ORNL/TM-6698

Breeder Reactor Safety and Core Systems Programs Progress Report for July–September 1978

M. H. Fontana
J. L. Wantland

[redacted]

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BREEDER REACTOR SAFETY AND CORE SYSTEMS PROGRAMS PROGRESS REPORT FOR JULY—SEPTEMBER 1978

M. H. Fontana J. L. Wantland

Date Published: February 1979

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Prepared by the
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PREVIOUS BRP SAFETY PROGRESS REPORTS*

Prior information on these projects was presented in the following:†

Period Covered	Report No.
September—October 1972	ORNL/TM-4075
November—December 1972	ORNL/TM-4088
January—February 1973	ORNL/TM-4148
March—April 1973	ORNL/TM-4261
May—June 1973	ORNL/TM-4331
July—September 1973	ORNL/TM-4417
October—December 1973	ORNL/TM-4505
January—March 1974	ORNL/TM-4630
April—June 1974	ORNL/TM-4727
July—September 1974	ORNL/TM-4776
October—December 1974	ORNL/TM-4877
January—March 1975	ORNL/TM-4980
April—June 1975	ORNL/TM-5076
July—September 1975	ORNL/TM-5197
October—December 1975	ORNL/TM-5431
January—March 1976	ORNL/TM-5513
April—June 1976	ORNL/TM-5699
July—September 1976	ORNL/TM-5753
October—December 1976	ORNL/TM-5785
January—March 1977	ORNL/TM-5940
April—June 1977	ORNL/TM-6020
July—September 1977	ORNL/TM-6158
October—December 1977	ORNL/TM-6288
January—March 1978	ORNL/TM-6439
April—June 1978	ORNL/TM-6558

*Reported previously as LMFBR Safety and Core Systems Program Progress Reports.

†Information on the Breeder Reactor Project for 1968 through August 1972 is available in ORNL Nuclear Safety Research and Development Program Bimonthly Progress Reports.

SUMMARY

1. Program Coordination

Coordination was maintained with the Department of Energy, Division of Reactor Research and Technology (DOE-RRT), and the national breeder reactor safety community.

2. THORS Analysis and Data Evaluation

A description and tabulation of 23 THORS Bundle 3C boiling tests are presented. Typical transient temperatures of one run are shown in 2-D and 3-D graphs. Preliminary posttest analysis indicates that, during 8 1/2 min of boiling (maximum continuous boiling of 28 sec), no dryout and no bulk flow instability occurred.

The Safety Analysis Section (SAS) 3A computer code was used to study the interaction between two heated bundles operated hydraulically in parallel with a bypass representing many nonboiling channels. These results indicate that the parallel channels are strongly coupled if unequal impedance to the inlet flow of one channel exists. If both channels have equal inlet impedance, then the coupling is negligible and each channel behaves as if it were a single channel in parallel with a bypass.

3. THORS Facility Operation

The test facility has operated for a total of 26,400 hr. Test bundles have been operated at varying power levels for ∿2300 hr; sodium boiling tests have been conducted in four bundles.

The boiling propagation tests for Bundle 3C have been completed. The facility has been shut down for modification and for installation of test Bundle 9, a 61-pin bundle that is of prototypic length and configuration of a CRBR fuel subassembly. A new Hastelloy-X test section housing designed to operate up to 1000°C (1800°F) will be installed during the modification period. The piping to the new 2.0-MW heat dump will also be installed, giving the facility full 2.0-MW(e) input and heat-removal capability.

Assembly of Bundle 9 is 90% complete. Photographs taken during the bundle assembly are included.

4. Sodium Boiling Test Facility (SBTF)

Boiling conditions were first achieved in the SBTF on August 11, 1978. Approximately 2 hr of boiling time at test section powers between 0.4 and 1.0 kW were logged before distortion of the test section forced shutdown for repair. A modified test section design has been installed, and testing will resume, pending DOE approval.

5. Material for Fuel Pin Simulation

There was no activity in this area during this reporting period.

6. Neutronics of a Disrupted Core

Work on the SPQR quasi-static Monte Carlo Code has been halted. A new, modified, probability table method has been developed and used for calculating bubble collapse reactivities. The adjoint version of the MELT computer code is being debugged and tested for use in the safety sensitivity studies.

7. Environmental Assessment of Alternate FBR Fuels

Two reports were submitted during this period for publication as ORNL technical reports.

Sites for weather monitoring equipment in the Lemhi Pass thorium resource region were selected following an aerial and ground survey.

Assembly of a tube furnace, glove box, and auxiliaries is nearly complete, and instrument calibration is being done preparatory to starting quantitative studies of burning properties of thorium-based FBR fuel. The objective of this work is to determine source terms appropriate to fuel-burning accident conditions.

8. Model Evaluation of Breeder Reactor Radioactivity Releases

A method of determining atmospheric stability based on the standard deviation of the wind direction at the point of release is a more accurate predictor of normalized air exposure via the Turner Workbook than stabilities based on vertical temperature gradients. A model developed by Chamberlain for the fraction of radioactivity deposited by the atmosphere which is retained on vegetation has been compared with field measurements. A model for the transport of ^{131}I in the pasture-cow-milk pathway has been compared with data from the Dresden and Monticello nuclear power plants.

9. Nuclear Safety Information Center

The NSIC serves the nuclear community through the collection, analysis, and dissemination of relevant safety information. During the report period July—September 1978, NSIC processed 2748 documents, responded to 230 inquiries, serviced 377 SDI subscribers, and provided a variety of other services, as well as pursuing in-house work on several reports and the technical progress review, *Nuclear Safety*. Three reports were issued during the quarter, and work is continuing or was initiated on several other reports. Work was also undertaken on the material for several issues of *Nuclear Safety*, as required by its publication schedule.

10. Central Computerized Data Base for Breeder Reactor Safety Codes

Version I of SACRD has been reviewed. Based on these reviews, updates are presently being incorporated into SACRD. The minutes of the Fifth Biannual SADCG Meeting were distributed.

11. Breeder Reactor Reliability Data Analysis Center

To date, less than half the reviewers of the CREDO specifications proposal have replied. Contacts have been made with Byron Jackson, Babcock and Wilcox, and Allis-Chalmers. Data collection and evaluation of

historical data on sodium components continue. The data base management system is being defined.

GLOSSARY OF ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
AI	Atomics International, Division of Rockwell Corporation
CREDO	Centralized Reliability Data Organization
CRBR	Clinch River Breeder Reactor
DAS	Data Acquisition System
DISSPLA	Display Integrated Software System and Plotting Language
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor — II
EOS	Equation of state
ETEC	Energy Technical Engineering Center (formerly LMEC)
FBR	Fast Breeder Reactor
FFTF	Fast-Flux Test Facility
FOP	First-order perturbation
FPS	Fuel pin simulator
FTS	Federal Telecommunications System
GA	General Atomic Corporation
GE	General Electric Corporation
HEDL	Hanford Engineering Development Laboratory
HPS	Health Physics Society
HTGR	High-Temperature Gas-Cooled Reactor
INEL	Idaho National Engineering Laboratory
LASL	Los Alamos Scientific Laboratory
LMEC	Liquid Metal Engineering Center (now ETEC)
LMFBR	Liquid-Metal Fast Breeder Reactor
LOA	Lines of assurance
LOPI	Loss-of-pipe integrity
NCSR	National Center of Systems Reliability (UK)
NPRDS	Nuclear Power Reliability Data Service
NRC	Nuclear Regulatory Commission
NSIC	Nuclear Safety Information Center
NSMH	Nuclear Systems Materials Handbook
ORCHIS	Oak Ridge Computerized Hierarchical Information System
ORNL	Oak Ridge National Laboratory

PLSA	Properties for LMFBR safety analysis
PMD	Pressure measuring device
RECON	Remote Console Bibliographic Retrieval System (DOE)
RRT	Reactor Research and Technology (Division of)
RTD	Resistance temperature device
SACRD	Safety Analysis Computerized Reactor Data
SADCG	Safety Analysis Data Coordinating Group
SBTF	Sodium Boiling Test Facility
SDI	Selective Dissemination of Information
THORS	Thermal-hydraulic out-of-reactor safety
TIC	Technical Information Center
TOP	Transient overpower
TREAT	Transient Reactor Test Facility
TUC	Transient undercooling
TUCOP	Transient undercooling and overpower
UKAEA	United Kingdom Atomic Energy Authority
USAEC	United States Atomic Energy Commission
WARD	Westinghouse Advanced Reactor Division

Computer Codes

AIRDOS II	Air Dose
ANISN	Anisotropic SN
COBRA	Coolant Boiling in Rod Arrays
KENO	None (Monte Carlo Code)
MELT	None (neutronic thermal-hydraulic code)
SABRE	Subchannel Analysis of Blockages in Reactor Elements
SPQR	Stochastic Program for Quasistatic Research

BREEDER REACTOR SAFETY AND CORE SYSTEMS PROGRAMS PROGRESS REPORT FOR JULY—SEPTEMBER 1978

M. H. Fontana J. L. Wantland

ABSTRACT

This report covers progress made during July—September 1978 by the Breeder Reactor Safety and Core Systems Programs at ORNL. Topics discussed are program coordination, THORS analysis and data evaluation, THORS Facility operation, SBTF, material for fuel pin simulation, neutronics of a disrupted core, environmental assessment of alternate FBR fuels, model evaluation of breeder reactor radioactivity releases, NSIC, central computerized data base for breeder reactor safety codes, and breeder reliability data analysis center.

The completed THORS Bundle 3C (8% of flow area blocked) boiling tests are summarized, and typical data are presented. Preliminary analysis indicates that no dryout and no bulk flow instability occurred during sodium boiling.

The SAS 3A computer code was used to study the interaction between two heated bundles operated in parallel with a bypass flow channel. These results indicate that bundle interaction depends on bundle inlet flow impedance.

The THORS Facility has operated for a total of 26,400 hr. Test bundles have been operated at varying power levels for ∿2300 hr; sodium boiling tests have been conducted in four bundles.

The boiling propagation tests for Bundle 3C have been completed. The facility has been shut down for modification and for installation of the 61-pin Bundle 9. The piping to the new 2.0-MW heat dump will also be installed, giving the facility a full 2.0-MW(e) input and heat-removal capability. Assembly of Bundle 9 is 90% complete.

The SBTF became operational during this reporting period. Approximately 2 hr of boiling time was logged before distortion of the test section forced shutdown for refurbishment.

Work on the quasi-static Monte Carlo Code SPQR has been halted. Present efforts in this task are directed to safety sensitivity analysis.

Two reports on environmental assessment of alternate FBR fuels were submitted during this period for publication as ORNL technical reports.

Sites for weather monitoring equipment in the Lemhi Pass thorium resource region were selected following an aerial and ground survey.

Assembly of a tube furnace, glove box, and auxiliaries is nearly complete, and instrument calibration is being done preparatory to starting quantitative studies of burning properties of thorium-based FBR fuel.

Output from a number of environmental transport models is being compared with field data.

During this report period, the NSIC processed 2748 documents, responded to 230 inquiries, and serviced 377 SDI subscribers. Three reports were issued, and work was initiated or is continuing on several other reports. Work was also undertaken on the material for several issues of *Nuclear Safety*, as required by its publication schedule.

Version I of SACRD has been released to reviewers, and their comments are presently being incorporated.

Continued progress is reported on the development of a centralized data base for collection, storage, and dissemination of breeder reactor component reliability data. Primary emphasis currently is placed on the collection and evaluation of existing data.

1. PROGRAM COORDINATION

M. H. Fontana

The Breeder Reactor Safety and Core Systems Program at ORNL, funded by DOE-RRT, comprises seven tasks: (1) the THORS Facility (189a OH044), (2) neutronics of a disrupted core (189a OH069), (3) environmental assessment of alternate FBR fuels (189a OH1488), (4) model evaluation of breeder reactor radioactivity releases (189a OH136), (5) NSIC (189a OH020), (6) central data base for breeder reactor safety codes (189a OH099), and (7) breeder reactor reliability data analysis center (189a OH143).

The THORS sodium-flow facility has the capability of simulating breeder reactor fuel subassemblies using electric cartridge heaters. Prototypic configuration, power density, pressure, specific flow, and temperature are obtained. The objectives of the program include (1) investigation of flow and temperatures within fuel rod assemblies under normal and accident conditions, such as partial flow blockages and flow-power mismatches; (2) detection of events that could lead to failure; (3) loop system dynamics; and (4) extrapolation of experimental results to predict the behavior of full-size reactors under hypothetical accident conditions.

The purposes of the neutronics analysis of a disrupted core task are to provide a rigorous assessment of the criticality of a disrupted breeder reactor core using 2-D and 3-D techniques, to provide an uncertainty margin for the reactivity calculation, and to develop a transient Monte Carlo code with the appropriate feedback necessary for accident analysis that will interface with existing thermal-hydraulic codes.

The objective of the environmental assessment of the alternate FBR fuels task is to determine the environmental consequences of substituting carbide, nitride, or carbonitride fuel for reference oxide fuel in breeder reactors.

The model evaluation of breeder reactor radioactivity releases task is to identify, evaluate, and recommend models and parameters that are best suited for predicting individual and population exposures resulting from routine and accidental breeder reactor radioactive discharges and to verify models to the extent possible.

The NSIC disseminates safety information on the breeder and other reactor programs to (1) the nuclear community through reports and the bimonthly publication *Nuclear Safety* and (2) individuals through SDI, searches, and consultation.

The central data base for the breeder reactor safety codes task encompasses the definition, development, and maintenance of a central computerized data base for breeder reactor safety computer codes.

The purposes of the breeder reactor reliability data analysis center are to define, develop, and maintain a reliability analysis center for use in breeder reactor safety and licensing and to develop the methodology necessary to process the data for validated and accurate reliability analysis.

During this reporting period, coordination was maintained with DOE-RRT and the national breeder reactor safety community.

1.1 THORS and SBTF

Meetings were held with J. Dickson and S. D. Rose of the UKAEA Winfrith Laboratories on July 12–14, 1978, to discuss possible areas of

cooperation between the THORS Program and the UKAEA on sodium-boiling research and development (R&D).

Arrangements were made with Oak Ridge Associated Universities (ORAU) for assigning an Iranian attaché to the THORS Program.

Contributions were made to a joint U.S. paper on sodium boiling R&D to be presented at the European Nuclear Society (ENS) Meeting on Reactor Safety, held in Brussels October 12–14, 1978.

A Quality Assurance (QA) audit meeting was held with CRBR and DOE Headquarters representatives on August 7, 1978. No deficiencies were found.

A review held in Germantown on August 9, 1978, of the Argonne National Laboratory (ANL) Sodium Loop Safety Facility (SLSF) P-4 experiment was attended by THORS staff.

A review of the THORS Program with A. Opdenaker of RRT-HQ was held at ORNL on August 10, 1978.

M. H. Fontana attended Behaviour of Sodium (BONA) Task Force Meetings held at Massachusetts Institute of Technology (MIT) on August 11, 1978, and on September 21, 1978.

D. L. Clark and P. A. Gnadt met with GE personnel at Sunnyvale and Oakland, California, to discuss the 600-gpm electromagnetic sodium pump delivery on August 30, 1978.

Descriptions of THORS and SBT facilities were sent to M. Oren of HEDL for inclusion in a National Directory of Breeder Reactor Program facilities.

Communication was maintained with Foster Wheeler Energy Company and ANL on the subject of Intermediate Heat Exchanger (IHX) tests and Component Mixing (COMMIX) computer code validation. R. J. Ribando attended a meeting at ANL on the same subject on September 15, 1978.

T. A. Shih of GE Advanced Reactor Systems Department met with the THORS staff at ORNL on September 18, 1978, to discuss the GE sodium boiling (SOBOIL) code and relevant THORS sodium-boiling results.

Discussions were held with M. Kazimi of MIT on the possibility of the use of ORNL computer facilities by the MIT DOE-funded Sodium Boiling Analysis Project.

Work on identifying sodium-boiling R&D needs related to LOA-2 requirements and event sequences began.

1.2 Neutronics of Disrupted Core

"Analysis of Neutronics and Reactivity Effects Associated with Severe Core Disruptive Accidents" — E. T. Tomlinson and G. F. Flanagan attended a meeting at DOE Headquarters on September 15, 1978, in regard to zero power reactor-9 (ZPR-9) experimental analysis.

"Safety Analysis Sensitivity Studies" — E. T. Tomlinson and C. V. Parks visited the Reactor Analysis and Safety (RAS) Division at ANL on September 5–8, 1978, to discuss the applicability of generalized perturbation methods to Fission Gas Release and Swelling (FRAS2) and Prediction of Molten Fuel Behavior under Top Conditions (PLUTO2).

1.3 Environmental Assessment of Alternate FBR Fuels

On July 19, 1978, E. S. Bomar, H. R. Meyer, and V. J. Tennery met with G. E. Start, National Oceanic and Atmospheric Administration (NOAA) Office in Idaho Falls and made an aerial survey of the Lemhi Pass thorium resource site. This was followed by a ground-level trip into the same area with L. Peterson of Idaho Falls. The information forwarded in the letter to I. Van der Hoven was obtained from this field trip.

1.4 Model Evaluation of Breeder Reactor Radioactivity Releases

On July 27, 1978, F. O. Hoffman and C. W. Miller met with members of the Environmental Planning staff of the Tennessee Valley Authority (TVA). This meeting was to discuss possible cooperative programs in the area of model validation.

F. O. Hoffman has been invited by the Fachverban für Strahlenschutz, Federal Republic of Germany (FRG), to present a paper for the plenary session of the group's Twelfth Annual Symposium, "Radiation and Environment." The paper, to be entitled "An Evaluation of the Uncertainties of Radioecological Models," will be co-authored by C. W. Miller and

D. L. Shaeffer. Travel expenses and per diem during the conference will be paid by the Fachverband.

D. L. Shaeffer and C. A. Little gave a presentation entitled "Effects of the Ratio of Two Measured Environmental Variables with Truncated Lognormal Frequency Distributions" at the mathematical ecology section of the Ecological Society of America meeting in Athens, Georgia, on August 22, 1978.

1.5 Nuclear Safety Information Center

The NSIC has processed 2748 documents, responded to 230 inquiries, serviced 377 SDI subscribers, and pursued in-house work on several reports and on the technical progress review, *Nuclear Safety*.

W. B. Cottrell attended a Bund Minister for Research and Technology (BMFT) Experts Meeting on Reactor Safety Research, Bonn, FRG, August 31–September 1, 1978, and an International Atomic Energy Agency (IAEA) Specialists' Meeting on Power Supply Arrangements for Nuclear Power Plants, Stockholm, Sweden, September 5–8, 1978.

1.6 Central Data Base for Breeder Reactor Safety Codes

Version I of SACRD is being updated. The minutes of the Fifth Biannual SADCG Meeting were distributed.

1.7 Breeder Reactor Reliability Data Analysis Center

In addition to meeting with FFTF staff, one staff member, T. F. Bott, met with NCSR members in Culcheth, United Kingdom. In addition to discussing numerous problems with development and operation of a large reliability data effort, participants also received NCSR data on sodium components. T. F. Bott attended the Fifth Symposium on Reliability Technology at the University of Bradford, where he presented a paper on CREDO operation and initial sodium valve data that have been collected.

1.8 Reports and Major Correspondence Issued

1.8.1 THORS and SBTF

Comments on Proposed SLSF P-4 Test, letter, M. H. Fontana to D. Lennox (July 13, 1978).

Comments on June 1978 Draft of "LMFBR Safety and Sodium Boiling — A State-of-the-Art Report," letter, M. H. Fontana to W. D. Hinkle (July 25, 1978).

Comments on Safe Shutdown Reliability Program for Large Breeders, letter, M. H. Fontana to J. D. Griffith (July 27, 1978).

Data transmittal of Bundle 3C boiling results sent to Westinghouse, Advanced Reactors Division, fulfilling a requirement of Section 10 of Development Requirement Specification (DRS) 02.01 (Rev. 3) (Aug. 14, 1978).

J. T. Han, *A Fluid Mechanics Model to Estimate the Leakage of Incompressible Fluids Through Labyrinth Seals*, ORNL/TM-6373 (August 1978).

THORS Bundle 9, Phase 1 — Steady-State Test Program was transmitted for external review on Sept. 19, 1978.

P. W. Garrison, *SIMBO — A Simple Boiling Model of the Response of a Simulated Sodium-Cooled Breeder Reactor Subassembly to an Undercooling Transient*, ORNL/TM-6343 (September 1978).

1.8.2 Neutronics of disrupted cores

C. V. Parks, "Derivation and Implementation of Sensitivity Theory to MELT-IIIA," Intralaboratory Memo (Sept. 29, 1978).

1.8.3 Environmental assessment of alternate FBR fuels

Letter from V. J. Tennery to I. Van der Hoven (NOAA) describing field trip to Lemhi Pass thorium resource region and giving specific coordinates for candidate locations for a mine near the continental divide and a mill on the eastern watershed of the Beaverhead Mountains (July 27, 1978).

V. J. Tennery et al., *Environmental Assessment of Alternate FBR Fuels: Radiological Assessment of Reprocessing and Refabrication of Thorium/Uranium Carbide Fuel*, ORNL/TM-6493 (August 1978).

The following papers have been accepted for presentation at the Conference on Radioactivity and the Environment, Norderney, FRG, October 2–6, 1978:

S. V. Kaye and J. E. Till, "Environmentally Important Radionuclides in Nonproliferative Fuel Cycles."

J. E. Till and G. G. Killough, "Scenarios for ^{14}C Release to the Atmosphere by the World Nuclear Industry and Estimated Radiological Impacts."

1.8.4 Model evaluation of breeder reactor radioactivity releases

C. W. Miller, "A Critique of the Determination Atmospheric Stability Categories for Assessing Airborne Releases of Radionuclides," *Health Phys.* 34, 489–92 (May 1978).

C. W. Miller, "An Examination of Gaussian Plume Dispersion Parameters for Rough Terrain," *Atmos. Environ.* 12, 1359–64 (1978).

C. W. Miller, "An Application of the ATDL Simple Dispersion Model," *J. Air Poll. Contr. Assoc.* 28(8), 798–800 (August 1978).

C. W. Miller, F. O. Hoffman, and D. L. Shaeffer, "The Importance of Variations in the Deposition Velocity Assumed for the Assessment of Airborne Radionuclide Releases," *Health Phys.* 34(6), 730–34 (June 1978).

D. L. Shaeffer, "Modelling Dresden/Monticello Data for ^{131}I Transport from Pasture to Milk," accepted for publication in *Trans. Am. Nucl. Soc., 1978 Winter Meeting*, Washington, D.C. (November 1978).

The following papers have been accepted for presentation at the Conference on Radioactivity and Environment, Norderney, FRG, October 2–6, 1978:

F. O. Hoffman, D. L. Shaeffer, and C. W. Miller, "An Evaluation of Uncertainties of Radioecological Models."

C. W. Miller and F. O. Hoffman, "A Critique of Methods for Estimating Plume Depletion and Deposition of Airborne Radionuclides."

D. E. Dunning, Jr. and G. G. Killough, "Internal Radiation Dose Calculations with the INREM-II Computer Program."

H. R. Meyer and J. E. Till, "Anticipated Radiological Impacts from the Mining and Milling of Thorium for the Nonproliferative Fuels."

1.8.5 Nuclear Safety Information Center

J. R. Buchanan and G. W. Keilholtz, *HTGR Safety 1: Review of Current Issues and Bibliography of Literature, 1960–1977*, ORNL/NUREG/NSIC-128 (NUREG/CR-0214).

W. B. Cottrell, Foreign Trip Report, ORNL/FTR-436 (Sept. 12, 1978).

E. W. Hagen, Foreign Trip Report, ORNL/FTR-457 (Sept. 25, 1978).

2. THORS ANALYSIS AND DATA EVALUATION

J. L. Wantland
N. E. Clapp
J. F. Dearing
P. W. Garrison
J. T. Han
N. Hanus
G. A. Klein
R. H. Morris
W. R. Nelson
R. J. Ribando

2.1 Boiling Tests in THORS Bundle 3C

N. Hanus

Analyses have begun on the boiling tests recently completed with THORS Bundle 3C. The objective of this test series was to evaluate the boiling stability in the wake zone downstream of a six-channel central blockage. Preliminary analysis indicates that no dryout and no bulk flow instability occurred in 23 boiling runs in which there was continuous boiling for up to 28 sec.

Bundle 3C is composed of 19 electrically heated, 5.83-mm-diam (0.230-in.) FPSs surrounded by 12 heated outer edge FPSs and a scalloped duct wall. The FPSs are separated from each other and from the duct wall by 1.42-mm-diam (0.056-in.) helically wrapped wire spacers on a 305-mm (12.0-in.) pitch. The unheated stainless steel blockage plate, 6.4 mm (0.25 in.) thick, blocks the six central subchannels (8.1% of total flow area) at 381 mm (15.0 in.) from the start of the heated section. The heated section of Bundle 3C is 533 mm (21 in.) long with a uniform axial power distribution. The unheated simulated fission gas plenum (SFGP), located downstream of the heated section, is 1.6 m (62 in.) long. Figures 2.1 and 2.2 show the instrumentation layouts for the heated section and the SFGP.

This test bundle is an extension of a previous blocked bundle (Bundle 3B)[1] whose 19 inner pins were identical to the present bundle but whose 12 outer edge pins were unheated and were set flush against the scalloped duct wall.

Table 2.1 summarizes the 25 tests conducted, 23 of which were thought to have produced boiling. (Test 20, Runs 101 and 231 did not produce boiling.) Total boiling time is estimated at ∿8 1/2 min, with continuous boiling ranging from 18 (Run 500) to 28 sec (Run 531).

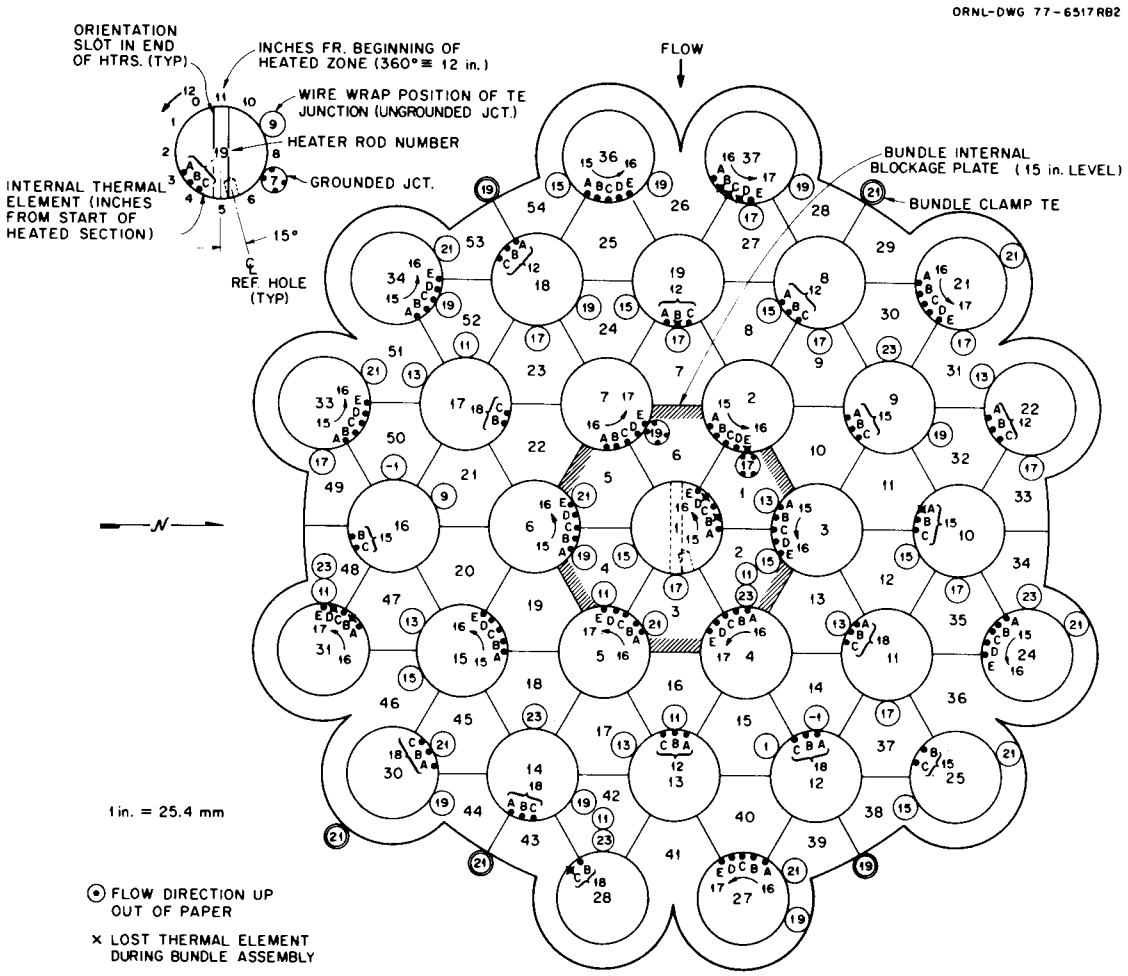

Fig. 2.1. THORS Bundle 3C heater pin bundle instrumentation.

At each setting of edge pin power in Table 2.1 (i.e., 100%, 80%, etc., of central pin power), a higher run number indicates an increased boiling intensity in the blockage wake and an increased tendency for boiling propagation. The run sequence for a particular edge pin power setting was terminated when, in the immediate posttest analysis, there was evidence that the saturation temperature in any of the unblocked flow channels was reached either because the bulk temperature was sufficiently near the saturation temperature or because there may have been radial boiling propagation. Run 101 (Test 21) was an unsuccessful attempt to produce dryout and therefore constitutes the most severe conditions imposed on the bundle.

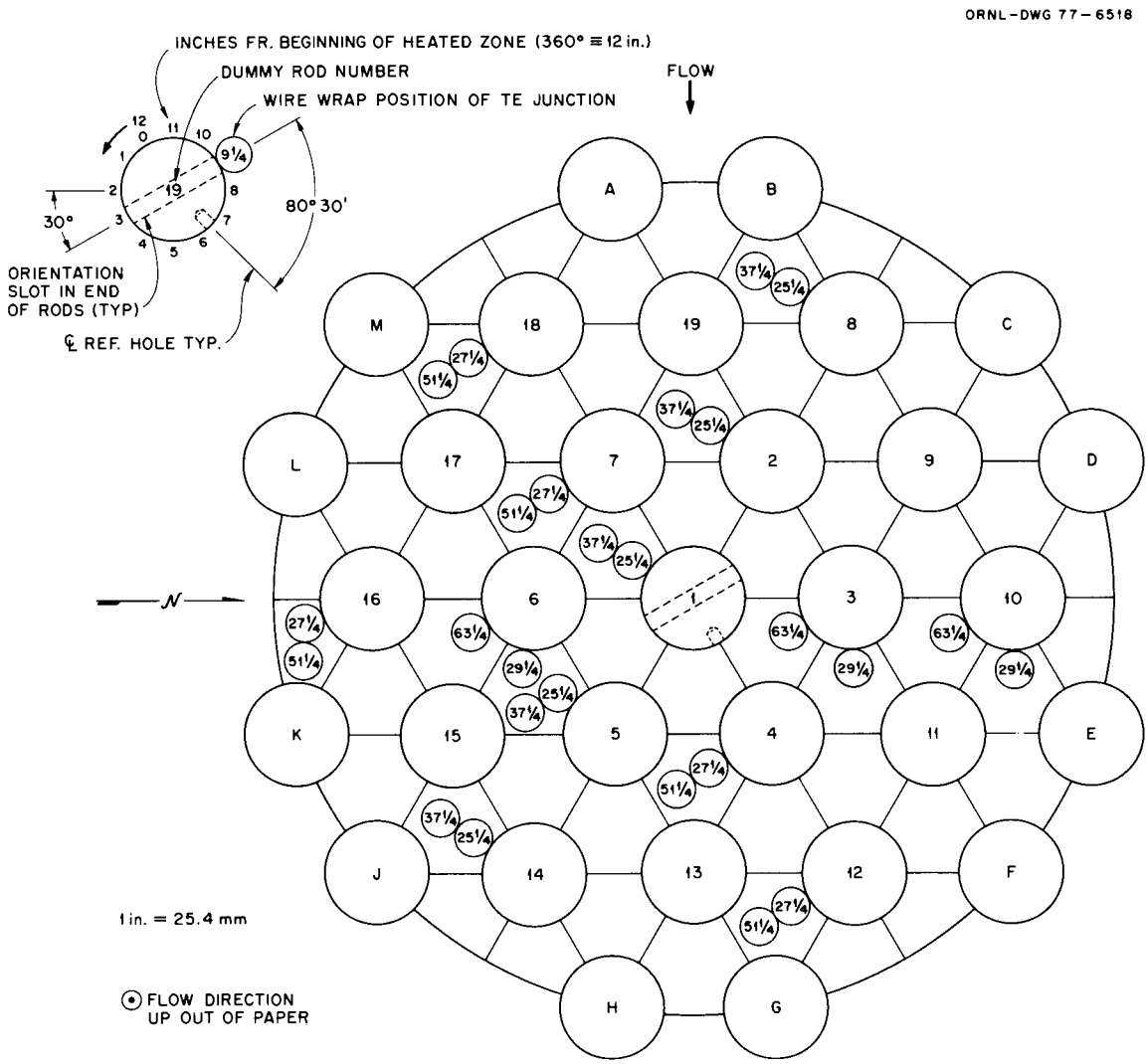

Fig. 2.2. THORS Bundle 3C SFGP instrumentation.

All tests were conducted in a similar manner. With an 8:1 bypass-to-test-section flow ratio, equilibrium conditions of test-section flow, power, and inlet temperature were established to produce a bulk mean test-section exit temperature of 700°C (1300°F). The preprogrammed flow reduction was initiated, and the flow remained at its low value for up to 24 sec (established so as not to exceed allowable bundle housing temperature limits). The flow was then automatically raised to its final value. The transient flow sequence is shown in Fig. 2.3 for a typical boiling run (Run 232).

Table 2.1. Test series 10 (Bundle 3C); Test 20 — boiling propagation tests (preliminary)

Run	Date	Inlet temperature [°C (°F)]	Flow prior to flow reduction (gpm)[a]	Reduced flow (average) (gpm)[a]	Flow after flow increase (gpm)[a]	Central pin power (19 pins) (kW/pin)[b]	Edge pin power (12 pins) (kW/pin)[b]	Total power (31 pins) (kW)	Time at reduced flow (sec)
					100% edge pin power				
101	4-18-78	425 (797)	14.2	9.2	17.5	8.14	8.05	251	35
102[c]	4-18-78	446 (834)	16.0	9.2	20.0	9.20	9.14	285	24
103[c]	4-18-78	447 (837)	17.5	9.6	19.0	9.51	9.42	294	24
104	4-18-78	450 (842)	17.2	9.0	17.5	9.69	9.60	299	24
101[d]	4-27-78	441 (825)	16.9	8.4	17.5	9.79	9.79	303	24
					80% edge pin power				
211[c]	4-19-78	444 (831)	16.2	9.0	18.0	9.60	7.67	274	24
221[c]	4-19-78	444 (832)	17.0	9.3	18.3	9.89	7.89	282	24
231	4-19-78	448 (839)	16.5	9.0	17.5	10.12	8.09	289	5[e]
232	4-19-78	449 (840)	16.5	9.0	17.5	10.12	8.08	289	24
241[f]	4-25-78	448 (839)	16.8	9.0	17.5	10.11	8.09	289	24
					60% edge pin power				
311[c]	4-20-78	429 (804)	14.2	9.4	17.5	10.06	6.06	264	24
321[c]	4-20-78	435 (815)	15.0	9.2	17.5	10.25	6.18	269	24
331	4-20-78	441 (825)	15.1	9.0	17.5	10.27	6.19	269	24
341	4-25-78	454 (850)	15.9	9.0	17.5	10.15	6.03	265	24
351	4-25-78	432 (809)	13.9	8.0	17.5	9.95	5.92	260	24
					40% edge pin power				
401[c,f]	4-24-78	412 (774)	12.1	8.3	18.0	9.24	3.65	219	24
411[c,f]	4-25-78	409 (769)	12.4	7.9	18.8	10.19	4.04	242	24
421[f]	4-24-78	413 (775)	12.4	7.9	17.5	10.07	3.97	239	24
431[f]	4-24-78	417 (783)	12.7	8.0	17.5	10.33	4.08	245	24
441[f]	4-25-78	429 (804)	13.2	8.0	17.5	10.37	4.11	246	24
					0% edge pin power				
500[f]	5-26-78	378 (713)	9.1	7.2	18.2	9.82	0.0	186	24
501[c,f]	5-26-78	381 (718)	9.1	7.3	19.0	10.43	0.0	198	24
511[c,f]	5-26-78	388 (730)	9.9	7.8	19.0	10.96	0.0	208	24
521[f]	5-26-78	389 (732)	10.3	8.0	17.5	11.69	0.0	222	24
531[f]	5-26-78	396 (744)	10.8	8.0	17.5	11.83	0.0	224	24

[a] 1.0 gpm = 0.063 ℓs^{-1}.

[b] Heated length of pin = 0.533 m (1.75 ft).

[c] Selected data are being transmitted for these runs.

[d] Test 21.

[e] Automatic scram; excessive incipient superheat caused high-temperature limit scram.

[f] Flow reduction slope = 1 gpm/sec; all other runs have slope = 2 gpm/sec

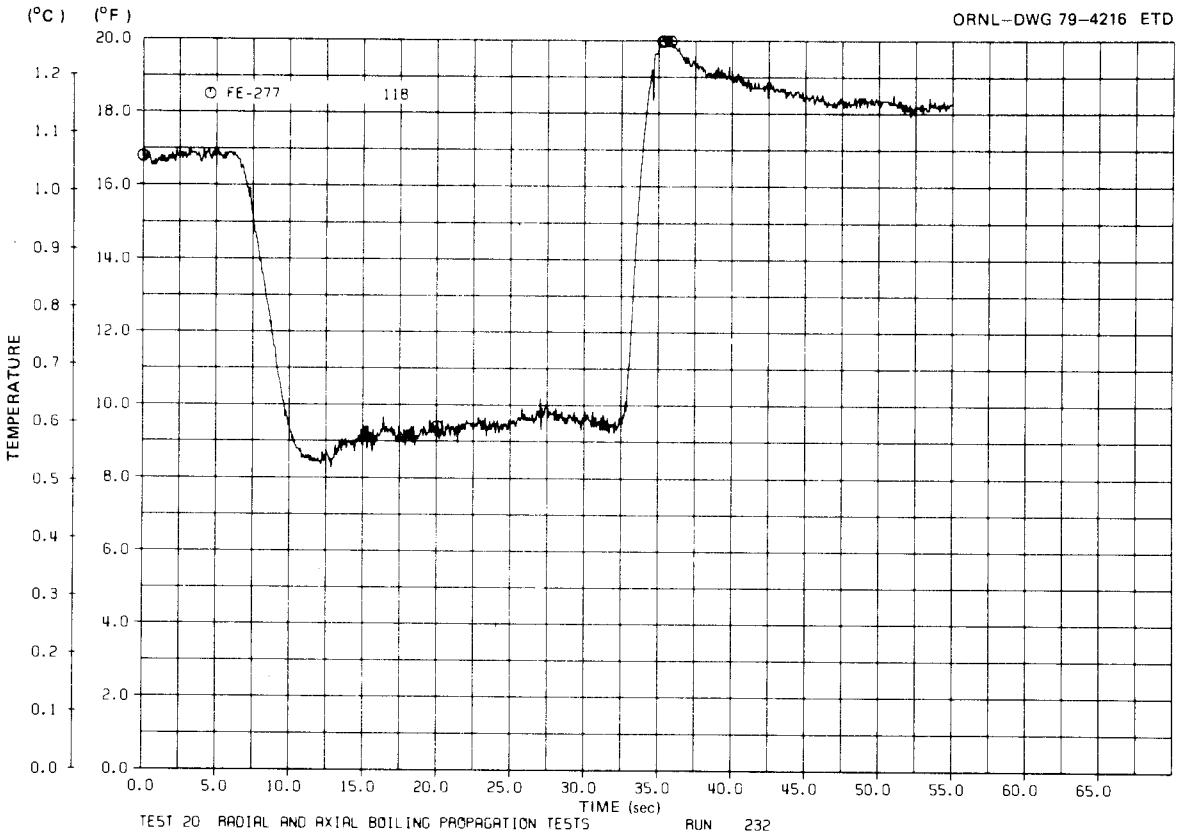

Fig. 2.3. Test section inlet flow during Test 20, Run 232 (Bundle 3C).

Typical transient temperature responses for Run 232 are shown in Figs. 2.4 and 2.5. Figure 2.4 shows two temperatures from heater-internal (attached to inside of clad) thermocouples; thermocouple R0417DE is located on rod 4 at 429 mm (16.9 in.) from the start of the heated section and appears to indicate boiling from 13 to 33 sec. Thermocouple R3316DE is located on rod 33 at 404 mm (15.9 in.) from the start of the heated section and does not indicate boiling.

Figure 2.5 shows a 3-D temperature plot at fixed times during the transient. (All wire-wrap and heater internal temperatures have been plotted.) In Fig. 2.5, Z refers to axial distance from the start of the heated section and radius refers to the radial distance from the center of the bundle. The discontinuities in the figures (from Z = 0 to Z = 6 in.; at Z = 30 in., r = 0; and at 48 < Z < 58 in., r = 0) are a result of insufficient instrumentation at these locations. Note that the peak

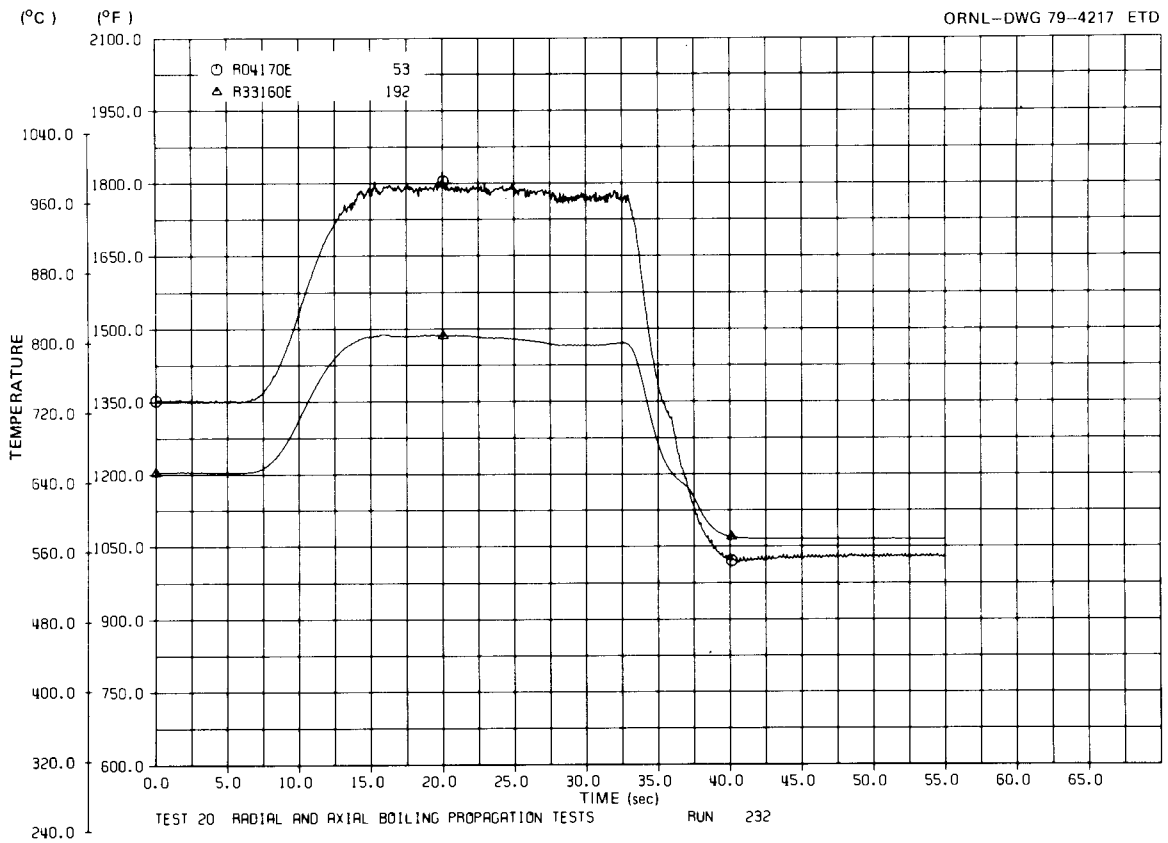

Fig. 2.4. Two-heater internal thermocouple responses from Test 20, Run 232 (Bundle 3C). R0417DE denotes the thermocouple located on Rod 4, 429 mm (16.9 in.) from the start of the heated section; R3316DE denotes the thermocouple located on Rod 33, 404 mm (15.9 in.) from the start of the heated section (see Fig. 2.1 for thermocouple locations).

temperature (saturation temperatures) occurs at $16 \leq z \leq 22$ in. at $0 \leq r \leq 0.2$ in., corresponding to the blocked channels downstream of the blockage, which is situated at 380 mm (15 in.). This boiling zone stays relatively constant for the 12 sec shown (total boiling time for this run is estimated at 21 sec).

For the ongoing data analysis, temperature responses such as those shown in Fig. 2.4 are used to obtain a detailed quantitative description of the boiling and temperature zones as a function of time. Three-dimensional plots like Fig. 2.5 are invaluable in obtaining quantitative overall pictures of boiling propagation and temperature fields.

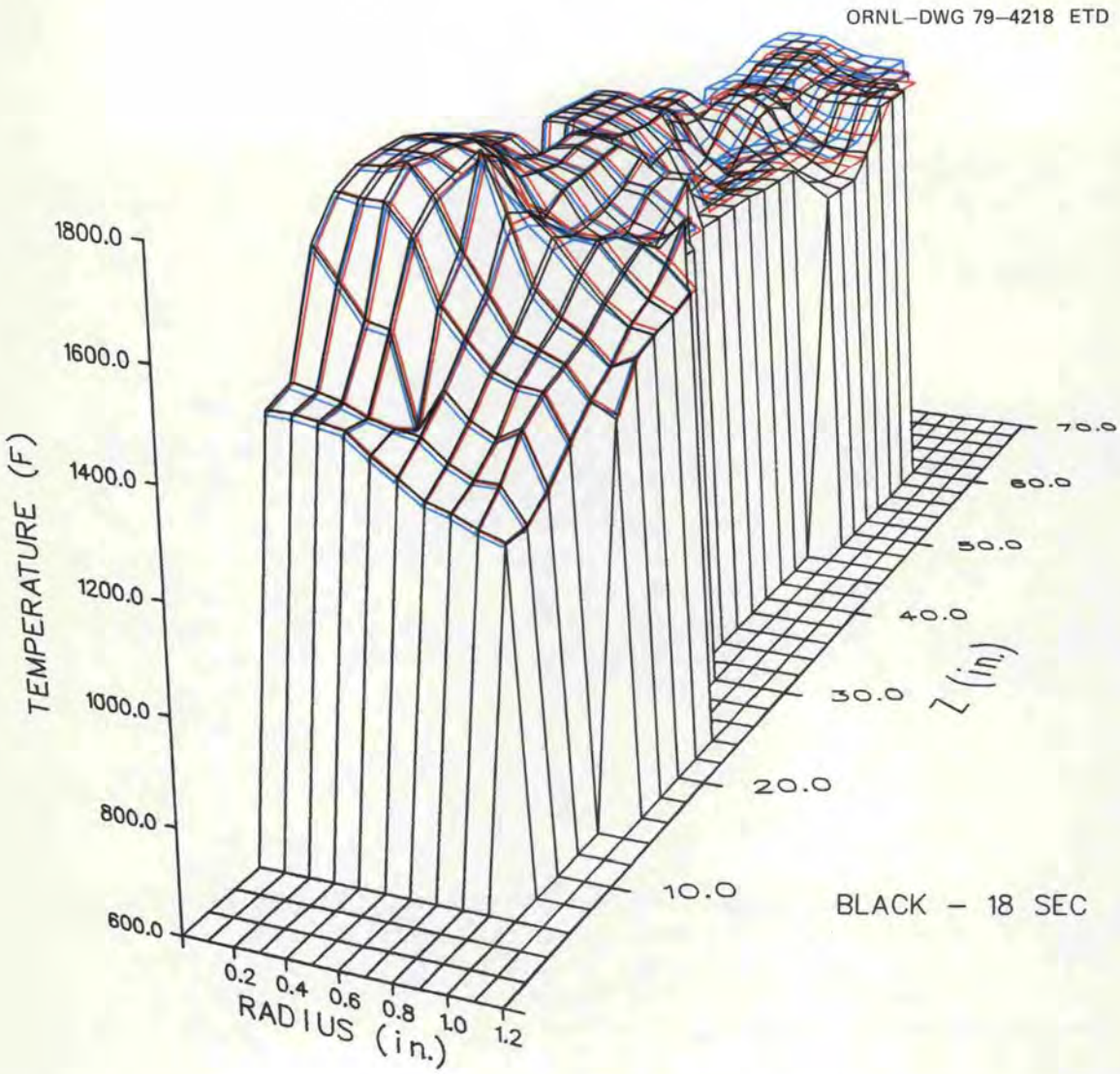

Fig. 2.5. Temperature plots of heater internal and wire-wrap thermocouples at 18, 24, and 27 sec for Test 20, Run 232 (Bundle 3C). The r–z origin of the figure corresponds to the center of the bundle at the start of the heated section (1 in. = 25.4 mm).

2.2 Single vs Parallel Channel Boiling Stability Studies Predicted by SAS 3A

G. A. Klein

The SAS 3A[2] computer code was used to study the interaction between two heated flow bundles operated hydraulically in parallel with a bypass representing many nonboiling channels. In the boiling flow regime, a

comparison is made between the recurring oscillatory flow behavior determined when uniform input parameters are applied to both bundles and the oscillatory flow behavior of a single bundle with identical channel inputs. These parallel bundle flow results are used as a comparative basis for evaluating the changes in flow behavior which occur when unequal channel input parameters are applied to the parallel-channel system, as in a parfait core geometry.

In the present SAS flow model, the flow channels have a constant exit pressure with a compressible inlet plenum and hydraulic coupling to the bypass channel. Parameters that would enable the recurring oscillatory flow behavior to be studied were chosen. The input data for the single- and parallel-channel control case were based on a modification of data taken from THORS Bundle 6, Test 73E, Run 102.[3] For this test, the bundle power was 290 kW with 15.3 kW/pin. The test-section low flow was 0.49 ℓ/s (7.7 gpm), and the test was run to dryout. The modifications include the following changes in the input data: (1) an addition of an inlet plenum volume equal to one-half the expansion tank volume with coupling of this inlet plenum to the system's channels; (2) an inlet orifice coefficient set equal to 15 for the control case; and (3) the incorporation of only the low flow value of the velocity in the evaluation of the pump head input parameter vs time. The channel input parameters were equally applied to each channel of the parallel-channel system and were identical to that applied to the single-channel system.

The parallel-channel interactions reported by previous investigators[4] indicated flow oscillations that were 180° out of phase between channels, while the oscillation periods were essentially identical for the one- or two-channel system. It was uncertain, however, whether the coupling between the parallel channels altered the amplitude of oscillation between the one- or two-channel systems. If so, the impact this would have on the stability boundaries could not be readily predicted. The SAS runs for parallel-channel systems, outlined in Table 2.2, indicate nearly the same stability boundaries between parallel- and single-channel systems, and it was found that the oscillation periods and amplitudes of both systems were nearly identical. The boiling stability

Table 2.2. Comparison of single- and parallel-channel system

Variable	Time to boiling (sec)		Time to dryout minus time to boiling, Δt_D (sec)		Time to clad melt minus time to boiling, Δt_{cm} (sec)	
	Single channel	Parallel channel	Single channel	Parallel channel	Single channel	Parallel channel
Control case	6.34	6.36	1.2	1.20	2.95	3.0
$K = 1/2\ K_{control}$	5.66	5.66	1.1	1.10	2.85	2.85
Single-phase inlet velocity $= 1/2 \times \left(\text{single-phase inlet velocity}\right)_{control}$	1.22	1.22	0.85	0.8	2.30	2.25
$q = 2 \times q_{control}$	0.54		0.38		0.99	
Upper and lower plenum volume $= 1/30 \times \left(\text{upper and lower plenum volume}\right)_{control}$	6.42	6.35	>48	>16	>48	>16
Length fission gas plenum $= 2 \times \left(\text{length fission gas plenum}\right)_{control}$	6.03		1.09		2.55	
$q = 1.05 \times q_{control}$		4.31		1.20		2.85
$q = 0.95 \times q_{control}$		>48				

boundary trends, outlined in Table 2.2, are consistent with those determined experimentally. The time to boiling initiation is determined by heat input, single-phase inlet velocity, inlet and exit pressure drop, orificing, and channel length. However, unlike the experimental findings, the parallel-channel SAS runs did not indicate a 180° phase shift between channels. In the presence of flow reversal, it was noted that there is some correlation between the amplitude and period of the flow oscillations immediately following the onset of boiling and the magnitudes of Δt_D and Δt_{cm} shown in Table 2.2.

The thermodynamic coupling between two parallel channels was examined for unequal channel input parameters, as shown in Table 2.3. The SAS output indicates that the channels are strongly coupled if an unequal impedance to the inlet flow of any one channel (created by vapor generation, orificing, etc.) exists. The mechanism (i.e., large density change) generating the density wave-type oscillations is initially established in the channel with the larger flow inlet impedance. As the impedance to the flow becomes greater in this channel due to boiling, the coolant mass flow is increased in the alternate channel, thereby delaying the onset of boiling in that channel. Thus the boiling stability trends outlined in Table 2.2 can be significantly altered in a parfait core geometry simply by an appropriate change in the magnitude of the inlet impedances to the channels.

In summary, the time to boiling initiation is determined by heat input, single-phase inlet velocity, inlet and exit pressure drop, orificing, and channel length; there also appears to be some correlation between the amplitude and period of the flow oscillations and the magnitude of Δt_D and Δt_{cm}, as predicted by SAS. The SAS output indicates that the parallel channels are strongly coupled if unequal impedance to the inlet flow of one channel (created by vapor generation, orificing, etc.) exists. If both channels have equal inlet impedances, then the coupling is negligible and each channel behaves as if it were a single channel in parallel with a bypass.

Table 2.3. Parallel-channel parfait core system

Variable	Channel No.	Time to boiling (sec)	Time to dryout minus time to boiling, Δt_D (sec)	Time to clad melt minus time to boiling, Δt_{cm} (sec)
Case 1				
$q_1 = q_{control} \times 1.05$	1	4.31	1.15	2.85
$q_2 = q_{control} \times 0.95$	2	>38	>38	>38
Case 2				
$q_1 = 1.05 \times q_{control}$	1	3.95	2.8	3.9
$q_2 = 1.05 \times q_{control}$	2	>38	>38	>38
Upper and lower plenum volume $= 1/30 \times \begin{pmatrix}\text{upper}\\ \text{and}\\ \text{lower}\\ \text{plenum}\\ \text{volume}\end{pmatrix}_{control}$				
Case 3				
$q_1 = q_2 = q_{control}$	1	3.83	1.1	2.8
$vel_1 = vel_2 = vel_{control}$				
$K_1 = 15.0$	2	>11		
$K_2 = 7.5$				
Case 4				
$q_2 = q_1 = q_{control}$	1	>14	>14	>14
$vel_{in\ s.f.}\big\|_1 = vel_{in\ s.f.}\big\|_{control}$				
$vel_{in\ s.f.}\big\|_2 = 1/2\ vel_{in\ s.f.}\big\|_{control}$	2	1.18	0.85	2.29
Case 5				
$q_1 = q_{control} \times 1.05$	1	3.2623		2.70
$q_2 = 0.95 \times q_{control}$				
$K_1 = 15.0$	2	>10.827		
$K_2 = 7.5$				
Case 6				
$q_1 = 1.05 \times q_{control}$	1	4.2065		2.80
$q_2 = 0.95 \times q_{control}$				
$K_1 = 7.5$	2	4.6405		3.90
$K_2 = 15.0$				

References

1. J. L. Wantland et al., *Sodium Boiling Tests in a 19-Rod Bundle with a Central Blockage — Record of Experimental Data for Fuel Failure Mockup Bundle 3B*, ORNL/TM-5458 (October 1976).

2. F. E. Dunn et al., *The SAS 3A LMFBR Accident Analysis Computer Code*, ANL/RAS 75-17 (April 1975).

3. THORS Staff, *Sodium Boiling in a Full-Length 19-Pin Simulated Fuel Assembly (THORS Bundle 6A)*, ORNL/TM-6553 (to be issued).

4. S. Lee et al., "Sustained and Transient Boiling Flow Instabilities in Two Parallel Channel Systems," *Two-Phase Flows and Heat Transfer, Vol. I*, p. 467, eds. S. Kakac and F. Mayinger, Hemisphere Publishing Company, Washington, D.C., 1977.

3. THORS FACILITY OPERATION

P. A. Gnadt B. H. Montgomery
N. E. Clapp A. Golshani*

3.1 Test Facility Operation

The THORS Facility has operated for a total of 26,389 hr. The THORS operating history is summarized in Table 3.1. Eleven bundle configurations have been operated at varying levels of power for a total of 2280 hr. Bundle 3C, test series 10, was completed during the last week of April. Currently, the facility is being modified for test Bundle 9 operations.

A new Hastelloy-X test-section housing that is designed to operate up to 982°C (1800°F) and that will accommodate a full-length CRBR simulated fuel assembly will be installed. Because the new test section is longer, the expansion tank and centrifugal pump will be raised to give the proper reactor upper plenum simulation. The piping to the new 2.0-MW heat exchanger will also be installed. This modification, which will permit 2.0 MW of electrical power input to the test bundles along with the 2.0-MW heat removal capability, is scheduled for completion by January 1, 1979.

3.2 Test Bundles

3.2.1 Bundle 3C LOA-2 local fault accommodation

The test program for this bundle has been completed. The bundle and associated housing have been removed from the facility, but the bundle has not yet been removed from its housing for examination. The design for the bundle is described in detail in Ref. 1. Photographs taken during the assembly of the test bundle and a detailed description of the test program, test series 10, are presented in Ref. 2. Preliminary results of the test program are described in Refs. 3 and 4.

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*ORAU trainee from Iran.

Table 3.1. THORS operating experience

Bundle identification	Duct configuration	Blockage configuration	Heated length (mm)	Fission-gas simulators	Isothermal operation (hr)	Operation with power (hr)
1A	Scalloped (bare edge rod)	None	610	No	1,300	200
2	Hex (full-size edge gaps)	Channels 13–14 (inlet)	533	No	3,010	470
1B	Scalloped (bare edge rod)	None	610	No	894	350
3A	Scalloped (wrapped edge rods)	Channel 6 (heated zone)	533	No	3,039	151
3B[a,b]	Scalloped (wrapped edge rods)	Channel 6 (heated zone)	533	No	537	51
5A, 5B, 5C	Hex (half-size edge gaps)	Channel 12[c] (edge gap)	457	No	3,252	391
5D[a]	Hex (half-size edge gaps)	None	457	No	2,827	97
6[d]	Hex (half-size edge gaps)	None	914	FFTF length	7,864	380
3C[e]	Scalloped (wrapped edge rods)	Channel 6 (heated zone)	533	FFTF length	3,666	190
					26,389	2,280

[a]Boiling tests (with and without gas injection).

[b]Boiling tests included 45 sec with full bundle (heater 7 opened during boiling tests with gas injection) and 11 min, 15 sec with rod 7 inoperative; 10 min, 16 sec of this time was continuous (without gas injection).

[c]A 12-channel edge blockage flush with duct for Bundle 5A, flush followed by 0.36-mm displacement for 5B, removed for Bundle 5C.

[d]A total of 38 tests were run in which boiling occurred; five of these runs were to dry out (unstable) conditions; total boiling time for the bundle was ∿11.5 min.

[e]A total of 24 tests were run in which boiling occurred; total boiling time for the bundle was ∿10 min.

3.2.2 Bundle 9 LOA-2 LOPI accommodation

Bundle 9 is a 61-pin, full-length bundle designed to simulate a larger segment of an LMFBR fuel subassembly. It is the largest bundle fabricated for use in the THORS Facility. The design of this bundle is discussed in detail in Ref. 5. Photos of the bundle parts prior to assembly are available in Ref. 3.

Figures 3.1, 3.2, and 3.3 show the instrumentation in the bundle. Figure 3.4 is a schematic diagram of the facility and associated instrumentation as planned for the Phase 1 (steady-state) tests planned for the bundle. Figure 3.5 is a schematic diagram of the facility and associated instrumentation planned for the Phase 2 (transient flow and boiling tests) portion of the tests.

The Phase 1 tests are primarily steady-state thermal-hydraulic tests which will be used to compare results from a 61-pin bundle with those of the earlier 19-pin bundles. Because of the high flow requirements for Phase 1 testing of Bundle 9, it has been necessary to remove the expansion tank from the system in order to control the distribution of the sodium inventory in the facility. The expansion tank contains a sodium inert gas interface, and a second liquid-gas interface exists in the ORNL-designed centrifugal sodium pump. Any pressure gradient between the expansion tank gas space and the pump gas space results in a shift in liquid inventory from one vessel to the other. During previous operation with 19-pin bundles, the sodium flows have been sufficiently low that only a modest pressure gradient existed between the expansion tank upstream of the pump and the pump itself. This gradient could be balanced with a modest sodium inventory transfer (increasing the expansion tank liquid level and decreasing the pump liquid level). Such is not the case at higher flows, and the second liquid-gas interface had to be eliminated for this phase of operation.

A new electromagnetic pump has been ordered and will be installed prior to the start of the boiling tests. The centrifugal pump will be valved out of the system, and the new electromagnetic pump will be used for the transient boiling tests. The expansion tank will be reinstalled in the system to simulate the upper reactor plenum during the boiling

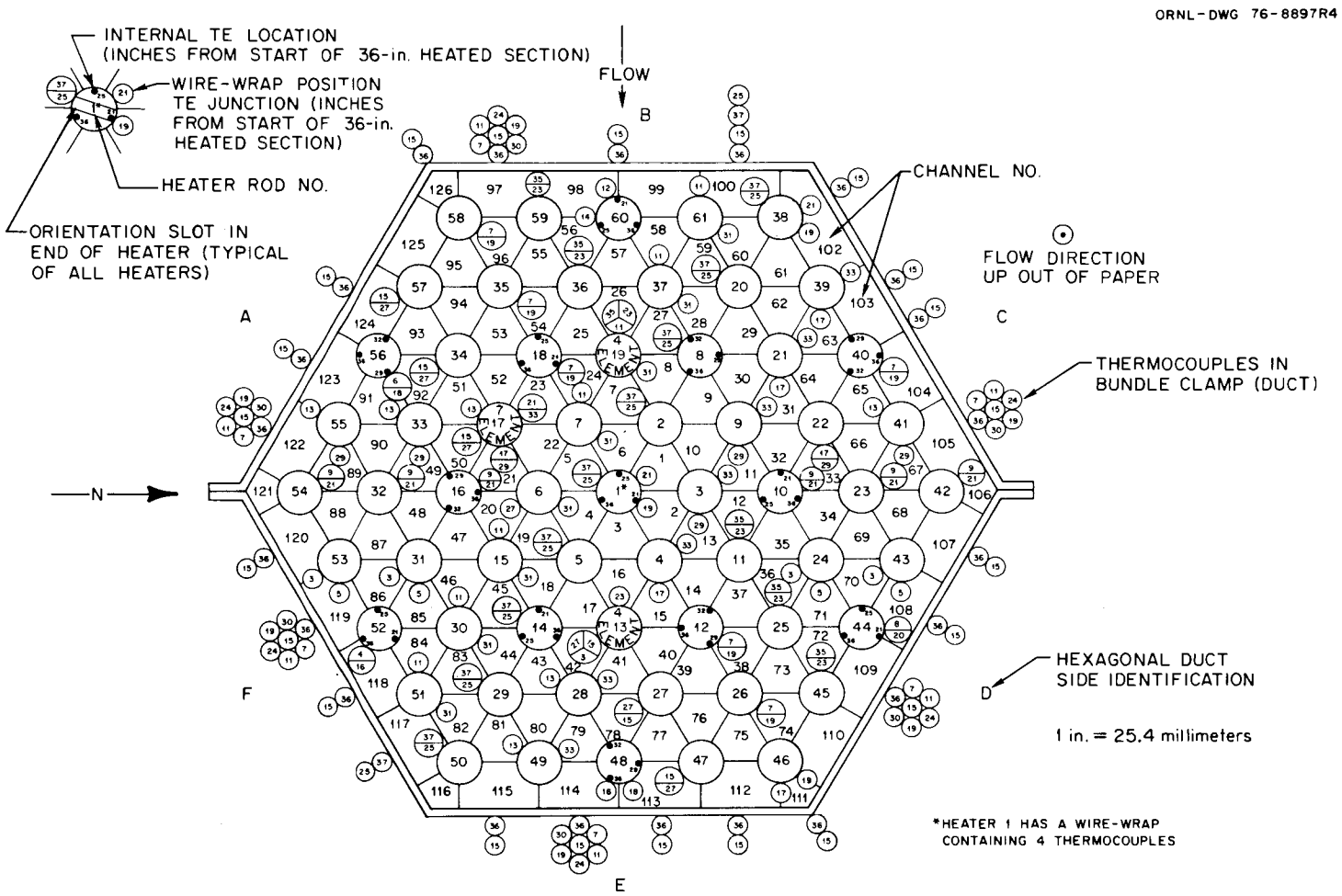

Fig. 3.1. THORS Bundle 9 heater rod bundle instrumentation.

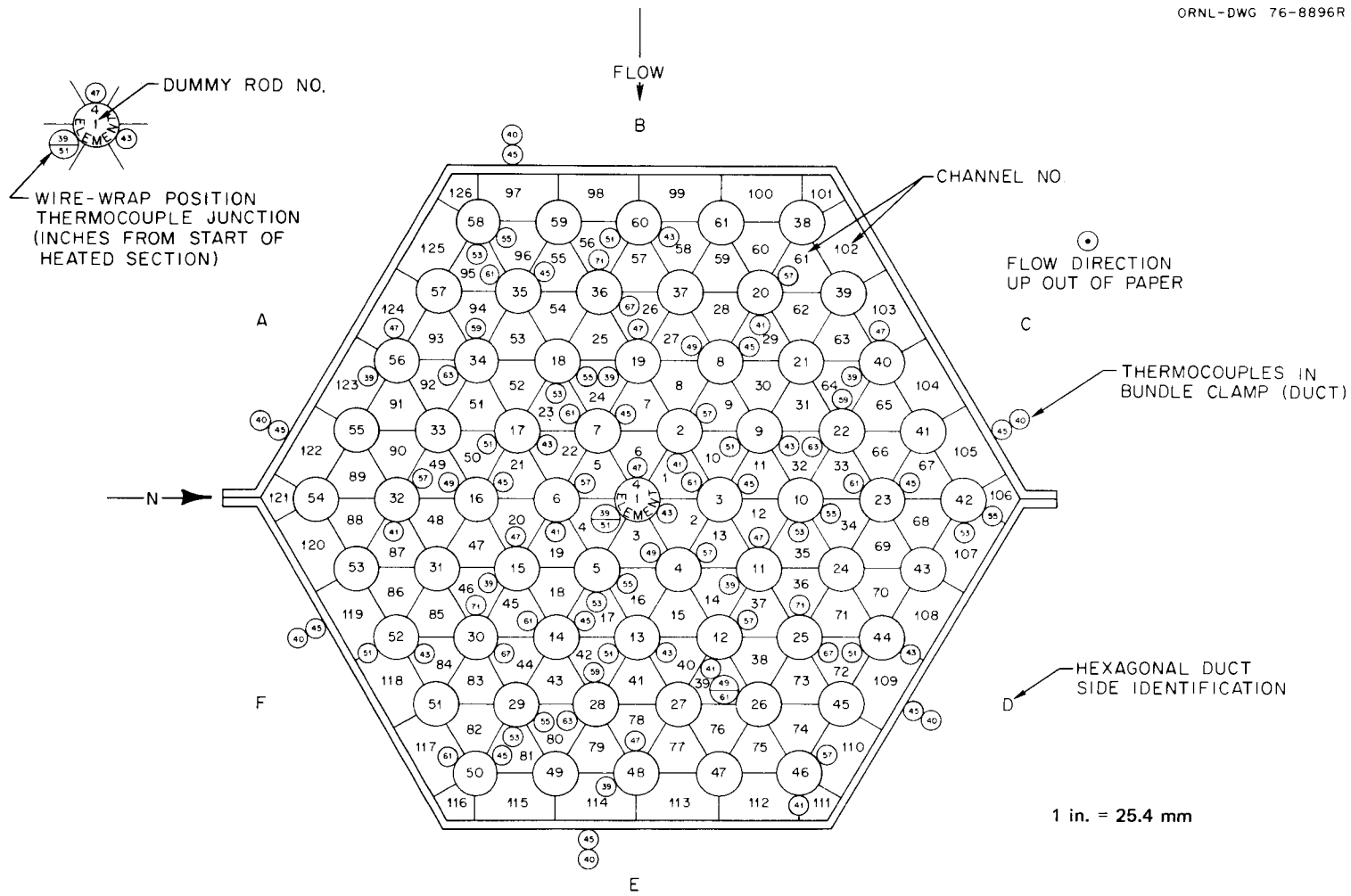

Fig. 3.2. THORS Bundle 9 fission gas simulator bundle instrumentation.

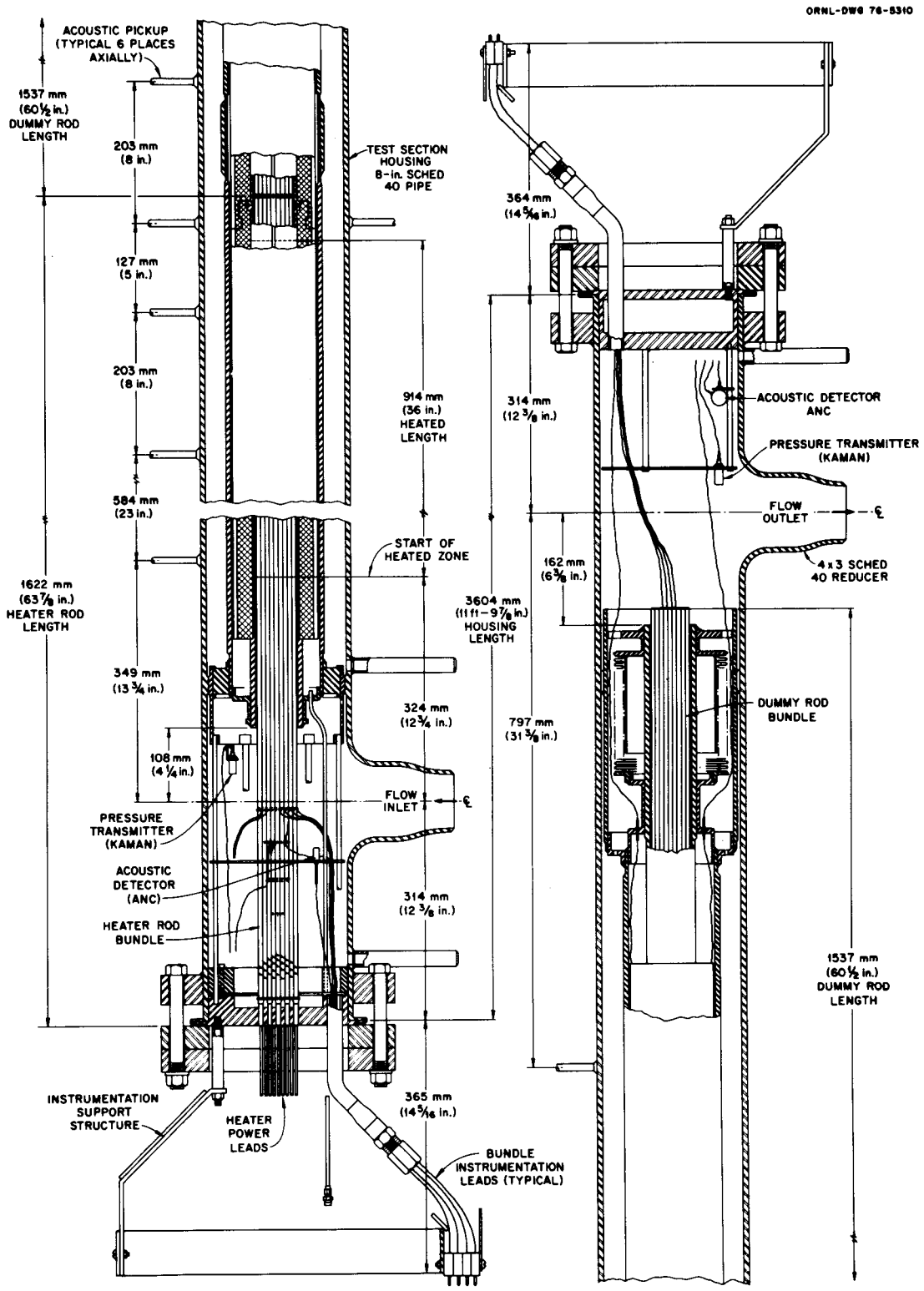

Fig. 3.3. THORS Bundle 9 test-section assembly.

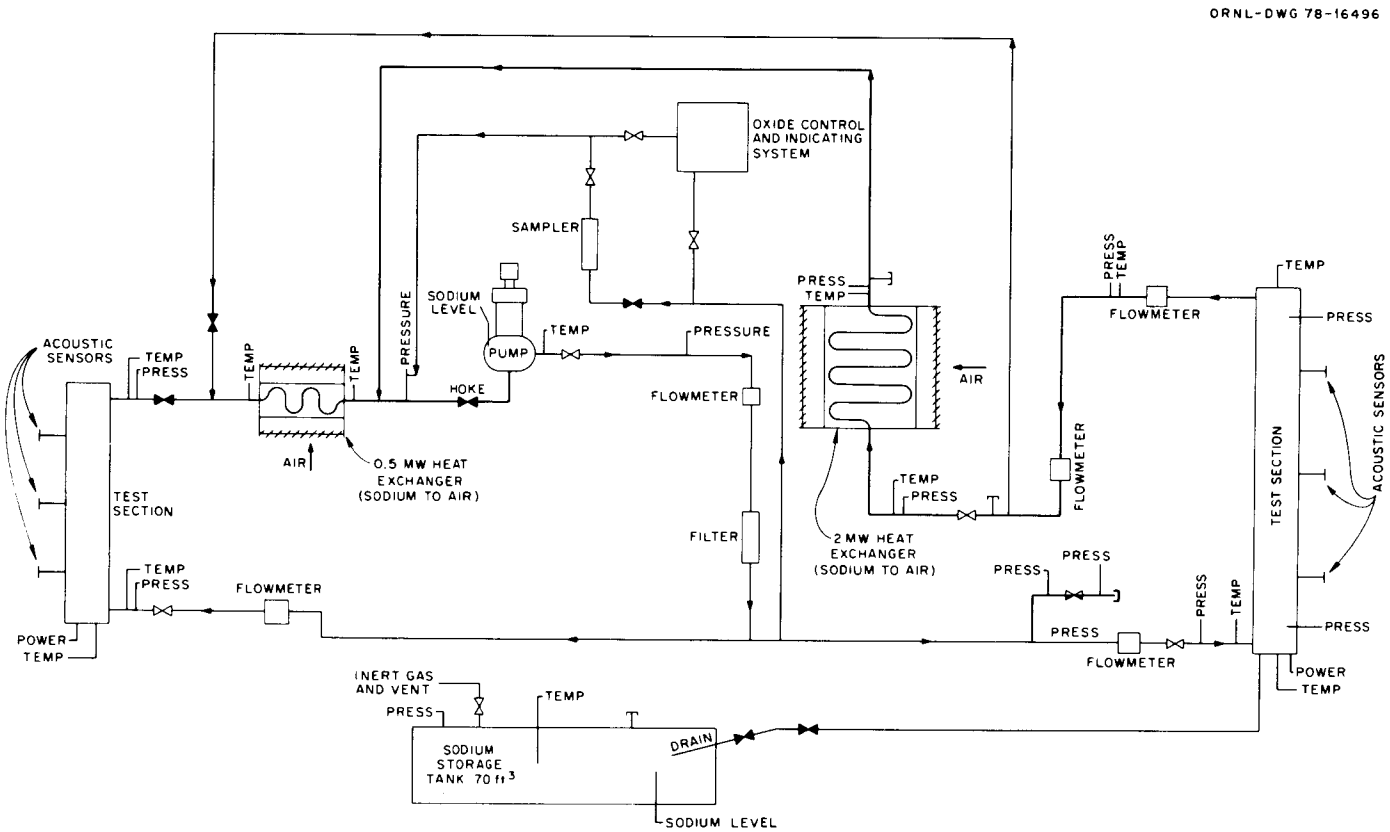

Fig. 3.4. Schematic diagram of THORS system for Bundle 9 Phase 1 (steady-state) tests.

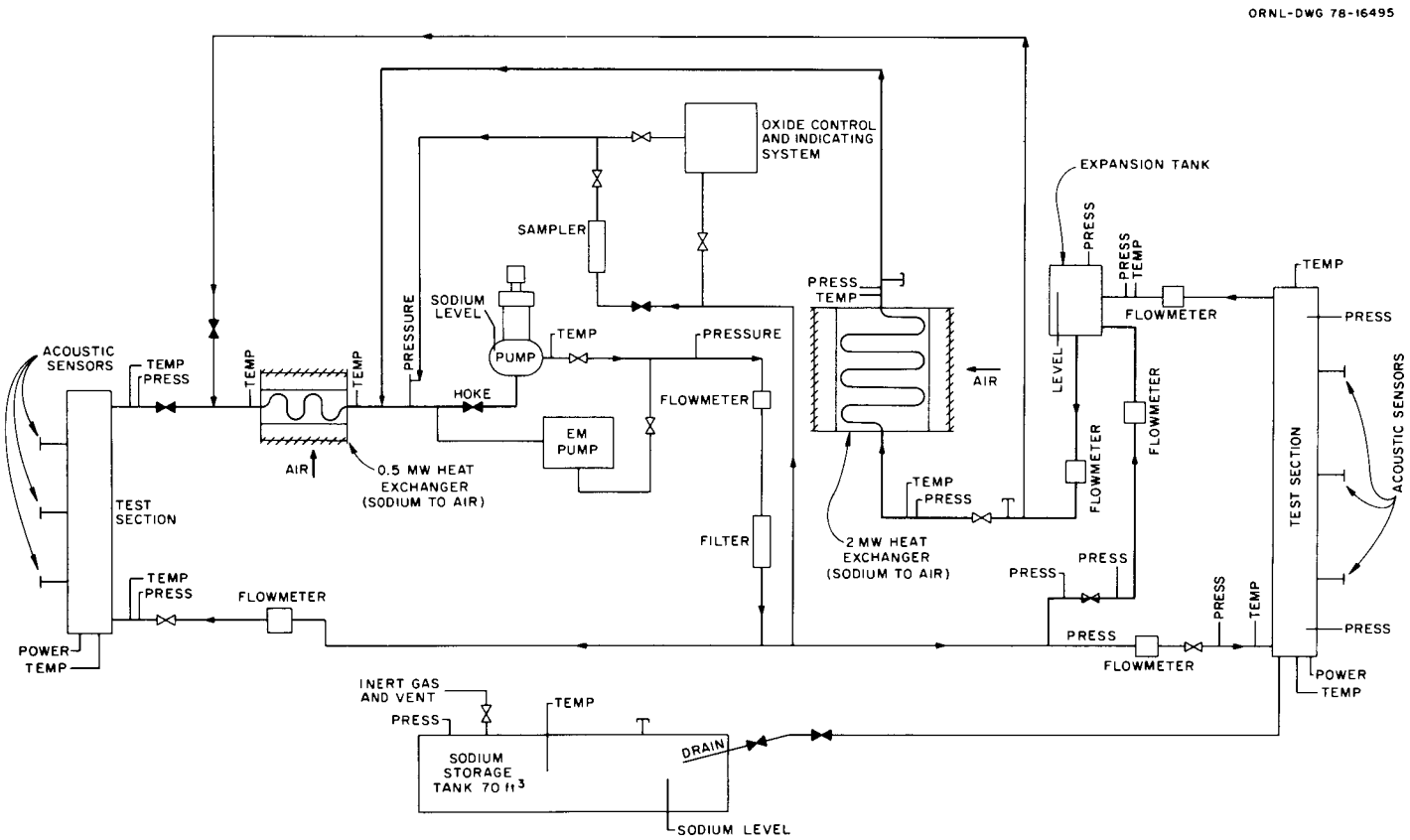

Fig. 3.5. Schematic diagram of THORS system for Bundle 9 Phase 2 (transient-boiling) tests. New electromagnetic pump is shown and expansion tank has been reinstalled to simulate the reactor upper plenum.

tests; however, use of the electromagnetic pump and valving out the centrifugal pump will result in a system with only one liquid-gas interface, and inventory transfer will not occur.

Figures 3.6 through 3.20 are photos taken of the bundle during the assembly process. (See Ref. 3 for photographs taken during earlier stages of assembly.) Figure 3.6 is a photograph of the SFGP assembly in the fixture prior to installation of the top half of the sodium duct. Figure 3.7 is a close-up photograph of the joining of the SFGP to the end of FPSs. Figure 3.8 is a close-up photograph showing how the SFGPs are held in the proper orientation for making the attachment to the FPSs. Figure 3.9 is a photograph taken from the opposite direction of that in Fig. 3.8. This photograph shows the wire-wrap thermocouples that are attached to the SFGPs as they come through the rear of the orientation fixture. Figure 3.10 is a photograph showing the bundle assembly after thermal insulation has been installed around the sodium duct. Also shown is one side of the labyrinth seal before it has been installed into proper position on the assembly. Figure 3.11 is a photograph of the male side of the labyrinth seal after final machining and before installation. The grooves in the part, when inserted into the close-fitting matching part, create a pressure drop that greatly reduces the flow through the labyrinth.[3]

Figure 3.12 is a close-up photograph of one of the duct wall thermocouples. The sheathed thermocouple assembly is formed to fit into a groove in the thermal insulation and is installed in a hole drilled in the outside of the sodium duct. The bottom of the hole is located between 0.1 and 0.2 mm (0.005 and 0.010 in.) from the inner surface of the sodium duct. The sheet metal and wire are used to force the top of the thermocouple against the bottom of the hole. A view of the thermal insulation and thermocouples prior to installation of the outer container is shown in Fig. 3.13. Figure 3.14 is a photograph of the bulkhead to which the bundle thermocouples are brazed. The outer container will be welded to this bulkhead at a later date. Figure 3.15 is a close-up view of this bulkhead and shows the braze material surrounding the thermocouples. The clamp shown in Figs. 3.14 and 3.15, which is used during assembly to relieve the strain from the brazed area, will be removed before operation.

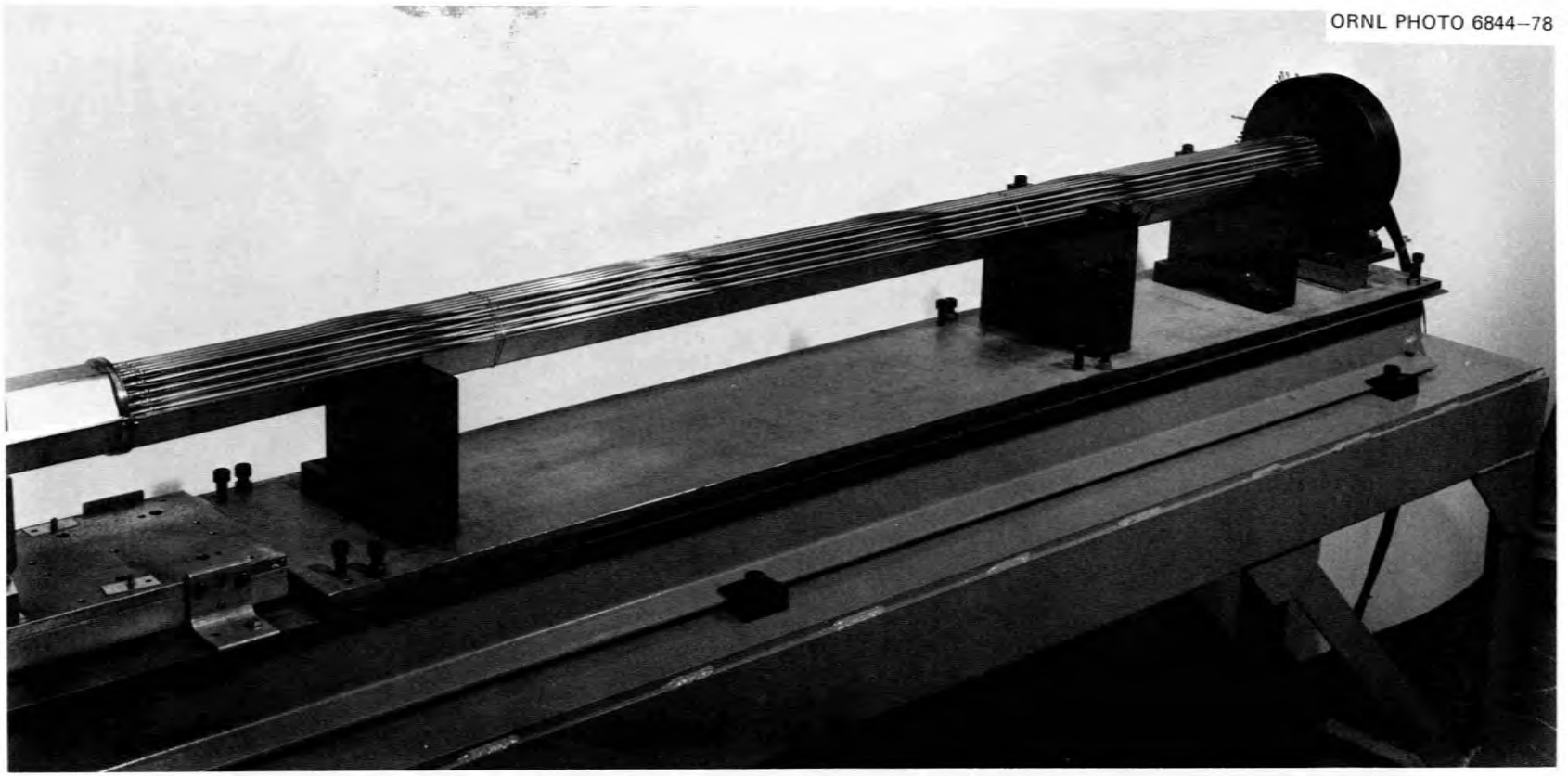

Fig. 3.6. THORS Bundle 9. Photograph showing SFGP assembly in fixture.

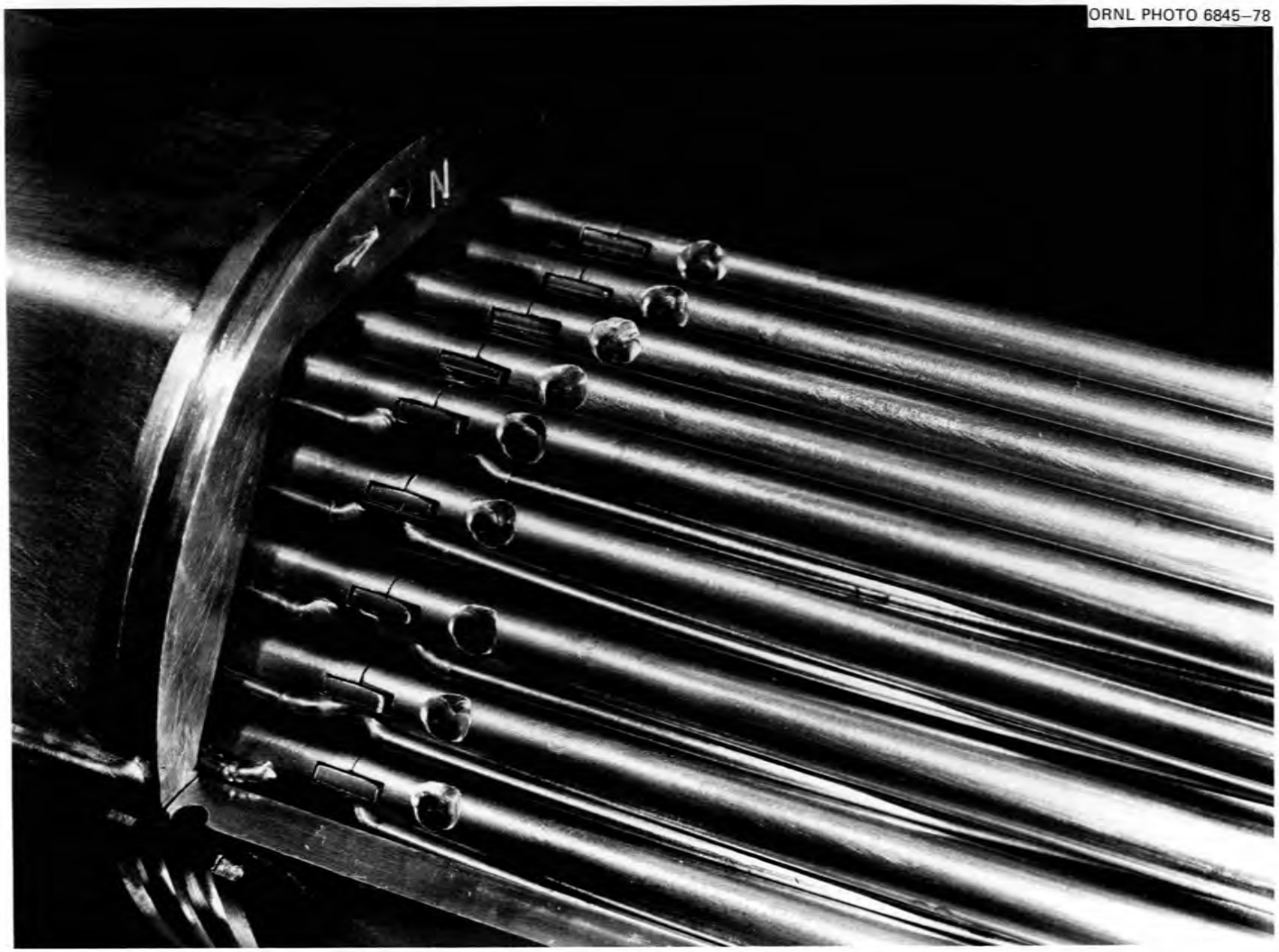

Fig. 3.7. THORS Bundle 9. Photograph taken during assembly showing joint between FPSs and SFGPs.

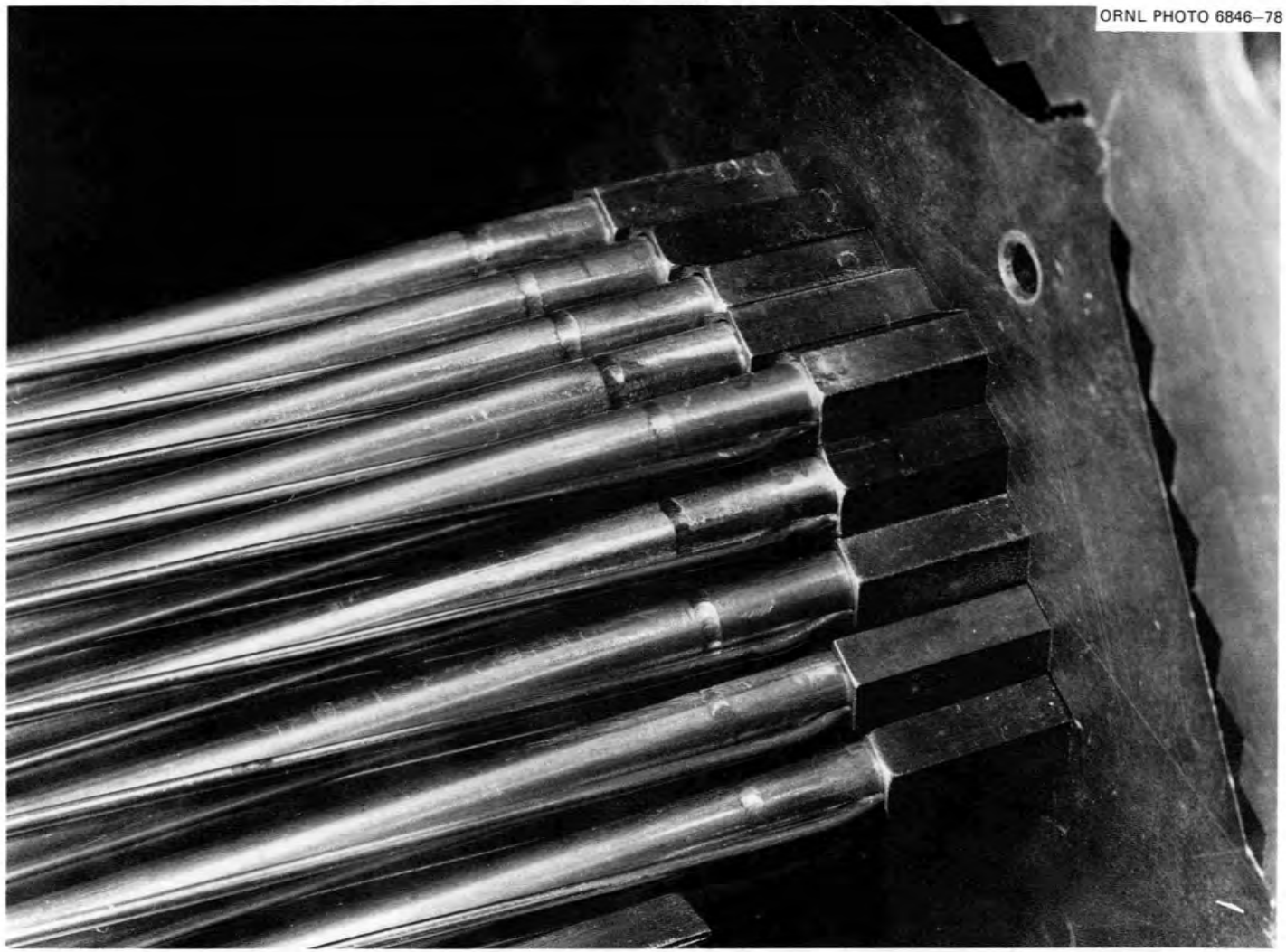

Fig. 3.8. THORS Bundle 9. Close-up photograph showing fixture that oriented fission gas plenum extension simulators to FPSs.

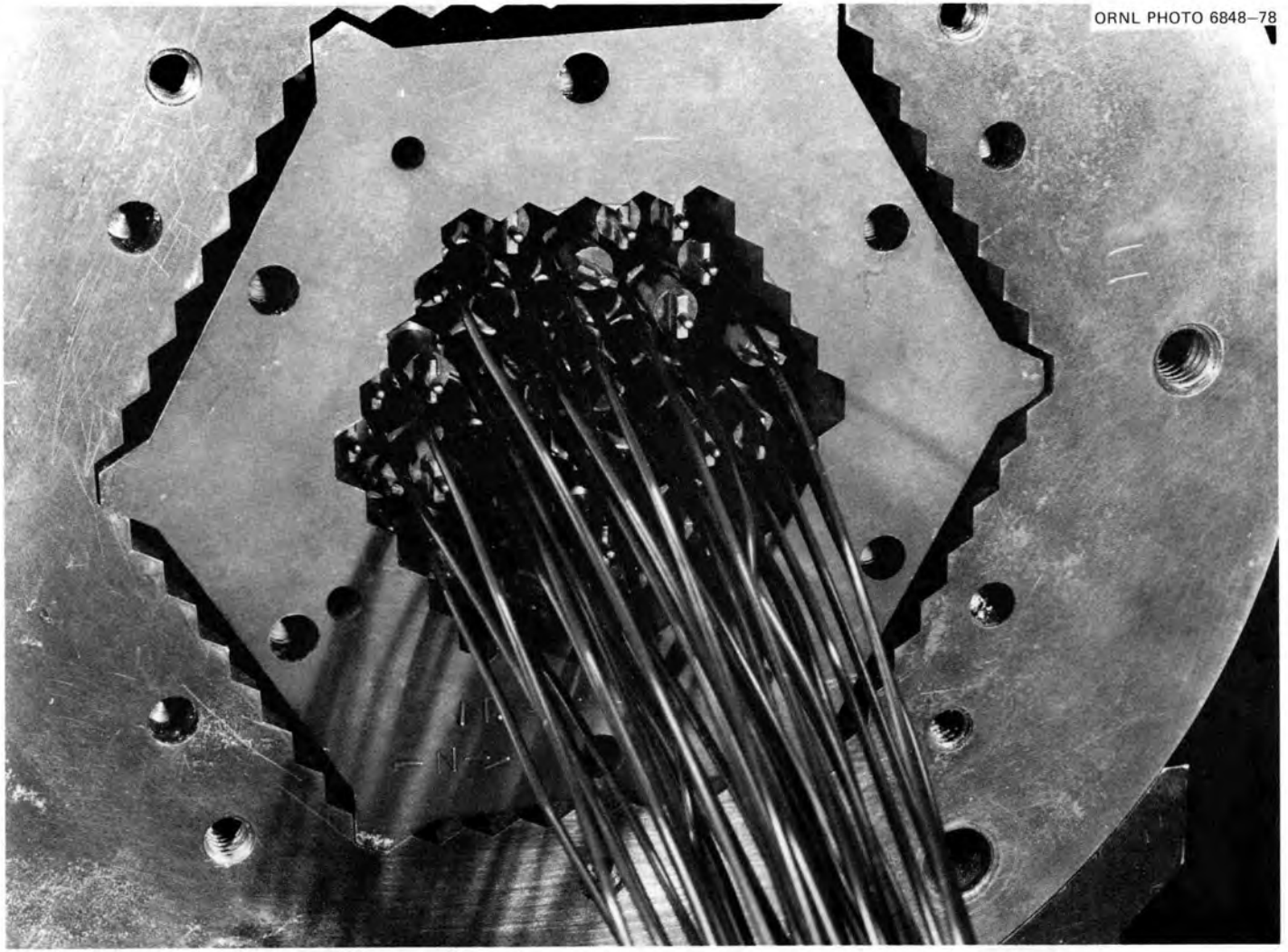

Fig. 3.9. THORS Bundle 9. Close-up photograph of bundle showing assembly after thermal insulation had been installed around the sodium duct.

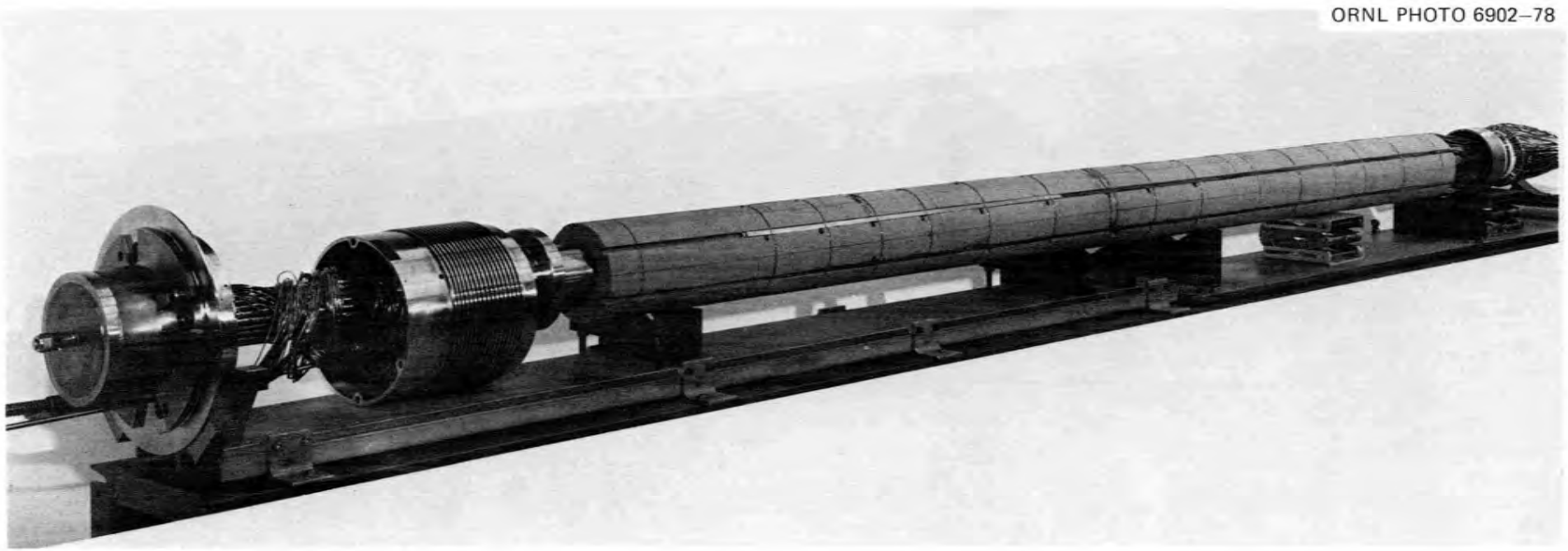

Fig. 3.10. THORS Bundle 9. Photograph of bundle showing assembly after thermal insulation had been installed around sodium duct.

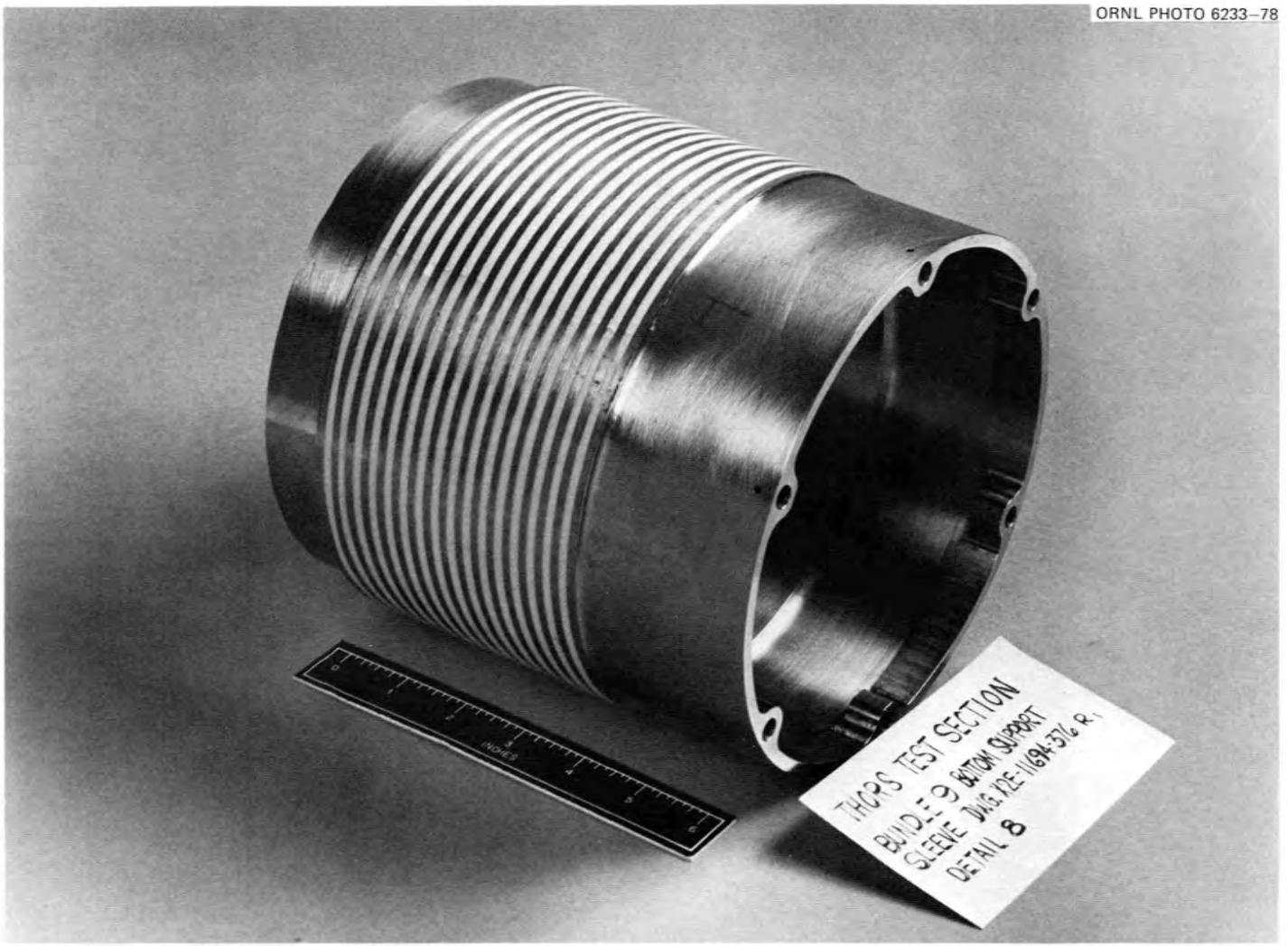

Fig. 3.11. THORS Bundle 9. Close-up photograph of male side of labyrinth seal.

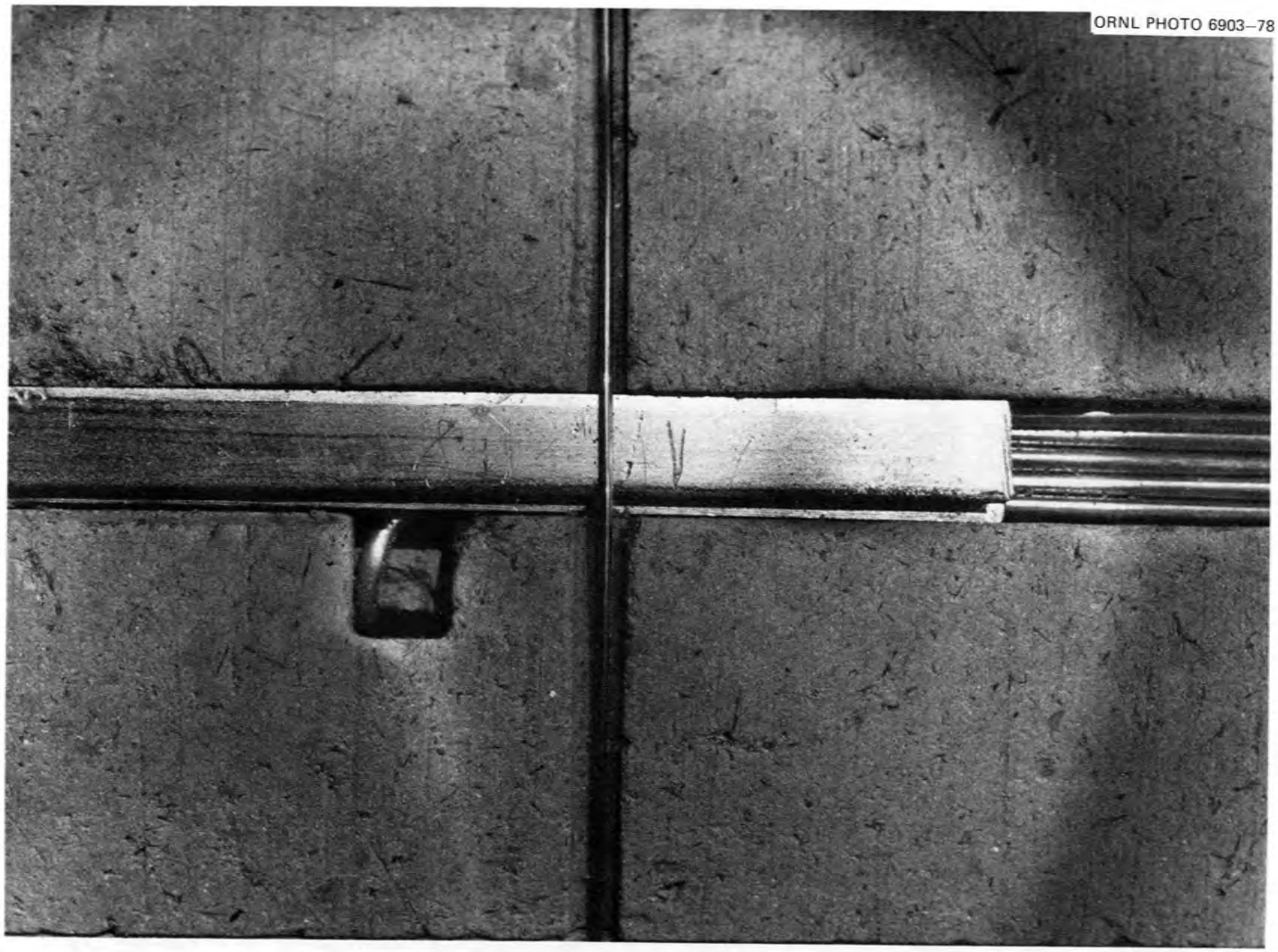

Fig. 3.12. THORS Bundle 9. Close-up photograph showing installation of duct wall thermocouples through thermal installation. Detail method of holding thermocouple junction in close contact with sodium duct is also shown.

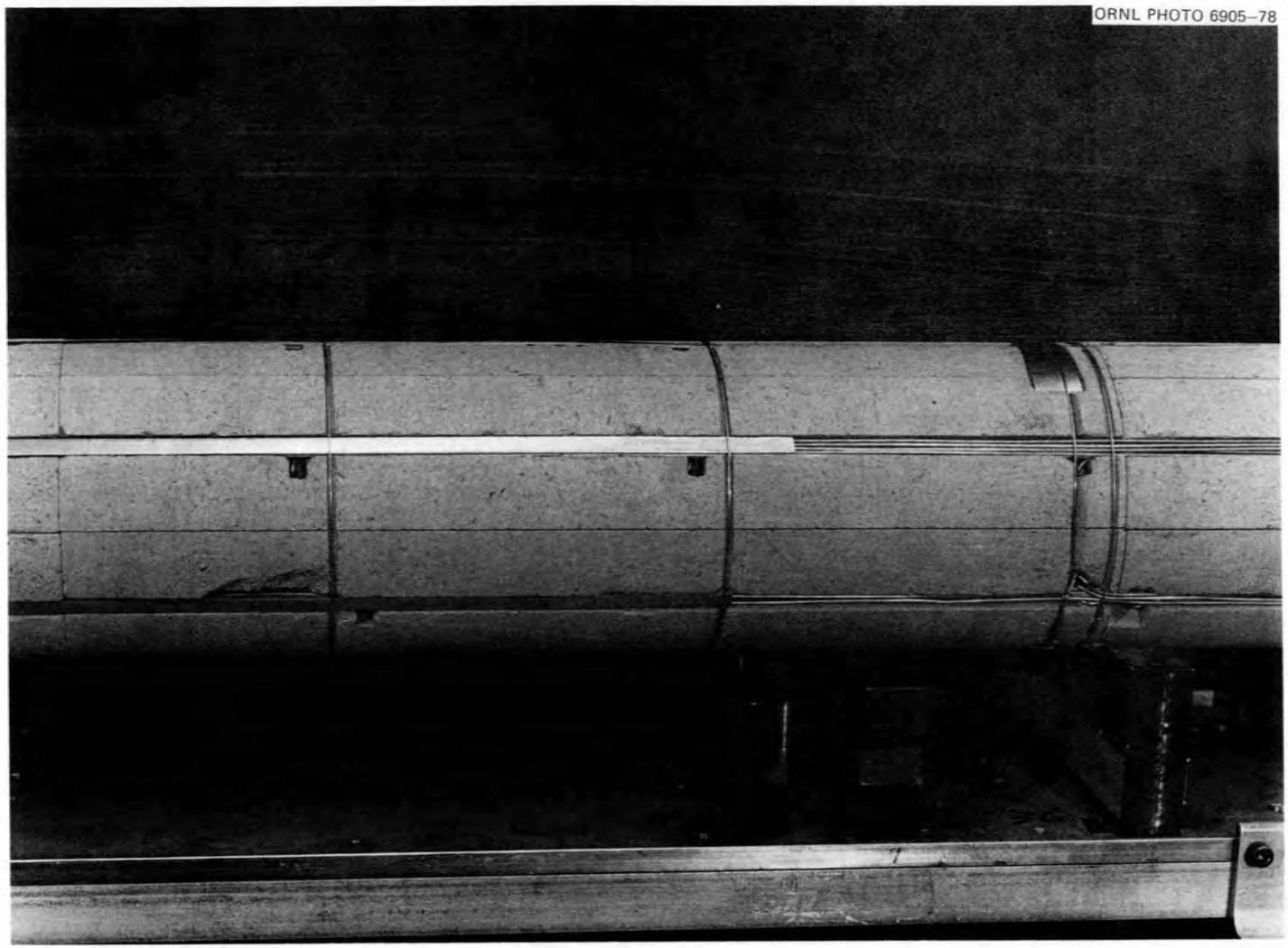

Fig. 3.13. THORS Bundle 9. Photograph taken during assembly showing duct wall thermocouple installation.

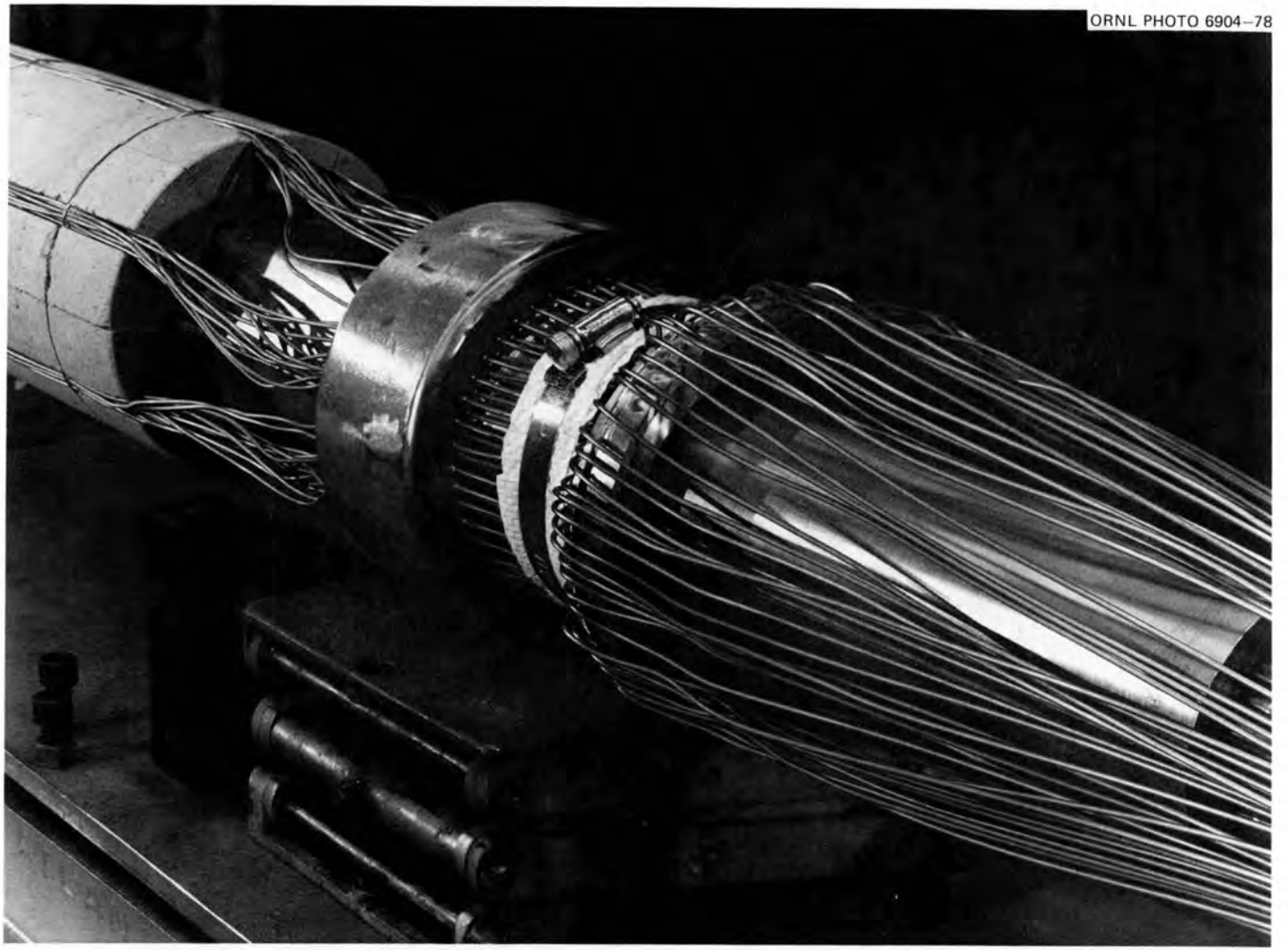

Fig. 3.14. THORS Bundle 9. Photograph showing bulkhead brazement for 90 duct wall thermocouples. An outer jacket was installed over thermal insulation at a later date.

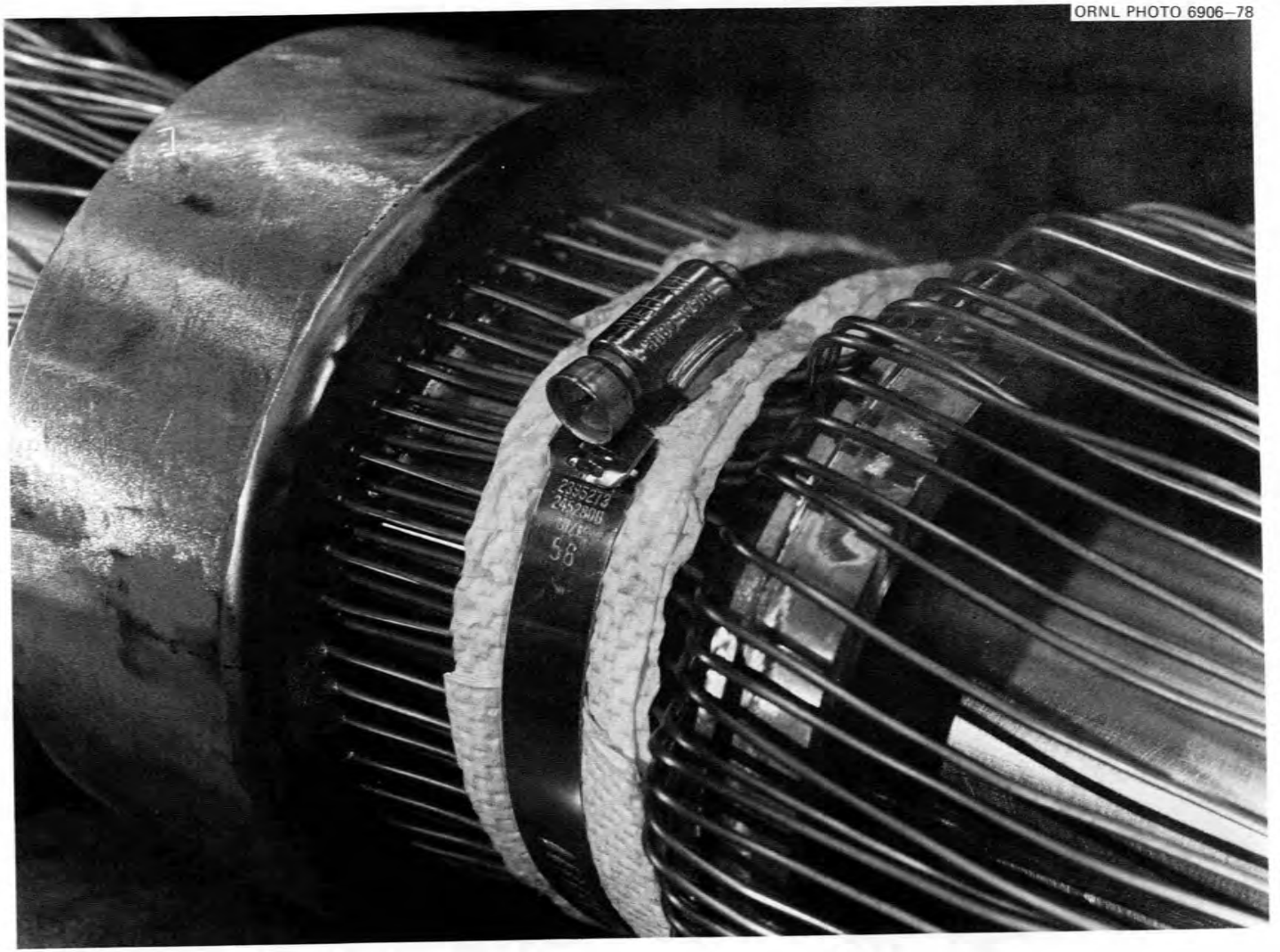

Fig. 3.15. THORS Bundle 9. Close-up photograph of 90 duct wall thermocouples that are brazed into a bulkhead.

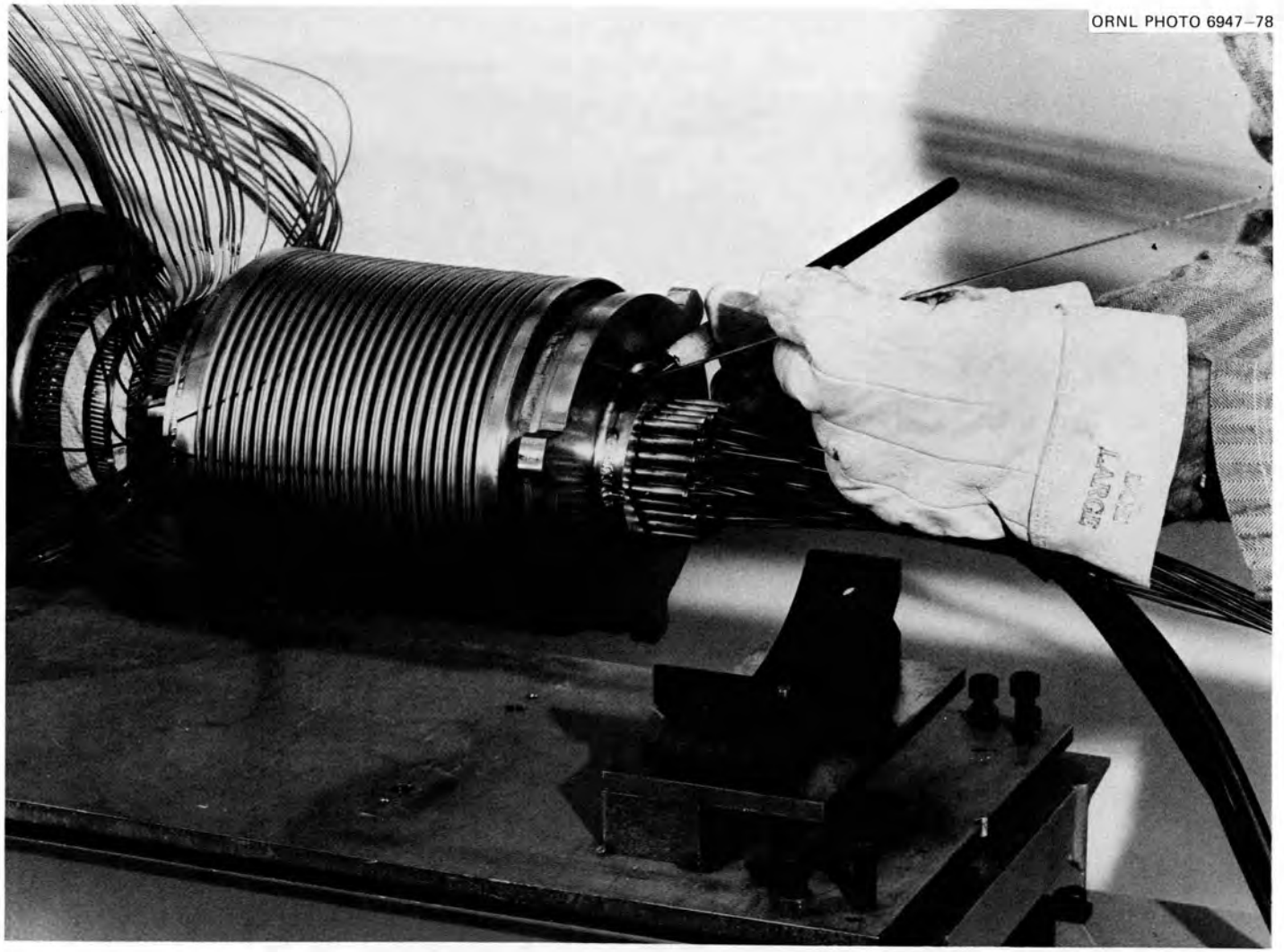

Fig. 3.16. THORS Bundle 9. Photograph of bellows used at top end of bundle to accommodate thermal expansion between sodium duct and outer jacket of thermal insulation.

Fig. 3.17. THORS Bundle 9. Photograph of bellows and duct wall thermocouples at top end of bundle assembly.

ORNL PHOTO 6946-78

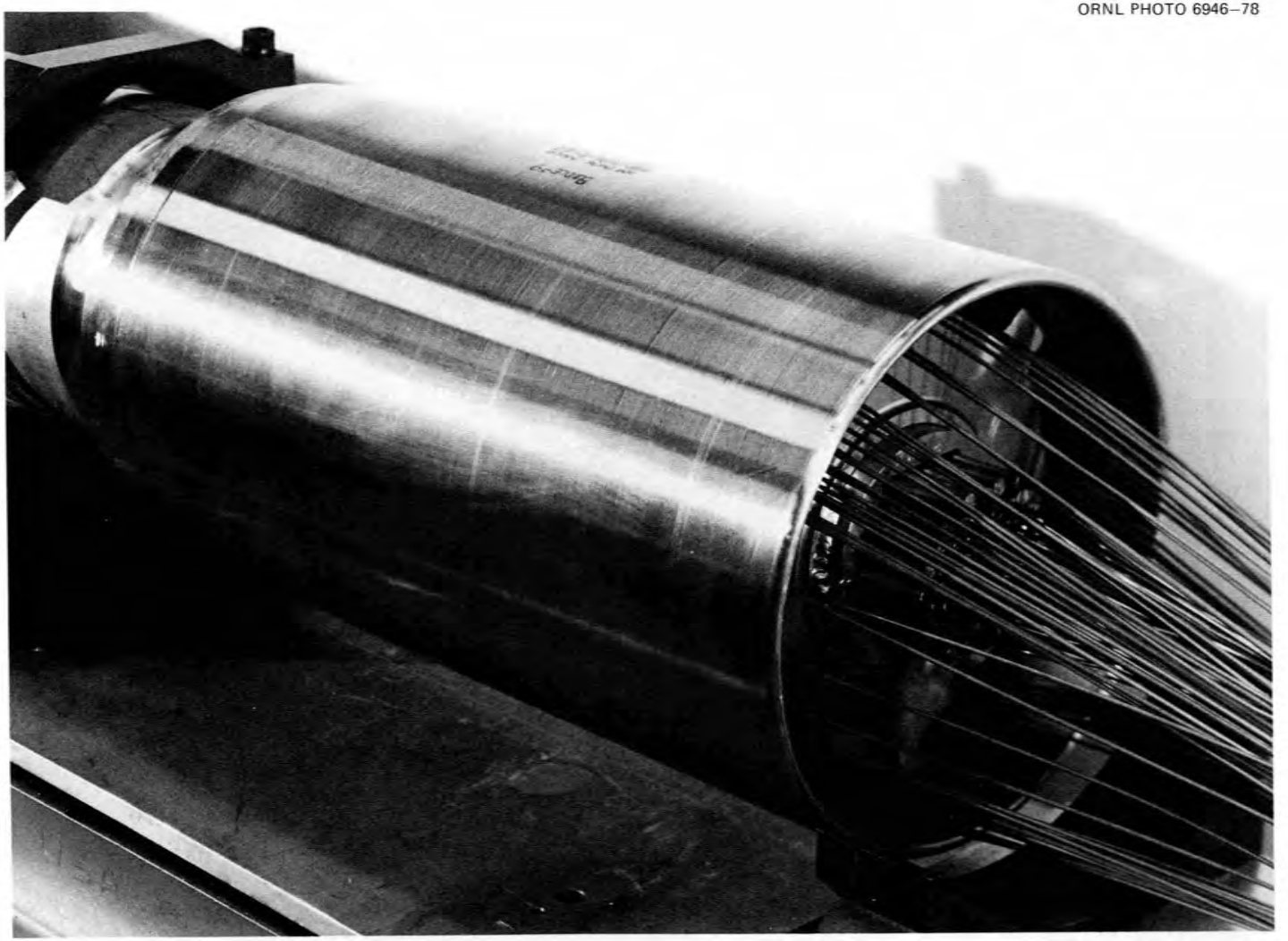

Fig. 3.18. Photograph of same area as in Fig. 3.17 with guide sleeve installed over bellows.

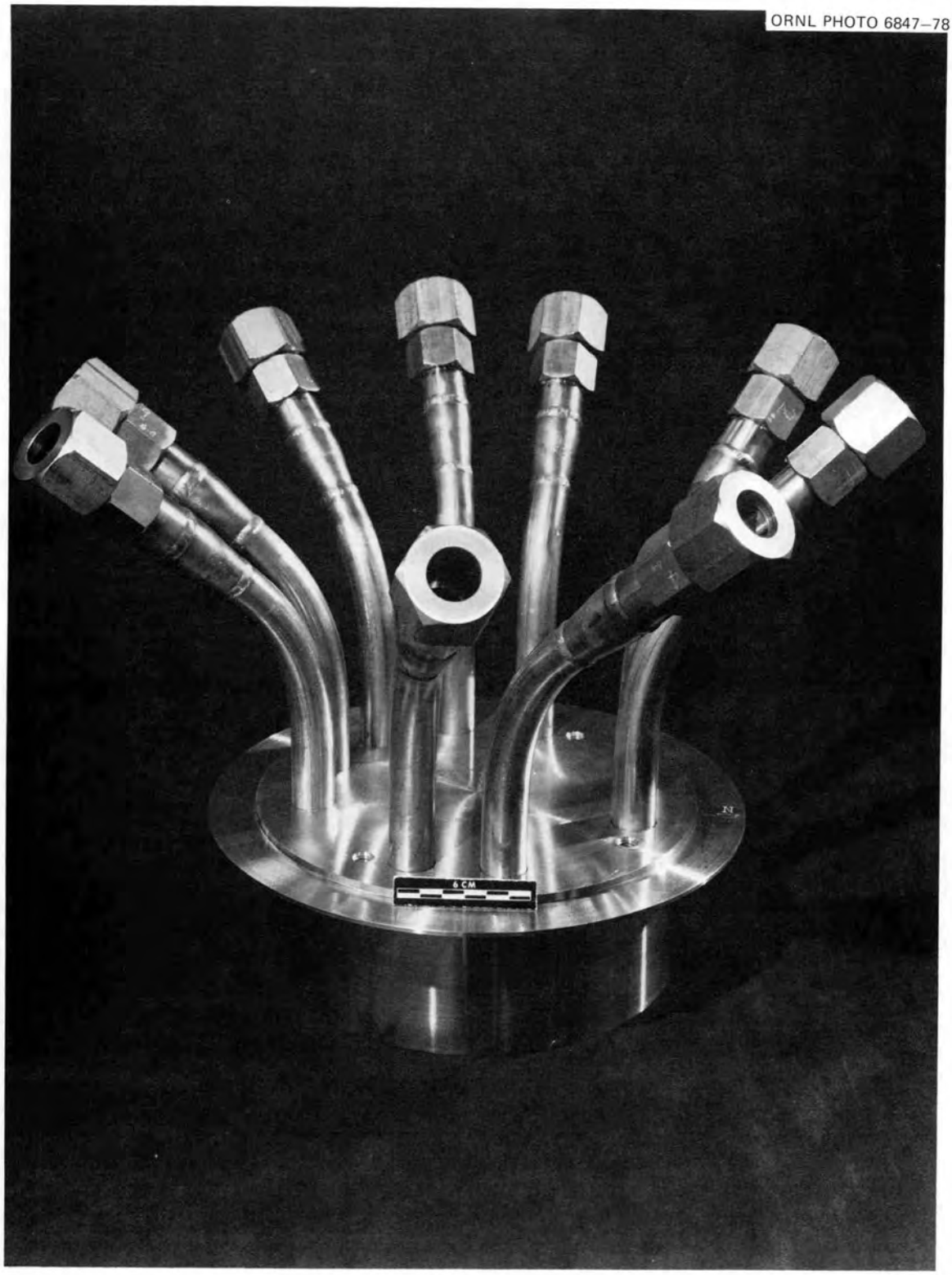

Fig. 3.19. Photograph of upper closure to be installed into bundle housing. Conax seal glands shown at top are used to seal duct wall SFGP wire-wrap thermocouples as they exit sodium containment system.

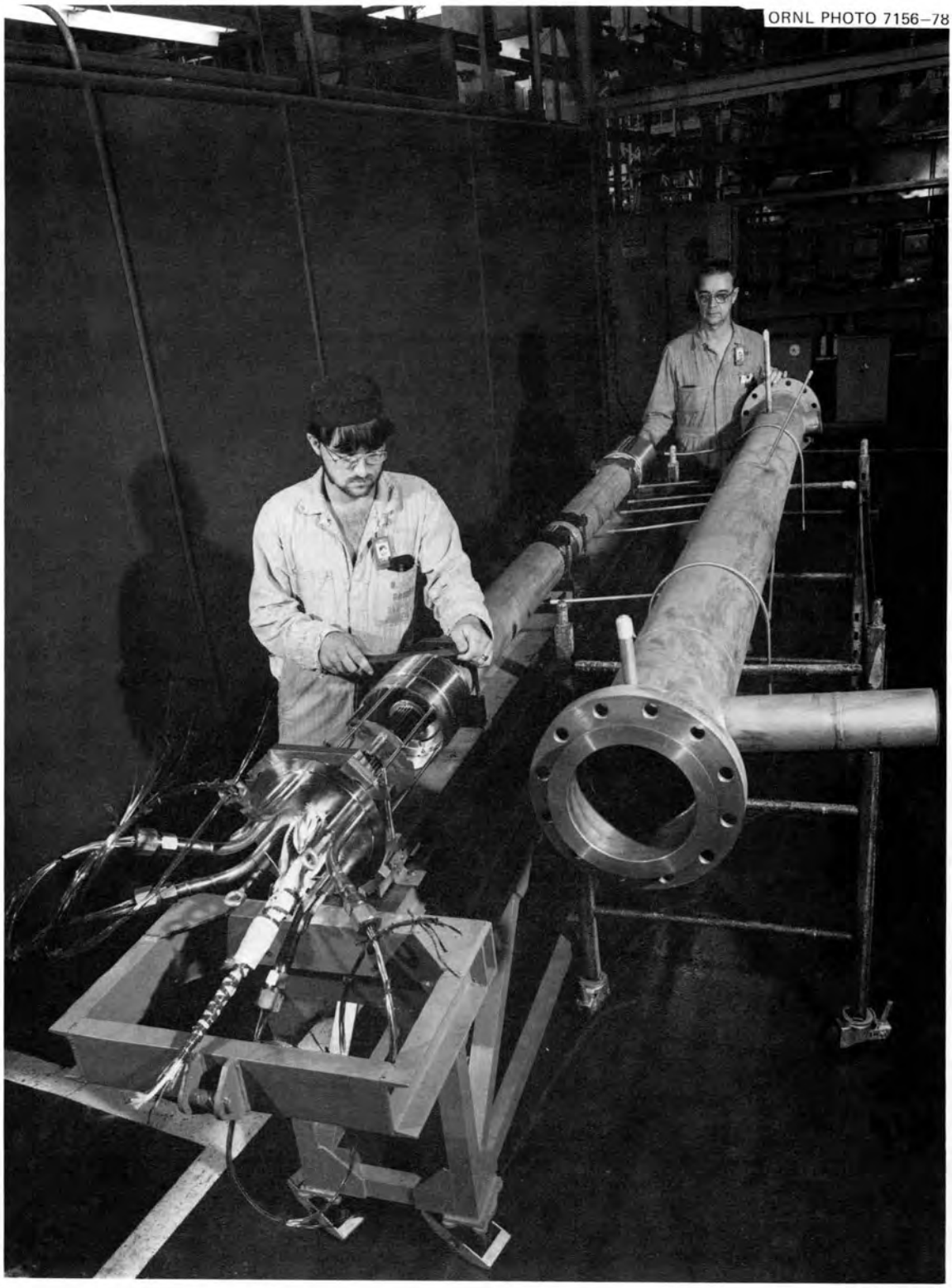

Fig. 3.20. Photograph of bundle assembly before installation into containment housing.

Figure 3.16 is a photograph of the end of the bundle and shows the thermocouple bulkhead brazement, the bellows that is used to accommodate the thermal expansion between the sodium duct, and the outer container that surrounds the thermal insulation. Figure 3.17 is a photograph of the duct wall thermocouples and the bellows prior to the installation of a protective guide sleeve over the bellows.

Figure 3.18 is a photograph of the protective guide sleeve that has been installed around the bellows. The wires at the right side of the picture are the wire-wrap thermocouples from SFGPs and the duct wall thermocouples. Figure 3.19 is a photograph of the upper closure for the test bundle containment housing. The pipes and fittings projecting upward will accommodate the duct wall thermocouples and the wire-wrap thermocouples for the SFGPs. The sodium seal is made with a conax-type fitting that tightly crushes lava around the thermocouple wires. A sodium-freeze seal on the pipe extension is used as further protection against sodium leaks. Figure 3.20 is a photograph of the final assembly before installation in the containment housing and shows the jacket over the thermal insulation, the male portion of the labyrinth seal, and the bellows used to accommodate the thermal expansion. The 309 thermocouples attached to the bundle assembly are also shown.

The computer-controlled data acquisition system (CCDAS) is being modified to provide the capability needed for Bundle 9. The modification consists of increasing the number of data channels from 256 to 512; adding a magnetic tape unit, 192 channels of active filters, and a cathode-ray tube (CRT) terminal; and changing the signal lead termination connectors. The increase in the number of data channels is required to measure the additional thermocouple signals installed in the Bundle 9 assembly. The new magnetic tape unit will allow the CCDAS to copy the data tape before it is sent to the computer center for processing, a QA requirement. The CRT terminal is being added to provide hard-copy graphic capability at the test facility. The signal lead connectors are being changed to increase the reliability of the connection point used to switch the CCDAS from one test facility to another.

At present, there are five 65.6°C (150°F) thermocouple reference boxes installed for the DAS to ensure a constant reference temperature.

Each unit is capable of accepting 50 signals. In order to accommodate the additional thermocouples in Bundle 9, five new reference boxes have been installed; ambient-temperature reference boxes are used instead of the previously used 65.6°C (150°F) heated boxes. Each new reference box contains a resistance temperature device (RTD) to provide a means of monitoring the reference temperature. The existing reference boxes will be modified by installing an RTD in each box and disabling the heating circuit.

References

1. M. H. Fontana and J. L. Wantland, *LMFBR Safety and Core Systems Programs Progress Report for July–September 1976*, ORNL/TM-5753.
2. M. H. Fontana and J. L. Wantland, *Breeder Reactor Safety and Core Systems Programs Progress Report for July–September 1977*, ORNL/TM-6158.
3. M. H. Fontana and J. L. Wantland, *Breeder Reactor Safety and Core Systems Programs Progress Report for April–June 1978*, ORNL/TM-6558.
4. M. H. Fontana and J. L. Wantland, *LMFBR Safety and Core Systems Programs Progress Report for April–June 1976*, ORNL/TM-5699.
5. M. H. Fontana and J. L. Wantland, *Breeder Reactor Safety and Core Systems Programs Progress Report for January–March 1978*, ORNL/TM-6439.

4. SODIUM BOILING TEST FACILITY

P. W. Garrison P. A. Gnadt
R. H. Morris B. H. Montgomery

The SBTF is a single-channel sodium loop developed at ORNL to investigate free and low flow forced convection boiling dynamics and heat transfer for conditions representative of hypothetical breeder reactor LOPI accidents. This facility is described in detail in Refs. 1 and 2. An elevation view of the SBTF is presented in Fig. 4.1.

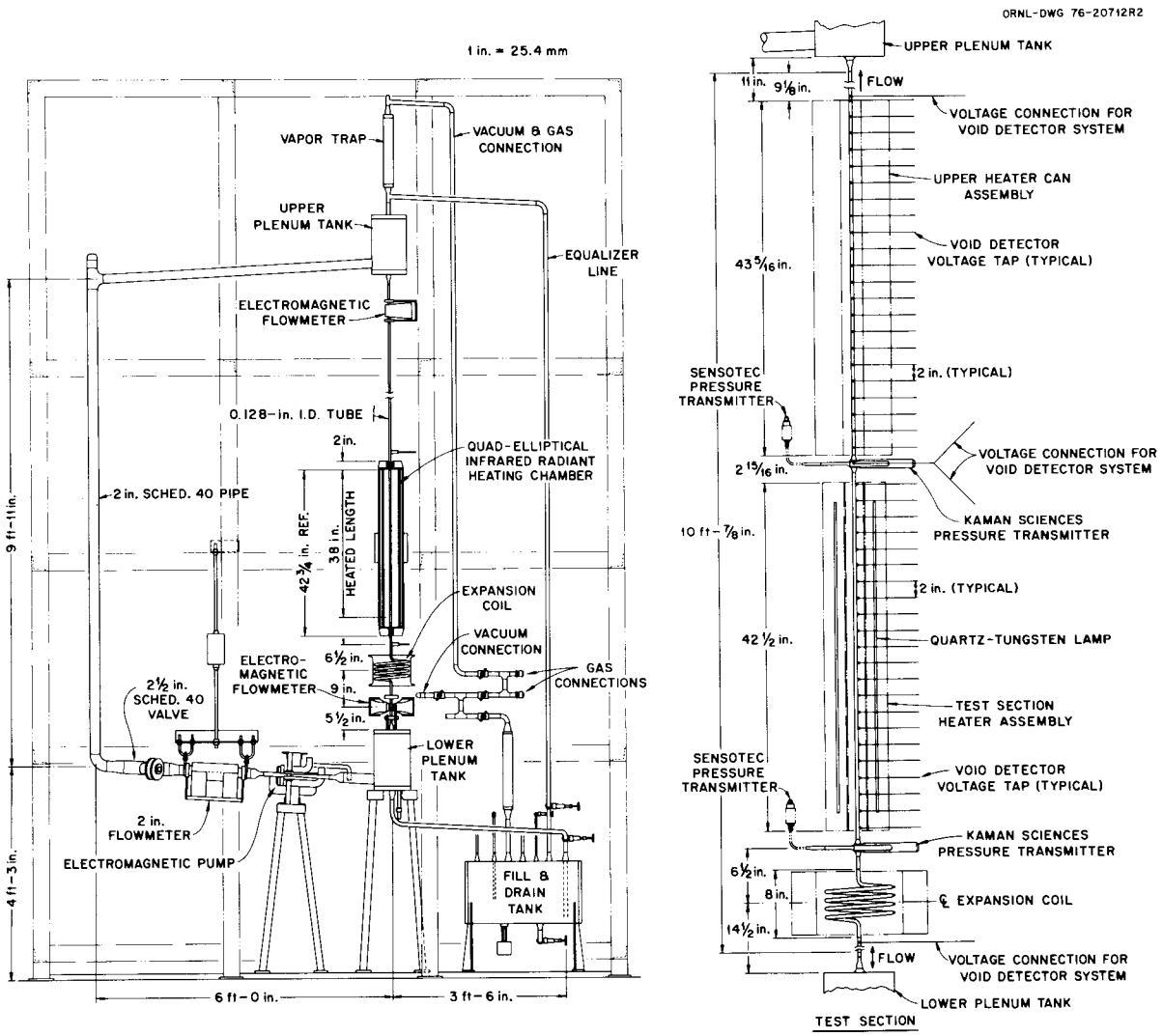

Fig. 4.1 SBTF — elevation view (1 in. = 25.4 mm).

Facility checkout and calibration operations were completed, and the Phase I T-100 test series (Ref. 1) was begun during this period. Boiling conditions were first achieved August 11, 1978. Boiling data were taken at a series of test-section powers between 0.4 and 1.0 kW. Throughout this range, the boiling was unstable, characterized by periods of boiling followed by periods of single-phase liquid flow. An example of this behavior is given in Fig. 4.2. The test-section power for this run was 0.46 kW. At boiling inception, the inlet (FE-1B) and outlet (FE-5B) flowmeters suggest the expulsion of liquid upstream and downstream of the boiling region. The temperature data are taken from thermocouples attached to the outer surface of the channel and located in the heated region. Thermocouple TE109 is located near the inlet of the heated zone. The void detection systems consist of a series of voltage taps located along the length of the test section; output from a series of void detectors in the heated region is presented.

Data taken at higher powers were qualitatively similar, but the duration of the single-phase liquid flow period decreased as test-section power was increased. At a test-section power of 0.8 kW, near-continuous boiling was indicated for a period of ∿45 min.

During a test conducted on August 24, 1978, at a test-section power of 1.0 kW, the system momentarily reverted to single-phase liquid flow after a period of ∿12 min of continuous boiling. The subsequent rise in wall temperature tripped the high-temperature scram limit 1010°C (1850°F) and shut down the system. Posttest inspection revealed that the test section had buckled, as indicated in Figs. 4.3 and 4.4. This buckling was caused by all or a combination of the following factors: (1) compressive load imposed by inlet coil; (2) eccentric load associated with the upper instrumentation adapter; and (3) larger-than-expected high-temperature creep behavior of the test-section material (Hastelloy X). The function of the inlet coil is to accommodate the thermal growth of the test section that occurs over the wide operating temperature range. Pretest analysis indicated that the load imposed by the coil was less than 90 N (20 lb_f), which is significantly less than the predicted buckling load on available Hastelloy-X properties. Information on the high-temperature behavior of Hastelloy X is limited.

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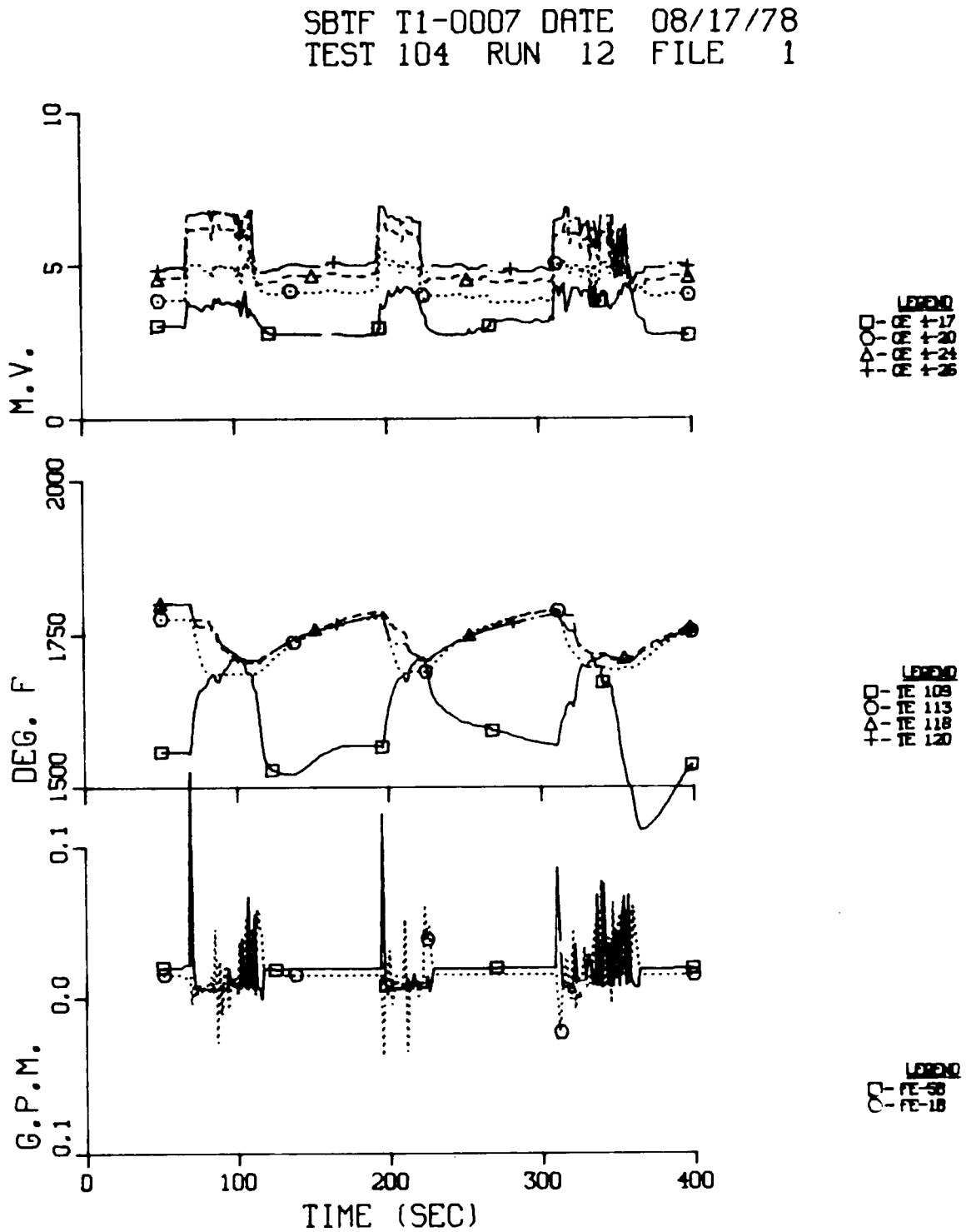

Fig. 4.2. SBTF test 104 run 12, free convection boiling at test section power of 0.46 kW and pressure of 109 kPa abs (15.8 psia).

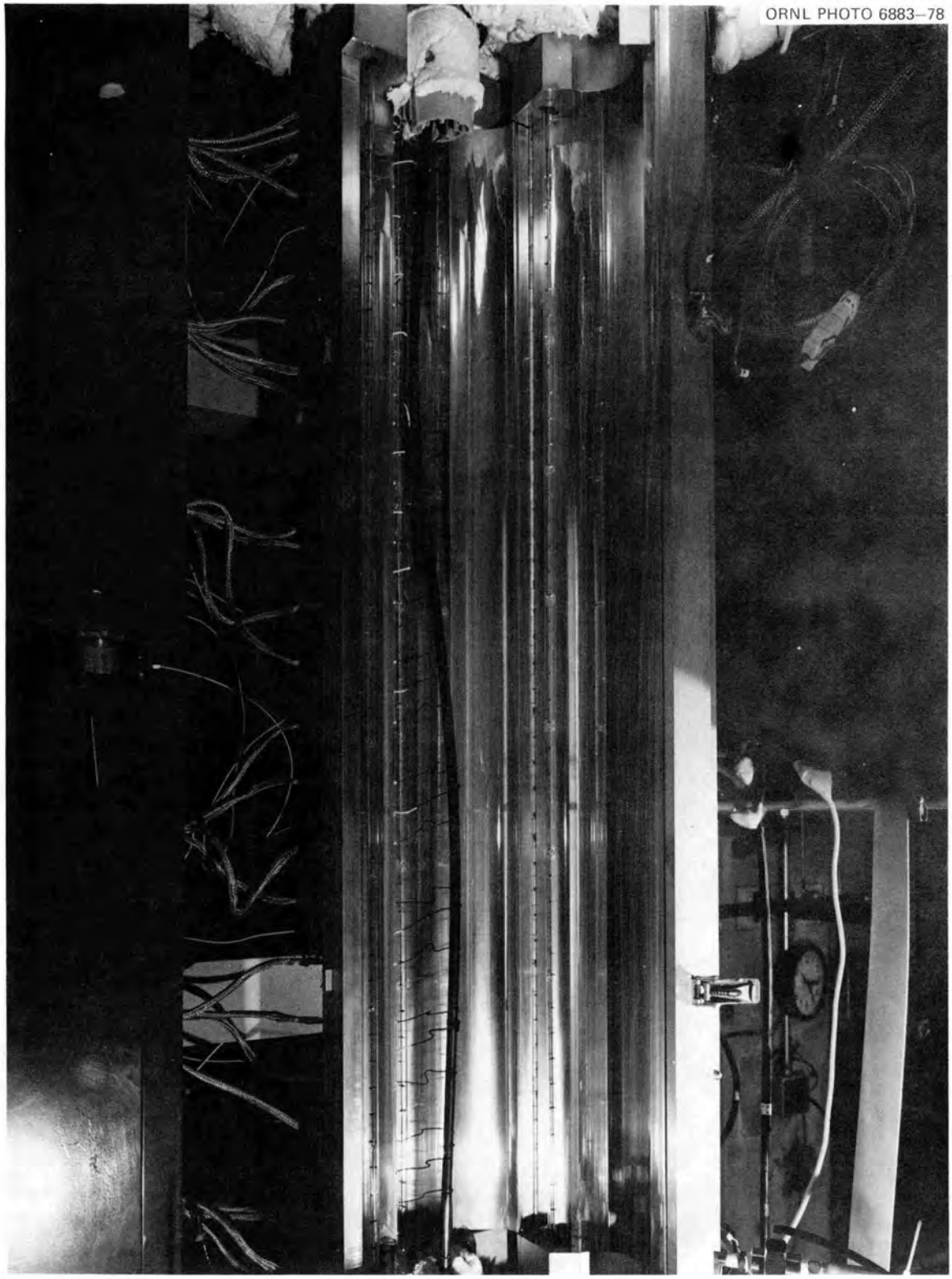

Fig. 4.3. Photograph of SBTF test section and radiant heat furnace showing buckling that occurred during testing.

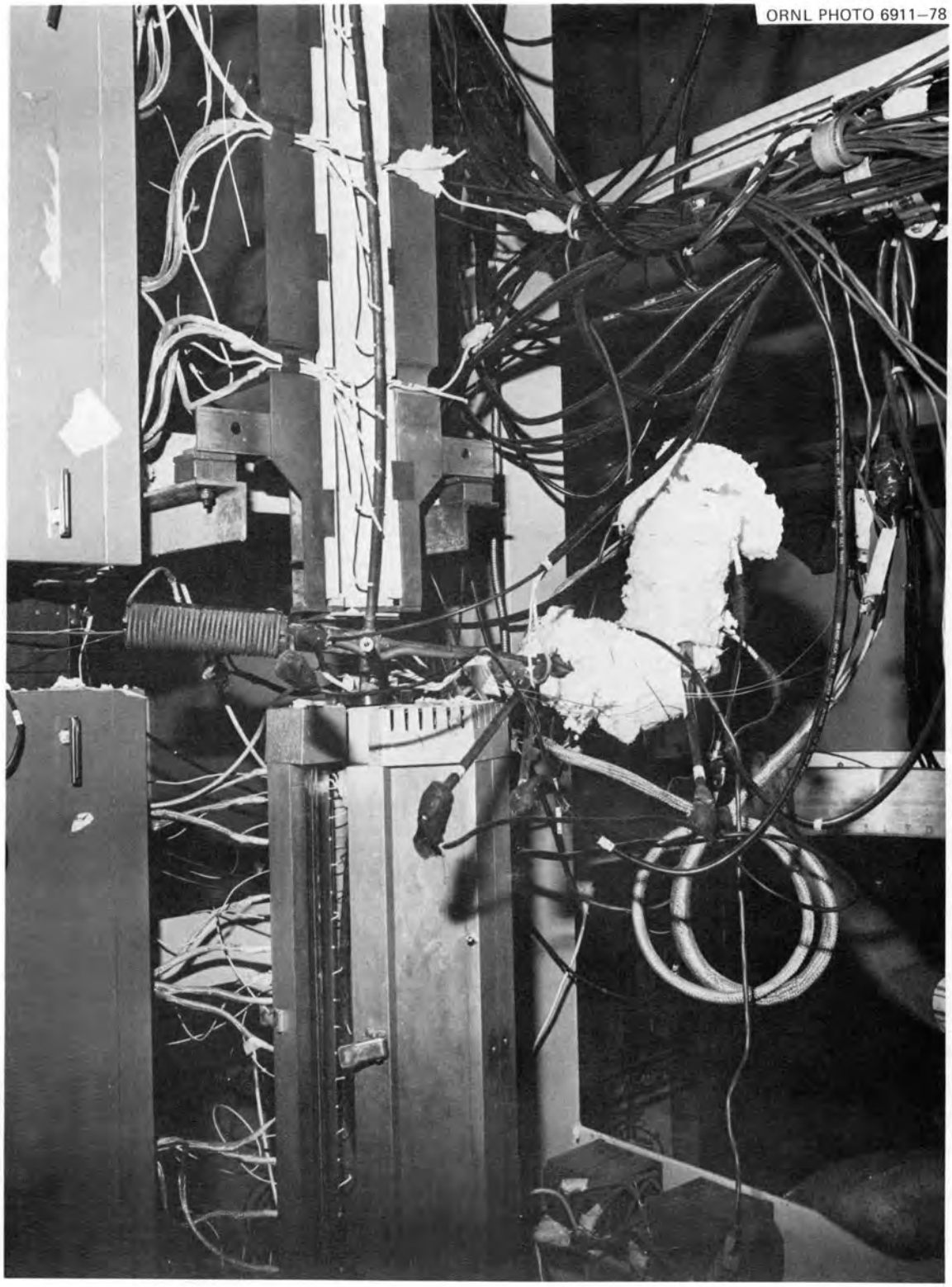

Fig. 4.4. Photograph of SBTF showing buckling that occurred during testing. Instrumentation adapter that contributed to the buckling moment is shown just above test section furnace.

The buckled test section has been removed and replaced with a new test section. The upper instrumentation adapter has been eliminated to reduce eccentric loading of the test section. A counterweight system (Fig. 4.5) has been installed between the inlet coil and the furnace

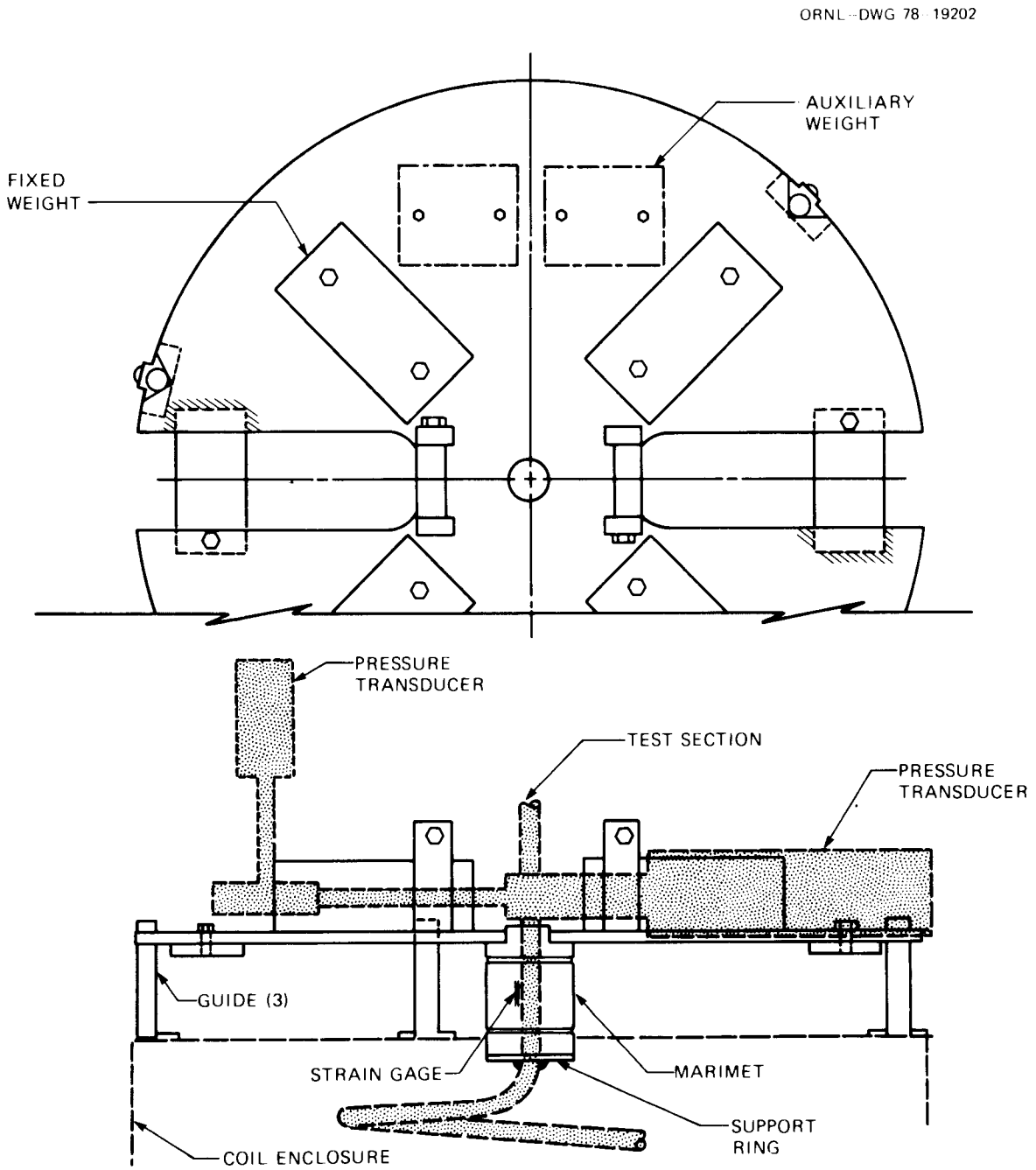

Fig. 4.5. SBTF — test section counterweight.

such that the test section remains in tension throughout the operating regime. A strain gage has also been installed to monitor the test-section stress condition.

Phase I testing will resume pending DOE approval.

References

1. M. H. Fontana and J. L. Wantland, *Breeder Reactor Safety and Core Systems Programs Progress Report for October–December 1977*, ORNL/TM-6288.

2. M. H. Fontana and J. L. Wantland, *Breeder Reactor Safety and Core Systems Programs Progress Report for January–March 1978*, ORNL/TM-6439.

5. MATERIALS FOR FUEL PIN SIMULATION

There was no activity in this area during the reporting period.

6. NEUTRONICS OF A DISRUPTED CORE

G. F. Flanagan E. M. Oblow
S. N. Cramer C. V. Parks
T. J. Hoffman E. T. Tomlinson

The purpose of this project is to develop a three-dimensional quasi-static Monte Carlo code that is appropriate for coupling with existing hydrodynamic codes for use in breeder reactor core disassembly calculations, to test and verify neutronics methods for the analysis of a disrupted core using existing methods, and to use generalized sensitivity methods to assess the sensitivity of safety calculations to input parameter variations.

6.1 Development of a Transient Monte Carlo Code

Due to difficulty in obtaining sufficiently small statistical errors in the calculation of reactivity, further development of SPQR has been halted. At present, work is directed toward completion of the previously described small problems with the current version of the code. The use of Monte Carlo methods for reactivity calculations in systems where only absorption or fission cross sections have been altered has met with some success; however, changes in scattering cross sections, as in accident simulation calculations, create unacceptably high statistical errors. Recent work by E. M. Gelbard at ANL illustrates these difficulties (see ANL/FRA/TM-111).

6.2 Reactivity Determination of a Disrupted Core

Several slab penetration calculations with explicit bubble distributions have been performed using the Monte Carlo method. In order to conform to Behren's model, the bubble positions were randomly selected in the slab volume. The calculations serve as benchmark calculations for the probability table method. The results of these calculations indicate that some modification of the probability table method is necessary when neutrons cross an interface between bubbly and nonbubbly media. Bubble-worth calculations have been performed using the probability table

method, the Benoist method, and the modified-probability table method. A paper on this work is being prepared with E. V. Vaughan of Atomics International.

6.3 Safety Analysis Sensitivity Studies

The sensitivity methodology for a time-dependent heat conduction/convection problem, with radial heat conduction in the pin and axial heat conduction and convection in the coolant, has been developed. Report No. ORNL/TM-6303, entitled "Sensitivity Theory for Reactor Thermal-Hydraulics Problems," by E. M. Oblow, which represents our initial approach, has been submitted for publication to *Nuclear Science and Engineering*. The methodology has been reviewed with respect to application to the MELT neutronics, thermal-hydraulics code. Basically, equations are first derived for $dT/d\alpha$ [the derivative of a temperature field with respect to some general parameter (α) for both the fuel and coolant] and its adjoint. The MELT and ORNL/TM-6303 formulations are equivalent for the fuel region; however, differences arise in the coolant equations. The MELT formulation considers axial conduction in the coolant as well as convection compared to ORNL/TM-6303, where only the convective terms are considered. It also considers heat loss to structural material. Adjoint equations appropriate to the specific case in MELT have now been formulated, leading directly to expressions for response sensitivities that can be obtained independent of any recalculation.

The next step in the procedure is to derive suitable numerical schemes for the solution of the set of adjoint equations. Crank-Nicholson numerical schemes are employed, which evaluate time derivatives at the midpoint of the time step with spatial contributions coming from the endpoints of the interval. At the present time, these techniques are being implemented in the MELT code itself. From a programming point of view, several substantial questions still remain. These are primarily concerned with the volume of data generated (e.g., temperatures, pressures, flow rates, reactivity coefficients, etc.) at each of many (a few thousand) time steps internally selected within MELT for each channel, reorganization of the calculation to reduce the severe core storage

requirements, and the preservation of all the required information for the adjoint calculation. Currently, the "restart tape" (written in MELT) is planned as the source of input to the adjoint calculation. To reduce the volume of data treated, edits will be written at fewer time intervals and interpolated where necessary for the adjoint calculation if the behavior of the adjoint will be amenable to this procedure. Physical constants will be reevaluated in the adjoint calculation as necessary. This programming is now under way using the sample problem described below as an example.

The physical basis behind this problem (conveyed to ORNL by L. Stewart at HEDL) is to study the behavior of the hot pin in a two-channel mockup of the Fast Test Reactor (FTR) during a reactivity transient with scram. The steady-state conditions were calculated at 80,000 MWd/t burnup at a power level of 391.68 MW(t). The core was described as having two channels, the hot channel (217 pins) and the remainder of the core (15,624 pins). Full flow was assumed throughout the transient, which consisted of a $0.243/sec reactivity insertion for 10 sec. The reactor scrammed on power level at 455.4 MW(t) at 0.76 sec into the transient. The temperatures in the hot channel are followed for 100.0 sec.

Because the current state of the generalized perturbation capability does not include point kinetics and reactivity feedback models, it was decided to model this problem as a single hot channel driven by a power trace rather than a reactivity trace. To this end, a single-channel model of the hot subassembly was created. The steady state for the single-channel case was formulated to model the hot channel of the two-channel case exactly. The *total* power trace from the two-channel case was used to drive the single channel. The peak fuel temperatures at the beginning of the transient were in good agreement [<20°C (40°F)] with the two-channel case. Larger differences occurred at longer times due to differences in Doppler, sodium worth, etc. This sample problem will suit the needs as a test case for MELT.

Validation of the MELT sensitivity analysis for the sample problem described above will include direct recalculation for a limited number of selected perturbations in addition to detailed comparisons against the more limited physics-checking code developed at ORNL as part of this

effort. The local capability now includes calculation of temperature distributions and heat fluxes corresponding to the geometry described in ORNL/TM-6303. Algorithms and programming for the adjoint are now complete and are currently being debugged on an analytic problem. A code to fold the forward and adjoint distributions to generate sensitivity profiles is currently being written. Theoretical approaches toward inclusion of neutronics feedback in the sensitivity system are under development.

The local physics-checking thermal-hydraulic tool used to test the adjoint sensitivity theory as well as to validate the theory when implemented in MELT is now capable of treating the time-dependent, coupled conduction/convection problem. Work focused on providing the capability for this code to retrieve input from the steady-state analysis code STEADY (used to provide initial conditions for MELT). Additional effort was devoted to improving the iteration capability necessary for the solution of this nonlinear problem. The sample problem is being solved with this tool to provide a comparison for the same problem when solved using the adjoint version of MELT.

Much effort was spent in the initial programming of the adjoint version of MELT. The coding is based on the formulation of the adjoint temperature equations with appropriate boundary conditions, derived by C. V. Parks, for application to MELT. A memo describing this derivation is presently being prepared for internal distribution. Routines to solve for the transient adjoint temperatures $T^*(r,z,t)$ have been developed. Checkout of these routines is awaiting solution to various data management problems discussed below.

The data management problems which exist deal with obtaining the forward temperatures $[T(r,z,t)]$ on files so that they can readily be accessed during solution of $T^*(r,z,t)$. The forward temperatures are needed to compute the physical properties (which are temperature dependent). Presently, MELT can be modified to write the forward temperatures for each channel at every time step. For use in solving for $T^*(r,z,t)$, the forward temperatures need to be on a file listing each time with all the channel temperatures. This data management problem of writing and rewriting files will be solved next month.

A substantial amount of time went into rewriting various routines in MELT to delete the kinetics, pressure evaluation, fuel coolant intention (FCI), etc., that are not being handled presently in the adjoint solution. The routines were also adjusted to handle the adjoint solution based on input from a restart tape and some namelist input. The restart tape has all the data from the last time step of the forward solution. It is needed for various control parameters and constant physical properties. The namelist input was added for the adjoint to primarily input some geometrical and physical properties that are needed to solve for T*(r,z,t) but are not available on the restart tape. The conditions of T*(r,z,t) at the final time step [where solution to T*(r,z,t)] have also been programmed into the adjoint MELT (MELTADJ).

The SHUFL module, which rearranges channel-ordered files to time-step-ordered files, has been made operational.

The MELT code has been reprogrammed to calculate temperature adjoints. Debugging of MELTADJ began in mid-August after SHUFL was completed. The present phase involves testing and reprogramming MELTADJ to ensure that it is actually doing what it was programmed to do (by continually going through major subroutines step-by-step) and to ensure that the adjoint temperatures are physically reasonable (based on experience and physical insight). Final verification of the transient adjoints is planned after the sensitivity code is written by comparing sensitivity predictions with direct reruns for perturbations from a reference case. The physics testing code developed by Weber is still presently incompatible with MELT, because the former does not include sintered fuel, gap shrinkage, etc. These difficulties are being worked out and sensitivities from this code based on a MELT problem are hoped for in the next few months.

A trip to ANL was made to discuss the safety sensitivity efforts with members of the RAS Division. The PLUTO2 code will also be obtained from H. Wider to assess the applicability of generalized perturbation methods to the detailed fuel motion models used in PLUTO2.

7. ENVIRONMENTAL ASSESSMENT OF ALTERNATE FBR FUELS

V. J. Tennery J. C. Mailen
W. D. Bond P. S. Rohwer

Two technical reports were prepared for publication during this period. The first, ORNL/TM-6474, *Environmental Assessment of Alternate FBR Fuels: Radiological Assessment of Airborne Releases from Thorium Mining and Milling*, V. J. Tennery et al., was submitted in May in its prepublication form to the Assistant Director, Reactor Safety, RRT, thus satisfying the requirement of milestone H.8. Radiological dose commitments resulting from airborne effluents from the mine/mill facilities were based on the best available site-specific data for Lemhi Pass, Idaho, and Wet Mountains, Colorado. The principal dose-contributing radionuclides at both sites were found to be ^{228}Ra and ^{220}Rn, plus the daughters of ^{220}Rn. Total body doses to the maximally exposed individuals at the respective sites were 2.4 and 3.5 millirems, while population dose commitments were calculated to be 0.05 and 0.3 man-rems.

The second report, ORNL/TM-6493, *Environmental Assessment of Alternate FBR Fuels: Radiological Assessment of Reprocessing and Refabrication of Thorium/Uranium Carbide Fuel*, V. J. Tennery et al., completed the requirements of milestone H.9 and was issued in late August. Tritium was found to be the principal contributor to dose to the maximally exposed individual and the general population; ^{14}C, ^{137}Cs, and ^{232}U also made significant contributions. Reprocessing was the principal source of exposure resulting in total body doses of 3.1 man-rems to the maximally exposed individual and 39.2 man-rems to the general population.

A paper entitled "Thorium Mining and Milling — An Analysis of Potential Radiological Impacts," by H. R. Meyer, J. E. Till, M. G. Yalcintas, E. S. Bomar, and W. D. Bond, was presented by H. R. Meyer at the Health Physics Society Annual Meeting, Minneapolis, Minnesota, June 18–23, 1978. A more detailed version of the paper entitled "An Analysis of Radioactive Airborne Effluents from a Hypothetical Thorium Mine and Mill" is currently undergoing internal ORNL review and is scheduled to be submitted to the journal *Nuclear Safety*. An abstract entitled "Anticipated Radiological Impacts from the Mining and Milling of Thorium for the Nonproliferative

Fuels" has been submitted and will be included in the German-Swiss Society for Radiation Protection Meeting on "Radioactivity and Environment," October 2–6, 1978. J. E. Till will present the paper, a version of which will be published in the conference proceedings.

V. J. Tennery, E. S. Bomar (Metals and Ceramics Division), and H. R. Meyer [Health and Safety Research Division (HASRD)] participated in a successful survey of the Lemhi Pass thorium resource region, which is located on the continental divide between Idaho and Montana in the Bitterroot Mountains. The primary purpose of the trip, siting for meteorological survey equipment, was accomplished by aerial and ground surveillance.

Installation of a tube furnace in a glove box for the study of the pyrophoricity of alternate FBR fuels is essentially complete, and the calibration of flow meters and thermometry equipment is being done.

8. MODEL EVALUATION FOR BREEDER REACTOR RADIOACTIVITY RELEASES

F. O. Hoffman C. W. Miller
D. L. Shaeffer C. A. Little

8.1 Comprehensive Testing of Selected Models

A paper entitled "Validation of Turner Workbook Normalized Air Exposure Predictions for Surface Level Releases Using Hanford Dispersion Data" has been prepared by C. W. Miller, C. A. Little, and F. O. Hoffman. This manuscript will be published in the preprint volume of The American Meteorological Society's Fourth Symposium on Turbulence, Diffusion, and Air Pollution to be held in Reno, Nevada, in January 1979. Briefly, this report suggests that a method of determining stability based on the standard deviation of the wind direction at the point of release, σ_θ, is a more accurate predictor of normalized air exposure than stabilities based on vertical temperature gradients, $\partial T/\partial Z$. Further, the location difference between the predicted and observed maximum normalized air exposure was found to approximate a normal distribution for wind speeds <2 m/sec. The <2 m/sec location differences had a 95% confidence prediction, and observations were not normally distributed but ranged from −50° to +10°. In summary, regardless of the method of stability determination, predicting the magnitude of an air exposure appears to be easier than predicting the location of that exposure.

The fraction of radioactivity deposited by the atmosphere that is retained on vegetation r is a key parameter in models used to estimate the dose to man from contaminated food stuffs and pasture grass. Chamberlain has reviewed a number of experimental measurements of r on grasslands from which he has developed a model relating r to grass density Y_v. This model has been validated in this project with a number of other measurements of r reported in the literature. Model predictions have been found to agree very well with measured values of r for pasture grasses and clover but not as well for other types of vegetation considered. These results suggest that in the absence of site-specific r values, Chamberlain's model could be used to generate values of r for pastures to be used in assessment calculations.

A study has been completed on the importance of incorporating age-dependent data into population dose estimates. These results suggested that little variation occurs in the population dose when specific age-dependent groups are considered, if the source term is represented by a spectrum of radionuclides. However, a source term dominated by a nuclide with known age-dependent characteristics would necessitate further consideration of the dose estimate.

An evaluation of ^{232}Th dose-conversion factors has been performed. This evaluation includes the use of state-of-the-art dynamic models to describe the behavior of internally deposited ^{232}Th. The dose-conversion factors from several sources were compared, with ^{232}Th providing most of the dose given by nuclides in the chain.

D. L. Shaeffer has concluded a report entitled "Validation of a Two-Compartment Model for Transport of ^{131}I in the Pasture-Cow-Milk Pathway." The significant conclusion of this report is that the model-predicted equilibrium values of C_m(pCi/ℓ milk)/K_v(pCi/kg vegetation) were larger by factors of 2.8 and 1.5, respectively, than the time-averaged values reported in the literature for the Dresden and Monticello nuclear power plants. This discrepancy was due to the fact that the reported values were time-averaged over periods that were not sufficiently long. Information derived from a pulsed (accidental) situation can be used for chronic situations provided that the time-averaging interval is sufficiently long.

Two other reports have been completed and are awaiting submission to their respective journals. The first, by D. L. Shaeffer, is entitled "Model Evaluation Methodology Applicable to Environmental Assessment Models" and will be submitted to *Ecological Modelling*. The second, co-authored by D. L. Shaeffer and F. O. Hoffman, is entitled "Uncertainties in Radiological Assessment — A Statistical Analysis of Radioiodine Transport Via the Pasture-Cow-Milk Pathway," and it will be sent to *Nuclear Technology*.

A report entitled "The Validation of Selected Predictive Models and Parameters for the Environmental Transport and Dosimetry of Radionuclides" is nearing completion and will be placed into laboratory review. A draft

of this report will be tendered to G. Sherwood, DOE, as soon as it is available.

8.2 Long-Term Development of a Standard Radiation Dose Assessment Code

The feasibility of developing a standard radiation dose assessment code has been studied throughout the course of this project. After careful consideration, it is our judgment that the development of such a code is neither warranted nor feasible at this time.

There are a number of reasons for this conclusion. Computer codes are often designed for specific purposes and with specific input and output requirements. It would be very difficult to design a standard code to meet the needs of all users. As scientific advances are made in the field of radiation dose assessment, changes are required in the models and parameter values used in the assessment codes. Such changes are generally easier to make in codes that concentrate on specific pathways or processes than in larger, standard codes. In addition, such a standard code would be very large and quite likely would be difficult to debug and maintain. Such codes can easily become machine dependent and hence difficult to transfer from one computer to another.

If a comprehensive radiation dose assessment capability is needed by a given user, it is recommended that a modular concept be used. Pathway- or process-specific codes can be written in which the output of one code can be used directly as input to other codes in the package. The exact design of this package and its modules would depend on the needs of the user.

In any consideration of the radiation dose assessment process, it must be noted that the models and parameter values used are of prime importance rather than the computer codes that implement them. If two or more codes properly implement the same model using the same parameter values, then, ideally, they should give the same results regardless of how different the organization, structure, or output of the code may appear. In the real world, however, a complex model may give rise to numerous seemingly "proper" implementations which may produce different

results because of differences in numerical techniques and other assumptions. This variation ordinarily should be of lower magnitude than that attributable to different choices of models and parameter values. Because of these considerations, however, the mathematical models and parameter values in this project, rather than specific computer implementations, are of primary concern.

8.3 Meetings

On July 27, 1978, F. O. Hoffman and C. W. Miller met with members of the Environmental Planning Staff of TVA. The purpose of this meeting was to discuss possible cooperative programs in the area of model validation.

D. L. Shaeffer and C. A. Little gave a presentation entitled "Effects of the Ratio of Two Measured Environmental Variables with Truncated Lognormal Frequency Distributions" at the mathematical ecology section of The Ecological Society of America Meeting in Athens, Georgia, on August 22, 1978.

F. O. Hoffman, J. E. Till, and S. V. Kaye have been invited by the members of the Fachverband für Strahlenschutz, FRG, to participate in their Twelfth Annual Symposium, "Radioactivity and Environment," at Norderney, FRG, October 2–6, 1978. Full travel expenses for F. O. Hoffman and partial expenses for J. E. Till will be paid by the Fachverband. The following papers will be presented at this meeting:

F. O. Hoffman,* D. L. Shaeffer, and C. W. Miller, "An Evaluation of Uncertainties in Radioecological Models" (Plenary Session).

C. W. Miller and F. O. Hoffman,* "A Critique of Methods for Estimating Plume Depletion and Deposition of Airborne Radionuclides."

D. E. Dunning, Jr., and G. G. Killough, "Internal Radiation Dose Calculations with the INREM-II Computer Program" (presented by S. V. Kaye).

D. C. Kocher, "Calculations of Dose-Rate Conversion Factor for External Exposure to Photons and Electrons" (presented by S. V. Kaye).

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*Presents the paper.

H. E. Meyer and J. E. Till,* "Anticipated Radiological Impacts from the Mining and Milling of Thorium for the Nonproliferative Fuels."

S. V. Kaye and J. E. Till,* "Environmentally Important Radionuclides in Nonproliferative Fuel Cycles."

J. E. Till* and G. G. Killough, "Scenarios for ^{14}C Release to the Atmosphere by the World Nuclear Industry and Estimated Radiological Impacts."

8.4 Publications

C. W. Miller, "A Critique of the Determination of Atmospheric Stability Categories for Assessing Airborne Releases of Radionuclides," *Health Phys.* 34, 489–92 (May 1978).

C. W. Miller, "An Examination of Gaussian Plume Dispersion Parameters for Rough Terrain," *Atmos. Environ.* 12, 1359–64 (1978).

C. W. Miller, "An Application of the ATDL Simple Dispersion Model," *J. Air Poll. Contr. Assoc.* 28(8), 798–800 (August 1978).

C. W. Miller, F. O. Hoffman, and D. L. Shaeffer, "The Importance of Variations in the Deposition Velocity Assumed for the Assessment of Airborne Radionuclide Releases," *Health Phys.* 34(6), 730–34 (June 1978).

D. L. Shaeffer, "Modelling Dresden/Monticello Data for ^{131}I Transport from Pasture to Milk," (accepted for publication), *Trans. Am. Nucl. Soc. 1978 Winter Meeting*, Washington, D.C. (November 1978).

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*Presents the paper.

9. NUCLEAR SAFETY INFORMATION CENTER

W. B. Cottrell J. R. Buchanan

In 1963, the NSIC was established by the United States Atomic Energy Commission (USAEC) as a focal point for the collection, analysis, and dissemination of information for the benefit of the nuclear community. It aids those concerned with the analysis, design, and operation of nuclear facilities by providing, and/or identifying, relevant information on their nuclear safety problems. This may be done directly as through special searches, SDI, and consultation, or indirectly as through the publication of reports, bibliographies, and the technical progress review (*Nuclear Safety*). In addition, the Center's staff members participate in various special studies and evaluations. The NSIC is now jointly sponsored by NRC and DOE.

The capabilities of the Center's technical staff are augmented by a computer file that contains 100-word abstracts of over 136,000 nuclear safety documents, a total that is increasing at the rate of 12,000 documents per year. Pertinent abstracts are readily retrievable from the computer to meet a number of requirements. These files are frequently queried for retrospective bibliographies to aid the staff in answering technical inquiries. The files are also used extensively during preparation of NSIC reports and review articles for the bimonthly journal, *Nuclear Safety*. A routine form of reference output from the storage files is a biweekly SDI Program that selects references according to individual user requirements. NSIC's computerized capability has been extended by the inclusion of the NSIC file on the DOE-RECON system where it may be accessed by the RECON network.

This report summarizes the principal activities of the Center from July 1—September 30, 1978. Since mid-1975, the NSIC has operated under the DOE Information Center Pricing Policy, which was implemented on July 1, 1975, for certain NSIC services and subsequently extended to apply to SDI (i.e., biweekly mailings). This experience is reported, as well as the more customary NSIC activities including the preparation of reports and the journal, *Nuclear Safety*.

9.1 Routine Services

During the months of July, August, and September 1978, the staff of NSIC (1) entered 2748 documents, (2) responded to 230 inquiries (of which 137 involved technical staff), (3) made 57 computer searches (of which 5 were for paying customers), (4) provided SDI to 377 subscribers (including 57 paying subscribers), (5) received 31 visitors, (6) attended six meetings, and (7) issued two reports.

The DOE report on the usage of the 24 data bases on the RECON system for the period August 1, 1978, to September 1, 1978, is presented in Table 9.1. Note that NSIC is the fifth most widely used data base behind only Nuclear Science Abstracts (NSA), Water Resources Abstracts (EPA-WRSIC), Environmental Mutagens Information Center (EMIC), Energy Data Base (EDB), and Engineering Index (EIX).

In accordance with the DOE Information Center Pricing Policy, which has been applicable to NSIC, those organizations or persons other than Center sponsors and their direct subcontractors must pay for the services they receive. As noted above, NSIC performed five paid searches during the quarter and, at the end of the quarter, had 57 paid SDI subscriptions. However, these funds are not received by the Center but are collected by the National Technical Information Center — serving as the billing agent — and then credited to the DOE General Fund.

9.2 ORNL-NSIC Reports

Work was initiated, continued, or concluded on several reports as follows:

ORNL/NUREG/NSIC-128, *HTGR Safety: Review of Current Issues and Bibliography of Literature 1960—1977* (issued).

ORNL/NUREG/NSIC-144, *Reactor Operating Experiences 1975—1977* (in reproduction).

ORNL/NUREG/NSIC-148, *Bibliography on Common Cause — Common Mode Failures* (in preparation).

ORNL/NUREG/NSIC-149, *Annotated Bibliography of Licensee Event Reports in Boiling-Water-Reactor Nuclear Power Plants as Reported in 1977* (in preparation).

Table 9.1. RECON data base activity from 08-01-78 to 09-01-78
(23 operating days)

Data base identification	Data base name and supporting installation identification	No. of sessions	No. of expands	Citations printed
NSA	(TIC) Nuclear Science Abstracts	828	1,665	36,104
RIP	(ERDA) Energy Research in Progress	194	366	4,945
WRA	(WRSIC) Water Resources Abstracts	350	1,798	26,577
GAP	(ERDA) General and Practical Information	149	122	3,458
NSR	(NDP) Nuclear Structure Reference	43	47	283
ARF	(EMIC/ETIC) Agent Registry File	35	61	76
EMI	(EMIC) Environmental Mutagens Information	244	446	13,857
EDB	(TIC/CSD) ERDA Energy Data Base	2,809	5,295	160,233
ERD	(EISO) Energy R&D Projects	69	105	33
NBI	(NBIC) National Biomonitoring Inventory	29	17	1,604
DBS	(LLL) Data Base Survey	40	46	344
ESI	(EIC) Environmental Science Index	118	257	2,439
EIX	(CSD) Engineering Index	390	949	10,505
MOS	(LLL) Engineering and Environmental Data Base Modeling Survey	25	49	58
NSC	(NSIC) Nuclear Safety Information Center	202	296	7,765
WRE	(WRSIC) Water Resource Research	56	109	272
NRC	(LC) National Referral Center	55	52	111
NER	(EIC) National Energy Referral	49	73	280
RSI	(RSIC) Radiation Shielding Information	20	10	235
EIA	(EIC) Energy Information Abstracts	111	164	706
RSC	(RSIC) Radiation Shielding Codes	13	7	9
ESR	(ERDA) Federal Energy Environmental and Safety Research	37	32	4
ETI	(ETIC) Environmental Teratology	66	198	695
TUL	(U. TULSA) Tulsa Data Base	45	151	7

ORNL/NUREG/NSIC-150, *Annotated Bibliography of Licensee Event Reports in Pressurized-Water-Reactor Nuclear Power Plants as Reported in 1977* (in preparation).

ORNL/NUREG/NSIC-151, *Breeder Reactor Safety — Review of Current Issues and Bibliography of 1977 Literature* (in reproduction).

ORNL/NUREG/NSIC-154, *Index of Microfiched Foreign Reports Distributed Under the NRC Light-Water-Reactor Safety Research Foreign Technical Exchange Program* (in preparation).

ORNL/NUREG/NSIC-155, *Bibliography of Reports on Research Sponsored by the NRC Office of Nuclear Regulatory Research, January–June 1978* (in reproduction).

ORNL/NUREG/NSIC-156, *Reports Distributed Under the NRC Reactor Safety Research Foreign Technical Exchange Program, Vol. V, January–June 1978* (in composition).

ORNL/NUREG/NSIC-157, *Stress Analysis and Other Analytical Techniques* (in preparation).

ORNL/FTR-436, *The FRG-BMFT Research Program on the Safety of Light Water Reactors 1977–1980* (issued).

ORNL/FTR-457, *IAEA Specialists' Meeting on Power Supply Arrangements for Nuclear Power Plants* (issued).

9.3 Journal of *Nuclear Safety*

The preparation and review of material for the technical progress review, *Nuclear Safety*, continued apace with the requirements of its bimonthly publication schedule. *Nuclear Safety* 19(4) was distributed the latter part of July and *Nuclear Safety* 19(5) the last week of September. All technical articles for *Nuclear Safety* 20(1) were delivered to NRC, DOE, and TIC on September 22, 1978. Manuscripts of all the technical articles for *Nuclear Safety* 20(2) have been received and are being subjected to peer review. The regular bimonthly *Nuclear Safety* staff meeting was held August 2, 1978. Minutes of that meeting and a tentative outline for the next several issues of *Nuclear Safety* were reviewed and have since been disseminated.

10. CENTRAL COMPUTERIZED DATA BASE FOR BREEDER REACTOR SAFETY CODES

G. F. Flanagan
J. W. Arwood N. M. Greene
V. M. Forsberg G. B. Raiford

A review copy of Version I information was sent out on June 15, 1978, to various SACRD subcommittee chairmen. The information included a set of plots of tabular data in Version I.

A dump of NSMH data in SACRD, including plots, was sent to H. Alter (DOE) for review.

Work continued on inputting NSMH data received. In addition, stress rupture correlations for 20% CS SS316 have been received and are being worked on.

Several modifications were made to individual entries in the data base to correct, modify, or make extensions as suggested by reviewers of the material distributed in May 1978.

Updates and extensions were made to the abbreviations file used to catalog the eight character names which classify the SACRD data base. In particular, an attempt was made to ensure that all abbreviations that have been used in the data base are cataloged and identified.

A special NSMH plotting code was received from M. Marchbanks at HEDL. An effort was made to convert this code to IBM FORTRAN.

The SACRD plotting programs have been extensively revised, and preliminary versions are able to display graphical representations of data on a Techtronix remote display station.

The complete set of minutes from the Fifth Biannual SADCG Meeting were distributed.

The tentative Version I radiological data were put on microfiche for distribution to S. V. Kaye's committee for review purposes.

Corrections and modifications were made to the previous data base as directed by various reviewers. This included extensive revisions to the helium data, as suggested by GA.

In order to make the SACRD Version I Data Catalog more readable, the "abbreviations" file was updated to include all terms used to catalog the data.

11. BREEDER REACTOR RELIABILITY DATA ANALYSIS CENTER

G. F. Flanagan
T. F. Bott S. D. Hudson
N. M. Greene J. J. Manning
P. M. Haas M. A. Woodside

The purpose of this effort is to define, develop, and maintain a reliability data analysis center for use in FBR safety and licensing and to develop the methodology necessary to process the data for validated and accurate reliability analysis.

11.1 Scoping Study for Version I

To date, less than half the reviewers requested by DOE to comment on the CREDO proposal have responded. The remaining reviewers are assumed to have no significant comments and/or are generally in agreement with the proposed plan, and the review is therefore completed. The critical reviews received have been studied carefully, and responses to each significant comment are being prepared.

All respondents were, in general, in agreement with the basic concept of a centralized reliability data base for advanced reactors, though at least one did suggest a closer examination of the urgency of the project compared to current advanced reactor development schedules. Other major points that were raised by more than one reviewer included:

1. The need to produce "something of direct value" to contributing sites. Generally, two positive suggestions surfaced repeatedly: (a) special data and data-related services for specific needs of the contributing site — not merely generic failure rates for components or systems; (b) payoff in terms of improved regulatory requirements, QA requirements, etc. (at least in terms of having specific "reliability criteria" defined as "acceptable" to regulatory agencies).

2. Data Base Management System (DBMS) selection — several reviewers desired further justification for selection of the ORCHIS-based (JOSHUA/ORCHIS combination) DBMS. Most of those seemed to want more discussion of the relative merits of various commercially available and other existing DBMSs. A survey of existing systems was made early in the

CREDO program, and additional review of other possible systems has been carried out since then. Our judgment is still that the proposed system is the best option available.

3. The need to interface with and learn from experiences of other existing data bases. This need has been recognized and acted upon throughout CREDO planning, but more specific interfaces (such as compatible naming schemes and procedures for exchanging data) now need to be worked out.

4. The need for and the difficulty of accomplishing data validation. The difficulty of this problem has been recognized since early planning stages, and the need for guidance from the proposed committee of experts suggested in the CREDO plan has been noted previously. This committee should be organized by DOE as soon as possible.

Many of the other specific comments have been quite helpful both in reinforcing points learned during the scoping study and in indicating areas that need to be addressed more explicitly in the specifications for CREDO. The efforts of DOE staff and contributing reviewers in completing this critical review are sincerely appreciated.

11.2 Data Collection and Evaluation

Contacts were made with Byron Jackson, Babcock & Wilcox, and Allis-Chalmers companies concerning sodium pumps. Arrangements for acquiring data on sodium components collected by the LMEC data base were initiated. Work on sodium valve data collection at WARD was initiated by on-site personnel.

Evaluation of data collected on sodium valves continued with point failure rate estimates and hazard plots for several sets of data.

Collection of published data on sodium pumps continued throughout the month.

Evaluation and classification of data on EBR-II instrumentation continued this month. Identification of components and systems to be monitored on a first-priority basis at FFTF was completed and sent to DOE.

Initial arrangements for data collection through subcontracting at HEDL were made.

Collection and evaluation of historical data on sodium components, especially pumps, valves, and rupture disks, have continued. The subcontracted effort at WARD has identified 93 sodium valves of sizes 1 to 6 in. that have been operating for as long as 11 years. Engineering, event, and operating data are being accumulated, and qualitative descriptions of failures will be prepared. A subcontract for similar historic data collection on sodium loop components has been established at ETEC (formerly LMEC), and a temporary CREDO employee has been assigned to HEDL to collect data from HEDL test loops. Historical data collected from these loops and all available published sources, combined with historical data from EBR-II, will form the basis for building the comprehensive data base desired. (Unfortunately, plans for immediate participation of an EBR-II staff member on a part-time basis cannot be carried out this fiscal year due to lack of available personnel at EBR-II. Efforts are being made to establish such an on-site representative during FY 1979.)

A special request for available data on gas valves from C. Griffen at AI was made. The primary interest was in fairly large gas valves (2 to 4 in.) typical of those in an LMFBR primary system cover gas. Specific information was desired on relative reliability of manual and operator-controlled valves. A search was made of all known sources available on short-term notice, including EBR-II, ETEC test loops, and Mine Safety Appliance Research (MSAR) loops. Data were compiled on $\sim$500 valves ranging from 1/4 to 6 in. Although complete and detailed information was not readily available, the data and statistical analysis provided did indicate a significantly higher failure rate for operator-controlled than for manually operated gas valves. Estimated failure rates for both were on the order of 10^{-6}/hr.

Collection and evaluation of historical data on sodium components at test loops have continued. A thorough, detailed search of records and reports from the Transient Test Loop (TTL) at HEDL has produced rather complete documentation on engineering, operating, and event data from that loop for most of its operating history. Additional collection

at two other HEDL test facilities, the Composite Reactor Component Test Activity (CRCTA) and the Prototype Applications Loop (PAL) has been started. This effort is being accomplished by use of a temporary CREDO employee with assistance from highly experienced HEDL personnel. The success of the data collection work is due to the particular talents of the CREDO and HEDL staff members, the cooperation of HEDL management, and the fact that reasonably good loop documentation is available. However, this success does demonstrate the feasibility of gathering complete historical data from existing sites.

Subcontract work for data collection at AI (ETEC) and at WARD is continuing, and collection of information on heat exchangers and steam generators (primarily from records from Hallam, Fermi, and EBR-II) has been initiated.

11.3 Data Management System

Requirements of the DBMS to be fulfilled by ORCHIS, JOSHUA, and other software are being defined more specifically, as are the procedures for interfacing the different types of software. Classification schemes for both instrumentation and controls (I&C) and mechanical components and systems are being worked out in detail, and corresponding detailed specifications for the DBMS are being developed. The CREDO system is primarily "component based," that is, event, engineering, and operating data focus on items of hardware at some level designated by CREDO as a component, which is the lowest level for which complete data will be accumulated, because different users and different applications require information on hardware items at higher levels (i.e., subsystems and systems). In addition, at least key engineering and event data on "parts" (i.e., hardware items of a level below what has been designated by CREDO as a component) should be available to the user. Methods for assuring this upward and downward "mobility" from the component level are being devised.

Computer programs have been written that create a collection of CREDO records containing engineering, operations, and population data. Programs such as these will allow qualified DBMS personnel to enter data

into the data base from local computer terminals or through card readers. Simple programs to calculate failure rates using this experimental collection of records are now being written. These initial programs will be used primarily to gain experience with management of CREDO data, but they do form a basis for the full software system which is to be developed.

11.4 Development of Methodology

The computer code Goodness of Fit (GOF), which includes a number of routines for classical statistical analysis, has been obtained and is being reviewed for potential applicability to CREDO data analysis.

Some initial experience in developing trends analyses has been gained by analyzing sodium valve data in preparation for presentation of a paper at the Fifth Symposium on Reliability Technology, Bradford, United Kingdom, in September.

11.5 Interface and Coordination

Comments on the proposed list of priority items for data collection at FFTF, which was sent to DOE in June, have not yet been received. It is hoped that these items can be agreed upon by LOA groups, FFTF staff, and CREDO in the near future so that further interfacing between FFTF and CREDO can be worked out. We feel it is extremely important to begin data collection efforts at FFTF as early as possible. Similar interfacing is required at EBR-II, and preliminary discussions with EBR-II staff are being carried out.

A meeting was held with EBR-II staff at EBR-II to discuss further data collection. This was the first time that we have been able to meet with a number of the key management personnel at one time, including S. Barak, I&C Supervisor; D. Cissell, Acting Project Manager; R. McConnell, Associate Director for Engineering; and R. Smith, Operations Manager. A trip report of the meeting will be forwarded to DOE.

In summary, EBR-II staff members are willing to cooperate, assuming CREDO would be developed as outlined in the CREDO proposal. (D. Cissell was one of the DOE reviewers of the proposal.) They agreed that an

EBR-II staff member familiar with the plant would be the most effective personnel choice and suggested that a "trial" period of about one year should be adequate to establish the feasibility and adequacy of data collection procedures. Funding for this staff member would have to be established through DOE by mutual agreement among DOE, CREDO, and EBR-II.

EBR-II staff requested a list outlining the systems and components on which data are needed and the priorities in obtaining the data, much like the list prepared for FFTF. This list has been drafted and will be sent to DOE and EBR-II in the near future, along with more specific description of the proposed effort at EBR-II.

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