ORNL NUCLEAR SAFETY RESEARCH
AND DEVELOPMENT PROGRAM
BIMONTHLY REPORT
FOR JANUARY—FEBRUARY 1973
NON-LMFBR PROGRAMS

Wm. B. Cottrell

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Oak Ridge, Tennessee 37830
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PREFACE

The Nuclear Safety Program at ORNL was established when the scope of earlier safety investigations was enlarged in July 1961 (FY 62). At that time and for more than a decade thereafter, the program served a vital role in identifying new areas of research and in coordinating the many safety activities at ORNL both internally and externally. However, in recognition of the increasing maturity of the nuclear program, resulting in the isolation of the remaining problems, as well as increased Commission direction of all work, the Nuclear Safety Program administrative structure at ORNL has become of diminishing value in recent years and is being phased out.

All the individual projects which formerly constituted the Nuclear Safety Program are being continued. However, this will be the last progress report on these activities in this series. The work on the individual projects will continue to be reported — some in document series which are already being established and others by means yet to be established. Listed below are all the individual projects, the names of the project leaders, and the report series in which the project work will be reported. Groups and/or individuals will be added, if necessary, to the distribution of these progress reports in accordance with the RDT report distribution list. Other groups and/or individuals who have valid need for information on one or more projects should contact the project leaders in question for inclusion in that report distribution.

<u>Project</u>	<u>Project Leader</u>	<u>Where Reported</u>
1. PWR Blowdown Heat Transfer Separate Effects	H. W. Hoffman	To be determined
2. HTGR Safety Studies	P. R. Kasten H. J. deNordwall	Gas Cooled Reactor Program Annual Progress Report
3. HSST Program	F. J. Witt	HSST Program Semi- annual Progress Report
4a. Design Criteria for Piping, Pumps, and Valves	W. L. Greenstreet S. E. Moore	Annual Progress Report on Studies in Applied Solid Mechanics
4b. Experimental and Analytical Investigation of Nozzles	W. L. Greenstreet J. P. Callahan	Annual Progress Report on Studies in Applied Solid Mechanics

<u>Project</u>	<u>Project Leader</u>	<u>Where Reported</u>
5. RDT Standards Program	M. Bender W. F. Ferguson	Status Report of RDT Standards (issued bimonthly)
	R. A. Schmidt	Index of RDT Standards (issued bimonthly)
6a. Seismic Research Program*	Jerry Harbour AEC-RDT	
6b. Ignition of Charcoal Adsorbers by Fission Product Decay Heat	A. P. Malinauskas R. A. Lorenz	Reactor Chemistry Division Annual Progress Report
7. Nuclear Safety Information Center (including journal of <i>Nuclear Safety</i>)	W. B. Cottrell J. R. Buchanan	Operation of the Nuclear Safety Information Center (an annual report)

* Management of this project, formerly the responsibility of ORNL, will henceforth be handled directly by RDT.

Previous Progress Reports on the ORNL Nuclear Safety Program:

<u>Period Covered</u>	<u>Report No.</u>
January-June 1962	ORNL-3319
July-December 1962	ORNL-3401
January-June 1963	ORNL-3483
July-December 1963	ORNL-3547
January-June 1964	ORNL-3691
July-December 1964	ORNL-3776
January-June 1965	ORNL-3843
September-October 1965	ORNL-CF-65-11-2
November-December 1965	ORNL-CF-66-1-41
July-December 1965	ORNL-3915
January-February 1966	ORNL-CF-66-3-13
March-April 1966	ORNL-CF-66-5-20
May-June 1966	ORNL-CF-66-7-48
July-August 1966	ORNL-TM-1636
September-October 1966	ORNL-TM-1680
November-December 1966	ORNL-TM-1742
January-December 1966	ORNL-4071
January-February 1967	ORNL-TM-1792
March-April 1967	ORNL-TM-1864
May-June 1967	ORNL-TM-1913
July-August 1967	ORNL-TM-1986
September-October 1967	ORNL-TM-2057
November-December 1967	ORNL-TM-2095
January-December 1967	ORNL-4228
January-February 1968	ORNL-TM-2164
March-April 1968	ORNL-TM-2230
May-June 1968	ORNL-TM-2283
July-August 1968	ORNL-TM-2368
September-October 1968	ORNL-TM-2425
November-December 1968	ORNL-TM-2479
January-December 1968	ORNL-4374
January-February 1969	ORNL-TM-2533
March-April 1969	ORNL-TM-2588
May-June 1969	ORNL-TM-2663
July-August 1969	ORNL-TM-2718
September-October 1969	ORNL-TM-2777
November-December 1969	ORNL-TM-2829
January-December 1969	ORNL-4511
January-February 1970	ORNL-TM-2919
March-April 1970	ORNL-TM-2984
May-June 1970	ORNL-TM-3061
July-August 1970	ORNL-TM-3122
September-October 1970	ORNL-TM-3212
November-December 1970	ORNL-TM-3263
January-December 1970	ORNL-4647

<u>Period Covered</u>	<u>Report No.</u>
January-February 1971	ORNL-TM-3342
March-April 1971	ORNL-TM-3411
May-June 1971	ORNL-TM-3483
July-August 1971	ORNL-TM-3554
September-October 1971	ORNL-TM-3623
November-December 1971	ORNL-TM-3670
January-February 1972	ORNL-TM-3738
March-April 1972	ORNL-TM-3831
May-June 1972	ORNL-TM-3900
July-August 1972	ORNL-TM-3973
September-October 1972	ORNL-TM-4030
November-December 1972	ORNL-TM-4087

ABSTRACT

The accomplishments during January and February 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. Much of this work is in general support of water-cooled power reactor technology, including LOFT and industry programs, and reflects current safety problems such as blowdown heat transfer studies. Other major projects include experimental 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 projects are also conducted in support of the high-temperature gas-cooled reactor (HTGR). Seismic considerations of particular significance in nuclear plant design are evaluated in the seismic program. Several of the above projects are also directly related to another major undertaking, 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. The recent activities of the Nuclear Safety Information Center (NSIC) and the *Nuclear Safety* journal in behalf of the nuclear community are also discussed. Of particular interest in both of these activities is the emphasis on environmental effects.

This will be the last report in this series. Recipients of this document who wish to continue to follow developments on the individual projects reported here will have to make special arrangements as discussed in the Preface.

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SUMMARY

1. LWR SAFETY STUDIES

1.1 PWR-BDHT Separate-Effects Program Scope and Coordination

The status of the PWR-BDHT program was reviewed with RDT on February 26-27, 1973. Among commitments agreed to were (1) the final draft SDD will be made available by ORNL on April 16, 1973; (2) the revised program plan, including experimental and analytical requirements, will be made available by ORNL in June 1973; and (3) the earliest schedule for testing the 49-rod bundle, using direct heaters available from either GE or ORNL, appears to be October 1974. ORNL will reexamine the heater schedule in more detail; the schedule for using indirect heaters has not been determined, but schedule and costs appear to favor the indirect heater based on preliminary information and assuming that qualified indirect heaters are available in the next nine months.

1.2 Fuel Rod Simulation (Task 1)

Two additional blowdown tests were performed with ANC heater S/N-8. The blowdowns were from an initial power level of 66 kW, a system pressure level of ~1500 psia, and a coolant inlet temperature of 550°F. In one test a full area blowdown orifice was used, and in the other test the orifice was only half area. Analysis of the data is incomplete.

1.3 Experiment Plans and Evaluations (Task 2)

Modeling of the single-rod facility with RELAP 3 and THETA 1-B continued. Predicted pressure drops in the test section housing were about half those measured. Preliminary modeling of the multirod THTF is under way.

1.4 Multirod Transient and Blowdown Heat Transfer Experiments (Task 3)

During the remainder of FY 1973, major emphasis will be concentrated on those aspects of the Thermal-Hydraulic Test Facility (THTF) identified

as critical-path items: (1) the development and/or procurement of suitable direct heater rods, (2) the design of the heater-rod bundle, (3) the design of the flow shroud and pressure housing, (4) the design of control modifications for parallel operation of the multiple generators constituting the dc power supply, and (5) the completion of title I instrument application drawings for the several THTF systems (primary flow, bypass flow, cooling water, etc.). A concentrated effort will also be maintained on all supporting documentation.

Inquiries are in process with General Electric Co., San Jose, as a potential supplier of direct electrical heaters. Regardless of the source of heater rods (i.e., outside procurement or in-house fabrication), ORNL will be responsible for procurement of the heater element (tube). Accordingly, we are developing inspection and welding procedures for tapered Inconel-625 tubing.

Drawings for the various components of the test section shroud are now being reviewed. Tentatively, the top grid plate will consist of two plates: a top structural member of Inconel X-750 and a bottom current-carrying member of nickel 201. The use of four 90° angles for the ceramic shroud liners is being investigated as a means for reducing damage due to implosion during blowdown; the base design uses two U-channels.

The bundle spacer grid design, originally conceived as similar to the Westinghouse PWR FLECHT grid, is being modified to increase the bearing area between tube and grid and to constrain the tubes more tightly.

Preliminary schematics showing power, control, and interlock circuits for the dc power supply have been completed and are being reviewed.

Work continued on instrument application drawings for the THTF, since this affects specification of long-lead items.

1.5 Hybrid Computer Heat Transfer Program (Task 4)

During this period hybrid computations were suspended due to severe hardware problems with the analog-digital converter used as interface. The faulty converter is a new unit, and the factory is now assisting in making this equipment operational. In the interim, attention has been

concentrated on our other task of extending the capabilities of the present single-pin hybrid model, with the ultimate objective of creating a core thermal model on the hybrid machine.

2. HTGR SAFETY STUDIES

2.1 Reactions of Steam with Graphite

Comparison of experimental data reported for the studies made at 1000°C with H-327 graphite at steam partial pressures of 0.0450 and 0.0700 atm with results reported earlier for 0.095-atm steam partial pressure indicates a complicated dependence upon this parameter. It may be necessary to postulate that surface area development is due primarily to creation of new capillaries.

2.2 Fuel Integrity and Fission Product Release

A mathematical model of carbon transport is being used to study Tri- α - and Bi- α -coated particles. Inhibition of cesium release by addition of Al_2O_3 to the kernel is being studied.

2.3 Fission Products in the Coolant Circuits

The experiments for studying the desorption of iodine from HTGR coolant circuit materials in a flowing helium-steam-hydrogen atmosphere have resumed. Measurements of fission gases in the primary-coolant circuit of the Peach Bottom HTGR were made with the computer-gamma spectrometer system. A Peach Bottom fuel element was received, and examination is in progress.

2.4 HTGR System Safety Analysis

Preparation of our HTGR Safety Program Plan continues, and detailed proposals for FY 74 were prepared.

3. HEAVY-SECTION STEEL TECHNOLOGY PROGRAM

3.1 Program Administration and Procurement (Task A)

Two technical reports were published, two intermediate test vessels (6 in. thick) were received, and eight subcontracts are in force.

3.2 Material Inspection and Control (Task B)

The fabrication of the remaining 6-in.-thick intermediate test vessels remains a significant procurement item monitored under this task.

3.3 Variability in Plate, Heat-Affected Zones, Welds, and Forgings (Task D)

Several studies using precracked Charpy specimens were carried out. One in particular demonstrated the slow stable crack growth initiated at the beginning of the maximum load.

3.4 Fracture Mechanics Investigations (Task E)

Only five specimens remain to be tested in the investigation of pop-in crack arrest. An extensive evaluation will be undertaken once these results are obtained.

An investigation of the lower-bound fracture toughness of submerged-arc weld and an evaluation of plane strain fracture theories are in progress.

3.5 Fatigue and Crack Propagation (Task G)

Water reactor environments were found to be the major cause of increased fatigue crack growth at low cycling rates. No effect of cycling rates was found from tests in air, since tests in air at all cycling rates gave the same results as water reactor environments at high cycling rates.

3.6 Irradiation Effects (Task H)

Extensive fracture toughness and lower-bound fracture toughness testing on irradiated plate and weldment have been performed. In all cases high toughness levels were obtained at temperatures significantly less than operating temperatures. The irradiation of 4T compact-tension specimens continues. The nominal crack line temperature is 550°F.

3.7 Simulated Service Test (Task K)

Preparation is under way to test intermediate test vessels 5 and 6, and new techniques for growing flaws and monitoring flaw growth are being developed.

3.8 Specific Safety Research (Task L)

The investigations of extent of ductile pipe rupture and toughness of piping steels continue.

4. DESIGN AND ANALYSIS OF NOZZLES, PIPING, PUMPS, AND VALVES

4.1 Design Criteria for Piping, Pumps, and Valves

The ORNL piping program — Design Criteria for Piping, Pumps, and Valves — is part of a cooperative effort between the USAEC and private industry to study the structural behavior of piping system components and to develop basic stress analysis information for use in writing and improving design codes and standards. The ORNL program consists of carefully coordinated analytical and experimental stress analysis studies on piping system components such as straight pipe, lug attachments, elbows, tees, reducers, transition joints, and welds. Stress indices developed for many of these components have been incorporated into the ASME Boiler and Pressure Vessel Code. Close cooperation with private industry is maintained through the Pressure Vessel Research Committee (PVRC) of the Welding Research Council.

Experimental stress analysis tests are currently being conducted on 24-in. ANSI standard B16.9 tees under subcontract at Combustion Engineering. Elastic-response tests and low-cycle fatigue tests have been completed for three tees, and the low-cycle fatigue test of the fourth tee was started this period. The tee (T-12) is being pressure cycled between 100 and 1800 psi, and we expect the test to be completed in about two months (20,000 to 30,000 cycles). The fifth tee (T-16), presently being instrumented with strain gages for conducting the elastic-response tests, should be ready for testing within the next two months.

Finite-element computer code development work is proceeding according to schedule at the University of California, Berkeley. Main frame modifications to the elastic-response computer program, B16.9 TJ/SA, to accommodate elastic-plastic constitutive equations were completed this period and checked out. Constitutive equations for elastic-perfectly plastic materials behavior were written into the program and are currently being checked.

Elastic-response tests were completed for the third model in a series of four idealized thin-walled tees (cylinder-to-cylinder models) being tested at ORNL. All the data have been examined, and a number of wiring identification errors have been corrected. Comparisons between the experimental data and results from finite-element calculations were also completed, and stress plots and graphic illustrations are being prepared for the final report.

The fourth and last thin-walled tee model (ORNL model 2) planned for testing at ORNL is currently being instrumented. All 152 of the three-gage strain rosettes have been installed, and instrument lead wires have been attached to the inside surface gages (77 rosettes). We plan to attach the outside surface lead wires, close the model, and begin testing within the next two months.

During this period the elastic-response strain-gage tests were also completed for machined elbow model ME-4. This model, which was eccentrically bored and ovaled, developed about twice the maximum stress per unit load as the corresponding circular model, ME-3. Deflection-measurement tests will be completed during the next period.

The final report on an analytical study to develop stress indices for lug attachments to straight pipe for thrust and moment loadings was completed and is being reproduced for general distribution. The draft of a second report on stress indices for other loadings on lugs was also completed.

4.2 Experimental and Analytical Investigations of Nozzles

Experimental and analytical studies of single nozzles and clusters of nozzles attached radially and nonradially to flat plates, spherical shells, and cylindrical shells are continuing. The 7 7/8-in.-OD radial nozzle model 807-003 has been tested at the University of Tennessee and is being used to conduct acceptance tests on a new data acquisition system. Model 802-223, the third and final 2 5/8-in.-OD, 22 1/2° nonradial nozzle model, is ready for testing, and model 451, the only 7 7/8-in.-OD, 45° nonradial nozzle configuration scheduled for testing, is being fabricated at the University of Tennessee. Part A of a three-part report of the test results of the 2 5/8-in.-OD, 22 1/2° nonradial nozzle model 802-222 has been published. At Auburn University, all nozzle loadings planned for the flat-plate model having a five-nozzle cluster have been completed.

A final check of experimental data for ORNL thin-shell cylinder-to-cylinder model 4 has been completed, and comparisons have been made with analytical results. The joint ORNL-Battelle/Columbus Laboratories parametric study of inside and outside reinforcement for nozzle-to-spherical shell attachment is being extended to include a study of primary and secondary stresses.

5. RDT STANDARDS PROGRAM

5.1 Program Objectives and Activities

The program was managed through continued review of standards prepared by others, processing and control of publication of approved documents, and assignment of numbers for new standards proposed for preparation. Support to the American National Standards Institute (ANSI) continued.

5.2 Mechanical Equipment and Systems

Work continued in preparation and review of drafts and amendments for ASME code supplement and air cleaning standards. One amendment to a code supplement standard was approved and issued.

5.3 Instrumentation, Controls, and Electrical

Work continued in preparation, review, and publication of standards and amendments; two amendments and one new standard were published. ORNL review of four standards was completed.

5.4 Quality Assurance and Procedures

Participation in ANSI N45 committee and subcommittee activities and in the Nuclear Division Committee for Interface of the American Society for Quality Control (ASQC) continued. Two proposed amendments to RDT standards were prepared and submitted to RDT for approval.

5.5 Materials and Fabrication

One new standard, six revisions, and nine amendments were approved and issued; and four proposed amendments, three revisions, and four new standards were completed and submitted to RDT for approval.

6. GENERAL NUCLEAR SAFETY STUDIES

6.1 Seismicity and Seismic Resistance of Nuclear Power Plants

Work under the SW-AJA soils study subcontract has continued on the development of a method for determining the dynamic shear modulus in situ over a wide range of strains. Limitations on the loads that can be applied to the down hole continuous vibrating probe concept for the determination of this dynamic property have led to preliminary investigations of impact techniques. Shake-table tests have been initiated on sand samples as part of the multidirectional shaking to study liquefaction of

cohesionless soils. Work has continued on the preparation of a report on the recommended procedures for evaluating the seismic response of soil columns. ORNL involvement in the seismic program will be terminated this fiscal year.

6.2 Ignition of Charcoal Adsorbers by Fission Product Decay Heat

The apparatus for experimentally determining the consequences of the radioactive decay energy of iodine adsorbed on charcoal is being tested in the hot cell. Tests of the decomposition products from heating PdI_2 in air have shown several percent particulate iodine and essentially no methyl iodide.

7. NUCLEAR SAFETY INFORMATION

7.1 Nuclear Safety Information Center

The Nuclear Safety Information Center (NSIC) was established by the Commission to collect, evaluate, and disseminate nuclear safety information to governmental agencies, research and educational institutions, and the nuclear industry. The Center's basic computer file now contains references on over 72,100 documents. One new NSIC report has been published and may be purchased from NTIS. The 1970 Directory of Personnel Involved in the Development of Nuclear Standards in the United States (ORNL-NSIC-75) is currently being revised and updated. During the reporting period, NSIC responded to 261 inquiries, of which 151 involved the technical staff.

7.2 Technical Progress Review *Nuclear Safety*

The January-February 1973 issue of *Nuclear Safety*, a bimonthly Technical Progress Review sponsored by AEC's Office of Information Services, was distributed in February. All the topical articles, prepared by authors outside ORNL, discussed earthquake-resistant design of engineering structures, regulation of the environmental effects of nuclear power plants, probability analysis applied to LWR LOCAs, human engineering

factors in control-board design, and safety experience gained from operation of France's sodium-cooled Rapsodie reactor. Also included were the regular staff-prepared articles on safety-related occurrences, R&D project progress summaries, general administrative activities, action on projects in regulatory review, and proposed rule changes. Several miscellaneous announcements, book reviews, etc., were included.

1. LWR SAFETY STUDIES

1.1 PWR-BDHT SEPARATE-EFFECTS PROGRAM SCOPE AND COORDINATION

(AEC Activity 04 60 80 02 1.)

H. W. Hoffman C. G. Lawson

Studies within the scope of the PWR Blowdown Heat Transfer Separate-Effects Program emphasize programmed LOCA-type depressurizations in large (7×7) arrays of 12-ft-heated-length electric rods simulating nuclear fuel pins. Results will provide a measure of the behavior of rod assemblies experiencing blowdown transients as well as input for testing and upgrading existing analytical procedures for describing such transients. This work is supported by experiments evaluating the effectiveness of electric heaters of several types as simulators for nuclear fuel rods and by appropriate analyses.

Program Status

At a meeting with RDT on February 26-27, 1973, the following items were agreed on.

1. The RDT-ORNL representatives agreed to proceed concurrently to examine indirect as well as direct methods for heating the single rods and the 49-rod bundle.

2. ORNL will explore with Watlow and/or Westinghouse the availability of a full-power, 12-ft-long indirect heater for single-rod testing and will discuss their findings with RDT by March 9, 1973.

3. ORNL will continue to explore fabrication of direct heaters by GE and will review and approve the revised GE specifications by April 2, 1973, assuming input from GE by March 22, 1973. The revised GE specification will form the basis for cost and schedule estimates by GE to be made available by April 16, 1973.

4. A decision regarding heater fabrication method (direct vs indirect) should be made before either drawing tubes or contracting with GE. The decision is required probably by mid-summer 1973.

5. The small loop at ORNL will be made capable of testing 12-ft indirect heaters by May 15, 1973.

6. Comments issued by RDT on the draft SDD form the initial basis for proceeding with remaining procurement actions, with the following exceptions:

- a. ORNL will furnish results of an analysis of the THTF primary heat exchangers by March 9, 1973. This analysis is a control document requiring RDT approval for procurement of the heat exchangers.
- b. ORNL will provide a discussion by March 30, 1973, of valve procurement plans, applicable standards, and justification for procurement of spare valve operator parts. This discussion requires RDT approval and controls valve procurement actions.
- c. The ORNL preliminary draft MRD constitutes an acceptable basis for proceeding with preparation of the final draft MRD to be made available by May 1, 1973, and which will include improved symbol definition, keying of instruments to system location, instrument error analysis, and results of RELAP 3 studies. The MRD is a document requiring RDT approval and controls procurement actions on the data acquisition system.

7. The final draft SDD will be made available by ORNL on April 16, 1973.

8. The revised Program Plan, including experimental and analytical requirements, will be made available by ORNL in June 1973.

9. It is anticipated that the final drafts of the MRD, Program Plan, and SDD will be circulated for review and comment to a wider audience, including additional groups within RDT, REG, vendors, and the ERC.

10. RDT will provide guidance on the distribution list for informal monthly progress reports.

11. The earliest schedule for testing the 49-rod bundle, using direct heaters available from either GE or ORNL, appears to be October 1974. However, ORNL will reexamine this schedule in more detail. The schedule for using indirect heaters has not been determined, but schedule and costs appear to favor the indirect heater, based on preliminary information from Watlow and GE.

1.2 FUEL ROD SIMULATION (TASK 1)

(AEC Activity 04 60 80 02 1)

P. P. Holz R. L. Moore G. G. Lawson

This task considers the extent to which electrical heater rods experiencing blowdown transients simulate nuclear fuel rods in a similar environment. Single heater rods of both direct (ohmic wall heating) and indirect (internal resistance element heating) types will be tested in the Forced-Convection Test Facility (FCTF) to determine such factors as time to critical heat flux (CHF), cladding-temperature response, heat-flux variation with surface-to-coolant ΔT , and post-CHF heat transfer patterns. Results from the test sets (separate experiments utilizing a direct heater and an indirect heater designed for the same service) will be compared with the behavior predicted for a nuclear fuel pin.

Heater Rod Tests

Two additional blowdown runs were performed with ANC indirect heater S/N-8; these tests are identified and described in Table 1.1. The heater

Table 1.1. Additional Blowdown Tests with ANC Heater S/N-8

Identification Run	Heater	Rod Power		System Pressure (psia)	Coolant Inlet Temp. (°F)	Mass Flow at Blowdown (gpm, approx)	Blowdown Time (sec, approx)	Remarks
		%	kW					
73-2-1	S/N-8	100	66	1500	550	11.5	20	<i>a</i>
73-2-2	S/N-8	100	66	1500	500	11.5	40	<i>b</i>

^a Full-area blowdown orifice, 0.970 in. diam.

^b Half-area blowdown orifice, 0.685 in. diam.

operated without incident in each test. Analysis of the data is incomplete.

Instrumentation

Design is proceeding for the installation of a gamma densitometer and a drag-disk meter. As noted previously,¹ the densitometer, complete with readout instrumentation and power supply, was borrowed locally. Specifications for the drag-disk probes have been prepared, and related instrumentation will be obtained from other ORNL programs.

Reference

1. ORNL Nuclear Safety Research and Development Program Bimonthly Report for November-December 1972, Non-LMFBR Programs, USAEC Report ORNL-TM-4087, Oak Ridge National Laboratory, pp. 6, 7.

1.3 EXPERIMENT PLANS AND EVALUATIONS (TASK 2) (AEC Activities 04 60 80 02 1 and 04 40 03 01 1)

C. G. Lawson	J. P. Sanders
R. F. Bennett	H. W. Hoffman

This task provides design guidance and analytic support for the experimental tasks of the PWR-BDHT Separate-Effects Program. Particular attention is to be given to the modeling of the ORNL blowdown facilities and to the interpretation of the data obtained therefrom.

Forced-Convection Test Facility

Modeling the Forced-Convection Test Facility (for single-rod blowdown tests) is under development in order to permit extracting the test data from blowdown experiments in this facility and interpreting the results. This model will also serve to test the hybrid application to blowdown analysis (see Sect. 1.5) and indicate future directions for

computer expansion and procedure extension. Work on the modeling with both RELAP-3 and THETA 1-B continued. The calculated pressure trace was improved by the inclusion of the volume between the rupture disks and the critical-flow orifice. Predicted pressure drops were about half those measured.

Flow and enthalpy values at the loop 1 test channel boundaries as determined by RELAP calculations have been input to THETA. Resultant predictions of rod-surface temperature disagree with those estimated directly by RELAP. Thus the transient THETA calculation predicts the onset of film boiling and a consequent sharp rise in rod temperature. This contradicts both the RELAP results and the experimental evidence.

Thermal-Hydraulic Test Facility

Modeling the Thermal-Hydraulic Test Facility (for multirod blowdown tests) is being undertaken to translate reactor LOCA boundary conditions to the experiment and to interpret the resulting data. This effort will depend heavily on existing digital computer programs. Interfaces with the authors and users of the several available codes will assure not only the necessary rapid information exchange for experiment guidance and code development but also will provide the necessary perspective on the role of these studies in the national effort.

Preliminary modeling of the multirod THTF is under way.

1.4 MULTIROD TRANSIENT AND BLOWDOWN HEAT TRANSFER EXPERIMENTS (TASK 3) (AEC Activity 04 60 80 02 1)

R. E. Helms	R. H. Chapman
C. G. Lawson	M. I. Lundin
R. L. Moore	H. W. Hoffman

This task examines experimentally the thermal-hydraulic performance of large bundles of electrically heated rods during LOCA-type depressurization transients. A separate-effects approach is followed in which

bundle response to imposed boundary conditions, rather than to flow loop behavior, is observed. Initial emphasis is placed on the time to critical heat flux (CHF), on the CHF magnitude, and on the immediate post-CHF heat transfer (period prior to ECC injection) associated with blowdown transients in 12-ft-heated-length, 49-rod bundles operating at typical PWR power and pressure level and coolant velocities and enthalpies. The test bundle resembles, but in its limited radial extent is not, a model of a reactor core; neither does the system, by virtue of the decoupling of test bundle and flow loop, model a reactor system. The Thermal-Hydraulic Test Facility (THTF) is being modified to create a large, versatile test bed satisfying the needs of this task.

Test Facility Modifications

During the remainder of FY 1973, major emphasis will be concentrated on these identified critical-path items needed for modifying the THTF: (1) the development and/or procurement of direct heater rods, (2) the design of the heater-rod bundle, (3) the design of the flow shroud and pressure housing, (4) the design of control modifications for parallel operation of the multiple generators constituting the dc power supply, and (5) the completion of title I instrument application drawings for the several THTF systems (high-pressure primary system, demineralized coolant water system, etc.). A concentrated effort will also be maintained to identify long-delivery items and to produce the supporting documentation.

Test Section

The availability of both direct and indirect heaters is being examined to determine which type of heater may now be available to meet the THTF schedule and program requirements. Inquiries are in process with the General Electric Co., San Jose, as a potential supplier of direct electrical heaters.

Cost and delivery of indirect heaters are being obtained as input to making the final decision for the heaters needed for the test section. Computational procedures for determination of the magnetic field and force

vectors on the corner rod of a rectangular array with a uniform transverse current distribution were received from Karl Roesch, Inc. A source listing of a FORTRAN-IV computer program to effect these calculations was also included. Roesch presented results for a 7×7 square array having a rod-to-rod pitch of 0.563 in. (1.43 cm) and a uniform transverse current distribution.

In performing the calculations the following assumptions were made: (1) the current flow through each rod is taken as equivalent to current flowing through an infinitesimal line located at the rod center; (2) any effect of the metal shroud on the induced magnetic field is ignored; and (3) end effects at the top and bottom of the bundle are ignored.

Figure 1.1 shows the resultant magnetic force developed on a corner rod as a function of current per rod. At maximum power the current per rod should not exceed 1000 A, and the corresponding force on the corner rod should not be greater than approximately 10.6 lb_f/ft. Cost estimates for two designs of the Al₂O₃ (99.7% pure) ceramic shroud liners were received. These designs include the standard set of two channel-shaped sections and also a set of four corner-shaped sections for which interchangeability and annealing were specified.

Thirteen drawings of various components of the test section shroud assembly were received from Karl Roesch, Inc. Tentative design of the top grid plate now consists of two plates, a top structural support member made of Inconel X-750, and a bottom current-carrying member of nickel 201. The two-plate composite design was adopted because none of the commercially available materials that were investigated, with the exception of nickel 201, had high-temperature strength combined with low electrical resistivity, which is required for this design. Blowdown forces as well as the constant rod differential pressure force of ~23,000 lb must be considered in the design of the top grid structure. Plate deformation rather than strength is the limiting design parameter.

The shroud tube cover plates were increased from 1/4 to 1/2 in. in thickness to resist an external differential pressure of 1200 psi during blowdown operation. Although type 316 stainless steel has usually been specified for the shroud material, we are considering 17-4PH and

Fig. 1.1. Resultant Magnetic Force Developed on the Corner Rod as a Function of Current per Rod. The transverse power profile is assumed to be uniform, and the direction of the force is toward the center of the bundle.

other martensitic materials since their thermal expansion is much lower and therefore more compatible with our blowdown operating conditions.

Regardless of the source for obtaining direct heater rods (i.e., outside procurement or in-house fabrication), ORNL will be responsible for procurement of the tapered Inconel 625 tube used to fabricate the heater

rods. Accordingly, inspection procedures for tapered tubes are being developed. To date, direct heaters for blowdown heat transfer experiments have used stainless steel tapered tube, and no vendors have produced Inconel 625 tapered tubes. Therefore, to develop the preliminary inspection procedures for the Inconel 625 tapered tubes, several tapered stainless steel tubes were obtained from the Superior Tube Co., and tests are being performed to determine the variables associated with the inspection of these tubes. The inspection procedures for the Inconel 625 tapered tubes will include methods for wall thickness and eccentricity determinations, flaw and discontinuity detection, heat generation variations, and resistivity measurements. Development of procedures for the circumferential butt weld necessary to make the 12-ft-long heater is continuing, with emphasis on the welding parameters and on the influence of both a metallic insert and the ceramic support body. Brazing procedures are also being developed to make the rod extension to tapered tube joint.

Drawings for the various components of the test section shroud are being revised. The use of four 90° angles for the ceramic shroud liners is being investigated as a means for reducing damage due to implosion during blowdown; the base design uses two U-channels. Tentatively, the top grid plate will consist of two plates: a top structural member of Inconel X-750 and a bottom current-carrying member of nickel 201. The design could change in the event the final decision is to use indirect heaters. Tooling and equipment needed for in-house fabrication are being examined in the event the heater is produced in-house.

The bundle spacer grid design, originally conceived as similar to the Westinghouse PWR FLECHT grid, is being modified to increase the bearing area between tube and grid and to constrain the tube more tightly. Fabrication of the grid by an outside vendor is being investigated.

Valves for the High-Pressure Primary System

Preliminary inquiries have been made to establish cost and delivery of throttling control valves for the high-pressure primary system. Based on these preliminary results, it appears impossible to meet the present schedule for completion of the THTF (July 1, 1974) and purchase these

valves to RDT standards. Present plans are to purchase high-quality ASME N-stamped valves and to examine them prior to installation in the high-pressure primary system, thus minimizing the compromise in the reliability of the valves.

DC Power Supply

Preliminary schematics showing power, control, and interlock circuits for parallel operation at the dc power supply have been completed and reviewed. Construction drawings have been started.

The power decay and voltage control circuits, the commutator flash-over device, loss of synchronous motor, and overspeed device are yet to be specified.

In the event the decision is made to use indirect heaters the present plan for paralleling the four 2.5-MW generators would not be necessary.

Instrument Application Drawings

Preliminary drawings of the instrument application diagrams have been completed for the high-pressure primary system, the demineralized cooling water system, and the pressure-suppression system. These drawings will be mailed for project comments about March 2, 1973. The test section instrument application drawing has been started.

The bid for the FM tape recorder was canceled and will be rebid at a later date. This cancellation is due to failure to receive AEC approval of the specification with the 60-day limit which the vendor placed on acceptance of their bids.

1.5 HYBRID COMPUTER HEAT TRANSFER PROGRAM (TASK 4)

(AEC Activity 04 40 03 01 1)

R. S. Stone F. H. Clark

The hybrid computer heat transfer program has two objectives:
(1) to use the hybrid version of THETA and other hybrid programs as

available in support of the BDHT program, particularly at ORNL, and (2) to create a core thermal model on the ORNL hybrid computer exploiting the capabilities of this machine for easy interaction with the analyst and benefiting from its rapid and stable running characteristics.

Program Status

After two years of good performance, the hybrid began to experience intermittent interface problems in the early summer of 1972. A new converter unit was ordered as backup, and operation continued. Despite repairs, the old converter grew progressively worse, leading to complete dependence on the new unit on its delivery last December. This replacement unit occasionally commits gross conversion errors, however, making results undependable. Vigorous measures are being taken, at factory expense, to correct this situation. Most of the malfunctions have been resolved, and the equipment should be available for use very soon.

Meantime, effort which normally would have gone into hybrid operation has been diverted into speedup of our second objective, creation of a core thermal model. A self-consistent set of equations has been written incorporating phase separation and cross flow. When expressed in a hybrid program this development will permit more realistic single-pin heatup calculations and expansion to calculations of a multipin bundle or core. A technical memorandum has been written, ORNL-TM-4123, Model for Two-Phase Flow in Water, describing the model, and presenting the development of the equations. This memorandum has been submitted for publication and will be available in the near future.

2. HTGR SAFETY STUDIES

2.1 REACTIONS OF STEAM WITH GRAPHITE

(AEC Activity 04 60 01 10 1)

C. M. Blood A. P. Malinauskas*
H. J. deNordwall*

The reactions of steam with various graphites are being studied to develop methods for calculating the effects of steam on the core of an HTGR and the rates of hydrogen and carbon monoxide production. Investigations are centered upon the determination of rate constants and the understanding of the reasons why the observed rate of reaction varies with the extent of reaction. The results will be used in the analysis of accidents involving steam ingress.

Reaction rate data characteristic of H-237 graphite have been obtained at 800, 900, and 1000°C using flowing helium-steam streams containing approximately 0.0950-atm partial pressure steam, and at 1000°C at 0.0450- and 0.0700-atm partial pressure steam as well. In addition to obtaining the rate data, measurements also were made of the surface area (B.E.T. gas adsorption), permeability, and interdiffusion characteristics of the samples. All these data have been reported previously.¹⁻⁴

* Part time.

Two models have been developed to account for the experimental observations. One deals with the chemical kinetics involved and the dependence upon internal geometry, whereas the other is concerned with a description of the manner in which reaction (i.e., graphite removal) alters the geometrical features.⁵ We have used the experimental data to test the latter model.

Reassembly of the experimental apparatus has continued through this period.

Tests of the capillary model which has been proposed to account for the variation of geometrical parameters with burnoff indicate it to be inadequate for a detailed description of the various phenomena involved, even though some of the results of an application of the model appear to be qualitatively correct. For example, on the basis of the model, one predicts that the development of surface area S with the amount of graphite reacted g_R can be described by the relation

$$S^2 = S_0^2 + (4\pi N\ell/\rho_s)g_R, \quad (1)$$

where S_0 is the initial surface area of the graphite specimen, ρ_s is the density of solids, and N is the number of identical capillaries of length ℓ and radius r contained within the septum. The model assumes N and ℓ to remain constant during the course of reaction; hence Eq. (1) predicts S^2 to vary linearly with g_R . This turns out to be a fairly accurate expression for describing the experimental data over the range 0 to 10% graphite removal, although curvature in plots of S^2 vs g_R has been noted.

Equation (1) was derived in terms of the capillary model by assuming that the effect of reaction is the uniform enlargement of the pore radii, and a direct test of this assumption could be had by considering the results which were obtained for the viscous flow coefficient B_0 and the porosity-tortuosity factor (ϵ/q). The former parameter is obtained from gaseous permeability measurements, whereas the latter is measured via gaseous counterdiffusion experiments. According to the capillary model, one of several types of "effective pore radii" can be calculated from the

expression

$$r = [8B_0/(\epsilon/q)]^{1/2} \quad (2)$$

Values so derived are presented in Table 2.1 for one of the H-327 graphite samples which have been employed in this study. The results are, unfortunately, typical of the data obtained with other specimens; contrary to prediction, the characteristics or effective radii are observed to decrease with graphite removal. Thus to retain the capillary concept, it appears necessary to postulate that the surface area development is due primarily to the creation of new capillaries rather than to capillary enlargement (i.e., N increases with g_R). This aspect is currently being investigated.

Table 2.1. Observed Variation in Geometrical Parameters with Burnoff for H-327 Graphite Specimen 4C

Burnoff (%)	B_0 (cm ²)	$r = [8B_0/(\epsilon/q)]^{1/2}$ (μm)
	$\times 10^{-9}$	
0.0	1.07	12.9
0.2	1.19	12.7
1.2	1.41	12.6
2.4	1.60	12.0
4.1	1.78	10.9
7.6	2.14	9.7

References

1. C. M. Blood and A. P. Malinauskas, ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971, USAEC Report ORNL-TM-3411, Oak Ridge National Laboratory, pp. 45-47, July 1971.
2. C. M. Blood and A. P. Malinauskas, ORNL Nuclear Safety Research and Development Program Bimonthly Report for November-December 1971, USAEC Report ORNL-TM-3670, Oak Ridge National Laboratory, pp. 73-76, March 1972.

3. C. M. Blood and A. P. Malinauskas, ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1972, USAEC Report ORNL-TM-3831, Oak Ridge National Laboratory, pp. 65-69, July 1972.
 4. C. M. Blood and A. P. Malinauskas, ORNL Nuclear Safety Research and Development Program Bimonthly Report for November-December 1972, Non-LMFBR Programs, USAEC Report ORNL-TM-4087, Oak Ridge National Laboratory.
 5. C. M. Blood and A. P. Malinauskas, Nuclear Safety Program Annual Progress Report for Period Ending Dec. 31, 1969, USAEC Report ORNL-4511, Oak Ridge National Laboratory, pp. 84-88, March 1970.
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2.2 FUEL INTEGRITY AND FISSION PRODUCT RELEASE*

(AEC Activity 04 60 01 10 1)

H. J. deNordwall T. B. Lindemer
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J. H. Coobs

The goal of this program is to determine the cause and extent of fission product release from irradiated coated-particle fuel bodies at normal and abnormal reactor fuel temperatures. Knowledge of both the factors determining coating integrity and fission product transport mechanisms is necessary for the analysis of fission product release from fuel bodies and ultimately from the reactor core, since changes in the fraction of fuel with failed coatings can affect fission product transport substantially. Current emphasis is on describing internal carbon transport mechanisms and failure brought about by excessive temperature in an isothermal environment. Transport of fission products at normal operating temperatures must be considered in accident consequence analysis, since the fission product release from a reactor at the time of an abnormal temperature fluctuation would depend upon the initial fission product distribution.

*Work jointly supported by HTGR Base Program.

The information generated will be used in calculating fission product inventories of the core graphite and the coolant circuit and in estimating the changes in those inventories that may occur as a result of accidental temperature changes. To the extent that fission product release from the fuel determines the relatively higher risk inventory of the coolant circuit, these results will also be useful in analyzing the consequences of depressurization accidents and the radiation doses that might be received by maintenance crews.

Mass Transport in HTGR Particles in a Temperature
Gradient (Amoeba Effect)

We are continuing to interpret the very minimal available experimental data for carbon transport by the CO-CO₂ mechanism in oxide particles. Some data are available from a KFA report^{1,2} for carbon transport and subsequent kernel movement in as-coated UO₂ and UO_{2+x} particles. For such a system our mathematical model suggests:³

$$M = \frac{\epsilon'}{q'} kt \frac{dT}{dz} P_{CO} \frac{P_{CO} + P_{CO_2}}{\bar{P}_{CO} + P_{CO_2} + P_g},$$

in which

M = carbon transported, μm ,

ϵ'/q' = ratio of observed diffusion coefficient to that in free space, dimensionless,

k = rate constant for the C-CO-CO₂ system, $\mu\text{m of carbon/hr-(}^\circ\text{C/cm)-(atm-CO + CO}_2\text{)}$;

$\log k = -7.837 + 9452/T$,

t = time, hr,

dT/dz = temperature gradient, $^\circ\text{C/cm}$,

P_{CO}, P_{CO_2} = pressure of CO or CO₂, atm,

P_g = pressure of residual coating gases, usually 1 atm.

In the KFA experiments at 1700°C the as-coated value of O/U was 2.005; upon subsequent heating, the O/U would be reduced to 2.000 and

the hyperstoichiometric oxygen would react with carbon to form CO. The value of P_{CO} can be calculated from the ideal gas law; the ratio of free volume in the buffer layer to the superficial volume of the kernel calculated from the photomicrographs in the report, assuming a 50%-dense buffer layer, was 0.82. The value of P_{CO} was therefore 40.0 atm. In the KFA experiments, they observed metallographically that $M = 70 \mu\text{m}$ at 1700°C in 480 hr at $dT/dz = 200^\circ\text{C}/\text{cm}$. We can predict what M should be from the above equation with $\epsilon'/q' = 0.017$ obtained from our analysis of the Dragon Project empirical correlation;³ we calculate $M = 58 \mu\text{m}$, which is in good agreement with the observed behavior.

The KFA work with as-coated kernels having an O/U of 2.000 was performed over the temperature range of 1400 to 1800°C for times up to 720 hr at $dT/dz = 200^\circ\text{C}/\text{cm}$. The value of P_{CO} is not known, but values of 1 to 10 atm are indicated from the measurements of Bildstein and Strigl. At 1400°C the above equation would give 8 to $160 \mu\text{m}$ of carbon transported, depending on whether the low or high value of P_{CO} is used, while at 1800°C , 0.6 to $6.0 \mu\text{m}$ is predicted. It is doubtful whether $<10 \mu\text{m}$ of transported carbon would be observable metallographically, because the carbon transported to the cold side would probably just densify the buffer layer. In view of the uncertainty in the value of P_{CO} , it would appear that the lack of observable transport is not inconsistent with the predictions of the above equation.

We are currently applying the mathematical model to three different situations: to Triso and Biso particles during irradiation and to as-coated particles out of pile. To date, the Triso model used for the Dragon Project data³ and the as-coated model analyzed above appear to agree with experimental work; the Biso in-pile particle model does not agree to date, and we are working on this one as information becomes available.

Chemical Characteristics of Fuel and Fission Products

We have examined the thermodynamics of certain cesium-containing systems to ascertain the conditions for release of this element through pyrocarbon coatings. The phase diagram for the $\text{Cs}_2\text{O}-\text{Al}_2\text{O}_3-\text{SiO}_2$ system

contains some compounds with melting temperatures as high as 1700°C. This behavior prompted us to estimate the possibility that this system could be used to hold cesium in an oxide particle in a chemically combined form. This possibility was studied in detail on the basis of available data.

We conclude that even though the addition of Al_2O_3 to the oxide kernel may reduce P_{Cs} below that in equilibrium with the liquid cesium phase that may be formed from the fissioning of the fuel, it apparently will not reduce P_{Cs} below that necessary for the formation of intercalation compounds with carbon and thus probably not prevent the diffusion of cesium through the pyrolytic carbon coating of a Biso particle. However, other work indicates that Al_2O_3 is beneficial in the retention of strontium and barium within the coated oxide particle.

For that reason we have started preparation of sol-gel, highly enriched UO_2 containing additives that may be effective in retaining Cs, Sr, and/or Ba within the particle during irradiation. We have prepared 85% UO_2 -15% Al_2O_3 spheres (mole % basis) and have successfully dried the material. Other kernel compositions and heat treatments are to be performed so that additive-containing Biso-coated kernels will be available for future irradiation capsules.

Analysis of Data on Fission Product Behavior

Fission product release data obtained during the previous year are being systematically reviewed in order to select the best results for devising an improved diffusion model that will accurately describe the release behavior. Preliminary work covers two distinct areas: (1) documentation and augmentation of existing data, and (2) review of derivations and testing of various diffusion equations with emphasis on boundary conditions. Immediate objectives are to refine present data-correlation techniques and devise auxiliary experiments if necessary.

When the subject data were first obtained, it was desirable to report them promptly in compact form as "effective" diffusion coefficients. It was assumed that the transient diffusion equation was governed by constant-potential boundary conditions for at least the early release times of the

postirradiation diffusion anneal. Under these conditions, the cumulative release-vs-time curve would start out with a zero slope, bend upward (during the transient), and then approach a straight-line relationship (for steady flow). Extrapolation of this line back to a time t_0 at zero release would give the coefficient D , if the coating thickness ℓ were known, according to the equation:

$$D = \frac{\ell^2}{6t_0}.$$

Furthermore, the effective driving force C_0 in terms of count data could be determined from the slope of the curve.⁴ The most recent experiment with ^{137}Cs at 1600°C (see Fig. 2.1) indicates that depletion at the source boundary takes place as evidenced by the "S-shaped" curve. If t_0 occurs at about 100 hr, the curve would not approach a straight line until about 300 hr, but here the slope of the curve has already started to decrease.

As a first step toward accounting for depletion, diffusion equations incorporating this feature are under consideration. Ultimate application requires an equation for spherical coordinates, but slab geometry turns out to be complicated enough. We have assumed that the slab solution⁵ has the same general form as the spherical solution. Attempts are now being made to concoct a scheme (using the slab solution) whereby preliminary values of D , C_0 , and hopefully a ^{137}Cs partition coefficient can be extracted from the data by comparing integral forms of the solution to the experimental curves in such a way that computations are minimized.

References

1. A. Nauomidis, R. Rotterdam, and B. T. Thiele, Technical Report KFA-IRW-IB-2/72, pp. 13-16, January 1972.
2. See also a previous analysis of this paper in GCR-TU: 72-17, p. 35.
3. T. B. Lindemer and H. J. deNordwall, An Analysis of Chemical Failure of Urania Fuel Particles in the High-Temperature Gas-Cooled Reactor, to be published as an ORNL-TM report.

ORNL-DWG 73-3282

Fig. 2.1. Cesium Release During Postirradiation Anneal of LTI Pyrocarbon-Coated ThO_2 Particle at 1600°C .

4. J. Crank, *The Mathematics of Diffusion*, pp. 48, 95-96, Oxford University Press, New York, 1957.
5. H. S. Carslaw and J. C. Jaeger, *Conduction of Heat in Solids*, p. 128 [Eq. (8)], Oxford University Press, New York, 1959.

2.3 FISSION PRODUCTS IN THE COOLANT CIRCUITS

(AEC Activity 04 60 01 10 1)

F. F. Dyer	W. P. Teichert
J. O. Kolb	M. F. Osborne
H. J. deNordwall*	W. J. Martin

This section is concerned with describing and explaining the distribution and adhesion of fission products in the coolant circuits of HTGRs. It includes the activities associated with the HTGR fission product surveillance program.

The results of these studies will be used in the analysis of accidents involving depressurization and coolant composition change, and in testing codes used to predict the inventory and distribution of fission products in the coolant circuit. Recently, experimental and theoretical analyses of fission product distributions in a Peach Bottom element were begun, with the objective of describing the time variation of the source of fission products in the coolant circuit.

Iodine Adsorption and Desorption

Laboratory relocation and equipment modifications were completed, and experimental work has been resumed. A 10-mCi sample of ^{131}I was obtained and converted to the elemental form with 20 mg of normal iodine by the usual procedure: precipitation as PdI_2 , drying, and decomposition of the PdI_2 in vacuum.

* Part time.

Two samples of Fe_3O_4 have been loaded in vacuum at 250°C to about 0.25 monolayer of iodine for use in studying the effect of steam on iodine desorption from Fe_3O_4 . These desorption tests will be run in a flowing atmosphere of helium-0.1% hydrogen-3.0% steam. Adsorbed iodine is measured as a function of temperature in a linear transient ($\sim 1.5^\circ\text{C}/\text{min}$) beginning at 50°C , until >99% of the iodine has been desorbed.

An experimental investigation of the equilibrium adsorption of iodine on Fe_3O_4 at 400°C is proceeding satisfactorily. At the initial iodine source pressure of 10^{-11} atm, the system reached apparent equilibrium at an iodine coverage of about 0.01 monolayer, which is consistent with previous experiments under similar conditions. This experiment will continue to equilibria at successively greater iodine pressures of 10^{-10} , 10^{-9} , and 10^{-8} atm.

Fission Products in Peach Bottom Coolant Circuits

During the period February 12-21, 1973, we took our computer-gamma spectroscopy system to Peach Bottom to measure fission gases in the primary circuit, the total reactor purge system, and the purge from several experimental fuel elements. Measurement on the test element purge lines was a cooperative program between GGA and ORNL. Due to malfunction of the gamma-ray detector, no measurements were made on the test element purge lines. A malfunction of the pressure regulator on the reactor's total purge system also made it impossible to obtain purge-gas samples. Gamma spectra were obtained from primary circuit gas samples collected on charcoal as well as the loop 1 hot-valve cooling line. The hot-valve cooling line is accessible during reactor operation and contains circulating coolant at full pressure. Results of these measurements will be reported as soon as the gamma spectra are resolved.

The fabrication and testing of a replacement diffusion tube sampler (unit 2) for thermowell 205, situated in the loop 1 hot valve, was completed this month. The two samplers currently in the reactor in positions up- and downstream of the steam generator are now scheduled for removal in April.

Fuel Element Postirradiation Analysis

Driver element E06-01 was received in January 1973 and unloaded into cell A, Building 3026-D. This element had received 380 days exposure before its withdrawal from the reactor in April 1972. Examination of the exterior of the fuel element showed no apparent cracks in the graphite, though there were several scratches along the entire length of the element. A sooty deposit was seen on the upper reflector. Measurements of continuity and resistance to ground were made on the thermocouples by the Instrumentation and Controls Division. After removing both thermocouples, the top and bottom reflector sections were cut off, leaving the fueled section of the element. The fuel compacts were then pushed from the sleeve and canned separately for gamma scanning. The spines were canned separately.

Gamma scanning of fuel compacts and other fuel element components is in progress. Gamma-ray-emitting fission products in the driver element are being measured remotely through a collimator by high-resolution gamma spectroscopy. These results will be normalized to absolute amounts of fission products by destructive analysis of a limited number of specimens.

Computer Calculations of Fission Product Transport in Fuel Elements

The fission product transport code FIPER-Q has been developed by Gulf General Atomic to calculate one-dimensional diffusion of fission products in slab, cylindrical, and spherical geometries. GGA uses FIPER-Q to calculate the release to the coolant of hazardous alkali metal and alkaline-earth metal fission products, such as ^{137}Cs , ^{89}Sr , ^{90}Sr , and ^{140}Ba , during the normal fuel cycle and core overheating accident situations. The objectives of our use of FIPER-Q are (1) to determine the validity of the diffusion transport model from fission product concentration distributions in Peach Bottom HTGR test elements, (2) to determine appropriate coefficients under reactor conditions, and (3) to determine the diffusional properties of irradiated HTGR fuel particles from release measurements during annealing experiments.

To familiarize ourselves with the code and also any limitations involved, we have chosen several simplified geometry test cases for which analytical solutions exist. Table 2.2 shows the cases that have been checked to date by checking either the transient concentration profiles or the total fractional release, which is calculated from the time integral of the net current at the outer boundary. For desorption from initially uniformly loaded sources, mesh sizes had to be reduced to 1/400th of the radius or thickness for a solid sphere or slab and to 1/60th of the outer annulus thickness for the two-region sphere to keep errors in the release results below 1%. Small mesh sizes were necessary just in the area where steep concentration gradients existed from the step change in initial concentration.

Table 2.2. FIPER-Q Test Cases

Geometry	Desorption of Initial Inventory (Uniform C at $t = 0$)	Uniform Production, No Decay ($C = 0$ at $t = 0$)
<u>Solid sphere</u>		
$C_b = 0$, all time	a, b, c	c, d
$J_b = K(C_b - C_0)$		
<u>Two-region sphere</u>		
$C_a = 1.0, C_b = 0$, all time	b, c	
<u>Solid cylinder</u>		
$C_b = 0$, all time	a, c	
<u>Solid slab</u>		
$C_b = 0$, all time	a, b, c	

^aTransient concentrations checked.

^bTotal fractional release vs time checked.

^cAsymptotic concentrations checked.

^dTotal fractional release vs time does not agree with analytical solution until ~20% of total release is reached.

An additional case that will be checked against analytical results is the desorption from a solid sphere with a net current at the outer boundary, $J_b = K(C_b - C_0)$, where K is a transfer coefficient and C_b is the outer boundary concentration. Also the solid sphere with uniform generation, radioactive decay, and a zero outer boundary concentration will be examined.

Several FIPER-Q program modifications being investigated are the removal of limitations on calculating an evaporative outer boundary condition on the IBM 360 system and the calculation of the sorption ratio parameter at material interfaces from surface temperatures rather than from an assumed constant value.

2.4 HTGR SYSTEM SAFETY ANALYSIS

(AEC Activity 04 60 01 10 1)

H. J. deNordwall P. Rubel

The objectives of this task are (1) to provide logical guidance in the planning of the HTGR Safety Program and (2) to assess probable courses and hence end points for HTGR accidents. This information, in combination with the descriptive material available from other tasks, will enable one to formulate the risks associated with an HTGR.

Detailed proposals for safety analysis work to be conducted through FY 74 were developed. Effort is generally to be focused on questions of core cooling following reactor shutdown. In addition to being of paramount safety importance, this area is of sufficient scope to demonstrate the effectiveness and need for the analysis methods being proposed. The major tasks will be: (1) to execute decision trees, representing the possible accident main event sequences, for example, following delayed start of auxiliary cooling loops; (2) to execute system fault trees, representing system events such as component failures that may contribute

to the accident primary sequences; (3) to arrange for, conduct, and interpret plant transient analyses which will describe safety-related conditions during accident sequences; (4) to acquire and interpret information relevant to components' response prospects under emergency conditions for all components exposed to the transient; and (5) to perform preliminary probabilistic analyses to gain some insight regarding relative importance of various safety considerations.

Additional editing and rewriting of the Program Plan draft were done. For task area 1, systems and safety analysis, we attempted to clarify the proposal to perform safety evaluations on the basis of central or "best" estimates of the parameters involved, rather than using all pessimistic "upper-limit" assumptions. The central estimates are to be complemented by evaluations of overall problem uncertainty, via error propagation methods.

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3. HEAVY-SECTION STEEL TECHNOLOGY PROGRAM

3.1 PROGRAM ADMINISTRATION AND PROCUREMENT (TASK A)

(AEC Activity 04 60 80 03 1)

F. J. Witt

The assessment of the potential of failure by either bursting or leakage of in-service water reactor pressure vessels is one of the major engineering capabilities the HSST program is seeking to establish. This capability is in the latter stages of evolvement, with current interest focused on testing a series of ten 6-in.-thick flawed pressure vessels. Major complementary efforts include the investigation of the effects of fast-neutron irradiation on the fracture toughness of vessel product forms and fatigue crack growth characteristics under in-service conditions.

Two technical reports were published,^{1,2} two are in publication, and several are in preparation. Eight subcontracts are in force, but final reports are being prepared for three of them.

The major procurement items remain the 6-in.-thick, 39-in.-OD intermediate test vessels. The remaining two vessels of the first order of six vessels were recently received. These vessels, shown in Fig. 3.1, were both fabricated from ASTM A508, class 2 forging steel. The nozzle in vessel V-5 is 21 in. in outside diameter and 6 in. thick; vessel V-6, without a nozzle, contains a longitudinal submerged-arc weld seam intersected by a submerged-arc girth seam weld at the middle of the vessel. Four additional vessels are still in the component stages of fabrication.

Fig. 3.1. Intermediate Test Vessels V-5 (Foreground) and V-6.

The activities of the HSST program are carried out under a series of 12 tasks, 4 of which are completed. The activities under the remaining seven tasks are discussed below.

References

1. J. G. Merkle, A Review of Some of the Existing Stress Intensity Factor Solutions for Part-Through Surface Cracks, USAEC Report ORNL-TM-3983, Oak Ridge National Laboratory, January 1973.
2. J. G. Merkle, An Elastic-Plastic Thick-Walled Hollow Cylinder Analogy for Analyzing the Strains in the Plastic Zone Just Ahead of a Notch Tip, USAEC Report ORNL-TM-4071, Oak Ridge National Laboratory, January 1973.

3.2 MATERIAL INSPECTION AND CONTROL (TASK B)

(AEC Activity 04 60 80 03 1)

The major activity of this task remains the fabrication and inspection monitoring of the intermediate test vessels. The components of the seventh intermediate test vessel were inspected during this reporting period.

3.3 VARIABILITY IN PLATE, HEAT-AFFECTED ZONES, WELDS, AND FORGINGS (TASK D)

(AEC Activity 04 60 80 03 1)

Several series of precracked Charpy specimens were tested as part of the materials investigation of the intermediate vessel tests, and a study was made of the initiation of slow stable crack growth in precracked Charpy

specimens. The results summarized in Fig. 3.2 show that slow stable crack growth initiates at the beginning of maximum load.

3.4 FRACTURE MECHANICS INVESTIGATIONS (TASK E)

(AEC Activity 04 60 80 03 1)

Some 20 edge-notch bend specimens have been tested at Materials Research Laboratory as part of the investigation of pop-in arrest. Five specimens remain to be tested, and the data must be interpreted. An investigation of the lower-bound toughness of submerged-arc weld is in progress at Westinghouse Electric Corporation as well as a study for evaluating plane strain fracture theories.

3.5 FATIGUE AND CRACK PROPAGATION (TASK G)

(AEC Activity 04 60 80 03 1)

The enhancement of fatigue-crack growth propagation at low cycles and in a reactor environment excluding irradiation was found to be due almost totally to the environment, as low cycling crack growth in air gave the same results as high cycling rates in air or in the "reactor" environment.¹ Tests are currently in progress at still lower cycling rates and lower stress intensity values. The reactor-environment tests were performed by Westinghouse Electric Corporation.

Reference

1. ORNL Nuclear Safety Research and Development Program Bimonthly Report for November-December 1972, Non-LMFBR Programs, USAEC Report ORNL-TM-4087, Oak Ridge National Laboratory.
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Fig. 3.2. Slow Stable Crack Growth in Precracked Charpy Specimens Obtained by Stopping the Load at Various Deflections. Specimens tested in the slow-bend mode.

3.6 IRRADIATION EFFECTS (TASK H)

(AEC Activity 04 60 80 03 1)

Extensive fracture-toughness and lower-bound fracture-toughness testing of irradiated specimens continues at Hanford Engineering Development Laboratory (HEDL). A summary of recent data is shown in Fig. 3.3.

The irradiation of the 4T compact-tension specimens continues in the Battelle Research Reactor. The nominal crack line temperature is 550°F. This work is subcontracted to Westinghouse Electric Corporation.

3.7 SIMULATED SERVICE TEST (TASK K)

(AEC Activity 04 60 80 03 1)

Preparation is under way to test intermediate vessels 5 and 6 (see Fig. 3.1). Techniques for fatigue sharpening inside surface flaws have been developed and are being validated in a cylinder prolongation. The extensions on the prolongation are for developing actual procedures for the vessel. Several 1/7-scale models are being prepared for testing. Improved techniques for accurate flaw fatigue growth monitoring are under development.

3.8 SPECIFIC SAFETY RESEARCH (TASK L)

(AEC Activity 04 60 80 03 1)

The investigation of the extent of ductile pipe rupture continues at Battelle Columbus Laboratories. Precracked Charpy specimens from two piping materials (four heats) have been obtained, and testing is in progress.

Fig. 3.3. Unirradiated and Irradiated Fracture Toughness of ASTM A533, Grade B, Submerged-Arc Weldment Materials from HSST Weldment Section 51A.

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4. DESIGN AND ANALYSIS OF NOZZLES, PIPING, PUMPS, AND VALVES

4.1 DESIGN CRITERIA FOR PIPING, PUMPS, AND VALVES

(AEC Activity 04 60 80 03 1)

W. L. Greenstreet S. E. Moore

The ORNL piping program,¹ a USAEC-supported program, is carried out by the Oak Ridge National Laboratory to study the structural behavior of nuclear reactor-piping system components. This program is the U.S. Atomic Energy Commission's portion of a cooperative effort with private industry to develop basic structural response information for critical system piping components, pumps, and valves. The primary objective is to provide information for use in writing and improving design and construction codes and standards for commercial power reactor systems. However, because of the fundamental nature of the investigations a substantial portion of the information is directly applicable to other critical piping systems as well.

The program consists of about 30 carefully coordinated experimental and analytical studies, with many of the individual tasks carried out at universities and industrial laboratories under subcontract to ORNL. At present the experimental work is confined to tees, branch connections, and elbows, while analytical studies are being carried out for most of the standard piping components including straight pipe and auxiliary attachments, socket welding and threaded fittings, reducers, weld joints, transition joints, and flanged joints as well as elbows, tees, and branch connections. The results of these studies are used in evaluating various

2

analytical stress analysis methods and in developing stress indices, flexibility factors, and design rules to be used in the codes and standards.

Close cooperation between the ORNL piping program and the industrial code-writing bodies is maintained through the Pressure Vessel Research Committee (PVRC), and an important aspect of our management responsibility to the AEC is active participation in the work of the PVRC and the ASME code committees. The work under the ORNL program as well as other related work is discussed at regular meetings of these groups. Three committee meetings were held during January and February: the ASME Section III Working Group on Piping on January 8; the PVRC Subcommittee to Develop Stress Indices for Piping, Pumps, and Valves on January 30; and the PVRC Design Division on January 31, 1973. Progress and status reports on the ORNL piping program work were presented at these meetings.

Following is a brief summary of progress made under the ORNL piping program during January and February 1973.

Structural Analysis of Tees

These studies include experimental and analytical stress analyses of a series of 17 ANSI standard B16.9 tees; experimental and analytical stress analyses of a series of four very carefully made idealized thin-shell tee models; and analytical studies of small-branch connections which meet the fabrication requirements of the ASME Boiler and Pressure Vessel Code, Section III.²

The work on B16.9 tees includes experimental elastic stress analysis and low-cycle fatigue-to-failure tests on the 17 tees listed in Table 4.1, photoelastic stress analysis on one-half-scale models of 4 of the tees, experimental thermal stress analysis on 2 of the tees, and the development of finite-element computer codes for both elastic and elastic-plastic stress analysis of B16.9 tees. The photoelastic studies were completed under subcontract at Westinghouse Research Laboratories and reported previously. Experimental stress analysis and low-cycle fatigue tests on five of the tees have also been completed under subcontract at Southwest Research Institute. Work is currently under way at Combustion Engineering, Inc. on the 24-in. tees; at Oak Ridge National Laboratory on the thermal

Table 4.1. Experimental Studies for ASA B16.9 Tees

Tee No.	Manufacturer	Nominal Size (in.)	Nominal Wall Thickness (sched)	Material ^a	Type of Fatigue Test ^b	Type of Photoelastic Test ^c	Thermal Stress Test ^d
T-1	I	6 × 6 × 6	40	SS	M		
T-2	I	6 × 6 × 6	160	C	P		
T-3	I	6 × 6 × 6	40	SS	M		
T-4 ^e	I	12 × 12 × 12	80	C	M		
T-5	II	12 × 12 × 12	80	C	P		
T-6 ^{e,f}	III	12 × 12 × 12	80	C	M	M	8A-T
T-7 ^{e,f}	II	12 × 12 × 12	160	SS	M	P	
T-8 ^{e,f}	II	12 × 12 × 6	40	SS	M	M	8A-T
T-9 ^f	II	12 × 12 × 6	160	SS	P	P	
T-10 ^g	III	24 × 24 × 24	40	C	M		
T-11 ^g	III	24 × 24 × 24	160	C	P		
T-12 ^g	III	24 × 24 × 10	40	C	P		
T-13 ^g	III	24 × 24 × 10	160	C	P		
T-14	III	12 × 12 × 6	40	SS	U		
T-15 ^e	I	12 × 12 × 6	40	SS	M		
T-16 ^g	III	24 × 24 × 24	10	SS	M		
T-17 ^g	III	24 × 24 × 10	10	SS	M		

^aSS = type 304L stainless steel; C = A-106 grade B carbon steel.

^bThe loading for the fatigue test is designated as M for moment loading, P for internal pressure, and U as yet undecided.

^cPhotoelastic analysis with either internal pressure (P) or out-of-plane bending moment loading on the branch (M).

^dSteady-state temperature distribution thermal stress tests.

^eSteel models tested at SWRI.

^fPhotoelastic models tested at Westinghouse.

^gBeing tested at Combustion Engineering at Chattanooga.

stress tests; and at the University of California on the development of finite-element computer programs.

Experimental Studies of B16.9 Tees (Combustion Engineering)

Experimental studies being carried out under subcontract at Combustion Engineering include elastic-response tests and low-cycle fatigue tests on the six 24-in. nominal diameter tees listed in Table 4.1 (T-10 through T-14, T-16, and T-17). The elastic-response tests have been completed for the first four (T-10, T-11, T-12, and T-13), and the low-cycle fatigue tests have been completed for three of these (T-10, T-11, and T-13). The fatigue test of T-12 is currently under way. We plan to conduct the elastic-response tests and fatigue test of T-16 this fiscal year and to test T-17 next fiscal year.

The low-cycle fatigue test of T-12, with a cyclic internal pressure between limits of 100 and 1800 psi, was started during this reporting period. Complete sets of strain-gage data (230 three-gage rosettes) were obtained for the first 21 cycles while the tee was still in the elastic-response loading frame. The test assembly has now been moved to the pressure burst pit, where the fatigue test will be completed. Only a limited amount of strain data will be obtained during the remainder of the fatigue test. Several strain gages, located in regions of maximum stress, will be monitored and read during each pressure cycle. We expect the fatigue test of T-12 to be completed in about two months.

The test assembly for the stainless steel tee, T-16 (24 × 24 × 24-in., sched 10), is currently being prepared for testing. During this reporting period the assembly was mounted in the elastic-response loading facility, and the inside surface strain gages (115 three-gage rosettes) were connected to the data acquisition system. The gages and lead-wire connections are presently being checked out prior to making the final closure weld on the test assembly. We plan to install the outside surface strain gages and start the elastic-response test during the next two months.

Finite-Element Stress Analysis of B16.9 Tees
(University of California)

Concurrently with the experimental studies of tees, finite-element computer code development is under way at the University of California, Berkeley. The objective of this work is to develop computer programs for both elastic and inelastic (plasticity) stress analysis of ANSI standard B16.9 tees. The computer codes will be used at ORNL primarily for conducting analytical parameter studies after they have been fully validated by comparing calculated results with experimental data. The elastic analysis computer program, identified as B16.9 TJ/SA, was delivered to ORNL in July. It is now operational on the ORNL IBM 360 computer, and output plotting packages have been written. Currently the program is being used at ORNL to make comparisons with experimental data obtained from the 12-in. B16.9 tees tested earlier at Southwest Research Institute (tees T-4, T-6, T-7, T-8, and T-15).

The elastic-plastic finite-element program, B16.9 TJ/EPsA, currently being written at Berkeley³ is expected to be completed by the end of FY 1974. During this period expansion and modification of the basic elastic program (B16.9 TJ/SA) to remove the superposition feature of the code were completed and checked out. Since superposition is not valid for the elastic-plastic case, certain loadings require that the entire tee-pipe assembly must be analyzed rather than the one-quarter symmetry section used for the elastic analyses. This requires additional computer core storage and/or more complex dynamic storage. We also added constitutive equations for elastic-perfectly plastic materials behavior and operated the program successfully. These latter modifications are currently being checked out.

Idealized Thin-Walled Tee Studies (ORNL)

The four idealized thin-walled tees (cylinder-to-cylinder models) listed in Table 4.2 are being experimentally and analytically stress analyzed at ORNL to obtain high-quality data for use in developing better analytical design methods for nozzles in cylindrical pressure vessels and for branch connections in piping systems. We reported earlier⁴ that all

Table 4.2. Thin-Shell Cylinder-to-Cylinder Tee Models

Model No.	Major Dimensions ^a (in.)				Dimensionless Parameters			
	D_0	d_0	T	t	d_0/D_0	D_0/T	d_0/t	s/S^b
1	10.0	5.0	0.1	0.05	0.5	100	100.0	1.00
2	10.0	10.0	0.1	0.1	1.0	100	100.0	1.00
3	10.0	1.29	0.2	0.168	0.129	50	7.7	0.154
4 ^c	10.0	1.29	0.2	0.064	0.129	50	20.2	0.40

^a D_0 is the outside diameter of the larger cylindrical shell, d_0 is the outside diameter of the nozzle (the smaller cylinder), T is the wall thickness of the cylindrical shell, and t is the wall thickness of the nozzle.

^bThe dimensionless parameter s/S is the ratio of membrane stress in the nozzle to the membrane stress in the cylindrical shell due to internal pressure loading; in terms of other parameters, $s/S = d/D \times T/t$.

^cModel 4 was made by boring out the nozzle of model 3.

the test work including elastic-response strain gage and deflection-measurement tests were completed for model 4, although a number of apparent strain-gage wiring errors had been discovered. After the tests were completed, the model was opened, and all the strain-gage wiring was traced and checked through the data acquisition system. A number of gage identification errors were corrected and appropriate corrections made in the data. Comparisons between the experimental data for all 13 loading conditions (internal pressure; in-plane, out-of-plane, and torsional moments; and in-plane, out-of-plane, and axial force loadings on the branch and on the run) have now been completed. The stress plots and other graphic illustrations are currently being prepared for the final report.

Instrumentation lead wires are presently being installed on model 2, the last model in this series to be tested. This model is instrumented with one hundred fifty-two 0.030-in. gage-length three-gage strain rosettes, 77 on the inside surface and 75 on the outside surface. All the inside surface instrumentation (gage installation and lead wire attachment) has been completed. During the next two months we plan to close the model, attach the outside surface lead wires, and begin the series of elastic-response tests.

Elbows, Curved Pipe, and Miter Bends

Elastic Response of Machined Elbows (ORNL)

Elastic-response tests were completed for machined-elbow model ME-4 (see Table 4.3), and the strain-gage data were diagnostically examined for linearity and "bad" gages. We expect to complete the deflection-measurement tests within the next two or three weeks. With the exception of the low-cycle fatigue test of model ME-2, to be done next fiscal year, this will complete the experimental work for machined-elbow models as originally planned.¹ Analytical stress analyses of the different models are currently under way.

Model 4, mounted in the loading frame as shown in Fig. 4.1, was instrumented with 88 three-gage rosettes, 72 on the outside surface and 16 on the inside surface, and was tested with internal pressure and in-plane, out-of-plane, and torsional moments applied through the flange on the upper end of the model. Moment loadings were applied both individually and in combination with internal pressure. Whereas previous models had been tested with 188,160 in.-lb of moment and 1305 psi internal pressure, the maximum internal pressure for this model (ME-4) was limited to 600 psi because of the higher stresses induced by the effects of out of roundness. The magnitudes of the combined loadings were also limited for the same reason. For the in-plane bending with internal pressure test combination the maximum loads were: (1) 188,160 in.-lb and 200 psi, (2) 125,440 in.-lb and 400 psi, and (3) 62,720 in.-lb and 600 psi. Maximum loads for the out-of-plane moment with internal pressure test cases as well as the maximum loads for the torsional moment with internal pressure combinations were also reduced.

Structural Analyses of Other Piping Components

Integral Attachments

The final report⁵ on an analytical study to develop stress indices for integral attachments to straight pipe (lugs) that are loaded with axial

Table 4.3. Machined-Elbow Test Models

Model Designation	Identification	Model Parameters
ME-1	Ideal torus	90° bend, long radius, 10.750-in. nominal OD, uniform wall 0.365 in. thick
ME-2 ^a	Ovaled torus	90° bend, long radius, 10.750-in. nominal OD, uniform wall 0.365 in. thick, flattened 5 to 8% of the diameter
ME-3	Thinned elbow	90° bend, long radius, 10.750-in. nominal OD, bore eccentric 0.093 in., minimum wall thickness 0.272 in., maximum wall thickness 0.458 in.
ME-4	Ovaled and thinned elbow	90° bend, long radius, 10.750-in. nominal OD, bore eccentric 0.093 in., minimum wall thickness 0.272 in., maximum wall thickness 0.458 in., flattened 5 to 8% of the diameter

^aIn addition to the elastic strain-gage tests, this model will be fatigue loaded to failure with a cyclic internal pressure loading.

Fig. 4.1. Machined-Elbow Model Mounted in the Loading Frame for an Out-of-Plane Bending Test.

thrust T , in-plane bending moment M_L , or out-of-plane bending moment M_C was completed this period and is currently being printed for general distribution. The study is based on the theoretical works of Bijlaard, which were programmed for the ORNL IBM 360/91 computer and used to make parameter studies. Simplified formulas to represent the maximum stress intensities were developed and used to define corresponding stress indices.

The draft of a second report⁶ on stress indices for lugs was also completed this period. This report considers other loading conditions which may also exist on a piping lug and contains specific recommendations for ASME code consideration.

References

1. S. E. Moore and W. L. Greenstreet, Program Plan for the ORNL Piping Program — Design Criteria for Piping, Pumps, and Valves, July 1, 1970 (unpublished).
 2. ASME Boiler and Pressure Vessel Code, Section III-1971, Nuclear Power Plant Components, American Society of Mechanical Engineers, New York, 1971.
 3. W. L. Greenstreet and S. E. Moore, Design Criteria for Piping, Pumps, and Valves, ORNL Nuclear Safety Research and Development Program Bimonthly Report for November-December 1972, Non-LMFBR Programs, USAEC Report ORNL-TM-4087, Oak Ridge National Laboratory.
 4. W. L. Greenstreet and S. E. Moore, Design Criteria for Piping, Pumps, and Valves, ORNL Nuclear Safety Research and Development Program Bimonthly Report for September-October 1972, Non-LMFBR Programs, USAEC Report ORNL-TM-4030, Oak Ridge National Laboratory, pp. 67-74.
 5. W. G. Dodge, Secondary Stress Indices for Integral Structural Attachments to Straight Pipe, USAEC Report ORNL-TM-3476, Oak Ridge National Laboratory (to be published).
 6. E. C. Rodabaugh, W. G. Dodge, and S. E. Moore, Stress Indices at Lug Supports on Piping Systems (to be published).
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4.2 EXPERIMENTAL AND ANALYTICAL INVESTIGATIONS OF NOZZLES

(AEC Activity 04 06 70 01 1)

B. L. Greenstreet J. P. Callahan

The distribution of stresses in the region of attachment of nozzles to pressure vessels is of particular interest to pressure vessel designers because of the complex geometries employed and the high stresses produced during normal vessel operation. The ORNL nozzles program is assigned the tasks of developing and verifying analytical methods for accurately determining these stresses. Included in the program are studies of single nozzles and multiple-nozzle clusters attached radially and nonradially to flat plates, cylindrical shells, and spherical shells as well as studies of perforated flat plates and spherical shells. Each study consists of an analysis development phase and an experimental phase to supply reliable data for verifying the corresponding analysis. In addition, such factors as superposition of nozzle loadings, mutual interactions of adjacent openings and nozzles, and interpretation of the resulting information in light of current design practice are under investigation. The information developed under this program and other technically related programs is evaluated and prepared in forms suitable for use in revising existing design rules or in developing new rules for both industrial and AEC-RDT codes and standards.

These studies are conducted in cooperation with the Pressure Vessel Research Committee (PVRC) of the Welding Research Council. Program personnel actively participate in the PVRC to provide communication between the ongoing work and the pressure vessel industry. The PVRC also serves as a means for transmitting information to appropriate code bodies for their consideration. The incorporation of this information into design practice provides added assurance that pressure vessels will perform reliably and safely.

Project Management

As a member of the Pressure Vessel Research Committee, ORNL is assigned the tasks of coordination, review, and evaluation of ongoing research programs for the PVRC Subcommittee on Reinforced Openings and External Loadings. To perform this management function ORNL (1) directs AEC-sponsored work, including work both at ORNL and under subcontract; (2) conducts parameter studies in support of code-oriented evaluation and correlation work in progress at Battelle/Columbus Laboratories; (3) prepares progress summaries and recommendations on PVRC-sponsored work; and (4) solicits comments and recommendations from the PVRC Subcommittee concerning AEC-sponsored studies. ORNL recently participated in the January 1973 meetings of the Subcommittee on Reinforced Openings and External Loadings and the PVRC Design Division, where the status of the various studies was discussed.

Single Nozzles in Spherical Shells

The University of Tennessee has been provided a 90-channel data acquisition system by ORNL for use in testing the two remaining nonradial nozzle models. Acceptance tests for the new system will be conducted by repeating selected loadings of the radial nozzle model 807-003 ($d_i/D_i = 0.249$, $D_i/T = 80.35$, and $d_i/t = 40$) which was recently tested using the older data acquisition equipment. Data from the acceptance tests will also provide a check of part of the data obtained previously. Once the acceptance tests have been completed, the 22 1/2° nonradial nozzle model 803-223 ($d_i/D_i = 0.083$, $D_i/T = 80.35$, and $d_i/t = 40$) will be mounted in the loading frame for testing. This model is the last of the three 22 1/2° nozzle configurations to be tested at the University of Tennessee.

Fabrication of the 7 7/8-in.-OD, 45° nonradial nozzle model 807-451 is also in progress at the University of Tennessee. Only one model configuration, consisting of a flush nozzle having a d_i/D_i of 0.236, a D_i/T of 80.35, and a d_i/t of 19.0, will be tested. This is the final model of the planned nonradial nozzle series.

The first volume of a three-volume report containing the experimental data for the second 2 5/8-in.-OD, 22 1/2° nonradial nozzle configuration, model 802-222, has been published.¹ Part A of the report contains the data in tabular form for an internal pressure loading, an axial loading of the nozzle, and a pure torque applied in the plane of the nozzle cross section. Part B will contain the data in tabular form for the pure bending moment loadings applied in selected longitudinal planes of the nozzle, and Part C will contain summary data plots for all loadings. The last two volumes of the report are currently being printed.

Single Nozzles in Cylindrical Shells

Model 4 ($d_0/D_0 = 0.129$, $D_0/T = 50$, and $d_0/t = 20.16$) of the ORNL series of thin-shell cylinder-to-cylinder models has been tested, and a final identification and resistance check of the 480 individual strain-gage elements attached to the model was completed. Preliminary results, together with analytical results obtained using an improved version of the closed-form solution for the governing differential equations, were reported previously.² This analysis was developed for internal pressure loadings by A. J. Edmondson, an ORNL consultant from the University of Tennessee. The basic geometry of the model and associated nomenclature used in the analysis are shown in Fig. 4.2. Figures 4.3 and 4.4 are revised plots of the experimental and analytical results. Figure 4.3 contains the results for the longitudinal plane of the model ($\phi = 0$ and 180°), and Fig. 4.4 contains the corresponding results for the transverse plane ($\phi = 90$ and 270°). Both longitudinal and transverse stresses are shown in each figure. In general, the experimentally determined stresses on the two sets of symmetrical planes, consisting of the 90 and 270° planes and the 0 and 180° planes, respectively, are in good agreement. In several cases the agreement between experimental and theoretical results has been improved with respect to the earlier comparisons. These comparisons will be included in a TM report being prepared on the improved closed-form analysis. The first draft of this report has been reviewed, and the report is being revised.

Fig. 4.2. Nomenclature for the Cylinder-to-Cylinder Model Configuration.

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Fig. 4.3. Comparisons of Analytical and Experimental Results for the 0 and 180° Planes of ORNL Cylinder-to-Cylinder Model 4 ($D_0/T = 50$, $d_0/t = 20.6$, and $d_0/D_0 = 0.129$) under 300 psi Internal Pressure Loading.

Fig. 4.4. Comparisons of Analytical and Experimental Results for the 90 and 270° Planes of ORNL Cylinder-to-Cylinder Model 4 ($D_0/T = 50$, $d_0/t = 20.6$, and $d_0/D_0 = 0.129$) under 300 psi Internal Pressure Loading.

Cluster of Nozzles

All moment and axial thrust nozzle loadings of the flat-plate model having an attached five-nozzle cluster have been completed at Auburn University, and the model has been repositioned in the loading frame in preparation for applying both 1:1 and 2:1 biaxial loadings to the plate. These loadings will complete the planned tests of the five-nozzle flat-plate model. Once this test has been completed, the hemispherical shell model having an attached four-nozzle cluster will be ready for testing. The instrumentation wiring for this model is being subjected to a final check-out.

Reinforcement Study

A series of parameter studies of inside vs outside reinforcement has been conducted at ORNL in conjunction with code-rule development work being done under subcontract at Battelle/Columbus Laboratories. As part of this work, a series of 22 models was analyzed using a finite-element method.³ These models represented various combinations of inside and outside reinforcement.

As a further extension of this work, the total calculated stresses for the six models listed in Table 4.4 were reduced to the three basic components defined according to the ASME Boiler and Pressure Vessel Code as: (1) primary stress — statically determinate stresses in a structure such as the membrane stress in a shell structure; (2) secondary stress — statically indeterminate stresses which are developed by the self constraint of a structure, and (3) peak stress — stresses at notches or other local discontinuities which do not cause noticeable distortion and are additive to the primary and secondary stresses in a structure.

These three basic stress components can be identified from the calculated total stresses using the procedures and formulas in paragraph NB-3653.2, Section III of the ASME Boiler and Pressure Vessel Code. The primary stress σ_p is calculated by

$$\sigma_p = \frac{1}{t} \int_{-t/2}^{t/2} \sigma(y) dy ,$$

Table 4.4. Nozzle-to-Spherical Shell Models Used in the Study of Primary, Secondary, and Peak Stresses

Model No.	D_i/T	d_i/D_i	d_i/t	s/S	Reinforcement ^a
S-24D	24	0.2	11.43	1.0	44% outside fillet
S-24H	24	0.2	11.43	1.0	112% inside
S-24M	24	0.2	11.43	1.0	65% outside fillet and 35% inside
S-72D	72	0.2	35.7	1.0	100% outside fillet
S-72F	72	0.2	35.7	1.0	100% inside
S-72H	72	0.2	35.7	1.0	65% outside fillet and 35% inside

^aReinforcement percentages are based on ASME Boiler and Pressure Vessel Code Replacement Rules (Winter 1971 Addenda to both Sections III and VIII).

where t is thickness of the cross section, $\sigma(y)$ is stress as a function of thickness, and y is thickness coordinate. The secondary stress σ_s was calculated by

$$\sigma_s = \frac{6}{t^2} \int_{-t/2}^{t/2} y\sigma(y) dy$$

This study was undertaken to determine whether the models that had the higher maximum total stress intensities will also have the maximum primary plus secondary stress intensities. The results of this investigation will be included in the ORNL report⁴ of the study of inside versus outside reinforcement.

References

1. R. L. Maxwell and R. W. Holland, Experimental Stress Analysis of the Attachment Region of Hemispherical Shells with Attached Nozzles, Part 4a, Nonradial Nozzle at 22 1/2 Degrees, 2.625 in. OD-2.500 in. ID, 4.00 in. Penetration, Report ME-7-72-1, The University of Tennessee Department of Mechanical and Aerospace Engineering, September 1972.
2. W. L. Greenstreet and J. P. Callahan, Experimental and Analytical Investigations of Nozzles, ORNL Nuclear Safety Research and Development Program Bimonthly Report for September-October 1972, Non-LMFBR Programs, USAEC Report ORNL-TM-4030, Oak Ridge National Laboratory, pp. 74-81.
3. D. C. Cornell, SAFE-2D, A Computer Program for the Stress Analysis of Plane and Axisymmetric Composite Structures, A User's Manual, USAEC Report GA-9076, Gulf General Atomic, February 1969.
4. R. C. Gwaltney and J. M. Corum, Analytical Study of Inside and Outside Compact Reinforcement for Radial Nozzles in Spherical Shells, USAEC Report ORNL-4732, Oak Ridge National Laboratory (to be published).

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5. RDT STANDARDS PROGRAM

5.1. PROGRAM OBJECTIVES AND ACTIVITIES

(AEC Activity 04 60 80 01 1)

M. Bender	P. C. Keeble
W. F. Ferguson	R. A. Schmidt

The prime responsibility for managing the organization, preparation, and distribution of engineering standards was assigned to the Oak Ridge National Laboratory by the AEC Division of Reactor Development and Technology. ORNL is the responsible standards office for RDT standards applicable to general nuclear technology and specifically to all systems except liquid-metal-cooled systems. Those standards prepared exclusively for liquid-metal-cooled systems are under the purview of the Liquid-Metal Engineering Center (LMEC), which is the responsible standards office for them.

The Bimonthly Program Management Letter No. 6 was prepared and submitted to RDT, and the Bimonthly Status Report and Index of RDT Standards were prepared and issued. Following discontinuance of the ORNL Nuclear Safety Research and Development Program Bimonthly Report (see Preface), the Index of RDT Standards and Status Report of RDT Standards may be used to obtain information regarding the status of RDT documents being prepared. The Index includes those standards which have been approved for use and are available for general distribution. The Status Report covers standards

planned for the program as well as work in progress on amendments and revisions to the approved standards. The Index is available upon request to the ORNL Standards Office, P. O. Box X, Oak Ridge, Tennessee 37830. Distribution of the Status Report is limited to participants in the RDT Standards Program.

Support to American National Standards Institute (ANSI)

The ORNL Standards Group continued to provide support to the ANSI Technical Advisory Board (NTAB) through preparation of a draft of the ANSI-NTAB Status Report, which included priority projects for power plant and other nuclear standards. The draft report was presented for review at the NTAB Priorities Committee and Executive Committee meetings held in January and February and will be updated to incorporate review comments and nonpriority standards prior to issue and publication.

A list of the national standards that are supplemented by or referenced in RDT standards was also prepared for ANSI-NTAB for presentation at the February meeting of the Executive Committee. This list revealed that 421 national standards are currently invoked by the RDT standards.

Additional support is being provided through comprehensive compilations of a list of nationally recognized standards for nuclear application for which there is an established need. This list, when completed, will be published by ANSI and will include source and cost information for the individual standards.

Status of Standards Being Processed

During this period, 15 approved standards (6 of which supersede previous issues) and 15 amendments to approved standards were issued; 8 draft standards (3 of which will supersede a previous issue), 6 proposed amendments to approved standards, and 1 proposed amendment to a superseded standard were prepared by ORNL and submitted to RDT for approval. RDT numbers and titles were assigned for 4 new standards, the titles for 4 standards were revised, and 5 proposed standards were withdrawn from the program.

A total of 267 tentative standards are currently available for use, and 22 draft standards, revisions, and amendments prepared by ORNL are awaiting approval by RDT. The mailing lists, which include AEC installations and contractors, codes and standards organizations, architect-engineers, consultants, nuclear components manufacturers, and electric utilities, now contain 416 addressees who receive 797 copies of the standards and 612 addressees who receive 1217 copies of the Index of RDT Standards.

In response to telephone and memorandum requests, 2319 copies of RDT publications were mailed to 153 organizations, 25 of which were added to the distribution list to receive future issues of the standards or the bimonthly index.

The following approved RDT standards were issued.

<u>Number</u>	<u>Title</u>	<u>Standards Writer</u>
C 7-4T	Thermocouple Material, Copper-Constantan Mineral-Oxide Insulated, Sheathed, Amendment 1, January 1973	ORNL
C 7-5T	Thermocouple Material, Chromel-P and Alumel, Solid Conductor (Bare, Fiberglass Insulated, and Sheathed over Fiberglass Insulation), January 1973	ORNL
C 7-15T	Thermocouple Connectors and Thermocouple Connector Panels, Amendment 1, January 1973	ORNL
C 17-5T	Metal-Sheathed, Mineral-Insulated Cable Bulk Material, February 1973	LMEC
E 1-6T	Automatic Spring-Loaded Safety Valves, Amendment 1, January 1973	ANC
E 1-37T	Balanced Safety Relief Valves for Liquid Metal Facility Gas Systems, February 1973	HEDL
E 3-2T	Centrifugal, Free-Surface, Sodium Pump with Electrical Drive, Amendment 1, February 1972	ARD
E 4-7T	Sodium to Air Heat Exchanger, Amendment 2, December 1972	HEDL

<u>Number</u>	<u>Title</u>	<u>Standards Writer</u>
E 6-13T	Core Support Structure for Sodium Cooled Reactors (Fabrication Only), Amendment 1, December 1972	ARD
E 6-17T	Core Restraint Mechanism for Sodium Cooled Reactors (Fabrication Only), October 1972	ARD
E 10-6T	Tank for Gas Service, November 1972	HEDL
E 11-2T	Filter for Sodium Service, January 1973	HEDL
F 11-4T	Determination of a Figure of Merit for $\text{PuO}_2\text{-UO}_2$ Fuel Pellet Homogeneity by Use of an Electron Microprobe, September 1972	HEDL
F 11-6T	Ceramographic Preparation of Mixed Oxide Fuel Pellets, January 1973	HEDL
M 2-2T	Stainless and Low Alloy Steel Forgings (ASME SA-182 with Additional Requirements), Amendment 1, January 1973	ORNL
M 2-3T	Carbon and Alloy Steel Welding Fittings (ASME SA-234 with Additional Requirements), January 1973 Revision	ORNL
M 2-15T	Nickel-Chromium Alloy Bars, Forgings, and Forging Stock (ASTM A637 with Additional Requirements), Amendment 3, January 1973	ORNL
M 3-2T	Stainless and Alloy Steel Seamless Tubes (ASME SA-213 with Additional Requirements), Amendment 3, February 1973	ORNL
M 3-4T	Nickel Alloy Seamless Tubes (ASME SB-163 with Additional Requirements), Amendment 1, January 1973	ORNL
M 3-5T	Austenitic Stainless Steel Welded Tubing (ASME SA-249 with Additional Requirements), Amendment 2, February 1973	ORNL
M 3-9T	Nickel-Iron-Chromium Alloy Seamless Pipe and Tubing (ASME SB-407 with Additional Requirements), January 1973 Revision	ORNL
M 3-10T	Nickel-Molybdenum-Chromium Alloy Seamless Pipe and Tubes (ASME SB-167 with Additional Requirements), January 1973 Revision	ORNL

<u>Number</u>	<u>Title</u>	<u>Standards Writer</u>
M 3-18T	Nickel-Molybdenum-Chromium Alloy Seamless Tubes (ASME SB-163 with Additional Requirements), Amendment 1, February 1973	ORNL
M 3-27T	Seamless and Welded Small Diameter Austenitic Stainless Steel Tubing (ASTM A632 with Additional Requirements), January 1973	ORNL
M 3-29T	Nickel-Chromium-Molybdenum-Columbium Alloy Seamless Tubes (AMS 5589 with Additional Requirements), Amendment 1, February 1973	ORNL
M 3-30T	Nickel-Chromium-Molybdenum-Columbium Alloy Seamless Tubes (AMS 5590 with Additional Requirements), Amendment 1, February 1973	ORNL
M 3-32T	Seamless Medium Carbon Steel Boiler and Superheater Tubes (ASME SA-210 with Additional Requirements), Amendment 1, February 1973	ORNL
M 5-4T	Nickel-Chromium-Iron Alloy Plate, Sheet, and Strip (ASME SB-168 with Additional Requirements), January 1973 Revision	ORNL
M 7-4T	Nickel-Chromium-Iron Alloy Rod and Bar (ASME SB-166 with Additional Requirements), February 1973 Revision	ORNL
M 8-1T	Helical Age-Hardenable Nickel-Chromium-Iron Alloy Springs, February 1973 Revision	ORNL

The following documents prepared by ORNL were submitted to RDT for approval.

<u>Number</u>	<u>Title</u>
C 7-17	Platinum Resistance Thermometer, November 1972 draft
E 15-2T	Requirements for Nuclear Components (Supplement to ASME Boiler and Pressure Vessel Code, Section III), Amendment, Feb. 14, 1973, draft
F 3-37T	Special Requirements for Metal Products, Amendment, Feb. 20, 1973, draft
F 5-1T	Cleaning and Cleanliness Requirements for Nuclear Components, Amendment, Jan. 19, 1973, draft

<u>Number</u>	<u>Title</u>
F 6-1T	Welding (Superseded Standard), Amendment, Jan. 22, 1973, draft
F 6-5T	Welding Qualifications (Supplement to ASME Boiler and Pressure Vessel Code Section IX), Amendment, Nov. 10, 1972, draft
F 6-5T	Welding Qualifications (Supplement to ASME Boiler and Pressure Vessel Code Section IX), Amendment, Jan. 17, 1973, draft
M 1-13	Aluminum and Aluminum Alloy Welding Rods and Bare Electrodes (ASME SFA-5.10 with Additional Requirements), February 1973, draft
M 2-10	Aluminum Alloy Welding Fittings (ASTM B361 with Additional Requirements), February 1973 draft
M 2-13	Aluminum Alloy Forgings (ASME SB-247 with Additional Requirements), February 1973 draft
M 2-15	Nickel-Chromium Alloy Bars, Forgings, and Forging Stock (ASTM A637 with Additional Requirements), Revision, February 1973 draft
M 3-8T	Zirconium and Zirconium Alloy Tubes (ASTM B353 with Additional Requirements), Amendment, Jan. 29, 1973, draft
M 3-14	Aluminum Alloy Seamless Tube (ASME SB-234 with Additional Requirements), February 1973 draft
M 6-3T	Alloy Steel Bolting Material for High Temperature Service (ASME SA-193 with Additional Requirements), February 1973 Revision, draft
M 6-6T	High-Strength, High-Temperature Bolting Materials (ASME SA-453 with Additional Requirements), February 1973 Revision, draft

Numbers and titles for the following new standards were assigned.

<u>Number</u>	<u>Title</u>	<u>Standards Writer</u>
C 17-9	Foil-Shielded Instrumentation Cable	ANL
E 7-10	Flow Restrictor for Liquid Sodium Lines	HEDL
F-3-14	Acceptance Testing of Liquid Metal Systems	LMEC
F 9-3	Energy Dependent Damage Functions	HEDL

The titles for the following standards were changed.

<u>Number</u>	<u>Title</u>	<u>Standards Writer</u>
E 1-11	Rupture Disk Devices for Liquid Metal Systems	HEDL
E 6-10	Temperature and Liquid Level Control Monitor, Port Plug (Fabrication Only)	HEDL
E 14-1	Static Seals	ORNL
P 4-2	High-Temperature Heaters, Molded Ceramic, Elec- trical Resistance Type	ORNL

The following proposed standards were withdrawn from the Standards Program.

<u>Number</u>	<u>Title</u>	<u>Standards Writer</u>
E 6-12	Simulated Test Position Assembly for Use in Nuclear Reactor Core	HEDL
E 6-37	Control Rod Driveline Disconnect Tool Assembly	HEDL
E 14-2	Solid Metal, Heavy Cross-Section Gaskets	ORNL
E 14-3	Hollow Metallic O-Rings	ORNL
E 14-4	Spiral-Wound, Asbestos-Filled, Metal Gaskets	ORNL

5.2. MECHANICAL EQUIPMENT AND SYSTEMS

(AEC Activity 04 60 80 03 1)

H. G. Arnold	J. H. Evans
C. A. Burchsted	W. R. Gall
W. A. Bush	R. W. Dehoney

Nuclear Code Supplements

A proposed amendment to E 15-2T, Requirements for Nuclear Components (Supplement to ASME Boiler and Pressure Vessel Code, Section III), was

prepared and submitted to RDT for approval. This amendment included changes required to update the standard to the Winter Addenda to the Code, several other minor changes that had been recommended, and a proposed solution to the weld shrinkage problem.

A proposal concerning possible areas in which more than one level of design may be appropriate for class 2 or 3 components was submitted to RDT. This proposal was prepared to assist RDT with the presentation of a proposal to ASME relative to cases in which a more detailed design is required for class 2 and 3 components, but which do not warrant a complete class 1 treatment.

Air Handling

An ORNL proposal to revise the following previously submitted draft standards to accommodate the developments reported at the Government-Industry Conference on Filters and Filter Media in August 1972 and to strengthen the specification on filter-pack tightness was submitted for RDT concurrence.

- E 9-1 Air Filter, HEPA, for High Humidity, High Temperature Application (January 1972 draft and August 1972 proposed changes)
- E 9-2 Air Filter, HEPA, General Purpose (October 1970 draft and August 1972 proposed changes)
- M 16-3 HEPA Filter Media, Glass Fiber (January 1972 draft and August 1972 proposed changes)

Consistent with the stated RDT policy to use adequate nationally recognized standards where they exist, a recommendation was made by ORNL to supersede RDT E 9-3T, High-Efficiency Pleated-Bed Gas-Phase Adsorber Cells, by the recently published AACC CS-8, High-Efficiency Gas-Phase Adsorber Cells.

A proposed amendment to M 16-1T, Gas-Phase Adsorbents for Trapping Radioactive Iodine and Iodine Compounds, is being prepared. Updating of this standard is required to reflect state-of-the-art changes, particularly

with respect to testing and test procedures, as well as other changes that were recommended by industry sources.

5.3. INSTRUMENTATION, CONTROLS, AND ELECTRICAL

(AEC Activity 04 60 80 03 1)

K. R. Carr T. L. Hudson
S. J. Ditto C. S. Lisser
J. A. Russell

The advanced draft of RDT P 4-4, Electric Heater — Simulated Pressurized Water Reactor Fuel Pins, incorporating comments from ORNL, RDT, and Idaho Operations Office (ID), was reviewed, and ORNL comments were prepared and submitted to Aerojet Nuclear Company (ANC).

Work continued on preparation of the draft of C 4-3, Orifice Flange Union, Weld Neck, Stainless Steel for 300-Pound Service, for submittal to RDT for approval.

Assistance was provided with the evaluation of RDT P 4-1T, Electric Heaters — Simulated LMFBR Fuel Pins, and recommendations were made to:

1. institute closer control in processing, handling, and storing boron nitride insulation;
2. increase in-process inspection of each component and strengthen documentation of test and inspection results;
3. improve infrared scanning and neutron radiography examination;
4. incorporate detailed quality verification documentation requirements which are presently included in ORNL Reactor Division Job Specification JS-174-246 and being used for procurement of the heaters.

As listed in Sect. 5.1, three thermocouple standards were published during the report period following approval by RDT.

RDT C 17-4T, General Instrumentation, and draft C 4-8, Orifice Assemblies for Nuclear Systems, were reviewed.

5.4. QUALITY ASSURANCE AND PROCEDURES

(AEC Activity 04 60 80 03 1)

J. W. Anderson	F. L. Hannon
H. G. Arnold	R. M. Fuller
C. A. Burchsted	R. M. Holmes
G. L. Copeland	T. W. Pickel
W. C. T. Stoddart	

Quality Assurance

The following activities were performed in support of the ANSI N45 subcommittee and work groups.

1. The January meeting of the N45-2 Subcommittee was attended, and minutes of the meeting were distributed to the subcommittee members. In addition, a list of the action items resulting from the meeting was prepared and distributed; and a list of the numbers, titles, and responsibility scopes for the subcommittee and work groups was corrected and distributed.

2. The proposed charter for the Area Management Advisory Group (AMAG) was reviewed and returned to the chairman of the N45-2 Subcommittee.

3. The Manual on Operating Practices and Procedures for the subcommittee was revised in accordance with comments received at the meeting in January, and the completed manual was distributed to the subcommittee members.

4. Draft 5 of ANSI N45-2.8 (Mechanical Equipment) was marked up to reflect resolution of comments and distributed to the subcommittee. The comments received from the subcommittee were distributed to the work group.

5. The proposed draft of an AEC Regulatory Guide covering preoperational and startup testing was also submitted to the N45-2.8 work group for review.

6. Draft standards N45.2.9 (Records) and N45-2.12 (Audits) were reviewed; and comments were returned to the work group.

7. The draft outline for proposed standard Quality Assurance Program Requirements and Their Implementation for Design, Construction (and Operation) of Nuclear Fuel Reprocessing Facilities was reviewed, and comments were returned to the N46-2 Subcommittee.

A meeting of the ASQC Interface Subcommittee was attended in January to resolve comments from the Nuclear Power Technical Committee received during balloting of the Matrix of Nuclear Quality Assurance Program Requirements. Typing and reproduction of the matrix were coordinated, and tentative arrangements have been made to have the matrix published and offered for public sale by the ASQC.

Construction, Installation, and Transportation

An informal copy of a revised draft of F 8-7, Transportation of Critical Components and Equipment, was sent to ANC and ID after consideration of their earlier comments. Following reconciliation of these comments, the draft will be submitted to RDT for approval.

A meeting was held at ORNL to review user problems with RDT F 8-6T, Hoisting and Rigging of Critical Components and Related Equipment. Problems arise mostly among routine operations. Other users have pointed out that test frequency should probably be reduced for routine operations and that load testing should probably be at 110% of rated capacity, rather than 125%, following the initial acceptance test.

Testing

A proposed amendment to RDT F 3-37T, Special Requirements for Metal Products, was prepared and submitted to RDT for approval. This amendment was prepared to correct certain differences in a previous amendment and to include provisions for angle beam scanning of bolts and studs.

Air Cleaning

A proposed amendment to RDT F 5-1T, Cleaning and Cleanliness Requirements for Nuclear Components, was prepared and submitted to RDT for approval. This amendment covers inspection and recleaning in the event the packaging of a component is found to be violated on delivery.

The latest draft of ANSI N45-8.2, Testing of Nuclear Air Cleaning Systems, was reviewed, and comments were returned to the subcommittee.

A draft of N45-8.1, Requirements for Nuclear Power Plant Air Systems, is still in preparation.

5.5. MATERIALS AND FABRICATION

(AEC Activity 04 60 80 03 1)

R. M. Fuller F. L. Hannon
O. A. Kelly

The previously submitted revision of RDT F 3-6T, Nondestructive Examination, was approved by RDT; however, incorporation of additional changes by users of the standard will require resubmittal to RDT prior to publication of the revision.

A response to the LMEC request to initiate an amendment to the appropriate RDT standard to cover diametral weld shrinkage was prepared which outlined a program for reviewing the problem and developing the required amendments. After studying the problem ORNL proposed an amendment and submitted it to RDT. A meeting was also attended at HEDL in which discussions were focused on control or analysis of diametral weld shrinkage and qualification of welding procedures using an automatic welding machine associated with fabrication of FFTF components.

Sixteen materials standards were approved and published during this period, and another 11 were prepared and submitted to RDT for approval. All of these are included in the summary list in Sect. 5.1.

6. GENERAL NUCLEAR SAFETY STUDIES

6.1 SEISMICITY AND SEISMIC RESISTANCE OF NUCLEAR POWER PLANTS (AEC Activity 04 60 40 01 1)

J. Foster G. C. Robinson
 G. D. Whitman

The seismic research program is concerned primarily with studies in soils engineering with emphasis on the effects soils may have on the characteristics of earthquake motions and the potential for soils failure from earthquakes. This work is performed under subcontract with Shannon & Wilson-Agbabian Jacobsen Associates (SW-AJA) and the University of California, Berkeley. Three principal tasks are involved: (1) in-situ determinations of dynamic soil properties, (2) a study on the liquefaction potential of cohesionless soils, and (3) the development of recommended procedures for predicting the dynamic response of soil deposits. No additional work is being planned on the development of requests for proposals on the intercomparison study of soil testing methods and the study on evaluation and validation of analytical methods used in design.

ORNL management of the seismic research program and the subcontract activities is to be terminated this fiscal year. AEC-RDT will assume responsibility for the further development and administration of the program.

In-Situ Determination of Dynamic Soil Properties

The development of methods for determining soil properties in situ involves techniques for obtaining dynamic shear moduli. The approach has been to drive a down hole probe anchored in a bore hole so that time of flight determination of shear waves may be made in adjacent bore holes at varying depths. Shear wave velocities and shear moduli can then be calculated as a function of strain level for use in a dynamic analysis of the soil column. The feasibility of the concept has been demonstrated, with good results having been obtained up to strain levels of $5.1 \times 10^{-4}\%$. Unfortunately, inordinately high forces would be required to achieve the upper range of strain levels, 0.1%, required, so the practicability of a continuously excited device cannot be demonstrated. In the course of checking the anchor system in the bore hole and calibrating the instrumentation in adjacent holes, it was determined that impact loadings on the drill rod produced high strain levels. Strains in the range of 10^{-3} to $10^{-2}\%$ were achieved with relatively light blows ranging from a hammer to a drop weight producing about 600 ft-lb of energy. This increase in strain level is to some degree due to a higher frequency of excitation. Preliminary analysis of the data and theoretical treatment of this approach indicate no frequency dependence of the shear modulus. The approach involves highly simplified techniques and offers greater flexibility to work at considerable depth. An analysis has shown that an impact system more nearly matched to a soil stiffness can produce greatly improved results. The concept is being evaluated in greater detail.

Liquefaction Potential of Cohesionless Soils

The liquefaction potential of cohesionless soils is being studied with large shake-table specimens at the University of California, Berkeley, Richmond Field Station. A dual approach is being pursued in which dry sand specimens are being used to study the effects of multidirectional shaking and saturated sand specimens are being used to examine the effects of relative density and surcharging on liquefaction potential.

A dry sand sample 36 in. in diameter and 3 in. thick has been successfully produced, and one-directional horizontal shaking tests have been performed. This is being done prior to installing the specimens and excitation system on the large 20- by 20-ft shake table to achieve another horizontal and a vertical motion, thereby producing a three-dimensional system of loading. These tests will provide data on the comparative behavior of sands subjected to one- and multidimensional shaking.

The saturated sand specimen designed to study the effects of relative density and surcharging is 90 in. long, 42 in. wide, and 2 in. thick. High shear strains will be achieved in the sample by placing a flexible mass over the top of the specimen in the form of small lead shot contained in a grating. The instrumentation for measuring pore pressure, strain, deformation, and acceleration has been acquired, and no further difficulty has been incurred in depositing a uniform sample with a modified soil spreader.

This series of tests on large samples will minimize the boundary effects that exist in small specimens and more nearly simulate the state of stress that exists under field conditions. Through this work a more quantitative understanding of the parametric relationships which govern liquefaction of sands under earthquake shaking will be established.

Dynamic Response of Soil Deposits

The outline for the report on recommended procedures for evaluating the seismic response of soil deposits has been completed, and two of the five sections have been written. A first draft of the complete report is scheduled to be issued for review on April 1, 1973.

6.2 IGNITION OF CHARCOAL ADSORBERS BY FISSION PRODUCT DECAY HEAT

(AEC Activity 01 03 00 00)

R. A. Lorenz W. J. Martin
H. Nagao*

An experiment to demonstrate the consequences of iodine decay heating of a charcoal bed is being assembled and tested in a hot cell for safe operation. The objective of this work is to provide information to enable confident prediction of the desorption of iodine, and possible ignition of the charcoal, at abnormally low flow rates. The experiments will ultimately use several thousand curies of ^{130}I to provide radioactive decay energy in a charcoal bed of approximately 28 cm^3 volume.

Three dummy irradiation capsules were welded and inspected. All passed the helium leak test and hydraulic pressure-cycle leak test, but the weld penetration was less than desired.

In order to test the hot-cell charcoal filters for I_2 adsorption efficiency, we decomposed PdI_2 in the cell atmosphere to obtain a radio-traced source of I_2 . We found that approximately 2% of the iodine was particulate. A special test for the decomposition products was then performed with the recirculating loop components using a series of filter papers, 1-cm-long unimpregnated charcoal cartridges, and 1-cm-long impregnated charcoal cartridges in an unheated assembly. The dry, filtered air flow was 4.92 liters/min (25°C), and the decomposition of 13.8 mg PdI_2 was completed in 30 min. Air flow continued for 2 hr as the decomposition furnace cooled. We found that 5% of the iodine was collected with particles on the first AP-20 glass-fiber filter, and an additional 3 or 4% (probably adsorbed as I_2) on each of the two Millipore AP-20 and two Cambridge IG high-efficiency filter papers. Only $2 \times 10^{-4}\%$ of the iodine reached the impregnated charcoal, indicating negligible methyl iodide. The particulate iodine was not associated with palladium.

*Visiting scientist from NAIG, Nuclear Research Laboratory, Kawasaki, Japan.

We have added a filter holder to the recirculating loop between the decomposition furnace and the test bed to hold several disks of HEPA filter paper. This will prevent particles from reaching the test bed, and, by monitoring with an ion chamber, we can observe the adsorption/desorption behavior of the highly radioactive iodine on the filter media.

The calculated rate of heat release from oxidation of charcoal by dry air is shown in Fig. 6.1 for several of our previously reported¹ laboratory test runs. The calculation is based on a heat balance for the charcoal bed. For these test runs (which used electric heat input in simulation of iodine decay heat), the computer program was adjusted to solve for oxidation heat as the only unknown. For the runs in the hot cell, which use decay heat from iodine as the heat source, the computer program will be changed to treat the oxidation heat rate as a known quantity and will solve for the decay heat.

For the few samples of charcoal tested we observed that the addition of iodine tends to decrease the chemical reactivity to air (runs 5 and 8 with BC-573) and that the reactivity is higher for charcoals with high potassium content. We have also observed that second and third heatings of a given bed show successively lower reactivity.

For experiments conducted in our apparatus at low air flow rates, we have defined "ignition temperature" as the temperature at which the rate of rise first equals $20^{\circ}\text{C}/\text{min}$. When these ignition temperatures are plotted on the extrapolated heating rate curves, the oxidation heating rate is close to $15 \text{ cal min}^{-1} \text{ g}^{-1}$ charcoal for all of the runs. In contrast, if the charcoal heat capacity were $0.3 \text{ cal g}^{-1} (^{\circ}\text{C})^{-1}$ and there were no heat transfer to the quartz container, the thermocouples, or the flowing air, only $6 \text{ cal min}^{-1} \text{ g}^{-1}$ charcoal would have been required to heat the charcoal at a rate of $20^{\circ}\text{C}/\text{min}$.

Reference

1. R. A. Lorenz and W. J. Martin, Ignition of Adsorbers by Fission-Product Decay Heat, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May-June 1972, USAEC Report ORNL-TM-3900, Oak Ridge National Laboratory, pp. 122-125.

Fig. 6.1. Heat Release Rate from Oxidation of Charcoal in Dry Air.

7. NUCLEAR SAFETY INFORMATION

7.1 NUCLEAR SAFETY INFORMATION CENTER

(AEC Activity 04 60 70 01 1)

J. R. Buchanan Wm. B. Cottrell

The Nuclear Safety Information Center (NSIC) was established in 1963 by the USAEC Division of Reactor Development and Technology (RDT) to collect, evaluate, and disseminate nuclear safety information to governmental agencies, research and educational institutions, and the nuclear industry. The Center's basic computer file now contains references on over 72,100 documents.

Computer Programming and Processing

Major development on the NSIC system was completed before 1970; however, a performance improvement effort has continued. This comparatively low level of effort in conjunction with computer hardware and software enhancements have significantly improved performance and reliability.

As faster and more reliable input/output devices were made available and as new or improved file management techniques were applied, the computer costs associated with the entry of a document description were reduced by approximately 50%.

In addition, the following functional improvements have been made within the past year:

1. NSICIRK, the interactive query program designed for typewriter type terminals was modified to provide paging when used on CRT terminals.
2. The AUTHFIND search program was modified to optionally permit the user to search on a portion of an author's name. The name used as a search key may be truncated at any point to the right of the first character.
3. The KWIC list of keywords was modified to provide greater readability.
4. The initial programs were implemented using the IBM COBOL (F level). IBM now supports the American National Standard (ANS) COBOL. A recent conversion to this language gained some improved functions and greater error checking capability.

SDI Program

During this period, 37 new participants were added to the SDI program and 16 were removed, for a total of 2135 participants. Thirty profiles were revised during the period.

PPIF Program

Four new participants were added to the Program and Project Information File during this period, for a total of 284 PPIF users.

Reports Status

The following report has been published and is available for purchase from the National Technical Information Service, U.S. Department of Commerce, Springfield, Virginia 22151:

<u>Report No.</u>	<u>Title and Author</u>	<u>Price</u>
ORNL-NSIC-53	Radiography Incident and Overexposures, R. L. Scott and R. B. Gallaher	\$10.00

Nuclear Standards Activities

A preliminary machine-generated printout of data from the 1972 revision of the Directory of Personnel Involved in the Development of Nuclear Standards in the U.S. was submitted to AEC-RDT on February 14. The printout listed individuals and their standards effort as sorted by affiliated company, organization type, and address. Final data entry and debugging are in process, and the revised directory is scheduled to go to reproduction by mid-March.

First typing of updated information for the 1972 revision of the *Compilation of Nuclear Standards, Part I: U.S. Activities* has been completed. Following proofing and correction, it will be sent to reproduction for distribution in early April.

Some data for the 1972 Foreign Standards Compilation have not yet been received. All data that are on hand have been transferred to master sheets for subsequent typing. Missing data have been urgently requested and are expected during March.

Summary of Information Activities

During the report period, the following requests were received and answered by NSIC personnel:

Accessions added to the computer file	1122
Technical inquiries (letter or telephone)	151
Requests for NSIC reports and other routine inquiries	110
Additions to NSIC reports announcement list	5
Visits for consultation or use of reference material	15
Additions to SDI program	37
Additions to PPIF program	4

7.2 TECHNICAL PROGRESS REVIEW *NUCLEAR SAFETY*

(AEC Activity 04 60 70 01 1)

J. P. Blakely Wm. B. Cottrell

Nuclear Safety, a Technical Progress Review, is prepared and edited at ORNL on a bimonthly schedule under contract with the AEC Office of Information Services. The January-February 1973 issue (Vol. 14, No. 1, 80 pages) contained articles by authors from the University of North Carolina, Consolidated Edison of New York, the TÜV (Technischer Überwachungs-Verein Rheinland e.V. Geschäftsleitung), Gulf General Atomic, the CEA (Commissariat à l'Energie Atomique, Cadarache), and the *Nuclear Safety* staff.

The lead article (by C. B. Bell, University of North Carolina) summarized a two-week meeting on Earthquake-Resistant Design of Engineering Structures, held at the Berkeley Campus of the University of California from June 19-30, 1972. The principal topics covered included seismology and strong-motion records, response of soils and dams, finite-element applications in structures and foundations, a new large shaker, and probabilistic design.

Joyce P. Davis, an attorney with Consolidated Edison, discussed the regulation of the environmental effects of nuclear power plants. This was the first of a series of two articles, and it presented a survey of the regulation of environmental features of nuclear power plants. Receiving particular attention was the jurisdiction of the Atomic Energy Commission under the Atomic Energy Act of 1954 and its expansion under the National Environmental Policy Act of 1969.

K-H. Lindackers and W. Stoebel, of the TÜV, presented a probability analysis applied to light-water reactor loss-of-coolant accidents. In the past, it has been the practice in the Federal Republic of Germany to base site selection and engineering safeguards for nuclear power plants on the so-called maximum-credible-accident (MCA) concept. In the opinion of the authors, it will now be necessary to use a more quantitative approach, as developed by the United Kingdom Atomic Energy Authority, because nuclear power plants must be constructed in more densely populated areas.

GGA's M. H. Raudenbush, who is assigned to the Fort St. Vrain HTGR site, evaluated control-board layout design trends in the nuclear power industry from a human engineering standpoint. The article includes both an extensive review of existing literature and a subjective discussion of operator-automation interfaces which suggests the direction future control-board design should take to assure optimum operator control and safety.

Safety experience gained from operation of France's experimental fast reactor Rapsodie was discussed by J. P. Argous, H. Chantot, and J. Petit (all of CEA, Cadarache) in an article that was adapted by S. J. Stachura of Atomics International's Fast Breeder Reactor Engineering Group. This article briefly describes and analyzes three operational incidents at the Rapsodie.

The issue concluded with regular review articles by *Nuclear Safety* staff members: Safety-Related Occurrences Reported in August and September 1972 (W. R. Casto); Progress Summary of Nuclear Safety R&D Projects (Myrtle Sheldon); and General Administrative Activities, Action on Research and Other Projects Undergoing Regulatory Review or Consideration, and Proposed AEC Rule Changes as of October 15, 1972 (J. P. Blakely). Also included were two book reviews (by D. N. Hess and R. D. Birkhoff), announcements of five meetings or short courses, a list of available NSIC documents, and an erratum. The "editorial" space contained a discussion of NSIC's technical inquiry service, and a photograph and abstract of ORNL-NSIC-74, the plutonium dose report, were included on the inside front cover.

Photoready copy for *Nuclear Safety*, 14(2), March-April 1973, was scheduled to be sent to the printer in mid-March for distribution in mid-April. Draft material for the May-June issue of *Nuclear Safety*, 14(3), was distributed for external review in early March. Subsequent issues of *Nuclear Safety* through January-February 1974 are in various stages of development.

Frank N. Browder, Chemical Technology Division, replaced Boyd Weaver as editor for Plant Safety Features in February.

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