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ORNL NUCLEAR SAFETY RESEARCH AND DEVELOPMENT PROGRAM

BIMONTHLY REPORT FOR JULY-AUGUST 1971

Wm. B. Cottrell

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ORNL NUCLEAR SAFETY RESEARCH AND DEVELOPMENT PROGRAM BIMONTHLY REPORT FOR JULY—AUGUST 1971

Wm. B. Cottrell

OCTOBER 1971

OAK RIDGE NATIONAL LABORATORY
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ABSTRACT

The accomplishments during the months of July and August in the research and development program under way at ORNL as part of the U.S. Atomic Energy Commission's Nuclear Safety Program are summarized. 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. Several projects are also conducted in support of the high-temperature gas-cooled reactor (HTGR) and the liquid-metal fast breeder reactor (LMFBR) programs. Other major projects are those on primary piping and steel pressure-vessel technology. They also 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. 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 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 recent emphasis on environmental effects.

Previous Reports in This Series

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September–October 1965	ORNL-CF-65-11-2
November–December 1965	ORNL-CF-66-1-41
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–February 1967	ORNL-TM-1792
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January–February 1968	ORNL-TM-2164
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September–October 1970	ORNL-TM-3212
November–December 1970	ORNL-TM-3263
January–February 1971	ORNL-TM-3342
March–April 1971	ORNL-TM-3411
May–June 1971	ORNL-TM-3483

SUMMARY

1. LMFBR SAFETY STUDIES

1.1 FFM Program Coordination

Weekly telephone contacts have been maintained with the FFTF project. Monthly management activity reports have been sent to HEDL, with copies to RDT, WARD, and ANL. The instrumentation layout and test program for bundle 2A (to determine base-line thermal-hydraulic behavior of the FFTF rod bundles) has been reviewed by FFTF, WARD, and RDT. Prepublication preliminary results from single-rod tests in bundle 1A have been transmitted to interested parties. A heater development program has been formally initiated, and a program plan is being formed. A program has been initiated to determine the physical response of LMFBR stainless steel cladding under the temperature transients in LMFBR perturbations such as partial blockages and partial flow coastdown.

1.2 Analytical Studies

Evaluation of data from bundle 1A (unblocked base-line tests) has started. The internal flow mixing as determined by single-rod-heated tests can be predicted reasonably well by the corrected ØRRIBLE code using the coefficients for turbulent cross flow, diversion cross flow, and flow sweeping of C_t = 0.015, C_d = 0.6, and C_s = 0.5 respectively. (See the text for definitions of these factors.) Evaluation of multirod experiments was started (18 rods were heated because rod 10 failed fairly early in the test program). The temperatures measured by the thermocouples within the wire wrap were consistently higher (up to 70°F) than expected sodium temperature. The temperature differences do not follow simple correlation with heat flux and show an effect of temperature that becomes larger at higher temperature. The temperature differences are consistent and reproducible for each point but are not uniform from point to point. Postulated reasons for this might be reductions in contact resistance between the clad and the BN directly under the wire because of the pressing effect by the wire due to differential thermal expansion of the clad and wire and of the BN and the cladding. Until disproved by further tests,

it would be prudent to assume that these hot spots would also occur in an LMFBR core.

1.3 Water System Mockup of the FFM Facility

Results from tests with ten-channel and six-channel inlet blockages show no deleterious effects within the heated zone. Experiments are beginning with partial six-channel symmetrical blockage in the heated zone. These results will anticipate FFM results with bundle 3A.

1.4 FFM Noise Analysis

Cross correlation of thermocouple response to residual power fluctuations shows that determination of internal flow profiles may be possible by these techniques.

1.5 FFM Facility Activities

Tests scheduled for bundle 1A (previously bundle 1) were completed. In addition, it was shown that the system operated stably at very low flow and power (1 gpm and 6 kW), so that determination of temperature profiles under conditions simulating degraded emergency cooling in the FFTF could be determined. Post-test examinations showed that a slow failure propagation initiated by failure of rod 10 was probably responsible for two other rod failures. In addition, two heating elements failed internally at the closed end. The design for subsequent heaters is different at this point. Some discoloration of the stainless steel cladding was discovered downstream from the heated section. Chemical analyses are being performed to determine its nature. No failures or other difficulties were experienced with the test loop itself during testing with bundle 1A. It was demonstrated that defective heaters could be removed from the bundle by grinding the heater sheath-to-header weld.

Assembly of bundle 2A is nearly complete, and procurement and manufacturing for bundle 3A have been initiated.

1.6 Heater Development

A separate heater development program was started to assure a viable supply of electrical fuel-rod simulators for HEDL, ANL, and ORNL. The

intent is to develop a phase II heater which can withstand transients to the point of local boiling at a heat flux of 1,000,000 Btu hr^{-1} ft^{-2} and a phase III heater which can withstand sodium expulsion and reentry (for 1-sec durations) at 1,000,000 Btu. The prototype heater for bundle 2A finally failed after 5.2 hr at 14.5 kW/ft and 1200°F sodium temperature. Cumulative operating time on this heater exceeded 500 hr at power levels in excess of 9.0 kW/ft and temperatures in excess of 900°F. Development of sheathed intrinsic thermal elements within the heater clad is progressing. Evaluation of the problem of maintaining high-purity boron nitride insulation both as powder and as crushable free-standing forms has been started.

1.7 LMFBR Cladding Response under Transient Heating Conditions

We have initiated a study to quantitatively define the behavior of irradiated stainless steel cladding under transient temperature conditions. A three-part program is being developed consisting of: (1) rapid transient annealing of irradiated materials for subsequent electron microscopic examination and evaluation of damage recovery, (2) tests of tensile or bend specimens under rapid heating conditions, and (3) burst testing of irradiated cladding tubes to define their irradiation temperature deformation and burst characteristics as a function of fluence, irradiation temperature, heating rate, and internal pressure.

2. LWR SAFETY STUDIES

2.1 Failure Modes of Zircaloy-Clad Fuel Rods

All experimental work on fuel rod failure and its effects on emergency cooling was terminated at ORNL on June 30, 1971. The experimental facilities developed by the program are being held for possible future use. Analysis of data obtained near the end of the fiscal year has continued, and topical reports on several studies are being prepared.

2.2 High-Temperature Properties and Metallurgical Evaluation of Zircaloy Cladding

All experimental work at ORNL on fuel rod failure and its effects on emergency cooling was terminated on June 30, 1971, and analysis of data

obtained near the end of the fiscal year has continued. In particular, examination of the final 32-rod transient burst test (1000 psig, 25°F/sec) has been completed. Although only half of the rods deformed to rupture, coolant channel blockage was almost 70%. A 16-rod test run for Combustion Engineering at 1200 psig and 25°F/sec exhibited at least this degree of blockage. Further analysis on the embrittlement of Zircaloy cladding has demonstrated that there is an acceleration of the rate of embrittlement related to deformation of the cladding and to diffusion criteria. Diffusion equations and assumptions used by some in the past have been in error. Topical reports are being prepared on all phases of the Failure Modes Program.

2.3 Heater Rod Performance

In view of the need to employ electrically heated rod bundles to simulate fuel bundles in accident transient studies, an experimental program which will include simulation of length, diameter, linear heat output, surface heat flux, surface heat transfer characteristics, internal gas pressure, stored energy, material properties, etc., has been established.

2.4 Loss-of-Coolant Heat Transfer from Rod Bundles

Because of the importance in the loss-of-coolant heat transfer from rod bundles in determining the emergency core cooling requirements, this program has been initiated to supplement the large-scale tests at NRTS. This program will use an existing facility slightly modified to accommodate a 5 × 5 full-length bundle operated up to 12 kW/ft. The associated program, now under review, would determine at least the fuel rod behavior during the loss-of-coolant accident, including the effect of emergency core cooling.

3. HTGR SAFETY STUDIES

3.1 Program Objectives and Coordination

The HTGR safety program studies reactions between steam and graphite-based fuel elements, coated particle integrity under abnormal conditions, and fission product behavior in the coolant circuit.

3.2 Reactions of Steam with Graphite

Reaction rate experiments with H-327 graphite have continued at 800 and 900°C with about 0.095 atm partial pressure steam and at 1000°C with 0.045 and 0.070 atm. Comparison of some of the experimental data previously obtained at 1000°C with results derived at Gulf General Atomic using the OXIDE calculational program suggests that the calculations are conservative by about a factor of 10.

The experimental part of the in-pile steam graphite program is complete. Experiment SG-4 was found to have sustained severe fuel damage and extensive removal of the matrix carbon after 17 hr exposure to steam. Surface temperatures ranged from 600 to 800°C and center-line temperatures from 800 to 1300°C.

3.3 Fuel Integrity and Fission Product Release

Comparison of cesium transmission through pyrocarbon deposited from propylene and methane continues. High-temperature isotropic coatings derived from methane are proving to be significantly better containers for cesium, as had been expected.

3.4 Fission Products in the Coolant Circuit

Iodine desorption experiments have continued. Desorption spectra have been obtained for iodine deposited on Fe_3O_4 loaded at a concentration of 0.29 monolayer. In addition, isothermal desorptions were performed to compare rates of desorption at 250 and 400°C.

Data obtained from surveillance experiments at Peach Bottom continue to be processed.

The ^{137}Cs and ^{131}I concentrations in the helium leaving the core during the first cycle of Core II operation were found to have fallen below the value determined at a comparable stage of Core I and considerably below the values existing at the end of Core I, when there were many broken elements. Cesium deposited during Core I was probably only reentrained to a small extent.

4. HEAVY-SECTION STEEL TECHNOLOGY PROGRAM

4.1 HSST Program Administration and Procurement (Task A)

The major items to be procured for the Heavy-Section Steel Technology (HSST) Program continue to be the 6-in.-thick pressure vessels. Flaw indications in the head-to-cylinder welds have delayed the delivery of the first vessel until mid-September. Investigations under five subcontracts continue.

4.2 Material Inspection and Control (Task B)

A summary report which identifies the steel set aside for surveillance monitoring is being prepared.

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

Final tests planned on HSST plates were completed and results are being collated. Investigation of effects on heat treatment parameters will commence in the near future.

4.4 Fracture Mechanics Investigations (Task E)

Analytical work and gross strain and crack arrest investigations are continuing. Testing of the large tensile specimens was completed, and upper and lower bounds on K_{Ic} up to 550°F have been deduced from the data.

4.5 Fatigue and Crack Propagation (Task G)

The final report on crack propagation in reactor environment is being prepared, and preparations are being made to study propagation rates of irradiated steel.

4.6 Irradiation Effects (Task H)

Gamma heat studies in the Engineering Test Reactor and the Babcock and Wilcox Test Reactor have been completed. The use of either reactor for irradiating 4-in.-thick compact tension specimens presents problems. Further evaluation is continuing.

Fifty Charpy impact and Charpy thickness compact tension specimens were successfully machined from 20 broken halves of irradiated 1-in.-thick compact tension specimens. Testing of these specimens is scheduled to begin in September.

4.7 Complex Stress State (Task I)

Twenty-two epoxy pressure vessels containing nozzles having corner flaws were tested, and the stress intensity factor as a function of flaw size was determined from the results. Six 1-in.-thick steel pressure vessels were tested.

4.8 Simulated Service Test (Task K)

Twenty-six 1-in.-thick flawed tensile specimens are being tested. The six tested thus far exhibit fracture behavior similar to 1-in.-thick compact tension specimens. Three 6-in.-thick flawed submerged-arc weld tensile specimens were tested at temperatures for frangible, transitional, and toughness behaviors. The initial tests of the study of specimen behavior in Charpy thickness (0.394 in.) have been completed.

4.9 Specific Safety Research (Task L)

Testing of the series of Charpy thickness specimens is in progress, and the regular Charpy and the slow-bend tests have been completed.

5. DESIGN AND ANALYSES OF NOZZLES, PIPES, PUMPS, AND VALVES

5.1 Experimental and Analytical Investigations of Nozzles

Experimental and analytical studies of single radial and nonradial nozzles and clusters of nozzles attached to flat plates, spherical shells, and cylindrical shells are continuing. Current experimental work includes application of the final series of loadings to the modified 22 1/2° nonradial nozzle at the University of Tennessee, and completion of retesting of the Auburn University flat-plate model having a single attached nozzle. Experimental models in various stages of preparation include the last of the planned series of single radial nozzles attached to a spherical shell at the University of Tennessee, the four-nozzle cluster attached to a

spherical shell, and the flat-plate models with two- and five-nozzle clusters at Auburn University. The computerized data reducing and plotting procedures developed at the University of Tennessee are now being used with Auburn University experimental studies.

At ORNL a parametric study of fillet reinforcement of radial nozzles attached to spherical shells is being conducted for the internal pressure loading case using a computer code based on a numerical shell analysis. This information is being used in code-rule development work at Battelle Memorial Institute.

5.2 Design Criteria for Piping, Pumps, and Valves

The ORNL piping program, Design Criteria for Piping, Pumps, and Valves, is the AEC-supported portion of a joint AEC (ORNL)—industry (PVRC) cooperative program to develop stress indices and simple analytical procedures for use in codes and standards, as well as to develop fundamental information on the structural behavior of piping system components. The ORNL program consists of a number of different tasks, some of which are carried out at Oak Ridge and others which are performed by vendors under subcontract.

During the present reporting period subcontract work was completed at Southwest Research Institute on the experimental stress analysis and low-cycle fatigue tests of five 12-in. ASA B16.9 tees, and at Westinghouse Research Laboratories on the photoelastic stress analysis of four one-half-scale models of 12-in. ASA B16.9 tees. Final reports are being prepared by both of these subcontractors.

Other subcontractors include Combustion Engineering, Inc., where 24-in. tees are being tested, and the University of California, Berkeley, where finite-element computer codes are being developed for the elastic response analysis of tees. At Combustion Engineering, preparations were completed for conducting an internal pressure low-cycle fatigue test on ORNL tee T-11 (24 × 24 × 24 in., sched 160), and strain-gage layout work was completed for tee T-13 (24 × 24 × 10 in., sched 160). At the University of California, most of the service function subroutines were completed, including a rather general input preparation package and a stress output package, and the entire program was assembled and operated.

Work continuing at Oak Ridge includes thermal stress tests on two 12-in. ASA B16.9 tees, thin-walled idealized tee studies, and experimental and analytical stress analyses of machined elbows. The thermal stress tests were reactivated, and one of the models is now ready for installation in the test frame. Test loadings were completed on the first of four machined elbows, and strain-gage instrumentation was completed on the third of four thin-walled tee models. Strain-gage installation was also started on the second machined elbow and on the fourth thin-walled tee model.

6. RDT STANDARDS PROGRAM

6.1 Program Objectives and Activities

This activity includes the organization, preparation, and distribution of the RDT standards. During this period, 14 tentative standards (7 of which superseded previous issues) and 4 amendments to tentative standards were issued; 35 draft standards (32 of which when approved will supersede previous issues) and 11 proposed amendments to tentative standards were submitted to RDT for approval; 2 draft standards were submitted to ANC for use on the LOFT project, and 2 draft standards were submitted to HEDL for use on the FFTF project; RDT numbers were assigned for 9 new standards; and drafts of 3 standards prepared by others were reviewed by ORNL.

6.2 Reactor Coolant System and Equipment Standards

Work continued in preparing and reviewing drafts and in reviewing and evaluating comments received on valve, pump, heat exchanger, nuclear-code supplement, reactor internal, vessel, and demineralizer standards. One valve standard was submitted to HEDL for use on the FFTF project, and one valve standard prepared by ANC was issued as tentative; one heat exchanger standard and one demineralizer standard were submitted to ANC for use on the LOFT project; and two code-supplement standards were issued as tentative.

6.3 Instrumentation and Controls and Electrical Standards

Work performed on instrumentation and electrical standards during this period included preparation of drafts; distribution of drafts for external review and comments; review and evaluation of comments received on standards prepared by ORNL; review of standards prepared by others and consolidation of comments for submittal to the requester; and submission of drafts to RDT for approval.

6.4 Programmatic and Procedural Standards

Work continued in the preparation of draft standards and amendments, review and evaluation of comments, and revision of quality assurance, maintenance, cleaning, air cleaning, and procedural standards. Four procedural standards were issued for trial use.

6.5 Material and Fabrication

Work involved updating existing tentative standards to conform with the requirements of the 1971 edition of Section III of the ASME Boiler and Pressure Vessel Code. Six revised tentative standards were issued for trial use; and 31 draft standards were submitted to RDT for approval.

7. GENERAL NUCLEAR SAFETY STUDIES

7.1 Seismicity and Seismic Resistance of Nuclear Power Plants

Drafts of the Shannon and Wilson—Agbabian Jacobsen Associates (SW-AJA) report "Soil Behavior under Earthquake Loading Conditions" were completed and are being reviewed. The SW-AJA interim report on soil analytical models, in-situ soil tests, shake table tests, and laboratory soil liquefaction tests is also essentially complete. The final SW-AJA program definition report has been issued. The May 7, 1971, revision of the proposed plan for USAEC Nuclear Safety Seismic Research has been distributed for comments. The UCLA final report on the EGCR tests is being prepared in draft form for review.

7.2 Ignition of Charcoal Adsorbers by Fission Product Decay Heat

An experiment is being developed to demonstrate the consequences of heat generated in a charcoal bed by the radioactive decay of adsorbed iodine. The flow diagram of the experiment has been designed for completely metal-enclosed primary and secondary containment.

8. NUCLEAR SAFETY INFORMATION

8.1 Nuclear Safety Information Center

The Nuclear Safety Information Center, established by the Commission, collects, evaluates, and disseminates nuclear safety information to governmental agencies, research and educational institutions, and the nuclear industry. The Center's basic computer file now contains references on more than 56,700 documents. E. W. Hagen presented a paper entigled "Reactor Availability and Station Reliability for Continuity of Service," at the ANS Reactor Operations Topical Meeting, Denver, Colorado, August 8–11. During the report period, 7 reports were published, and NSIC responded to 206 inquiries of which 137 were technical.

8.2 Technical Progress Review *Nuclear Safety*

The July–August 1971 issue of *Nuclear Safety*, a bimonthly Technical Progress Review sponsored by AEC's Division of Technical Information, was distributed in mid-August. The lead article reviewed the apparent public impact of some 800 items, in the daily and periodical press, pertaining to the nuclear industry. Other articles were prepared by 2 non-ORNL and 11 ORNL authors, including 7 *Nuclear Safety* and Nuclear Safety Program staff members. Subsequent issues of *Nuclear Safety* for July–August 1972 are in various stages of development.

1. LMFBR SAFETY STUDIES

1.1 FFM PROGRAM COORDINATION
(AEC Activity 04 60 01 09 1)

M. H. Fontana R. E. MacPherson

The fuel failure mockup (FFM) is a large high-temperature sodium-cooled facility in which simulated LMFBR core segments (ten-pin bundles) may be subjected to thermal and hydraulic testing at typical operating conditions. The core segment is simulated by cartridge electric heaters fabricated to duplicate a fuel-pin geometry and assembled with spirally wrapped wire like spacers containing internal thermocouple junctions. The program will provide information to be used by projects outside of ORNL, such as the FFTF project (and LMFBR manufacturers in the future), and therefore requires close coordination to remain responsive to program needs.

During this reporting period we have continued to maintain weekly telephone conversations with the FFTF project and to supply them with monthly management progress reports.

The instrumentation layout and preliminary test program for bundle 2A has been sent to FFTF and WARD for comments. Both organizations have provided significant input to the test program to assure that needs of the core design effort will be met.

Along with PNL, WARD, and AI, we have been asked to prepredict the temperature distribution in bundle 2A. Of particular interest is the temperature distribution across the bundle from flat to flat of the hexagonal can. We have supplied the necessary parameters to PNL and AI.

Testing with bundle 1A with no blockage has been completed, and the data are being analyzed.

We have been requested to expedite the evaluation of a bundle design with replaceable heaters. In addition to evaluating designs utilizing a sodium freeze seal, we have demonstrated that all heaters could be removed from bundle 1A in one day by grinding out the tube-to-header weld. This approach could qualify the present design as being capable of having heaters replaced.

Firm commitments have been received from RDT and HEDL to fund a heater development program at ORNL. A program plan is being prepared for evaluation and review. Specifications for manufacturing and quality assurance of heaters have been prepared, and heaters have been ordered for initial testing. Efforts have started on assuring the purity of boron nitride insulating powder and on evaluating preformed BN shapes for use in heaters. An ORNL quality assurance representative has been stationed at Watlow Manufacturing Company, St. Louis, Missouri, to monitor heater manufacturing for ORNL, HEDL, and ANL as required. A meeting was held with ANL on heater problems they are encountering in their sodium boiling studies.

A program was initiated to evaluate the physical response of stainless steel cladding to temperature transients that might be experienced in LMFBR perturbations such as partial blockages and partial flow coastdown. A program plan is being prepared.

1.2 ANALYTICAL STUDIES

(AEC Activity 04 60 01 09 1)

M. H. Fontana L. F. Parsly
J. L. Wantland

Analytical studies are being performed to plan the FFM experiments, interpret the results, and evaluate the performance of components. During this report period, evaluation of experimental data from bundle 1A was

begun, corrections were made to the ØRRIBLE code, and progress was made in making the CØBRA III code operable on ORNL's IBM 360 computers.

Preliminary Results, Bundle 1A Single-Rod Data

Using the corrected ØRRIBLE code,[1] the experimental results for the central rod heated case were best correlated by using the following parameters: $C_t = 0.015$, $C_d = 0.6$, and $C_s = 0.5$, where C_t is a turbulent cross-flow factor defined as $W_{T_\ell} = C_t\,(|W_\ell| + |W_m|)$; C_d is a multiplier on the diversion cross-flow term described in the following subsection of this report; and C_s is a factor on the flow sweeping defined as

$$W_{S_{\ell m}} = C_s W_\ell \frac{\pi(D + G)}{2\theta}\,[1 + \cos(3\theta)] \ .$$

In the above

$W_{T_{\ell m}}$ = local turbulent cross-flow mass velocity between channels ℓ and m, $lb_m\ hr^{-1}\ ft^{-2}$;

W_ℓ, W_m = the local values of the axial mass velocities in channels ℓ and m, $lb_m\ hr^{-1}\ ft^{-2}$;

$W_{S_{\ell m}}$ = the sweeping cross flow from channel ℓ to channel m, $lb_m\ hr^{-1}\ ft^{-2}$;

D = rod diameter, ft;

G = wire wrap diameter (gap thickness), ft;

θ = wire wrap azimuthal position relative to rod-to-rod center line.

Comparison between experimental and calculated data using these parameters is shown in Fig. 1.1. The rod numbering layout and identification of thermocouple positions are shown in Fig. 1.2. These results supersede those reported in the March–April bimonthly.[2]

Analyses of subsequent single-rod experiments referred to in the last bimonthly[3] have been completed. In these experiments all rods were heated individually, and the response of the thermocouples on all other rods was recorded and compared with calculated results. Some representative results are shown in Table 1.1. Representation of all the data would

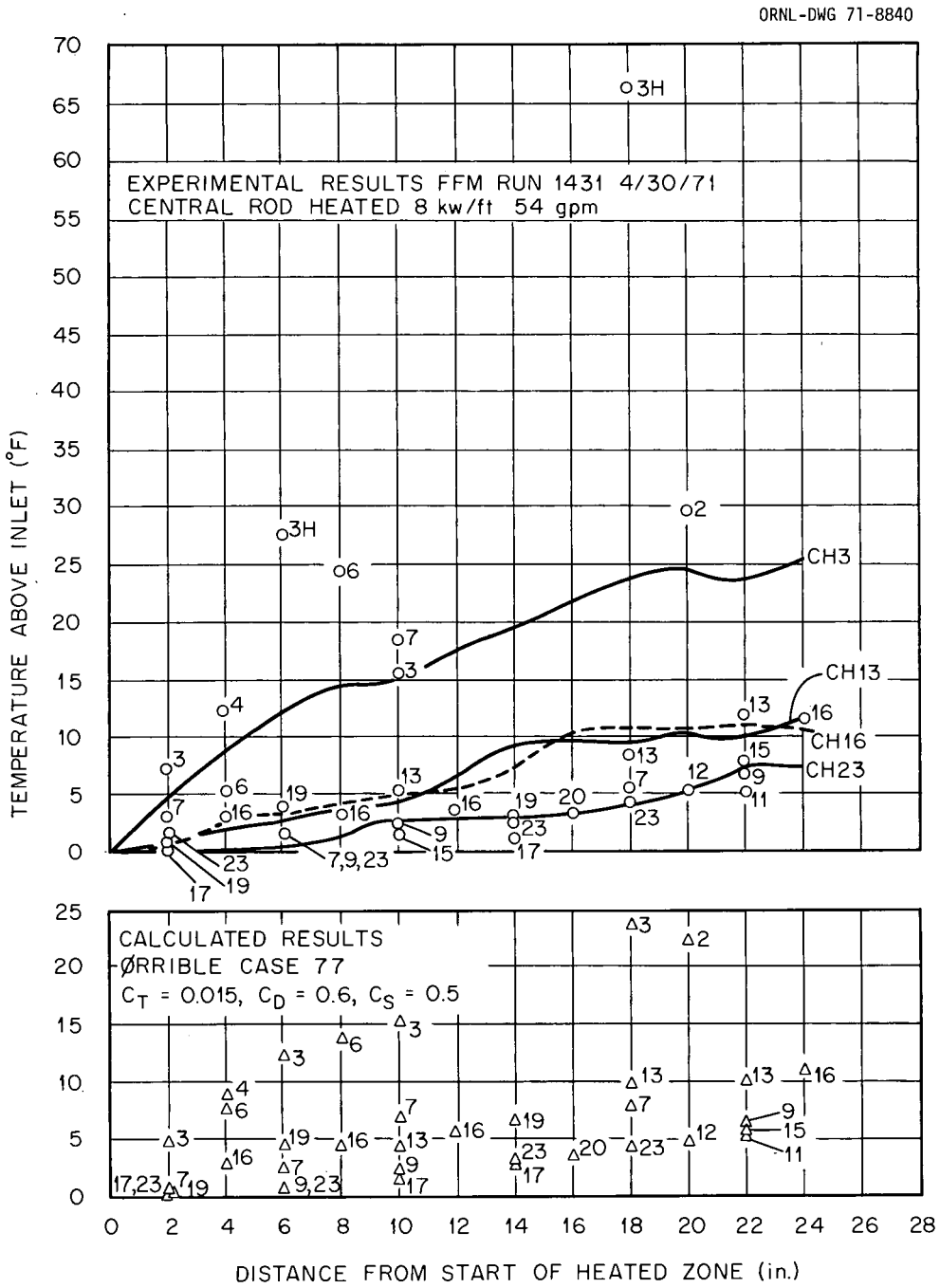

Fig. 1.1. Experimental points compared with calculated points for the case with the control rod heated only.

Table 1.1. Comparison of results of single-rod-heated experiments with calculated prediction

Temperature measured and calculated at location indicated

Heated rod	2-in. Level, rod 5, channel 3		6-in. Level, rod 1, channel 3		22-in. Level, rod 9, channel 9		14-in. Level, rod 17, channel 23		18-in. Level, rod 18, channel 23	
	E*	A†	E	A	E	A	E	A	E	A
1	21	24	92‡	57	34	33	23	16	23	23
2	3	0	9	5	194	121	12	15	26	26
3	4	3	21	19	122	65	3	3	5	6
4	26	23	58	58	3	4	1	0	1	0
5	71‡	24	93	52	8	7	13	6	14	9
6	11	4	23	13	11	9	56	39	51	46
7	3	1	7	4	40	31	93	90	150	108
8	2	0	2	0	149	136	3	7	12	16
9	2	0	3	1	185‡	126	1	2	5	5
10	3	0	3	1	64	44	1	0	3	1
11	3	0	5	6	18	19	1	0	1	1
12	4	0	5	5	5	5	1	0	3	1
13	6	3	11	14	3	3	2	1	3	2
14	6	0	6	3	3	1	6	2	9	3
15	4	0	6	3	5	2	16	9	20	12
16	3	0	3	1	2	0	42	30	46	35
17	3	0	3	1	20	9	174‡	103	129	114
18	3	0	2	0	48	21	114	102	183‡	121
19	3	0	3	0	82	61	19	34	64	54

*Experimental.

†Analytical.

‡Thermocouple is on heated rod and therefore is expected to read higher than sodium temperature.

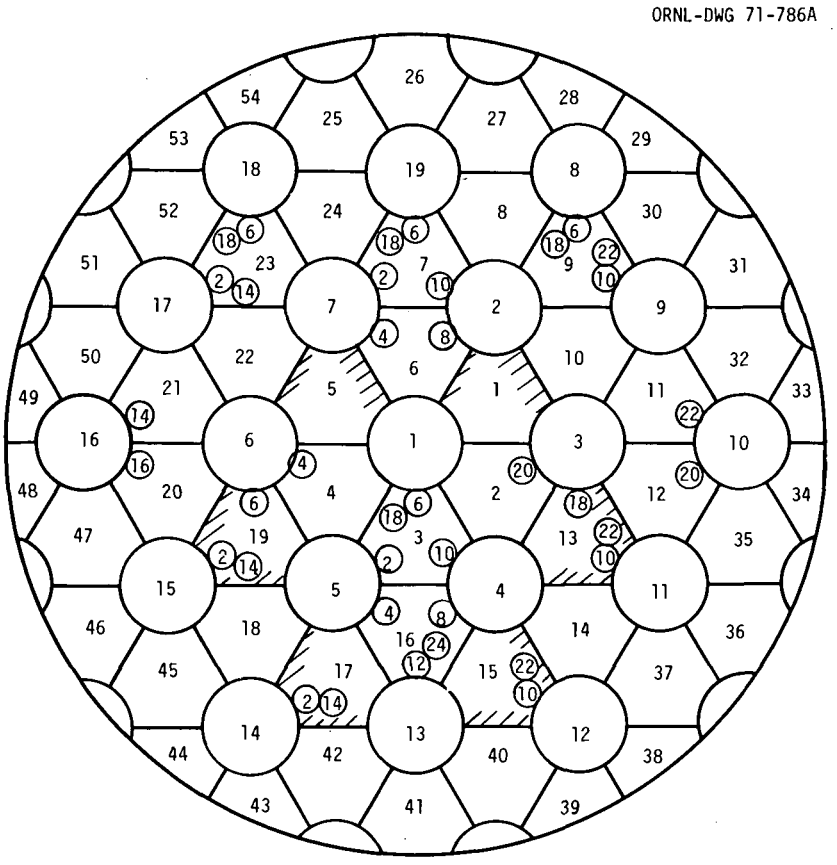

Fig. 1.2. Wire-wrapped thermocouple locations in the FFM bundle 1 test section. The small circles around the rods show the rotational or orientations of the thermocouples, and the numbers in the circles indicate their axial locations, in inches, downstream from the start of the heated section.

require 36 columns, and only a few are shown here. The full set will be published in a forthcoming topical report. For the experiment the power input was held constant at ∿8 kW/ft, and the flow was held at ∿11 gpm (∿1/5 normal flow) to allow easily detectable temperature spreads.

Multirod Experiments

The experiments performed with bundle 1A are listed in Section 1.5. We have not had time to evaluate all the data. However, preliminary analyses of some of the experiments with 18 rods heated have been initiated. (Rod 10 failed before fuel bundle experiments started.)

From analyses of single-rod experiments it was expected that the temperatures read by the wire-wrap thermocouples on heated rods would be significantly higher than the local sodium temperature. In order to develop a set of initial reference points, the temperature distribution expected in a 19-rod bundle with rod 10 unheated was calculated with the ØRRIBLE code using the values of C_t = 0.015, C_d = 0.6, and C_s = 1.0. (The temperature readings indicated by thermocouples on unheated rods are expected to be reasonably close to that of the local sodium.) The calculated results for representative channels 3, 16, 15, 21, 11, 40, and 41 are given in Fig. 1.3. The calculated radial temperature profiles at various axial positions are plotted in Fig. 1.4. Channels 3, 16, 15, 40, and 41 were chosen as being representative of channels belonging to the central, second, third, fourth, and fifth "ring" of channels.

A comparison of temperatures indicated by the wire-wrap thermocouples with calculated sodium temperature at the same point is shown in Fig. 1.5. Notice that the measured temperatures are consistently higher than the calculated temperatures. Also, the experimental points for channel 11 at 22 in. and channel 12 at 20 in. downstream from the entrance to the heated zone are obtained for the wire wrapped on rod 10, which was inoperative. One would expect these experimental readings to be closer to sodium temperature. This is indicated to some extent in Fig. 1.5.

The differences between calculated and measured temperature are shown in Fig. 1.6 for three cases at power levels of 7.65, 5.64, and 3.93 kW/ft. Notice that the relationship between power and ΔT is evident. Somewhat puzzling is the evidence that the ΔT is greater at longer distances into the bundle, which run hotter. One explanation for this might be the differential thermal expansion of stainless steel and boron nitride, which could cause a decrease in contact resistance between the clad and the BN except where held in by the wire-wrap pressure. This effect would be more noticeable at higher temperatures. A test of this might be to plot the ΔT versus the local measured temperature for various power levels. Values of ΔT relatively independent of power level would indicate an overriding effect of temperature.

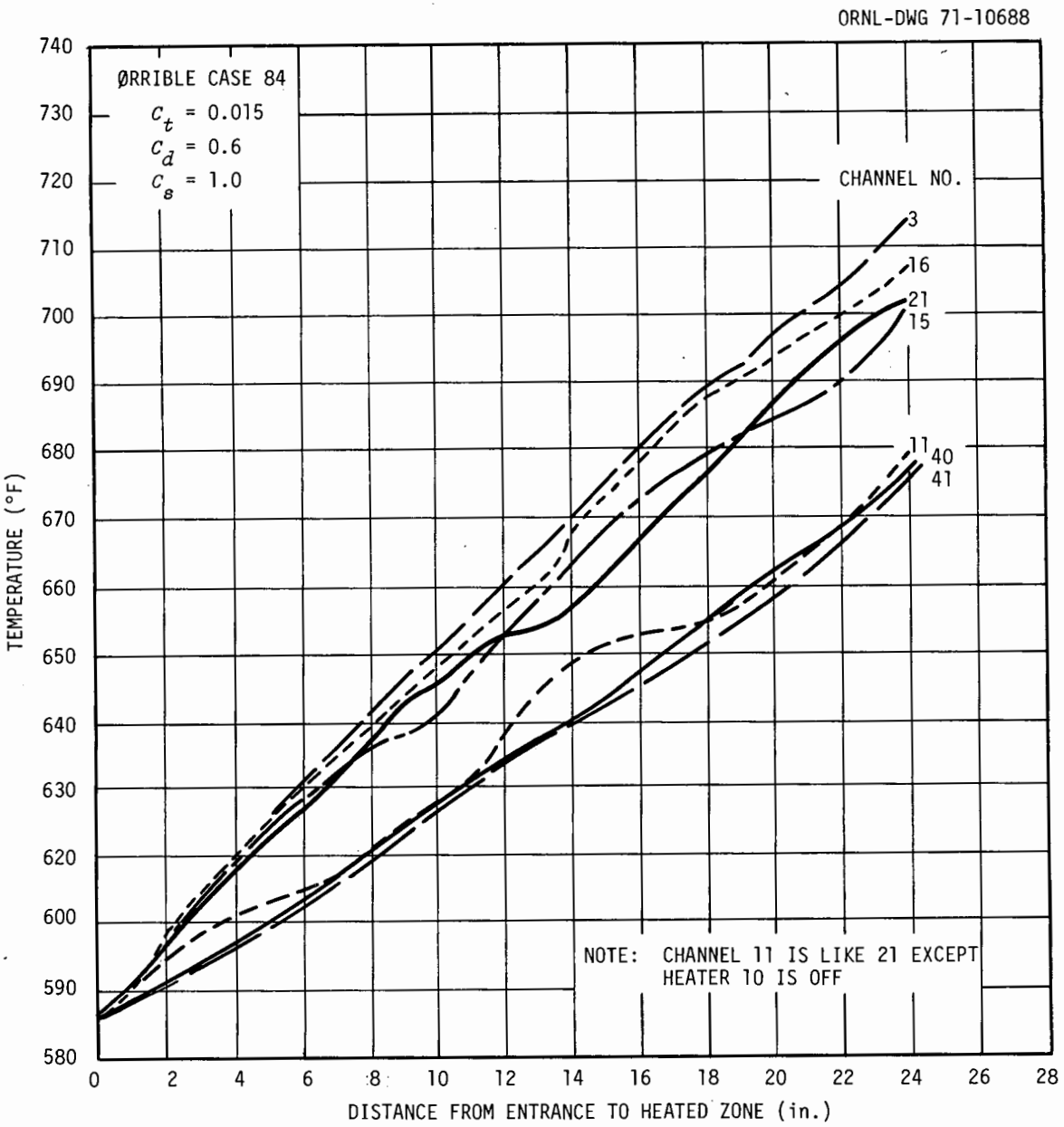

Fig. 1.3. Calculated temperature distribution for conditions equivalent to experimental run 2C (18 rods heated at 5.64 kW/ft average, rod 10 out, 54 gpm flow, 596°F inlet temperature).

Figure 1.7 shows this plot for runs at 7.65 and 5.825 kW/ft with similar inlet temperatures (799.9 and 781.1°F) and for runs at 5.64 and 3.93 kW/ft at 597.2°F inlet temperature. Notice that the definite trend is that ΔT increases with local temperature and that the effect of power level is more pronounced at the lower temperature. This would indicate

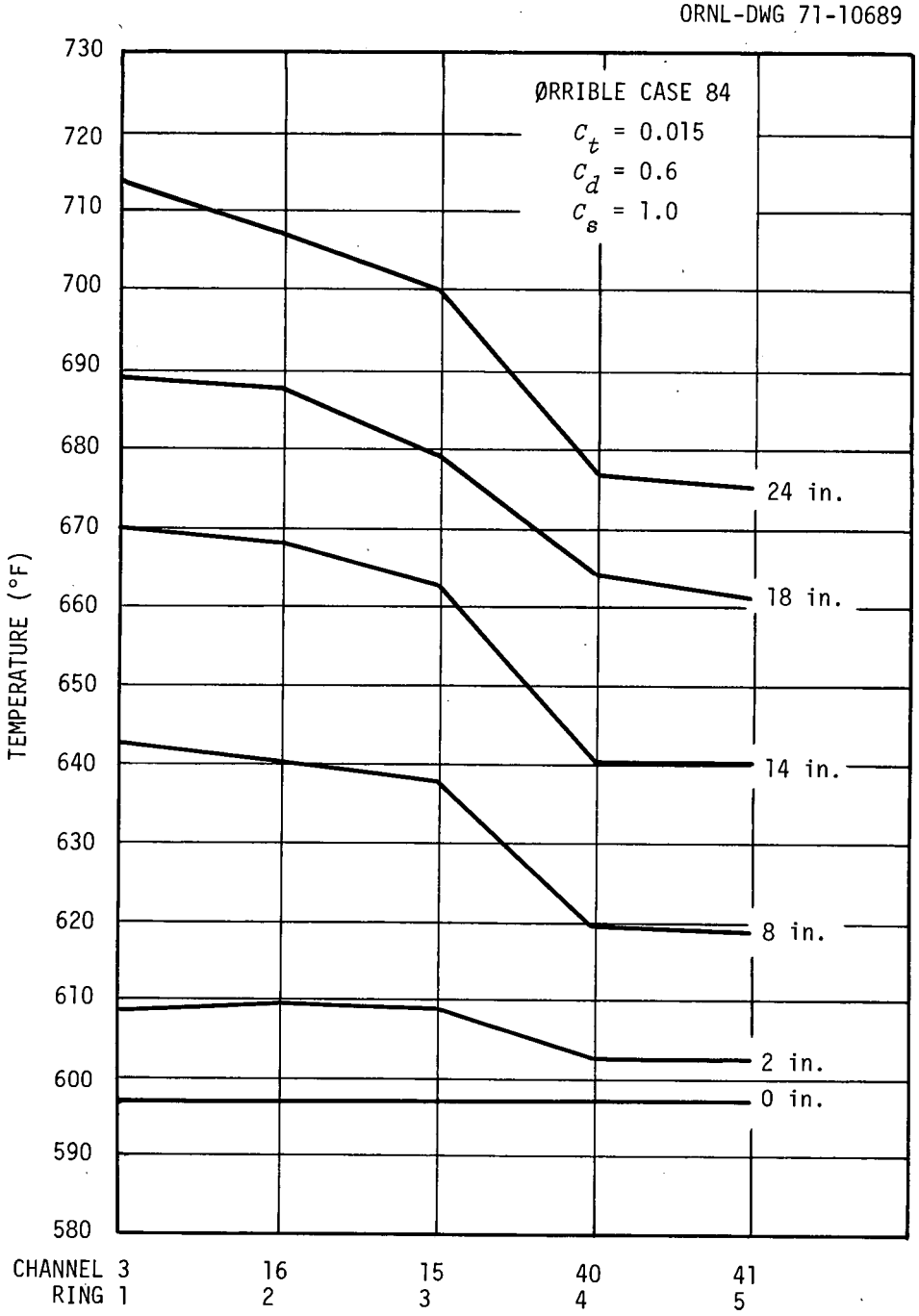

Fig. 1.4. Radial temperature profile in 19-rod bundle. Channels shown are representative of others of same distance from center.

ORNL-DWG 71-10690

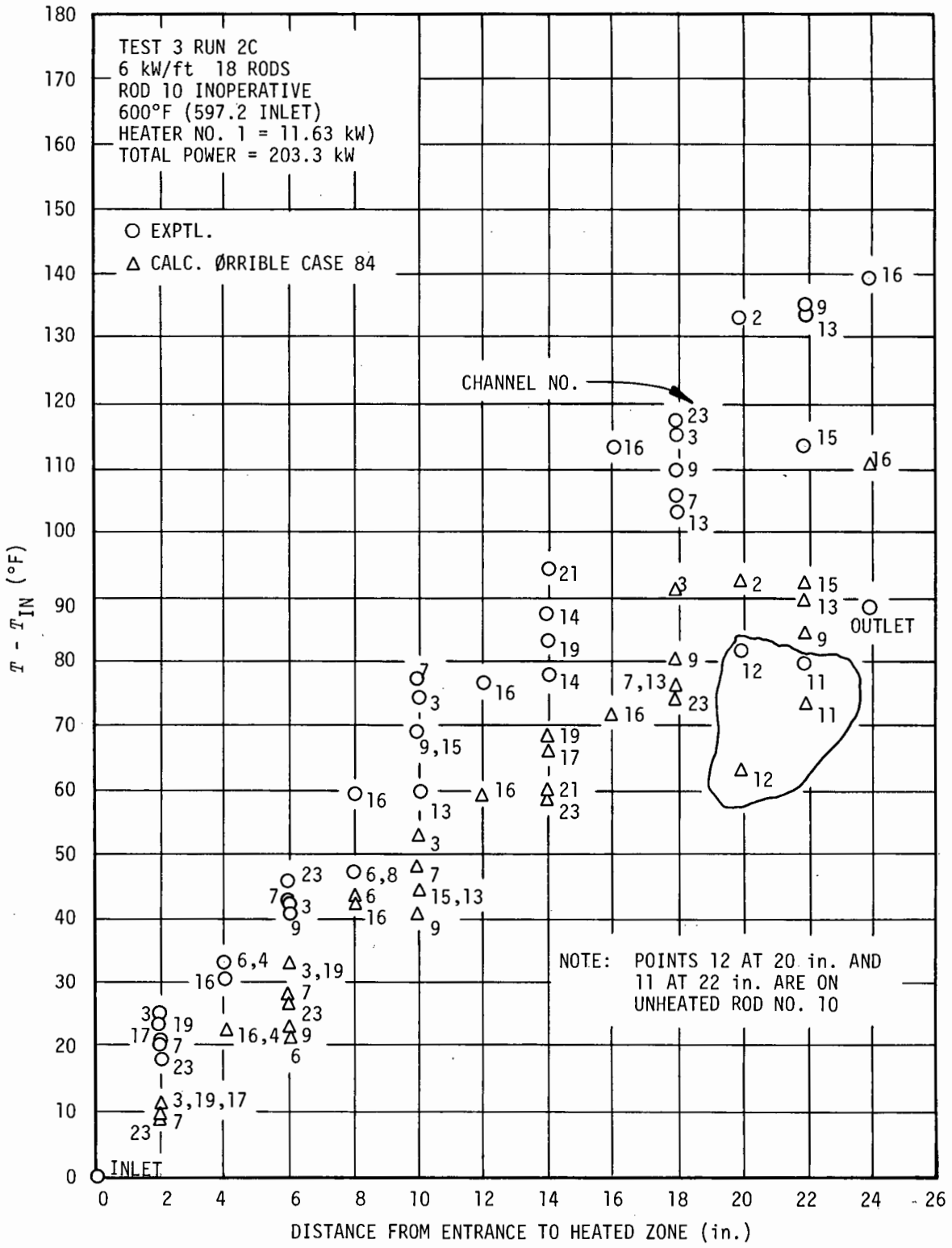

Fig. 1.5. Measured wire-wrap temperature versus calculated sodium temperature at same positions.

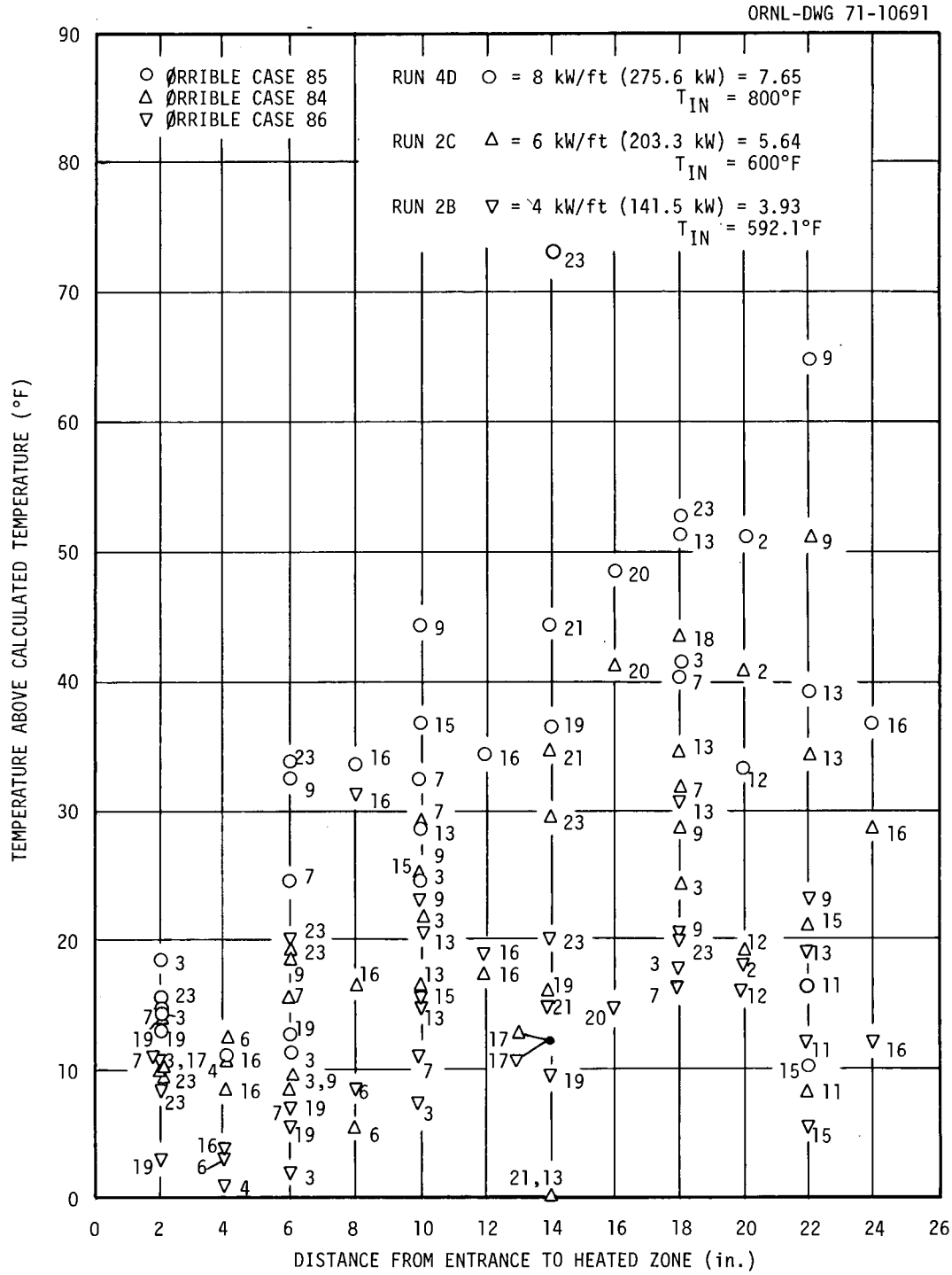

Fig. 1.6. Temperature difference between measured and calculated temperatures for three different power levels.

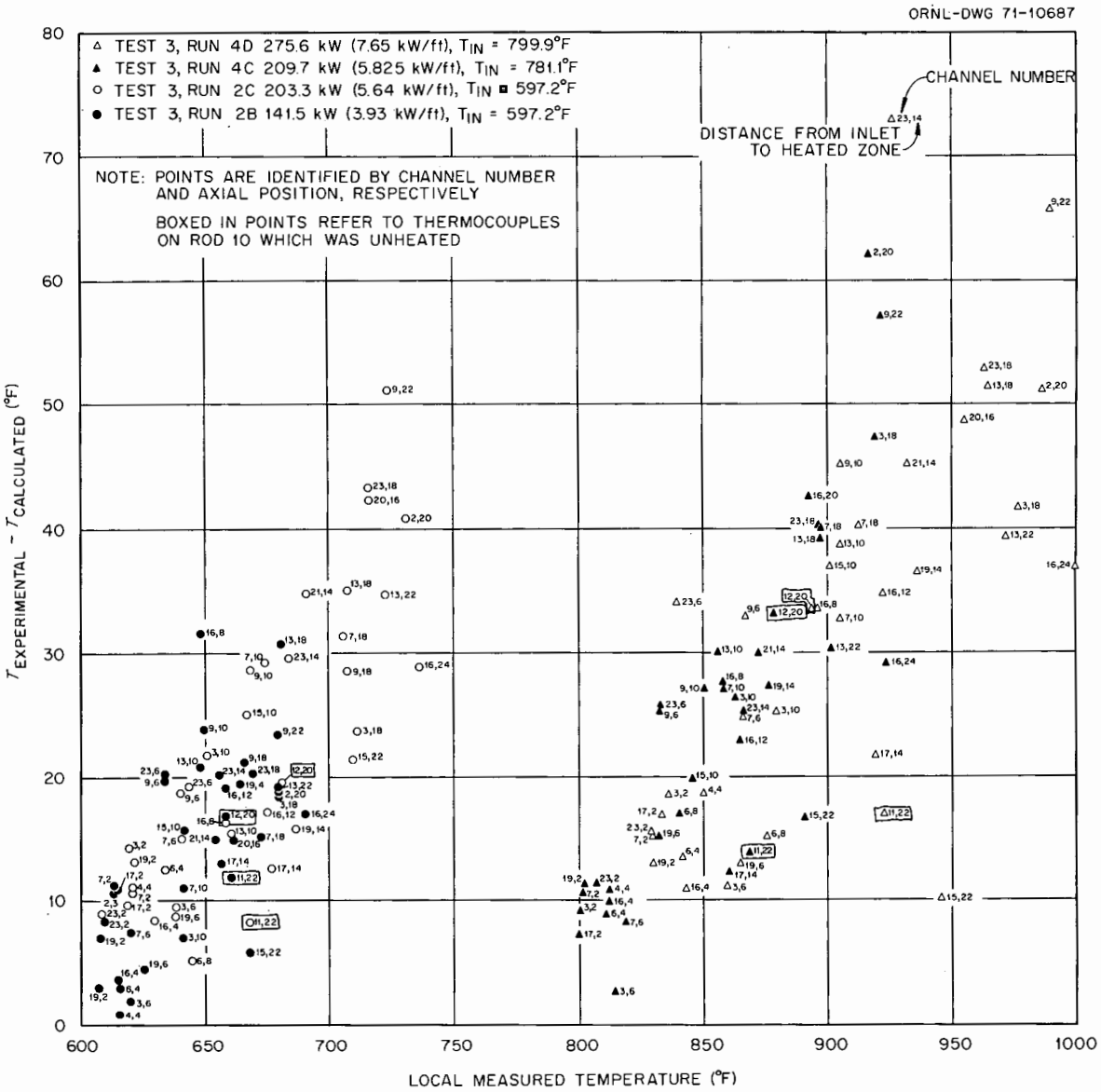

Fig. 1.7. Difference between measured and calculated temperatures plotted as a function of local temperature.

that unknown temperature effects become more important relative to the expected power effect at higher temperatures.

One would expect that the ΔT would be linearly proportional to power level. A check of this is shown in Fig. 1.8, which shows the effect of power level on the differences between measured and calculated temperatures in channels 3, 7, and 9. If the ΔT were due only to heat transfer effects, the temperature excess of each point should be on a straight

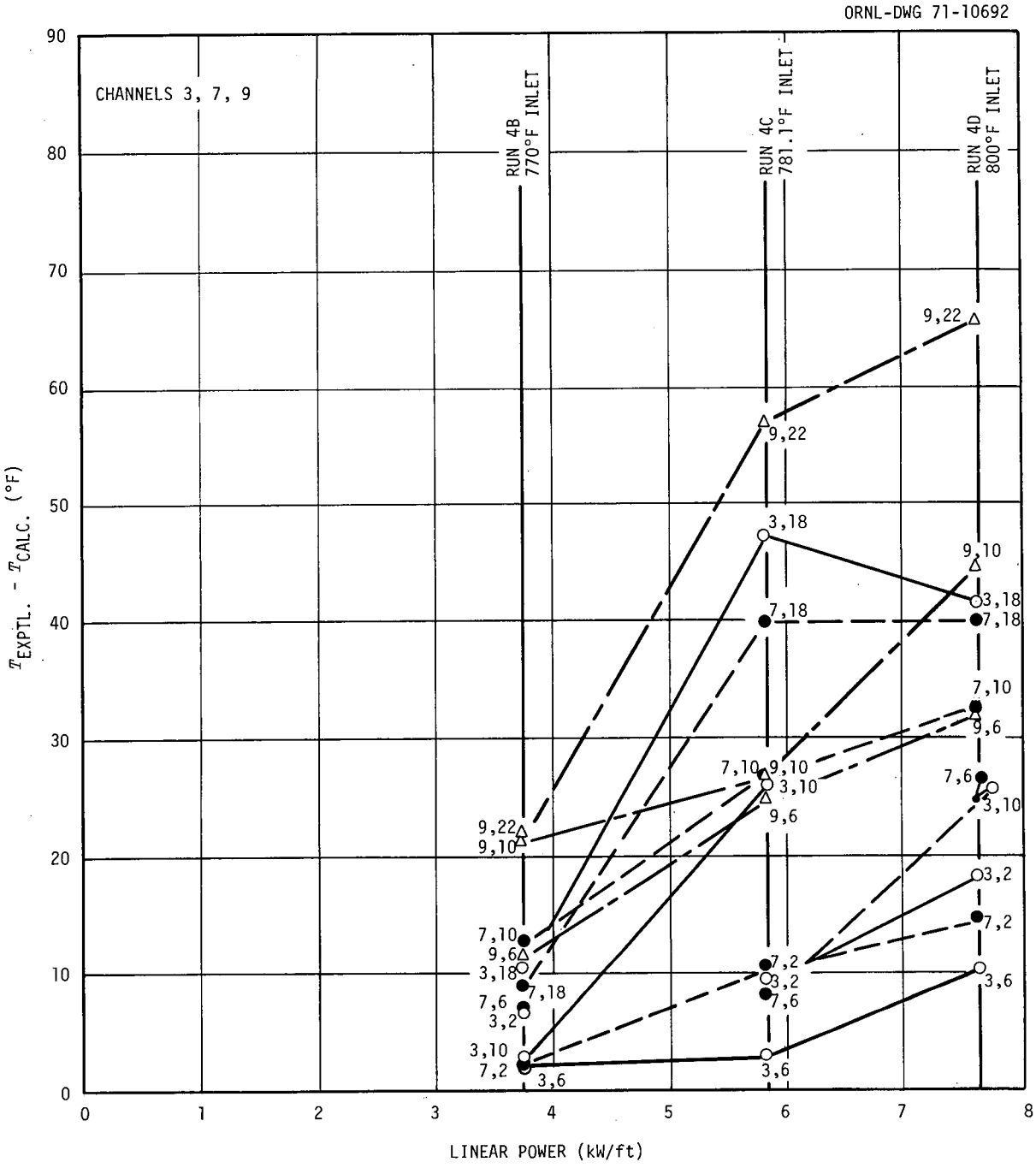

Fig. 1.8. Effect of power level on difference between experimental and measured temperatures.

line with an intercept at zero. It is apparent that this is nearly so in only a few cases and that the linearity becomes worse at higher power levels and at positions at further distances from the inlet to the heated zone (which are hotter for all runs).

It is apparent that these data are not completely understood. We must extrapolate the temperature excesses obtained from single-rod runs to the multiple-rod case; we must repeat the calculations with varying parameters to attempt a more consistent fit with the data. However, the following statements are probably valid.

1. The temperatures read by the thermocouples are probably accurate indications of the temperature within the wire wrap. The thermocouples were calibrated and corrected by isothermal tests; they appeared to behave consistently when not subjected to heat flux, returned to initial values after each experiment, and were reproducible under identical conditions. The effect where thermocouple readings are adversely affected by longitudinal temperature gradients in unannealed thermal element wires should not be present here because the wires were annealed prior to installation and were subjected to isothermal soak at 1000°F prior to any power runs.

2. The wire wraps can be at temperatures as much as 70°F above sodium temperature. Even if one has no faith in the calculated results from ØRRIBLE, one can determine the maximum possible sodium temperature in the central six channels by simply calculating the temperature rise assuming no mixing. The temperatures obtained are consistent with the ØRRIBLE results for those channels and are still lower than measured temperatures by 40°F at certain points.

3. Each wire wrap behaved differently. The calibration of one with respect to power level could not be extrapolated to others.

4. The restraining effect of the wire wrap on the cladding under differential thermal expansion decreases the initial resistance at the point beneath the wire and causes a hot spot.

5. In future tests, the wire wrap should be spot welded to this heater cladding to prevent wandering of the junction.

6. Until proved otherwise, it would be prudent to assume that these hot spots exist in the FFTF core.

We will continue to analyze the data, and the complete results from bundle 1A will be reported in a topical report.

Mathematical Models of Flow in Rod Bundles

An error was found in the derivation of the expression for pressure diversion cross flow previously presented.[4] This error has been corrected, and the resulting derivation for pressure diversion cross flow in terms of axial mass velocities follows. As before, it is assumed that the relationship derived for cross flow between two otherwise isolated channels is applicable to a multichannel system.

Consider two parallel interconnected channels 1 and 2 with the same cross-sectional areas and wetted perimeters. Channel 1 has an inlet blockage, so that flow will be diverted from channel 2 into channel 1.

The relationship usually used for this pressure diversion cross flow is

$$P_2 - P_1 = C_D \frac{W_P^{\,2}}{2\rho g_c} , \tag{1}$$

where

P_1, P_2 = pressure in channels 1 and 2, lb_f/ft^2;

C_D = diversion cross-flow coefficient, dimensionless;

W_P = pressure diversion cross flow from channel 2 to channel 1, $lb_m\ hr^{-1}\ ft^{-2}$;

ρ = fluid density, lb_m/ft^3;

g_c = gravitational conversion factor, $4.170 \times 10^8\ lb_m\text{-}ft/lb_f\text{-}hr^2$.

The equations of conservation of axial momentum for the two channels are

$$\frac{dP_1}{dx} = \frac{1}{\rho g_c}\left[\frac{-f_F\bar{p}}{2A} W_1^{\,2} - \rho^2 g - \frac{dW_1^{\,2}}{dx} + \frac{G}{A} W_T(W_2 - W_1) + \frac{G}{A} W_P W_2\right] , \tag{2}$$

and

$$\frac{dP_2}{dx} = \frac{1}{\rho g_c}\left[\frac{-f_F\bar{p}}{2A} W_2{}^2 - \rho^2 g - \frac{dW_2{}^2}{dx} + \frac{G}{A} W_T(W_1 - W_2) - \frac{G}{A} W_P W_2\right] , \qquad (3)$$

where

x = axial distance in the direction of flow, ft;
f_F = fanning friction factor, dimensionless;
$\bar{p}$ = wetted perimeter of either channel, ft;
A = cross-sectional area of either channel, ft^2;
W_1, W_2 = axial mass velocity in channels 1 and 2, $lb_m\ hr^{-1}\ ft^{-2}$;
G = width of the gap separating the two channels, ft;
W_T = turbulent interchange between the two channels, $lb_m\ hr^{-1}\ ft^{-2}$;
g = gravitational constant, 4.17×10^8 ft/hr^2.

The derivative of Eq. (1) with respect to x is taken, and this is combined with Eqs. (2) and (3) to eliminate pressure as a variable, yielding

$$C_D W_P \frac{dW_P}{dx} = \frac{f_F\bar{p}}{2A} (W_1{}^2 - W_2{}^2) + \frac{dW_1{}^2}{dx} - \frac{dW_2{}^2}{dx} + \frac{2G}{A} [W_T(W_1 - W_2) - W_P W_2] . \qquad (4)$$

The boundary conditions are $W_1 = 0$ at $x = 0$, $W_2 = 2W_\infty$ at $x = 0$, and $W_1 = W_2 = W_\infty$ at $x \to \infty$, where W_∞ is a constant. From conservation of mass,

$$W_1 + W_2 = 2W_\infty \qquad (5)$$

and

$$\frac{dW_1}{dx} = -\frac{dW_2}{dx} = \frac{G}{A} W_P . \qquad (6)$$

The turbulent interchange relationship is

$$W_T = C_T (W_1 + W_2) , \qquad (7)$$

where C_T is turbulent cross-flow coefficient (dimensionless). Substituting Eqs. (5), (6), and (7) into Eq. (4) and putting into dimensionless form by letting

$$s = \frac{xG}{A} \tag{8}$$

and

$$y = \frac{W_1}{W_\infty} , \tag{9}$$

we obtain

$$\frac{C_D}{2} \frac{dy}{ds} \frac{d^2y}{ds^2} = \left(\frac{f_F \bar{p}}{G} + 4C_T\right)(y - 1) + y \frac{dy}{ds} , \tag{10}$$

with the boundary conditions $y = 0$ at $s = 0$ and $y = 1$ at $s \to \infty$. Letting

$$\frac{dy}{ds} = n$$

so that

$$\frac{d^2y}{ds^2} = n \frac{dn}{dy} ,$$

Eq. (10) becomes

$$\frac{C_D}{2} n^2 \frac{dn}{dy} = \left(\frac{f_F \bar{p}}{G} + 4C_T\right)(y - 1) + yn , \tag{11}$$

with the boundary condition $n = 0$ at $y = 1$. Letting

$$q = \left(\sqrt{\frac{C_D}{6}}\, n\right)^3 \tag{12}$$

and

$$C = \sqrt{\frac{C_D}{6}} \left(\frac{f_F \bar{p}}{G} + 4C_T\right) , \tag{13}$$

Eq. (11) becomes

$$\frac{dq}{dy} = C(y - 1) + yq^{1/3} , \tag{14}$$

with the boundary condition $q = 0$ at $y = 1$.

Equation 14 was solved numerically for $0 \leq y \leq 1$ and $0.04 \leq C \leq 0.4$. The resulting family of curves was fitted by the empirical expression

$$q^{1/3} = 0.75C^{0.44}(1 - y)^{0.80C^{-0.32}} ,$$

or in terms of the original equations,

$$W_P = 0.375 \sqrt{\frac{6}{C_D} \frac{W_2{}^2 - W^2}{|W_2 - W_1|}} \, C^{0.44} \left(\frac{|W_2{}^2 - W_1|}{W_2 + W_1}\right)^{0.80C^{-0.32}} , \tag{15}$$

where C is defined by Eq. (13).

Equation (15) is the relationship for pressure diversion cross flow used in the present version of ØRRIBLE. This is not a completely satisfying procedure, and work is continuing on this aspect. The numerical solution of Eq. (15) was not well behaved in the neighborhood of $y = 1$, but otherwise consistent results were obtained with different numerical techniques. It is tacitly assumed that the results for two interconnected channels are applicable to multichannel systems. Other procedures may be more valid; some of these are being investigated.

CØBRA III Computer Program

The CØBRA III computer program was received at the beginning of the present reporting period. Several minor changes were necessary to convert it from the FØRTRAN V language in which it was written to FØRTRAN IV for ORNL's IBM 360 computers. FØRTRAN V allows nonexecutable statements to be specified in a subprogram and to be added to the other routines by a single-card INCLUDE statement. FØRTRAN IV does not have this feature, and it was necessary to prepare a package of nonexecutable statements and to insert this into the main program and subroutines. Since experience has shown this is generally necessary in iterative calculations, all

real variables were declared to be double precision. An early test run indicated that some variables were being used in the program without being defined, and coding was added to clear the memory area used by the program to zero when starting a run.

The test case included in the program package has been run, and agreement between results obtained at ORNL and at PNL is excellent.

A modified version of the program for use with the FFM has been prepared. The array sizes had to be changed to accommodate 19 rods, 54 flow channels, and 72 gaps for FFM bundle 1. Also the input format for physical property data had to be changed to permit using "E" format for sodium vapor pressure. Testing of the modified version will start August 30.

References

1. J. L. Wantland, *ØRRIBLE — A Computer Program for Flow and Temperature Distribution in LMFBR Fuel Rod Bundles*, ORNL-TM-3516, to be issued.

2. M. H. Fontana et al., *ORNL Nuclear Safety Research and Development Bimonthly Report for March–April 1971*, ORNL-TM-3411, p. 26.

3. M. H. Fontana et al., *ORNL Nuclear Safety Research and Development Bimonthly Report for May–June 1971*, ORNL-TM-3483, p. 24.

4. Wm. B. Cottrell, *ORNL Nuclear Safety Research and Development Program Bimonthly Report for March–April 1971*, ORNL-TM-3411, pp. 22–23.

1.3 WATER SYSTEM MOCKUP OF THE FFM FACILITY

(AEC Activity 04 60 01 09 1)

W. R. Gambill

Simulations using water coolant will provide an independent check on the FFM sodium data with respect to the effect of inlet and internal partial flow blockages on the downstream flow distribution, as well as an indication of optimal locations for future temperature measurements.

During the reporting period, data reduction for tests 11 through 13 with the ten-channel inlet blockage was completed, as shown in Table 1.2

Testing with the six-subchannel inlet blockage plate was completed with tests 14 through 16. Test conditions and results are summarized in Tables 1.3 and 1.4 respectively.

Table 1.2. Test results for ten-channel inlet blockage in FFM water mockup[a]

Test No.	Maximum value of $t_{b,s} - t_{b,csc}$ (°C)	$\frac{\text{Max } (\Delta t_{\text{radial}})_{\text{exptl}}}{\text{Calc } (\Delta t_{\text{radial}})_{\text{unmixed}}}$ (%)	$\left(\frac{dt_{\text{exptl}}}{dt_{\text{unmixed}}}\right)_{\text{S1 to S3}}$ (%)
11	−0.2	18–25	12
12	−0.5	26	24
13	−0.2	44–45	4–8

[a]The symbol $t_{b,s}$ denotes the station-average water temperature (six readings) and $t_{b,csc}$ the central-subchannel average water temperature (three readings); dt is the axial temperature rise of the water between stations 1 and 3 for the central subchannels.

Table 1.3. Test conditions[a] for six-channel inlet blockage experiments in the FFM water mockup

Test No.	Q (gpm)	V_{mn} (fps)	Mean $t_{b,\text{in}}$ (°C)	$N_{\text{Re,in}}$	ϕ_0 (Btu hr^{-1} ft^{-2})	Max Δt_{radial} (°C)		
						S1	S2	S3
14	109.3	5.7	28.8	0.20×10^5	2.13×10^5	0.9	1.5	3.7
15	64.5	3.4	30.0	0.12×10^5	1.58×10^5	0.6	1.9	3.4
16	113.4	5.9	29.7	0.22×10^5	2.13×10^5	1.0	1.3	2.4

[a]Q denotes the volumetric flow rate; V_{mn} the mean coolant velocity in the bundle based on Q and the net flow area; $t_{b,\text{in}}$ the inlet bulk water temperature; $N_{\text{Re,in}}$ the inlet Reynolds number; ϕ_0 the heater-rod heat flux; and Δt_{radial} the radial difference in bulk water temperature at each of the three axial stations (S1, S2, and S3). Station 1 is 21 in. above the rod-bundle base, and stations 2 and 3 are 39 and 51 in., respectively, above the base.

Table 1.4. Test results[a] for six-channel inlet blockage experiments in the FFM water mockup

Test No.	Mean inferred velocity[b]		Av deviation of V_{mn} from inferred V_{mn} (%)	$\frac{V_{mn}}{(V_x)_{inf,min}}$	$K_{adjusted}$	Maximum value of $t_{b,s} - t_{b,csc}$ (°C)	$\frac{\text{Max } (\Delta t_{radial})_{exptl}}{\text{Calc } (\Delta t_{radial})_{unmixed}}$ (%)	$\left(\frac{dt_{exptl}}{dt_{unmixed}}\right)$ S1 to S3 (%)
	V_{inf} (fps)	Number of points						
14	5.4	18	+5.1	1.18	0.0425	−0.1	23–36	14
15	3.6	18	−6.3	1.04	0.0380	−0.3	23–26	20
16	5.8	18	+1.0	1.13	0.0408	−0.4	21–24	6

[a] V_{inf} denotes inferred velocity and V_x a local (inferred) velocity; K is the numerical factor in the velocity inference equation, $V = KF_{pp}h^{1.205}$, in which F_{pp} is a physical properties factor and h the local heat transfer coefficient; the symbol $t_{b,s}$ denotes the station-average water temperature (six readings) and $t_{b,csc}$ the central-subchannel average water temperature (three readings); dt is the axial temperature rise of the water between stations 1 and 3 for the central subchannels.

[b] Using $K = 0.0404$, as for previously reported data (W. R. Gambill, *ORNL Nuclear Safety Research and Development Bimonthly Report for January–February 1971*, ORNL-TM-3342, pp. 27–30).

The 1/5-scale Lucite shroud model, when internally pressurized, fractured at 380 psig, preferentially along a longitudinal bonded seam. The effort to determine the ultimate bursting pressure of this model has been discontinued for the present, since operation of the prototype FFM mockup shroud at $P \leq 3$ atm absolute allows adequate nonboiling film temperature differences of less than about 10^5 to be attained for N_{Re}.

The amplitude of the helical profile imparted to each Lucite rod by the attached spiral spacer wire was measured in two perpendicular planes at several axial positions for each of four rods selected at random. The overall average deviation of the rod center line from the center line of a perfectly straight rod is $\sim$0.080 in.

Design, fabrication, and installation of the first internal blockage element, covering the six central subchannels at mid-heater length, were completed. The first bond formed between the concave edges of the blockage plate and the adjacent Lucite rods was imperfect, but the second attempt produced an apparently sound joint. The bundle was reassembled and reinstalled in the system, and the recorder and flow meter were calibrated. Thermal testing was begun with test 17 ($N_{Re} = 1.24 \times 10^4$, $\phi = 1.56 \times 10^5$ Btu hr^{-1} ft^{-2}) and test 18 ($N_{Re} = 2.15 \times 10^4$, $\phi = 1.85 \times 10^5$ Btu hr^{-1} ft^{-2}). We are reducing the data from these tests before beginning test 19, for which N_{Re} will be $\sim 10^5$ and ϕ will be $\sim 5 \times 10^5$ Btu hr^{-1} ft^{-2}.

1.4. FFM NOISE ANALYSIS
(AEC Activity 04 60 01 09 1)

D. N. Fry

Cross correlations were performed of thermocouple response to the power pulses normally imposed to individual heaters in the FFM by the SCR power control system. It appears that these correlations can be used to determine local velocity profiles and flow regime (i.e., laminar vs turbulent) within the rod bundle.

Because of a shift in scheduling of other experiments in SEFOR, time was not available to complete the analysis for this bimonthly. It will be reported in the next issue and in a separate topical report.

1.5 FFM FACILITY ACTIVITIES

(AEC Activity 04 60 01 09 1)

R. E. MacPherson P. A. Gnadt
A. G. Grindell

Test Operation with Bundle 1A

The tests scheduled for bundle 1A (previously bundle 1) were completed. The data points taken for test series 1 (tests 1, 2, and 3) are listed in Tables 1.5—1.7. Figure 1.9 shows the locations of the wire-wrap thermocouples installed for these tests. Figure 1.10 shows the other system parameters measured during each of the tests, superposed on a schematic drawing of the FFM sodium circulation system.

Several additions were made to the planned test program during this course of operation. The results of all the tests are discussed in Section 1.2. The electromagnetic flow meter was calibrated over the range 1 to 18 gpm by heat transport based on measurements of sodium temperature and test section input power, which was varied from 6 to 110 kW. The tests were conducted at approximately 650°F inlet temperature with a nominal temperature difference of 150°F across the test section. Good agreement with flow meter calibration curves was obtained. The system operated stably at ∿1 gpm and 6 kW total test section power.

Special single-heater-only tests were run on heaters 2, 4, 5, and 7 at 700°F sodium inlet temperature, 2, 4, 5, and 8 kW/ft, and at 54 gpm sodium flow to calibrate the wire-wrap thermocouples. Single-heater-only tests were run on each of the 19 rods at 700°F sodium inlet temperature, 8 kW/ft, and 10 gpm for the same reason.

Table 1.5. Test series 1 — test 1

Purpose: To measure test section pressure drop as a function of flow rate over the operating temperature range without a test piece installed

Run No.	Test section			
	Inlet temperature (°F)	Outlet temperature (°F)	Flow (gpm)	Power (kW/ft)
1-A	600	600	20	0
1-B	600	600	30	0
1-C	600	600	40	0
1-D	600	600	50	0
1-E	600	600	60	0
1-F	600	600	70	0
2-A	900	900	20	0
2-B	900	900	30	0
2-C	900	900	40	0
2-D	900	900	50	0
2-E	900	900	60	0
2-F	900	900	70	0
3-A	1000	1000	20	0
3-B	1000	1000	30	0
3-C	1000	1000	40	0
3-D	1000	1000	50	0
3-E	1000	1000	60	0
3-F	1000	1000	70	0
	1040	1040	150	0
	1040	1040	200	0
	1040	1040	227	0

Measure: Test section inlet pressure, psig; test section outlet pressure, psig.

Special instructions: Test piece not installed.

A special series of tests was run to check flow through the center of the bundle in which heater 1 (in the center of the bundle) was disconnected electrically. All other heaters except heaters 9 and 10, which had shorted internally during previous runs, were operated sequentially at 2, 4, and 6 kW/ft, with the sodium inlet temperature at 600°F and a sodium flow of 54 gpm.

Table 1.6. Test series 1 — test 2

Purpose: To perform a comparative calibration of all test piece thermocouples; to determine the effect of heat flux, flow, and temperature level on the temperature of the wire-wrap thermocouple relative to the bulk sodium temperature; to obtain insight into flow patterns in an unblocked rod bundle

Run No.	Test section			
	Inlet temperature (°F)	Outlet temperature (°F)	Flow (gpm)	Power[a] (kW/ft)
1-A	600	600	54	0
1-B	900	900	54	0
1-C	1200	1200	54	0
2-A	700		54	2
2-B	700		54	4
2-C	700		54	6
2-D	700		54	8
2-G	700		54	0
3-A	700		43.2(80%)	8
3-B	700		32.4(60%)	8
3-C	700		21.6(40%)	8
3-D	700		10.4(20%)	8
4-A	800		54	10
4-B	900		54	10
4-C	1000		54	10
4-D	1100		54	10
5-A	700		54	8(2)[a]
5-B	800		54	8(2)[a]
5-C	900		54	8(2)[a]
6-A	700		54	8(3)[a]
6-B	700		32.4(60%)	8(3)[a]
6-C	700		10.4(20%)	8(3)[a]

Measure: Test section outlet temperature, °F; test piece temperatures, °F.

[a]Special instructions: 1. Power applied only to rod 1 (central rod) except as noted. 2. Power applied to pins 1 through 7. 3. Power applied to pins 1 through 4.

Table 1.7. Test series 1 — test 3

Purpose: To obtain base-line pressure drop and temperature data on the unblocked bundle over a range of power levels and flow rates

Run No.	Test section			
	Inlet temperature (°F)	Outlet temperature (°F)	Flow (gpm)	Power (kW/ft)
1-A	600	600	70	0
1-B	600	600	60	0
1-C	600	600	50	0
1-D	600	600	40	0
1-E	600	600	30	0
1-F	600	600	20	0
1-G	600	600	10	0
1-H	600	600	17	0
2-A	600		54	2
2-B	600		54	4
2-C	600		54	6
2-G	600	(1)[a]	27	3.0
3-A	800	800	70	0
3-B	800	800	60	0
3-C	800	800	50	0
3-D	800	800	40	0
3-E	800	800	30	0
3-F	800	800	20	0
3-G	800	800	17	0
4-A	800		54	2
4-B	800		54	4
4-C	800		54	6
4-D	800		54	8
4-G	800	(2)[a]	(∿40)	6.0
4-H	800	(2)[a]	(∿27)	4.0
4-I	800	(2)[a]	(∿14)	2.0
5-A	1000	1000	70	0
5-C	1000	1000	50	0
5-E	1000	1000	30	0
5-F	1000	1000	20	0
6-A	1000		54	2
6-B	1000		54	4
6-C	1000		54	6
6-D	1000		54	8
6-E	1000		54	10
6-F	1000		54	10.7
6-G	1000	(3)[a]	(∿40)	7.5
6-H	1000	(3)[a]	(∿27)	5.0
6-I	1000	(3)[a]	(∿14)	2.5

Repeat test 2-C and compare with original data; repeat test 4-D and compare with original data; repeat test 6-E and compare with original data.

Measure: Test section outlet temperature, °F; test piece temperatures, °F; test section inlet pressure, °F; test section outlet pressure, °F; test section flow, gpm.

[a]Special instructions: 1. Outlet temperature same as 2-C. 2. Outlet temperature same as 4-D. 3. Outlet temperature same as 6-E.

ORNL—DWG 71-786

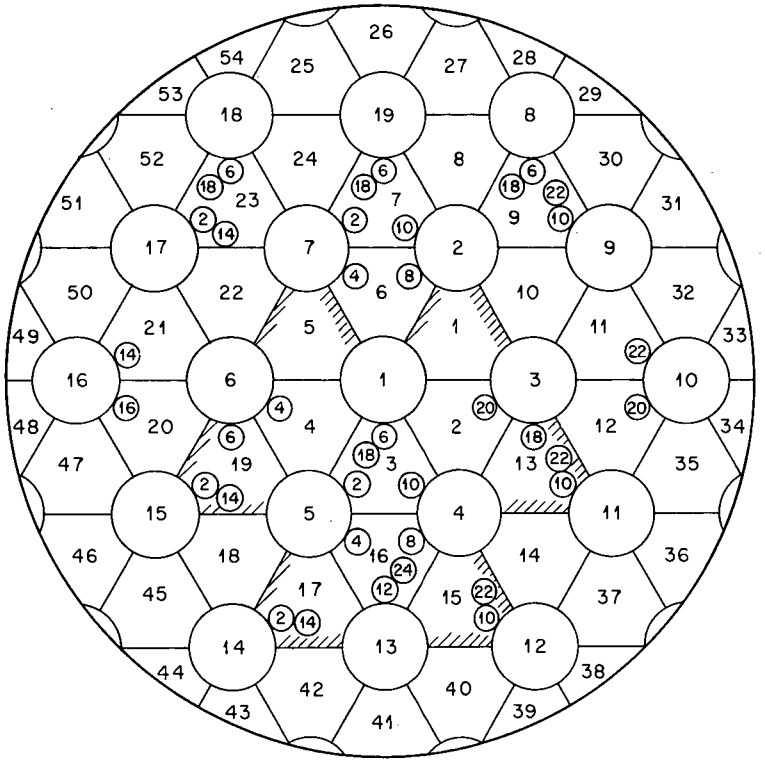

Fig. 1.9. Location of temperature measurement sensors for FFM bundle 1A. The axial position of a thermocouple, in inches, as measured from the upstream boundary of the heated section of the rod is inscribed in the small circle. The plan of the rod bundle is as oriented from the downstream end.

ORNL-DWG 70-495R

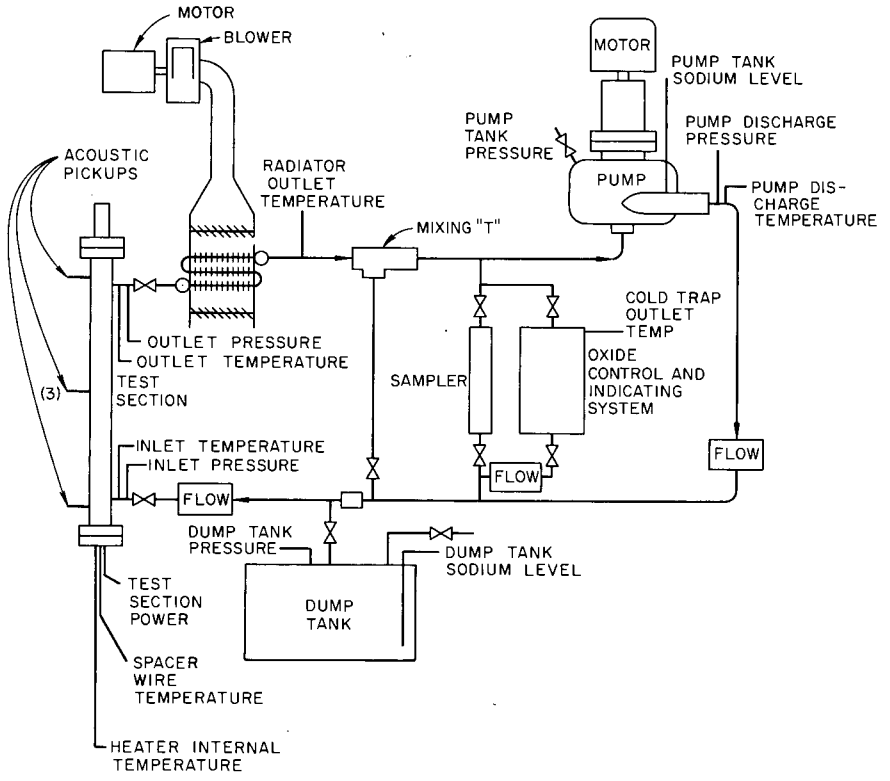

Fig. 1.10. Location of various test measurement sensors during operation of FFM bundle 1A.

Post-Test Observations of Bundle 1A

After draining the system sodium to the storage tank and prior to removing bundle 1A, the test section was isolated with block valves. The test section was then evacuated at approximately 1000°F to distill residual sodium to an external vessel. Figure 1.11 shows the bundle immediately after removal from the facility, and Fig. 1.12 shows the heater rods after the duct clamp had been removed. During the operation of this bundle six heater failures (10, 9, 6, 2, 11, 17) were experienced. The bundle has now been removed from the system, and an examination of x-rays taken of the failed heaters indicates that:

1. The sheath failure of heater 10 damaged the sheaths of heaters 9 and 2 and probably resulted in their eventual failure. The evident buildup and possible melting of the sheaths of heaters 1 and 3 may also have resulted from the failure of heater 10. These incidents occurred downstream of the bundle discharge and in a region of relatively static sodium. Figure 1.13 shows the area where heater 10 failed and shows some of its effects on heater 9.

2. Heaters 11 and 17 showed an open heating element at the closed (low-temperature, upstream) end of heater. Evidently brittle fracture occurred in the Nichrome V wire where the configuration of the element changes from a helical coil to a near straight run for attachment to the heater end plug. Several other heaters showed indications of cracks in the same area as noted for heaters 11 and 17.

3. Heater 6 has a damaged sheath, or foreign material has intruded into the boron nitride insulation at the point where the mica seal at the lead end of the heater had been swaged. Heater 10 also has the same symptoms.

Discoloration of the heaters and the bundle duct at its downstream end was noted. Figure 1.14 shows that all other parts of the bundle were clean. All thermocouples (ungrounded junction) were operative at the completion of the test program. Visual observation indicates that a slight bowing toward the bundle clamp may have occurred in some heaters. Measurements will be made to determine if this observation is correct.

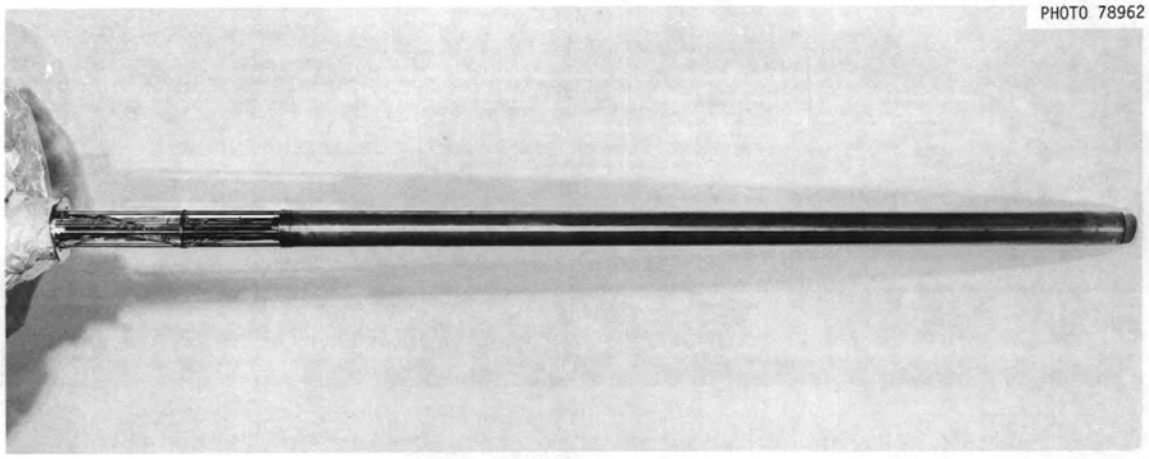

Fig. 1.11. FFM test bundle 1A immediately after removal from the FFM following completion of sodium operation.

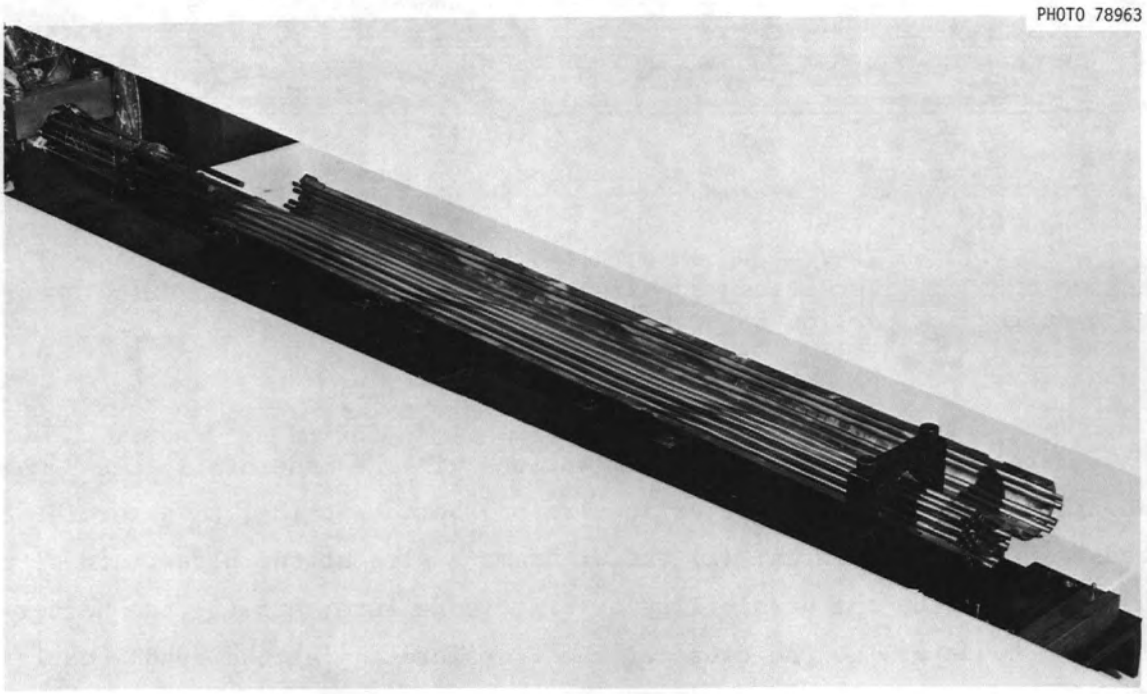

Fig. 1.12. FFM bundle 1A after removal from FFM. Duct clamp clamshell removed. Note dummy partial fuel pins in clamp.

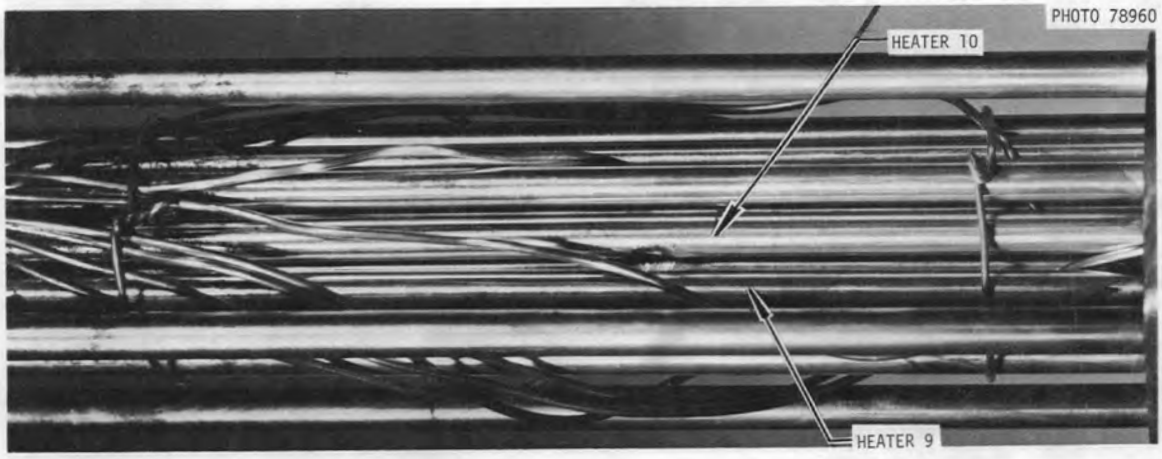

Fig. 1.13. Damage to sheaths of heaters 9 and 10, FFM bundle 1A.

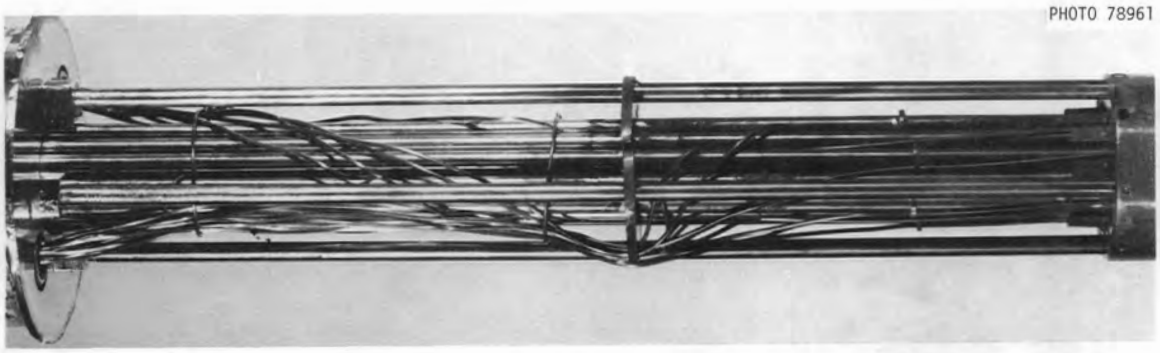

Fig. 1.14. View of upper portions of FFM bundle 1A downstream of heated section. Note clean appearance of static portion and discoloration of dynamic flow portion of bundle.

Metallographic and other examinations will be made of (1) the damaged sheath and the internal heater insulation, boron nitride, to determine the cause of sheath failures; (2) the Nichrome V wire at the closed end of the heater to pursue the possibility of brittle failure; and (3) the heater sheaths to determine the cause of the discoloration at the bundle outlet. Chemical analysis of system sodium samples will be performed to determine sodium purity now that the tests with bundle 1A are completed. Measurements will be taken of all rods to determine if they are bowed after test operation of bundle 1A. A complete checkout of heater protection systems

will be made, since at least one protective circuit was not effective in preventing sheath damage.

No failures or operating difficulties were experienced with the sodium or inert-gas valves during operation with this bundle. Our sodium operating experience with Mason-Neilan, Hoke, and Nupro valves and a throttling valve of ORNL design was discussed with the WADCO representative concerned with compiling a contemporary history of such experience.

Replacement of Damaged Bundle Heaters

Tooling was developed to remove damaged heaters from bundle 1 without damaging adjacent rods. The present estimate is that the welds which attach the 19 heaters to the tube sheet could be machined in one day and the heaters removed. Preparation of the tube sheet for rewelding would require about one day; rewelding of all 19 heaters, including replacement heaters, would probably require three days (including a trip to Los Angeles for electron beam welding). The greatest delay in heater replacement will be encountered in removing and reinstalling the heater and lead-wire extensions and their support brackets.

It may be possible to remove a single heater from a bundle configuration having wire wraps by spreading the bundle; however, displacement of adjacent wire-wrap couples or nicking and bending of adjacent rods would be highly probable. The problems and downtime associated with removal of extra rods to permit safe and convenient removal of a damaged rod should be bearable. We will explore this approach to heater replacement as well as the frozen sodium seal.

We will continue the development program to replace damaged heaters by welding. Our plans include making welds to a tube sheet from which the trepanned boss has been (1) partially and (2) completely machined away in the process of removing a damaged heater.

Preparation of Bundle 2A

Assembly of bundle 2A (previously bundle 2) is continuing. Figure 1.15 shows the bundle configuration when installed in the test section. Figures 1.16 and 1.17 show the temperature instrumentation for the new

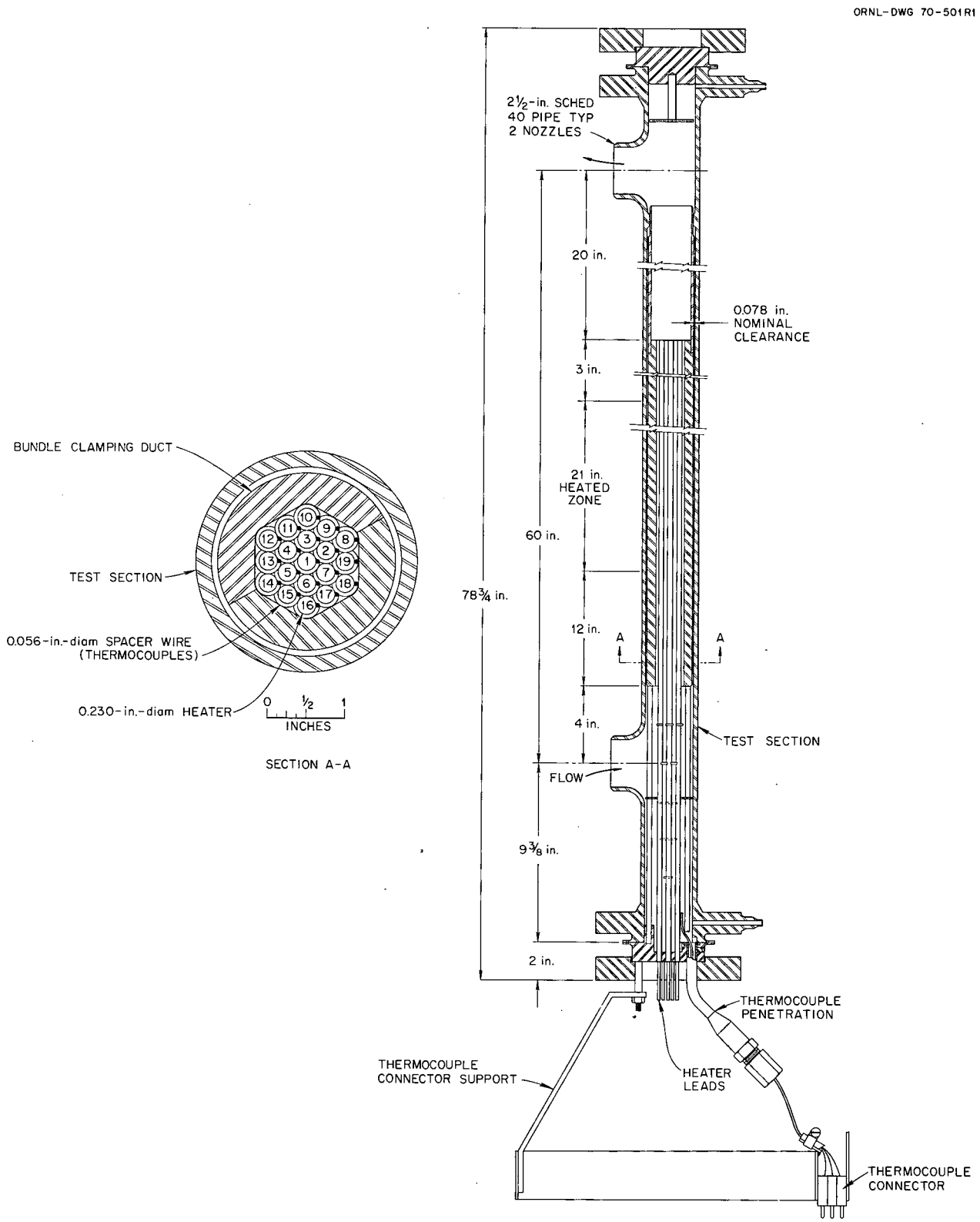

Fig. 1.15. FFM bundle 2A installed in test section. Inset shows bundle cross section and heater rod numbering scheme.

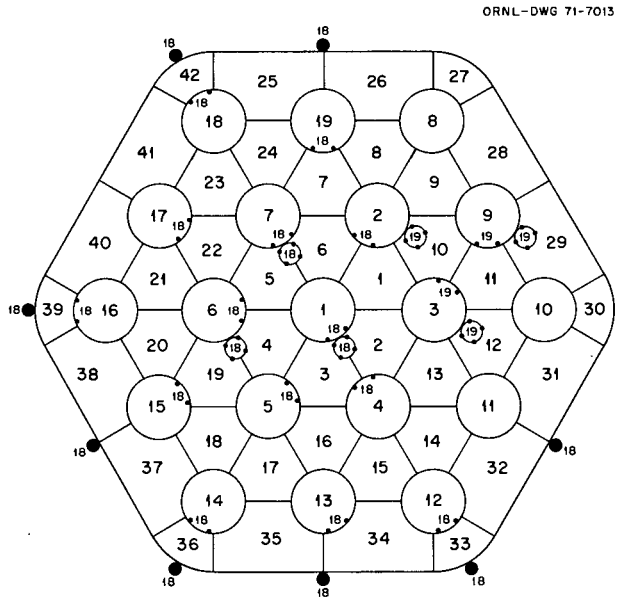

Fig. 1.16. Location of temperature measurement sensors for FFM bundle 2A. Locations of the grounded thermocouples and the duct thermocouples at the 18- and 19-in. levels. The dots indicate alternate Chromel and Alumel wires, with the junction being in between the wires. The small filled circles indicate locations of the duct thermocouples at the 18-in. level.

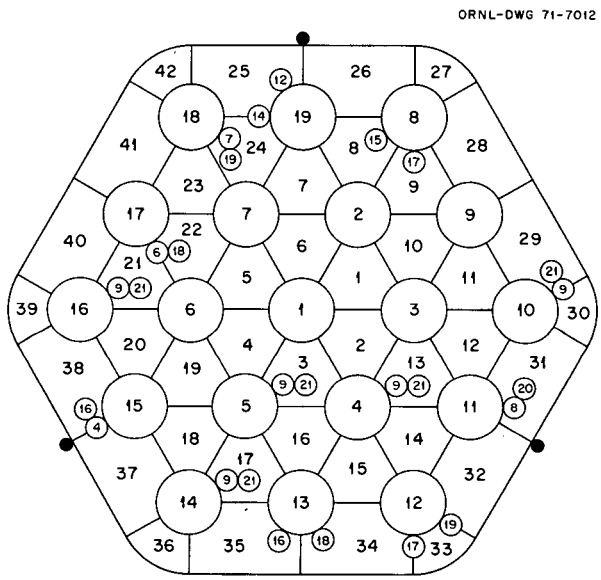

Fig. 1.17. Location of instrumentation temperature measurement sensors for FFM bundle 2A. Locations of ungrounded wire-wrap thermocouples. The small circles indicate the location of the wrap for each junction, and the numbers in the rod opposite them indicate the axial positions. The small filled circles indicate the locations of duct thermocouples at levels of 4, 8, 12, 16, 21, and 23 1/8 in.

bundle. In addition to the instrumentation shown in these figures, a thermocouple rake containing 12 thermocouples is being designed and will probably be installed in the test section in an attempt to measure individual channel outlet temperatures and to obtain a temperature profile immediately downstream of the bundle exit. A request for additional thermocouples or relocation of existing thermocouples in the hexagonal bundle duct has been received and is being evaluated. Relocation of couples by modifying the existing hex duct appears promising.

Installation of the wire-wrap thermocouples is progressing. The ungrounded junction thermocouples (Semco) have been straightened and rechecked for continuity. Trimming these thermocouples to length and installing end caps are in progress. No losses have been experienced.

Some difficulties were experienced with the grounded junction thermocouples supplied by both Sensor Dynamics and Semco. Breakage of thermal element wires in Sensor Dynamics thermocouples was experienced during the straightening operation. Approximately 30% of these assemblies had at least one open thermal element. Fifty percent breakage at the weld joints where the solid wire extension is attached to the thermocouple sheaths was experienced with the Semco grounded junction thermocouples during the straightening operation. Examination disclosed that unannealed wire had been used for the extension. The combination of the stiff unannealed wire and the reduced sheath wall section at the weld is the probable cause of the breakage. Annealing of all these couples and the extension wires has eliminated breakage during the straightening process. The couples are being dye-penetrant inspected and helium leak tested to determine their usability. Development work by ORNL Metals and Ceramics Division was initiated to determine if the broken attachments can be reinstalled locally. The weld joint design is being improved, and laser beam welding shows some promise. This development work will be coordinated into the LMFBR thermocouple development program. It will be necessary to use a combination of Semco and Sensor Dynamics thermocouples to provide temperature instrumentation for bundle 2A.

There was some indication during the operation of bundle 1A that the wire-wrap thermocouples had moved slightly from their assigned positions on the heaters during either assembly or operation. Attempts are

being made to develop a method of positive attachment of the wrap, at or near the thermocouple junction, to the heater sheath. A wire wrap was spot welded manually to a prototype heater that was operated subsequently in the heater test facility. In this particular attempt no special electrodes were used, no special effort was made to control the pressure between the two parts, and a 50 Wsec power setting was used. Although no failure to either the heater or thermocouple sheaths was experienced during the operation of the heater, the development of a more carefully controlled welding procedure was deemed advisable.

In response, the Metals and Ceramics Division modified an existing capacitor discharge welding machine (Unitek spot welder) to produce constant pressure and power and developed procedures that have yielded consistently satisfactory attachment welds between wire-wrap and heater sheaths. Specially shaped electrodes were devised to fit the two different sheaths. The photomicrograph, Fig. 1.18, indicates no detectable welding damage to either sheath.

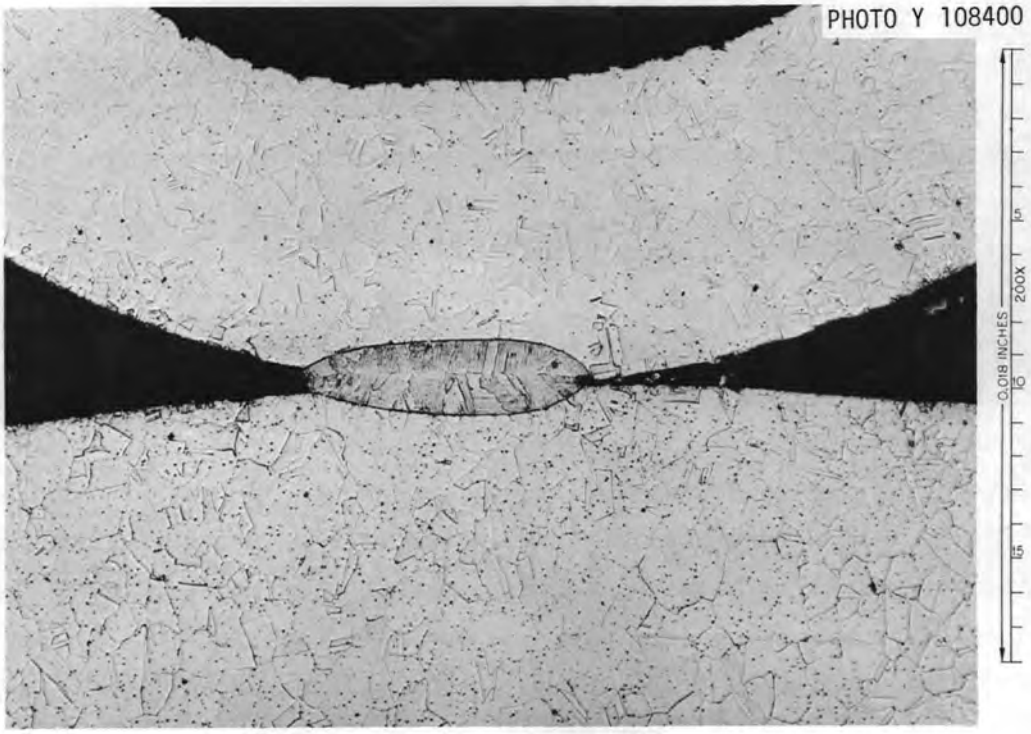

Fig. 1.18. Photomicrograph of capacitor discharge weld attaching wire-wrap sheath to heater sheath.

The heaters were annealed during the course of the bundle assembly. Inspection of the internal thermocouples in the heaters revealed that some thermal elements were lost and some additional secondary junctions were produced during the anneal. In 12 of the 19 heaters, at least one primary junction was viable. In the other 7 heaters, secondary junctions in the pertinent thermal elements occurred at locations from 5/16 to 3/4 in. from the primary junction (toward the terminal end of the heater). The output signal from these thermal elements will be somewhere between the signal generated by the primary junction and the secondary junction. Data from these thermal elements will be taken, but they will require careful interpretation. The thermocouple locations are being reevaluated, and initial indications are that most heaters will have at least one usable thermocouple junction.

Facility Preparation for Bundle 2A

The test section was removed from the FFM, and the weld joints were prepared for reinstallation. A new guard heater for the test section is being designed and will be installed. A new computer program is being developed to read out the additional temperature measurements required for this bundle. The requisite computer hardware and components for the temperature readout are being purchased. A numbering system for the additional thermocouples and the Dextir data points for their retrieval have been established. Data printout for blower speed and air flow will be available.

The design of the controls for facility operation is being simplified based on bundle 1A. A complete checkout of the heater protective circuitry (crowbar) has been initiated, since at least one heater sheath was not protected from damage during internal grounding of the heater element.

The vent line from the OCI unit, which is required during the sodium filling operation, is being routed directly to the pump bowl gas space instead of the present separate vent and overflow tank. The drive motor for the radiator induced draft air blower is being rewound for 1200-rpm operation to provide additional cooling air for the radiator. The Maintenance Manual for the facility and individual maintenance procedures are being prepared.

Preparation of Bundle 1B

A decision was made to reuse the tube sheet from bundle 1A to proof test the replacement of damaged heaters by rewelding to a previously used tube sheet.

Examination of the heater failures in bundle 1A tends to indicate that new heaters should be ordered to higher quality assurance standards for this test. Thirteen of the bundle 1A heaters are still operative and could be reused if necessary. The bundle 1A duct clamp (simulated infinite array) will be used for this bundle.

Location of the wire-wrap thermocouples will be identical to bundle 1A locations to take advantage of existing design drawings. It will be necessary to order new ungrounded junction thermocouples. The wire wrap will be attached to heaters using the capacitor discharge weld procedure developed for bundle 2A. Thus the possible shifting of thermocouple junctions during assembly and operation should be avoided.

The schedule for this bundle is not firmly established, since it will be necessary to order some new heaters. The tentative schedule for completion of the assembly is late in the calendar year.

Preparation of Bundle 3A

Pertinent information about heater dimensions and internal thermal elements contained in the procurement action for the 30 heaters (including spares) for bundle 3A (previously bundle 3) are listed below:

Diameter, in.	0.230
Heated length, in.	21
Total length, in.	Same as bundle 2A
End plug length, in.	To give overall length of bundle 1A
Thermal elements/heaters	
12 heaters	5
18 heaters	3
Thermal element separation	15° circumferentially 1/4 in. axially

Thermal element location

Six 5-element heaters	15, 15 1/4, 15 1/2, 15 3/4, 16
Six 5-element heaters	16, 16 1/4, 16 1/2, 16 3/4, 17
Six 3-element heaters	12
Six 3-element heaters	15
Six 3-element heaters	18

The five-element heaters will be used in the central seven rods of the bundle.

Establishment of the thermocouple locations for the bundle is in progress. A decision was made to use the duct clamp design for the simulated "infinite" bundle (bundle 1A), except that the dummy spacer rods are to be machined into the inside diameter of the duct clamp. The detail design for machining tube duct clamp is in progress.

An assembly fixture similar to the one used for bundle 1A will be required for this bundle and for bundle 1B. A design analysis is in progress to determine if the bundle 2 assembly fixture can be modified to accommodate the assembly of bundle 3A. Should this be impossible, two new assembly fixtures of bundle 1A design will be required, because assembly of the two bundles probably will overlap.

1.6 HEATER DEVELOPMENT

(AEC Activity 04 60 01 09 1)

D. L. Clark R. E. MacPherson

The intent is to develop a phase II heater which can withstand transients to the point of local boiling at a heat flux of 1,000,000 Btu hr^{-1} ft^{-2} and a phase III heater which can withstand sodium expulsion and reentry (for 1-sec durations) at 1,000,000 Btu.

Prototype Heaters for FFM Bundle 2A

The prototype heater for FFM bundle 2A failed at 14.5 kW/ft and 1200°F sodium outlet temperature 5.2 hr after the power was raised to

this level from 12.5 kW/ft at the same temperature. Failure was due to melting of the heater element approximately 2 in. from its end near the closed (higher-temperature) end of the heater. There were other areas where degradation of the inner diameter of the heater element was evident, including one severe instance approximately 8 in. from the connection to the heater lead-in conductor. This heater will be analyzed in detail to determine the cause of the previously reported decrease in the overall electrical resistance of the heater element during operation.

There was no sign of shorting to the heater sheath, and all thermal elements were operating properly after the failure. Apparently the melting occurred from the inside diameter of the Nichrome V wire. The prototype heater accumulated more than 500 hr of operation as follows: 185.1 hr at 9.0 kW/ft and 900°F sodium outlet temperature, 104.9 hr at 9.0 kW/ft and 1200°F sodium outlet temperature, 101.4 hr at 12.5 kW/ft and 900°F sodium outlet temperature, 117.6 hr at 12.5 kW/ft and 1200°F sodium outlet temperature, and 5.2 hr at 14.5 kW/ft and 1200°F sodium outlet temperature. A second bundle 2 heater has been installed in the heater test facility and will be operated at various conditions.

Watlow Manufacturing Company is fabricating three heaters using molybdenum elements as part of the work they are doing for European installations. This design is similar in length and operating characteristics to our bundle 2 heaters. Watlow will provide us with one or more of these heaters at no charge for test and evaluation.

Procurement of Heaters for Bundle 3A

Reactor Division Job Specification JS-174-246, entitled "Heaters for FFM Bundle No. 3," has been prepared and is being reviewed. Information obtained during meetings mentioned in the section entitled "Information Exchange" was of considerable value in the preparation of the specification. We intend also to use this specification to procure two prototype category II heaters, one having a helically slotted tubular heater element and the other a helically coiled ribbon heater element. We had intended to use RDT Specification P4-1 in this procurement action, but it

is not yet available for this purpose. When it is available, it will be used as a basis for the future procurement of pin heaters.

Inspection of tubing to be used as sheaths for the bundle 3A heaters is scheduled for the last part of this month. Part of this tubing will be provided to Watlow for the fabrication of heaters for ANL.

Intrinsic Thermal Elements

Both ORNL and Argonne have experienced considerable difficulty with secondary junctions on the intrinsic thermal elements installed inside cartridge heaters. In general, these secondary junctions have occurred within 1/2 in. of the primary junctions and have been observed both before and after annealing of the heater. A change made by Watlow, the heater manufacturer, in the installation technique for the thermal elements was not successful in three of the four replacement heaters they fabricated for ANL.

An alternate thermocouple installation design has been discussed in detail with Watlow and with the other users. This design utilizes individual 0.015-in.-OD sheathed thermal elements (TE) and retains the intrinsic thermocouple design. The sheath will be removed for approximately 1/4 in. at the hot junction end, and the bare element wire will be spot welded to the inside diameter of the heater sheath. Extreme care will be exercised to prevent shorting between the TE and the TE sheath. Watlow has previously demonstrated the ability to keep the TE sheath in close contact with the heater sheath in the heaters they fabricated for Idaho Nuclear for the LOFT program. This capability will be used in the design and fabrication of the FFM bundle 3 heaters.

Further investigations have been made of the possibility of utilizing a thermocouple consisting of one thermal element and the heater sheath. This work, performed in the Instrumentation and Controls Division indicates that the scheme will not function properly when the heater sheath is in contact with sodium. However, the two-element intrinsic thermocouple will function satisfactorily if the two junctions are in close proximity and are subjected to approximately the same temperature.

Boron Nitride Investigation

We are investigating boron nitride (BN) insulation as used in various heaters. We have obtained from Watlow samples of the Union Carbide produced BN in both the "as received" and Watlow's "reprocessed" conditions. Supposedly comparable grades of BN powder have been obtained from Carborundum Corporation and from the Oak Ridge Y-12 Plant. We are endeavoring to determine the purity level and the kinds and amounts of impurities in each grade. These data will be studied to discern the effects of probable manufacturing processes on BN purity. From this information, we can evaluate the feasibility of enhancing BN purity. The screen and particle size for various samples of BN are presented in Table 1.8.

We are also investigating the use of crushable BN tubes as the insulant in swaged heaters. This use would permit close examination of the insulation before installation; would eliminate the need for the powder tamping tool in the fabrication processes and thus the possibility of slivers and other wear particles from the electrical lead, the heater element, or the tamping tool becoming sources of subsequent shorting of element to sheath; and would provide closer control of concentricity between element and sheath. The British are doing similar work at the Risley Engineering Materials Laboratory (REML). Contacts are being established with REML and with a possible British commercial supplier of BN crushables, Science Workshop, Ltd. In addition, we are presently preparing at ORNL a limited number of BN crushable tubes of various densities using samples of the BN powder mentioned above.

Information Exchange

An information gathering and exchange visit was made by D. L. Clark to each of five users of cartridge heaters which simulate LMFBR fuel pins and to two prospective manufacturers. The manufacturers were Sensor Dynamics, Inc., Addison, Illinois, and Aero Research Instrument Company, Franklin Park, Illinois. The users visited included Argonne National Laboratory, Aerojet General Nuclear at the NTR site, WADCO at Hanford,

Table 1.8. Comparison of BN powder sizes

Source of powder	M&C No.	Weight % powder in various size fractions				
		(>147 μ) +100	(105–147 μ) –100 +200	(74 to 105 μ) –140 +200	(44 to 74 μ) –200 +325	(<44 μ) –325
Watlow TS-945	331	56.95	35.55	4.2	0.9	1.91
Y-12 Plant	332					100.0
Carborundum 8153	333	66.98	19.42	7.96	2.52	1.11
Watlow TS-945	334	Same values as for No. 331				
Watlow TS-945	335					

General Electric Company at San Jose, and Atomics International at Canoga Park. A report covering these meetings will be issued.

1.7 LMFBR CLADDING RESPONSE UNDER TRANSIENT HEATING CONDITIONS

(AEC Activity 04 60 01 09 1)

D. O. Hobson P. L. Rittenhouse

If substantial partial blockage at the inlet or within the heated zones of fast reactor fuel subassemblies should occur, temperatures approaching the melting point of the cladding could be achieved in short time periods. We are in the initial stages of an investigation of the deformation and rupture behavior of irradiated stainless steel undergoing such transients. The effects of irradiation damage on the microstructure and mechanical properties of stainless steel, as well as the effects of isothermally annealing such material, are being investigated at a number of installations. Our study differs from these in that it is concerned with the effect of rapid heating rates on the deformation and rupture characteristics of the material. We are initiating a program consisting of three parts: (1) rapid-transient annealing of irradiated material for subsequent electron microscopic examination and evaluation of damage recovery, (2) dead-loading tests of irradiated tensile or bend specimens under rapid heating conditions, and (3) burst testing of irradiated cladding tubes to define their irradiation temperature deformation and burst characteristics as a function of fluence, irradiation temperature, heating rate, and internal pressure.

The goal of this study is to quantitatively define the behavior of irradiated stainless steel cladding under transient temperature conditions. Very little is presently known; it is probable that the rapid heating rates involved (up to 500°F/sec) will produce results not directly predictable from isothermal data. It is important that such non-steady-state information be available.

We are awaiting delivery of the radiant heating furnace (promised delivery on August 16, 1971) for the transient annealing tests, and we are designing the dead-load devices for the hot cell. Specimens of type 304 stainless steel are presently available with a fast fluence of up to approximately 10^{23} neutrons/cm^2. Specimens of irradiated type 316 stainless steel, material more appropriate for this study, will be available later this fall.

2. LWR SAFETY STUDIES

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

P. L. Rittenhouse

The most serious accidents postulated in light-water reactors (LWRs) are those caused by rupture of any one of the pipes of the primary coolant system. Following rupture, system pressure and coolant are expelled, and the temperature of the Zircaloy fuel cladding rises as a result of redistribution of the heat contained in the fuel rods. Emergency core cooling systems (ECCS) are provided in LWRs to terminate the temperature transient associated with this highly unlikely but potentially serious loss-of-coolant accident (LOCA). However, since fuel rod failure could cause damage sufficient to adversely affect ECCS performance, experiments have been performed to (1) determine the modes of failure of Zircaloy-clad fuel rods as affected by conditions related to both normal operation history and the LOCA, (2) evaluate what effect these failures might have on emergency cooling, and (3) determine margins of safety relative to (1) and (2). All experimental work on this subject was terminated at ORNL on June 30, 1971. Section 2.2 presents results obtained too late in FY 71 to be included in the last report. The work outlined in Sects. 2.3 and 2.4 is new work specifically concerned with heat transfer characteristics of rod bundles during transients.

In addition to the new data reported in Sect. 2.2 reports are being prepared on several aspects of the Failure Modes Program, including the following:

1. single-rod transient burst tests, D. O. Hobson and P. L. Rittenhouse;
2. cladding embrittlement, D. O. Hobson and P. L. Rittenhouse;
3. multirod testing, P. L. Rittenhouse et al. (examination, data reduction, and analysis of last test in progress);
4. summary of all FRF work, P. L. Rittenhouse et al.;
5. irradiated single rod tests, M. F. Osborne et al.;
6. irradiated multirod tests (could be combined with preceding topical), M. F. Osborne et al.;
7. small-scale Blowdown Test Facility, C. G. Lawson;
8. experimental program in Blowdown Test Facility, C. G. Lawson;
9. increase in the pressure drop across rod bundles after deformation in simulated LOCA, W. R. Gambill.

The topical report on the second TREAT experiment has been completed.[1]

At the end of FY 71, preparations were under way for the second of two 21-rod bundle tests including nine irradiated rods. All components except the nine irradiated tubes were prepared. However, dissolution of the fuel from the nine irradiated tubes proceeded much more slowly, and required a higher temperature, than the dissolution of unirradiated fuel conducted previously. More than 90% of the irradiated fuel has been removed, and the dissolution, which was suspended in July, will be completed. Operation of the second test assembly appears doubtful unless further support becomes available. However, the components will be retained in readiness.

Reference

1. R. A. Lorenz and G. W. Parker, *Final Report on the Second Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT*, ORNL-4710.

2.2 HIGH-TEMPERATURE PROPERTIES AND METALLURGICAL EVALUATION OF ZIRCALOY CLADDING

(AEC Activity 04 60 01 10 1)

D. O. Hobson P. L. Rittenhouse

The type and severity of the failure of Zircaloy-clad fuel rods during an LOCA are dependent on reactor operating conditions at the time of the accident, the history of the core, and the magnitude of the accident. The activities described in this section were directed toward understanding and characterizing the behavior of the cladding and its effect on emergency cooling.

Multirod Testing

A series of multirod burst tests were run to evaluate the degree and extent of coolant channel blockage that might result from swelling and rupture of fuel rods during the LOCA temperature transient. Details of the final 32-rod test (test 10 at 1000 psig and 25°F/sec) were given in the last report.[1] Examination of the epoxy-cast bundle has shown that the maximum coolant channel blockage in any cross section was 68% (see Fig. 2.1); the maximum in any individual channel was 92%. Extrapolation of the blockage data using coolant channel coefficients[2] yielded a value of 77%. These figures compare with a maximum theoretical blockage (based on experimental data for cladding swelling in single rod tests) of about 90%.

Close examination of the bundle cross sections revealed that only half of the rods had ruptured during the experiment. The violence with which these ruptures occurred "encouraged" premature shutoff of power to the rods. The 16 rods that ruptured (see Fig. 2.2) deformed from 50 to more than 100% (circumferential expansion) prior to rupture, and nine others expanded 20 to 50% but did not fail. None of the other seven rods deformed as much as 20%; most were nearer 5%. This "incomplete" failure was probably the result of several factors, not the least of which was the early cutoff of power. However, it is felt that both damage to heaters caused by violent rupture of adjacent rods and pressure losses in some rods contributed to the result.

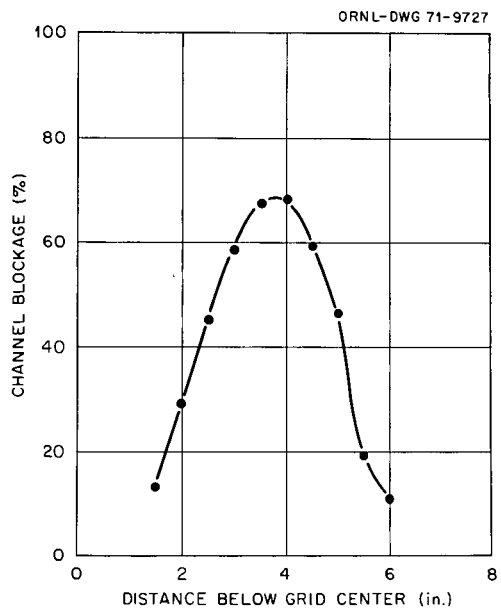

Fig. 2.1. Coolant channel blockage in test 10.

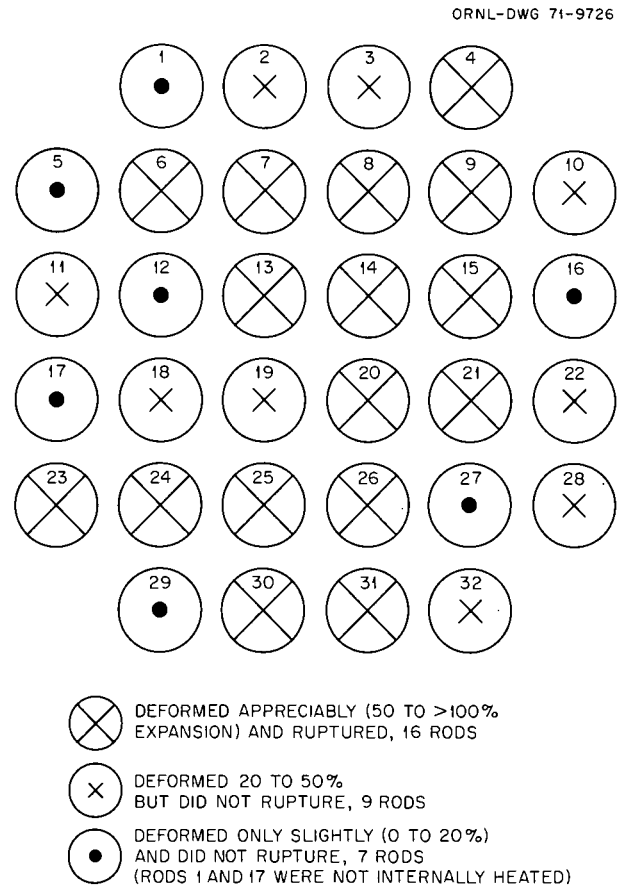

Fig. 2.2. Deformation and rupture in test bundle 10. This schematic of the 32-rod bundle describes the deformation behavior of the individual cladding tubes.

Rough calculations could be performed to show that, had all rods failed, total blockage would have occurred. Although this is certainly overly pessimistic, it is likely that coolant channel blockage would have more nearly approached the theoretical maximum had all rods been stressed to rupture.

A 16-rod bundle (4 × 4 array) of Zircaloy tubes was tested for Combustion Engineering (CE) at an internal pressure of 1200 psig and a heating rate of 25°F/sec. Examination of the tested bundle is currently being performed at CE. Twelve of the sixteen rods are known to have ruptured before the test was terminated by shorting of the power supply; the unruptured rods were those at the four corners of the assembly. Coolant channel blockage was at least at great as in test 10.

Cladding Embrittlement Due to an LOCA

This is expected to be the final bimonthly report on the embrittlement studies performed in connection with the Fuel Rod Failure Program at ORNL. An ORNL topical report on this subject is in preparation.

The previous report[3] in this series discussed the integration of the equation for the rate of oxygen penetration (migration of the brittle oxygen-stabilized α phase) with time in terms of temperature, thus providing an estimate of the embrittlement produced by a complete LOCA transient. The equation for brittle phase migration as a function of time was erroneously reported in the penultimate report[4] of this series. The correct equation is an order of magnitude less; so that, including a change to make it read as a first derivative, it should be

$$\frac{d\xi}{d\sqrt{t}} = 1.08 \times 10^{-6}T + 5.15 \times 10^{-6}e^{0.0168(T - 2200)} - 1.70 \times 10^{-2} , \qquad (1)$$

where ξ is in centimeters, t is in seconds, and T is in degrees Fahrenheit. This correction does not materially affect the results of the integrations reported previously.

When rupture occurs in a fuel tube the cladding thickness decreases in the vicinity of the rupture. This is illustrated in Fig. 2.3, which

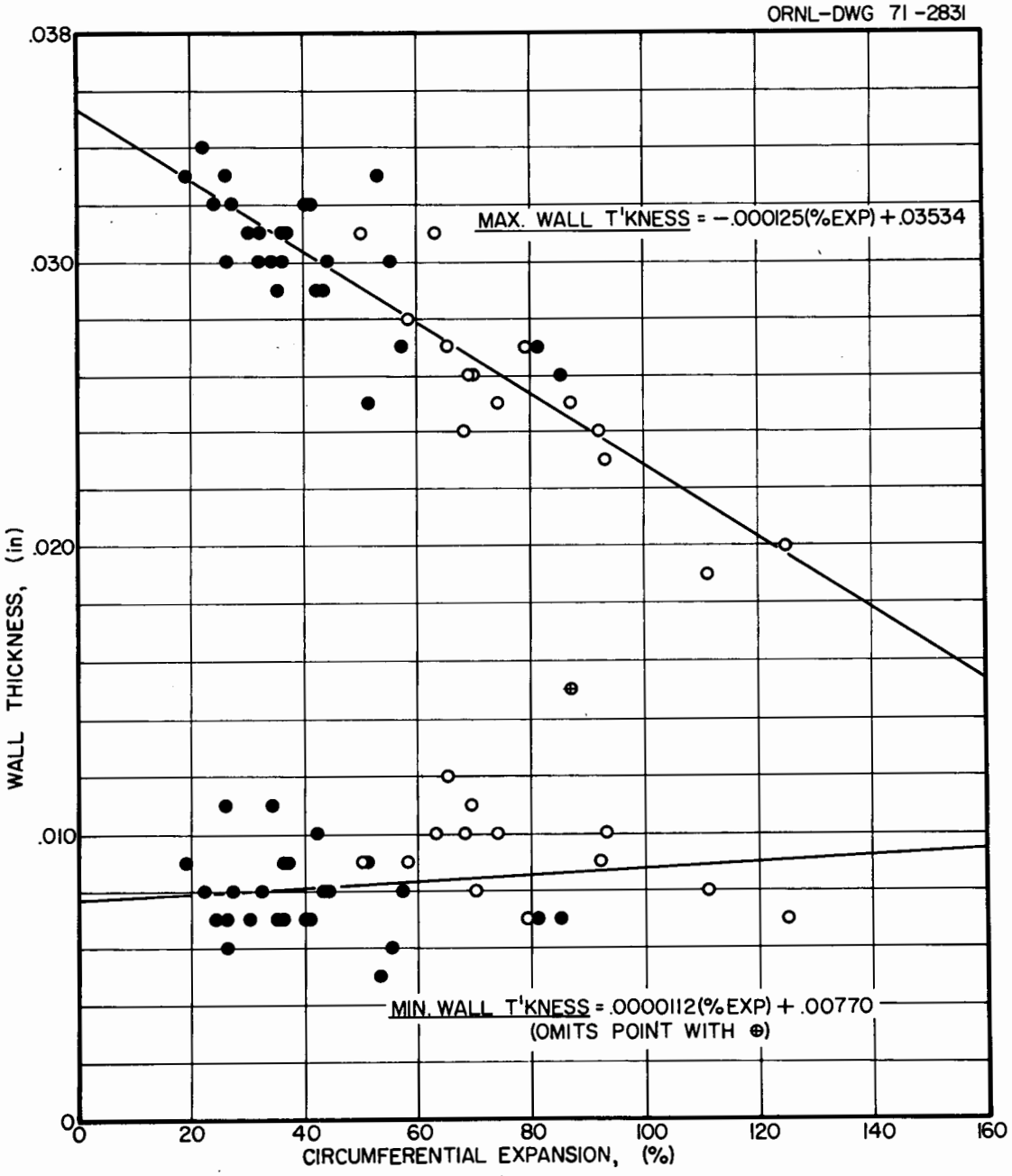

Fig. 2.3. Wall thickness as a function of circumferential expansion undergone by tubes during rupture.

is a plot of maximum and minimum wall thicknesses at the rupture as a function of the amount of circumferential expansion produced in the tube. The maximum thickness generally occurred 180° around the circumference from the rupture. The minimum thickness was measured approximately 0.015 in. away from the edge of the rupture lip. It was shown previously that tubing ductility is related to the ratio (F_w) of prior-β-phase thickness to total wall thickness. Obviously, if the wall thins the decrease must be absorbed by the prior-β-phase thickness for a given time and temperature exposure; that is the thinner the wall the less distance the brittle phases will need to penetrate to effect complete embrittlement of the wall.

In addition to the strictly geometrical considerations of embrittlement discussed above, there is an acceleration of the rate of embrittlement related to diffusion criteria. Early work[5] by BMI experimentally established diffusion coefficients for oxygen in α- and β-Zircaloy 2 and 4 at temperatures between 1000 and 1500°C. The diffusion coefficient of oxygen in α-Zircaloy (D_α) was determined by a moving boundary technique described by Jost.[6] The displacement ξ produced by the diffusion of a single species is given by

$$\xi = 2\gamma\sqrt{Dt} \ , \tag{2}$$

where

ξ = phase boundary displacement, cm,
γ = dimensionless parameter,
D = diffusion coefficient, cm^2/sec,
t = time, sec.

According to BMI, γ is defined by the equation (sic)

$$\frac{C_s - C_{II,I}}{C_{II,I} - C_0} = \sqrt{\pi\gamma}\, e^{\gamma^2} \operatorname{erf} \gamma \ , \tag{3}$$

where C_s, $C_{II,I}$, and C_0 are concentrations of the diffusing species at various phase boundaries. The value of γ calculated from this equation, using concentrations corresponding to the temperature being considered, allowed D to be obtained from a plot of experimental moving boundary data

(ξ vs $\sqrt{t}$). The slope of the straight line, from Eq. (2), was $2\gamma\sqrt{D_\alpha}$. Equation (3), as printed in Ref. 5, is in error according to Jost.[6] The γ parameter should not be under the square root sign.

Equation (3) is applicable for diffusion of a single species into a heterogeneous system of given overall concentration, and, by initially completely saturating the β phase at the experimental temperature, the Zircaloy-oxygen diffusion system could be made to behave as a heterogeneous system. The values of γ calculated from Eq. (3) represent diffusion conditions for which the moving boundary velocity is at a maximum for a given temperature. This equation, and the values of γ calculated from it, do not represent the real situation encountered in an LOCA. Jost[6] presents an equation for diffusion of a single species into a homogeneous system, as shown below:

$$C_{II,I} - C_{I,II} = \frac{(C_s - C_{II,I})e^{-\gamma^2}}{\sqrt{\pi}\,\gamma\,\mathrm{erf}\,\gamma} - \frac{(C_{I,II} - C_0)e^{-\gamma^2}}{\sqrt{\pi\phi}\,\gamma\,[1 - \mathrm{erf}\,(\gamma\sqrt{\phi})]}\,, \qquad (4)$$

where, in addition to those terms described for Eq. (3), ϕ is the ratio of the diffusion coefficients in the two phases, D_α/D_β, and $C_{I,II}$ is another concentration term.

Figures 2.4*a* and *b* illustrate the two diffusion situations. In situation 2*b* it is obvious that if C_0 is equal to $C_{I,II}$ the right-hand term of Eq. (4) becomes zero and Eq. (3) is applicable. Equation (4) is strictly applicable to a semiinfinite diffusion geometry, in which the base composition, C_0, is constant. In the thin-walled tubes under consideration, oxygen diffusing into the tube wall will raise C_0 to values anywhere between the original oxygen concentration of the material and composition $C_{I,II}$. Increases in C_0 effect increases in γ, which in turn increase the velocity of ξ and therefore the rate of embrittlement of the tube wall. A simplistic explanation for this effect is that the greater the difference between C_0 and $C_{I,II}$, the greater the driving force for oxygen atoms to cross the α-β interface. As C_0 approaches $C_{I,II}$, that driving force lessens and oxygen atoms diffusing through the α phase to the boundary are available for more rapid boundary movement. As stated earlier, when C_0 equals $C_{I,II}$ the boundary movement is at its maximum velocity for that particular temperature and microstructure.

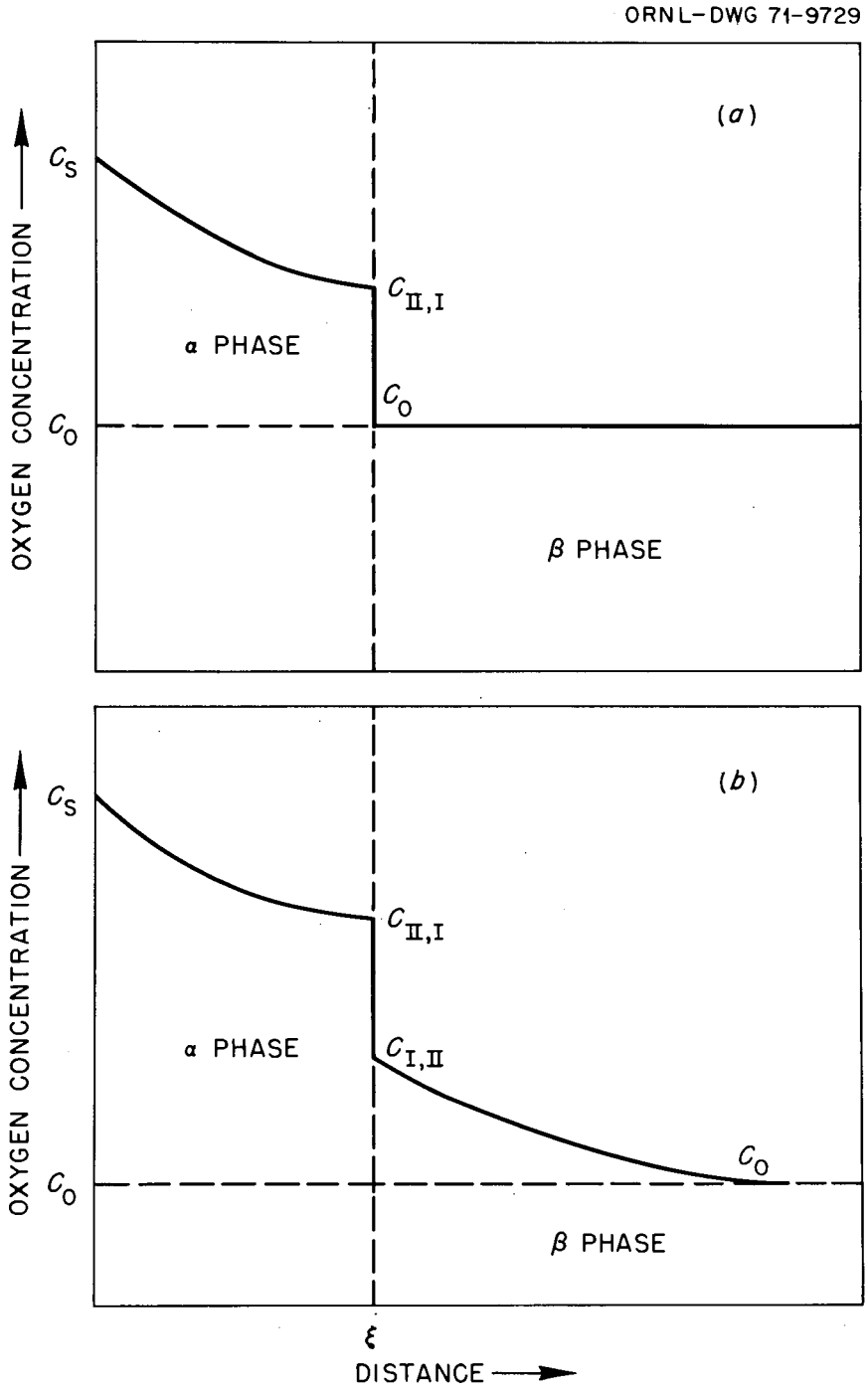

Fig. 2.4. Concentration vs distance diagram for diffusion of a single species into a heterogeneous system of given overall concentration (*a*) and for diffusion into a homogeneous system containing a concentration gradient (*b*). From Jost, Ref. 6.

A qualitative argument can now be presented which has direct bearing on the embrittlement of tubes with varying wall thicknesses such as would exist after tube rupture. A plot of ξ versus $\sqrt{t}$ is a straight line for isothermal exposure conditions and for C_0 constant; the slope of the line is $2\gamma\sqrt{D_\alpha}$. Consider a sealed tube being held in an isothermal steam environment at a temperature sufficiently high for surface oxidation and oxygen diffusion to occur. After an arbitrary time a diffusion gradient will have been established, extending into the wall from the outer surface. Consider now that the tube ruptures. Two new conditions for increasing the rate of embrittlement become operative: oxidation is occurring on both surfaces and the diffusion path has shortened due to wall thinning. Both conditions also increase the likelihood that C_0 will change, in which case the plot of ξ versus $\sqrt{t}$ should no longer be a straight line but should curve upward as shown in Fig. 2.5. The embrittlement increase

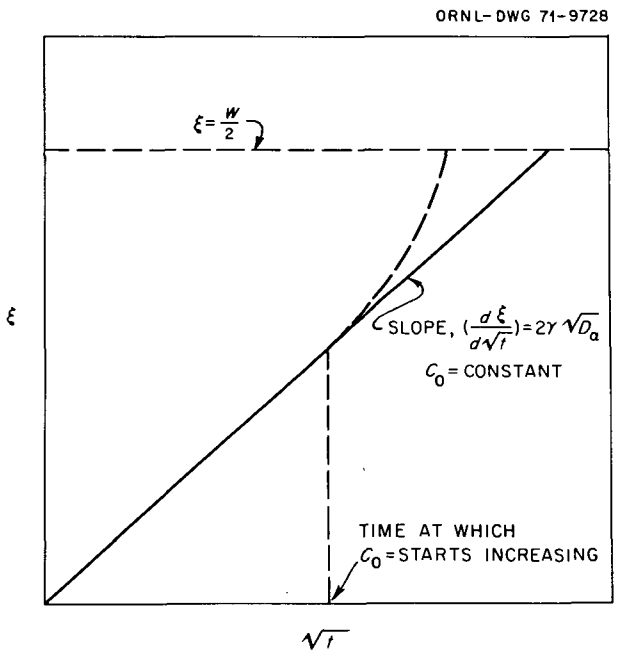

Fig. 2.5. Phase boundary movement ξ as a function of $\sqrt{t}$. The straight line represents conditions under which C_0, and therefore γ, remains constant. The curved dashed line represents the changes in $d\xi/d\sqrt{t}$ that could occur if C_0 increased with time. The top dashed line represents a condition in which oxygen diffusing from both surfaces of the tube has caused the phase boundaries to meet at the midwall of the tube.

caused by the increase in phase boundary velocity is in addition to the embrittling effects of wall thinning and two-surface oxidation, which affect F_w.

The effects of wall thinning on embrittlement and the effects of exposure of tubing to one- and two-surface oxidation during a thermal transient have not been studied experimentally. Such work needs to be performed if a realistic description of the effects of an LOCA on reactor cladding is to be realized.

References

1. D. O. Hobson et al., *ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1971*, ORNL-TM-3483, pp. 2–6.

2. P. L. Rittenhouse et al., *ORNL Nuclear Safety Research and Development Program Bimonthly Report for November–December 1970*, ORNL-TM-3263.

3. D. O. Hobson, W. R. Gambill, and P. L. Rittenhouse, *ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1971*, ORNL-TM-3483, pp. 6–9.

4. D. O. Hobson and P. L. Rittenhouse, *ORNL Nuclear Safety Research and Development Program Bimonthly Report for March–April 1971*, ORNL-TM-3411, pp. 2–9.

5. A. W. Lemmon, Jr., *Studies Relating to the Reaction Between Zirconium and Water at High Temperatures*, BMI-1154, pp. B-1–B-14 (Jan. 3, 1957).

6. W. Jost, *Diffusion*, pp. 71ff, Academic, New York, 1952.

2.3 HEATER ROD PERFORMANCE

(AEC Activity 04 06 01 10 1)

C. G. Lawson H. W. Hoffman

Out-of-site techniques for modeling reactor behavior commonly uses electrical heaters to simulate nuclear fuel rods. Heat is generated in the heater rod by passing an electric current through an internal resistor embedded in a dielectric (such as BN, MgO, or Al_2O_3) or through the tube

wall. Both types of heater have been successfully used to simulate fuel rods in experiments leading to steady-state (single-phase or two-phase) heat transfer correlations, and both are now proposed (or are already being used) for experiments studying the effects of transients in the primary variables (power, flow, pressure). Hence this program has been instituted to establish the extent to which the thermal response of these heaters duplicates that anticipated for the nuclear fuel rod undergoing similar depressurization transients. Specifically, data such as time to critical heat flux (CHF), variation of heat flux with rod-surface to coolant-temperature difference, and general post-CHF behavior are needed for thermal performance comparisons between the several electrical heaters and nuclear fuel elements.

Both directly heated (ohmic or skin heating by passage of an electric current through the tube wall) and internally heated (interior resistance wire electrically isolated from the tube wall by a ceramic insulator) rods in several variants will be tested. All experiments will be performed with single rods 3, 5 1/2, or 12 ft long (primarily) in the Forced-Convection Test Facility (depressurization loop) following modification to accept the longer rods. These heaters, of various clad materials with various internal ceramic bodies, will be procured from current users and suppliers or constructed at ORNL. Results with 3-ft-long rods should be of particular early significance, since they can be compared directly with data from Power Burst Facility nuclear experiments.

Development of instrumentation for measuring rod surface temperatures and heat fluxes and for establishing the channel flow regime and direction will be carried out as an accompanying aspect of this task. The results of related efforts of Idaho Nuclear Corporation under the LOFT program will be utilized to the maximum.

The data obtained in evaluating heater performance will also be used to test predictive codes available from outside sources or developed at ORNL.

2.4 LOSS-OF-COOLANT HEAT TRANSFER FROM ROD BUNDLES

(AEC Activity 04 60 01 10 1)

C. G. Lawson R. H. Chapman
H. W. Hoffman

The temperature of the fuel rods during the loss-of-coolant accident is fundamental to the integrity of the fuel rods and hence the consequences of the accident. The detailed objective of this program, currently being developed, is to conduct experiments to determine the cooling behavior of full-length heater assemblies (simulating reactor fuel bundles) undergoing LOCA-type pressure and flow transients. Throughout these experiments we will concentrate on acquiring data with full-length geometries of sufficient array to include radial flow effects. A concurrent analytic effort will permit explanation, extrapolation, and generalization of measured data and, equally important, prediction of similar situations in reactor systems.

The energy removed from a simulated PWR fuel bundle during a postulated LOCA, including such hydrodynamic aspects as variations in axial pressure distributions induced by acoustic and fluid inertia, will be determined. The approach to the measurement of energy removal during depressurization will depend on the coolant state within the flow channels. During the time in which liquid coolant is present (even to just a few weight percent), cooling is adequate; and the axial pressure difference across the core dominates the flow pattern. In the interval between the complete loss of primary liquid and the return of the core to coolant temperatures following emergency coolant injection, radial and axial flows are both important to the cooling process. Significant in establishing radial flow magnitudes are the temperature level of each fuel assembly and the temperature differences between fuel assemblies. Where axial pressure differences control, tests should be made with full-length heater rods in a 5 × 5 array.

Fuel assemblies will be tested in the large Thermal-Hydraulic Facility constructed for General Reactor Technology studies (189a No. 10123). This facility — a multipurpose high-pressure closed-cycle circulating loop with extensive control and data instrumentation and a potential power input

capability of 10 MW dc — will require some modification for blowdown studies. Currently, loop capability is restricted by the 2.5-MW high-pressure heat exchanger (5 MW dc is installed); the addition of a second high-pressure heat exchanger with associated controls and some slight power modifications would permit testing of 5 × 5 arrays. Regardless of the array dimensions, a low-pressure system, a blowdown assembly (rupture disks), and an emergency-coolant injection system will be required; these are conveniently accommodated below and/or adjacent to the existing facility.

In support of the experimental program, appropriate analyses will be conducted to (1) abstract generalizations from the measured data, (2) extend results over a greater range than actually covered by measurements (conversely, reduce the number of expensive measurements required to cover a given range of variables), (3) explain anomalous results and/or indicate areas where additional measurements are required, and (4) predict, from data with simulated fuel bundles, occurrences in similar situations for a reactor system.

3. HTGR SAFETY STUDIES

3.1 PROGRAM OBJECTIVES AND COORDINATION
(AEC Activity 04 60 10 01 1)

H. J. de Nordwall

The present HTGR safety program is concerned with the physical and chemical mechanisms that determine the transport of fission products in the coolant circuit, the integrity of coated particle fuel at abnormal temperatures, and the interaction of steam with a graphite core. Our approach has been and will continue to be that safety evaluations require a wide knowledge of the response of the plant to thermal or mechanical perturbations. An empirical approach is therefore unacceptable, since extrapolation of empirical data is illogical. We have also tried to avoid having our results become victims of the obsolescence produced by the continual design and material changes that are to be expected in any growing technology.

To ensure that the acid test of "How does the plant behave?" is never too far away, we have begun what we hope will become a joint fission product surveillance program in collaboration with Gulf General Atomic, Inc., and the Philadelphia Electric Company, who own the Peach Bottom HTGR. This program has the overall objective of monitoring fission product distribution in a number of first-of-a-kind HTGRs under normal operating conditions to establish the predictions made in various safety analysis reports. At present it is restricted to the Peach Bottom HTGR. During FY 72 agreement will need to be reached about the extent of surveillance in a more realistic prototype, namely a first-of-a-kind commercial HTGR,

if the program is not to suffer unnecessarily from difficulties associated with unpremeditated access to the primary coolant circuit.

In addition, we would emphasize that ORNL is the repository of a large body of information about gas-cooled reactors which is freely available to those contemplating HTGR or nuclear power. We encourage electric utilities and other interested parties to use this resource, because we believe that a rapidly expanding nuclear industry needs well-informed customers.

While it is recognized that fission products represent the principal peril from nuclear reactors, and the HTGR is no exception, we must point out that our program does not include any engineering evaluations of the mechanisms or frequencies of events that can cause abnormal fission product release. Because this type of analysis is lacking, we have been unable to specify the performance of engineered safeguards, though these areas have been recognized in the Preliminary HTGR Safety Program Plan and in the formulation of quality control procedures. We are now conducting a pilot program to analyze an important engineered safeguard utilizing ORNL's subscription to System Reliability Service, a commercial venture of the United Kingdom Atomic Energy Authority. Our objective is to determine the potential value of reliability analysis to safety systems evaluation.

For the present we will continue to report our studies under three main headings: (1) Reactions of Steam with Graphite, (2) Fuel Integrity and Fission Product Release, and (3) Fission Products in the Coolant Circuit, which includes surveillance studies.

3.2 REACTIONS OF STEAM WITH GRAPHITE

(AEC Activity 04 60 10 01 1)

C. M. Blood
H. J. de Nordwall*
S. H. Freid
A. P. Malinauskas*

The reactions of steam with various graphites are being studied to develop methods for calculating the effects of steam on the core of an

*Part time.

HTGR and the rates of hydrogen and carbon monoxide production. Laboratory investigations are centered on determining rate constants and understanding the reasons why the observed rate of reaction varies with the extent of reaction. The in-pile program is devoted to testing the validity of the synthesis of laboratory data under realistic conditions.

Laboratory Investigations of the Steam-Graphite Reaction

Reaction rate experiments using H-327 graphite have continued. Specimens were exposed to approximately 0.095 atm partial pressure steam at 800 and 900°C and to 0.045 and 0.070 atm at 1000°C. (Experiments involving 0.095 atm partial pressure steam at 1000°C have been completed and reported.)[1] A cursory examination of the surface area data obtained thus far indicates that the surface area development displayed by all the samples investigated is virtually identical when viewed in terms of graphite removal. Thus there is reason to suspect that a single mechanism persists over the 800 to 1000°C temperature range being investigated.

Experimentally determined parameters required as input data for Gulf General Atomic's OXIDE computational program were transmitted to D. Sedgley for use in the program, and we have received the results of the computation.[2] Experimental results and the values derived from the OXIDE program are compared in Fig. 3.1. These data are representative of the actual results obtained at (nominally) 1000°C and (nominally) 0.095 atm partial pressure steam. (Note that the data listed in Ref. 1 have been mathematically reduced to correspond to *exactly* 1000°C and 0.095 atm partial pressure steam.) It is quite obvious from an examination of Fig. 3.1 that, with regard to graphite removal (or hydrogen production), at 1000°C the OXIDE program appears to yield results that are conservative by about a factor of 10.

In-Pile Experiments

The fourth and final experiment in this series has been completed. In this experiment moderate (∿800°C) to low (∿600°C) graphite surface temperatures were maintained during oxidation, thus emphasizing fuel and

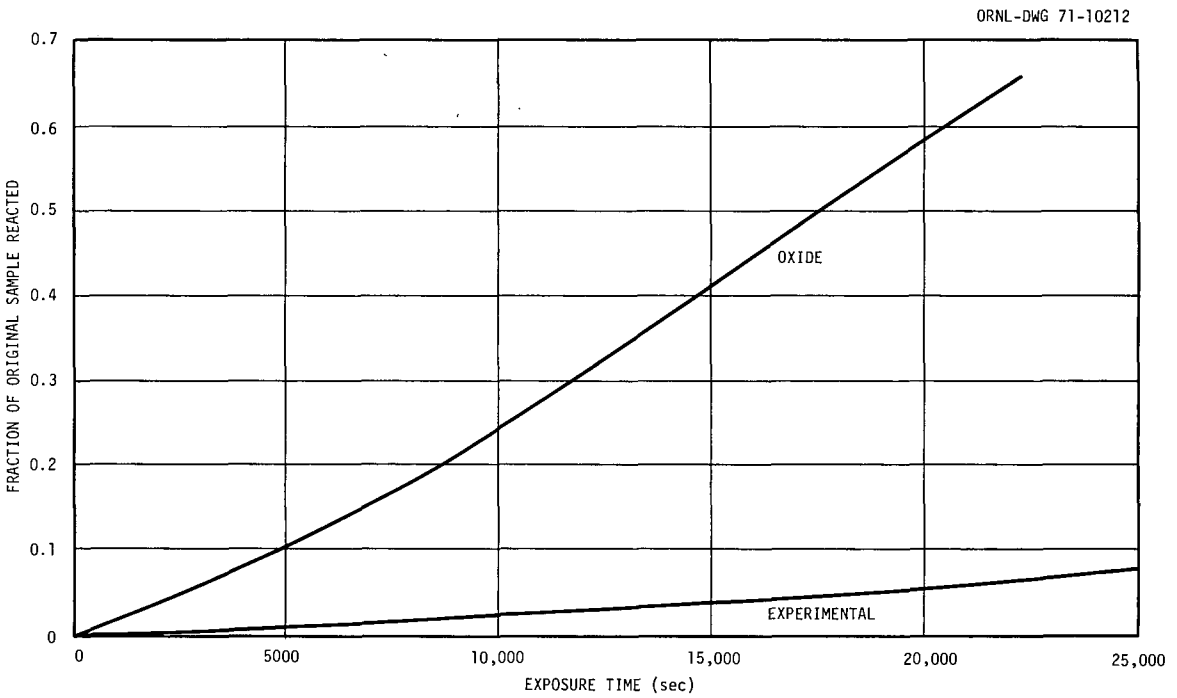

Fig. 3.1. Comparison of experimental reaction data with computational results from Gulf General Atomic's OXIDE program for the reaction of H-327 graphite with 0.095 atm partial pressure steam at 1000°C.

fuel matrix reactivity. The highest temperature in experiment SG-4, 800°C, overlapped the lowest temperature of experiment SG-3. We have therefore covered the steam oxidation rate of fuel elements covered with Great Lakes Carbon's type H-327 graphite over the surface temperature region 600 to 1000°C. We estimate that the center-line temperatures ranged from approximately 800 to 1300°C. (These estimates will be refined later.) The history of experiment SG-4 is shown in Table 3.1.

We estimate that approximately 8% of the total graphite was removed during exposure to steam. The carbon removed was divided as follows: approximately 1% of the H-327 casing, almost all the binder material, and about one-fifth of the pyrocarbon surrounding both the ThO_2 and UC_2 particles. Hot-cell examination of the fuel element revealed many bare ThO_2 cores. Although many shards of PyC coatings from the UC_2 particles were seen, very few UC_2 cores were visible. This is almost certainly due to the rapid hydrolysis of the exposed UC_2 to form UO_2. We are analyzing the fine debris (<80 mesh) for uranium to support this conclusion. Corrosion

Table 3.1. Irradiation history of steam-graphite oxidation experiment SG-4

Time interval (hr)	Surface temperature (°C)	Steam (mole fraction)
122.2	800	0.00
4.0	820	0.11
42.8	800	0.00
3.3	820	0.10
20.4	800	0.00
71.5	710	0.00
2.9	730	0.098
3.0	730	0.055
22.5	710	0.00
20.2	608	0.00
3.5	630	0.10
115.8	608	0.00

of the PyC on both UC_2 and ThO_2 coated particles seems to have been initiated by a pitting mechanism.

As in the previous experiments, we have noticed a very strong increase in the fission gas release when steam is admitted to the fuel, followed by a moderate decrease when the steam addition is halted. Figure 3.2 shows release-to-birth rate ratios (R/B) for ^{88}Kr and ^{89}Kr during experiment SG-4. The overall rate of release for ^{88}Kr during the experiment was about 470 times larger than at the inception of the experiment, but during the last two steam additions, the rates of rare-gas release were larger than the initial rates by factors of about 3300. Ratios for ^{88}Kr at the end of the experiment indicated that more than 10% of the fuel particles had broken.

Initial hot-cell examination[3] of experiment SG-3, whose surface temperatures had been higher, also showed essentially complete binder removal. However, in this case a much larger fraction (∿5%) of the casing had been removed. The ThO_2-coated particles were in much better shape than those in SG-4, with only slight pitting observed. Again, many PyC shards from UC_2 particle coatings were evident but no UC_2 cores.

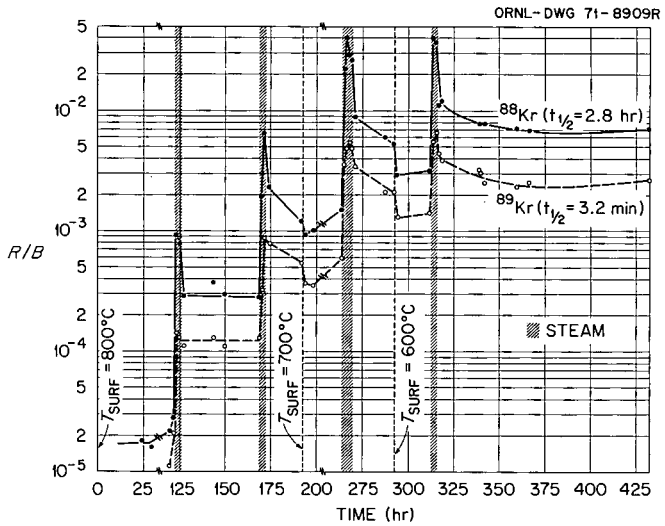

Fig. 3.2. ^{88}Kr, ^{89}Kr *R/B* ratios during in-pile steam-graphite oxidation experiment SG-4.

The experimental portion of this program is completed, and the data are being processed and evaluated. It should be noted that the effective exposures to steam during these experiments correspond to many times those expected in the most probable reactor accidents, because an operator will always cool the reactor core as rapidly as possible to a temperature at which reaction ceases.

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2. Wayne Bell, Gulf General Atomic, Inc., private communication to H. J. de Nordwall, June 15, 1971.

3. C. M. Blood, A. P. Malinauskas, and S. H. Freid, *Nuclear Safety Research and Development Program Bimonthly Report for May–June 1971*, ORNL-TM-3483, pp. 49–51.

3.3 FUEL INTEGRITY AND FISSION PRODUCT RELEASE

(AEC Activity 04 60 10 01 1)

M. T. Morgan R. L. Towns
C. D. Baumann

The purpose of this program is to determine the reasons for failure of irradiated coated particle fuels above normal operating temperatures and the rates at which fission products are released before and after failure. Knowledge of when a fuel element is likely to fail after a thermal perturbation is necessary for the design of engineered safeguards and for the proper analysis of the risks involved from fission product release.

Fission Product Release from Coated Fuel Particles in a Bonded Bed During Postirradiation Anneals

These experiments were undertaken to determine the cesium and strontium distribution in a bonded bed after high-temperature operation and to determine the effects of bonding on coated particle stability and fission product release. Early experiments[1] on bonded beds of TRISO-coated particles showed that 3% of the coated particles in a bonded bed completely failed during a 6-hr anneal at 1700°C and many more failed during a 6-hr anneal at 2000°C. The exact number of failures was not determined since the large amount of ^{85}Kr released caused the detection system to saturate, and a subsequent mishap released the gas to the stack before it could be collected for further analysis. The bonded beds used in these experiments had been irradiated to 20% FIMA in the ORR and contained TRISO-coated particles with sacrificial outer layers.

Further experiments with these bonded beds were postponed in order to study the stability of individual loose TRISO-coated particles from the same batch as those in the bonded beds and irradiated in the same capsule. In the course of these experiments it was found that most of the coated particles tested remained intact at temperatures of up to 2000°C but the SiC coatings of some particles cracked, releasing cesium but not krypton. Out of 25 loose coated particles annealed at 2000°C or higher for periods of from 20 min to 11 hr, only one coating failed completely.[2] This is a better performance than that of the coated particles in the

bonded bed. We need to verify the difference by repeating the 2000°C anneal of a bonded bed to determine the exact number of coating failures and then, if the difference still exists, study the reasons for it. This investigation would include x-ray and metallographic examinations of the coated particles and analysis of the matrix to determine if metallic contamination were present.

At present, the high-temperature furnace is occupied with experiments to determine the diffusion coefficients for cesium and strontium in pyrocarbon coatings (reviewed below). We expect to resume high-temperature experiments on bonded beds of TRISO-coated particles in about six weeks.

The 1250°C anneal of the B-19 bonded bed containing LTI*-BISO-coated UO_2 particles irradiated to 10% FIMA in the B9-42 capsule has now reached 430 hr. The furnace components are being analyzed.

Diffusion of Cesium in PyC Coatings

The postirradiation annealing experiment comparing LTI-PyC-coated particles with HTI*-PyC-coated particles was continued. Both samples, irradiated to a fast fluence of 4×10^{21} neutrons/cm^2 in experiment HT-1, have now accumulated 950 hr at 1400°C. Cesium has not yet penetrated the HTI coatings. Diffusion through the LTI coatings occurred in 130 hr, as reported previously.[3]

The LTI-PyC-coated UO_2 particles, irradiated to 10 and 20% FIMA in experiments B9-42 and B9-41, respectively, have been annealed 35 hr at 1400°C. The cesium release at 31 hr averaged 0.45% for the coated particles irradiated to 20% FIMA and 0.05% for the coated particles irradiated to 10%, but the data are widely scattered. The anneals will continue until the cesium release is sufficient to plot graphs of the fractional release versus annealing time.

LTI pyrocarbon does not appear to be a useful container for cesium in HTGR fuels.

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*LTI = low temperature isotropic, HTI = high temperature isotropic.

References

1. M. T. Morgan, R. L. Towns, C. D. Baumann, *Gas-Cooled Reactor Semiannu. Progr. Rep. Mar. 31, 1969*, ORNL-4424.

2. M. T. Morgan et al., *Nuclear Safety Program Annual Progress Report for Period Ending Dec. 31, 1970*, ORNL-4647, p. 107.

3. C. M. Blood and A. P. Malinauskas, *ORNL Nuclear Safety Research and Development Program Bimonthly Report for March–April 1971*, ORNL-TM-3411, p. 45.

3.4 FISSION PRODUCTS IN THE COOLANT CIRCUIT

(AEC Activity 04 60 10 01 1)

F. F. Dyer H. J. de Nordwall*
J. O. Kolb E. Hoinkis†
M. F. Osborne

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.

Iodine Adsorption and Desorption

During this reporting period, desorption experiments from Fe_3O_4 were performed with iodine deposited at 250°C, the temperature of the coolest part of an HTGR steam generator.

During the adsorption stage of one experiment, the surface of the Fe_3O_4 sample was loaded unintentionally rapidly with 0.43 monolayer of iodine. The source was immediately cooled to −196°C and isothermal desorption at 250°C observed for 40 min, during which time 10% of the iodine desorbed at an average rate of 0.25%/min. To remove more iodine the system was then heated to 400°C. During this period of changing temperature,

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*Part time.

†Visiting scientist from Germany.

65% of the *original* iodine desorbed. The desorption rate at 400°C then declined slowly. Figure 3.3 illustrates the course of this experiment. We must emphasize that this result should not be considered final; it is merely indicative of the sensitivity of the desorption rate to temperature at this surface loading. It is reported because it is of topical interest. We would expect iodine that had been adsorbed for a very long time to desorb more slowly, rather than more rapidly. At lower surface concentrations desorption would not have begun until approximately 350°C.

Iodine desorption spectra were also measured for Fe_3O_4 covered with 0.29 monolayer of iodine at 250°C. The results are shown in Figs. 3.4 and 3.5. The reproducibility of the differentiated desorption spectra in Fig. 3.5*a*, in which specimen heating rates were kept the same, should be noted. The maximum desorption rate occurred at a higher temperature

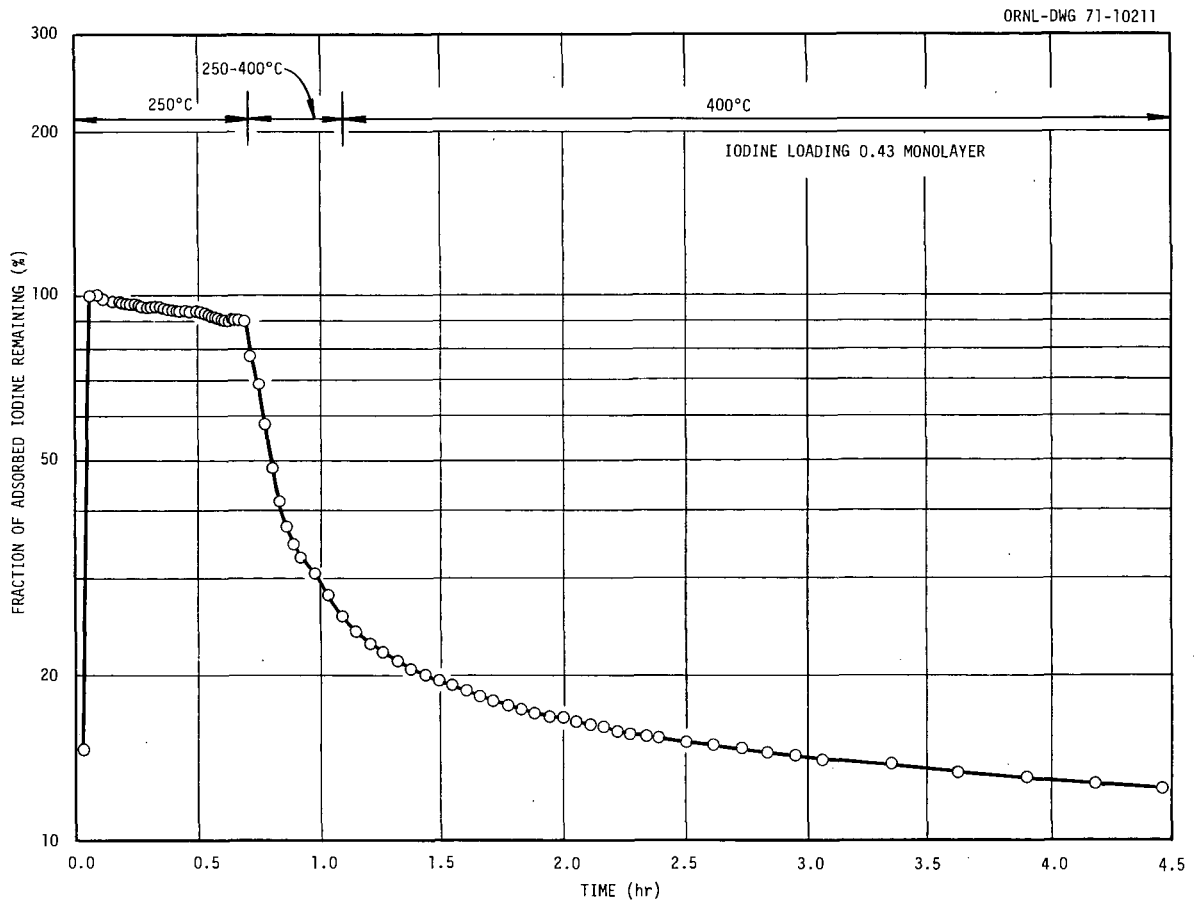


Fig. 3.3. Isothermal adsorption and desorption of iodine from Fe_3O_4.

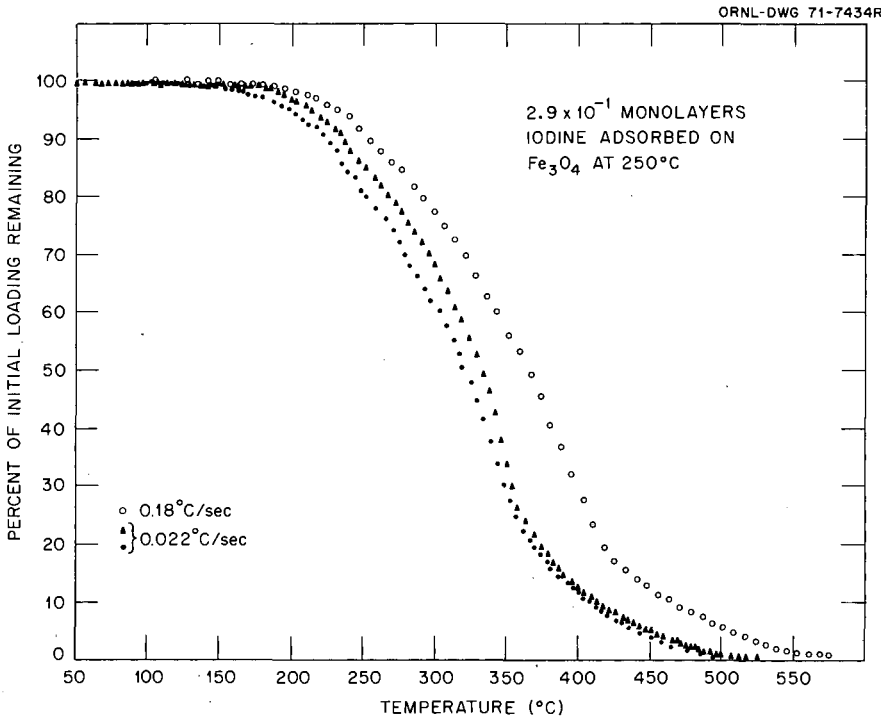


Fig. 3.4. Desorption spectra for iodine desorbing from Fe_3O_4 at 250°C at various heating rates into a vacuum.

when the specimen was heated more rapidly (cf. Fig. 3.5*b*). The spectral shift will be correlated with the preexponential component of the rate equation in subsequent analyses.

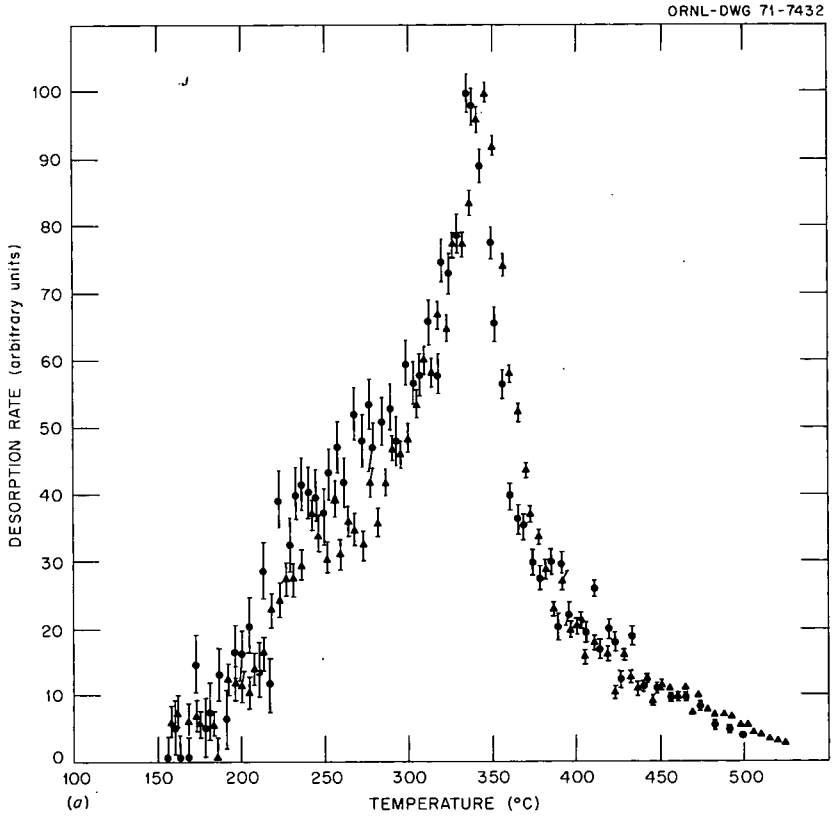
Desorption experiments from surfaces with low iodine coverages are planned for the next period.

Fission Product Surveillance at the Peach Bottom HTGR

Data obtained during visits to Peach Bottom are being evaluated.

Rare gases

Figure 3.6 shows a comparison between R_i/B_i, the steady-state fractional release into the purge of isotope i, and R_i'/B_i, the steady-state fractional release into the primary coolant, using our data and those obtained by Philadelphia Electric Company. The most striking point is the efficiency of the purge system; R_i/R_i' is approximately 5000.



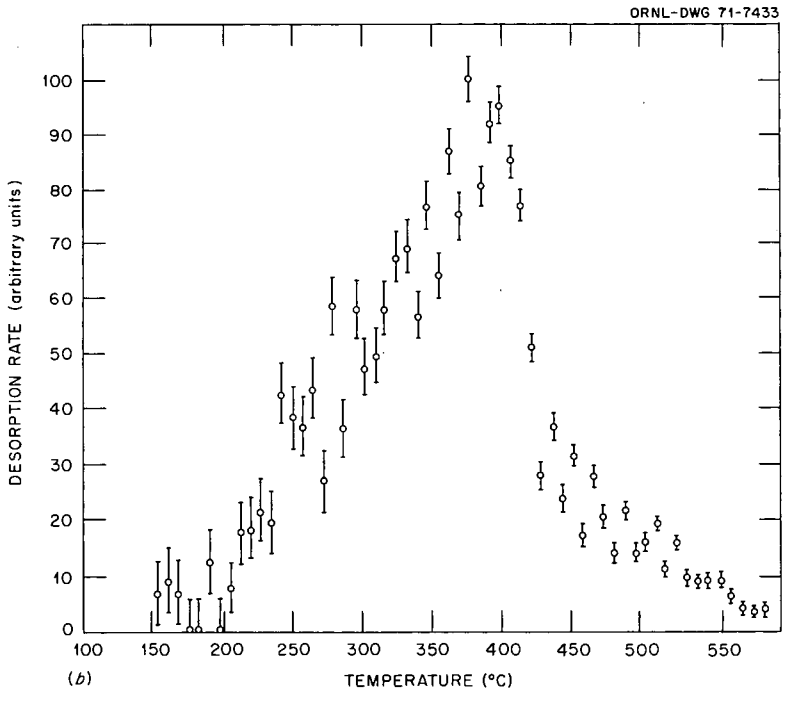


Fig. 3.5. Differential desorption spectra for iodine desorbing from Fe_3O_4 into vacuo. 0.29 monolayer of iodine initially deposited at 250°C. (*a*) Heating rate 0.022°C/sec; (*b*) heating rate 0.18°C/sec.

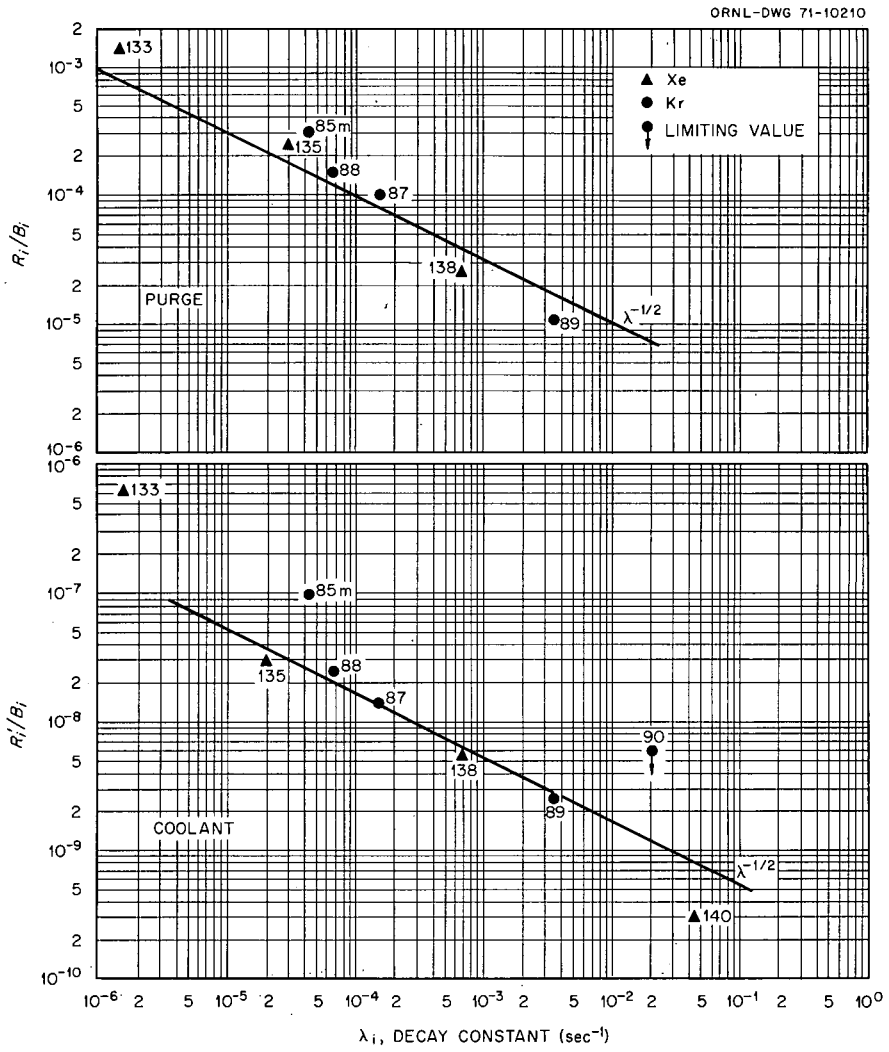


Fig. 3.6. Rare-gas releases into purge and primary circuit of peach bottom HTGR, Dec. 4, 1970.

Other interesting features are the similarity of release of xenon and krypton, which we cannot explain convincingly as yet in relation to the observed behavior of other systems, and the persistence of the $\lambda^{-1/2}$ dependence over such a wide range of λ. This latter point would seem to indicate that the purge depends more on dilution of the krypton and xenon in the space between the fuel and the sleeve than on diffusional delay in the graphite sleeve. Our ^{140}Xe measurement would appear to fall significantly below this line, implying that a delay of the order of 20 sec was being experienced by xenon atoms traversing the sleeve. An effective

average diffusion coefficient for comparison with that quoted by J. Norman and D. D. Busch[1] will be calculated when the data have been checked further. The primary circuit activity that one would expect if the Peach Bottom HTGR were to operate without a purge, with helium purification being provided only by the steam generator purge of 175 lb/hr, may be calculated from the following relationships.

With no purge R_i', the number of atoms of i entering the primary coolant per second, will be equal to R_i, the number leaving the fuel. Now, from mass balance considerations we may write, under steady-state conditions,

$$R_i' = R_i = \lambda_i N_i + f N_i \ ,$$

where f is the probability of an atom entering a 100% efficient purification plant per second and N_i is the number of atoms of i in the primary coolant. Thus

$$f = \frac{\text{purification flow rate}}{\text{coolant inventory}}$$

$$= \frac{175}{910 \times 3600} \text{ sec}^{-1} = 0.000053 \text{ sec}^{-1} \ .$$

(For comparison, f for Fort St. Vrain is 0.000041 sec^{-1}.)[2] Whence

$$\lambda_i N_i = \frac{\lambda_i R_i}{3.7 \times 10^{10} \ (\lambda_i + 0.000053)} \text{ Ci} \ .$$

Since what one measures is $\lambda_i N_i / V$, where V is the volume of the reactor, it is evident that measurement of coolant activity will be much easier (requiring 1/5000 of the counting time) during operation without a purge. An analysis of the practical value of observations on short-lived gas release in a failed-fuel location is being considered.

One should note that these comments apply only to unadsorbed gases. Without a purge, R' for iodine would increase by 5000 times if iodine behaved exactly like xenon in passing through the fuel sleeve. The direct release of metallic fission products to the coolant, with the possible exception of cesium, would not be expected to be changed by purge omission.

Iodine

The average partial pressure of iodine (^{131}I) in the helium in the core exit duct was found to be 4×10^{-18} atm over the last 30 days of the first cycle of Core II operation. Since the fraction plated out per pass is not known, one cannot deduce a release rate without assuming iodine circulates less than once around the circuit.

This preliminary estimate was based on an ^{131}I determination on diffusion probe 4DP-2, which was built by Gulf General Atomic, Inc., and operated by Philadelphia Electric Company. This value is about 1/20 that found by Busch[3] during the first 168 power days of Core I (8×10^{-16}atm)[4] and considerably lower than the partial pressure measured at the end of Core I. No ^{131}I was assumed to have passed through the diffusion tube.

Metallic fission products and impurities

Cesium-137 has been measured in the helium leaving the core during the first cycle of Core II. The average partial pressure was determined by integrating the area under the plot of ^{137}Cs activity per unit length versus position shown in Fig. 3.7 for diffusion tube 4DP-2; $\overline{P}_{Cs}$ was found to be 6×10^{-15} atm. Again this value was lower than that found at the beginning of Core I (1×10^{-14})[3] and considerably lower than that found at the end of Core II (1×10^{-13}).[4]

Contributions to ^{137}Cs activity from ^{137}Xe could be excluded. Some contribution from dust-borne cesium would seem to have been present because diffusion tube No. 0, whose flow was smaller, showed a flatter, more disorderly, concentration profile. Silver also exhibited a profile steep enough to be ascribed to a molecular species. The ratio of the slopes of the lines for silver and cesium are approximately consistent with what one would expect from a comparison of the atomic radii of the two atoms.

The most important observation is that the cesium partial pressure can fall when a clean core replaces an emitting core. Until the source of cesium at the end of Core I has been defined, speculation on the practical significance of this observation is profitless.

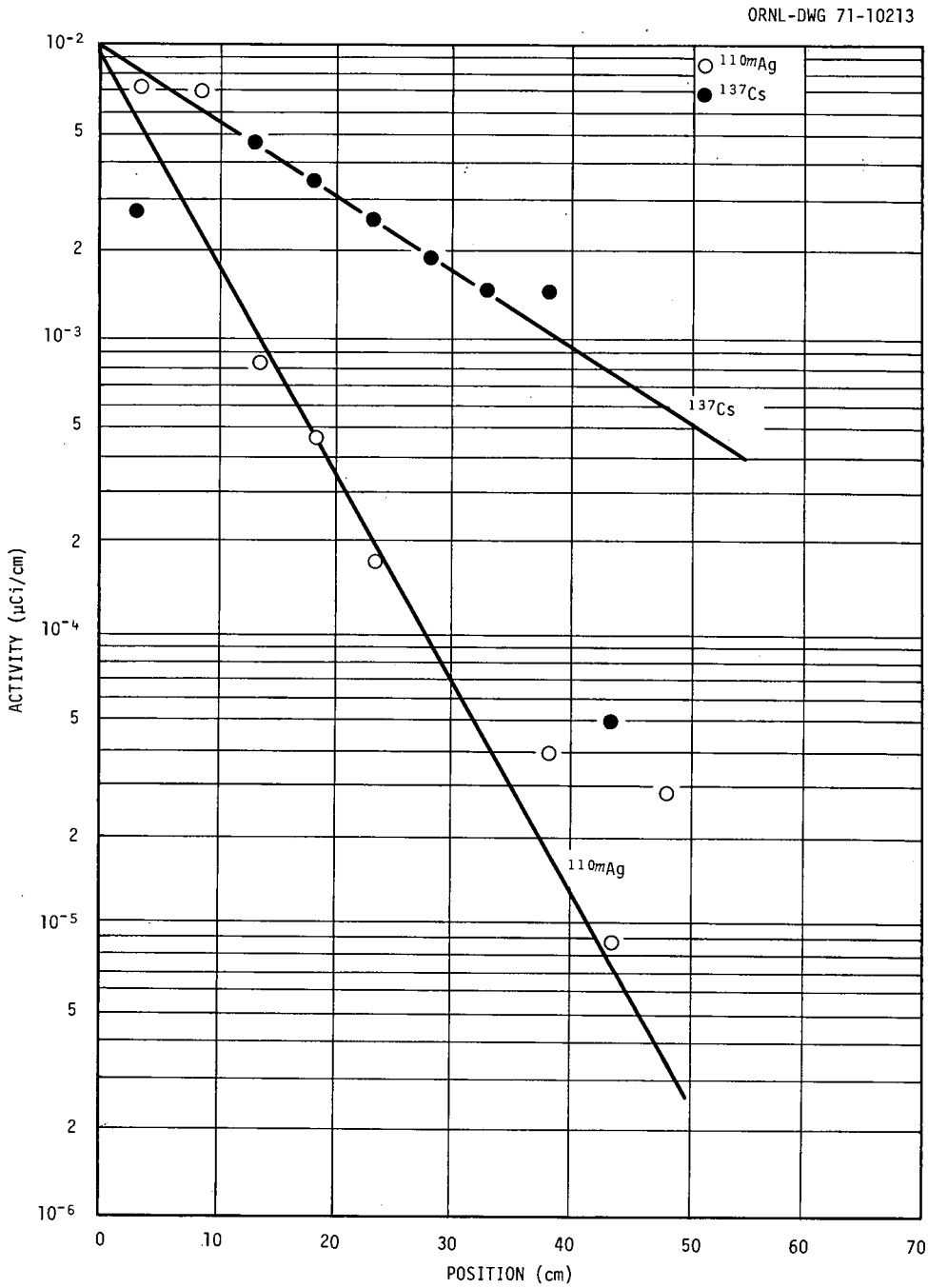


Fig. 3.7. Distributions of ^{110}Ag and ^{137}Cs in diffusion tube 4DP-2.

The concentrations of strontium have yet to be determined. Other metallic fission products were present only in very small amounts. The fuel debris that was found was insufficient to explain away the cesium found. The concentrations of ^{59}Fe, ^{51}Cr, ^{54}Mn, and ^{60}Co observed corresponded to <1000 atoms/cm^3 at 1000°K and 24 atm.

Work in progress includes examination of dust from Cores I and II and the determination of counting efficiencies for the primary gas duct experiments previously reported.

References

1. *40-MW Prototype HTGR Post-Construction R&D Development Program Progress Report*, GA-10400, p. 40 (1970).

2. *Final Safety Analysis Report*, Fort St. Vrain HTGR—Public Service Company of Colorado.

3. *40-MW Prototype HTGR Post-Construction R&D Development Program Progress Report*, GA-9797, p. 49 (1969).

4. D. D. Busch, Gulf General Atomic, private communication to H. J. de Nordwall, Aug. 31, 1971.

4. HEAVY-SECTION STEEL TECHNOLOGY PROGRAM

4.1 HSST PROGRAM ADMINISTRATION AND PROCUREMENT (TASK A)
(AEC Activity 04 60 80 03 1)

F. J. Witt

The Heavy Section Steel Technology (HSST) Program is concerned with the primary vessels for water reactors, concentrating mainly on the effects of flaws, material inhomogeneities, and discontinuities on the behavior of the vessels under normal operating and accident conditions. Emphasis is placed on developing a technology that can be used to establish the margin of safety against failure of the vessels, particularly catastrophic failure beyond the control of plant safety features.

Fracture phenomena under the environmental conditions of interest and the effects of section thickness on the fracture behavior of the large reactor vessels are being examined through a systematic progression of specimen sizes up to the equivalent of a full section thickness of 12 in. The program will culminate in a series of simulated service tests in which deliberately flawed vessels of substantial thickness will be tested to fracture.

There are currently eight technical research subcontracts in force. Of these, the research work in three is completed and final reports are being or have been written. Extensions of two of the remaining five subcontracts are currently being negotiated, and two new subcontracts are anticipated by midyear. Two programmatic manuscripts were issued,[1,2] one technical report received a final review, and a progress report was prepared for publication.

The major procurement effort remains the fabrication of six 6-in.-thick, 39-in.-OD pressure vessels. One such vessel being prepared for x-ray examination is shown in Fig. 4.1. Some nonhomogeneity or flaw indications were found in the head-to-cylinder welds. Repairs have delayed delivery by some four to six weeks. The hydrostatic and leak-tightness tests on the first vessel are scheduled for late August.

The research efforts of the HSST program are carried out under 12 tasks, 3 of which have been completed. The activities of the HSST program are discussed under the other active task headings that follow.

References

1. N. Krishnamurthy, *Three-Dimensional Finite Element Analysis of Thick-Walled Vessel-Nozzle Junctions with Curved Transition,* ORNL-TM-3315 (July 1971).

2. C. W. Hunter and J. A. Williams, *Fracture and Tensile Behavior of Neutron-Irradiated A 533-B Pressure Vessel Steel,* HEDL-TME-71-76 (Feb. 5, 1971).

4.2 MATERIAL INSPECTION AND CONTROL (TASK B)

(AEC Activity 04 60 80 03 1)

C. E. Childress

The major effort in this task remains the monitoring of the fabrication of the intermediate test vessels. A report was written and is being reviewed which identifies the material which has been set aside for use as surveillance monitoring steel.

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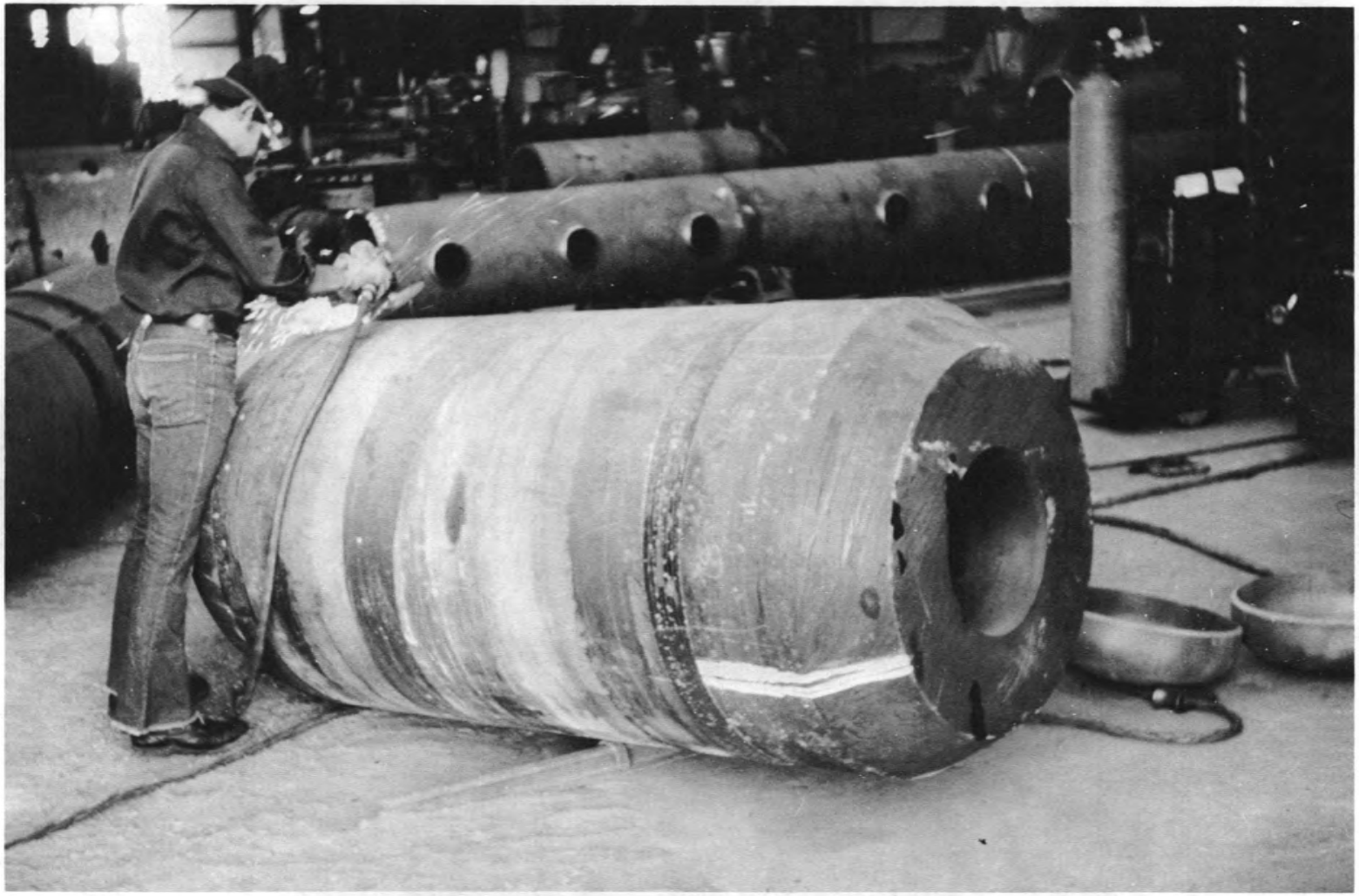

Fig. 4.1. Welds on 6-in.-thick vessel being prepared for x-ray inspection.

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

(AEC Activity 04 60 80 03 1)

R. G. Berggren D. A. Canonico

All tests planned on HSST plates are now completed and the results are being collated. The variability in steel produced by various heat treatment parameters will be investigated once the current investigation of Charpy thickness specimens is completed.[1]

Reference

1. W. B. Cottrell, *ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1971*, ORNL-TM-3483, p. 84.

4.4 FRACTURE MECHANICS INVESTIGATIONS (TASK E)

(AEC Activity 04 60 80 03 1)

J. G. Merkle

The three-dimensional elastic-plastic analyses (Brown University), the gross strain investigations (TRW Systems), and the crack arrest investigations (Materials Research Laboratory) continue. The testing of large compact tension specimens up to 550°F has been completed. Typical behavior of the load-deflection curves of these specimens is shown in Fig. 4.2. A summary of the fracture toughness in terms of K_{Icd} and volumetric energy ratios (see Ref. 1) is given in Table 4.1, and the volumetric energy ratios are plotted in Fig. 4.3. In the figure it is seen that as temperature increases, the relative behavior between specimens of different thicknesses tested at the same temperature improves. The data have been used to calculate upper and lower bounds on K_{Ic} up to 550°F. These results are given in Fig. 4.4. Here K_{Ic} is seen to increase to a maximum lower bound value of around 450,000 psi $\sqrt{\text{in.}}$ at 200°F and to decrease to around 400,000 psi $\sqrt{\text{in.}}$ at 550°F.

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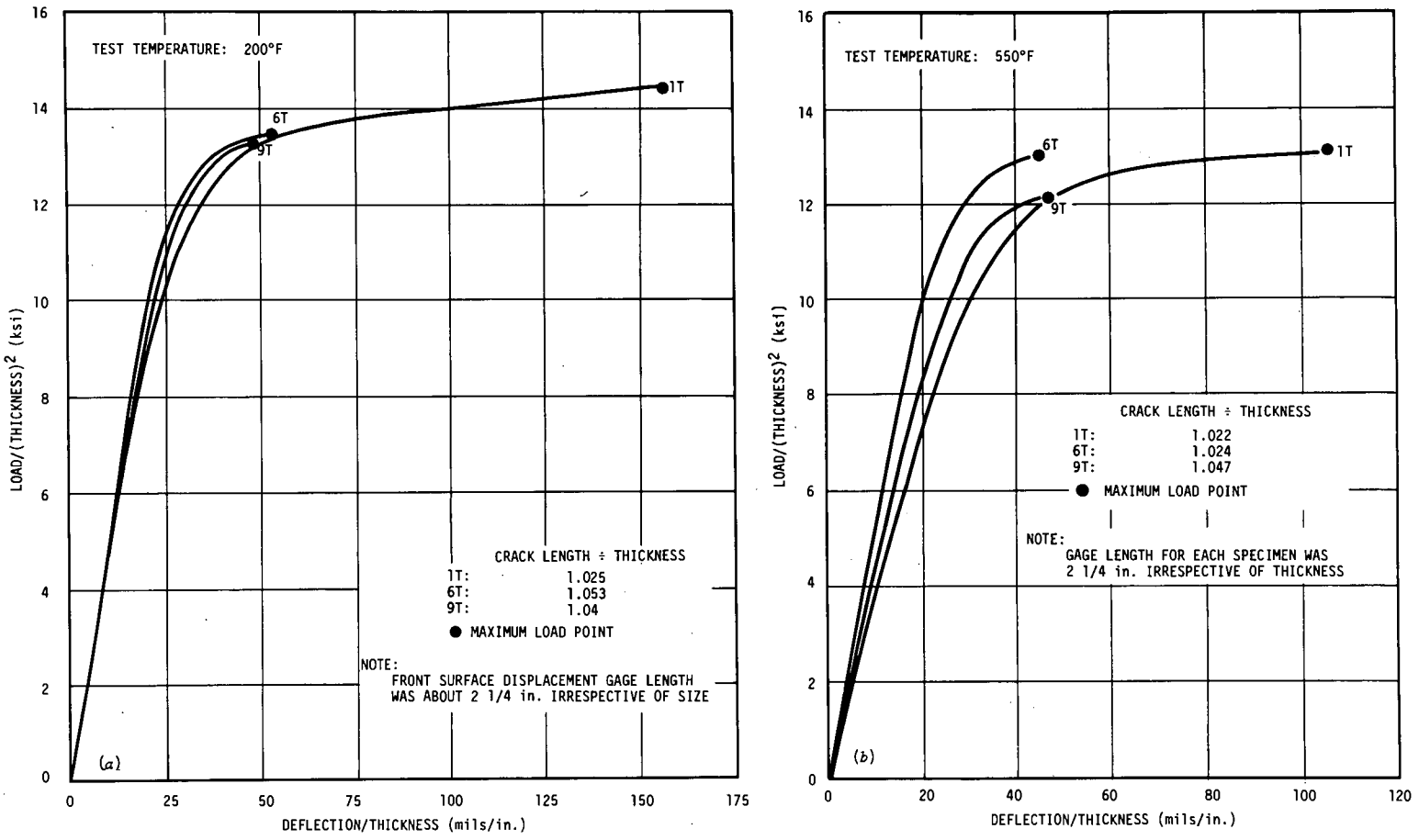

Fig. 4.2. Normalized load deflection curves for 1 T (in. thick), 6 T, and 9 T compact tension specimens tested at 200 and 550°F (ASTM A 533, grade B, class 1 steel, longitudinal orientation).

Table 4.1. Summary of the K_{Icd} and volumetric energy ratio results for ASTM A 533, grade B, class 1 steel HSST plate 03, longitudinal orientation

Test Temperature (°F)	K_{Ic1} (ksi $\sqrt{in.}$)	K_{Ic6} (ksi $\sqrt{in.}$)	$s_{1,6}$*	K_{Ic9} (ksi $\sqrt{in.}$)	$s_{1,9}$*
100	202	232	4.6		
130	312†	374	4.15		
200	246, 254	408	2.24	447	2.81
550	196, 212	362	1.91	394	2.41

*The average of the K_{Ic1} values, where more than one was obtained, was used in the calculation. These values were obtained from the K_{Icd} data in Ref. 1.

†This value was the one obtained at 125°F from Ref. 1.

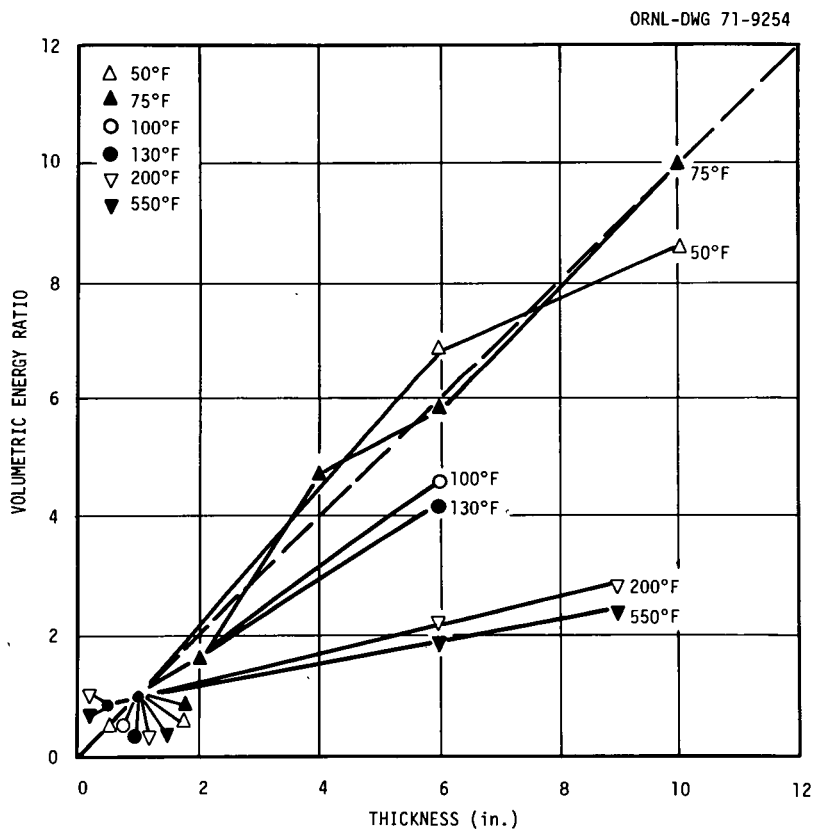

Fig. 4.3. Variations of volumetric energy ratios as a function of thickness at different temperatures normalized to a value of 1 for a 1 T specimen.

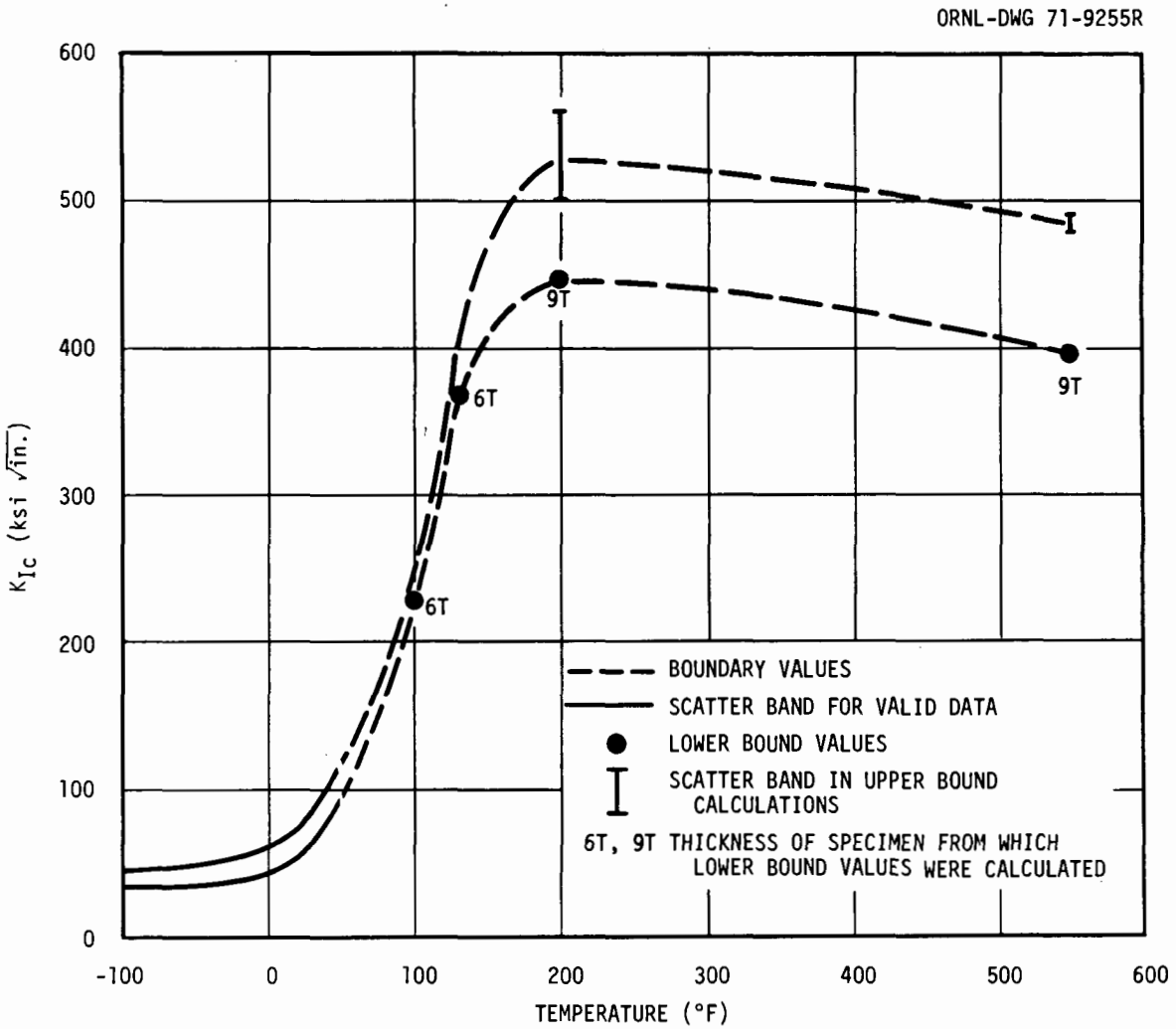

Fig. 4.4. Upper and lower bounds on K_{Ic} for ASTM A 433, grade B, class 1 steel (HSST plates 02 to 75°F, HSST plate 03 from 100 to 550°F, longitudinal orientation).

Reference

1. F. J. Witt and T. R. Mager, "Fracture Toughness K_{Icd} Values for Temperatures up to 550°F for ASTM A 533 Grade B, Class 1 Steel," *Nucl. Eng. Des.*, 17(1), in press.

4.5 FATIGUE AND CRACK PROPAGATION (TASK G)

(AEC Activity 04 60 80 03 1)

L. F. Kooistra

The main results of the investigations of crack propagation in reactor environment have been previously summarized,[1,2] and a final report is being prepared.

Preparation is under way to investigate crack propagation rates of irradiated pressure vessel steel.

References

1. W. B. Cottrell, *ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1971*, ORNL-TM-3483, pp. 84–85.

2. W. B. Cottrell, *ORNL Nuclear Safety Research and Development Program Bimonthly Report for March–April 1971*, ORNL-TM-3411, pp. 53–61.

4.6 IRRADIATION EFFECTS (TASK H)

(AEC Activity 04 60 80 03 1)

F. J. Witt

The 4 T gamma heat investigations of 4-in.-thick compact tension specimens in the Engineering Test Reactor (ETR) and the Babcock and Wilcox Test Reactor (BAWTR) have been completed. In summary, the maximum temperature difference across the crack line of a specimen was found to be between 168 and 210°F in the ETR and around 100°F in the BAWTR. An experiment in the ETR would require the design of a variable-temperature-controlled capsule. Less sophistication would be required for a BAWTR specimen capsule. The usefulness of specimens machined from broken halves of specimens irradiated in the ETR would be almost nil, and much of the material irradiated in the BAWTR would be of marginal value. The irradiation of 4-in.-thick specimens has been rescheduled for early winter, pending further evaluations.

The machining of 50 specimens from 20 broken halves of 1 T compact tension specimens (irradiated to 2×10^{19} neutrons/cm^2, $E > 1$ MeV at 550°F) has been completed. The 30 Charpy-thickness (0.394 in.) compact tension specimens are shown in Fig. 4.5. Only one specimen (see the asterisk in Fig. 4.5) will be excluded from testing.

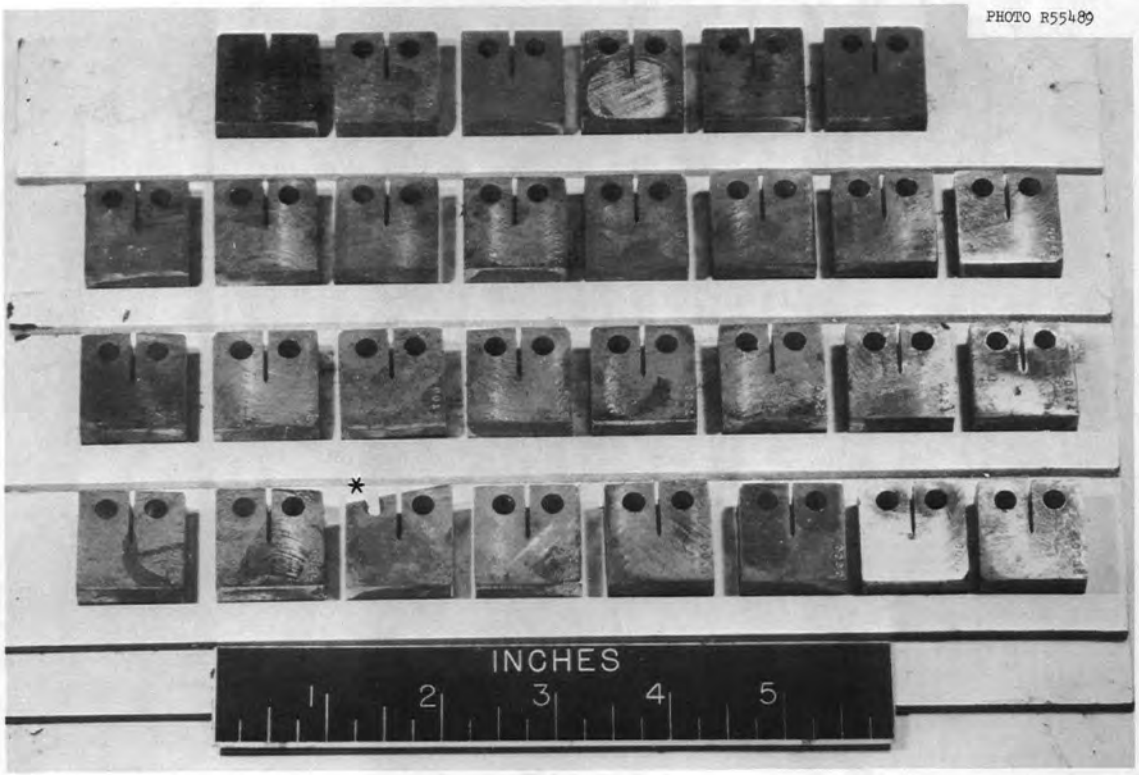

Fig. 4.5. Photograph of 30 Charpy-thickness compact tension specimens machined from broken halves of irradiated 1 T compact tension specimens.

4.7 COMPLEX STRESS STATE (TASK I)

(AEC Activity 04 60 80 03 1)

R. W. Derby

A series of 22 epoxy pressure vessel models (0.75 and 0.55 in. thick) each containing a nozzle have been tested (see Fig. 4.6). Flaws in the

Fig. 4.6. The 0.55-in.-thick (larger vessel) and 0.75-in.-thick (smaller vessel) epoxy pressure vessels used to determine the stress intensity factor in nozzle corners.

nozzle corner in the axial plane of the nozzle and shell were fatigue sharpened to different depths, and the vessels were then pressurized to fracture. Using the nominal hoop stress, σ_h, in the vessel, stress intensity factors were calculated as a function of flaw size for each series of vessels. The results are plotted in Fig. 4.7, where a is the crack depth and r_z is a nominal nozzle radius. The toughness value K_I was obtained by testing notched bend specimens. The data are compared with that obtained by Yukawa from a finite element analysis.[1]

Six flawed steel pressure vessels 1 in. thick were tested to fracture between 50 and 200°F, and three more vessels are being prepared for testing. The results currently available are being collated.

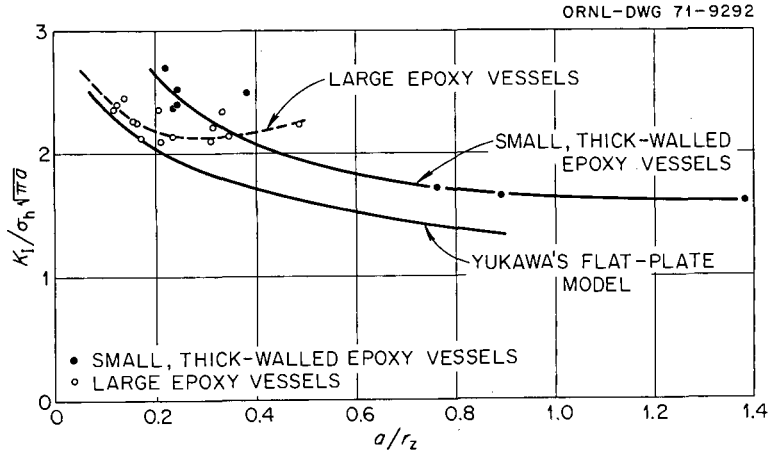

Fig. 4.7. Stress intensity factors in nozzle corners as determined from the two series of epoxy pressure vessels.

Reference

1. S. Yukawa, General Electric Company, Schenectady, N.Y., personal communication.

4.8 SIMULATED SERVICE TEST (TASK K)

(AEC Activity 04 60 80 03 1)

G. C. Robinson R. W. Derby

A series of twenty-six 1-in.-thick flawed tensile specimens are to be tested to failure. To date six have been tested, and five more are scheduled for testing in September. Preliminary analyses of these results indicate a consistent pattern of behavior between these specimens and 1-in.-thick compact tension specimens.

Three 5 5/8-in.-thick flawed submerged-arc weld tensile specimens were tested at Southwest Research Institute. Because of the anticipated superior toughness around room temperature, the test temperatures were 0, 155, and 75°F, listed in order of testing. The fracture surfaces are shown in Fig. 4.8. The gross section strain was about 0.2% at 0°F, 1.5%

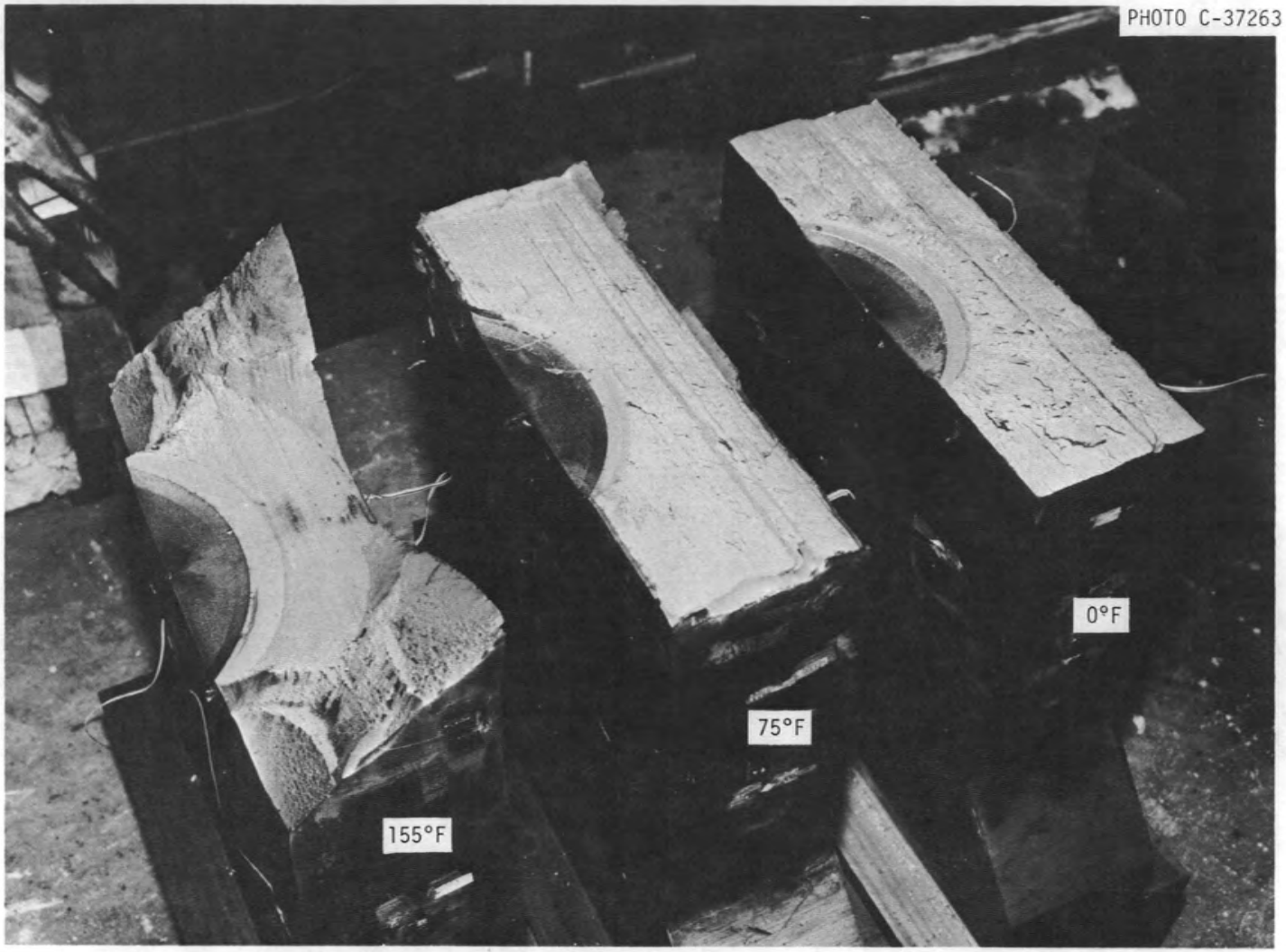

Fig. 4.8. Fracture surfaces of three submerged-arc weld flawed tensile specimens (test temperatures shown in figure).

at 155°F, and 1% at 75°F. Fracture loads were about 6×10^6 lb in all three cases.

4.9 SPECIFIC SAFETY RESEARCH (TASK L)
(AEC Activity 04 60 80 03 1)

R. W. Derby J. G. Merkle

The Charpy thickness specimens are being tested.[1] The regular Charpy tests and slow-bend tests using Charpy specimens have been completed, and the compact tension specimen series featuring fatigue sharpened cracks is ready for testing.

Reference

1. W. B. Cottrell, *ORNL Nuclear Safety Reserach and Development Program Bimonthly Report for May–June 1971*, ORNL-TM-3483, pp. 87–88.

5. DESIGN AND ANALYSES OF NOZZLES, PIPES, PUMPS, AND VALVES

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

B. L. Greenstreet J. P. Callahan

The juncture of a nozzle with a pressure vessel is a highly stressed region when the pressure vessel is subjected to in-service operating conditions. Consequently, in the case of nuclear reactor pressure vessels, the designer must be capable of accurately predicting the magnitudes and locations of maximum stresses in these highly stressed regions. The nozzles program is assigned the tasks of developing and verifying analytical procedures capable of identifying these maximum stresses. Included in the program are studies of single nozzles attached either radially or nonradially to spherical shells, perforated flat plates, and spherical shells; clusters of nozzles attached to flat plates and spherical shells; and nozzles attached to cylindrical shells. In addition to the analytical development phase, each study includes an experimental phase designed to supply highly reliable data for use in verifying the corresponding analysis. Other factors being investigated include the effect of superposition of various loadings on the nozzle, mutual interactions between adjacent openings and nozzles, and the relationship of the information resulting from this program to current design practices.

Close cooperation is maintained with appropriate subcommittees of the Pressure Vessel Research Committee (PVRC) at the Welding Research Council. The findings of the research efforts are prepared in forms

suitable for use in establishing or improving design rules for both industrial and AEC-RDT codes and standards. Application of this information to the design of pressure vessels will provide added assurance that reliability and safety standards are met.

As a member of the Pressure Vessel Research Committee, ORNL has been assigned the tasks of coordinating, reviewing, and evaluating the research programs for the PVRC Subcommittee on Reinforced Openings and External Loadings. Included in this management function are: (1) directing AEC-sponsored work, including both the work at ORNL and the work under subcontract; (2) generating parameter studies in support of code-oriented evaluation and correlation work being performed under subcontract at Battelle Memorial Institute; (3) reporting and making recommendations to the subcommittee on non-AEC-sponsored projects; and (4) soliciting comments and recommendations from the PVRC Subcommittee concerning AEC-sponsored work.

The next meeting of the PVRC Subcommittee on Reinforced Openings and External Loadings is scheduled for September 28, 1971, in New York. A progress report is being prepared for distribution to the Subcommittee members prior to the meeting. Personnel from ORNL will attend the meetings of the Subcommittee, the Design Division, and the Main Committee of the PVRC.

Single Nozzles in Spherical Shells

The 22 1/2° nonradial nozzle model 802-222 (Ref. 1), with an outside diameter of 2.625 in., has been prepared at the University of Tennessee for application of moment loadings. This is the last of the planned loadings for this particular configuration. Moments will be applied in the 0 and 90° planes, as identified in Fig. 5.1. Initiation of the moment loading test has been delayed by unanticipated problems with data acquisition equipment.

Single Radial Nozzles in Cylindrical Shells

We have nothing to report at this time.

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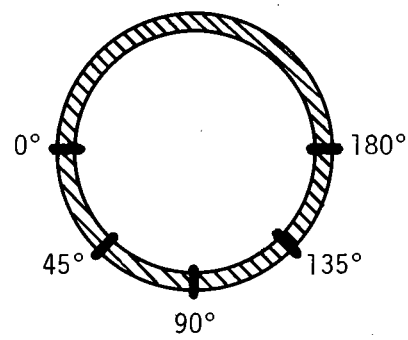

SECTION "AA"

NOTE: ALL PLANES IN THE HEMISPHERE PRODUCE GREAT CIRCLES ON THE DOME.

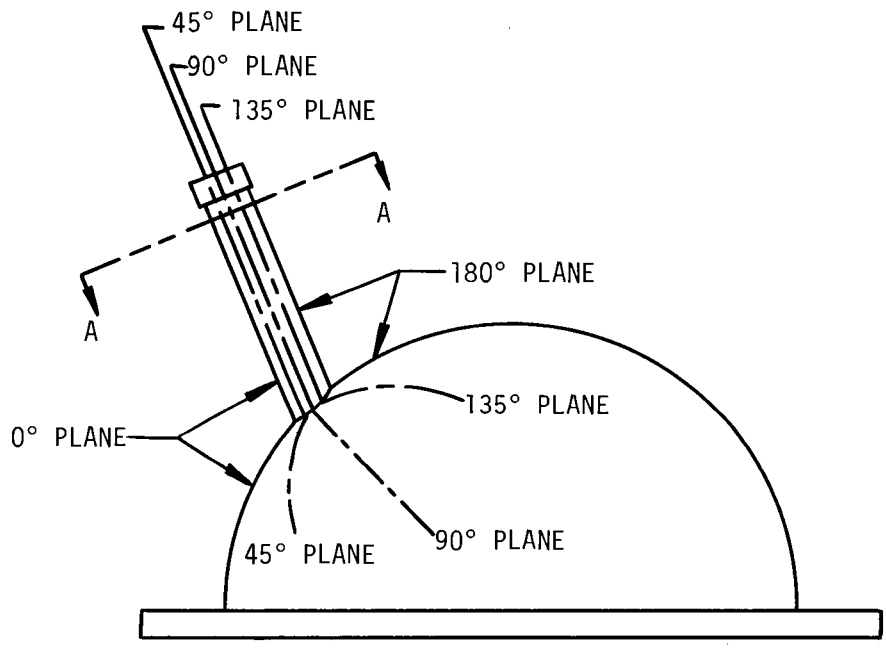

Fig. 5.1. Strain gage reference plane locations for the 22 1/2° non-radial nozzle model.

Clusters of Nozzles

At Auburn University, the program for testing clusters of nozzles attached to flat plates and hemispherical shells is continuing. The computerized data reduction and plotting procedures developed at the University of Tennessee[2] in connection with the single-nozzle model tests have been adapted for use with the Auburn University test data. Initially, the procedures are being used to reduce the internal pressure test data for the hemispherical shell model having a two-nozzle cluster.

Analyses are being made at ORNL using the computer code INTRTIE,[3] developed at Battelle Memorial Institute for multiple-nozzle flat-plate models. Thus far the computer program is being used to analyze one- and two-nozzle clusters for a one-to-one biaxial loading. In addition, a flat-plate model having the same nozzle geometry and dimensions as the two-nozzle hemispherical shell model tested at Auburn University is being analyzed for a one-to-one biaxial loading. The results of this analysis should provide a reasonably good means for comparing the internal-pressure experimental results of the hemispherical model.

All data from the one-nozzle flat-plate model tests have been examined, and loadings showing questionable data have been repeated. The one-nozzle model will be replaced in the loading frame by the two-nozzle flat-plate model for which instrumentation and wiring of the 492 strain gages has been completed.

The loading frame designed for testing the four-nozzle hemispherical shell model has been fabricated at Auburn University. Instrumentation of the model with strain gages is in progress.

The flat-plate model with a five-nozzle cluster, the last in the current flat plate model series scheduled for testing, has been transported to Auburn University from ORNL. All strain gages to be located inside the nozzles were installed and wired at ORNL. In addition, all strain gages located in relatively inaccessible regions between the nozzles have been installed. The balance of the gage installation and wiring will be done at Auburn University.

Reinforcement Studies

Prior to initiation of a parameter study of fillet reinforcement for radial nozzle-to-spherical shell attachments, a study was made to determine the most suitable analytical method. The finite-element method of analysis has been thoroughly validated for this particular problem;[4] however, it requires excessively long running times on the computer and therefore becomes quite expensive to use for the large number of calculations needed for a parameter study. Consequently, another method of analysis suitable for use in the prescribed range of parameters has been investigated at ORNL. The previously validated finite-element analytical results were used as the standard by which the other analytical method was compared. Analytical results obtained using the computer code STATIC,[5] which is based on a numerical shell analysis, were compared with both experimental and analytical results for model S-24E (Ref. 4) and with analytical results for model S-72D. Model S-24E is a relatively thick-shell type having diameter-to-thickness ratios D_i/T = 24 and d_i/t = 11.4 for the shell and nozzle, respectively, and a nozzle-to-shell stress ratio of s/S = 0.95. This model contains 65% outside fillet reinforcement, based on Section III of the ASME Boiler and Pressure Vessel Code cross-sectional area replacement rule.

Model S-72D is a relatively thin reinforced model having diameter-to-thickness ratios D_i/T = 71.9 and d_i/t = 35.7 and a nozzle-to-shell stress ratio s/S = 1.0. This model has 100% outside fillet reinforcement, based on the area replacement rule.

Comparisons of results for the two models are shown in Figs. 5.2 and and 5.3. The results for the relatively thick-walled model S-24E (Fig. 5.2) agree reasonably well, since this model represents the upper limit of thicknesses to be considered in the parameter study. In the case of the thin-walled model (Fig. 5.3) the agreement is very good. On the strength of these comparisons, the planned parameter study consisting of the analysis of 32 models under internal pressure loading is now being conducted using the shell code STATIC.

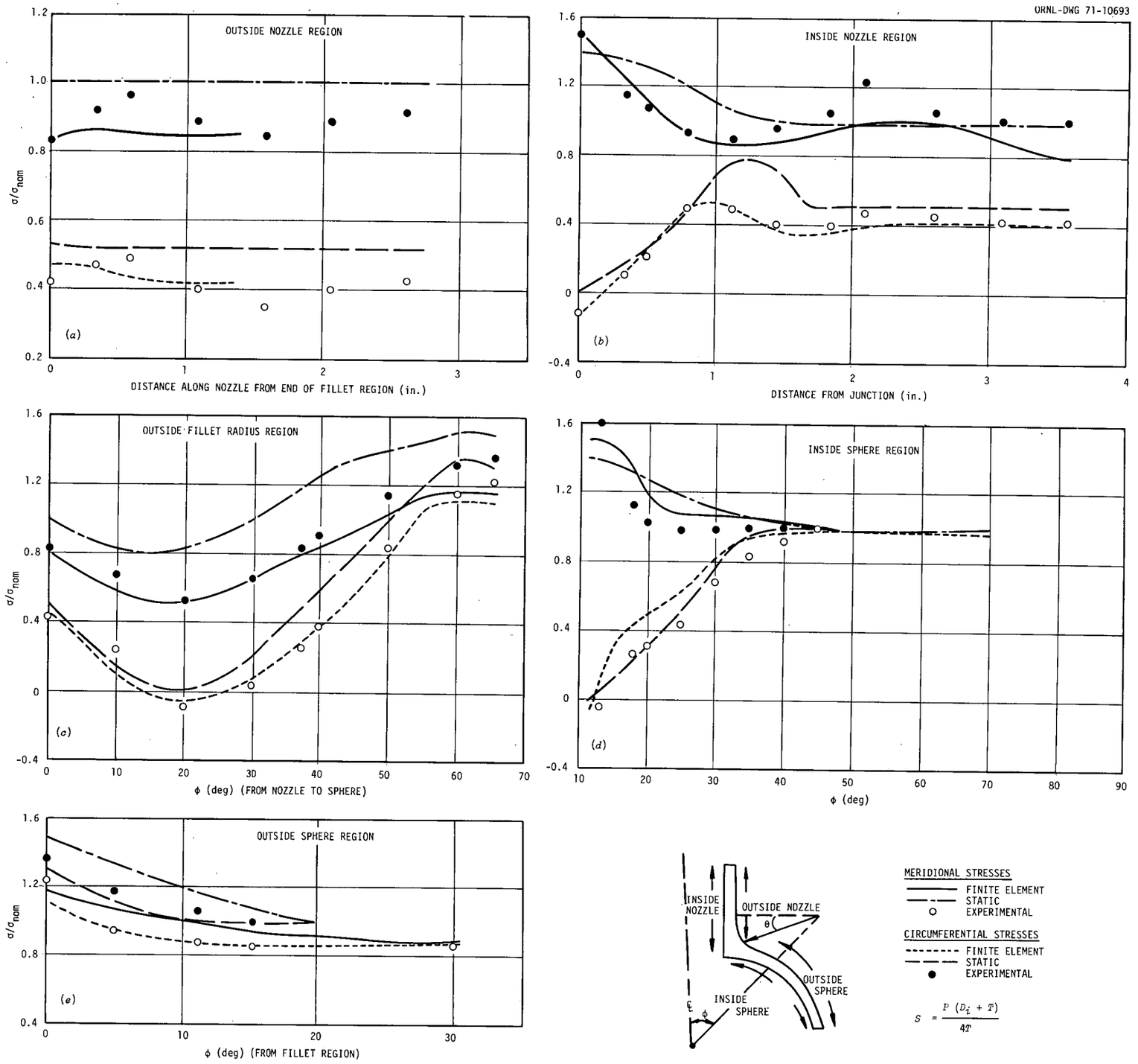

Fig. 5.2. Comparisons of theoretical and experimental stresses for model S-24E.

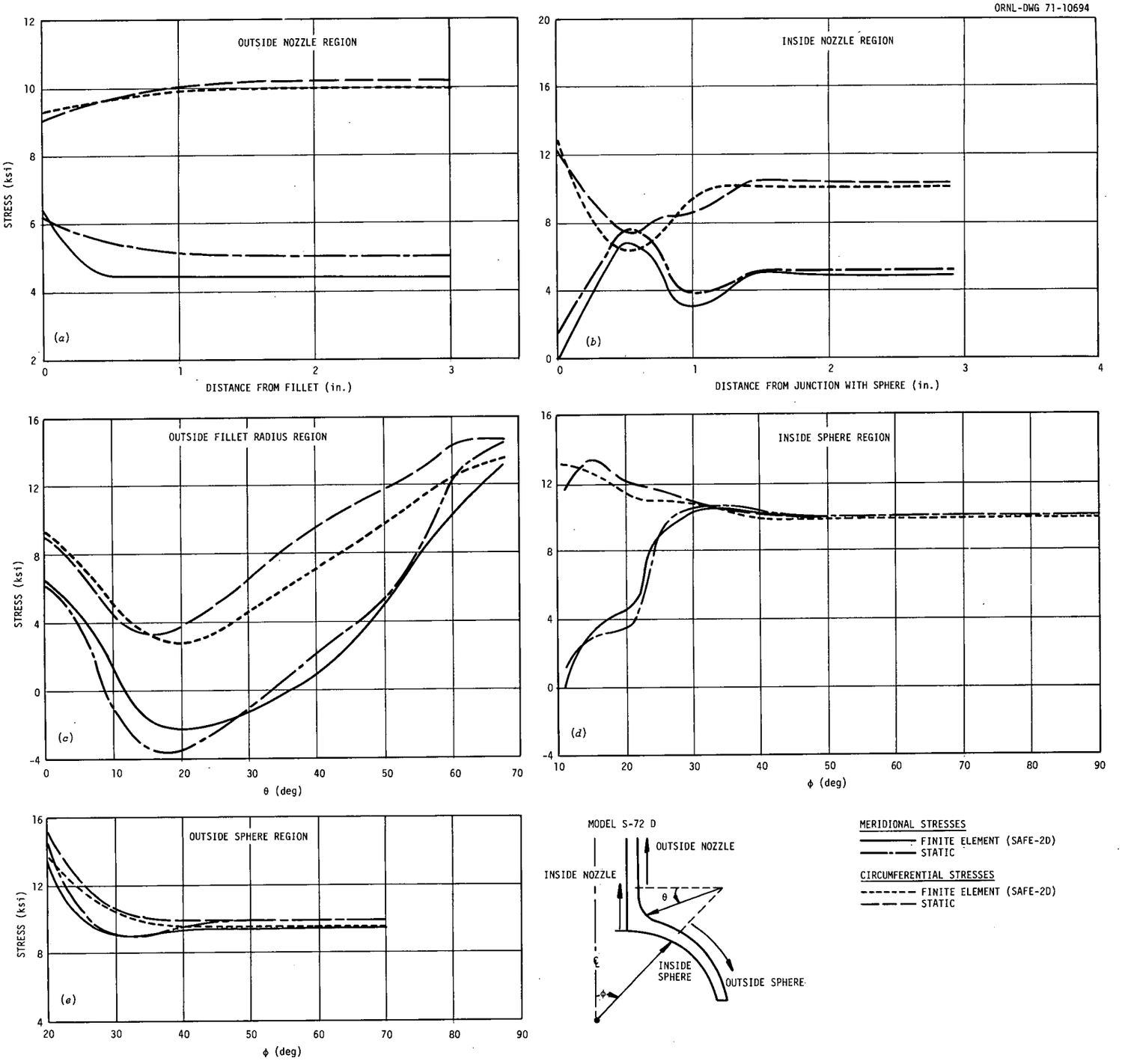

Fig. 5.3. Comparisons of finite-element and static analytical results for model S-72D.

References

1. W. L. Greenstreet, S. E. Moore, and J. P. Callahan, *Second Annual Progress Report on Studies in Applied Solid Mechanics*, ORNL 4693 (July 1971).

2. G. R. Stengl, *A Computer Program for the Generation of Stress Versus Profile Drawings*, ME7-70-3, The University of Tennessee Department of Mechanical and Aerospace Engineering (June 1970).

3. L. E. Hulbert, A. T. Hopper, and E. F. Rybicki, *Second Annual Report on the Stress Analysis of Perforated Plates with Single and Clustered Nozzles*, Battelle Memorial Institute (Oct. 9, 1970).

4. R. C. Gwaltney and J. M. Corum, "Analytical Investigations of Compact Reinforcement for Radial Nozzles in Spherical Shells," ASME Paper No. 71-PVP-26, to be published in the Journal of Engineering for Industry.

5. A. Kalnins, "Analysis of Shells of Revolution Subjected to Symmetrical and Nonsymmetrical Loads," to be published in the ASME Journal of Applied Mechanics (September 1964).

5.2 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] is an AEC-supported program within the overall nuclear safety program as part of a large AEC-industry cooperative effort to develop basic structural behavior and component design information for nuclear system piping, pumps, and valves. This information, when reduced to convenient form for inclusion in industry and AEC-RDT codes and standards, will provide for the continued adequate and safe design of modern nuclear power plant piping systems.

Principal tasks within the program are thorough and accurate experimental and analytical stress analyses of pipe fittings; data correlations and evaluations; and the reduction of this information to design charts, graphs, tables, and design rules to be used by the various code bodies. Data will be presented in terms of stress indices and flexibility factors for direct use. At present the experimental work is confined to tees,

branch connections, and elbows, while analytical studies are planned for all standard pipe fittings. Overall interpretive reports will also be written on work sponsored under this program plus that done by others.

The ORNL management responsibility for the AEC-funded portion of the AEC-industry cooperative program includes the provision for keeping both AEC and industry fully informed through close cooperation with the Pressure Vessel Research Committee (PVRC) of the Welding Research Council and the code committees of the American National Standards Institute (ANSI) and the American Society of Mechanical Engineers (ASME). Recently the piping design groups of ANSI and ASME code committees were combined and their membership organized into Task Group I of the PVRC Subcommittee to Develop Stress Indices for Piping, Pumps, and Valves. The three groups are therefore now able to act as a unified body in promoting the research effort and applying the results in improved design rules and practices. The next meeting will be held on September 14, 1971, in New York.

Survey Reports

We have nothing to report this period.

Structural Analysis of Tees

One of the major tasks under the ORNL piping program is concerned with structural investigations of ASA standard B16.9 tees, since essentially no information existed previously that could be used to develop stress indices for design analysis applications. Studies include full-scale experimental and analytical stress analyses of tees ranging in nominal size from 6 to 24 in., with wall thicknesses ranging from sched 10 to sched 160, and branch-to-run diameter ratios from 10/24 to 1/1. Each of 17 tees in this series is to be instrumented with strain gages, tested with various loading conditions, and finally tested to failure in a low-cycle fatigue test. Tees of type 304L stainless steel and of type A106 grade B carbon steel are being tested, and the fatigue loading conditions include both internal pressure and applied bending moments. The experimental work also includes steady-state thermal stress tests and photoelastic model studies to determine through-the-wall thickness stress distributions.

The main thrust of the analytical work is directed toward developing accurate and efficient finite-element computer codes that will be used in parameter studies to increase the coverage beyond the scope of the experimental studies. Not included within the present AEC funding level is a study on the behavior of mixing tees. The PVRC is attempting to obtain support for this study from private industry.

Portions of the work on ASA standard B16.9 tees are being done at Oak Ridge National Laboratory and under subcontract at the University of California, Southwest Research Institute (SwRI), Combustion Engineering (C.E.), and Westinghouse Research Laboratories. Experimental and analytical studies on thin-walled tees, branch connections, and laterals (idealized cylinder-to-cylinder connections) are being done at Oak Ridge and at Auburn University.

Elastic response and fatigue tests of ASA B16.9 tees

To date, 9 of the 17 tees in this series[2] either have been or are being tested under subcontract to ORNL for elastic response and low-cycle fatigue failure. Five of these, all 12-in. tees, are being tested at SwRI, and the remaining four, which are 24-in. tees, are being tested at C.E. During the present reporting period the fifth and final fatigue test to be conducted at SwRI was completed, preparations were completed at C.E. for conducting the second 24-in. tee fatigue test, and preparations were completed for instrumenting the third 24-in. tee (T-13) for the elastic response tests.

The final topical reports are being written at SwRI for all their tests, including both elastic response and low-cycle fatigue tests. All the fatigue tests (including one completed at C.E. on a 24-in. tee) were fully reversed deflection-controlled bending tests with the load applied to the branch pipe extension and a constant internal pressure in the tee. In each case the loading produced a maximum cycling equivalent elastic principal stress of ±83,500 psi at the gage site with the highest stresses. Based on the experimental correlations of Markl,[3] failure, defined as a through-the-wall crack as evidenced by a leak, was expected in 7000 cycles. The results of the fatigue tests are summarized in Table 5.1.

Table 5.1. Low-cylce fatigue failure results for 12- and 24-in. ASA B16.9 tees

Tee No.	Nominal size	Cycling Moment Load	Constant internal pressure (psi)	Cycles at failure
T-4	12 × 12 × 12 in., sched 80	Out of plane	1925	2,062
T-6	12 × 12 × 12 in., sched 80	Out of plane	1925	1,309
T-7	12 × 12 × 12 in., sched 160	In plane	3240	11,475
T-8	12 × 12 × 6 in., sched 40	Out of plane	950	8,979
T-10	24 × 24 × 24 in., sched 40	In plane	1025	18,532
T-15	12 × 12 × 6 in., sched 40	Out of plane	950	10,200

The last tee to be tested at SwRI, T-6, failed after a rather low number of cycles compared with the target value of 7000 cycles (less than 20% of the expected life). However, when this test is evaluated according to the design procedures of ASME Code Case 1441 and recently revised design procedures[4] of the piping code ANSI B31.7, the test results tend to substantiate the conservatism of the new code rules. According to these rules the design life of this tee would be only N_c = 38 cycles. Thus the experimental life was 35 times the code allowable design life, which is more than the safety factor of 20 used in ASME fatigue design curves to account for scatter in experimental data. Nevertheless, an extensive postfailure examination of the tees and fatigue cracks will be made.

The fatigue test presently being started at C.E. on ORNL tee T-11 (24 × 24 × 24 in., sched 160) will be conducted with an internal pressure cycling between 100 and 7000 psi. To our knowledge this will be the first piping tee to be fatigue tested to failure with a cyclic pressure load. Failure is expected between 20,000 and 30,000 cycles, based on correlations between the maximum experimental stresses and cyclic fatigue life curves. The test will take approximately three months to complete.

Photoelastic tests on models of 12-in. tees

Frozen stress photoelastic analyses using one-half-scale models of the four tees listed in Table 5.2 are being conducted at Westinghouse Research Laboratories under subcontract to ORNL. All the experimental work was completed during this reporting period, and the final report is being written. Surface stresses were determined along lines 22 1/2° apart around the branch to correspond with the strain-gage layout for the prototype tees tested at SwRI, and stress distributions through the wall were determined at three maximum stress locations for each tee. Through-the-wall thickness distributions were obtained in order to provide a rational basis for dividing the maximum stress intensity value into the three categories of design stress — primary, secondary, and peak. The results for moment loading on the branch pipe for tees T-6 and T-8 indicated linear stress variations through the wall of the tee, whereas the stress distributions for internal pressure loading on tees T-7 and T-9 indicate a considerable amount of nonlinearity. The peak stress component at the inside surface of the tee was about one-half as large as the bending component and about one-fifth as large as the maximum stress intensity in the model. These results, along with the actual pressure fatigue data from tee T-11, currently under test at C.E., will provide an empirical basis for K_1, the peak-stress index of the piping codes.

Thermal stress tests on 12-in. tees

Thermal stress tests are being conducted at Oak Ridge on two 12-in. tees that are essentially identical to the tees, T-6 and T-8, tested for

Table 5.2. Photoelastic test models for 12-in. ASA B16.9 tees

Tee No.	Nominal size	Loading condition
T-6	12 × 12 × 12 in., sched 80	Out-of-plane moment on branch
T-7	12 × 12 × 12 in., sched 160	Internal pressure
T-8	12 × 12 × 6 in., sched 40	Out-of-plane moment on branch
T-9	12 × 12 × 6 in., sched 160	Internal pressure

elastic response at SwRI. The test models were fabricated in FY 70 by welding 2-ft-long segments of pipe to each outlet and grinding the welds flush on both the inside and outside surfaces. Eighty-six high-temperature strain-gage rosettes were installed on T-6, and both tees were held in storage until the beginning of FY 72 (July 1971).

During the current reporting period work was resumed on preparing the models for testing. Tee T-6 was instrumented with thermocouples placed at the strain-gage sites, and the various pieces of test equipment were reassembled and checked. Connectors are presently being attached to the strain-gage lead wires in preparation for connecting the instrumentation with the data acquisition system. We plan to begin calibration tests during the next reporting period.

Finite-element stress analysis

Concurrently with the experimental studies discussed above, a finite-element computer code is being developed at the University of California, Berkeley, specifically for the elastic stress analysis of B16.9 tees. Loadings of interest are internal pressure, moment and thrust loadings applied to the branch and run pipe, arbitrary combinations of these loadings, and loadings arising from steady-state and transient temperature distributions. Detailed stress analyses of three of the 12-in. tees tested at SwRI (T-6, T-7, and T-8) and comparisons with the experimental data are included in the scope of the work as partial proof of the analysis. The final version of the computer program is expected to require only a small amount of input data and to produce calculated stress and flexibility results in easily interpreted tabular and graphical form.

During the current reporting period many of the service function subroutines, including the automatic mesh generation and input preparation packages and the stress output packages, were completed. The assembled computer program was operated successfully through its entire sequence for the first time. During the next few weeks a number of minor features will be added, the programming will be optimized and cleaned up, and the graphical output package will be developed. Present plans are to draw contour plots and pseudo-isoclinic plots of the principal stresses directly on orthographic projections of the tee.

A good deal of effort has been expended lately in developing the input preparation package to enable analyses for the variety of geometrical shapes allowed under the ASA B16.9 standard. As a result one can either develop the finite-element mesh automatically using as many as nine or ten different options, or develop the input manually using measured dimensional information from the tee being analyzed. Currently tee T-7 is being analyzed with both automatically and manually prepared input for comparison with the experimental data obtained at SwRI.

Thin-walled tees and branch connections

The series of four idealized thin-walled tees (cylinder-to-cylinder connections) referred to in earlier reports[2] is being experimentally and analytically stress analyzed at ORNL as a means of obtaining high-quality data for use in developing better theoretical solutions. Two of the models were analyzed previously, and the remaining two, model 1 with a diameter ratio d/D of 1.0 and model 4 with a d/D of 0.129, are scheduled for study this fiscal year. Model 4 has now been fully instrumented for testing and mounted in the loading frame. The strain-gage lead wires are being connected to the new computer-controlled data acquisition system described previously.[5] Strain gages are being installed on model 1, and testing is scheduled to begin in May 1972.

Elbows, Curved Pipe, and Miter Bends

Elastic response analysis of machined elbows

The elastic response tests were run for the first of four machined 10-in. long-radius elbows being stress analyzed at Oak Ridge, and the strain-gage data are being analyzed. This model, ME-1 (see Ref. 2), has an outside diameter of 10.750 in. and a uniform wall thickness of 0.365 in. Eighty-three 45° strain-gage rosettes were mounted on the model, with 16 of the rosettes on the inner surface and 67 on the outside surface. Thirteen individual loading conditions were run: internal pressure; in-plane, out-of-plane, and torsional bending moments applied to the free end of the assembly; and combinations of internal pressure and bending moments as listed in Table 5.3.

Table 5.3. Elastic response test loadings for machined elbow model ME-1

Loading	Maximum load	Pressures for combined loadings			
		0	$1/3P_{max}$	$2/3P_{max}$	P_{max}
Internal pressure	1,305 psi				
In-plane bending	94,080 in.-lb	0	435	870	1305
Out-of-plane bending	94,080 in.-lb	0	435	870	1305
Torsional bending	94,080 in.-lb	0	435	870	1305

Combined loadings were conducted by first pressurizing the model to the indicated pressure and then adding the moment loading in four steps to the indicated value. A slight overload was applied and the model was unloaded, with data taken at the same loads used during loading. The strain-gage data were thus scanned by the data acquisition system at 8 load values for each of the 13 different loadings. Strain readings were punched on IBM cards, examined for linearity, and used to calculate stresses. The stresses are being plotted as a function of position on the model for further examination.

References

1. S. E. Moore and W. L. Greenstreet, "Program Plan for the ORNL Piping Program — Design Criteria for Piping, Pumps, and Valves" (unpublished).

2. W. L. Greenstreet, S. E. Moore, and J. P. Callahan, *Second Annual Progress Report on Studies in Applied Solid Mechanics (Nuclear Safety)*, ORNL-4693 (July 1971).

3. A. R. C. Markl, "Fatigue Tests of Piping Components," *Trans. ASME* 74, 287–303 (1952).

4. E. C. Rodabaugh and S. E. Moore, *Phase Report No. 115-10 — Comparisons of Test Data with Code Methods for Fatigue Evaluation*, ORNL-TM-3520 (to be published).

5. B. L. Greenstreet and S. E. Moore, "Design Criteria for Piping, Pumps, and Valves," *ORNL Nuclear Safety Research and Development Program Bi-monthly Report for January–February 1971*, ORNL-TM-3342, pp. 63–72.

6. RDT STANDARDS PROGRAM

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

W. F. Ferguson R. A. Schmidt

The prime responsibility for the organization, preparation, and distribution of engineering standards was undertaken by ORNL at the request of the AEC Division of Reactor Development and Technology. These standards provide sound bases from which nuclear reactors and related facilities under the guidance of RDT can be planned, designed, constructed, and operated. The RDT Standards Program at ORNL is responsible for the preparation of standards applicable to water-cooled reactor systems and for standards covering liquid-metal-cooled systems. Standards for liquid-metal-cooled systems are prepared in cooperation with the Liquid Metal Engineering Center (LMEC), which has the prime responsibility for standards applicable to these systems. The RDT Standards Program is a source of proven engineering practices for designers, builders, and operators of nuclear reactors and related facilities that will improve quality assurance, reactor safety, and functional reliability. Current activities include:

1. analyzing RDT project needs for standards;
2. working cooperatively with projects under the direction of RDT in preparing and applying standards;
3. developing program plans, schedules, goals, and details;
4. preparing and distributing new standards and reviewing and updating existing standards;
5. maintaining close liaison with the LMEC Standards Office;

6. reviewing and providing advisory services on standards prepared by other participating projects and organizations;
7. working with RDT technical branches in the development of a more effective standards program;
8. preparing guidelines for format, definitions, titling, and numbering;
9. participating in committee and task group activities of recognized voluntary codes and standards agencies;
10. consulting with reactor designers, builders, operators, AEC agencies, and contractors;
11. writing and distributing reports, indexes, and tabulations pertaining to the program;
12. collecting and analyzing reports of reactor experience and technical data;
13. analyzing and preparing summaries of RDT incident reports.

Organization, Scheduling, and Manpower

Personnel assignments and schedules were reviewed and adjusted, and assistance requirements for on-going projects were reviewed for the fiscal year. Increased effort was directed toward expediting the updating of published standards to conform to the 1971 edition of the ASME Boiler and Pressure Vessel Code, amending standards to facilitate project applications, and providing review of standards being prepared by on-going projects. Negotiations for the subcontract providing financial support to ANSI are nearing completion.

Activities

A number of material, process, and component standards are being amended or revised to conform to the 1971 edition of Section III, Nuclear Power Plant Components, of the ASME Boiler and Pressure Vessel Code. Thirty-two revisions to existing tentative standards were submitted to RDT for approval during this period, and eleven documents were issued as tentative standards for trial use.

The LMEC recommendation for a standard compliance statement to be used in the scope section of all RDT standards for components that may be designed and built in accordance with the ASME Code was evaluated by ORNL. The ORNL proposal on this recommendation was prepared and submitted for RDT appraisal.

At the request of RDT, ORNL is providing assistance to the Grinding Wheel Institute to establish standards and criteria for the composition of grinding wheels used on stainless steels for nuclear installations.

Representatives from the Standards Office continued participation in ANS-7, Subcommittee on Reactor Component Criteria; ANS-11.13, Concrete Radiation Shields; ASTM A-10, Subcommittee IV-408, Stainless Steel Corrosion; ASTM C-16, Committee on Thermal Insulation; ASTM D-28, Activated Carbon; ANSI Nuclear Technical Advisory Board; ANSI N45-3.0, Subcommittee on Nuclear Quality Assurance Standards; N45-9, Subcommittee on Air Cleaning Systems; N45 Ad Hoc Committees 3.1 through 3.8; ASME Boiler and Pressure Vessel Working Groups on Core Support Structures (SC-III) and Quality Assurance and Stamping (SC-III); and ISA SP-60, Control Centers Committee.

Considerable effort was extended toward assisting in the preparation of the ANSI N45 standards related to the construction phase of nuclear power plants and assisting the ANSI Nuclear Technical Advisory Board as requested.

As agreed with RDT, most of the documents previously identified as Engineering Technology References (ETR) will be published as ORNL reports. Six of the documents scheduled for publication as ETRs had been previously published as ORNL reports, and the piping and heat transfer documents are now being edited and prepared for publication.

Status of standards being processed

A total of 112 tentative standards are currently available for use, and 58 draft standards are awaiting approval by RDT. The ORNL mailing list for tentative standards now contains 294 addressees, including AEC installations and contractors, codes and standards organizations, architect-engineers, consultants, nuclear component manufacturers, and electrical utilities who receive a total of 568 copies.

In response to telephone and memo requests, 1218 copies of RDT documents and 15 copies of the standards index were mailed to 48 organizations, 10 of which were added to the distribution list to receive future issues of the standards or the index.

The following approved tentative RDT standards were issued.

		Issued by
E 1-13T	Valve, Isolation, Butterfly Type Rubber Seated Containment Heating and Ventilating Systems for Light Water Nuclear Power Plants	ANC
E 3-2T	Centrifugal Free-Surface Sodium Pump with Electrical Drive	HEDL
E 15-2T	Requirements for Nuclear Components (Supplement to ASME Boiler and Pressure Vessel Code, Section III)	ORNL
*F 3-6T	Nondestructive Examination	ORNL
F 3-37T	Special Requirements for Metal Products	ORNL
F 6-5T	Welding Qualifications (Supplement to ASME Boiler and Pressure Vessel Code, Section IX)	ORNL
F 6-10T	Repair of Materials by Welding	ORNL
*M 1-2T	Stainless Steel Welding Rods and Bare Electrodes (ASME SFA-5.9 with Additional Requirements)	ORNL
*M 2-2T	Stainless and Low Alloy Steel Forgings (ASME SA-182 with Additional Requirements)	ORNL
*M 2-5T	Austenitic Stainless Steel Welding Fittings (ASME SA-403 with Additional Requirements)	ORNL
*M 3-2T	Stainless and Alloy Steel Seamless Tubes (ASME SA-213 with Additional Requirements)	ORNL
*M 3-3T	Austenitic Stainless Steel Pipe (ASME SA-376 with Additional Requirements)	ORNL
*M 5-1T	Stainless Steel Plate, Sheet, and Strip (ASME SA-240 with Additional Requirements)	ORNL

*Revision.

M 12-2T	Calcium Silicate Block and Pipe Thermal Insulation (ASTM C533 with Additional Requirements)	ORNL

The following approved amendments to tentative standards were issued.

	Issued by
Amendment 1 to E 15-2T, Requirements for Nuclear Components (Supplement to ASME Boiler and Pressure Vessel Code, Section III)	ORNL
Amendment 1 to F 1-1, Preparation of RDT Standards	ORNL
Amendment 1 to F 2-2T, Quality Assurance Program Requirements	ORNL
Amendment 1 to F 6-5T, Welding Qualifications (Supplement to ASME Boiler and Pressure Vessel Code, Section IX)	ORNL

The following standards were prepared and submitted to RDT for approval

C 7-5	Thermocouple Material, Chromel-P Vs. Alumel, Magnesium-Oxide Insulated, Stainless Steel Sheathed
F 6-14	Brazing Fabrication Requirements
*M 1-1	Stainless Steel Covered Welding Electrodes (ASME SFA-5.4 with Additional Requirements)
*M 1-3	Mild Steel Covered Welding Electrodes (ASME SFA-5.1 with Additional Requirements)
*M 1-4	Low Alloy Steel Covered Welding Electrodes (ASME SFA-5.5 with Additional Requirements)
*M 1-5	Welding Rods and Electrodes, Surfacing (AWS A5.13 with Additional Requirements)
*M 1-6	Mild Steel Electrodes and Welding Rods (ASME SFA-5.18 with Additional Requirements)
*M 1-7	Copper and Copper Alloy Welding Electrodes (ASME SFA-5.6 with Additional Requirements)
*M 1-8	Copper and Copper Alloy Welding Rods (ASME SFA-5.7 with Additional Requirements)
*M 1-9	Brazing Filler Metal (ASME SFA-5.8 with Additional Requirements)
*M 1-10	Nickel and Nickel Alloy Covered Welding Electrodes (ASME SFA-5.11 with Additional Requirements)

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*Revision.

*M 1-11	Nickel and Nickel-Alloy Bare Welding Rods and Electrodes (ASME SFA-5.14 with Additional Requirements)
*M 2-1	Carbon Steel Forgings (ASME SA-105 with Additional Requirements)
*M 2-3	Carbon and Alloy Steel Welding Fittings (ASME SA-234 with Additional Requirements)
*M 2-4	Alloy Steel Forgings (ASME SA-336 with Additional Requirements)
*M 2-7	Carbon and Alloy Steel Forgings, Vacuum Treated (ASME SA-508 with Additional Requirements)
*M 2-8	Alloy Steel Forgings (ASME SA-541 with Additional Requirements)
*M 3-1	Carbon Steel Seamless Pipe (ASME SA-106 with Additional Requirements)
*M 3-4	Nickel Alloy Seamless Tubes (ASME SB-163 with Additional Requirements)
*M 3-5	Austenitic Stainless Steel Welded Tubing (ASME SA-249 with Additional Requirements)
*M 3-6	Austenitic Stainless Steel Pipe (ASME SA-312 with Additional Requirements)
*M 3-7	Austenitic Stainless Steel Welded Pipe, Large Diameter (ASME SA-358 with Additional Requirements)
*M 3-11	Carbon and Low Alloy Steel Welded Pipe (ASME SA-155 with Additional Requirements)
*M 3-12	Ferritic Alloy Steel Seamless Pipe (ASME SA-335 with Additional Requirements)
M 3-20	Polyethylene Instrument Tubing
*M 4-2	Austenitic Stainless Steel Castings (ASME SA-351 with Additional Requirements)
*M 5-2	Carbon Steel Plates (ASME SA-516 with Additional Requirements)
*M 5-3	Low Alloy Steel Plates (ASME SA-533 with Additional Requirements)
*M 5-5	Low Alloy Steel Plates (ASME SA-387 with Additional Requirements)
*M 5-7	Nickel-Iron-Chromium Alloy Plate, Sheet, and Strip (ASME SB-409 with Additional Requirements)

*Revision.

*M 6-1 Alloy Steel Bolting Material for Low Temperature Service (ASME SA-320 with Additional Requirements)

*M 6-3 Alloy Steel Bolting Material for High Temperature Service (ASME SA-193 with Additional Requirements)

*M 6-4 Alloy Steel Nuts for Bolting for High Temperature Service (ASME SA-194 with Additional Requirements)

*M 7-3 Stainless Steel Bars and Shapes (ASME SA-479 with Additional Requirements)

M 12-3 Mineral Fiber, Hydraulic-Setting Insulating and Finishing Cement

The following amendments to approved tentative standards were submitted to RDT for approval.

Amendment	Standard	Standard title
1, 2	F 1-1T	Preparation of RDT Standards
2, 3, 4, 5	F 2-2T	Quality Assurance Program Requirements
1	F 5-1T	Cleaning and Cleanliness Requirements for Nuclear Reactor Components
1	F 7-2T	Preparation for Sealing, Packaging, Packing, and Marking of Components for Shipment and Storage
1	F 7-3T	Requirements for Identification Marking of Reactor Plant Components and Piping
1	M 12-2T	Calcium Silicate Block and Pipe Thermal Insulation (ASTM C533 with Additional Requirements)
1	F 2-4T	Quality Verification Program Requirements

The following standards were revised to the 1971 ASME Code and submitted to ANC for use on the LOFT project.

E 4-17 Heat Exchanger, Class 2, Water-to-Water, Straight or U-Tube

E 11-1 Ion Exchanger, Non-Regenerative Type

The following standards were prepared and submitted to HEDL for use on the FFTF project.

E 1-35 Inert Gas Valves

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*Revision.

M 2-15	Nickel-Chromium Alloy Bars, Forgings, and Forging Stock (ASTM A637 with Additional Requirements)

RDT numbers were assigned to nine new standards.

		RSO
C 7-17	Resistance Temperature Detector	ORNL
C 7-18	Thermowell Systems	ORNL
C 10-1	Signal Conditioner	ORNL
C 12-1	Gas Chromatograph for Sodium Cover Gas Service	LMEC
C 16-3	Plant Protection System Buffer	ORNL
C 17-7	Removable Instrument Assembly for Liquid Metal Service	LMEC
E 1-36	Reactor Refueling Floor Valve for LMFBR	LMEC
E 6-34	Core Component Pot for LMFBR	LMEC
E 6-35	In Vessel Handling Machine (Design Only)	LMEC

The following drafts were reviewed by ORNL, and comments were prepared and submitted to the requester.

C 17-6	Cable for Nuclear Radiation Detectors
E 10-4	Pressure Vessel, Class C, Nuclear
E 13-15	Fuel Assemblies for Pressurized Water Reactors

The comments received on the following drafts are being evaluated for incorporation into the standards.

F 3-11	Helium Mass Spectrometer Leak Testing for Instruments
F 4-20	Operation and Maintenance Manuals
M 12-2T	Chloride Silicate Block and Pipe Thermal Insulation (ASTM C533 with Additional Requirements)

The comments received on the following documents were incorporated into the standards.

C 7-5	Thermocouple Material, Chromel-P Vs. Alumel, Magnesium-Oxide Insulated, Stainless Steel Sheathed
C 15-3	Current Pulse Preamplifiers for Use with Fission Counters
E 4-1T	Steam Generator for Pressurized Water Reactors
M 3-20	Polyethylene Instrument Tubing

The following standard is being rewritten to incorporate the comments received.

P 3-1 Electrical Penetration Assemblies for Nuclear Reactor Containment Structures

In-house review of the following standard prepared by ORNL is in progress.

P 4-2 High-Temperature Trace Heaters, Molded Ceramic, Electrical Type

ORNL completed in-house review of the following standard.

C 10-1 Signal Conditioner

Drafts of the following documents were distributed for review and comments.

C 2-3 Thermocouple Time Response Test

C 7-7 Thermocouple Materials, Platinum and 90% Platinum-10% Rhodium Wires, Noninsulated Reference and Standard Grades

C 10-1 Signal Conditioner

6.2 REACTOR COOLANT SYSTEM AND EQUIPMENT STANDARDS

(AEC Activity 04 60 80 03 1)

H. G. Arnold
W. A. Bush
R. W. Dehoney
W. R. Gall
D. L. Gray
J. M. O'Keefe
L. F. Parsly
T. H. Row

Piping, Valves, and Rupture Disks

RDT E 1-35, "Inert Gas Valves," was completed and submitted to HEDL. When approved, this standard will be used to procure valves for the FFTF project. It is based on the 1971 edition of Section III of the ASME Boiler and Pressure Vessel Code and RDT E 15-2T, "Requirements for Nuclear Components (Supplement to ASME Boiler and Pressure Vessel Code, Section III)." Review and approval of E 1-35 will be handled through project channels by HEDL.

RDT E 1-13T, "Valve, Isolation, Butterfly Type Rubber Seated Containment Heating and Ventilating Systems for Light Water Nuclear Power Plants," was issued as a tentative standard for trial use. This standard was

prepared by Aeroject Nuclear Corporation (ANC) as the first user and standard writer. In accordance with agreements with ANC, AEC-RDT, and ORNL, future responsibility as standard writer was transferred to ORNL.

A preliminary proposal on rules governing structural analysis of class I valves of 4 in. nominal size and smaller was prepared and submitted to RDT for information.

Pumps

Effort continued toward conversion of the FIRL "COPATS" computer program from UNIVAC 1108 language to ORNL's IBM 360/91. All runs to date on the automatic mesh generator program have been aborted because program statements were unacceptable to the 360/91. Consultation with FIRL and programming assistance from ORNL are being used to complete the conversion.

Work continued on preparation of draft standard E 3-1, "Vertical, Canned Motor, Single Stage, Centrifugal Reactor Coolant Pump," with emphasis being placed on correlating requirements with those of the 1971 edition of Section III of the ASME Boiler and Pressure Vessel Code. To permit the wet motor concept to be covered by this standard, it is planned to remove "canned motor" from the title.

Heat Exchangers

RDT E 4-17, "Heat Exchanger, Class 2, Water-to-Water, Straight or U-Tube," was revised for use with the 1971 edition of Section III of the ASME Boiler and Pressure Vessel Code and submitted to ANC. Approval of this document will be requested by ANC.

An amendment to RDT E 4-1, "Steam Generator for Pressurized Water Reactors," is in preparation. It incorporates comments from reviewers and correlates requirements with those of the 1971 edition of Section III of the ASME Boiler and Pressure Vessel Code.

Nuclear Code Supplements

RDT E 15-2T, "Requirements for Nuclear Components (Supplement to ASME Boiler and Pressure Vessel Code, Section III)," and F 6-5T, "Welding

Qualifications (Supplement to ASME Boiler and Pressure Vessel Code, Section IX)," were issued as tentative standards. These standards are current for use with the 1971 edition of the Code. References to the RDT material standards and F 3-6T, "Nondestructive Examination," require the issues bearing the date of July 1971 or later.

Work was begun on preparation of a supplement to Subsection NE, Containment Components, of Section III of the ASME Boiler and Pressure Vessel Code. This supplement will be an amendment to RDT E 15-2T.

Reactor Internals

Draft standard E 13-15, "Fuel Assemblies for Pressurized Water Reactors," was reviewed by ORNL, and comments were prepared and submitted to ANC.

Vessels

Draft standard E 10-4, "Pressure Vessel, Class C, Nuclear," was reviewed by ORNL, and comments were prepared and submitted to ANC.

Demineralizers

RDT E 11-1, "Ion Exchanger, Non-Regenerative Type," was revised to conform with the requirements of the 1971 edition of Section III of the ASME Boiler and Pressure Vessel Code and submitted to ANC. Approval of this document will be requested by ANC.

6.3 INSTRUMENTATION AND CONTROLS AND ELECTRICAL STANDARDS

(AEC Activity 04 60 80 03 1)

R. K. Adams
A. E. G. Bates
J. T. DeLorenzo
O. A. Kelly
C. S. Lisser
J. A. Russell
T. L. Hudson

Instrumentation

Drafts of the following standards were completed and distributed for external review and comments.

C 2-3	Thermocouple Time Response Test
C 7-7	Thermocouple Materials, Platinum and 90% Platinum-10% Rhodium Wires, Noninsulated Reference and Standard Grades
C 10-1	Signal Conditioner

Review comments were incorporated into C 7-5, "Thermocouple Material, Chromel-P Vs. Alumel, Magnesium-Oxide Insulated, Stainless Steel Sheathed," and M 3-20, "Polyethylene Instrument Tubing," and the revised documents were submitted to RDT for approval.

Draft standard RDT C 7-6, "Thermocouple Material, Nuclear Grade, Chromel-P Vs. Alumel, Stainless Steel Sheathed, Magnesium Oxide Insulated," was revised. Three quality levels of thermocouples are now covered in this document.

Comments are being evaluated and incorporated into draft standard F 3-11, "Helium Mass Spectrometer Leak Testing for Instruments."

RDT C 15-3, "Current Pulse Preamplifiers for Use with Fission Counters," was revised in accordance with RDT comments.

A meeting was attended in Richland, Washington, to discuss proposed standard C 13-1, "General Purpose, Stored Program, Digital Computer."

The ORNL review of C 17-6, "Cable for Nuclear Radiation Detectors," was completed, and comments were prepared and submitted to HEDL.

Electrical

Work continued on the revision of P 3-1, "Electrical Penetration Assemblies for Nuclear Reactor Containment Structures." This document

is being revised to incorporate additional requirements for FFTF application for electrical penetrations. Review comments are also being evaluated and factored into this version.

RDT draft standard P 4-2, "High-Temperature Trace Heaters, Molded Ceramic, Electrical Type," was completed and is being reviewed in-house.

6.4 PROGRAMMATIC AND PROCEDURAL STANDARDS

(AEC Activity 04 60 80 03 1)

J. W. Anderson
H. G. Arnold
C. A. Burchsted
N. E. Dunwoody
F. L. Hannon
R. M. Holmes
T. W. Pickel
W. C. T. Stoddart
W. W. Goolsby

Quality Assurance

A meeting was attended in Los Angeles, where the Quality Assurance Working Group met to discuss application, interpretation, amendments, and suggested revisions to RDT standards F 2-2T, "Quality Assurance Program Requirements," and F 2-4T, "Quality Verification Program Requirements." The quality assurance requirements of Section III of the ASME Boiler and Pressure Vessel Code, 10 CFR 50, and the ANSI standards were also reviewed in this meeting.

Amendment 1 to RDT F 2-2T, "Quality Assurance Program Requirements," was approved by RDT and issued for trial use. This amendment requires that the approval of the purchaser be obtained for all dispositions of nonconformances which result in either repair or waiver of a requirement, regardless of the magnitude of the nonconformance.

Amendment 1 to F 2-4T, "Quality Verification Program Requirements," was prepared and submitted to RDT for approval. This amendment included a revision to subsection 1.3 concerning the purchase of complex engineered components and an editorial change to subsection 4.

Proposed amendments 2, 3, 4, and 5 to F 2-2T, "Quality Assurance Program Requirements," were prepared and submitted to RDT for approval.

These amendments, which were requested by RDT, will

1. require independent design reviews of engineered components and systems by or for the responsible design organization to verify the adequacy of design;
2. clarify the intent of paragraph 4.10.1, concerning receipt inspection of procured items, and revise paragraph 4.10.2 to emphasize the requirements for contractor review and approval of supplier-generated documents before the acceptance of procured items;
3. require that the contractor employ procedures to verify that the configuration of the purchased item conforms to applicable codes, specifications, drawings, and other contractural requirements, including authorized changes;
4. stipulate that the contractor institute methods for the coordination and interchange of information to the maximum extent practicable on supplier capabilities with respect to products, processes, services, and quality performances in order to eliminate unnecessary and costly duplication of supplier surveys and evaluations.

At the request of RDT, an additional amendment to F 2-2T, paragraph 7.7, is to be proposed by ORNL. This amendment will strengthen the requirements for contractor quality surveillance, particularly during experiment handling and maintenance operations.

Maintenance

Two unresolved comments are being incorporated into the draft of F 4-20, "Operation and Maintenance Manuals." This standard will be submitted to RDT for approval after resolution of these comments.

Cleaning

Amendment 1 to F 5-1, "Cleaning and Cleanliness Requirements for Nuclear Reactor Components," was prepared and submitted for approval. This amendment included a number of substantive comments that had been received since the standard was issued. Consideration for an additional amendment to cover application requirements of FFTF/LMFBR will be evaluated with LMEC following completion of Amendment 1.

Air Cleaning

Data were solicited from major manufacturers of carbons presently being used in iodine adsorption systems. In view of data showing that elemental iodine apparently converts to an organic form when exposed to carbon and oxygen in a radioactive field, RDT M 16-1, "Activated Carbon, for Gas-Phase Iodine Adsorption," will be modified to cover only impregnated carbons. It is intended to provide requirements for three or four grades, based on physical properties of carbons made from various base materials.

Procedural

Amendment 1 to RDT F 1-1, "Preparation of RDT Standards," was prepared and submitted to RDT, approved, and issued for trial use. This amendment provides (1) a revised procedure for processing amendments to approved RDT standards and (2) a new procedure for processing project modifications for special applications to RDT standards.

In response to agreements reached in the June Management Meeting at RDT, the comments and proposals received from several sources for generally updating RDT F 1-1 were evaluated, and a proposed Amendment 2 was prepared and submitted to RDT for approval.

Amendment 3 to F 1-1, which will be an appendix of standard paragraphs to be used in RDT standards, is being prepared.

The following documents, which are for use with the 1971 edition of Section III of the ASME Boiler and Pressure Vessel Code, were issued as tentative standards.

F 3-6T	Nondestructive Examination
F 3-37T	Special Requirements for Metal Products
F 6-5T	Welding Qualifications (Supplement to ASME Boiler and Pressure Vessel Code, Section IX)
F 6-10T	Repair of Materials by Welding

Amendment 1 to F 7-2T, "Preparation for Sealing, Packaging, and Marking of Components for Shipment and Storage," and Amendment 1 to RDT F 7-3T, "Requirements for Identification Marking of Reactor Plant Components and

Piping," were revised to incorporate comments and submitted to RDT for approval.

6.5 MATERIALS AND FABRICATION

(AEC Activity 04 60 80 03 1)

J. W. Anderson
R. O. Carden
R. M. Fuller
F. L. Hannon
O. A. Kelly
J. N. Robinson

The following standards were rewritten to conform to the materials requirements of the 1971 edition of Section III of the ASME Boiler and Pressure Vessel Code and were submitted to RDT for approval. After approval and issue, these documents will supersede the existing issues.

M 1-1	Stainless Steel Covered Welding Electrodes (ASME SFA-5.4 with Additional Requirements)
M 1-3	Mild Steel Covered Welding Electrodes (ASME SFA-5.1 with Additional Requirements)
M 1-4	Low Alloy Steel Covered Welding Electrodes (ASME SFA-5.5 with Additional Requirements)
M 1-5	Welding Rods and Electrodes, Surfacing (AWS A5.13 with Additional Requirements)
M 1-6	Mild Steel Electrodes and Welding Rods (ASME SFA-5.18 with Additional Requirements)
M 1-7	Copper and Copper Alloy Welding Electrodes (ASME SFA-5.6 with Additional Requirements)
M 1-8	Copper and Copper Alloy Welding Rods (ASME SFA-5.7 with Additional Requirements)
M 1-9	Brazing Filler Metal (ASME SFA-5.8 with Additional Requirements)
M 1-10	Nickel and Nickel Alloy Covered Welding Electrodes (ASME SFA-5.11 with Additional Requirements)
M 1-11	Nickel and Nickel-Alloy Bare Welding Rods and Electrodes (ASME SFA-5.14 with Additional Requirements)
M 2-1	Carbon Steel Forgings (ASME SA-105 with Additional Requirements)
M 2-3	Carbon and Alloy Steel Welding Fittings (ASME SA-234 with Additional Requirements)

M 2-4	Alloy Steel Forgings (ASME SA-336 with Additional Requirements)
M 2-7	Carbon and Alloy Steel Forgings, Vacuum Treated (ASME SA-508 with Additional Requirements)
M 2-8	Alloy Steel Forgings (ASME SA-541 with Additional Requirements)
M 3-1	Carbon Steel Seamless Pipe (ASME SA-106 with Additional Requirements)
M 3-4	Nickel Alloy Seamless Tubes (ASME SB-163 with Additional Requirements)
M 3-5	Austenitic Stainless Steel Welded Tubing (ASME SA-249 with Additional Requirements)
M 3-6	Austenitic Stainless Steel Pipe (ASME SA-312 with Additional Requirements)
M 3-7	Austenitic Stainless Steel Welded Pipe, Large Diameter (ASME SA-358 with Additional Requirements)
M 3-11	Carbon and Low Alloy Steel Welded Pipe (ASME SA-155 with Additional Requirements)
M 3-12	Ferritic Alloy Steel Seamless Pipe (ASME SA-335 with Additional Requirements)
M 4-2	Austenitic Stainless Steel Castings (ASME SA-351 with Additional Requirements)
M 5-2	Carbon Steel Plates (ASME SA-516 with Additional Requirements)
M 5-3	Low Alloy Steel Plates (ASME SA-533 with Additional Requirements)
M 5-5	Low Alloy Steel Plates (ASME SA-387 with Additional Requirements)
M 5-7	Nickel-Iron-Chromium Alloy Plate, Sheet and Strip (ASME SB-409 with Additional Requirements)
M 6-1	Alloy Steel Bolting Material for Low Temperature Service (ASME SA-320 with Additional Requirements)
M 6-3	Alloy Steel Bolting Material for High Temperature Service (ASME SA-193 with Additional Requirements)
M 6-4	Alloy Steel Nuts for Bolting for High Temperature Service (ASME SA-194 with Additional Requirements)
M 7-3	Stainless Steel Bars and Shapes (ASME SA-479 with Additional Requirements)

The following documents, also revised for use with the 1971 edition of the ASME Boiler and Pressure Vessel Code, were issued as tentative standards. These revisions supersede the documents issued in February 1969.

M 1-2T	Stainless Steel Welding Rods and Bare Electrodes (ASME SFA-5.9 with Additional Requirements)
M 2-2T	Stainless and Low Alloy Steel Forgings (ASME SA-182 with Additional Requirements)
M 2-5T	Austenitic Stainless Steel Welding Fittings (ASME SA-403 with Additional Requirements)
M 3-2T	Stainless and Alloy Steel Seamless Tubes (ASME SA-213 with Additional Requirements)
M 3-3T	Austenitic Stainless Steel Pipe (ASME SA-376 with Additional Requirements)
M 5-1T	Stainless Steel Plate, Sheet, and Strip (ASME SA-240 with Additional Requirements)

Review comments were incorporated into RDT M 12-3, "Mineral Fiber, Hydraulic-Setting Insulating and Finishing Cement," and the advanced draft of this standard was submitted to RDT for approval. As a result of discussions with RDT to coordinate the requirements of M 12-3 and M 12-2, "Calcium Silicate Block and Pipe Thermal Insulation (ASTM C533 with Additional Requirements)," Amendment 1 to RDT M 12-2 was also prepared and submitted for approval. This amendment will prohibit the addition of leachable sodium compounds other than leachable sodium silicates during the manufacture of insulation and will require the seller to furnish certification of conformance to the standard and the referenced ASTM standard.

M 2-5T, "Austenitic Stainless Steel Welding Fittings (ASME SA-403 with Additional Requirements)," was issued as a tentative standard.

The draft of RDT M 2-15, "Nickel-Chromium Alloy Bars, Forgings, and Forging Stock (ASTM A637 with Additional Requirements)," was revised and submitted to HEDL. RDT approval of this document will be requested by HEDL.

An amendment to RDT M 6-2T, "Mechanical Locking Devices," was prepared to update the standard to Section III of the ASME Boiler and Pressure Vessel Code.

7. GENERAL NUCLEAR SAFETY STUDIES

7.1 SEISMICITY AND SEISMIC RESISTANCE OF NUCLEAR POWER PLANTS

(AEC Activity 04 01 91 01 1)

J. Foster G. D. Whitman

The seismic program at ORNL, supported by the AEC Division of Reactor Development and Technology, is concerned primarily with identifying and developing information required to assure that proper seismic provisions are made in the siting, design, construction, and operation of nuclear power plants. Subcontract activities are carried out in both the earth sciences and the mechanical sciences related to structural dynamics.

Soil Behavior

Completed draft copies of the Shannon and Wilson—Agbabian Jacobsen Associates (SW-AJA) report "Soil Behavior under Earthquake Loading Conditions: State of the Art — Evaluation of Soil Characteristics for Seismic Response Analyses" were received for review and comment. Copies have been sent to AEC offices and ORNL consultants concerned with the seismic program. ORNL will assemble and furnish comments and recommendations to SW-AJA, after which the report will be issued in final form.

The draft of the SW-AJA interim report is essentially completed and will be sent to ORNL for review and comment prior to final issue. This report describes work that has been done on soil analytical models, in-situ test procedures, feasibility of shake table and large-scale simple

shear soil tests, and laboratory liquefaction tests on cohesionless soils. Final copies of the SW-AJA program definition report, revised several months ago, were received from SW-AJA and distributed to AEC offices and to the ORNL seismic program consultants.

Seismic Program Plan

The May 7, 1971, revision of the proposed plan for USAEC Nuclear Safety Seismic Research was sent to a number of firms in the nuclear power plant industry for review and comments. Copies were also sent to members of the National Research Council Committee on Seismology for review and comment, and the first few responses have been received. ORNL will prepare an interpretation of all the comments received for use in further developing the proposed plan.

A draft of the final report, "Simulating Strong Motion Earthquake Effects on Nuclear Power Plants Using Explosive Blasts," on the explosively excited dynamic tests at the EGCR is being prepared by UCLA for distribution and review in accordance with their subcontract with the AEC. This report will contain descriptions of the test programs and the data obtained. It will also contain the results of the analyses of the EGCR site, structures, and equipment to ground shaking accelerations at the site with peak values in the soil exceeding 0.5 g. There is presently no further activity on the earthquake simulation tests at the EGCR, for which a preliminary proposal was submitted earlier this year.

7.2 IGNITION OF CHARCOAL ADSORBERS BY FISSION PRODUCT DECAY HEAT

(AEC Activity 02 40 20 90)

R. A. Lorenz R. P. Shields
R. J. Davis

An experiment is being developed to demonstrate the consequences of heat generated in a charcoal bed due to radioactive decay of adsorbed

iodine. Briefly, the experiment will involve (1) preparation of radioactive ^{130}I by irradiation of ^{129}I in the HFIR in the chemical form of PdI_2; (2) release of ^{130}I as I_2 by heating the PdI_2 to 350°C, the decomposition temperature; (3) transfer of the I_2 to an instrumented charcoal test bed; and (4) observation of bed temperatures and location of adsorbed iodine during experiments at low air flow. Air entering the charcoal bed will be preheated to 100°C. The heat from radioactive decay of the ^{130}I will then be sufficient to heat the charcoal bed and flowing air to temperatures above 300°C, where desorption or ignition can occur. Temperature behavior in the charcoal bed will be compared with the results of the CHART computer program,[1] which calculates the temperature of large-size charcoal adsorbers exposed to reactor accident conditions.

The flow diagram for the experiment has been revised, as shown in Fig. 7.1. A recirculating system completely enclosed in metal will provide positive containment for the experiment in the hot cell. A stainless steel glove box will be modified to form a secondary containment system. To minimize the chance of leakage, the secondary containment box will operate at a pressure lower than the hot cell, and the primary containment loop will be maintained at an even lower pressure.

Provision is being made to add a steam generator and a condenser so that future experiments can be performed with moist air. To reduce plateout and assure high purity of molecular iodine, the line between the PdI_2 decomposition furnace and the charcoal bed will be constructed of quartz rather than stainless steel.

Another design improvement to help reduce iodine plateout and undesirable reactions is the placement of PdI_2 in seven capillary-size quartz capsules (1 mm ID) in the metal irradiation capsule. The capsule shear/furnace combination was tested and found to work satisfactorily. The procedure will be to load the irradiated capsule into the shear (Fig. 7.1) and cap the shear. The hot-cell manipulators will be used to rotate the outer portion of the shear so that the tips of the metal capsule and the seven quartz capillary capsules will be sheared off and the capillary capsules will be located in front of the air outlet line. After sealing the loading port in the secondary containment box, the shear/furnace combination will be electrically heated above 350°C to decompose the PdI_2.

ORNL-DWG 71-10209

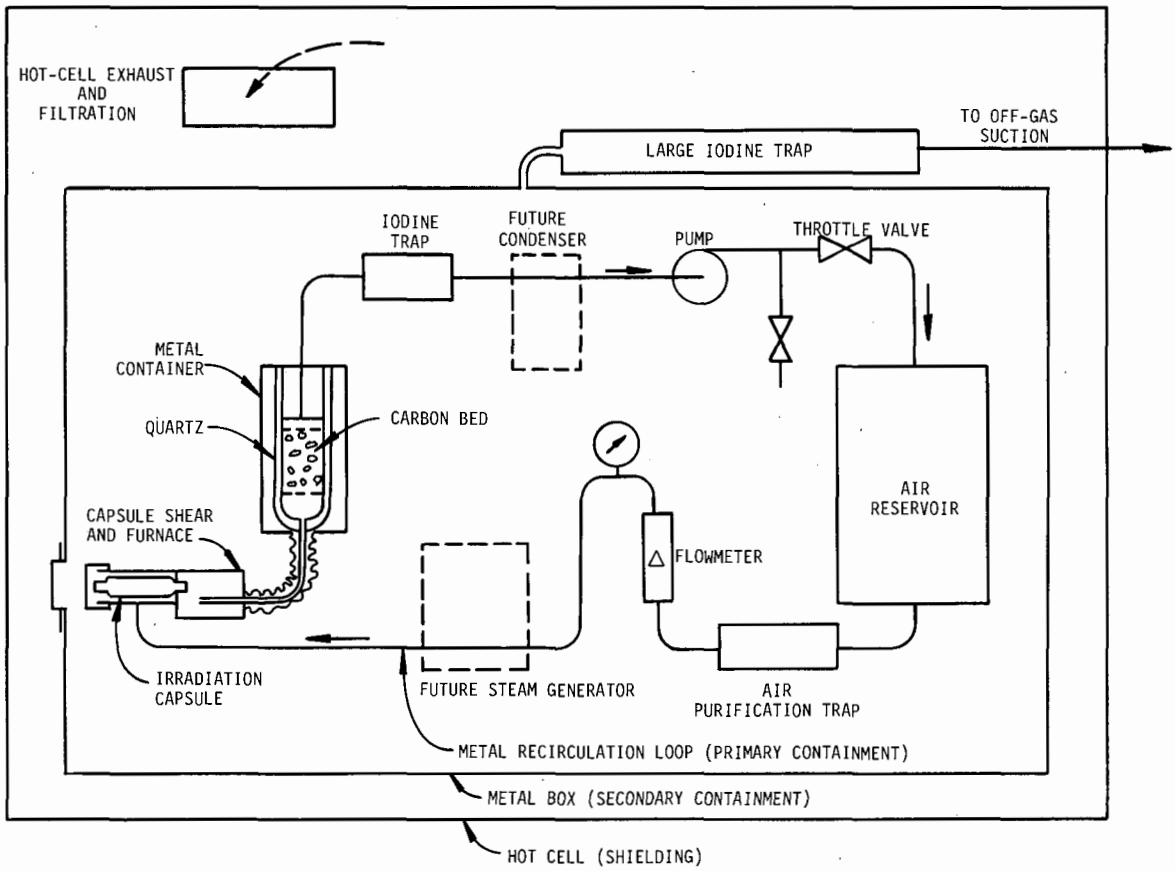

Fig. 7.1. Radioactivity effects on iodine adsorption.

Laboratory experiments to determine the efficiency of decomposition and iodine transport are in progress.

Reference

1. R. P. Shields and M. Siman-Tov, *The Effect of Iodine Decay Heat on Charcoal Adsorbers*, ORNL-4602 (January 1971).

8. NUCLEAR SAFETY INFORMATION

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

J. R. Buchanan Wm. B. Cottrell

The Nuclear Safety Information Center (NSIC), established in 1963 by the USAEC Division of Reactor Development and Technology, collects, evaluates, and disseminates nuclear safety information to governmental agencies, research and educational institutions, and the nuclear industry. The Center's basic computer file now contains references on more than 56,700 documents.

Computer Programming and Processing

Recently installed IBM-2314 disks at the Computing Technology Center are being used to improve the performance of the retrospective search program. Previously, magnetic tapes were used for batch sorting. However, with the availability of large-capacity disks, the NSIC reference files can now be sorted without mounting tapes for this purpose. Other program changes are being made to further improve system performance through greater use of these disks.

A new product created during this reporting period is the System/360 Key Words in Context (KWIC) listing of NSIC document titles. These indexes had been prepared on the IBM 7090.

SDI and PPIF Programs

During the period 34 new participants were added to the SDI (Selective Dissemination of Information) program and 13 were removed, bringing the total of users to 1967. Twenty-three existing profiles were revised.

One new participant was added to the PPIF (Program and Project Information File) program, bringing the total to 244, and three existing profiles were revised.

Meetings

E. W. Hagen of the NSIC staff attended the ANS Reactor Operations Topical Meeting, Denver, Colorado, August 8–11, and presented a paper entitled "Reactor Availability and Station Reliability for Continuity of Service." While in Denver, he also attended the AEC Reactor Safety Conference on August 12 and discussed the topic "Experience to Date on the Application of C16-1T/IEEE-279 to AEC-Owned Reactors."

Bibliographies and State-of-the-Art Reports

The reports Nuclear Reactor Siting Practice and Its Relationship to Population, ORNL-NSIC-66; Calculation of Doses Due to Accidentally Released Plutonium from an LMFBR, ORNL-NSIC-74; and Safety-Reliability of Reactor Systems, ORNL-NSIC-83, are all undergoing revision. Reports published since the last reporting period are ORNL-NSIC-48, Information Scanning and Processing at the Nuclear Safety Information Center, by Celia Murphy and Carol Julian; ORNL-NSIC-85, Compilation of U.S. Nuclear Standards, 7th ed., 1970, by J. P. Blakely (compiler); ORNL-NSIC-86, Index to *Nuclear Safety*, A Technical Progress Review by Chronology, Permuted Title, and Author, Vol. 1(1), through Vol. II(6), by J. P. Blakely and Ann Klein; ORNL-NSIC-87, Safety-Related Occurrences in Nuclear Facilities as Reported in 1969, by R. L. Scott and W. R. Casto; ORNL-NSIC-89, Biological Concentration and Turnover of Radionuclides in Food Chains: A Bibliography, by D. E. Reichle, D. J. Nelson, and P. B. Dunaway; ORNL-NSIC-92, NSIC Computerized Information Techniques, by Wm. B. Cottrell and J. R. Buchanan; ORNL-NSIC-95, Indexed Bibliography on Effects of Radionuclides and

Ionizing Radiation on Ecological Systems, by G. U. Ulrikson, C. D. Bopp, and R. M. Carroll.

Summary of Information Activities

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

Accessions added to the computer file	1590
Technical inquiries (letter or telephone)	137
Specific NSIC reports and other nontechnical inquiries	69
Additions to NSIC reports distribution	5
Visits for consultation or use of reference material	14
Additions to SDI program	34
Additions to PPIF program	1

8.2 TECHNICAL PROGRESS REVIEW *NUCLEAR SAFETY*

(AEC Activity 14 02 01)

J. P. Blakely Wm. B. Cottrell

Nuclear Safety is a Technical Progress Review that is prepared and edited at ORNL on a bimonthly schedule under AEC-DTI contract. The July–August 1971 issue (Vol. 12, No. 3, 119 pages, distributed August 15) contained articles by 2 authors from outside ORNL and 11 authors from ORNL, including 4 *Nuclear Safety* staff members and 3 Nuclear Safety Program staff members.

The lead article, by D. N. Hess (NSIC staff), from which the cover was adapted, was one in the Nuclear Power in Perspective series and dealt with the "Plight of the Benign Giant." Premised on the assumption that the public press is both a molder of opinion and a reflector of public interest and critique, nearly 800 items in the daily and periodical press pertaining to the nuclear industry and covering a period of about one year were examined for their philosophical and psychological impact on

the reader. Accordingly, this survey is a retrospective assessment in order of priority of the principal controversial issues confronting the nuclear community. It is hoped that, from this work, nuclear advocates and allied interests may find a firmer sense of direction and significant areas where special attention can be most profitably devoted to enable the public to feel at ease in the presence of the energy giant.

Other articles dealt with nuclear liability insurance and a brief history reflecting the success of nuclear safety (by Joseph Marrone, General Manager of the Nuclear Energy Liability Insurance Association); basic studies of fission-gas effects in reactor fuels (by R. M. Carroll, ORNL-RCD); risk minimization by optimum allocation of resources available for risk reduction (by Joseph B. Rivard, Sandia); the 1970 joint symposium of the IEEE power and nuclear science groups (by E. W. Hagen, *Nuclear Safety* section editor); ecological aspects of ^{3}H behavior in the environment (by J. W. Elwood, ORNL-HPD); the IAEA-WHO symposium on handling of radiation accidents (by P. S. Rohwer, ORNL-HPD); the regular current-events-type features by *Nuclear Safety* staff members W. R. Casto, A. W. Savolainen, and J. P. Blakely; and a new feature on legislative and regulatory activities affecting the environmental quality of nuclear facilities, by Nuclear Safety Program staff members R. H. Bryan, B. L. Nichols, and J. N. Ramsey. This was the first issue to carry a regular editorial-type feature on the Contents overleaf.

The special Geneva Issue of *Nuclear Safety*, September—October 1971, was printed and shipped to Switzerland in July but is not available for release until September. It will be discussed in the next bimonthly report.

Draft material for the November—December 1971 issue, Vol. 12, No. 6, was distributed for external review; comments are due in late August, with distribution scheduled for early November.

Draft articles for the January—February 1972 issue are being received and reviewed internally prior to distribution for external review.

Subsequent issues through July—August 1972 are in various stages of article assignment, receipt, and outline.