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ORNL Breeder Reactor Safety Quarterly Technical Progress Report for July–September 1980

M. H. Fontana J. L. Wantland

No longer AT.
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4/17/2008

[redacted]

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ORNL BREEDER REACTOR SAFETY QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY–SEPTEMBER 1980

M. H. Fontana J. L. Wantland

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PREVIOUS BREEDER REACTOR SAFETY PROGRESS REPORTS*

Prior information on these projects was presented in the following Breeder Reactor Safety and Core Systems Programs Progress Reports:†

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
July—September 1978	ORNL/TM-6698
October—December 1978	ORNL/TM-6851
January—March 1979	ORNL/TM-6947
April—June 1979	ORNL/TM-7147
July—September 1979	ORNL/TM-7229
October—December 1979	ORNL/TM-7301
January—March 1980	ORNL/TM-7348
April—June 1980	ORNL/TM-7489

*Reported previously as LMFBR Safety and Core Systems Programs 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. Task 01 — LOA 2 Tasks: THORS Program

Based on a preliminary analysis of Part A experimental results, Part B (the final section) of the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Bundle 9 Phase 2 test program was written as an addendum to the Part A test plan. The addendum is documented in this report.

All Bundle 9 Phase 2 transient boiling tests were completed during this report period. A total of 45 transient runs were conducted. Boiling was observed in 36 runs for a total boiling time of ~140 min at temperatures near 1000°C. Thirteen of these runs reached dryout conditions. Following completion of these tests, Bundle 9 had been operated in sodium for ~5600 h, 234 h of which were with power to the fuel pin simulators (FPSs).

The Bundle 9 Phase 1 (steady-state) record of experimental data report was published during this report period, and the Bundle 9 Phase 2 (transient boiling) record of experimental data report has undergone internal review and is in the final stages of preparation.

The computer code NATOF-2D has been transmitted by Massachusetts Institute of Technology (MIT) and made operational at Oak Ridge National Laboratory (ORNL). The code contains a two-dimensional (2-D) two-fluid model for sodium boiling in a Liquid-Metal Fast Breeder Reactor (LMFBR) fuel pin bundle geometry. Pending resolution of some problems with the code, it will aid in analysis of THORS Bundle 9 boiling test data.

Analysis of Bundle 9 sodium boiling data indicates that the static (Ledinegg) flow instability always occurs before film dryout. This instability, in turn, does not occur unless the boiling region occupies the entire bundle cross section. The inclusion of a simple non-dynamic boiling model in the transient single-phase SABRE-2 code has given an accurate three-dimensional (3-D) representation of transient boiling behavior to the point of dryout for both Bundle 6A and Bundle 9.

2. Task 02 — ORNL LOA 4 Tasks: Environmental Assessment of Alternate FBR Fuels

Preparation of a report describing oxidation behavior of the C-based Fast Breeder Reactor (FBR) fuels continued. Performance of an apparatus to measure the ^{3}H content of High-Temperature Gas-Cooled Reactor (HTGR) fuel particles was determined using a ^{3}H-labeled organic compound and fuel particles irradiated in the High Flux Isotope Reactor (HFIR). Particles irradiated in the Experimental Breeder Reactor—II (EBR-II) were provided by General Atomic Company (GA) and have been characterized by microradiography and gamma spectrometry preparatory to measurement of their ^{3}H contents.

Work continued on evaluation of dose conversion factors associated with releases of high-specific—activity radionuclides and ^{3}H. Several papers on environmental impact topics were presented verbally or for publication. A major contribution was made to organizing the Third Annual Health Physics Society (HPS) Summer School held in Seattle, Wash., July 14 through 18, 1980.

A recently purchased mass spectrometer failed to meet operational specifications. Corrective steps are being taken by the supplier's representative. A funding and time supplement was made to the contract with Colorado State University (CSU) to study source terms from a thorium ore pile.

3. Task 03 — ORNL LOA 4 Tasks: Model Evaluation of Breeder Reactor Radioactivity Releases

A report has been completed that makes recommendations concerning needs for model evaluations, environmental research, and biomedical research to support breeder reactor environmental radiological assessments. An attempt has been made to estimate the imprecision in the thyroid dose to children from elemental ^{131}I and the imprecision in the total-body dose commitment from the ingestion of a unit activity of soluble ^{137}Cs. A comparison has been made between total-body external gamma doses calculated by the AIRDOS-EPA and COMRADEX-IV computer codes. Preparations have been

completed for the *Symposium on Intermediate Range Atmospheric Transport Processes and Technology Assessment*.

4. Task 04 — ORNL LOA Support and Integration: Nuclear Safety Information Center

The Nuclear Safety Information Center (NSIC) serves the nuclear community through the collection, analysis, and dissemination of relevant safety information. During the report period July to September 1980, we processed 2375 documents, responded to 202 inquiries, serviced 396 Selective Dissemination of Information (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*. Two reports were sent to reproduction during the quarter, and work is continuing on several others. Work was also undertaken on the material for several issues of *Nuclear Safety* as required by its publication schedule.

5. Task 05 — ORNL LOA Support and Integration: Breeder Reactor Reliability Data Analysis Center

Testing of the Centralized Reliability Data Organization (CREDO) Data Base Management System (DBMS) with respect to searching capabilities and output generation has been completed on real data from EBR-II. Capabilities for generating failure rates on any set of qualified CREDO components now exist, and the capability for generating engineering and event inventories, engineering summaries, component failure summaries, and quantitative failure data summaries exists. Additionally, programs now exist for automatic data entry via interactive display terminals. Engineering data collection was initiated on three new systems at EBR-II. Engineering data collection was initiated on all valves in the Fast-Flux Test Facility (FFTF) primary and secondary heat transport systems as well as its dump heat system.

6. Task 06 — ORNL LOA Support and Integration: Central Data Base for Breeder Reactor Safety Codes

Work continued on entering *Nuclear Systems Materials Handbook* (NSMH) update package No. 19 and the saturated sodium properties file into the data base. A Safety Analysis Computerized Reactor Data (SACRD) newsletter was prepared and mailed during the quarter. Physical locations of many SACRD data files were changed as a consequence of a sequence of computer system errors that seriously threatened the integrity of the data base. Work was initiated toward converting JOSHUA to be operational on the IBM MVS Operating System that will replace the older MVT System sometime during the next quarter.

GLOSSARY OF ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
AI	Atomics International, Division of Rockwell Corporation
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ARAC	Atmospheric Release Advisory Capability
BGA	Bundesgesundheitsamt, German Federal Health Office
BN	boron nitride
BNBF	boron nitride backfilled
BNL	Brookhaven National Laboratory
BONA	boiling natrium
CCDAS	Computer-Controlled Data Acquisition System
CDS	Conceptual Design Study
CEC	Commission of the European Communities
CRCTA	Composite Reactor Components Test Activity (HEDL)
CREDO	Centralized Reliability Data Organization
CRBR	Clinch River Breeder Reactor
CSU	Colorado State University
DAS	Data Acquisition System
DBMS	Data Base Management System
DISSPLA	Display Integrated Software System and Plotting Language
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor-II
EG&G	Edgerton, Germeshausen, and Grier
EM	electromagnetic
ENS	European Nuclear Society
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
ERDS	European Reliability Data System
ESD	Energy Systems Division
ETEC	Energy Technical Engineering Center (formerly LMEC)
FBR	Fast Breeder Reactor
FFTF	Fast-Flux Test Facility
FPS	fuel pin simulator

FRSTMC	Fast Reactor Safety Technology Management Center
FTS	Federal Telecommunication System
GA	General Atomic Company
GE	General Electric Company
GMI	Groth-Mazur Industries
GOF	goodness of fit
HEDL	Hanford Engineering Development Laboratory
HFIR	High Flux Isotope Reactor
HPS	Health Physics Society
HTGR	High-Temperature Gas-Cooled Reactor
IAEA	International Atomic Energy Agency
ICE	implicit continuous Eulerian
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronics Engineers
INEL	Idaho National Engineering Laboratory
IO	input-output
LASL	Los Alamos Scientific Laboratory
LER	Licensee Event Report
LLL	Lawrence Livermore Laboratory
LMEC	Liquid-Metal Engineering Center (now ETEC)
LMFBR	Liquid-Metal Fast Breeder Reactor
LOA	lines of assurance
LOF	loss of flow
LOPI	loss of pipe integrity
LWR	light-water reactor
MIT	Massachusetts Institute of Technology
NCSR	National Centre of Systems Reliability
NOAA	National Oceanic and Atmospheric Administration
NPEC/SC-5	Nuclear Power Engineering Committee/Subcommittee 5
NPRDS	Nuclear Power Reliability Data Service
NRC	Nuclear Regulatory Commission
NSIC	Nuclear Safety Information Center
NSMH	*Nuclear Systems Materials Handbook*
OPERA	Out-of-Pile Expulsion and Reentry Apparatus
ORCHIS	Oak Ridge Computerized Hierarchal Information System

ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
1-D	one-dimensional
PAL	Prototype Applications Loop (HEDL)
PG	Pasquill-Gifford
PIC	particle in cell
PLSA	Properties for LMFBR Safety Analysis
PMD	pressure-measuring device
PNC	Power Reactor and Nuclear Fuel Development Corporation (Japan)
PO	Project Office
QA	quality assurance
RAS	Reactor Analysis and Safety (Division of ANL)
RECON	Remote Console Bibliographic Retrieval System (DOE)
RDMS	Reactor Data Management System (GE)
RRT	Reactor Research and Technology (Division of DOE)
RSR	Reactor Safety Research (Division of NRC)
RTD	resistance temperature device
SACRD	Safety Analysis Computerized Reactor Data
SADCG	Safety Analysis Data Coordinating Group
SBT	Sodium Boiling Test
SCTI	Sodium Component Test Installation
SDI	Selective Dissemination of Information
SFGP	simulated fission-gas plenum
SHRS	Shutdown Heat Removal System
SOR	Successive overrelaxation
SRE	Sodium Reactor Experiment
SRL	Savannah River Laboratory
SS	stainless steel
THORS	Thermal-Hydraulic Out-of-Reactor Safety
3-D	three-dimensional
TIC	Technical Information Center
TLD	Thermoluminescence dosimeter
TMC	Technology Management Center
TMI	Three Mile Island

TOP	transient overpower
TREAT	Transient Reactor Test Facility
TSO	time-sharing
TTL	Transient Test Loop (HEDL)
TUC	transient undercooling
TUCOP	transient undercooling and overpower
2-D	two-dimensional
UK	United Kingdom
UKAEA	United Kingdom Atomic Energy Authority
UOR	Unusual Occurrence Report
USAEC	United States Atomic Energy Commission
UT	The University of Tennessee
WARD	Westinghouse Advanced Reactor Division
WPAS	Work Package Proposal and Authorization System
ZPR	Zero Power Reactor

Computer Codes

ADJOINT	None
ADPIC	Atmospheric diffusion particle-in-cell
ADSEP	None
AIRDOS II	Air dose
AIRDOS-EPA	Developed for EPA
ANISN	Anisotropic SN
COBRA	Coolant Boiling in Rod Arrays
COMRADEX-IV	Containment and Meteorology on the Environmental Radiation Exposure
CRESTCO	None (a statistical analysis package)
DWNWND	None
JOSHUA	None (a data-base management program)
KENO	None (Monte Carlo Code)
LEXIBOSS	None
MATHEW	None
MELT	None (neutronic thermal-hydraulic code)
ORRIBLE/B6	None

SABRE	Subchannel Analysis of Blockage in Reactor Elements
SAS	Safety Analysis System
SCG	Sensitivity Coefficient Generator
SPQR	Stochastic Program for Quasi-static Research
THERMIT	None (thermal-hydraulic code developed at MIT)

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ORNL BREEDER REACTOR SAFETY — LOA 2 QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY—SEPTEMBER 1980

M. H. Fontana J. L. Wantland

AF 15 10 25 0

Task 01 — ORNL LOA 2 Tasks: THORS Program
[DRS 02.01]

OBJECTIVES

To determine the thermal-hydraulic response of simulated sodium-cooled fast reactor subassemblies at normal and off-normal operating conditions, to determine the characteristics of sodium boiling in a reactor core at low-flow and/or natural convection conditions, and to develop models to give a better understanding of sodium boiling and loop response to boiling conditions.

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ORNL BREEDER REACTOR SAFETY — LOA 2 QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY—SEPTEMBER 1980

M. H. Fontana J. L. Wantland

ABSTRACT

Based on a preliminary analysis of Part A experimental results, Part B (the final section) of the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Bundle 9 Phase 2 test program was written as an addendum to the Part A test plan. This addendum is documented in this report.

All Bundle 9 Phase 2 transient boiling tests were completed during this report period. A total of 45 transient runs were conducted. Boiling was observed in 36 runs for a total boiling time of ~140 min at temperatures near 1000°C. Thirteen of these runs reached dryout conditions. Following completion of these tests, Bundle 9 had been operated in sodium for approximately 5600 h, 234 h of which was with power to the fuel pin simulators.

The Bundle 9 Phase 1 (steady-state) record of experimental data report was published during this report period, and the Bundle 9 Phase 2 (transient boiling) record of experimental data report has undergone internal review and is in the final stages of preparation.

The computer code NATOF-2D has been transmitted by Massachusetts Institute of Technology and made operational at Oak Ridge National Laboratory. The code contains a two-dimensional two-fluid model for sodium boiling in a Liquid-Metal Fast Breeder Reactor fuel pin bundle geometry. Pending resolution of some problems with the code, it will aid in analysis of THORS Bundle 9 boiling test data.

Inclusion of a simple non-dynamic boiling model in the transient single-phase SABRE-2 code has given an accurate three-dimensional representation of transient boiling behavior to the point of dryout for both Bundle 6A and Bundle 9.

1. TASK 01 — ORNL LOA 2 TASKS: THORS PROGRAM

J. L. Wantland
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R. H. Morris
S. D. Rose

1.1 THORS Bundle 9 Phase 2 Part B Transient Boiling Test Plan

R. H. Morris J. F. Dearing
B. H. Montgomery

This section documents the final section (Part B) of the THORS Bundle 9 Phase 2 Test Plan. This portion was written after preliminary analysis of the results of Part A and is issued as an addendum to that test plan. Part A of the test plan is given in Ref. 1.

THORS BUNDLE 9, PHASE 2 TRANSIENT TEST PLAN, PART B

(Addendum to Part A, dated April 1, 1980)

Prepared by R. H. Morris

July 30, 1980

Approved: _____ 7-30-80
M. H. Fontana, Manager — Date
Breeder Reactor Safety Program

Approved: _____ 7-30-80
J. L. Wantland, Principal — Date
Investigator, THORS Program

Approved: _____ 8-11-80
B. H. Montgomery — Date
Project Engineer, THORS Facility

Approved: _____ 7/31/80
R. E. MacPherson, Head — Date
Experimental Engineering Section

Approved: _____ for C.A. MILLS 7-31-80
C. A. Mills — Date
ETD Quality Assurance Coordinator

Summary of Phase 2, Part B Testing

Tests 1 and 2 are identical to their counterparts in the Bundle 9 Phase 2, Part A test program.[1] Test 1 is run at steady-state conditions of 15 kW per pin, 7.4 ℓs^{-1} inlet flow and 370°C inlet temperature, as a comparison for documentation of changes in pin distortion during testing. Test 2 is a power-off transient, with initial conditions of 1.0 kW per pin, 0.49 ℓs^{-1} inlet flow and 370°C inlet temperature, in order to detect any further deterioration of the insulation during this portion of the test program.

Portions of the Phase 2, Part A test program will be repeated with minor modifications in the Phase 2, Part B test series. These tests have been proposed to determine the influence of an increase in the bypass flow ratio upon the amplitude and period of the flow oscillations and the resulting measured time to dryout. In addition, the influence of the expansion tank pressure upon the amplitude of the flow oscillations and time to dryout will be examined, and a determination will be made of the influence of changes in the inlet valve setting upon the stability of the density wave oscillations and resulting time to dryout.

Test 300 will provide data on the flow-pressure drop characteristics of the test section inlet valve, at three different valve settings for the range of test section inlet flows of the Part B program.

Test 301 will determine the maximum possible bypass to test section flowsplit as a function of test section flow, for the three test section inlet valve settings to be used during Part B testing.

Test 302 is designed to give information on setting the EM pump controller so that the specified low flow, for the controlled flow reduction tests can be obtained in the presence of buoyancy-induced flows.

The conditions for the transient tests (Tests 303–308) are summarized in Table 1.1. The initial conditions of Tests 303 and 304 are such that the buoyancy-induced differential static head over the elevation difference between the beginning of the heated section and the inlet to the expansion tank is responsible for a large fraction of the test section flow. The two-phase transient is induced by cutting power to the EM pump. Test section flow will decrease rapidly to some minimum value (determined by

Table 1.1. Summary of Bundle 9 Phase 2 Part B transient tests

Test	Run	Power		Test section inlet flow		Flow ramp down rate [ℓ/s (gpm/s²)]	Bypass to test section flow ratio[a]	Test section inlet valve pressure drop[b]	Expansion tank cover gas pressure [kPa (psi)]	Test section inlet temperature [°C (°F)]	Analogous Part A Run	
		Total (kW)	kW/pin	Initial [ℓ/s (gpm)]	Reduced [ℓ/s (gpm)]						Test	Run
303	(Natural convection test)											
	101	250	4.1	0.74 (11.7)		Cut power to pump	4:1	Normal	48 (7)	390 (730)	203	102
	102	250	4.1	0.74 (11.7)		Cut power to pump	4:1	Normal	69 (10)	390 (730)	203	101
	103	250	4.1	0.74 (11.7)		Cut power to pump	4:1	Maximum	48 (7)	390 (730)		
	104	250	4.1	0.74 (11.7)		Cut power to pump	Maximum	Normal	48 (7)	390 (730)		
	105	250	4.1	0.74 (11.7)		Cut power to pump	Maximum	Normal	69 (10)	390 (730)		
304	(Natural convection test)											
	101	287	4.7	0.84 (13.3)		Cut power to pump	Maximum	Normal	48 (7)	390 (730)	203	105
	102	287	4.7	0.84 (13.3)		Cut power to pump	Maximum	Normal	69 (10)	390 (730)		
305	(Controlled flow reduction test)											
	101	409	6.7	1.2 (19)	0.45 (7.1)	−0.40 (−6.4)	Maximum	Normal	48 (7)	390 (730)	204	107
	102	409	6.7	1.2 (19)	0.45 (7.1)	−0.40 (−6.4)	Maximum	Maximum	48 (7)	390 (730)		
	103	409	6.7	1.2 (19)	0.45 (7.1)	−0.40 (−6.4)	Maximum	Minimum	48 (7)	390 (730)		
	104	409	6.7	1.2 (19)	0.45 (7.1)	−0.013 (−0.2)	Maximum	Normal	48 (7)	390 (730)		
306	(Natural convection test with a 3:1 diametral power skew)											
	101	102		9.84 (156)		Cut power to pump		Normal	48 (7)	390 (730)		
307	(Natural convection test)											
	101	201	3.3	0.73 (11.5)		Cut power to pump	4:1	Normal	48 (7)	443 (830)		
	102	158	2.6	0.73 (11.5)		Cut power to pump	4:1	Normal	48 (7)	499 (930)		
308	(Controlled flow reduction test)											
	101	915	15	2.56 (40.6)	1.53 (24.3)	−0.40 (−6.4)	Maximum	Normal	48 (7)	390 (739)	205	1021

[a]Maximum bypass-to-test section flow ratio will be determined in Test 301.

[b]Test section inlet valve settings for normal, maximum, and minimum pressure drop will be determined in Test 300.

the initial buoyancy-induced head) and then slowly increase as temperatures rise in the simulated fission gas plenum. Tests 303 and 304 will be run at two different expansion tank cover gas pressures, both 4:1 and maximum bypass to test section flow ratio and one run with an increased test section inlet valve pressure drop. The results of these tests will then be compared to determine the influence of these parameters upon the amplitude of the flow oscillations and the time to dryout.

The initial conditions of Test 305 are such that the buoyancy-induced flow is a smaller fraction of the total test section flow. The two-phase transient is induced by a programmed reduction in the head provided by the EM pump. The pump controller is set so that the initial test section inlet flow is decreased over a specified time to a specified low flow. The head produced by the EM pump then remains constant until the transient is ended by a programmed flow increase or manual or automatic termination.

The initial conditions of Test 306 are such that there is almost no buoyancy-induced flow at the start of the transient. This test has a 3:1 diametral power skew across the bundle and a very high flow rate when power to the pump is cut. The very low thermal driving head and 3:1 power skew should allow the examination of intra-subassembly natural convection effects.

Test 307 is approximately the same as Test 303, but with a higher test section inlet temperature and a corresponding lower power. This test will investigate the effect of reduced inlet subcooling on boiling stability under natural convection conditions.

Test 308, the last proposed transient test for the Bundle 9 Phase 2 test series, is a high power controlled flow reduction test. This test is a repeat of Test 205 Run 1021 of the Phase 2, Part A test plan.

At any point in this test plan, the testing may be terminated due to loss of bundle instrumentation, damage to bundle or pins, or due to any other unforeseen event by the Project Engineer and the Program Coordinator. Also, the sequence of tests may be altered at the discretion of the above.

Test 1

Purpose: To determine the reproducibility of bundle temperature profiles throughout the test program. The steady-state temperature distribution in the bundle is documented at various times during the test program by recording the bundle temperature for 60 s while the bundle is being operated at the following conditions.

Procedure:
1. Open test section inlet valve.
2. Close bypass line valve.
3. Adjust inlet temperature to 370°C (700°F), test-section inlet flow to 7.38 ℓs^{-1} (117 gpm), and heater power to 15 kW/pin.
4. Establish steady-state conditions.
5. Establish a heat balance.
6. Record 60 s of instrument responses on magnetic tape in Fast Scan mode.
7. Adjust test section inlet temperature to 370°C (700°F), test section pressure drop to 295 kPa (42.7 psi) (uncorrected for head differential), and heater power to 15 kW/pin. Repeat procedures 4-6 above.

Test 2

Purpose: To determine the condition of the thermal insulation which insulates the pin bundle from the test section housing. Results of this test during the Phase 1 test program showed that sodium had leaked into the insulation. Any further deterioration will be evident by comparing results of this test run before and after Phase 2 Part B.

Procedure:
1. Open test section inlet valve.
2. Close bypass line valve.
3. Adjust heater power to 1.0 kW/pin, test section inlet flow to 0.49 ℓs^{-1} (7.8 gpm), and test section inlet temperature to 370°C (700°F).

4. Establish steady-state conditions.
5. Start recording instrument responses on magnetic tape in Fast Scan mode. After 30 s at steady-state conditions, shut off heater power (continue recording data in Fast Scan Mode).
6. Two minutes after shutting off heater power, discontinue data acquisition.

Test 300

Purpose: To determine the flow-pressure drop characteristics of the test section inlet valve.

Procedure:
1. Close test section bypass line valve.
2. Set test section inlet flow to 2.59 ℓ/s (41 gpm) and adjust test section inlet valve pressure drop to 37 kPa (5.3 psi) at 390°C (730°F) isothermal conditions.
3. Establish steady-state isothermal conditions at 390°C (730°F).
4. Record flows, temperatures, inlet valve setting, pressure drop across inlet valve and test section for the following test section inlet flows.

Run	Test Section Inlet Flow ℓ/s	(gpm)
101	2.52	(40)
102	1.89	(30)
103	1.26	(20)
104	0.95	(15)
105	0.63	(10)
106	0.32	(5)
107	0	(0)

5. Readjust inlet valve setting such that the valve pressure drop is twice that obtained in procedure 2 above. Repeat procedure 4 (starting with run number 201).
6. Repeat procedure 5 above with the inlet valve setting such that the valve pressure drop is one-half of that obtained in procedure 2 (starting at run number 301).

Test 301

Purpose: In order to simulate a constant pressure-drop boundary condition between heat section inlet and outlet, transient tests are run with a parallel bypass line. This test will determine the maximum bypass/test section flow split possible with the EM pump.

Procedure:

1. Set test section inlet valve setting to the normal pressure drop (as specified in Test 300, procedure 2).
2. Open test section bypass line valve fully.
3. Establish steady-state, isothermal conditions at 390°C (730°F).
4. Find the maximum bypass to test section flowsplit possible at each of the test section flow conditions listed below.
5. Record flows, temperatures, inlet valve setting, pressure drop across inlet valve and test section.

	Test Section Inlet Flow	
Run	ℓ/s	(gpm)
101	2.65	(42.)
102	1.53	(24.)
103	1.20	(19)
104	0.69	(11)
105	0.45	(7.1)

6. Readjust test section inlet valve setting to the maximum pressure drop (as specified in Test 300, procedure 5) then repeat 3-5 above (starting with run number 201) for test section inlet flows of 1.2 ℓ/s (19 gpm) and 0.45 ℓ/s (7 gpm), only.
7. Readjust test section inlet valve setting to the minimum pressure drop (as specified in Test 300, procedure 6), then repeat 3-5 above (starting with run number 301) for test section inlet flows of 1.2 ℓ/s (19 gpm) and 0.45 ℓ/s (7 gpm), only.

Test 302

Purpose: The EM pump controller will be set to achieve a specified low flow in order to obtain saturation conditions in the test section for the controlled flow reduction test. This test will provide data which will be helpful in determining the low-flow setting of the pump controller which will give a specified low flow, with natural convection effects taken into account.

Procedure:

1. Set the test section inlet valve to the normal pressure drop setting (as specified in Test 300, procedure 2).
2. Set the valve on the bypass line to obtain the bypass/test section flowsplit determined in Test 301 or 10:1, whichever is lower.
3. With a test section inlet temperature of 390°C (730°F), obtain steady-state conditions at each set of power/flow conditions listed below.
4. Record the settings for the EM pump controller, flows, temperatures, inlet valve and bypass valve settings, pressure drop across the inlet valve, test section and bypass valve.

	Test Section Inlet Flow		Bundle Power	
Run	ℓ/s	(gpm)	kW	kW/pin
101	1.53	(24.3)	525	8.6
102	0.45	(7.1)	153	2.5

5. Readjust test section inlet valve to the maximum pressure drop setting (as specified in Test 300, procedure 5), then repeat procedures 2-4 above for the power/flow conditions listed below.

	Test Section Inlet Flow		Bundle Power	
Run	ℓ/s	(gpm)	kW	kW/pin
201	0.45	(7.1)	153	2.5

6. Readjust test section inlet valve to the minimum pressure drop setting (as specified in Test 300, procedure 6), then repeat procedures 2-4 above for the power/flow conditions listed below.

	Test Section Inlet Flow		Bundle Power	
Run	ℓ/s	(gpm)	kW	kW/pin
301	0.45	(7.1)	153	2.5

Test 303

Purpose: The EM pump will be turned off after steady-state conditions are established at a test section inlet flow of 0.74 $\ell \cdot s^{-1}$ and a pin power of 4.1 kW. This test is similar to Test 55A, Run 101, run in Bundle 6A,[2] in which saturation conditions were obtained approximately 30 s after the pump rundown. Test 203 Runs 101 and 102 in the Bundle 9 Phase 2, Part A test program are analogous to the test proposed in the test series. These two tests had similar power/flow conditions but showed a significant difference in the amplitude of the flow oscillations during boiling. This test will repeat these two runs with the same power/flow conditions, expansion tank gas pressure and bypass-to-test section flow ratio in order to check the reproducibility of the data. Also, the effects of higher bypass to test section flow ratio and increased inlet valve resistance will be examined.

Procedure: 1. Set test section inlet valve (normal setting corresponds to that specified in Test 300, procedure 2 and maximum to Test 300, procedure 5), expansion tank gas pressure and the bypass-to-test section flow ratio (use the maximum ratio determined in Test 301) as given in the table below.

2. Obtain steady-state conditions of test section inlet temperature, outlet temperature and test section flow at the following conditions:

   test section inlet temperature ... 390°C (730°F)
   test section inlet flow ..... 0.74 $\ell \cdot s^{-1}$ (11.7 gpm)
   total bundle power ..... 250 kW

3. Record data in fast scan mode for 12 min, beginning 10 s before cutting power to EM pump.
4. Cut power to EM pump.
5. If dryout conditions do not prematurely terminate the test, end test by returning to initial flow conditions 11.5 min after cutting power to EM pump.

Run	Bypass To Test Section Flow Ratio	Expansion Tank Gas Pressure [kPa (psi)]	Test Section Inlet Valve Setting
101	4:1	48 (7)	normal
102	4:1	69 (10)	normal
103	4:1	48 (7)	max.
104	max.	48 (7)	normal
105	max.	69 (10)	normal

Test 304

Purpose: The EM pump will be turned off after steady-state conditions are established at a test section inlet flow of 0.84 ℓ/s and a pin power of 4.7 kW. This test is similar to Test 203 Run 105 of the Bundle 9 Phase 2, Part A test plan, in which violent boiling occurred ~8 s after pump rundown. The run was terminated at approximately 66 s by an erroneous signal from a failed thermocouple, just as the test section inlet flow decreased rapidly. This test will be repeated to determine the effects of a higher bypass-to-test section flow ratio. Also, if the results from Test 303 indicate a significant difference due to expansion tank cover gas pressure, a run will be made using the higher gas pressure.

Procedure:

1. Set test section inlet valve to the normal pressure drop setting (as specified in Test 300, procedure 2).
2. Obtain steady-state conditions of test section inlet temperature, outlet temperature and test section flow for the conditions given in table below.
3. Record data in the fast scan mode for 12 min, beginning 10 s before cutting power to EM pump.
4. Cut power to EM pump.
5. If dryout conditions do not prematurely terminate the test, end test by returning to initial flow conditions 11.5 min after cutting power to EM pump.

Run	Test Section Inlet Temperature [°C (°F)]	Test Section Inlet Flow [P/s (gpm)]	Bypass to Test Section Flow Ratio	Total Bundle Power (kW)	Expansion Tank Gas Pressure [kPa (psi)]
101	390 (730)	0.84 (13.3)	max.	287	48 (7)
102	390 (730)	0.84 (13.3)	max.	287	69 (10)

Test 305

Purpose: The EM pump controllers will be set to ramp down the initial test section flow of 1.2 ℓ/s (19 gpm) to a low flow of 0.45 ℓ/s (7.1 gpm) over a specified time (approximately −0.40 $\ell \cdot s^{-2}$) after steady-state conditions have been established at the initial flow rate. This test is similar to Test 204, Run 107 of the Bundle 9 Phase 2, Part A test program which produced violent boiling starting approximately 4 s after low flow conditions were achieved. The test was terminated at approximately 27 s by an erroneous signal from a failed thermocouple just as the test section inlet flow decreased rapidly. The power/flow conditions of Test 204 Run 107 will be repeated but with a higher bypass-to-test section flow ratio and both a higher and lower test section inlet valve pressure drop. Also, the effect of decreasing the flow ramp down rate on boiling behavior will be investigated.

Procedure: 1. Set the test section inlet valve (normal setting corresponds to that specified in Test 300, procedure 2, maximum to procedure 5 and minimum to procedure 6), and the flow ramp down rate for each run as specified in the table below.

2. Set the valve on the bypass line to obtain the maximum bypass-to-test section flowsplit.

3. Obtain steady-state conditions of test section inlet temperature, outlet temperature and test section flow at the following conditions.

test section inlet temperature ... 390°C (730°F)
test section inlet flow ..... 1.2 $\ell \cdot s^{-1}$ (19 gpm)
total bundle power ..... 409 kW (6.7 kW/pin)

4. Record data in fast scan mode for 2.5 min, beginning at 10 s before the programmed flow reduction begins.

5. Begin the transient by starting the programmed flow reduction to the low flow of 0.45 ℓ/s (7.1 gpm).

6. If dryout conditions do not prematurely terminate the test, end test by returning to initial flow conditions 2 min after the beginning of the programmed flow reductions.

Run	Test Section Flow		Test Section Inlet Valve Setting	Flow Ramp Down Rate
	Initial	Low		
	[ℓ/s (gpm)]	[ℓ/s (gpm)]		[ℓ/s^2 (gpm/s)]
101	1.2 (19)	0.45 (7.1)	normal	−0.40 (−6.4)
102	1.2 (19)	0.45 (7.1)	max.	−0.40 (−6.4)
103	1.2 (19)	0.45 (7.1)	min.	−0.40 (−6.4)
104	1.2 (19)	0.45 (7.1)	normal	−0.013 (−0.20)

Test 306

Purpose: The EM pump will be turned off after steady-state conditions are established at a test section inlet flow of 9.84 ℓ/s (156 gpm) and a total bundle power 102 kW with a 3:1 diametral power skew. Due to the very low thermal driving head in the

bundle, it is expected that the inlet flow will drop rapidly to almost zero and then slowly rise as a higher temperature gradient develops in the bundle. This will allow the examination of intra-subassembly natural convection effects due to the power skew impressed across the bundle.

Procedure:

1. Set the test section inlet valve to the normal pressure drop setting (as specified in Test 300, procedure 2).
2. Adjust the bypass line valve to obtain 9.84 ℓ/s (156 gpm) through the test section.
3. Obtain steady-state conditions of test section inlet temperature, outlet temperature and test section flow at the following conditions:

   test section inlet temperature ... 390°C (730°F)
   test section inlet flow ..... 9.84 ℓ/s (156 gpm)
   total bundle power ..... 102 kW

4. Record data in fast scan mode for 10 min, beginning 10 s before cutting power to EM pump.
5. Cut power to EM pump.
6. If dryout conditions do not prematurely terminate the test, end the test by returning to initial flow conditions 9.5 min after cutting power to EM pump.

Pin power (kw/pin)	Percentage maximum power	Pins heated	Power Zones	Total power to each zone (kW)
0.83	33	58, 59, 60, 61, 38	B	4.15
1.05	42	57, 35, 36, 37, 20, 39	C	6.30
1.26	50	56, 34, 18, 19, 8, 21, 40	D	8.82
1.46	58	55, 33, 17, 7, 2, 9, 22, 41	E	11.68
1.68	67	54, 32, 16, 6, 1, 3, 10, 23, (42)	F(&A)	13.44 (1.68)
1.88	75	53, 31, 15, 5, 4, 11, 24, 43	J	15.04
2.08	83	52, 30, 14, 13, 12, 25, 44	L	14.56
2.31	92	51, 29, 28, 27, 26, 45	K	13.86
2.51	100	50, 49, 48, 47, 46	H	12.55

Test 307

Purpose: This test is designed to investigate the effect of reduced inlet subcooling on boiling stability under natural circulation conditions. Run 101 starts with the test section inlet temperature 55°C (100°F) above nominal, Run 102 has an inlet temperature of 111°C (200°F) above nominal.

Procedure:

1. Set the test section inlet valve to the normal pressure drop setting (as specified in Test 300, procedure 2).
2. Set the valve on the bypass line to obtain a 4:1 bypass-to-test section flow ratio.
3. Obtain steady-state conditions of the test section inlet temperature, outlet temperature and test section flow for the conditions given in the table below.
4. Record data in the fast scan mode for 12 min, beginning 10 s before cutting power to the EM pump.
5. Cut power to the EM pump.
6. If dryout conditions do not prematurely terminate the test, end test by returning to initial flow conditions 11.5 min after the pump trip.

Initial conditions for Test 307

	TS Inlet Temperature [°C (°F)]	TS Inlet Flow [$\ell \cdot s^{-1}$ (gpm)]	Bundle Power (kW)	TS Outlet Temperature [°C (°F)]
Run 101	443 (830)	0.73 (11.5)	201	705 (1300)
Run 102	499 (930)	0.73 (11.5)	158	705 (1300)

Test 308

Purpose: The EM pump controller will be set to ramp down the initial test section flow of 2.56 ℓ/s (40.6 gpm) to a low flow of 1.53 ℓ/s (24.3 gpm) over a specified time (approximately −0.40 $\ell \cdot s^{-2}$) after steady-state conditions have been established at

the initial flow rate. This test is similar to Test 205 Run 1021 of the Bundle 9 Phase 2, Part A test program which produced dryout conditions at approximately 27 s after low flow conditions were achieved. The test was terminated by a high temperature signal from a heater internal thermocouple just as the test section inlet flow decreased rapidly. The power/flow conditions of Test 205 Run 1021 will be repeated but with a higher bypass-to-test section flow ratio.

Procedure:

1. Set the test section inlet valve to the normal pressure drop setting (as specified in Test 300, procedure 2).
2. Set the valve on the bypass line to obtain the maximum bypass-to-test section flowsplit.
3. Obtain steady-state conditions of test section inlet temperature, outlet temperature and test section flow at the following conditions:

   test section inlet temperature ... 390°C (730°F)
   test section inlet flow ..... 2.56 $\ell \cdot s^{-1}$ (40.6 gpm)
   total bundle power ..... 915 kW (15 kW/pin)

4. Record data in fast scan mode for 2.5 min, beginning at 10 s before the programmed flow reduction begins.
5. Begin the transient by starting the programmed flow reduction to the low flow of 1.53 ℓ/s (24.3 gpm).
6. If dryout conditions do not prematurely terminate the test, end test by returning to initial flow conditions 2 min after the beginning of the programmed flow reductions.

1.2 THORS Bundle 9 Phase 2 Testing

R. H. Morris A. E. Levin
B. H. Montgomery W. R. Nelson

The final section (Part B) of the THORS Bundle 9 Phase 2 test program was completed during this report period. Major parameters are summarized for Part A tests (Table 1.2) and Part B tests (Table 1.3).

During Phase 2 Part B testing, a total of 23 runs were conducted and 20 produced some sodium boiling. Eleven of these runs are thought to have reached dryout conditions.

For the entire Bundle 9 Phase 2 operation, a total of 45 transient runs were conducted. Boiling was observed in 36 runs, 13 of which reached dryout. The total boiling time was ~140 min at temperatures near 1000°C.

The Bundle 9 Phase 1 record of experimental data report[3] was published during this report period, and the Bundle 9 Phase 2 record of experimental data report has undergone coauthor and peer review and is in the final stages of preparation.

1.3 THORS Fabrication and Operation

B. H. Montgomery

Following completion of the Bundle 9 Phase 2 test program, Bundle 9 had operated in sodium for approximately 5600 h, 234 h of which were with power on the fuel pin simulators (FPSs).

The THORS Facility operating history is summarized in Table 1.4.

1.4 THORS Modeling with NATOF-2D

A. E. Levin

Developed at MIT, NATOF-2D is a computer code that incorporates a two-dimensional (2-D) two-fluid model for sodium boiling. Developed in cylindrical (r-z) geometry specifically for multipin bundles in a hexagonal array, the model is directly applicable to Liquid-Metal Fast Breeder

Table 1.2. Summary of THORS Bundle 9 Phase 2 Part A transient boiling tests

Test	Run	Date	Inlet temperature (°C)	Power		Test section flow (ℓ/s)		Bypass flow (ℓ/s)		Time at low flow (s)	Boiling		Comments
				Total (kW)	(kW/pin)	Prior to reduction	Reduced	Prior to reduction	Reduced		Duration (s)	Intensity	
202	(Natural convection test)												
	100	5/14/80	393	191	3.1	0.53	0.26	1.99	−0.42	58	None	None	*a*
	101	5/14/80	395	191	3.1	0.54	0.27	1.98	−0.27	41	None	None	*a*
	102	5/14/80	390	191	3.1	0.56	0.28	2.00	−0.39	265	None	None	
	103	5/22/80	389	204	3.3	0.62	0.31	2.22	−0.46	689	667	Intense	
	104	5/22/80	383	227	3.7	0.68	0.32	2.48	−0.38	689	675	Violent	
203	(Natural convection test)												
	100	5/15/80	378	254	4.2	0.75	0.31	2.88	−0.45	15	2	Incipient	*b*
	101	5/15/80	386	254	4.2	0.75	0.29	2.90	−0.43	43	30	Intense	*b*
	102	5/22/80	387	253	4.1	0.74	0.28	2.85	−0.38	686	676	Intense	
	103	5/22/80	386	275	4.5	0.83	0.31	3.15	−0.43	608	600	Intense	*a*
	104	5/27/80	392	290	4.8	0.85	0.29	3.31	−0.40	13	6	Intense	*a*
	105	5/27/80	386	289	4.7	0.83	0.28	3.33	−0.50	66	58	Violent	*a,c*
204	(Controlled flow reduction test)												
	101	5/28/80	390	411	6.7	1.18	0.98	4.87	3.98	119	None	None	
	102	5/28/80	393	413	6.8	1.22	0.84	4.83	3.38	119	None	None	
	103	5/28/80	391	410	6.7	1.22	0.76	4.89	2.96	116	31	Incipient	
	104	5/28/80	391	411	6.7	1.17	0.61	4.82	2.54	116	111	Intense	
	105	5/29/80	391	409	6.7	1.22	0.64	4.79	2.33	116	110	Intense	
	106	5/29/80	393	408	6.7	1.19	0.56	4.79	2.10	43	38	Violent	*a,c*
	107	5/29/80	392	407	6.7	1.20	0.45	4.85	1.60	27	23	Violent	*a,c*
205	(Controlled flow reduction test)												
	101	6/2/80	387	904	14.8	2.61	2.08	10.39	8.28	58	None	None	
	102	6/2/80	390	924	15.2	2.61	1.65	10.53	7.01	56	55	Intense	
	103	6/2/80	392	920	15.1	2.59	1.47	10.47	6.08	12	12	Dryout	*a,b,d*
	1021	6/2/80	394	920	15.1	2.56	1.53	10.49	6.68	27	27	Dryout	*b,d*

[a]Run terminated by erroneous signal from failed thermocouple.

[b]Run terminated by high-temperature signal from thermocouple.

[c]Test section inlet flow decreased rapidly prior to thermocouple failure.

[d]Test section inlet flow decreased rapidly prior to dryout.

Table 1.3. Summary of THORS Bundle 9 Phase 2 Part B transient boiling tests

Test	Run	Date	Inlet temperature (°C)	Expansion tank pressure (kPa)	Test section inlet valve setting	Power		Test section flow (ℓ/s)		Bypass flow (ℓ/s)		Test section flow ramp rate (ℓ/s²)	Time at low flow (s)	Boiling		Comments
						Total (kW)	kW/pin	Prior to reduction	Reduced	Prior to reduction	Reduced			Duration (s)	Intensity	
303	(Natural convection test)															
	101	8/12/80	382	49.1	Normal	253	4.1	0.73	0.33	2.92	−0.53	Cut power to pump	371	361	Intense	*a,b*
	102	8/12/80	390	68.8	Normal	253	4.1	0.73	0.38	2.95	−0.50	Cut power to pump	686	675	Violent	
	103	8/13/80	389	49.5	Maximum	249	4.1	0.74	0.30	3.03	−0.48	Cut power to pump	99	91	Dryout	*a,c*
	104	8/12/80	388	46.9	Normal	252	4.1	0.73	0.35	3.63	−0.60	Cut power to pump	226	216	Dryout	*c*
	105	8/12/80	394	67.8	Normal	253	4.1	0.74	0.37	3.61	−0.57	Cut power to pump	594	583	Dryout	*c*
	1011	8/25/80	393	52.8	Normal	250	4.1	0.76	0.38	2.94		Cut power to pump	664	652	Dryout	*c,d,e*
	1012	8/25/80	387	51.5	Normal	249	4.1	0.73	0.39	3.02		Cut power to pump	686	675	Violent	*d,e*
304	(Natural convection test)															
	101	8/14/80	388	48.8	Normal	285	4.7	0.84	0.37	4.21	−0.60	Cut power to pump	58	52	Dryout	*c*
	102	8/14/80	392	70.9	Normal	290	4.7	0.87	0.36	4.30	−0.43	Cut power to pump	74	66	Dryout	*c*
305	(Controlled flow reduction test)															
	101	8/18/80	388	52.8	Normal	409	6.7	1.23	0.48	9.70	2.22	−0.20	18	14	Dryout	*c*
	102	8/18/80	387	51.8	Maximum	408	6.7	1.21	0.42	8.64	2.26	−0.20	21	17	Dryout	*a,c*
	103	8/18/80	387	53.1	Minimum	411	6.7	1.20	0.43	8.56	1.63	−0.26	18	14	Dryout	*a,c*
	104	8/15/80	393	49.2	Normal	413	6.8	1.22	0.49	9.76	2.86	−0.012	0	18	Intense	*f,g*
	1041	8/15/80	386	49.1	Normal	411	6.7	1.24	0.35	9.67	2.50	−0.14	6	25	Dryout	*c,g*
306	(Natural convection test with a 3:1 diametral power skew)															
	101	8/19/80	395	48.5	Normal	99		9.79	0.13	5.83	−0.03	Cut power to pump	44		None	*b,h*
	102	8/19/80	391	50.9	Normal	99		9.66	0.12	6.09	−0.07	Cut power to pump	352		None	*b,h*
	103	8/22/80	389	50.4	Normal	102		9.80	0.10	5.22		Cut power to pump	689		None	*d,e,h*
307	(Natural convection test)															
	101	8/19/80	441	51.0	Normal	198	3.2	0.74	0.37	2.92	0.01	Cut power to pump	273	257	Violent	*i*
	102	8/19/80	510	52.9	Normal	160	2.6	0.73	0.31	2.75	−0.17	Cut power to pump	118	106	Violent	*b,i*
	1021	8/19/80	515	51.7	Normal	160	2.6	0.73	0.29	2.76	−0.35	Cut power to pump	66	55	Violent	*b,i*
	1011	8/22/80	458	51.5	Normal	196	3.2	0.74	0.32	2.86		Cut power to pump	687	676	Violent	*d,i,e*
	1022	8/22/80	486	49.8	Normal	161	2.6	0.75	0.29	2.98		Cut power to pump	685	667	Violent	*d,i,e*
308	(Controlled flow reduction test)															
	101	8/20/80	390	49.9	Normal	907	14.9	2.52	1.38	14.08	7.97	−0.38	9	9	Dryout	*c*

[a]Test section inlet valve settings: "Normal" - ΔP across valve is 115% of test section ΔP at 8.77 ℓ/s test section inlet flow. This setting results in a ΔP of 37 kPa across the valve at 2.59 ℓ/s. "Maximum" - ΔP across valve is twice "normal" (74 kPa at 2.59 ℓ/s); "Minimum" - ΔP across valve is one-half "normal" (18.5 kPa at 2.59 ℓ/s).

[b]Run terminated because of erroneous signal by failed thermocouple.

[c]"Dryout" indicated by rapid test section inlet flow decay due to void formation in bundle; termination of test accomplished by manual power trip.

[d]Run performed without high limit DAS termination capability.

[e]Operational problems prevented accurate recording of bypass flow during transient; expansion tank pressure, initial bypass flow, and inlet temperature taken from pretest operator's log.

[f]Run was terminated because of erroneous signal by failed thermocouple just as controlled flow ended.

[g]Note extremely slow flow decay for test.

[h]Pin power ranged from ∿0.83 to ∿2.5 kW/pin.

[i]Note elevated inlet temperature for test.

Table 1.4. THORS operating experience

Bundle identification	Duct configuration	Number of heated pins	Blockage configuration	Heated length (mm)	Fission gas simulators	Isothermal operation (h)	Operation with power (h)
1A	Scalloped (bare edge rod)	19	None	610	No	1,300	200
2A	Hex (full-size edge gaps)	19	None	533	No	3,010	470
2B	Hex (full-size edge gaps)	19	13- and 24-channel inlet				
1B	Scalloped (bare edge rod)	19	None	610	No	894	350
3A	Scalloped (wrapped-edge rods)	19	6 channel in heated zone	533	No	3,039	151
3B[a,b]	Scalloped (wrapped-edge rods)	19	6 channel in heated zone	533	No	537	51
5A, 5B, 5C	Hex (half-size edge gaps)	19	12-channel edge gap[c]	457	No	3,252	391
5D[a]	Hex (half-size edge gaps)	19	None	457	No	2,827	97
6A[d]	Hex (half-size edge gaps)	19	None	914	FFTF length	7,864	380
3C[e,f]	Scalloped (wrapped-edge rods)	31[f]	6 channel in heated zone	533	FFTF length	3,666	190
9[g]	Hex (full-size edge gaps)	61	None	914	CRBR length	5,602	234
					Total	31,991	2,514

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

[b]Boiling tests included 45 s with full bundle (heater 7 opened during boiling tests with gas injection) and 11 min 15 s with rod 7 inoperative; 10 min 16 s 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-in. displacement for 5B, removed for 5C.

[d]A total of 38 tests were run in which boiling occurred; five runs were to dryout (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.

[f]A major modification of bundle 3B, this was a 19-pin bundle surrounded by 12 heated edge pins replacing the scalloped duct configuration of Bundle 3B.

[g]A total of 36 tests were run in which boiling occurred; 13 of these were to dryout conditions; total boiling time for the bundle (Phase 2, Part A) was ~139 min.

Reactors (LMFBRs) or out-of-pile experiments, such as THORS, which use hexagonal simulated LMFBR fuel bundles. The final version of this code, written by M. R. Granziera as part of his doctoral dissertation[4] (supervisor — Professor M. S. Kazimi) has been transmitted to ORNL.

Previous modeling of THORS has been done with the SAS and SABRE computer codes.[1,5] SAS incorporates a one-dimensional (1-D) fixed regime boiling model. SABRE is written for single-phase (all liquid) flow; however, it has been modified at ORNL by the inclusion of a homogeneous mixture model for two-phase flow. SABRE, as modified, has been quite successful in modeling average conditions during THORS boiling tests, as discussed in Sect. 1.5. However, SAS has been less than satisfactory, primarily because boiling in THORS is not 1-D in nature. While SABRE predicts average conditions quite well, it cannot predict the rapid flow oscillations that occur upon boiling inception in THORS. These oscillations appear to have little effect on the time to dryout in the THORS boiling tests; however, an understanding of the mechanisms involved may assist in determining whether oscillations play a role in preventing early dryout, and how they do so. Because the boiling model is 2-D, NATOF-2D has the potential to model these oscillations and should be able to simulate THORS transients more accurately than SAS, giving a 2-D representation of the growth of the voided zone in the bundle. The two-fluid model in NATOF-2D treats the sodium liquid and vapor phases each as a separate fluid and attempts to model heat, mass, and momentum exchange between them, as well as tracking their temperatures, velocities, and volumetric amounts (void fraction) independently. This representation may be superior to mixture models (such as that used in the ORNL-modified SABRE code), provided the interaction models are sufficiently accurate.

After NATOF-2D was received, one minor error was found and corrected. Runs then produced the expected output from the sample problems. In reviewing the output, inconsistencies were found that could not be explained. Other errors and inconsistencies were found in Granziera's report and have been brought to Professor Kazimi's attention. In addition, NATOF-2D is currently written for nuclear fuel pins only. Although written for the THERMIT computer code, a model exists for the boron nitride—stainless steel (BN-SS) FPSs used in THORS. MIT has been asked to supply

a copy of this model to ORNL for implementation in the NATOF-2D code that will be used to model the THORS Bundle 9 boiling tests. If NATOF-2D is successful in this effort, serious consideration will be given to its use as a scoping tool in developing a test plan for the proposed THORS—Shutdown Heat Removal System (THORS-SHRS) experiments.

1.5 Computational Analysis of Sodium Boiling Experimental Data from THORS Bundles 6A and 9

J. F. Dearing S. D. Rose

1.5.1 Description of the computational model

Many corrections, modifications, and additions have been made to the copy of SABRE-2 (Ref. 6) (version dated June 1, 1978) that ORNL received last year. These changes have allowed accurate modeling of transient single-phase flow in THORS Bundle 9 (including natural convection with ΔP boundary conditions on the test section).[1] Recently, inspection of boiling data from Bundle 9 suggested that the inclusion of a relatively simple boiling model in SABRE-2 might have the potential to reflect the essential features of the static (Ledinegg) flow excursion to dryout. The essential features of the boiling model (which will herein be referred to as SABRE-2P) will be described in the remainder of this section, and comparisons of model results with experimental boiling data from THORS Bundles 6A and 9 will be given in the following two sections, respectively. Work is continuing to gain an understanding of why this model, which neglects some terms and approximates others, works so well.

Essential features of the SABRE-2P boiling model

1. A local two-phase quality is calculated on the basis of subchannel enthalpy. This quality is used to calculate a subchannel two-phase friction-factor multiplier (homogeneous viscosity corrected) for inclusion in the axial momentum equation.
2. The subchannel quality is also used to calculate a void fraction (slip = 1) which is used to modify the gravitational body force term in the axial momentum equation.

3. Thermal conductivity, density, and viscosity are evaluated as liquid properties at the saturation temperature. The enthalpy field solved by the energy equation is, however, that of the two-phase mixture (or, equivalently, a superheated liquid).
4. Note especially that the density change on boiling is not included in the continuity equation for the following reasons.
   a. Inclusion of this term results in flow oscillations that are extremely hard to model numerically, even in one dimension.
   b. Adequate correlations for interphase heat and momentum transfer during these oscillations do not exist.
   c. Even if these two problems were solved, the resulting three-dimensional (3-D) code would be so time consuming as to be impractical to use.

These oscillations about the mean flow are instead viewed as enhanced two-phase turbulence that affects the effective thermal diffusivity of the boiling region.

Preliminary thoughts on the validity of modeling assumptions

Inspection of data from Bundle 9 indicates that a successful model must

1. accurately predict heat transfer in the boiling region, both axial and radial, and
2. accurately predict the effect of boiling on the local axial flow. The two most important effects are the two-phase friction multiplier (which tends to decrease the flow) and the gravitational body force (which tends to increase the flow).

Factors (1) and (2) are strongly coupled to produce the 3-D growth rate of the boiling region. The model must then calculate both the transient flow redistribution around and through the boiling region and the associated change in the total inlet flow produced by a constant Δp boundary condition. This capability is SABRE's forté.

Many expressions and correlations exist for the estimation of the two-phase friction-factor multiplier used in factor (2). The viscosity-corrected homogeneous multiplier is used here.

The difficult part of the problem is the modeling of factor (1). As a first guess, liquid properties were assumed in the solution of the energy equation — with remarkably accurate results. The explanation for this agreement appears to be that the increased axial two-phase resistance of the homogeneous multiplier produces the correct axial distribution of *transverse mass flux* into and out of the boiling region. The transverse convection term is the dominant term in the energy balance in the boiling region until it completely encompasses the bundle cross section. Note that the model predicts neither the correct density or transverse velocity, but does predict correctly their product, the transverse mass flux, which convects the enthalpy.

1.5.2 Comparison of SABRE-2P results with sodium boiling data from THORS Bundle 6A

The one-twelfth section model of Bundle 6A used in SABRE-2P is shown in Fig. 1.1. Nodes 1 through 4 represent the subchannel arrangement used in the SABRE-2P thermal-hydraulic analysis. Nodes 5 through 12 represent the 1-D thermal conduction model of the Bundle 6A housing incorporated in subroutine B6SRC. This subroutine is called at each axial level at each

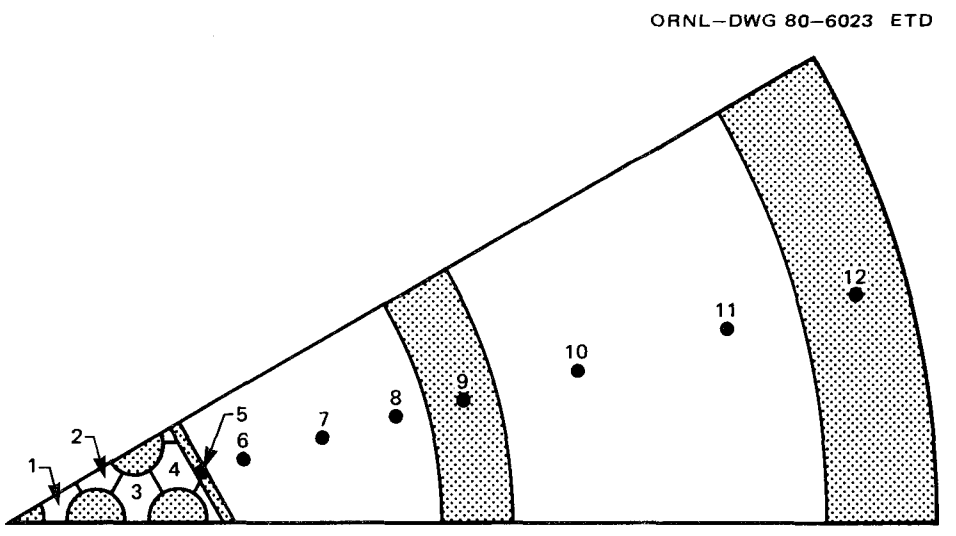

Fig. 1.1. One-twelfth-section thermal-hydraulic model of THORS Bundle 6A.

time step. The input argument is the temperature of the edge subchannel (node 4); the output is the heat flux from the duct wall (node 5) to the edge subchannel.

In the two-phase testing of 19-pin Bundle 6A, ~30 flow transients culminated in boiling, and 5 were continued to dryout. Test 71H Run 101 was the controlled flow reduction run at the lowest power that exhibited dryout, and Test 73E Run 102A was the highest power dryout run. These two runs were chosen for analysis with SABRE-2P, and their conditions are given in Table 1.5.

Table 1.5. Two bundle 6A boiling runs that reached dryout

	Test 71H Run 101	Test 73E Run 102A
Initial flow, ℓ/s	0.39	1.26
Power, kW/pin	6.8	15.3
Inlet temperature, °C	386.6	450.7

In this analysis, 24 axial nodes, each 0.1143 m in length, are used: 3 upstream of the heated section, 8 for the heated section, and 13 for the simulated fission-gas plenum (SFGP) region. A wire-wrap mixing model is not used in this version of SABRE; but a value for the turbulence mixing multiplier, FMIX = 10, is used which has been found suitable from single-phase analyses.[2] Subchannel areas and wetted perimeters are corrected on an average basis for the presence of the wire wraps.

A change in the code was made to represent the effects of transverse thermal conduction through the BN FPSs. This change was implemented by using a value of 2.3 for the shape thermal conduction factor calculated using a finite difference scheme for internal subchannels.[7] In SABRE, the heat source into the coolant is represented by a simple homogeneous two-region model for fuel pins and the coolant assuming a linear solution to the heat transfer between them.

Subroutine B6SRC calculates the radial heat conduction heat transfer in the duct wall and outer structure. Because of sodium leakage from the hexcan into the thermal insulation, an effective value of thermal conductivity of the insulation is used. Investigations with a run in which the temperature almost reached boiling (Test 41H Run 104) indicate that the value of the effective thermal conductivity of the sodium-permeated insulation should be 0.4. Close investigations of this run indicate that a 2-D radial heat conduction model is necessary to model exactly the temperatures in the outer structure. However, for the purposes of this investigtion, a 1-D model was found to be adequate.

The inlet valve of the bundle is represented in SABRE as a $K\rho V^2$ term on the first axial node. Pressure-drop investigations at a comparable flow indicate that a value of K = 8.5 represents the setting of the inlet valve.

Figure 1.2 shows the test section inlet flow together with the SABRE-2P prediction for Test 73E Run 102A, the highest power dryout run. (The flow is normalized in cubic centimeters per second per pin to enable

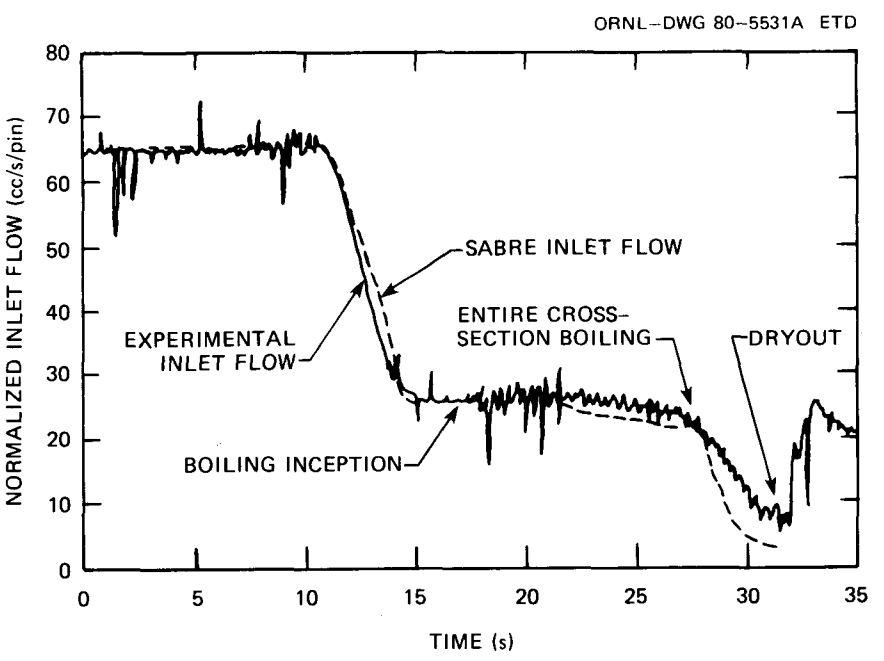

Fig. 1.2. Test section inlet flow: Test 73E, Run 102A, Bundle 6A — experimental data and SABRE-2P results.

direct comparison between Bundles 6A and 9.) The flow was reduced at about 10 s, and the present low flow is established 2 s later. At about 6.4 s into the transient (16.4 s), boiling starts centrally in the bundle near the end of the heated section. At about 28 s, the boiling region has grown radially to encompass the entire bundle cross section. The static (Ledinegg) flow instability then occurs and results in dryout at about 32 s. The flow decays rapidly once the complete cross section of the bundle is boiling, and the test is terminated once dryout has been detected.

The SABRE-2P calculation utilized a pressure-drop boundary condition. The test section pressure drop was reduced from the steady-state value to the low-flow value (based on a previous forced-flow boundary condition calculation) and then kept at this constant value. Because of a 5% reduction in the total test section power during the transient, three power ramps were used to model closely its change during this run.

Figure 1.2 indicates that the SABRE-2P predicts the Ledinegg flow excursion accurately. Boiling initiation is predicted at 1.4 s (experimentally detected at 16.4 s), and the entire cross section reaches boiling at 28 s in agreement with the experiment; the flow excursion to dryout is then predicted. Figure 1.3 shows schematically the development of the boiling region at 28 s, together with the positions of selected thermocouples at different radial and axial locations. In temperature plots for these thermocouples (Figs. 1.4 through 1.7), lines indicate the thermocouple traces and solid circles show the SABRE predictions.

Agreement between SABRE and the experimental results for this run is good, showing that the code accurately predicts the axial and radial development of the void. Reasons for the possible success of the SABRE-2P with a simple non-dynamic boiling model have already been discussed. Notice that when the complete cross section is boiling that the boiling zone extends axially between thermocouples H0523U and 0459U (Fig. 1.3) at the center of the bundle. The axial extent of the boiling zone is therefore about 0.3 m upstream from the end of the heated section to about 0.6 m downstream of the heated section into the SFGP region. Temperatures from thermocouples H0333U and H1633U are shown on the same plot to indicate the variation for thermocouples at the same radial position (midway radially from the center). The experimental variation in these temperatures at the

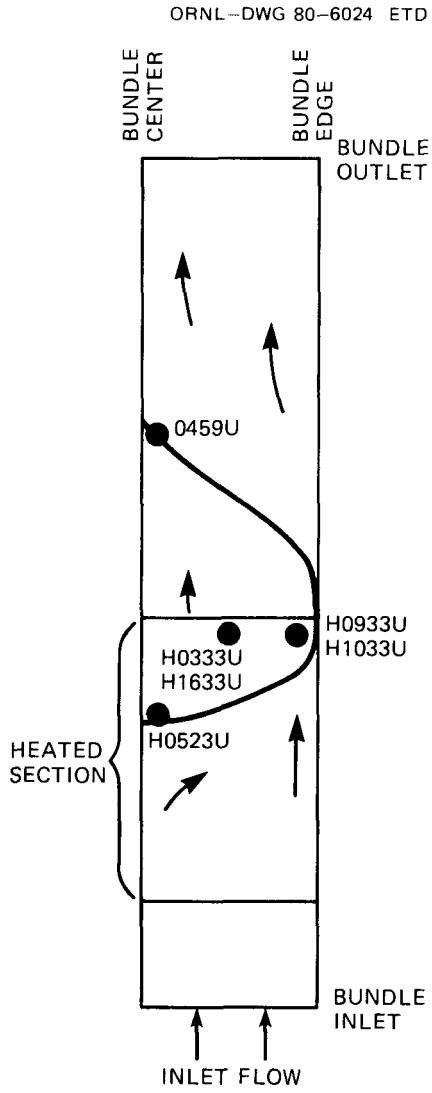

Fig. 1.3. Development of boiling region at 28 s in Test 73E, Run 102A, THORS Bundle 6A.

same radial positions may be caused by pin distortion, duct wall movement, and wire-wrap mixing — all of which reduce the symmetry of the temperature field in the bundle.

SABRE-2P was also used to model Test 71H Run 101, the lowest power dryout run for Bundle 6A. Figure 1.8 shows the test section inlet flow and the SABRE-2P prediction. Again a constant pressure-drop boundary condition was used. During the transient, a 7% power reduction occurred and was modeled by three power ramps. Experimentally, large oscillations existed on the inlet flow, some going to negative values. These (density

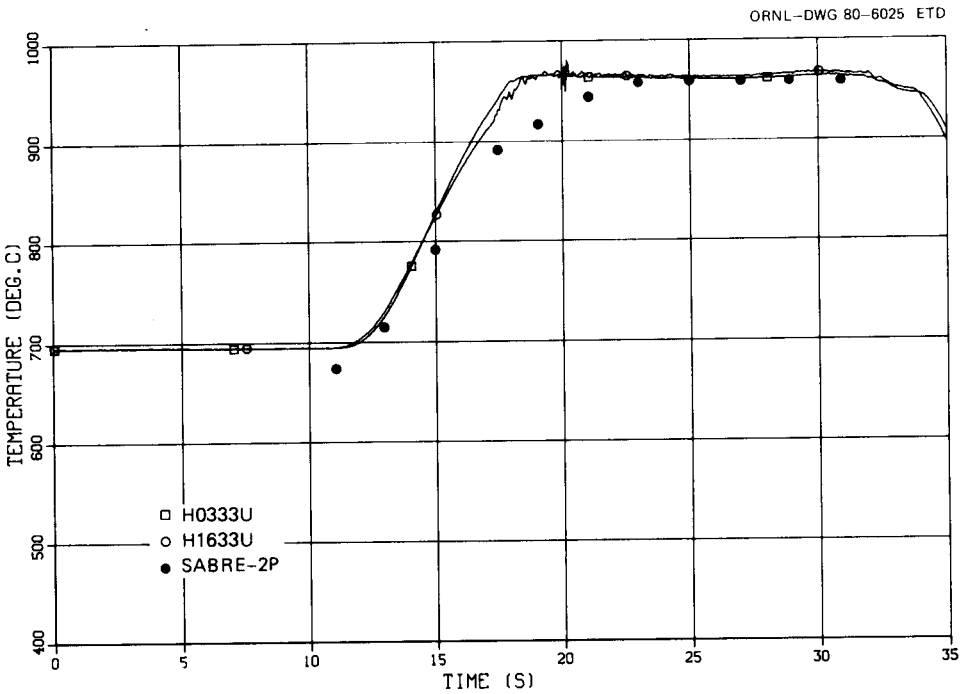

Fig. 1.4. Experimental vs SABRE-2P calculated transient temperatures for Test 73E, Run 102A (thermocouples H0333U and H1633U).

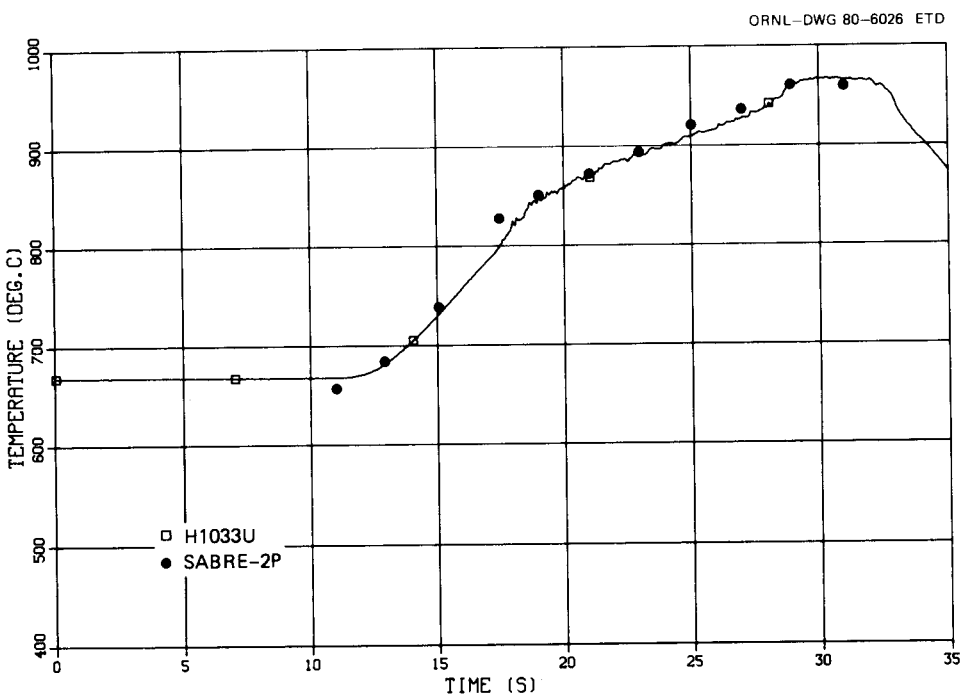

Fig. 1.5. Experimental vs SABRE-2P calculated transient temperatures for Test 73E, Run 102A (thermocouple H1033U).

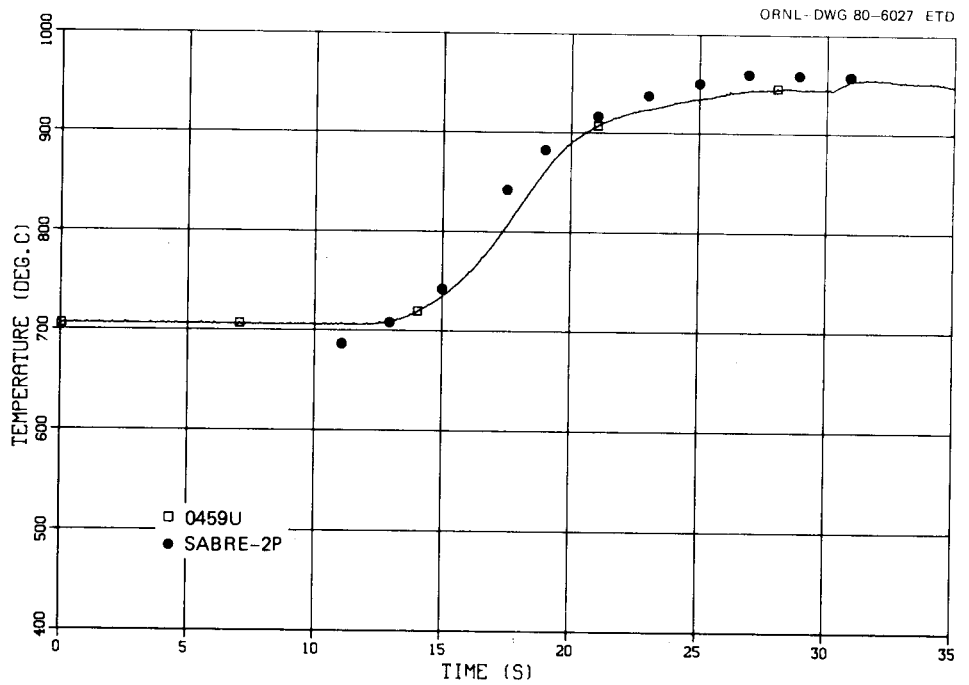

Fig. 1.6. Experimental vs SABRE-2P calculated transient temperatures for Test 73E, Run 102A (thermocouple 0459U).

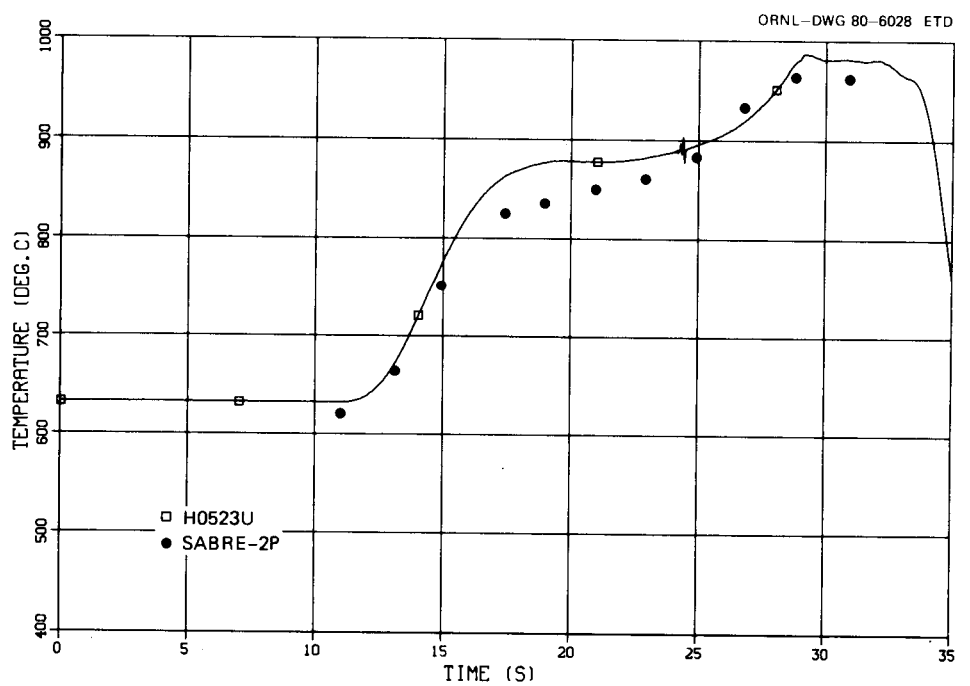

Fig. 1.7. Experimental vs SABRE-2P calculated transient temperatures for Test 73E, Run 102A (thermocouple H0523U).

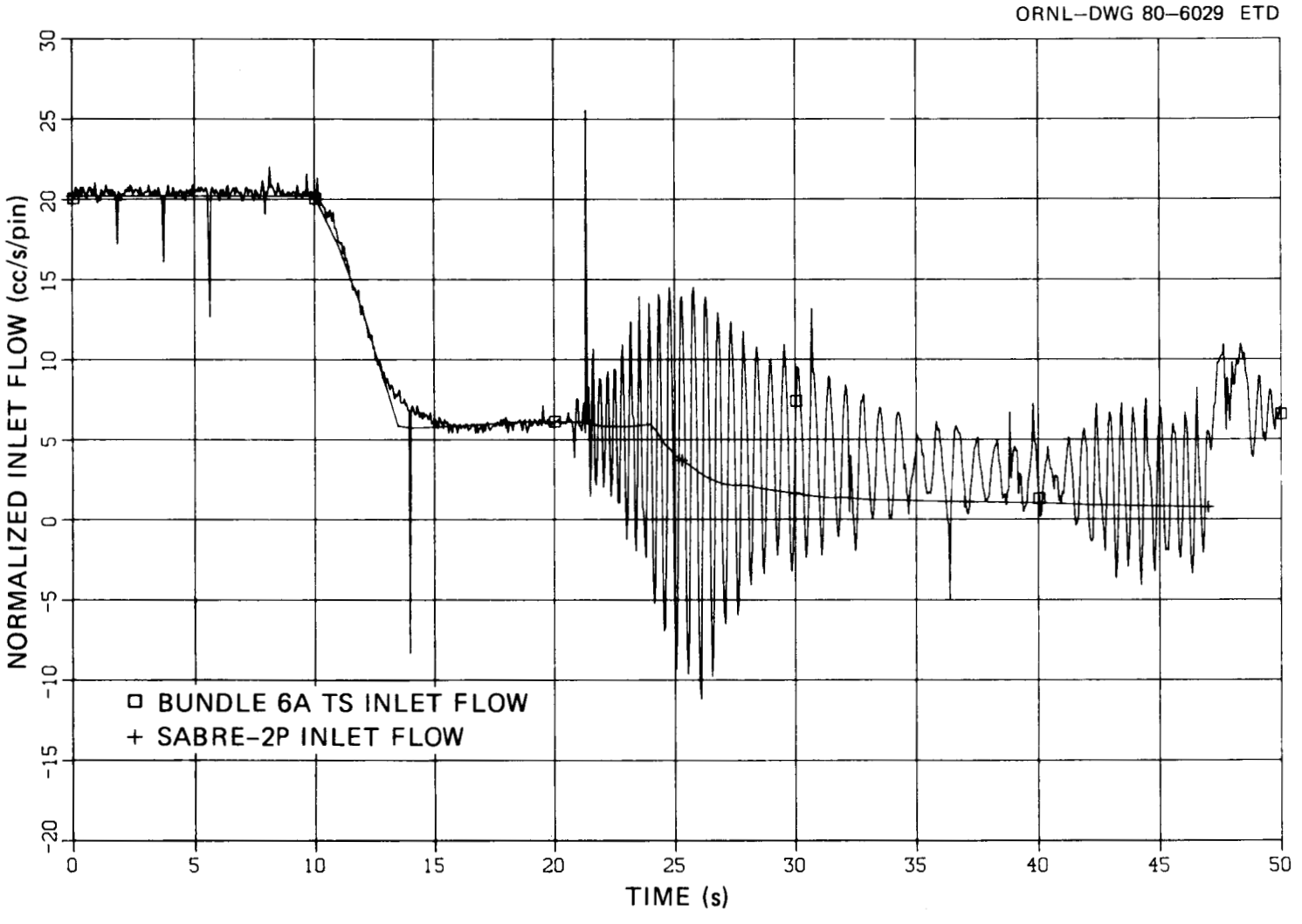

Fig. 1.8. Test section inlet flow: Test 71H, Run 101 — experimental data and SABRE-2P results.

wave) dynamic effects will not be shown with the non-dynamic boiling model in SABRE-2P. What is important is that SABRE-2P predicts the mean flow and accurately predicts the timing of the Ledinegg excursion. Boiling inception is at about 20.5 s (10.5 s into the transient). The complete bundle cross section is boiling by about 25 s, then the mean flow gradually decays as the boiling region grows. Figures 1.9 through 1.12 show bundle temperatures and the SABRE-2P predictions for thermocouples at different locations in the bundle.

The SABRE-2P results for Test 71H Run 101 do not match the experiment as well as the higher power Test 73E Run 102A. For Test 71H Run 101, SABRE-2P shows that the boiling region develops ~2 s ahead of the experiment. The SABRE-2P runs are sensitive to exact experimental conditions, particularly test-section power, and this has been modeled closely by three appropriate power ramps during the transient. In Test 71H Run 101, the boiling region is slower to develop because the run is at a lower power, and ~27 s of boiling occurred before permanent dryout; whereas in Test 73E Run 102A, about 16 s of boiling occurred before dryout.

1.5.3 Comparison of SABRE-2P results with sodium boiling data from THORS Bundle 9

SABRE-2P has been used in the preliminary analysis of four Bundle 9 boiling runs. Experimental conditions for these runs are summarized in Table 1.6. The first two runs are controlled forced flow reduction to boiling at conditions comparable to the two Bundle 6A boiling runs discussed in Sect. 1.5.2. The third run is a natural convection run in which the test-section inlet valve resistance was doubled from its nominal value of 1.15 times the test-section pressure drop at 0.73 ℓ/s. The fourth run is a controlled forced flow reduction run in which the rate of flow reduction is reduced.

The one-twelfth section model of Bundle 9 is shown in Fig. 1.13 (compare with the similar one-twelfth section of Bundle 6A shown in Fig. 1.1). The non-boiling aspects of this model are discussed in detail in Ref. 1.

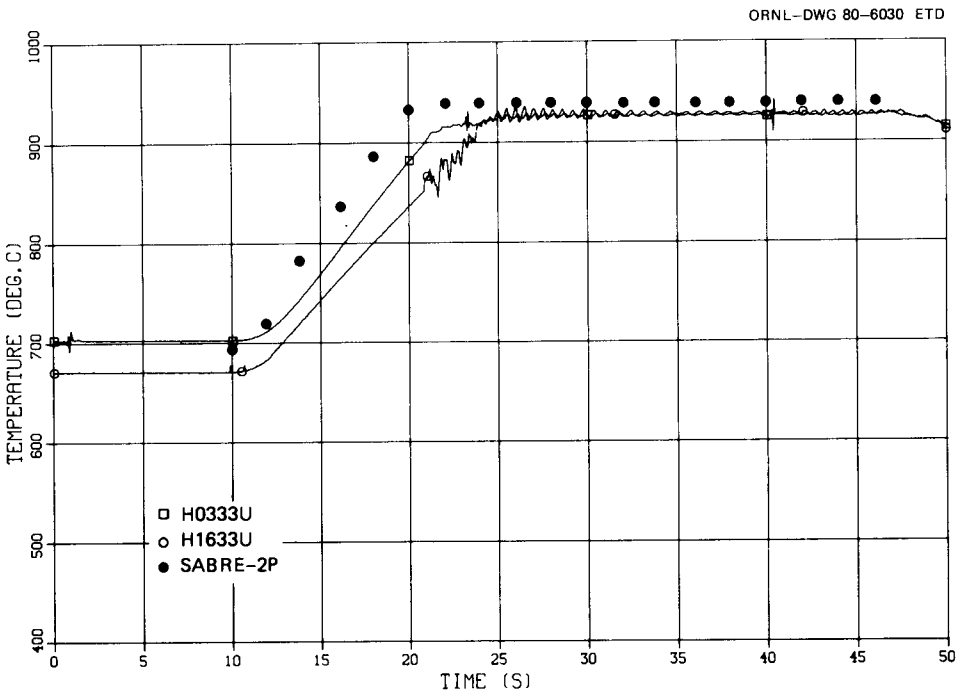

Fig. 1.9. Experimental vs SABRE-2P transient temperatures for Test 71H, Run 101, THORS Bundle 6A (thermocouples H0333U and H1633U).

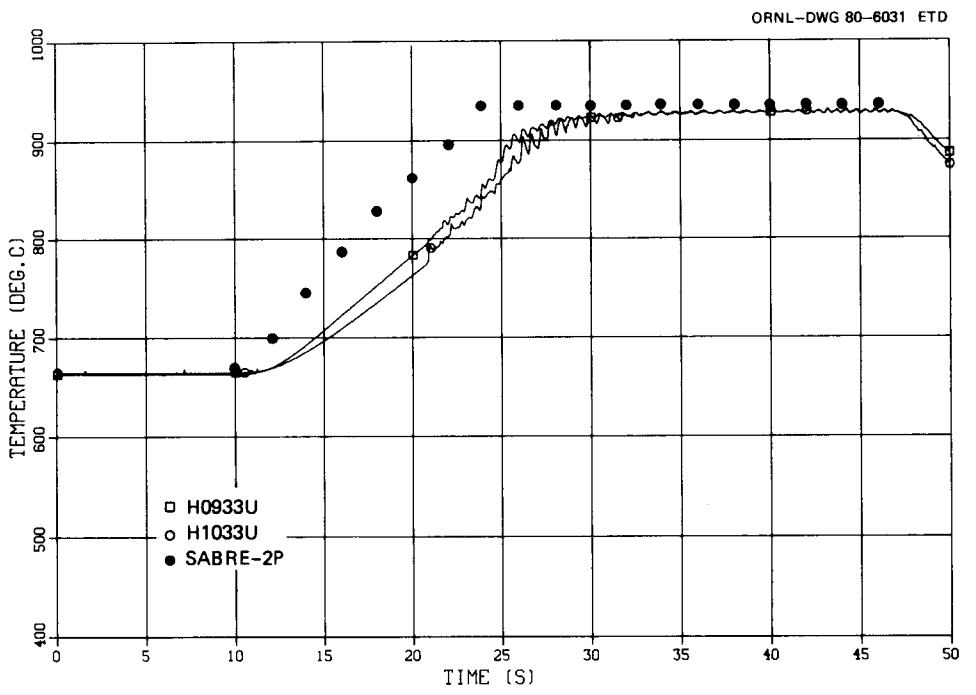

Fig. 1.10. Experimental vs SABRE-2P transient temperatures for Test 71H, Run 101, THORS Bundle 6A (thermocouples H0933U and H1033U).

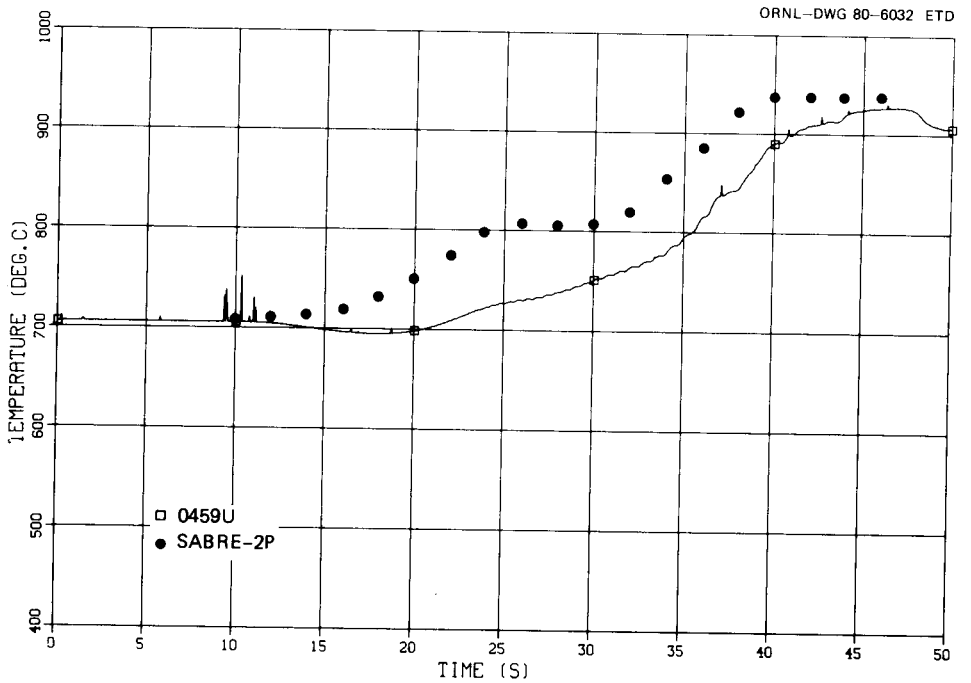

Fig. 1.11. Experimental vs SABRE-2P transient temperatures for Test 71H, Run 101, THORS Bundle 6A (thermocouple 0459U).

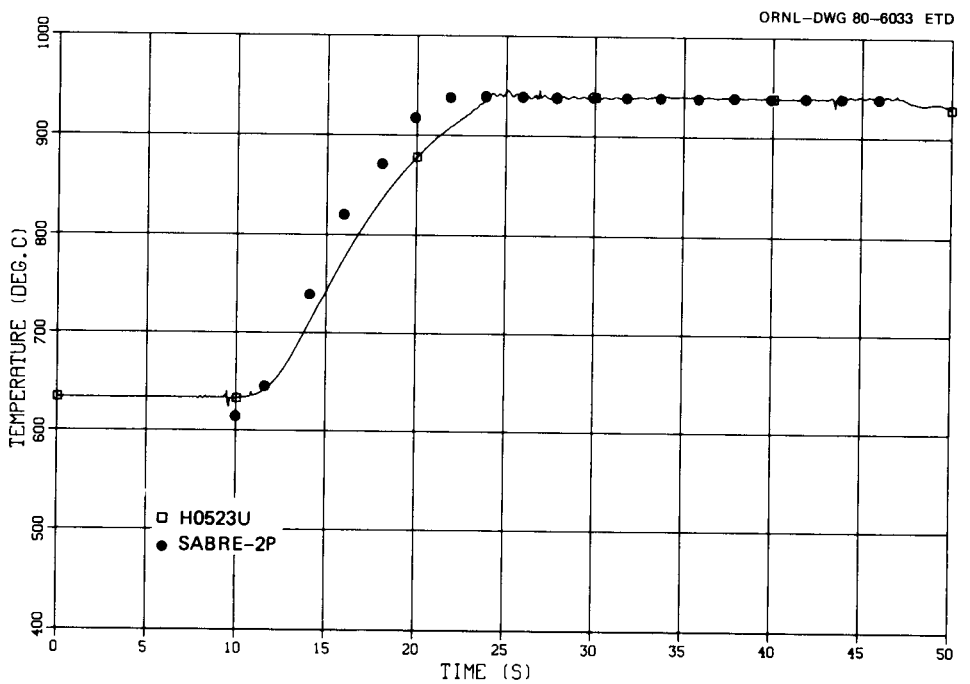

Fig. 1.12. Experimental vs SABRE-2P transient temperatures for Test 71H, Run 101, THORS Bundle 6A (thermocouple H0523U).

Table 1.6. Experimental conditions of four Bundle 9 sodium boiling tests

Test	Identification	Type	Pin power (kW/pin)	Initial test section flow (ℓ/s)
205	Run 103	Controlled forced-flow reduction	15.1	2.61
204	Run 107	Controlled forced-flow reduction	6.7	1.19
303	Run 103	Natural convection, increased test section inlet valve resistance	4.1	0.73
305	Run 1041	Extended controlled forced-flow reduction	6.7	1.20

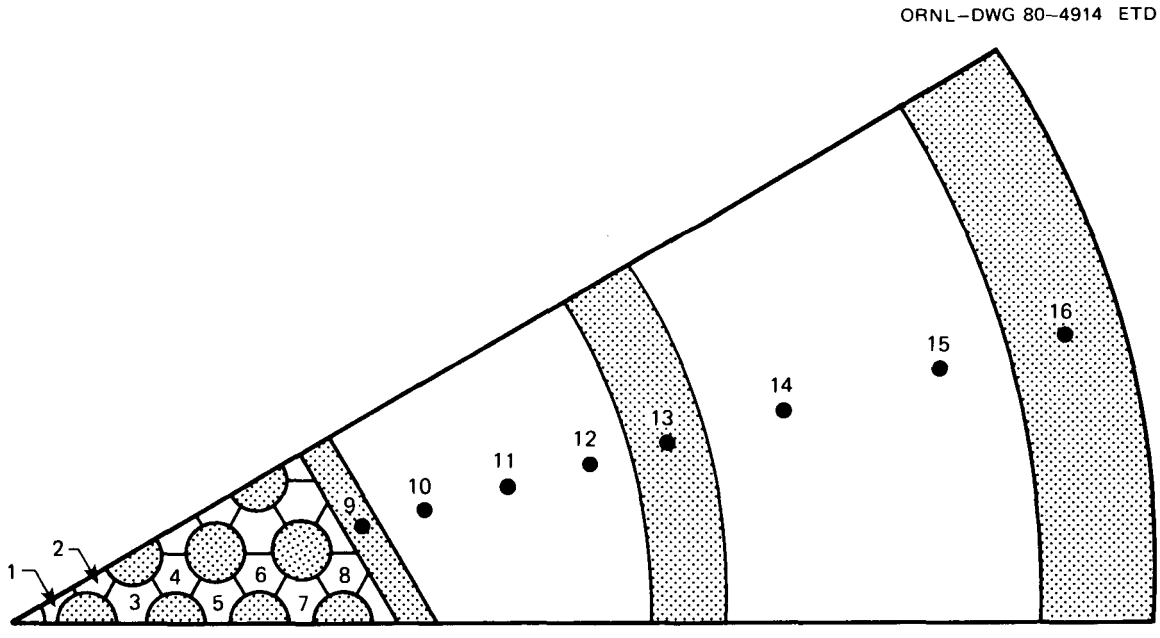

Fig. 1.13. One-twelfth-section thermal-hydraulic model of THORS Bundle 9.

Test 205 Run 103 (Controlled Forced Flow Reduction)

The experimental test section inlet flow for this run is compared with results of the SABRE-2P model (Fig. 1.14). Boiling begins in the bundle center near the end of the heated section shortly after controlled forced flow reduction (at ~15 s). The boiling region expands, both axially and radially, until it encompasses the entire bundle cross section near the end of the heated section at ~24 s. At this time the static instability causes a rapid excursion of the inlet flow to a low level that results in dryout at ~27 s. The SABRE-2P model accurately shows the 3-D flow diversion around the developing boiling region and the rapid reduction of the inlet flow when the boiling region encompasses the entire cross section.

The approximate development of the boiling region at 21 s is shown in Fig. 1.15. Arrows on this figure indicate velocity vectors at a few positions in the bundle. Also shown are the locations of four wire-wrap thermocouples. A comparison of the experimental thermal transients at these locations with SABRE-2P results is shown in Figs. 1.16 through 1.19.

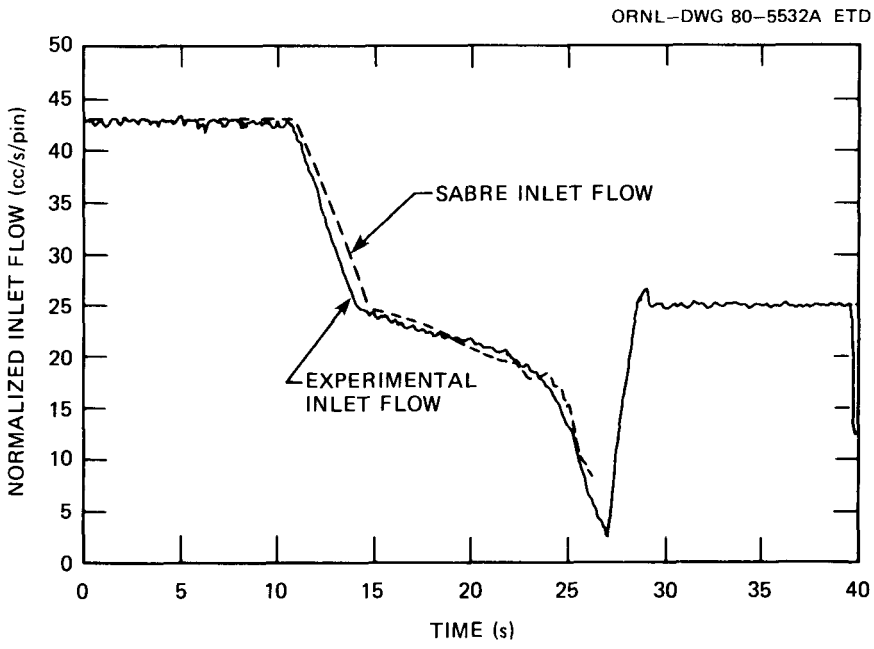

Fig. 1.14. Test section inlet flow: Test 205, Run 103, Bundle 9 — experimental data and SABRE-2P results.

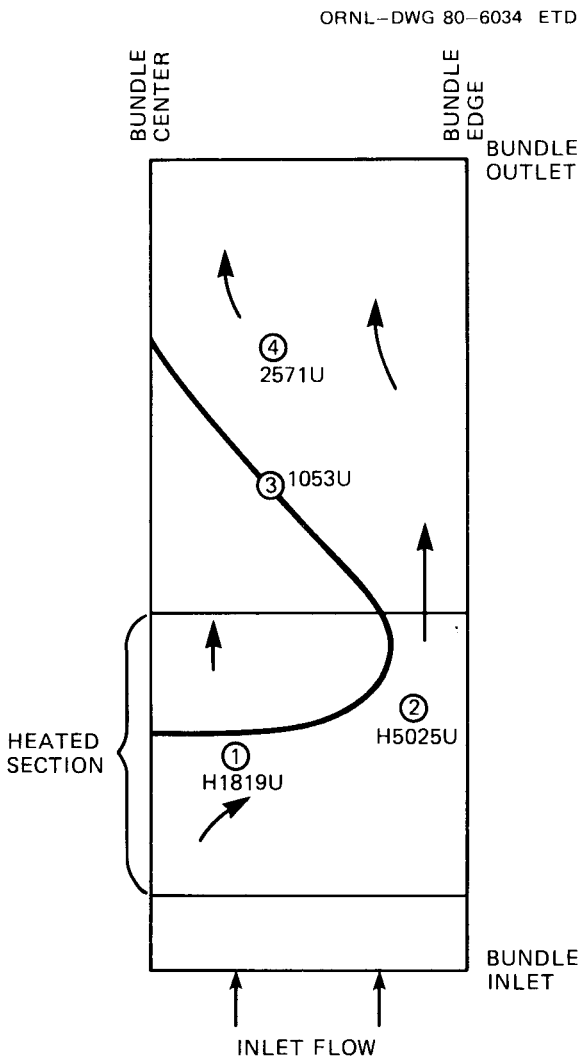

Fig. 1.15. Development of boiling region at 21 s in Test 205, Run 103, THORS Bundle 9.

Test 204 Run 107 (Controlled Flow Reduction)

A comparison of the experimental test section inlet flow for this run with results of the SABRE-2P model is shown in Fig. 1.20. Boiling begins near the bundle center near the end of the heated section at ~18 s. The boiling region has expanded to fill the entire bundle cross section at ~31 s, when the static flow excursion begins.

The approximate development of the boiling region at 28 s is shown in Fig. 1.21. A comparison of experimental vs SABRE-2P thermal transients is

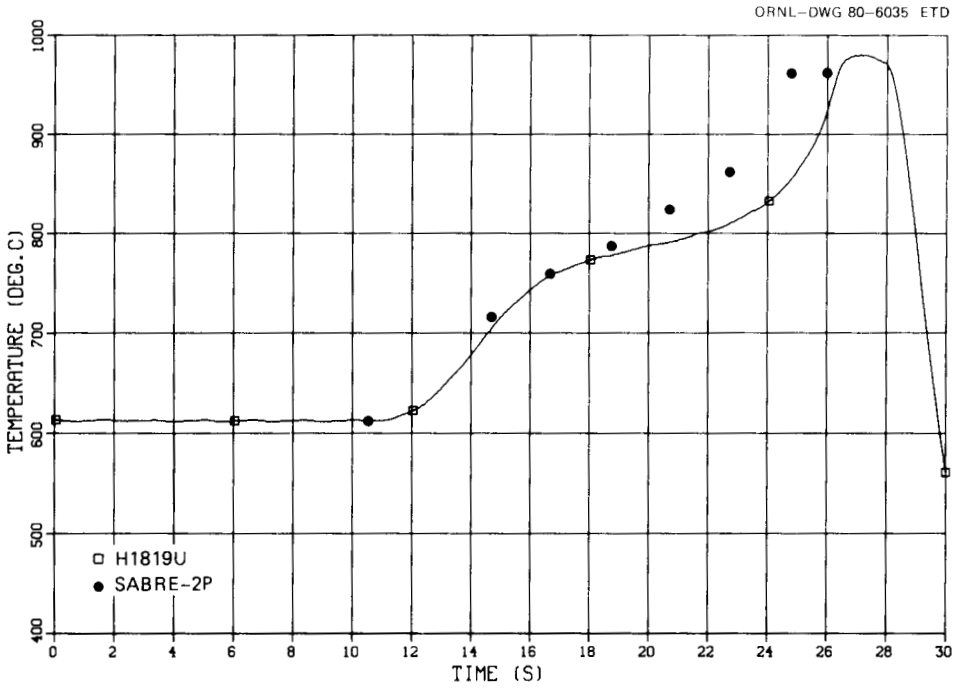

Fig. 1.16. Experimental vs SABRE-2P calculated transient temperatures for Test 205, Run 103, THORS Bundle 9 (thermocouple H1819U).

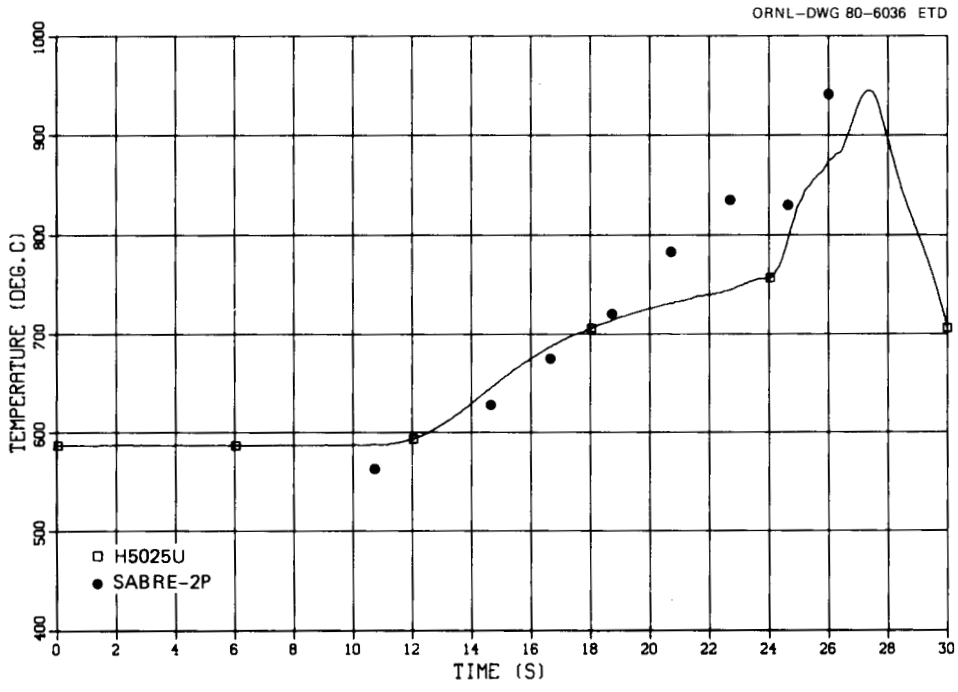

Fig. 1.17. Experimental vs SABRE-2P calculated transient temperatures for Test 205, Run 103, THORS Bundle 9 (thermocouple H5025U).

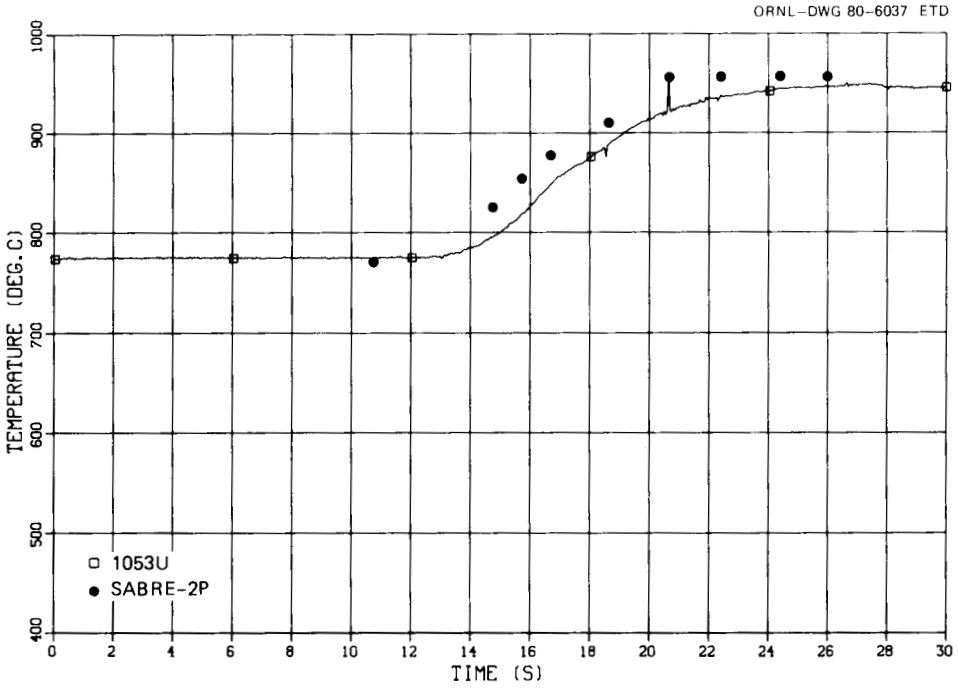

Fig. 1.18. Experimental vs SABRE-2P calculated transient temperatures for Test 205, Run 103, THORS Bundle 9 (thermocouple 1053U).

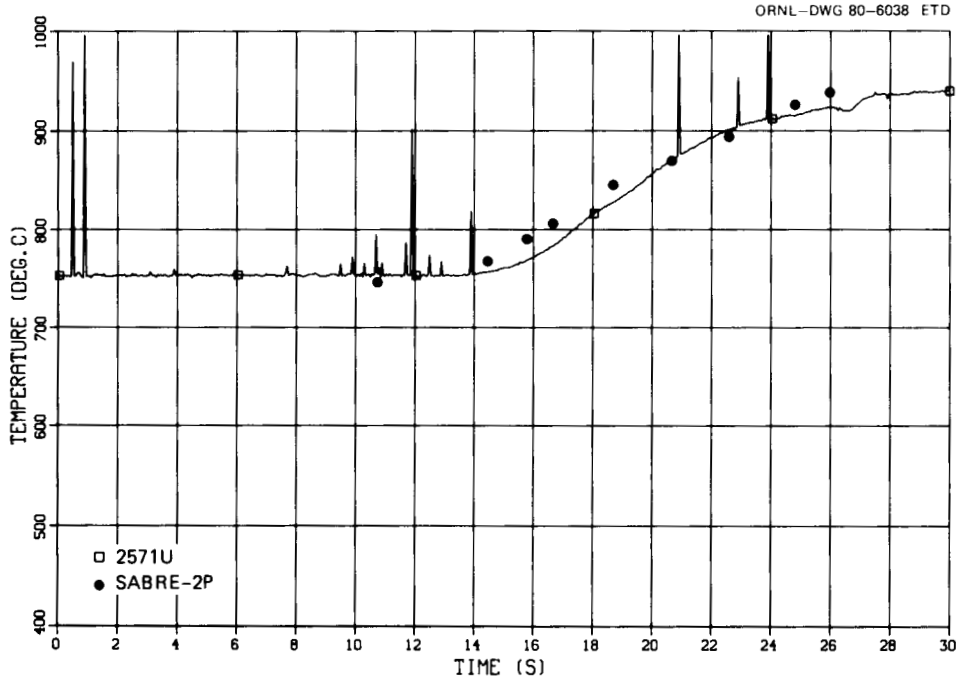

Fig. 1.19. Experimental vs SABRE-2P calculated transient temperatures for Test 205, Run 103, THORS Bundle 9 (thermocouple 2571U).

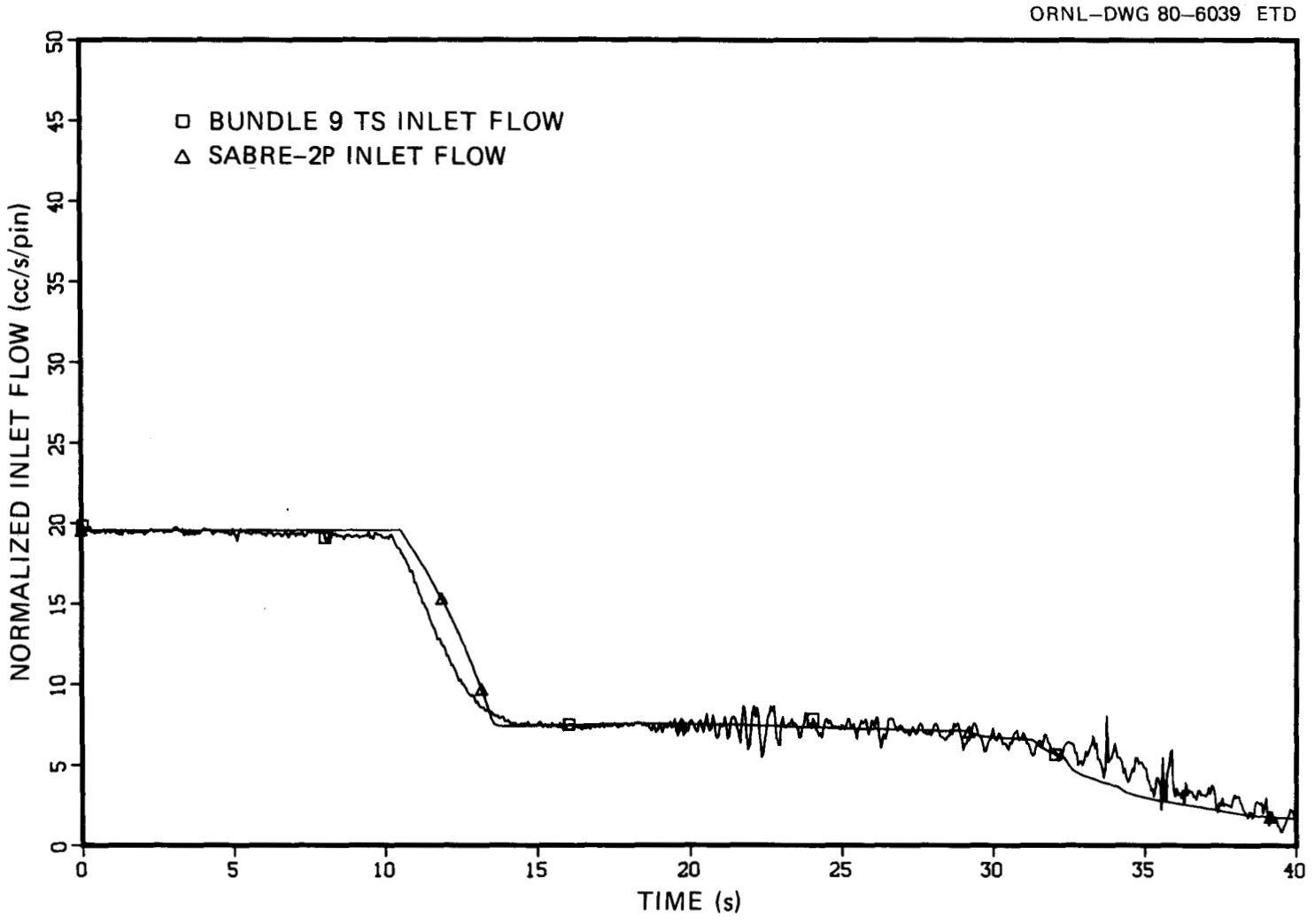

Fig. 1.20. Test section inlet flow: Test 204, Run 107, Bundle 9 — experimental data and SABRE-2P results.

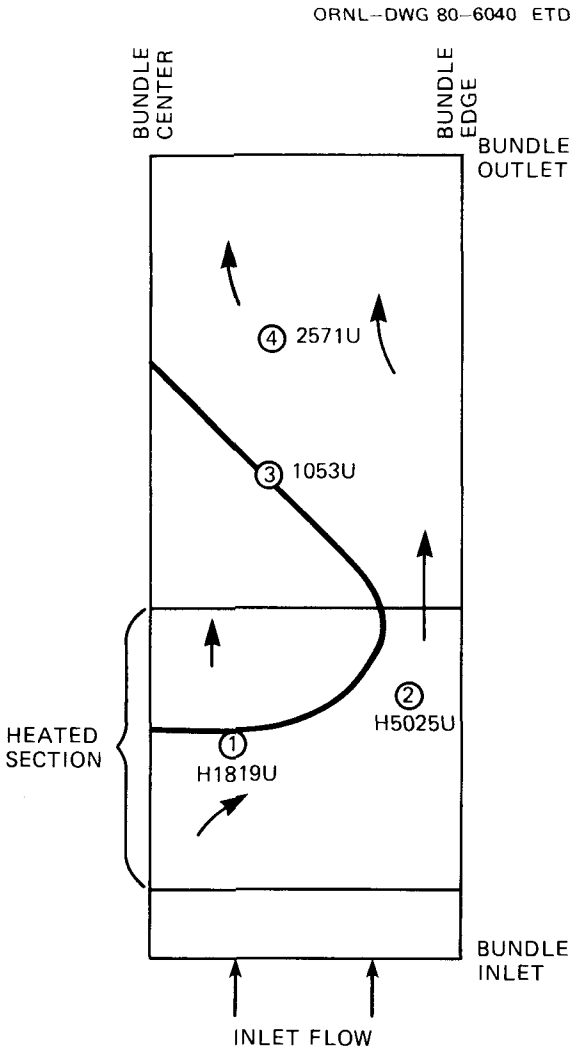

Fig. 1.21. Development of boiling region at 28 s in Test 204, Run 107, THORS Bundle 9.

shown in Figs. 1.22 through 1.25 at the same locations as in the previous section (Test 205 Run 103).

Test 303 Run 103 (Natural Convection Run)

This run was a pump-trip (natural convection) transient that was terminated manually because of low inlet flow at ~115 s into the transient, which included about 91 s of boiling. This run is particularly interesting because it shows that the occurrence of dryout in a natural convection run is strongly dependent on experimental conditions, in this

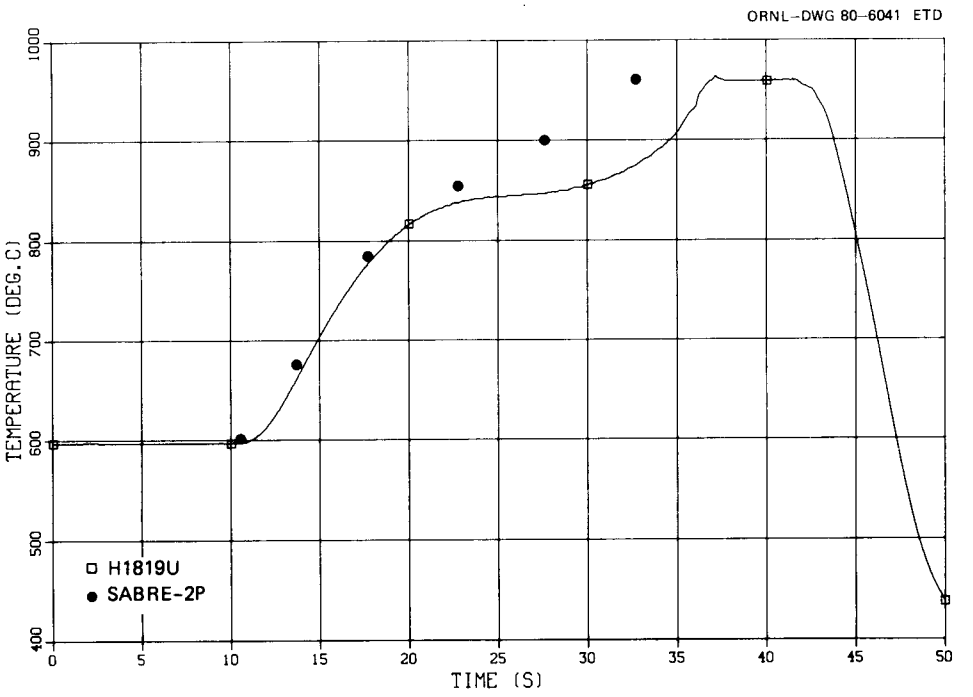

Fig. 1.22. Experimental vs SABRE-2P calculated transient temperatures for Test 204, Run 107, Bundle 9 (thermocouple H1819U).

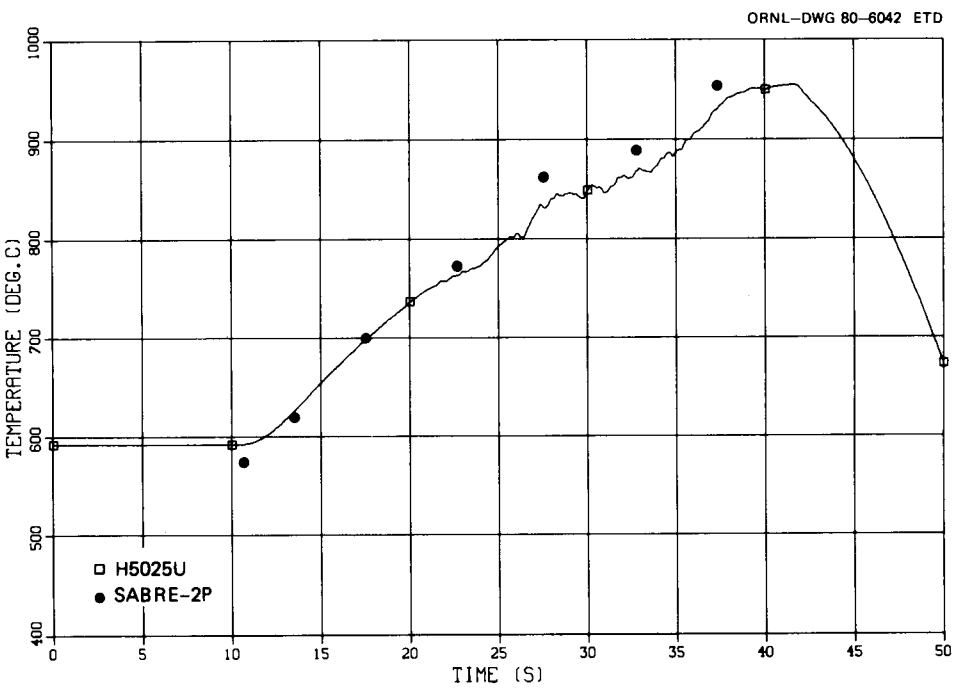

Fig. 1.23. Experimental vs SABRE-2P calculated transient temperatures for Test 204, Run 107, Bundle 9 (thermocouple H5025U).

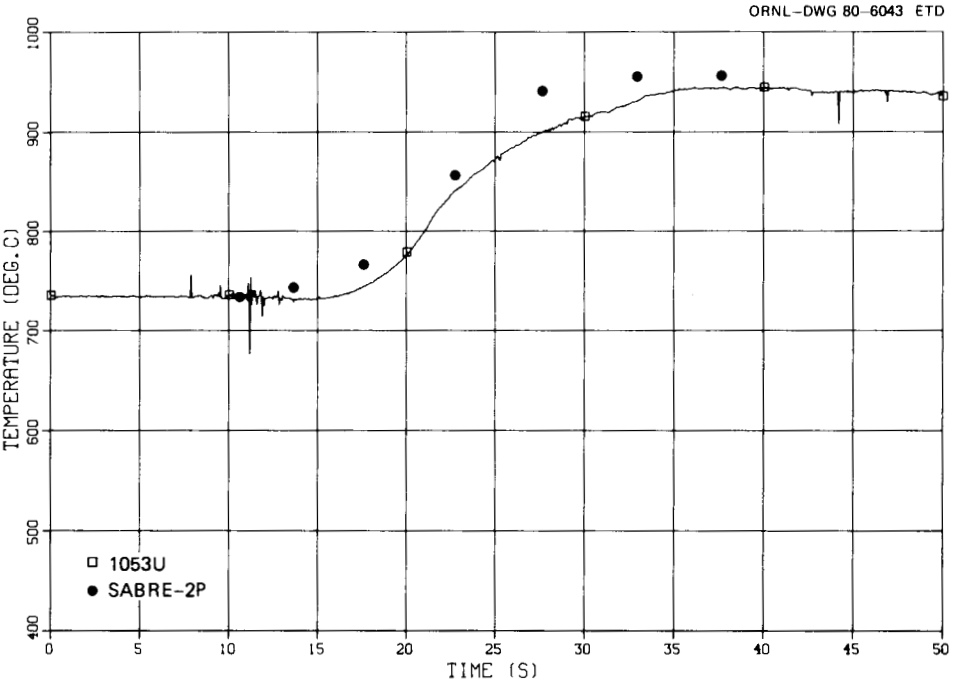

Fig. 1.24. Experimental vs SABRE-2P calculated transient temperatures for Test 204, Run 107, Bundle 9 (thermocouple 1053U).

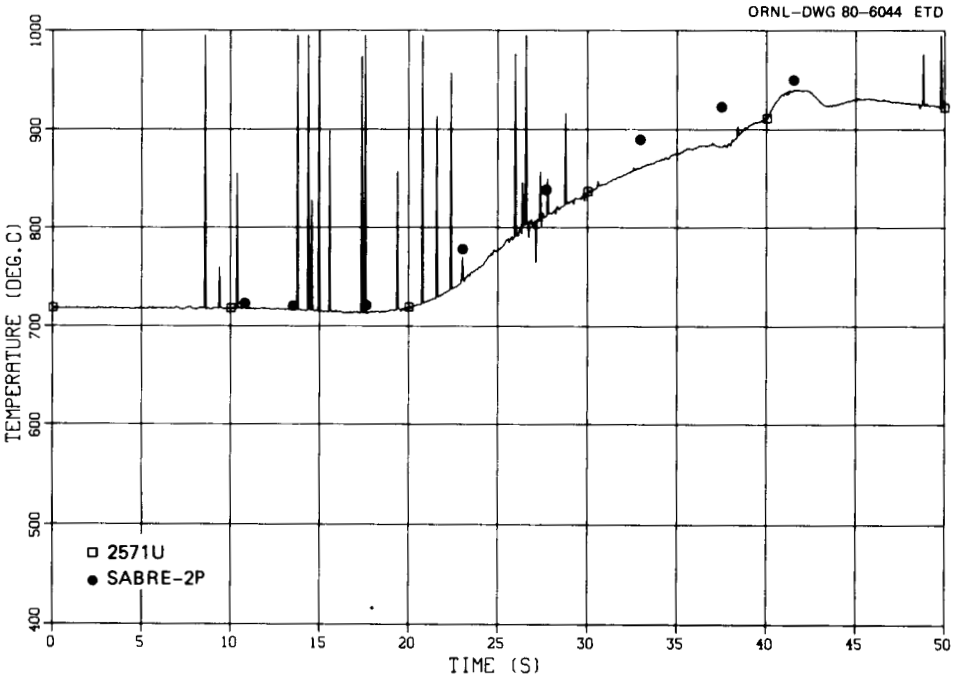

Fig. 1.25. Experimental vs SABRE-2P calculated transient temperatures for Test 204, Run 107, Bundle 9 (thermocouple 2571U).

case, the inlet valve resistance. The total time of the transient was less than 2 min, which also makes analysis with SABRE-2P not prohibitively expensive. Runs made with nominal inlet valve resistance took much more time to dryout or did not dryout at all in the 12-min test.

The inlet valve is represented in the SABRE model as a $k\rho V^2$ term on the first node to the bundle. The flow-pressure drop characteristics of the test section inlet valve were investigated in Test 300. Using a run from Test 300 with a corresponding flow to Test 303 Run 103, a value of $k = 25.4$ was calculated. The duct wall and outer structure is modeled by the radial heat conduction model B9SRC, which is very similar to the model used in Bundle 6A. Calculations with single-phase runs have shown that a value of 0.4 is suitable for the effective thermal conductivity of the sodium-permeated insulation.

To model the transient in SABRE-2P, a pressure-drop boundary condition was used. The test-section pressure drop was reduced from the steady-state value to the static-head value in 3 s, and then the pressure drop remained constant at this value. The inlet flow calculated by SABRE-2P is shown in Fig. 1.26 for the calculated value on inlet valve resistance $k = 25.4$, a slightly higher value of $k = 26$, and a slightly lower value of $k = 24$. With $k = 25.4$, the code accurately predicts the static (Ledinegg) excursion that leads to dryout. Boiling inception occurs at 22.3 s, and this is close to the prediction by SABRE of 21.9 s. For the higher inlet valve resistance ($k = 26$), the time to dryout is predicted to be reduced by about 20 s. For the lower inlet valve resistance ($k = 24$), the boiling region predicted by SABRE-2P has not developed significantly in 100 s, and boiling is quasi-steady-state. The predicted inlet flow is fairly constant for this case and shows only a slight reduction. For Test 303 Run 102, which had similar experimental conditions to this run except that the inlet valve was at the normal setting (with $k \cong 10.5$), boiling was quasi-steady-state at a fairly constant mean low-flow for 12 min, the complete time of the test.

In addition to dynamic oscillations that occur during boiling, the mean test section inlet flow (Fig. 1.26) shows a regular oscillatory variation about the predicted flow by SABRE-2P. This is probably a dynamic response of the test section with the rest of the loop. The test-

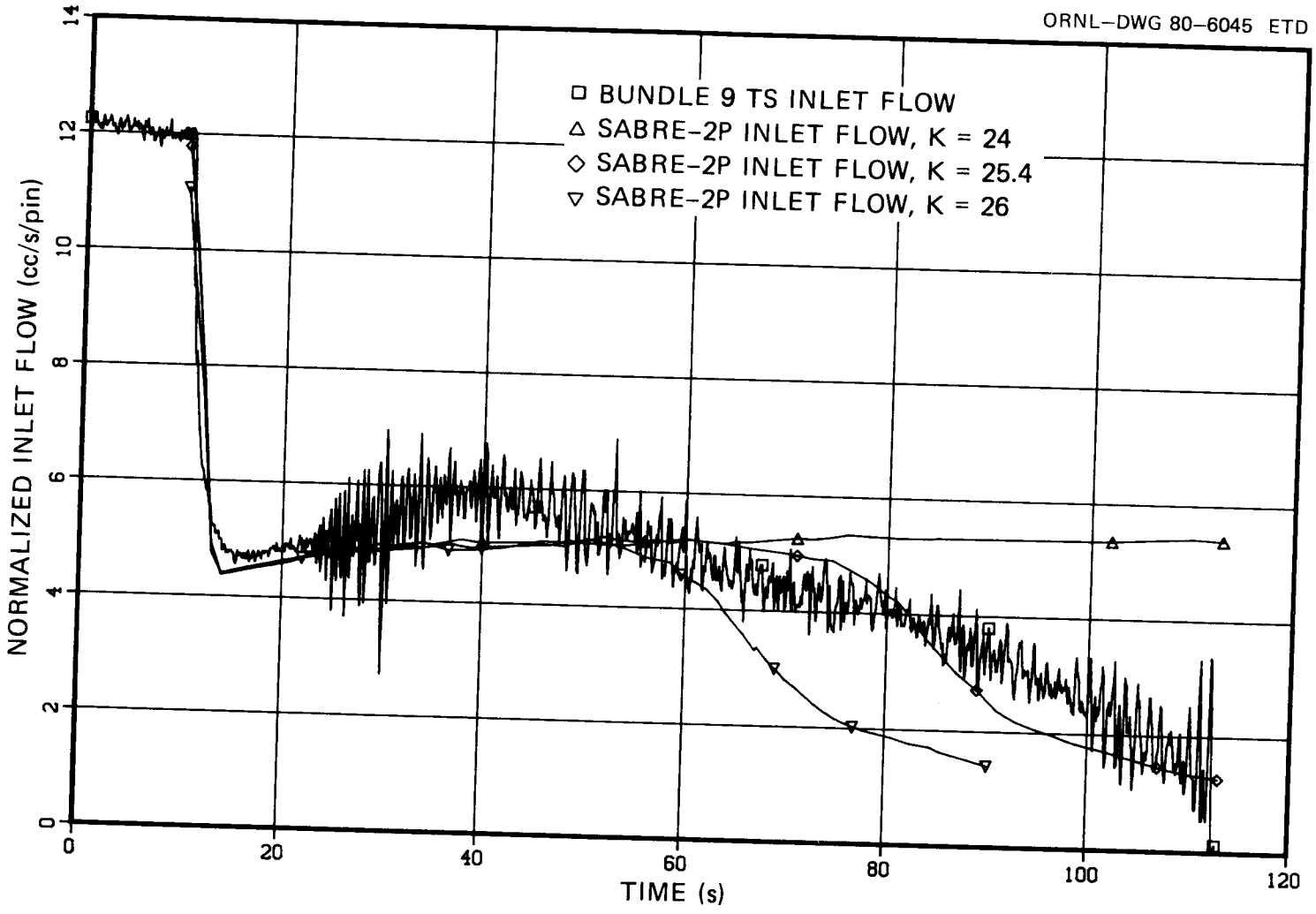

Fig. 1.26. Test section inlet flow: Test 303, Run 103, Bundle 9 — experimental data and SABRE-2P results with inlet valve resistance values K = 24, 25.4, and 26.

section outlet pressure shows a complementary oscillatory variation period of about 50 s with respect to the inlet pressure that remains fairly constant. This pressure variation occurs at the correct time to help explain the difference between experimental and predicted test-section inlet flow. Note that the Ledinegg flow excursion to dryout does not occur until the boiling region is across the entire cross section of the bundle. This is true of other Bundle 9 runs and Bundle 6A runs, as already discussed.

The accuracy with which SABRE-2P models the transient development of the two-phase region is shown in Figs. 1.27 through 1.33. Figure 1.27

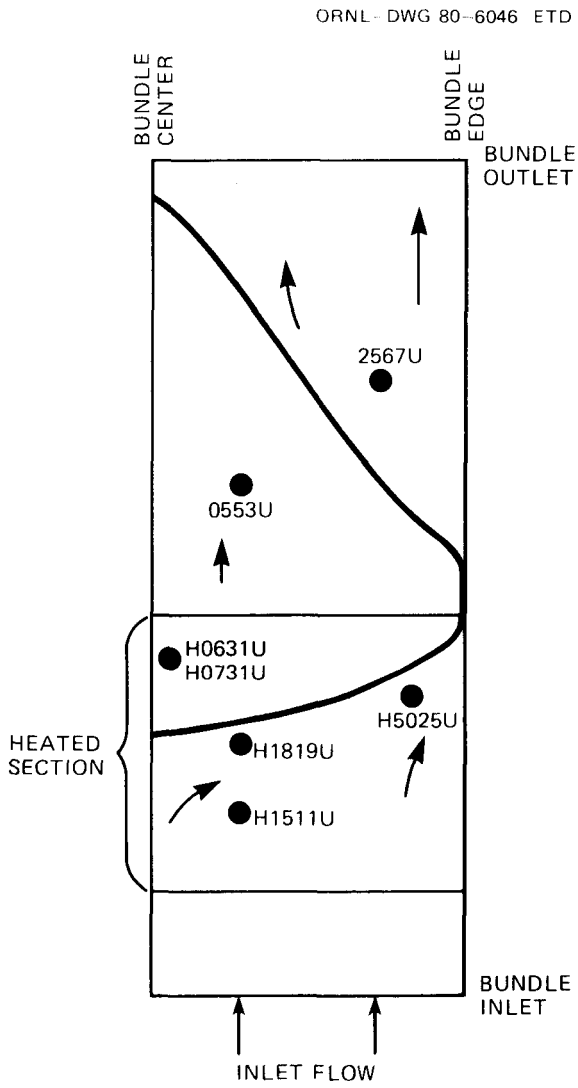

Fig. 1.27. Development of boiling region at 80 s in Test 303, Run 103, Bundle 9.

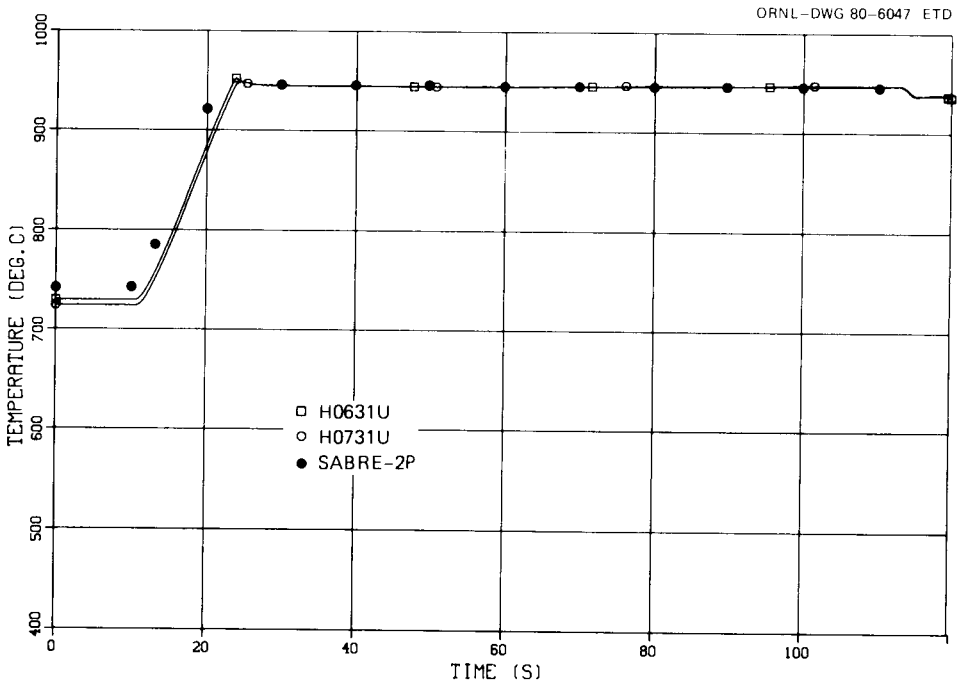

Fig. 1.28. Experimental vs SABRE-2P transient temperatures for Test 303, Run 103, THORS Bundle 9 (thermocouples H0631U and H0731U).

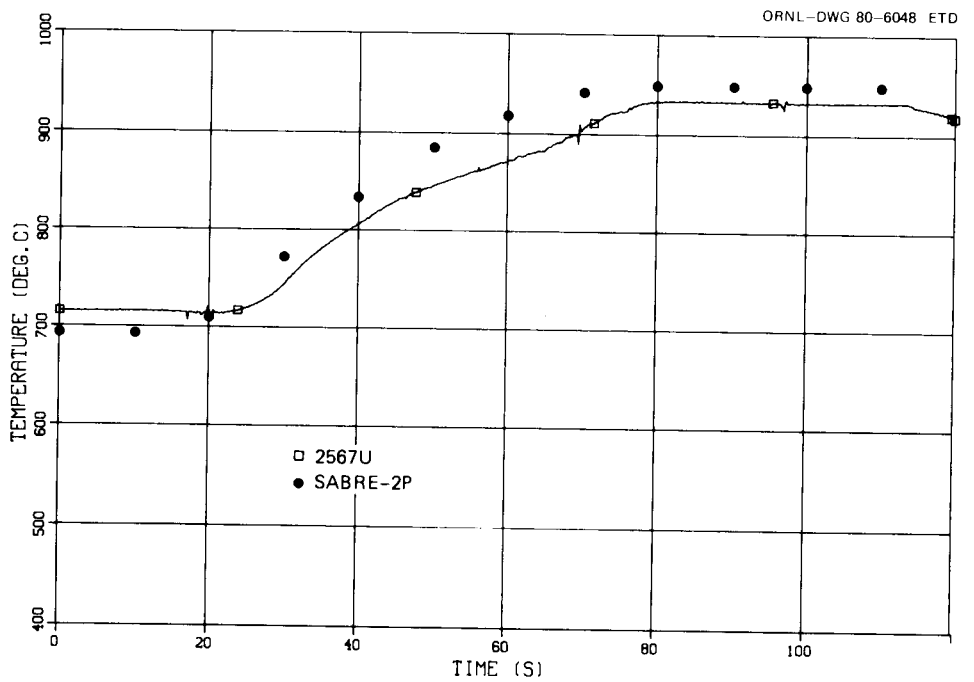

Fig. 1.29. Experimental vs SABRE-2P transient temperatures for Test 303, Run 103, THORS Bundle 9 (thermocouple 2567U).

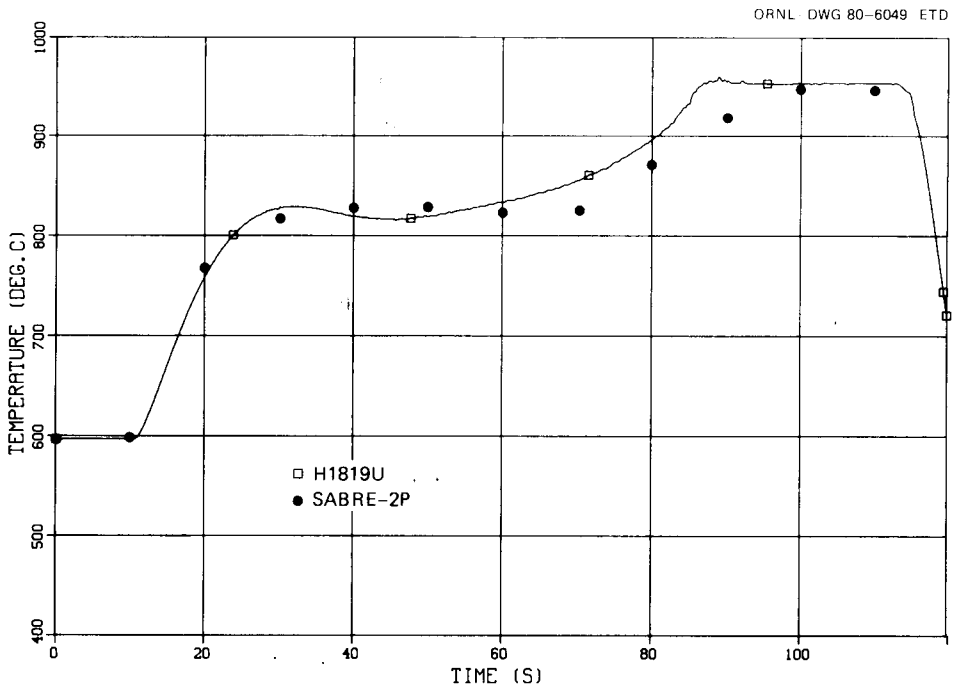

Fig. 1.30. Experimental vs SABRE-2P transient temperatures for Test 303, Run 103, THORS Bundle 9 (thermocouple H1819U).

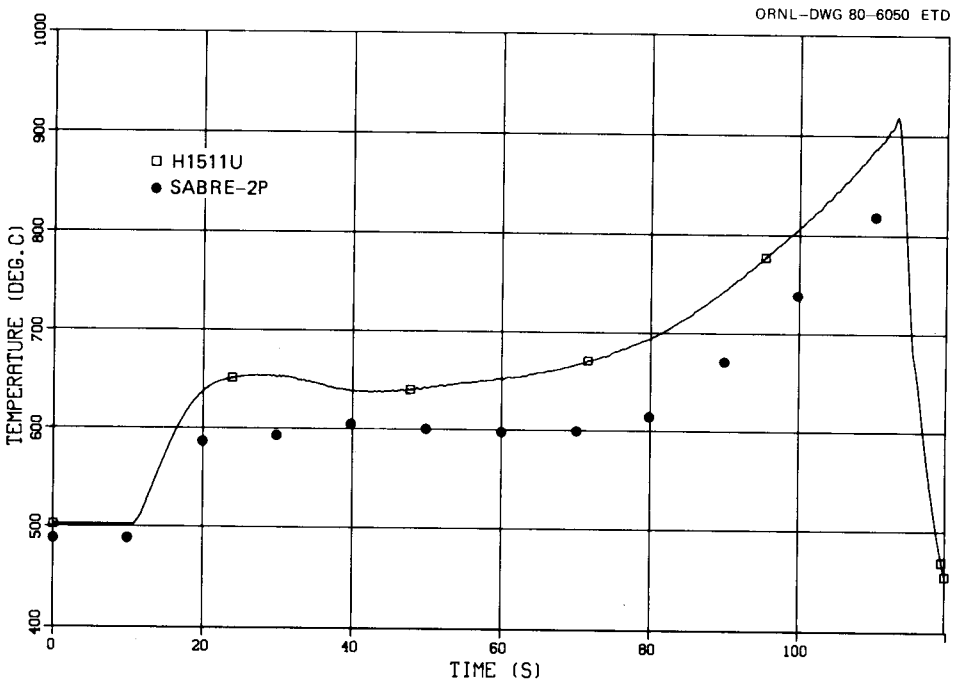

Fig. 1.31. Experimental vs SABRE-2P transient temperatures for Test 303, Run 103, THORS Bundle 9 (thermocouple H1511U).

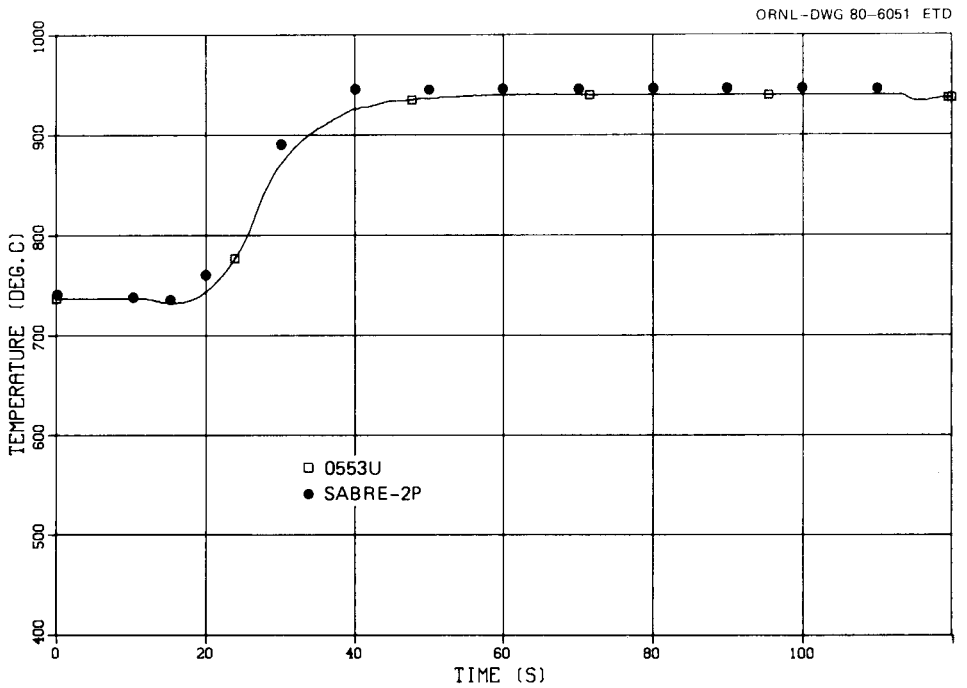

Fig. 1.32. Experimental vs SABRE-2P transient temperatures for Test 303, Run 103, THORS Bundle 9 (thermocouple 0553U).

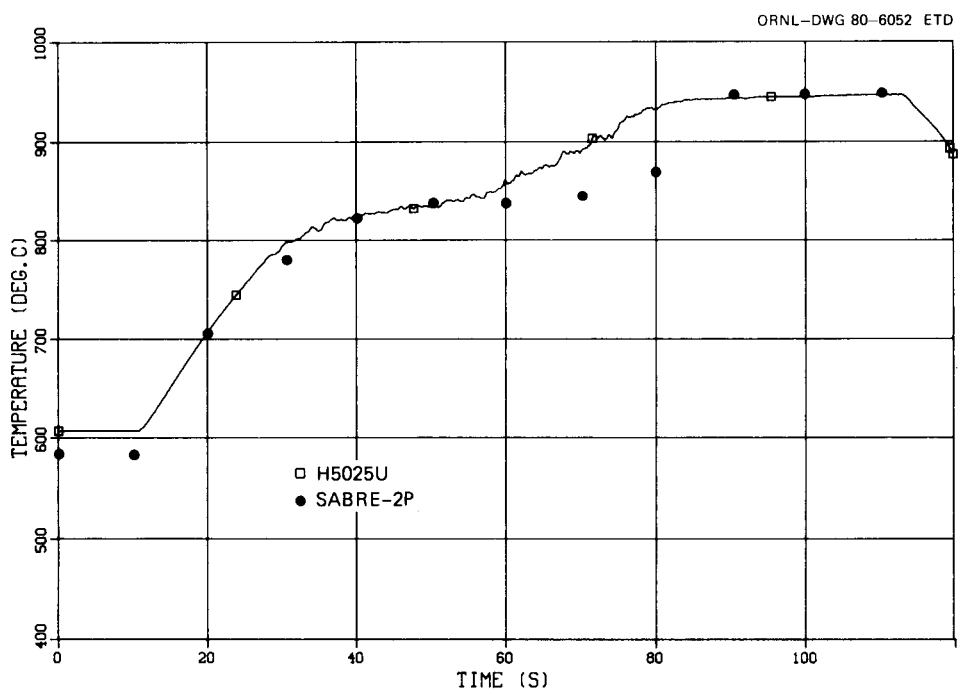

Fig. 1.33. Experimental vs SABRE-2P transient temperatures for Test 303, Run 103, THORS Bundle 9 (thermocouple H5025U).

shows schematically the development of the two-phase region at 80 s and the location of the wire-wrap thermocouples shown in the temperature plots. Data from these thermocouples are plotted along with SABRE-2P results (for k = 25.4) in Figs. 1.28 through 1.33; the lines are the experimental results, and the points are the SABRE-2P results. These thermocouples are widely spaced radially and axially, and the code shows well the development of the void growth. The axial extent of the boiling region at the center of the bundle is large, about 0.4 m upstream from the end of the heated section and ~1.4 m downstream, almost to the end of the SFGP. A comparison of the schematic diagram for Bundle 6A Test 73E Run 102A at 28 s (Fig. 1.27) with Fig. 1.3 shows similar development in Bundle 6A. At this time, the complete cross section of Bundle 6A is boiling. The boiling region in Bundle 6A is very similar to the region in the edge five subchannels of Bundle 9 that, in total, shows more extensive boiling because of the extra four inner subchannels. Notice the initial temperature decrease shown downstream of the heated section for the temperature plot of thermocouple 0553U on heater 5 at axial station 53. This decrease may well be caused by a flattening of the radial temperature profile at the reduced inlet flow. This temperature decrease is also predicted by SABRE-2P. The close comparison between code predictions and thermocouple data indicates that SABRE-2P is showing the correct 3-D growth of the boiling region.

Bundle 9 Test 305 Run 1041 (Extended Flow Reduction Run)

This run was a forced flow run in which the flow reduction ramp rate was reduced from previous runs. Instead of a rapid reduction to the low flow that usually takes about 3 s, the ramp was extended to take 60 s to investigate the boiling behavior with these different transient conditions. Because of operational problems in obtaining the exact required ramp rate, Test 305 Run 104 had a flow reduction that was more rapid than planned; this test was repeated as Test 305 Run 1041 in which the flow closely matched the flow reduction given by SABRE-2P in a pretest prediction (Fig. 1.34). In the SABRE-2P prediction, pressure drop was kept constant after the 60 s flow reduction. Boiling inception occurs before the

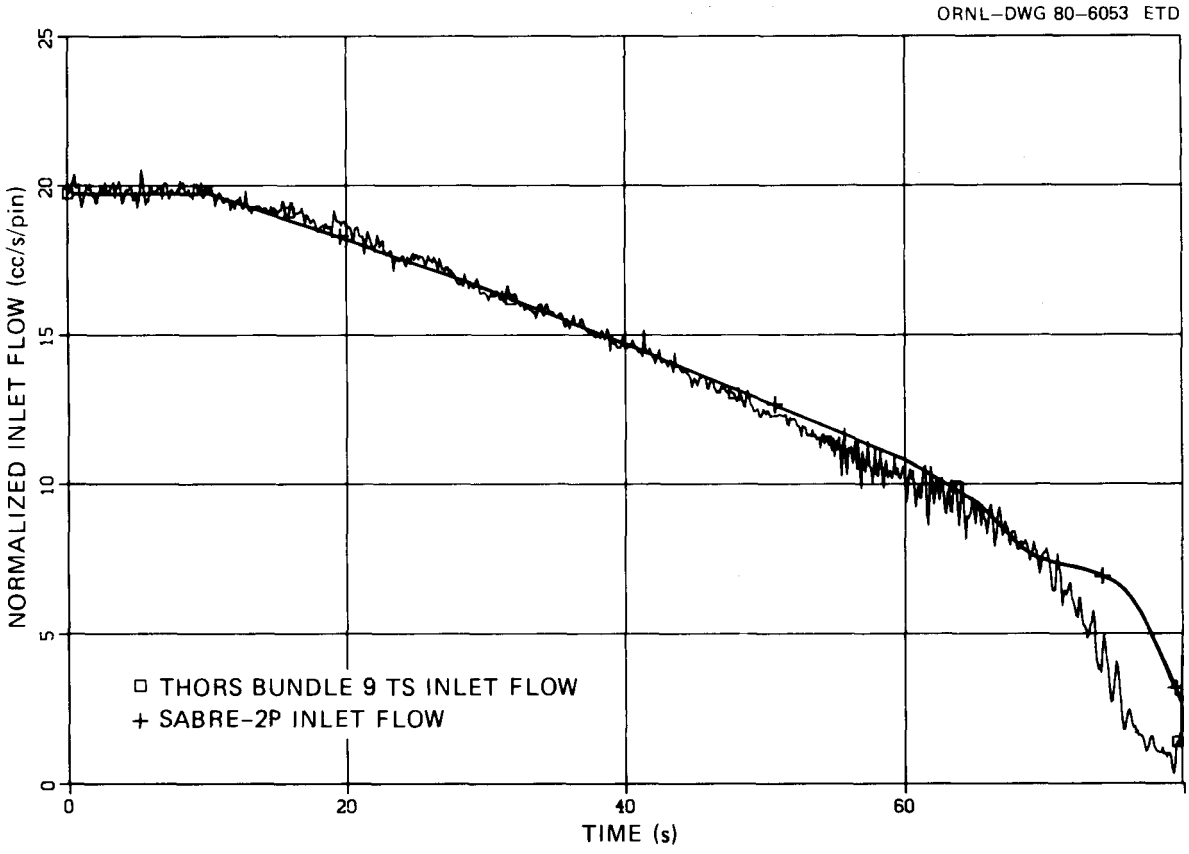

Fig. 1.34. Test section inlet flow: Test 305, Run 1041, Bundle 9 — experimental vs pretest computation SABRE-2P.

end of the flow reduction at about 53 s and is predicted at about 56 s near the end of the heated section in the bundle. After flow reduction, the Ledinegg flow excursion occurs almost immediately, whereas SABRE-2P shows a delay of about 3 s and then the flow drops rapidly. The test was terminated manually because of low inlet flow at 80 s.

Figures 1.35 through 1.40 show the experimental and predicted temperatures at six different locations in the bundle (see Fig. 1.27 for schematic of thermocouple positions). SABRE-2P predicts the growth of the boiling region reasonably well, which is encouraging for the modeling of a slow transient. In contrast, Test 305 Run 104 had similar experimental conditions but with a rapid flow reduction of about 3 s. In this run, the boiling region developed more rapidly, boiling started at 18 s, and dryout occured at 32 s.

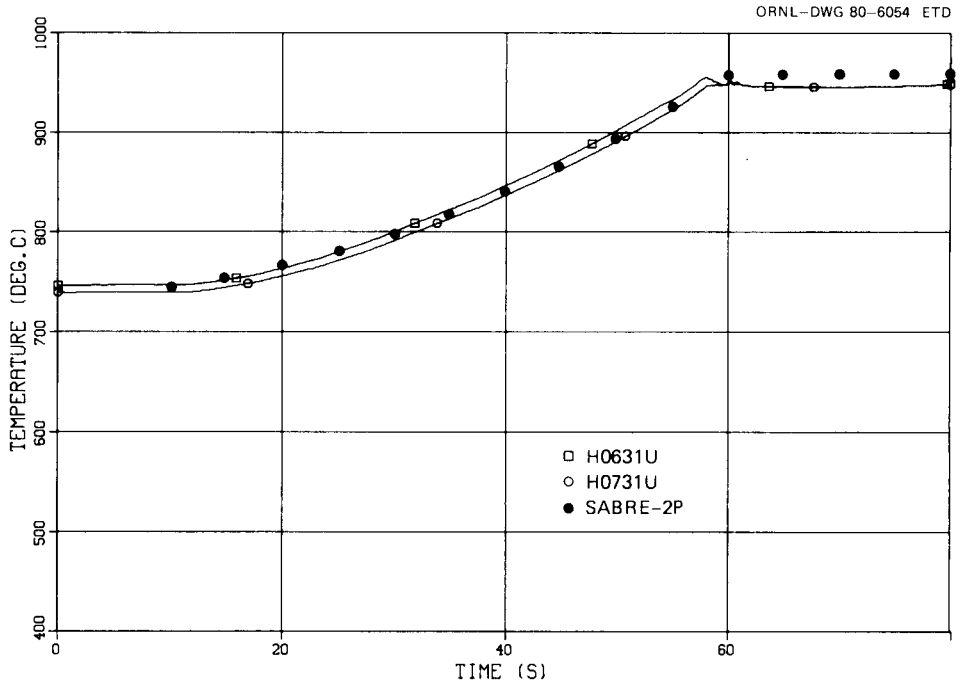

Fig. 1.35. Experimental vs SABRE-2P (pretest computation) transient temperatures for Test 305, Run 1041, THORS Bundle 9 (thermocouples H0631U and H0731U).

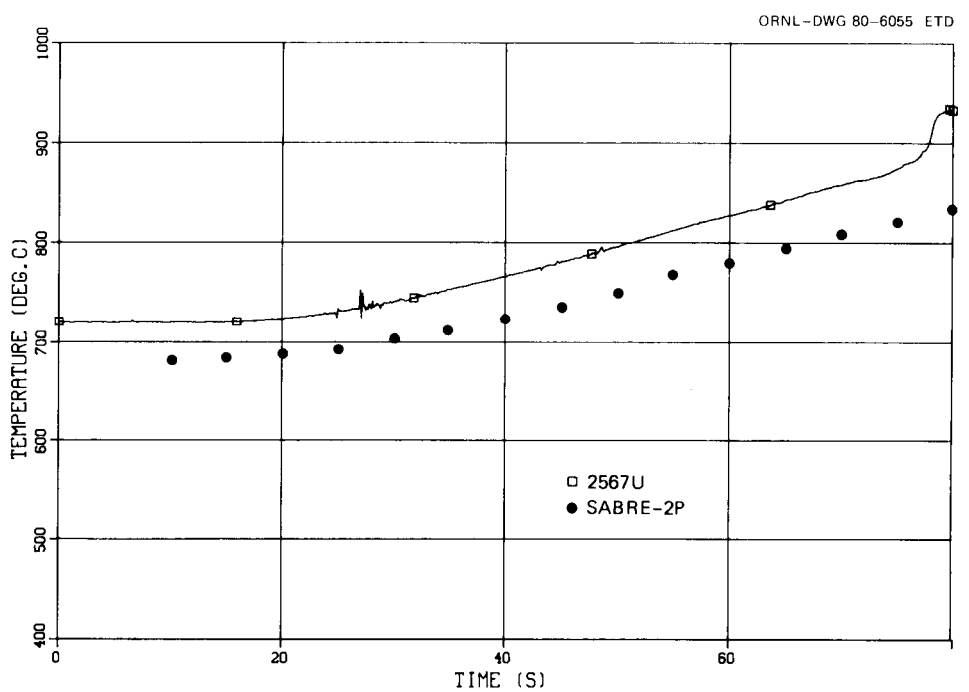

Fig. 1.36. Experimental vs SABRE-2P (pretest computation) transient temperatures for Test 305, Run 1041, THORS Bundle 9 (thermocouple 2567U).

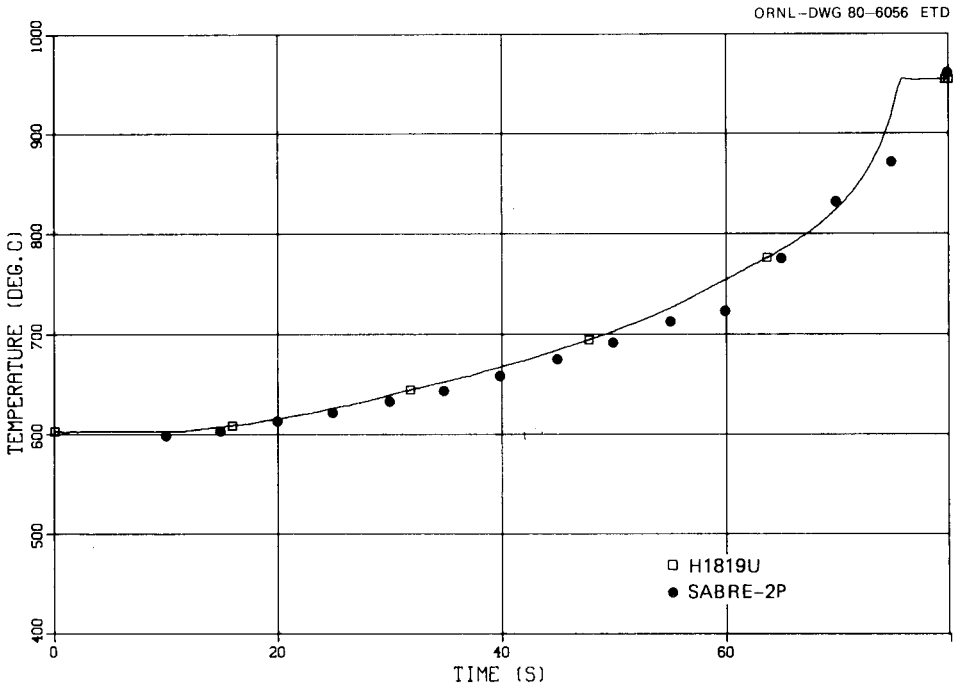

Fig. 1.37. Experimental vs SABRE-2P (pretest computation) transient temperatures for Test 305, Run 1041, THORS Bundle 9 (thermocouple H1819U).

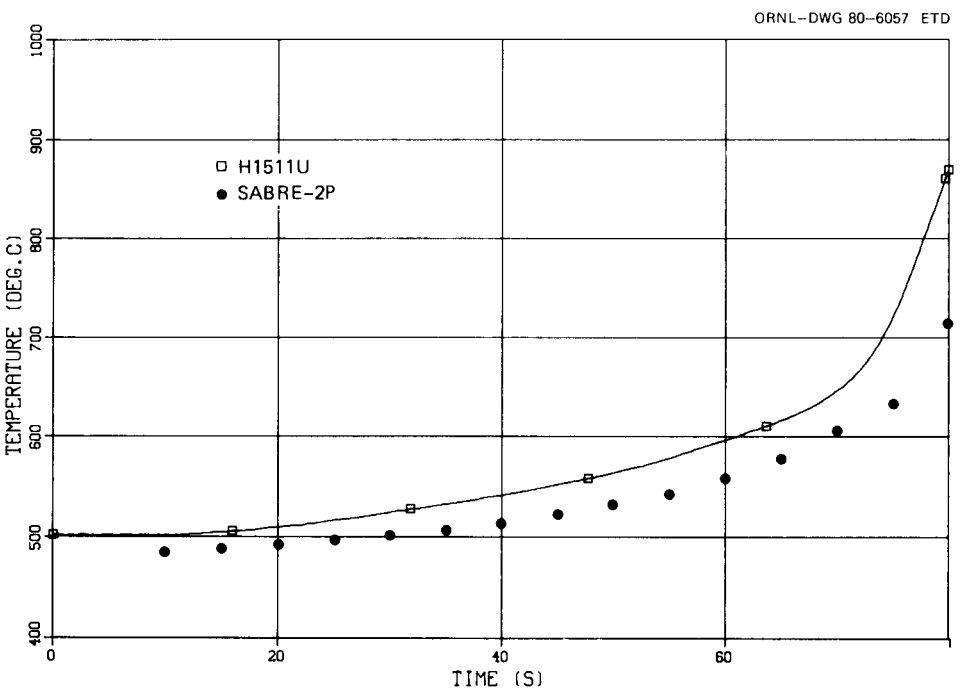

Fig. 1.38. Experimental vs SABRE-2P (pretest computation) transient temperatures for Test 305, Run 1041, THORS Bundle 9 (thermocouple H1511U).

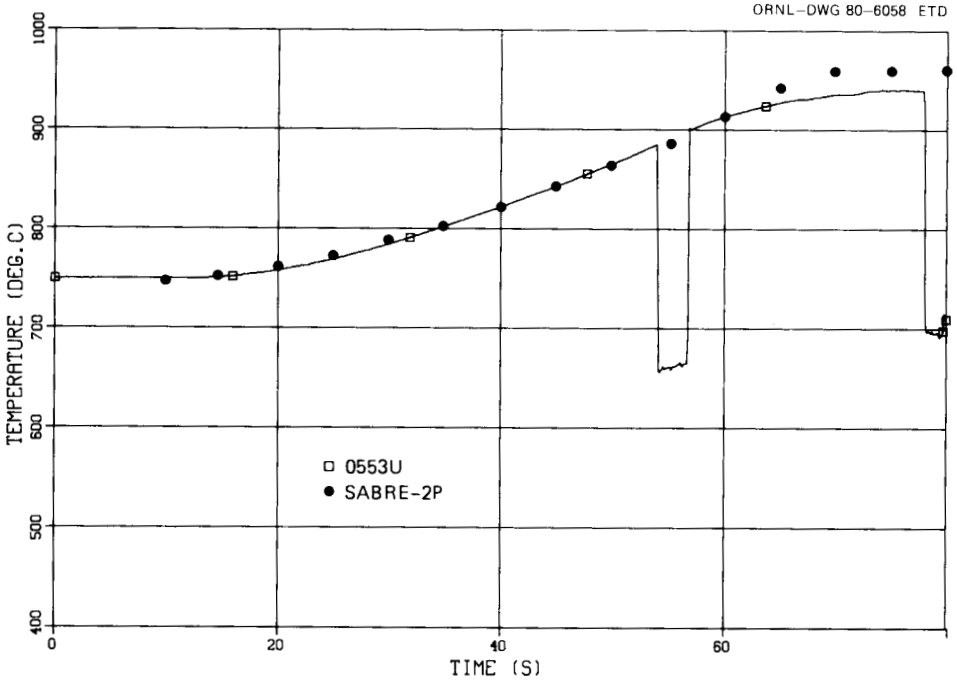

Fig. 1.39. Experimental vs SABRE-2P (pretest computation) transient temperatures for Test 305, Run 1041, THORS Bundle 9 (thermocouple 0553U).

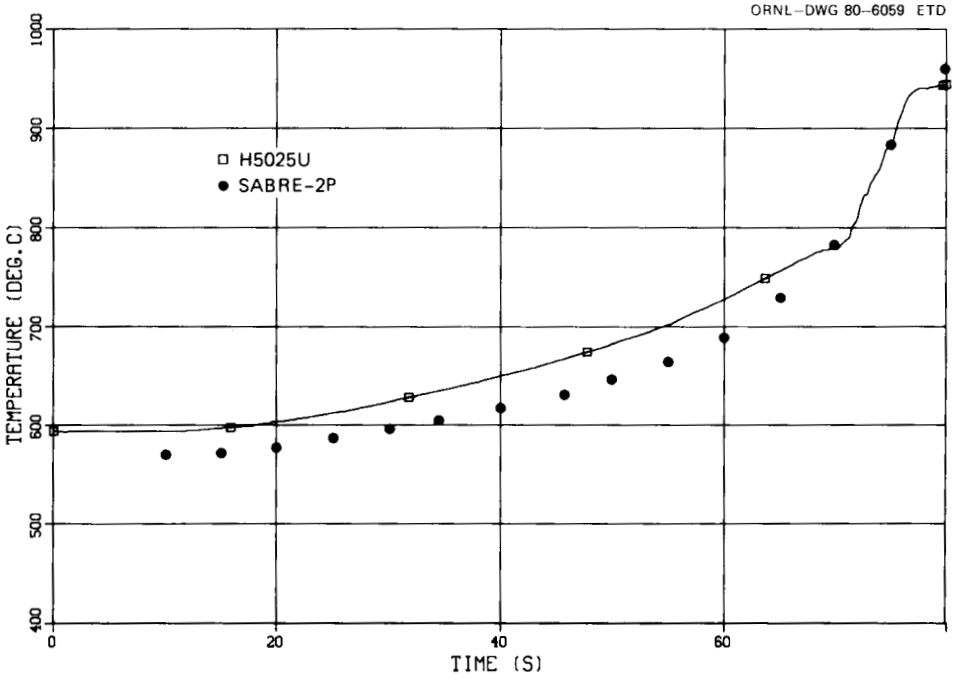

Fig. 1.40. Experimental vs SABRE-2P (pretest computation) transient temperatures for Test 305, Run 1041, THORS Bundle 9 (thermocouple H5025U).

This analysis is in its preliminary stage. However, the SABRE-2P code with a non-dynamic boiling model is able to predict the 3-D boiling characteristics in both Bundle 6A and Bundle 9 reasonably well. The runs discussed here have been made with different experimental conditions at both force flow and natural convection conditions. In all cases, the code supports the experimental observation that a Ledinegg flow excursion occurs once the boiling region has extended across the entire cross section of the bundle, and this culminates in dryout. Dryout has not occurred in any test before the onset of the Ledinegg flow excursion.

1.6 Reports and Major Correspondence Issued

R. H. Morris and W. R. Nelson et al., *Single-Phase Sodium Tests in a 61-Pin Full-Length Simulated LMFBR Fuel Assembly — Record of Phase 1 Experimental Data for THORS Bundle 9*, ORNL/TM-7315 (August 1980).

A. E. Levin, *Indirect Dryout Detection in a Simulated LMFBR Fuel Assembly*, ORNL/TM-7381 (September 1980).

1.7 Important Meetings

A THORS Program presentation was made to R. L. Rudman, Director of Planning, Electric Power Research Institute, and ORNL Engineering Technology Division Advisory Committee Member on July 18, 1980.

A THORS Program presentation was made to L. F. Lischer, Vice President, Engineering and Research (Retired), Commonwealth Edison Company, and ORNL Engineering Technology Division Advisory Committee Member on July 23, 1980.

E. U. Khan and J. M. Bates of Pacific Northwest Laboratories met with THORS Program Personnel to discuss experimental procedures and instrumentation on July 24–25, 1980.

A THORS Program presentation was made to T. J. Hanratty, Professor of Chemical Engineering, University of Illinois, and ORNL Engineering Technology Division Advisory Committee Member on August 11, 1980.

THORS Program staff members attended the ORNL Engineering Technology Division Annual Information Meeting, September 18–19, 1980. J. F. Dearing gave a formal presentation, "Sodium Boiling Behavior Under Simulated LMFBR Accident Conditions."

A. E. Levin attended an MIT Sodium Boiling Review Meeting, Cambridge, Massachusetts, September 24, 1980.

References

1. M. H. Fontana and J. L. Wantland, *ORNL Breeder Reactor Safety Quarterly Technical Progress Report for January–March 1980*, ORNL/TM-7348 (August 1980).

2. M. H. Fontana and J. L. Wantland, *LMFBR Safety and Core Systems Programs Progress Report for January–March 1977*, ORNL/TM-5940 (August 1977).

3. R. H. Morris and W. R. Nelson et al., *Single-Phase Sodium Tests in a 61-Pin Full-Length Simulated LMFBR Fuel Assembly – Record of Phase 1 Experimental Data for THORS Bundle 9*, ORNL/TM-7315 (August 1980).

4. M. R. Granziera and M. S. Kazimi, *A Two-Dimensional, Two-Fluid Model for Sodium Boiling in LMFBR Fuel Assemblies*, MIT-EL 80-011, Massachusetts Institute of Technology Energy Laboratory (May 1980).

5. M. H. Fontana and J. L. Wantland, *ORNL Breeder Reactor Safety Quarterly Technical Progress Report for October–December 1979*, ORNL/TM-7301 (June 1980).

6. J. D. MacDougall, *SABRE 2: A Computer Program for the Calculation of Transient Three-Dimensional Flows in Rod Clusters*, AEEW-R 1104, United Kingdom Atomic Energy Authority (1978).

7. M. H. Fontana and J. L. Wantland, *ORNL Breeder Reactor Safety Quarterly Technical Progress Report for July–September 1979*, ORNL/TM-7229 (May 1980).

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Part 2
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ORNL BREEDER REACTOR SAFETY — LOA 4 QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY—SEPTEMBER 1980

M. H. Fontana V. J. Tennery
E. S. Bomar

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Task 02 — ORNL LOA 4 Tasks: Environmental Assessment of Alternate FBR Fuels

OBJECTIVE

To determine the environmental consequences of substituting carbide fuels for reference oxide in fast breeder reactors.

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DEPARTMENT OF ENERGY

ORNL BREEDER REACTOR SAFETY — LOA 4 QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY—SEPTEMBER 1980

M. H. Fontana V. J. Tennery
E. S. Bomar

ABSTRACT

Preparation of a report is in progress describing the results of experiments to determine the oxidation characteristics of ThC-based fast breeder reactor fuels. Progress was also made in the measurement of ternary-fission yield of ^{3}H from ^{235}U, ^{238}U, and ^{232}Th in a fast-neutron flux. Dosimetry particles irradiated in the Experimental Breeder Reactor-II were characterized by microradiography and gamma spectrometry before measurement of their ^{3}H contents. Evaluation of dose-conversion factors for estimating health risks from high-specific-activity radionuclides was continued. Work was also continued on the impact of present and future releases of ^{3}H. A number of papers were presented or released for publication.

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2. TASK 02 — ORNL LOA 4 TASKS: ENVIRONMENTAL ASSESSMENT OF ALTERNATE FBR FUELS

V. J. Tennery H. R. Meyer
E. S. Bomar P. S. Rohwer
J. E. Till

A draft report has been written that describes the results of oxidation experiments with ThC-based fast breeder reactor (FBR) fuels.

Work directed at the confirmation of substantially enhanced ternary-fission yield of ^{3}H from ^{238}U and ^{232}Th in a fast-neutron flux has progressed through the assembly of an apparatus for recovery of the ^{3}H content of irradiated fuel particles and the early stages of making these measurements. This work was prompted by an environmental impact study[1] and experimental results reported by General Atomic Company (GA)[2] that show the importance of ^{3}H as a source of environmental impact and enhanced ternary-fission yield of ^{3}H in fuel irradiated in the EBR-II, respectively.

The ^{3}H content of High-Temperature Gas-Cooled Reactor (HTGR) fuel particles will be measured by breaking and oxidizing the particles in an

enclosed system purged with an oxidizing gas that contains added water. Oxidation of ^3H is ensured by passing the purge gas over heated copper oxide and then freezing tritiated water in a cold trap. The ^3H content of particles irradiated in the High Flux Isotope Reactor (HFIR) has been measured as a preliminary check of the analytical apparatus (Table 2.1). Recovery of less ^3H than indicated by the ^{137}Cs content of the particles is not considered a result of a deficiency in the analytical apparatus. Gainey has reported[3] the loss of 26 to 77% of the ^3H content of TRISO-coated UC_2 particles irradiated in the temperature range of 850 to 1250°C. These particles were exposed to temperatures of 1100 to 1300°C during irradiation. Additional confidence in the performance of the analytical apparatus resulted from analysis of a ^3H-labeled organic compound. Tritium recovery levels of 90 to 95% were obtained.

Analysis of the ^3H content of 28 HTGR-type fuel particles irradiated to various burn-up levels in the EBR-II will proceed. The particle kernels contain enriched ^{235}U, ^{238}U, or ^{232}Th. The number of fissions per particle on which to base ternary-fission yield will be determined from

Table 2.1. ^3H content of particles irradiated in the HFIR

Particles	^3H contents (moles)		Fraction
	Calculated from ^{137}Cs contents (10^{-11})	Analytical results (10^{-12})	Analytical/ Calculated
1	2.28	4.3	0.19
2	2.53	Nil[a]	
3	2.74	9.7	0.35
4	2.22	8.2	0.37
5	2.12	6.2	0.29
6	2.31	5.8	0.25

[a]This particle probably was not crushed during analysis.

the ^{137}Cs content of each particle, which in turn will be obtained from gamma-spectrometric measurements. The gamma scans have been completed, but calculation of ^{137}Cs contents awaits receipt of irradiation history and certain EBR-II neutronic data.

During this period, work also focused on evaluation of dose conversion factors currently employed in the estimation of health risks associated with release of high-specific-activity radionuclides in the event of Liquid-Metal Fast Breeder Reactor (LMFBR) accidents. Development and assessment work was continued with respect to tritium dosimetry and global modeling, with emphasis on a detailed study of the impact of current and predicted nuclear fuel cycle tritium releases. This latter study will be submitted for open literature publication in December 1980.

Several papers were presented or released for publication.

"Bounds for Global Tritium Population Doses," R. D. Gentry and C. C. Travis, was accepted for publication in *Health Physics*.

"Doses Due to Atmospheric Releases of Tritium and Carbon-14," by E. L. Etnier, will be published in a DOE Critical Review Series Report, "Models and Parameters for Environmental Radiological Assessments," C. W. Miller, Editor.

"Tritium — Potential Impacts of Nuclear Fuel Cycle Releases," by H. R. Meyer, J. E. Till, and E. L. Etnier, was presented by H. R. Meyer at the 25th Annual Health Physics Society (HPS) Meeting in Seattle, Washington, July 20–25, 1980.

"A Review of Methodologies for Calculating Dose from Environmental Releases of Tritium," by J. E. Till, E. L. Etnier, and H. R. Meyer, was accepted for publication in *Nuclear Safety*.

"Radiological Assessment of an Alternate Breeder Reactor Fuel Cycle," by H. R. Meyer, was accepted for presentation and publication in the *Proceedings of the Symposium on Intermediate Range Atmospheric Transport Processes and Technology Assessment, Gatlinburg, Tennessee, October 1–3, 1980*.

"Doses due to Environmental Releases of Tritium," by P. S. Rohwer, E. L. Etnier, and H. R. Meyer, *Recommendations Concerning Research and Model Evaluation Needs to Support Breeder Reactor Environmental Radiological Assessments*, ORNL/TM-7491, C. W. Miller, Editor (to be published).

The Third Annual HPS Summer School, "Assessment of Environmental Releases of Radioactivity," was held in Seattle, Wash., July 14 through 18, 1980. Technical content and faculty selection for the school were directed by J. E. Till and H. R. Meyer.

In an operational check of the mass spectrometer, originally scheduled for analysis of gaseous effluents from alternate fuel oxidation, the instrument failed to meet specifications. A field representative of the vendor CVC Products Incorporated made some corrective adjustments, but additional adjustments will have to be done to meet the purchase specifications.

A funding and time supplement was made to the contract with Colorado State University (CSU) for their work on "Gaseous and Particulate Source Terms for a Thorium Ore Pile," ORNL/Sub-7709.

References

1. J. E. Till et al., *Tritium — An Analysis of Key Environmental and Dosimetry Questions*, ORNL/TM-6990 (May 1980).

2. G. Buzzelli et al., "Tritium: Fast Fission Yields of ^{238}U and ^{232}Th," *Trans. Am. Nucl. Soc.* 24, 458 (1976).

3. B. W. Gainey, *A Review of Tritium Behavior in HTGR Systems*, GA-A 13461, Table 6, p. 21 (Apr. 30, 1976).

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Part 3
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ORNL BREEDER REACTOR SAFETY — LOA 4 QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY—SEPTEMBER 1980

M. H. Fontana R. O. Chester
C. W. Miller

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Task 03 — ORNL LOA 4 Tasks: Model Evaluation of Breeder Reactor Radioactivity Releases

OBJECTIVE

To develop and validate technology bases of analytical tools required to achieve LOA 4 goals for capabilities to predict individual and population exposures and health effects resulting from postulated radionuclide releases to the environment.

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ORNL BREEDER REACTOR SAFETY — LOA 4 QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY—SEPTEMBER 1980

M. H. Fontana R. O. Chester
C. W. Miller

ABSTRACT

Work is continuing on estimating the imprecision associated with calculations of dose to man and on comparing the AIRDOS-EPA and COMRADEX-IV computer codes.

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3. TASK 03 — ORNL LOA 4 TASKS: MODEL EVALUATION OF BREEDER REACTOR RADIOACTIVITY RELEASES

R. O. Chester C. W. Miller
J. T. Holdeman

3.1 Identification of Needs for Model Evaluations, Environmental Research, and Biomedical Research

The report, *Recommendations Concerning Research and Model Evaluation Needs to Support Breeder Reactor Environmental Radiological Assessments*, (ORNL/TM-7491) has completed technical and administrative review within the Health and Safety Research Division. Copies of this report have been forwarded to the Fast Reactor Safety Technology Management Center (FRSTMC) to complete milestone ORNL 4.2-1. The report will be sent to the Laboratory Records Department for final clearance and printing as soon as appropriate internal and external distribution lists can be completed.

The purpose of ORNL/TM-7491 is to make recommendations concerning needs for model evaluations, environmental research, and biomedical research to support breeder reactor environmental radiological assessments. This report has been prepared to assist the FRSTMC staff with their planning activities. Needs identified in this report are based on the results of the model evaluation project to date. However, other needs may be identified as new information is analyzed because this project is still under way.

The report contains summaries of model evaluation and research needs in the areas of atmospheric dispersion and deposition, terrestrial food chain transport, internal and external dosimetry, and environmental tritium dosimetry. These summaries reflect the work of those individuals who have contributed significantly to the model evaluation effort.

Numerous publications have been produced by the staff during this project's lifetime. Many publications include identification of relevant model evaluation and research needs. For the most part, this report represents a consolidation and compilation of the needs previously identified in these other publications. Consequently, no attempt has been made to justify rigorously each of the needs identified. Instead, the reader is referred to the cited literature for more information about any given topic.

Also, the report does not represent all of the model evaluation and research needs that have been suggested during the course of the project. Instead, the report attempts to present those needs that seem most relevant to the breeder reactor safety program.

3.2 Evaluation of Uncertainties Associated with Calculational Models

3.2.1 Imprecision in ^{131}I dose calculation

Estimations of the variability (or imprecision) of input parameters and implications of this variability on the predicted quantity have been recognized as very helpful tools for an evaluation of potential uncertainties in model predictions. A computer code has been developed to simulate the dose to man from air intake of radionuclides via multiple pathways by implementing the terrestrial food chain models described in the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.109, an atmospheric Gaussian dispersion model and a dosimetric model for ^{131}I. Some preliminary results of an imprecision analysis are summarized in Table 3.1 by giving the thyroid dose-to-air concentration ratio (D/X) for children of 0.5 to 2 years of age from a release of elemental ^{131}I. Evidently, the milk pathway dominates the total thyroid dose to children, subsequent to release of elemental iodine to the atmosphere. This conclusion is in

Table 3.1. Thyroid dose-to-air-concentration ratio (D/X) for children of 0.5 to 2 years of age from elemental $^{131}I_2$

Intake pathway	Median (X_m)	Sg^a	99th percentile (X_{99})
	mrem/year per pCi/m^3		
Milk	3,500	3.3	43,000
Meat	90	3.5	1,600
Nonleafy vegetables	32	2.7	300
Leafy vegetables	32	3.1	400
All pathways	3,800	3.2	44,000

[a]Geometric standard deviation.

agreement with various deterministic estimates. Imprecision in each pathway of a similar magnitude is also evident. Generally, the 99th percentile X_{99} of the distribution is an order of magnitude larger than the median value X_m.

3.2.2 Imprecision in ^{137}Cs dose calculation

An attempt has also been made to quantify the variability in human biological parameters determining dose to man from ingestion of a unit activity of soluble ^{137}Cs and the resulting imprecision in the predicted total-body dose commitment. The analysis is based on an extensive literature review along with the application of rigorous statistical methods to determine parameter variability, correlations between parameters, and predictive imprecision.

Imprecision in the quantities making up the dose equivalent conversion factor, and consequently in the total-body dose from cesium, is less than that observed in other biological parameters and dose or body burdens of other radioactive and nonradioactive pollutants. For example, the geometric standard deviation for parameters determining the thyroid dose from ^{131}I is on the order of 1.5 to 2.0, and the predicted geometric standard deviation in the thyroid dose is ~1.8 to 2.1 for several age

groups. However, variation in the adult parameters for cesium is on the order of 1.2 to 1.5, and the total-body dose to age groups of children/adolescents and adults exhibits a standard deviation of ~1.3 to 1.4. Consequently, the X_{99} of the predicted distribution of the dose from ^{137}Cs is ~2.1 times the mean $\overline{X}$ for both children/adolescents and adults. Corresponding ratios for thyroid dose from ^{131}I were predicted to be ~3 to 4. A summary of the estimated statistical properties of the total-body dose equivalent per unit activity intake of ingested ^{137}Cs is given in Table 3.2.

3.2.3 Symposium on atmospheric transport

Preparations are complete for the *Symposium on Intermediate Range Atmospheric Transport Processes and Technology Assessment* to be held in Gatlinburg, Tennessee, October 1 through 3, 1980. The Program Committee has examined the submitted abstracts, and the authors of accepted papers have been notified. At this time, 49 papers from the United States, Canada, West Germany, and the United Kingdom are expected at the meeting. A printed preliminary program and preregistration materials were distributed early in August.

3.3 Comprehensive Testing of Selected Models

Comparative work is continuing on the COMRADEX-IV and AIRDOS-EPA computer codes. A sample problem has been run considering four different radionuclides that might be released in a breeder reactor accident:

Radionuclide	Total release, Bq*
^{85m}Kr	1.7×10^{20}
^{131}I	7.8×10^{20}
^{134}Cs	1.1×10^{19}
^{238}Pu	1.0×10^{19}

*Bq = becquerel.

Table 3.2. Estimates of statistical properties of total-body dose equivalent per unit activity intake of ingested ^{137}Cs

[Estimates based on the assumption of a bivariate normal distribution between body mass and equivalent biological half-time (T_b)][a]

Age group (both sexes)	Unit	Mode X_p	Median X_m	Mean $\overline{X}$	Standard deviation	95th percentile X_{95}	99th percentile X_{99}	NRC value
Children and adolescents (0.1 to 18 years)	Sv/MBq	~0.009	0.009	0.009	0.004	0.015	0.018	0.012[b]
	rem/μCi	~0.032	0.032	0.034	0.014	0.056	0.074	0.046[b]
		(0.50)[c]	(0.50)	(0.56)				(0.88)
Adults (>18 years)	Sv/MBq	~0.010	0.010	0.010	0.003	0.016	0.021	0.014
	rem/μCi	~0.037	0.037	0.038	0.012	0.058	0.077	0.053
		(0.50)	(0.50)	(0.55)				(0.90)

[a]Table values are based on a unit of the predicted distribution in the total-body dose equivalent.

[b]Proposed value for individuals of age 1 to 11 years.

[c]Values in parentheses represent the cumulative probability $P(X \leq X_p)$.

Inhalation doses and external gamma doses were calculated at four downwind locations:

Location	Downwind distance, m
1	6.758×10^2
2	1.609×10^3
3	3.219×10^3
4	4.023×10^3

An instantaneous release directly to the environment was assumed for all four radionuclides.

Both COMRADEX-IV and AIRDOS-II normally base their external doses D_{ext} (from immersion in air containing gamma-emitting radionuclides) on the semi-infinite cloud model:

$$D_{ext} = 0.25\ (E_\gamma)(\chi)\ , \tag{1}$$

where

E_γ = average gamma energy released per disintegration,
χ = air concentration at the point of interest.

In COMRADEX-IV the dose in sieverts (Sv) is calculated directly using Eq. (1) and multiplying the product by a "local shield factor." The assumption is made that $E_\gamma \sim \Gamma/6$, where Γ is the radionuclide gamma dose rate constant C-m^2/Kg-Bq-s. This factor depends on the number of photons emitted per disintegration and their energy. A tabulated set of Γ values commonly used in COMRADEX-IV (derived from a number of sources) is in the associated code documentation. Because the Γ values are input parameters for the code, however, a user may choose another set of values if desired.

In AIRDOS-EPA, D_{ext} is not calculated directly from Eq. (1). Instead, the air concentration is multiplied by an external dose conversion factor (Sv/year per Bq/cm^3) that is an input parameter to the code. Values normally used are calculated using the semi-infinite cloud assumption, and they have been tabulated in ORNL/NUREG/TM-283, *Dose-Rate Conversion Factors for External Exposure to Photon and Electron Radiation from Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities*. However, other values could be input if desired.

Resulting total-body external doses calculated for this sample problem by both computer codes are shown in Table 3.3. All doses calculated with AIRDOS-EPA are lower than the corresponding doses calculated by COMRADEX-IV. While COMRADEX-IV did not calculate an external dose from ^{238}Pu, the ^{238}Pu external dose in this example is negligible compared to the total-body doses estimated for the other three radionuclides considered.

Dose conversion factors used in the AIRDOS-EPA computations are the organ dose-rate factors for total body. The report also contains photon dose-rate factors for tissue equivalent material at the body surface of an exposed individual. Organ dose-rate factors are based on these surface

Table 3.3. External gamma air-immersion doses calculated by COMRADEX-IV and AIRDOS-EPA for a hypothetical acute release of four radionuclides

Radionuclide	Location	Dose (Sv)	
		COMRADEX-IV	AIRDOS-EPA
^{85m}Kr	1	3.0E+03[a]	2.3E+03
	2	1.2E+03	9.7E+02
	3	1.0E+03	8.1E+02
	4	7.0E+02	5.5E+02
^{131}I	1	3.9E+04	2.6E+04
	2	1.7E+04	1.1E+04
	3	1.5E+04	1.0E+04
	4	1.0E+04	7.1E+03
^{134}Cs	1	2.2E+03	1.5E+03
	2	9.7E+02	6.5E+02
	3	8.6E+02	5.8E+02
	4	6.0E+02	4.1E+02
^{238}Pu	1	*b*	5.0E-02
	2	*b*	2.2E-02
	3	*b*	1.9E-02
	4	*b*	1.4E-02

[a] 3.0E+03 = 3.0×10^3.

[b] Indicates that no dose was calculated for this combination of conditions.

dose-rate factors and on estimates of absorbed dose rates in the body organs. An examination of the COMRADEX-IV documentation indicates that the COMRADEX-IV procedure for calculating external dose due to submersion in air is more likely to result in body surface dose rate than an average organ dose rate. This observation is supported by the fact that use of body surface dose-rate factors in AIRDOS-EPA results in calculation of external doses for radionuclides in the sample problem that are essentially identical to those calculated by COMRADEX-IV. However, organ dose-rate factors are considered to be more appropriate for assessment purposes than body surface dose-rate factors.

These results, as well as the results of the inhalation dose comparisons, are being documented in report ORNL/TM-6495, *A Comparison of the COMRADEX-IV and AIRDOS-EPA Methodologies for Estimating the Radiation Dose to Man from Radionuclide Releases to the Atmosphere*. This report is currently beginning technical and administrative review within the Health and Safety Research Division.

3.4 Meetings

3.4.1 Vienna, Austria

F. Owen Hoffman was in Vienna, Austria, June 23 through 27, 1980, to participate as a scientific consultant at the request and expense of the International Atomic Energy Agency (IAEA) and to serve as chairman of the food-chain working group in the Technical Committee Meeting on the "Assessment and Verification of Environmental Pathways." Twenty scientists representing eleven countries and three international organizations were convened to review a draft document entitled, "Generic Models and Parameters for Assessing the Environmental Transfer of Radionuclides in Predicting Exposures to Critical Groups from Routine Releases." This document presents a calculational approach that is simple to apply and is intended for preoperational evaluation of planned discharges of radioactive materials from nuclear facilities. The participants approved the principal concepts, models, and choices of default parameter values in the draft document. In the document, the models and parameter values were selected

to ensure a low probability of the dose limit being exceeded by any exposed individual when the dose estimated with the models remained at least one order of magnitude below the dose limit. When dose calculations exceeded this level, the participants recommended careful evaluation of the model and parameter values to examine their relevance to prevailing site-specific conditions. If such evaluations identify discrepancies between conditions represented by the models and parameter values and conditions prevailing at a given site, then site-specific investigations are recommended to improve the relevancy of dose predictions through alterations of model structure and parameter values and, when possible, through model validation experiments. The draft document was revised in both plenary and specialized working sessions and currently awaits technical editing before preparation of a final draft. Publication by the IAEA is expected before January 1981. The accomplished review and acceptance of the draft document by the participants represents an achievement that will complement model evaluation and development activities within the Department of Energy (DOE), the Environmental Protection Agency (EPA), the NRC, the National Council on Radiological Protection and Measurements, and the International Commission on Radiological Protection (ICRP). Publications from the model evaluation project provided a major source of supporting documentation for development of generic models and selection of parameter values.

3.4.2 Seattle, Washington

G. Schwartz attended the annual meeting of the HPS in Seattle, Washington, July 20 through 24, 1980. He presented a paper coauthored with F. O. Hoffman entitled, "Uncertainties in Dose Predictions: An Application of a Monte Carlo Simulation Technique for an Imprecision Analysis."

3.5 Publications

C. W. Miller, C. A. Little, and S. J. Cotter, "Comparison of Observed and Predicted Normalized Air Concentrations for 56-m Releases of Fluorescein Particles," pp. 634–41 in *Volume of Conference Papers: Second Joint Conference on Applications of Air Pollution Meteorology and Second Conference on Industrial Meteorology, New Orleans, Louisiana, Mar. 24–28, 1980*; American Meteorological Society, Boston, 1980.

C. W. Miller, S. J. Cotter, R. E. Moore, and C. A. Little, "Estimates of Dose to the Population Within Fifty Miles Due to Noble Gas Releases from the Three Mile Island Incident," pp. 1336–43 in *Proceedings of the American Nuclear Society/European Nuclear Society Topical Meeting Nuclear Reactor Safety, Knoxville, Tennessee, Apr. 6–9, 1980*, CONF-800403/V-II (1980).

K. Brüsserman et al., *Proceedings of a Workshop on Accuracy in Dose Calculations for Radionuclides Released to the Environment, Aachen, Federal Republic of Germany, Sept. 10–14, 1979*, June 1980.

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ORNL BREEDER REACTOR SAFETY — LOA SUPPORT QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY–SEPTEMBER 1980

M. H. Fontana W. B. Cottrell
J. R. Buchanan

AF 15 10 25 0

Task 04 — ORNL LOA Support and Integration:
Nuclear Safety Information Center

OBJECTIVE

To disseminate safety information on the breeder and other reactor programs to the nuclear community through reports and the bimonthly publication, *Nuclear Safety*, and to individuals through Selective Dissemination of Information, searches, and consultation. (NOTE: The Department of Energy pays for only 15% of the work described.)

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ORNL BREEDER REACTOR SAFETY — LOA SUPPORT QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY—SEPTEMBER 1980

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4. TASK 04 — ORNL LOA SUPPORT AND INTEGRATION: NUCLEAR SAFETY INFORMATION CENTER

W. B. Cottrell J. R. Buchanan

In 1963, the Nuclear Safety Information Center (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. The Center aids those concerned with analysis, design, and operation of nuclear facilities by providing and/or identifying relevant information on nuclear safety problems. This may be done directly through special searches, Selective Dissemination of Information (SDI), and consultation, or indirectly through the publication of reports, bibliographies, and the technical progress review (*Nuclear Safety*). Jointly sponsored by NRC and DOE, the Center participates in various special studies and evaluations.

Capabilities of the Center's technical staff are augmented by a computer file that contains 100-word abstracts and complete bibliographic data on over 150,000 nuclear safety documents and 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. The NSIC computerized capability has been extended by 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 to August 31, 1980. Since mid-1975, the Center has operated under

the DOE Information Center Pricing Policy, 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 Center activities, including the preparation of reports and the journal.

4.1 Routine Services

During the months of July, August, and September 1980, the staff of NSIC (1) entered 2375 documents, (2) responded to 202 inquiries (of which 136 involved technical staff), (3) made 57 computer searches (of which 8 were for paying customers), (4) provided SDI to 396 subscribers (including 48 paying subscribers), (5) received 23 visitors, (6) attended 10 meetings, (7) worked on 10 reports, and (8) worked on several issues of *Nuclear Safety*. The DOE report on the usage of the data bases on the RECON system for the period August 1 to August 29, 1980, is presented in Table 4.1. Note that NSIC is the fourth most widely used of the 26 data bases on RECON.

Table 4.1. RECON data base activity from August 1 to August 29, 1980 (21 operating days)

Date base identification	Supporting installation identification	Data base name	Number of sessions	Number of expands	Citations printed
EDB	TIC	DOE energy data base	4,342	6,380	160,637
NSA	TIC	Nuclear sciences abstracts	652	959	11,332
WRA	WRSIC	Water resources abstracts	230	575	10,062
RIP	DOE	Energy research in progress	177	230	1,795
NSC	NSIC	Nuclear Safety Information Center	169	313	12,773
GAP	DOE	General and practical information	150	134	196
EMI	EMIC	Environmental mutagens information	137	395	3,405
ETI	ETIC	Environmental teratology	117	663	4,597
FED	DOE/EIA	Federal energy data index	108	175	464
ESI	EIC	Environmental science index	93	217	390
WRE	WRSIC	Water resource research	67	192	909
EIA	EIC	Energy information abstracts	65	125	334
NRC	LC	National referral center	50	76	896
IPS	TIC	Issues and policy summaries	45	34	56
PRD	TIC/NRC	Power reactor dockets	36	40	114
CIM	DOE	Central inventory of models	31	35	82
ERG	BERC	Enhanced oil and gas recovery	31	42	221
TUL	U. TULSA	Tulsa data base	31	76	299
SLR	FRANKLIN	Solar data bases	27	25	110
API	API	American petroleum data base	26	64	438
NSR	NDP	Nuclear structure reference	25	21	109
NES	NESC	National energy software	24	51	20
RSI	RSIC	Radiation shielding information	23	60	552
EIS	TIRC	Epidemiology information system	20	11	3
ARF	EMIC/ETIC	Agent registry file	12	8	1
RSC	RSIC	Radiation shielding codes	9	4	

In accordance with the DOE Information Center Pricing Policy, which has been applicable to NSIC, those organizations or persons outside of our sponsors and their direct subcontractors must pay for the services received. As noted, NSIC performed 8 paid searches during the quarter and at the end of the quarter had 28 paid SDI subscriptions. (Note that the number of paid subscriptions is decreasing because we anticipate the elimination of this service in FY-1981.) In any event, these funds are not received by NSIC but, in accordance with DOE policy, are collected by the National Technical Information Service, serving as our billing agent, and then credited to the DOE General Fund.

4.2 ORNL-NSIC Reports

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

ORNL/NUREG/NSIC-161, *Nuclear Power and Radiation in Perspective* (in preparation)

ORNL/NUREG/NSIC-167, *Role of Probability in Risk and Safety Analysis* (in preparation)

ORNL/NUREG/NSIC-168, *Annotated Bibliography on the Transportation and Handling of Radioactive Materials* (in reproduction)

ORNL/NUREG/NSIC-172, *Annotated Bibliography on Fire and Fire Protection in Nuclear Facilities* (in preparation)

ORNL/NUREG/NSIC-177, *Bibliography of Microfiched Foreign Reports Distributed under the NRC Reactor Safety Research Foreign Technical Exchange Program* (in reproduction)

ORNL/NUREG/NSIC-178, *Summary and Bibliography of Safety-Related Events at Boiling Water Nuclear Power Plants as Reported in 1979* (in preparation)

ORNL/NUREG/NSIC-179, *Summary and Bibliography of Safety-Related Events at Pressurized Water Nuclear Power Plants as Reported in 1979* (in preparation)

ORNL/NUREG/NSIC-180, *Nuclear Power Plant Operating Experience — 1979 Annual Report* (in preparation)

ORNL/NUREG/NSIC-181, *Summary Report on Light Water Reactor Systems Survey* (in preparation)

ORNL/NUREG/NSIC-182, *Accident Sequence Precursors of Potential Severe Core Damage: Status Summary Report* (in preparation)

4.3 Journal of *Nuclear Safety*

Preparation and review of material for the technical progress review, *Nuclear Safety*, continued apace with the requirements of its bimonthly publication schedule. All technical articles for *Nuclear Safety* 22(1) were completed and mailed to NRC, DOE, and Technical Information Center (TIC) on September 19, 1980. The "current events" material (covering events that occurred in July and August) for *Nuclear Safety* 21(6) were completed by September 19 (except for data on operating power reactors, which were not yet available from NRC). Most technical articles for *Nuclear Safety* 22(1) have been received and are in various stages of preparation. Final copies of *Nuclear Safety* 21(5) were received from the printer on September 18, and the galleys for *Nuclear Safety* 21(6) will be sent to the printer by TIC early next month. The regular bimonthly *Nuclear Safety* staff meeting was held August 6. Minutes of that meeting and tentative outlines for the next several issues of *Nuclear Safety* were reviewed and distributed August 11 and 8, respectively. Concern for the continued participation of TIC in the preparation of *Nuclear Safety* was alleviated when DOE Resource Applications (to whom TIC reports) came up with the necessary funding.

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ORNL BREEDER REACTOR SAFETY — LOA SUPPORT QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY—SEPTEMBER 1980

M. H. Fontana G. F. Flanagan

AF 15 10 25 0

Task 05 — ORNL LOA Support and Integration: Breeder Reactor Reliability Data Analysis Center

OBJECTIVE

To define, develop, and maintain a central reliability data analysis center for use in dissemination of reliability data in support of safety programs.

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[redacted] distribution by any holder of the document or of the data [redacted] energy.

NOTICE This document is an internal report of the [redacted] Program and contains preliminary [redacted] information. It is subject to revision or correction [redacted] not represent a final report.

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ORNL BREEDER REACTOR SAFETY — LOA SUPPORT QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY—SEPTEMBER 1980

M. H. Fontana G. F. Flanagan

ABSTRACT

Initial development of the Centralized Reliability Data Organization (CREDO) Data Base Management System (DBMS) has been completed. Searching routines have been tested on real data from Experimental Breeder Reactor—II (EBR-II) and have demonstrated the capability of producing various standardized outputs. The collection of engineering, historic event, and operating data from EBR-II and engineering data from Fast-Flux Test Facility continues. Emphasis is being placed on developing user services in the area of statistical analysis of data which is output from searches done by CREDO's DBMS.

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5. TASK 05 — ORNL LOA SUPPORT AND INTEGRATION: BREEDER REACTOR RELIABILITY DATA ANALYSIS CENTER

G. F. Flanagan P. M. Haas
H. E. Knee S. D. Hudson
J. J. Manning N. M. Greene
G. W. Cunningham

5.1 Initial Data Collection and Evaluation

Engineering data collection for the Argon Distribution System of Experimental Breeder Reactor—II (EBR-II) continues. Engineering data collection was initiated on the following systems at EBR-II: (1) Primary Cover-Gas Monitoring System, (2) EBR-II's Silicone Loop, and (3) Cover-Gas Cleanup System.

EBR-II operating reports for the first two quarters of 1980 were incorporated with existing data. Follow-up data from EBR-II's Primary Sodium Storage and Processing System and the Fuel Failure Monitoring System is now being reviewed and incorporated into existing data.

Partial engineering data has been collected on all sodium valves in Primary and Secondary Heat Transport Systems and the Dump Heat System of

the Fast-Flux Test Facility (FFTF). These forms, along with other partially completed forms, are being compiled for transfer to FFTF for review and determination of missing information.

Work was reinitiated on the translation of FFTF data tapes into Centralized Reliability Data Organization (CREDO) format structures and on the establishment of data transfer procedures between CREDO and FFTF's Data Utilization Program via magnetic computer tape.

5.2 Development of the DBMS

Inventory report generation capabilities for outputs reviewed and approved by the CREDO Steering Committee have been programmed into CREDO's Data Base Management System (DBMS) and have been debugged.

Testing of the DBMS with respect to searching and output generation was initiated and completed this quarter. Existing capabilities include (1) the generation of failure rates on any set of qualified CREDO components and (2) the capability of generating engineering and event inventories, engineering summaries, component failure summaries, and quantitative failure data summaries. Programs also exist for automatic data entry via interactive display terminals.

The capability of calculating the Kolmogorov-Smirnov (K-S) statistic for the determination of the goodness-of-fit for the five different probability distribution functions within CRESTCO was initiated and completed. Work on the robust methods version of the K-S statistic was initiated.

Work was initiated on software that will provide individual component lifetimes. This program is needed to provide input to the CRESTCO statistics code.

5.3 Information and Analysis Services

Three data requests were processed this report period that pertain to the following: (1) data on large sodium valves, (2) data on large non-electromechanical pumps, and (3) data on sodium leaks.

5.4 Interface and Coordination

In response to two Action Items from the CREDO Steering Committee Meeting, the CREDO staff took the following action.

1. A letter to the CREDO Test Loop Representatives of the Steering Committee was prepared that summarized previous data collected and data to be collected and gave a manpower estimate for the remaining effort. Test Loop Representatives responded with their own estimates for this effort. In addition, CREDO formulated initial guidelines for the data collection efforts at the test loops and forwarded them to the representatives for review and comment.
2. A response to A. Torri's (GA) letter concerning statistically dependent failures was prepared and forwarded to the Chairman of the Data Evaluation Working Group [R. Waller, Los Alamos Scientific Laboratory (LASL)] and to other interested parties. The working group will respond with recommendations and guidelines concerning statistically coupled events.

The chairman of the Output Working Group [R. Simonelli, Westinghouse Advanced Reactor Division (WARD)] provided CREDO with some preliminary ideas concerning data security, user access, and data dissemination restrictions. These items are now being reviewed.

A member of the CREDO staff met with EBR-II personnel in Idaho Falls to organize data collection efforts there for FY-1981. Problem areas in prior collection efforts were discussed.

Subcontracts for the data collection efforts at FFTF and EBR-II have been renewed at the same funding level as FY-1980. Guidance from DOE indicates that increased funding will be received for FFTF data collection for FY-1981. The FFTF subcontract will be increased when the FY-1981 budget is approved.

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AF 15 10 25 0

Task 06 — ORNL LOA Support and Integration: Central Data Base for Breeder and Reactor Safety Codes

OBJECTIVE

To disseminate LOA technology through the development and maintenance of a central computerized data base (SACRD) for safety analysis computer codes.

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[redacted] This doc[redacted]rnal report of th[redacted]ctor [redacted]ns preliminary [redacted]tive information. It is su[redacted]ion or correction and therefo[redacted] present a fina[redacted]report.

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ORNL BREEDER REACTOR SAFETY — LOA SUPPORT QUARTERLY TECHNICAL PROGRESS REPORT FOR JULY—SEPTEMBER 1980

M. H. Fontana G. F. Flanagan

ABSTRACT

Work continued on entering *Nuclear Systems Materials Handbook* update information in the Safety Analysis Computerized Reactor Data base. Work from a recent Argonne National Laboratory evaluation was initiated. A newsletter was written. The *Interactive Terminal Users Guide* was rewritten and is being prepared for distribution as hard-copy handbooks.

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6. TASK 06 — ORNL LOA SUPPORT AND INTEGRATION: CENTRAL DATA BASE FOR BREEDER AND REACTOR SAFETY CODES

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6.1 LOA Integration-Technology Dissemination

Work continued on entering *Nuclear Systems Materials Handbook* (NSMH) update package No. 19 and the saturated sodium properties file into the data base. Additional uncertainty data were added. Some corrections and modifications to version 1 were made. The NSMH update package No. 20 was received during the latter part of the quarter.

A SACRD newsletter was prepared and mailed during the quarter. The physical locations of many SACRD data files were changed because of a sequence of computer system errors that seriously threatened the integrity of the data base. As a result, file backup procedures include several instances of redundancy that did not previously exist. Edits with which to prepare hard-copy handbooks have been made and are in the process of being matted for reproduction.

Work was initiated toward converting JOSHUA to be operational on the IBM MVS Operating System, which will replace the older MVT System sometime

during the next quarter. This task is required to continue to allow iterative communication with the SACRD data base.

6.2 Reports

SACRD Bulletin

ORNL/CSD-53, *SACRD — A Data Base for Fast Reactor Safety Computer Codes — Operational Procedures*

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