Calif.
279. J. J. Grob, Consolidated Edison Company, 4 Irving Place, New York, N.Y.

280. J. E. Grund, Phillips Petroleum Company, 621 Southwest Alder St., Portland, Oregon 97205
281. Bruno Guerrini, Universita Di Pisa, Instituto di Impianti Nucleari, Pisa, Italy
282. A. Guzday, Foster-Miller Associates, 183 Bear Hill Road, Waltham, Mass. 02154
283. L. L. Guzick, U.S. Navy Dept., Bureau of Ships Code 609.36, Standardization Societies, Liaison Section, Washington
284. J. C. Haire, Phillips Petroleum Company
285. O. Hagen, Westinghouse Electric Corporation, Cheswick, Pa.
286. D. B. Hall, Los Alamos Scientific Laboratory, P. O. Box 1663, Los Alamos, New Mexico 87544
287. H. L. Hamester, DRDT, AEC, Washington
288. S. H. Hanauer, 606 Dougherty Hall, University of Tennessee, Knoxville, Tennessee 37916
289. Harvard Air Cleaning Laboratory, Harvard University, 665 Huntington Ave., Boston, Massachusetts 02190
290. R. A. Hasty, Department of Chemistry, Montana State University, Bozeman, Montana 59715
291. R. L. Heath, Idaho Nuclear Corp., Box 1845, Idaho Falls, Idaho
292. F. M. Heck, Westinghouse Electric Corporation, Atomic Power Division, P. O. Box 355, Pittsburgh, Pa. 15230
293. W. G. Hegener, Sargent & Lundy, 140 So. Dearborn St., Chicago, Illinois
294. H. G. Hembree, DRDT, AEC, Washington
295. J. M. Hendrie, Brookhaven National Laboratory, Upton, New York 11973
296. R. L. Hervin, ORO, AEC
297. R. K. Hilliard, Battelle-Pacific Northwest Laboratory
298. P. G. Holsted, DRDT Senior Site Representative, Box 550, Richland, Wash.
299. A. B. Holt, SS, AEC, Washington
300. N. Horton, General Electric Company, San Jose, Calif. 95125
301. J. P. Howe, Gulf General Atomic, P. O. Box 608, San Diego, California 92112
302. P. W. Howe, DRL, AEC, Washington
303. F. C. Huffman, USAEC, Oak Ridge, Tennessee 37830
304. R. Impara, Division of Reactor Standards, USAEC, Washington, D.C. 20545
305. S. M. Ingeneri, Stone and Webster Engineering Corporation, Boston, Mass.
306-308. E. Irish, Battelle-Pacific Northwest Laboratory
309. H. Isbin, University of Minnesota, Minneapolis, Minnesota 55455
310. R. O. Ivins, Argonne National Laboratory
311. I. E. Jackson, Division of Reactor Development & Technology, USAEC, Washington, D.C. 20545
312. B. Japikse, Barnebey-Cheney Co., Cassady at 8th Ave., Columbus, Ohio 43219
313. A. A. Jarrett, Atomics International, P. O. Box 309, Canoga Park, California 91304
314. S. Jayne, Gulf General Atomic, P. O. Box 608, San Diego, California 92112

315. B. D. Johnson, Health and Safety Division, Health Physics Branch, USAEC, Idaho Operations Office, Idaho Falls, Idaho
316. O. H. Jones, Babcock & Wilcox Company, Box 785, Lynchburg, Va.
317. H. S. Jordan, Los Alamos Scientific Laboratory, P. O. Box 1663, Los Alamos, New Mexico 87544
318. M. E. Kantor, Gulf General Atomic, P. O. Box 608, San Diego, California 92112
319. Walter A. Kee, DTI, USAEC, Washington, D.C. 20545
320. C. E. Kent, MC-11, General Electric Company, 175 Curtner Ave., San Jose, Calif. 95125
321. D. King, Idaho Operations Office
322. R. E. Kettner, Consumers Power Company, 212 W. Michigan Ave., Jackson, Michigan 49201
323. E. E. Kintner, Division of Reactor Development & Technology, USAEC, Washington, D.C. 20545
324. R. W. Klecker, DRL, AEC, Washington
325. W. G. Knecht, Darling Valve & Mfg. Co., Williamsport, Pa.
326. L. Kornblith, Jr., Div. of Compliance, AEC, Washington
327. H. J. C. Kouts, Brookhaven National Laboratory, Upton, L.I., New York 11973
328. J. L. Kovach, North American Carbon, Inc., P. O. Box 212, Columbus, Ohio 43216
329. Howard Kraig, Westinghouse Electric Corp., Astronuclear Laboratory, P. O. Box 10864, Pittsburgh, Pennsylvania 15236
330. R. W. Kupp, S. M. Stroller Associates, 201 Park Avenue, South, New York, New York 10000
331. P. Lakey, Idaho Nuclear Corporation, Idaho Falls, Idaho
332. D. F. Landers, Teledyne Materials, 303 Bear Hill Road, Waltham, Massachusetts 02154
333. B. F. Langer, Westinghouse Electric Corporation, Pittsburgh, Pa.
334. D. B. Langmuir, TRW Inc., One Space Park, Redondo Beach, Calif.
335. W. J. Larkin, ORO, AEC
336. C. F. Larson, Welding Research Council, 345 E. 47th Street, New York, N.Y. 10017
337. A. H. Lazar, Babcock & Wilcox Company, Box 1260, Lynchburg, Va.
338. F. Leonard, Witco Chemical, 277 Park Ave., New York, N.Y. 10017
339. Saul Levine, DRL, AEC, Washington
340-341. L. Link, LMFBR Program Office, Argonne National Laboratory, 9700 S. Cass Avenue, Argonne, Illinois 60440
342. Liquid Metal Engineering Center, c/o Atomics International, P. O. Box 309, Canoga Park, California (Attention: R. W. Dickenson)
343. A. Lohmeier, Westinghouse Electric Corp., Heat Transfer Division, Lester Branch, P. O. Box 9175, Philadelphia, Pennsylvania 19113
344. P. Lottes, Argonne National Laboratory
345. J. D. Lubahn, Metallurgical Engineering Dept., Colorado School of Mines, Golden, Colorado
346. R. N. Lutman, General Electric Company, San Jose, Calif.
347. M. V. Malkmus, Tube Turns, Div. of Chetrom Corp., Louisville, Ky.
348. H. G. Mangelsdorf, 78 Knollwood Road, Short Hills, New Jersey 07078

349. W. D. Manly, Materials Systems Division, Union Carbide Corp., 207 Park Avenue, New York, N.Y.
350. M. M. Mann, Director of Regulation, U.S. Atomic Energy Commission, Bethesda, Maryland
351. R. I. Marble, Farr Company, P. O. Box 90187, Airport Station, Los Angeles, Calif. 90009
352. G. B. Matheny, Phillips Petroleum Company, Idaho Falls, Idaho
353. J. D. McAdoo, Westinghouse Electric Corporation, Atomic Power Division, Pittsburgh, Pa.
354. R. G. McAllister, Liberty Mutual Insurance Company, Research Center, Franklin Road, Hopkinton, Mass. 01748
355. S. J. McCool, Division of Operation Safety, USAEC, Washington, D.C. 20545
356. J. G. McCormack, Battelle-Pacific Northwest Laboratory
357. W. P. McCue, ARHCO Document Control, Atlantic Richfield Hanford Co., Federal Bldg., Richland, Washington 99352
358. C. R. McCullough, Southern Nuclear Engineering, 6935 Arlington Road, Bethesda, Maryland
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361. W. T. McIntosh, DRDT, AEC, Washington
362. J. T. McKeon, The M. W. Kellogg Company, Box 696, Piscataway, New Jersey 08834
363. W. N. McLean, Crane Co., 4100 South Kenzie, Chicago, Ill.
364. J. L. Mershon, DRDT, AEC, Washington
365. U. Merten, Gulf General Atomic, P. O. Box 608, San Diego, California 92112
366. G. S. Merz, Savannah River Laboratory, Aiken, S.C.
367-368. C. E. Miller, Jr., LMFBR Program Office, Argonne National Lab.
369. Walter Mitchell, III, Project Manager, Southern Nuclear Engineering, Inc., Box 10, Dunedin, Florida
370. T. R. Moffette, Gulf General Atomic, Inc., P. O. Box 608, San Diego, California 92112
371. A. T. Molin, United Engineers and Constructors, Inc., 1401 Arch St., Philadelphia, Pa. 19105
372. H. O. Monson, Argonne National Laboratory, Laboratory Director's Office, Argonne, Illinois 60439
373-374. H. Morowitz, Atomics International, Box 309, Canoga Park, Calif.
375. R. L. Morgan, Canoga Park Area Office, Box 591, Canoga Park, Calif. 91305
376. P. A. Morris, DRL, AEC, Washington
377. D. L. Morrison, Battelle Memorial Institute, Columbus, Ohio
378. F. M. Moschini, Westinghouse Electric Corp., Pittsburgh, Pa.
379. C. Muehlhause, National Bureau of Standards, Washington, D.C.
380. J. L. Murray, General Electric Company, San Jose, California
381. J. L. Murrow, Lawrence Radiation Laboratory, Livermore, Calif.
382. H. W. Newson, Department of Physics, Duke University, Durham, North Carolina 27700
383. R. Newton, DRDT, AEC, Washington
384-385. R. Nightengale, Battelle-Pacific Northwest Laboratory
386. D. A. Nitti, Babcock & Wilcox Company, Box 1266, Lynchburg, Va.

387. J. H. Noble, Stone & Webster Engineering Corp., Boston, Mass.
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391-392. W. E. Nyer, Aerojet General, Idaho Falls, Idaho 83401
393. D. T. Oakley, National Center for Radiological Health, 12720 Twinbrook Parkway, Rockville, Maryland 20852
394. A. A. O'Kelly, 2421 W. Rowland Ave., Littleton, Colorado 80120
395. David Okrent, Argonne National Laboratory, 9700 S. Cass Avenue, Argonne, Illinois 60439
396. K. R. Osborn, General Chemical Division Allied Chemical & Dye Corp., Rector Street, New York 6, N.Y.
397. M. N. Ozisik, North Carolina State University, Raleigh, N.C.
398-399. R. E. Pahler, DRDT, AEC, Washington
400. N. J. Palladino, The Pennsylvania State University, University Park, Pa. 16802
401-402. S. Paprocki, Battelle Memorial Institute
403. C. S. Patterson, Furman University
404. W. D. Pearl, General Electric Company, San Jose, Calif.
405. I. A. Peltier, Idaho Operations Office
406. D. B. Peterson, Westinghouse Electric Corporation, Box 79, West Mifflin, Pa. 15122
407. G. B. Pleat, Division of Production, USAEC, Washington, D.C. 20545
408. A. K. Postma, Battelle-Pacific Northwest Laboratory
409. N. F. Prescott, Atwood & Morrill Co., Salem, Mass.
410-411. A. J. Pressesky, DRDT, AEC, Washington
412. J. Proctor, Naval Ordnance Laboratory, White Oak, Silver Spring, Md.
413. Lt. Col. John B. Radcliffe, USAEC, San Francisco Operations Office, 211 Bancroft Way, Berkeley, California 94704
414. Hilda M. Reitzel, Librarian, Mine Safety Appliances Co., J. T. Ryan Memorial Lab., 100 N. Braddock Ave., Pittsburgh, Penn. 15208
415. M. E. Remley, Atomics International Division, North American Aviation, Inc., Canoga Park, California 91304
416. B. Resnick, Components Branch, DRDT, AEC, Washington
417. C. W. Richards, Canoga Park Area Office, Box 591, USAEC, Canoga Park, California 91304
418. R. J. Rickert, Nuclear Division, Combustion Engineering, Inc., P. O. Box 500, Windsor, Connecticut 06095
419. R. L. Ritzman, Battelle Memorial Institute, Columbus, Ohio
420. I. C. Roberts, DRDT, AEC, Washington
421. C. L. Robinson, USAEC, Richland Operations Office, Richland, Washington
422. T. Rockwell, III, MPR Associates, Inc., 815 Connecticut Ave., Washington, D.C.
423. E. C. Rodabaugh, Battelle Memorial Institute
424. C. J. Rogers, Battelle-Pacific Northwest Laboratory
425. G. Rogers, Battelle-Pacific Northwest Laboratory

426. M. A. Rosen, Division of Reactor Development & Technology, USAEC, Washington, D.C. 20545
427. Denny Ross, Reactor Projects Branch, DRL, USAEC, Washington, D.C. 20545
428. B. J. Round, Combustion Engineering, Windsor, Conn.
429. J. H. Russell, Los Alamos Scientific Laboratory
430. M. A. Salmon, IIT Research Institute, 10 W. 35th Street, Chicago, Illinois 60616
431. T. G. Schleiter, DRDT, AEC, Washington
432. V. E. Schrock, University of California, Institute of Engineering Research, Berkeley, Calif. 94704
433-435. Frank Schroeder, USAEC, Washington, D.C. 20545
436. W. Schultheis, General Electric Company, San Jose, Calif.
437. F. R. Schwartz, Jr., North American Carbon, Columbus, Ohio
438. L. C. Schwendiman, Pacific Northwest Laboratory
439. J. T. Sevier, NUS Corporation, 2351 Research Blvd., Rockville, Md. 20852
440. R. C. Shank, Idaho Nuclear Corp., Box 1845, Idaho Falls, Idaho
441. M. Shaw, DRDT, AEC, Washington
442. E. E. Sinclair, DRDT, AEC, Washington
443. M. Siegler, General Electric Company, APED, 175 Curtner Ave., San Jose, Calif.
444. H. Sindt, DRDT, AEC, Washington
445. J. M. Skarpelos, General Electric Company, San Jose, Calif.
446. W. L. Slagle, Phillips Petroelum Company, Idaho Falls, Idaho
447. C. M. Slansky, Idaho Nuclear Corporation
448. W. L. Smalley, ORO, AEC
449. Craig Smith, Nuclear Energy Laboratory, University of California, Los Angeles, Calif. 90024
450. M. L. Smith, Douglas United Nuclear Co., Richland, Wash.
451. W. R. Smith, Sr., General Electric Company, San Jose, Calif.
452. R. E. Soderberg, Walworth Company, Huff Ave., Greensburg, Pa. 15601
453. N. Sowards, Phillips Petroleum Company
454. J. C. Spanner, Battelle-Pacific Northwest Laboratory
455. J. R. Spink, Douglas United Nuclear, Inc., 1760 D. Bldg., Box 490, Richland, Wash. 99352
456-457. F. Standefer, DRDT, AEC, Washington
458. A. Stathoplos, Nuclear Technology Corp., 116 Main Street, White Plains, N.Y. 10601
459. L. E. Steele, DRDT, AEC, Washington
460. D. D. Stepnewski, General Electric Company, Richland, Washington, 99352
461. Virginia Sternberg, Westinghouse Electric Corp., West Mifflin, Pa.
462. H. B. Stewart, Gulf General Atomic, Inc., P. O. Box 608, San Diego, California 92112
463. R. C. Stratton, Apt. 5-J, 98 Garden Street, Hartford, Conn. 06100
464. W. R. Stratton, Los Alamos Scientific Laboratory, P. O. Box 1663, Los Alamos, New Mexico 87544

465-466. D. N. Sunderman, Battelle Memorial Institute
467. W. A. Sutherland, General Electric Company, San Jose, Calif.
468. S. A. Szawlewicz, DRDT, AEC, Washington
469. Jacob Tadmor, Assistant Director, Israel Atomic Energy Commission for Nuclear Safety, Yavne, Israel
470. S. W. Tagart, General Electric Company, San Jose, Calif.
471. M. A. Taylor, DRL, USAEC, Washington, D.C. 20545
472. Madam Thal, STEP, B.P. 2, 91-Gifs/Yvette, France
473. M. D. Thaxter, Lawrence Radiation Laboratory, Berkeley, Calif.
474. C. E. Thomas, Babcock & Wilcox Co., 1201 Kemper Street, Lynchburgh, Va. 24505
475. D. R. Thomas, Commonwealth Associates, 209 East Washington Ave., Jackson, Mich. 49201
476. T. J. Thompson, Massachusetts Institute of Technology, Cambridge, Mass.
477. J. R. Thorpe, First Atomic Ship Transport, Inc., Hoboken, N.J.
478. J. Tribble, Yankee Atomic Electric Co.
479. R. F. Turner, Gulf General Atomic, Inc., P. O. Box 608, San Diego, California 92112
480. USAEC, Division of Compliance, Washington, D.C. 20545
481. USAEC, Division of Reactor Development, Army Reactors, Washington, D.C.
482. USAEC, Division of Reactor Development & Technology, Maritime Reactors Branch, Washington, D.C.
483. M. F. Valerino, Combustion Engineering, Box 500, Windsor, Conn. 06095
484. S. Vandenberg, General Electric Company, San Jose, Calif.
485. H. E. Vann, United Engineers & Constructors, Inc., 1401 Arch St., Philadelphia, Pa. 19105
486. P. O. Vissat, Taylor Forge & Pipe Works, P. O. Box 485, Chicago, Ill.
487-489. R. C. Vogel, Argonne National Laboratory
490. W. R. Voight, Jr., Division of Reactor Development & Technology, USAEC, Washington, D.C. 20545
491. G. E. Wade, General Electric Company, San Jose, Calif.
492. George Wang, Bechtel Corp., Box 3965, San Francisco, Calif. 94119
493. B. Wascher, Babcock & Wilcox Co., 1201 Kemper St., Lynchburg, Va.
494. R. Waterfield, Division of Reactor Standards, USAEC, Washington, D.C. 20545
495. A. S. Waterhouse, Richland Operations Office, Box 550, Richland, Washington 99352
496. George Wehmann, USAEC, Idaho Falls, Idaho
497. L. R. Weissert, Babcock & Wilcox Company, Washington Operations Office, 1725 I Street, N.W., Washington, D.C. 20006
498. D. L. West, General Electric Company, Box 254, San Jose, California 95108
499. G. W. Wensch, Division of Reactor Development & Technology, Washington, D.C. 20545
500. K. T. Whitby, University of Minnesota, Minneapolis, Minn. 55455

501. J. F. White, General Electric Company, NMPO, P. O. Box 132, Cincinnati, Ohio 45215
502. M. J. Whitman, DRDT, AEC, Washington
503. R. A. Wiesemann, Westinghouse Electric Corp., Atomic Power Division
504. E. A. Wiggin, Atomic Industrial Forum, Inc., 850 Third Avenue, New York, N.Y. 10022
505-506. D. Williams, Idaho Operations Office
507. F. S. G. Williams, Taylor-Forge, P. O. Box 485, Chicago, Ill. 60690
508. H. V. Williams, Hartford Accident & Idemnity Co., 690 Asylum Avenue, Hartford, Conn. 06100 (Attention: Myron A. Snell)
509. R. Williams, Spraying Systems, 3201 Randolph Street, Bellwood, Illinois 60104
510-511. T. R. Wilson, Phillips Petroleum Company
512. R. P. Wischow, Nuclear Fuel Services, West Valley, N.Y. 14171
513. W. R. Wise, Ingersol-Rand Co., Research and Development Laboratory, Bedminster, N.J. 07921
514. B. Wolfe, General Electric Company, San Jose, Calif.
515. L. E. Wright, C. F. Brown & Co., Alhambra, Calif.
516. R. R. Wright, Idaho Operations Office
517. R. E. Yoder, Harvard University, 665 Huntington Avenue, Boston, Mass. 02115
518. J. R. Youngblood, Knolls Atomic Power Laboratory, Schenectady, N.Y.
519. W. A. Yuill, Phillips Petroleum Company
520. Sumio Yukawa, General Electric Company, Schenectady, N.Y.
521-535. C. W. Zabel, ACRS, Director of Research and Associate Dean of the Graduate School, University of Houston, Houston, Texas
536. T. A. Zaker, IIT Research Institute, 10 W. 35th Street, Chicago 16, Ill.
537. C. Zangar, Richland Operations Office, P. O. Box 550, Richland, Washington 99352
538. E. Zebroski, General Electric Company, San Jose, Calif.
539. T. Ziebold, Nuclear Engineering Dept., Massachusetts Institute of Technology, Cambridge, Mass.
540. S. M. Zivi, TRW Inc., TRW Systems Group, One Space Park, Redondo Beach, Calif. 90278
541. L. Zumwalt, North Carolina State University, P. O. Box 5636, State College Station, Raleigh, N.C. 27607
542-543. General Atomic Library, P. O. Box 608, San Diego, California (Attention: R. M. Fryar)
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