27
12-22-77
5 to UT 15

SAND77-1134
NUREG-0181-3
Unlimited Release

3151

Distribution
Category NRC-7

ADVANCED REACTOR SAFETY RESEARCH
PROGRAM QUARTERLY REPORT
APRIL - JUNE 1977

MASTER

Advanced Reactor Research Department

Prepared by Sandia Laboratories, Albuquerque,
New Mexico 87115 and Livermore, California 94500
for the United States Nuclear Regulatory Commission
under ERDA Contract AT(29-1)-789.

Printed November 1977

Sandia Laboratories

Nuclear Fuel Cycle Programs

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

Issued by Sandia Laboratories, operated for the United States
Energy Research & Development Administration by
Sandia Corporation.

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research and Development Administration, nor the United States Nuclear Regulatory Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Printed in the United States of America
Available from
National Technical Information Service
U. S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161
Price: Printed Copy \$7.75; Microfiche \$3.00.

9.25

SAND77-1134
NUREG-0181-3
Unlimited Release
Printed November 1977

Distribution
Category NRC-7

ADVANCED REACTOR SAFETY RESEARCH PROGRAM*
QUARTERLY REPORT
April - June 1977

Submitted by

Sandia Laboratories,** Albuquerque, New Mexico 87115

Principal Organizations:

Reactor Studies Division, R. L. Coats, Supervisor
Radiation Physics Division, J. E. Powell, Supervisor
Thermomechanical and Physical Research Division, B. M. Butcher, Supervisor
Reactor Safety Studies Division, D. A. Dahlgren, Supervisor
Design Technology Division, R. E. Nickell, Supervisor
Reactor Theory and Analysis Division, W. J. Camp, Supervisor
Environmental Research Division, B. D. Zak, Supervisor

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

APPROVED:

Manager, Advanced Reactor Research

Director, Nuclear Fuel Cycle Programs

*This work was supported by the U. S. Nuclear Regulatory Commission (Project Nos. A-1016, A-1054, A-1058, A-1172, A-1181, A-1186, A-1197, and A-1198).

**Operated for the U. S. Energy Research and Development Administration by Sandia Laboratories under Contract AT(29-1)-789.

CONTENTS

	<u>Page</u>
Introduction	7
Executive Summary	15
1. Accident Energetics	41
1.1 Prompt Burst Energetics	41
1.2 Equation of State	63
1.3 Transition Phase	70
1.4 Aerosol Source Normalization - In-Pile Studies	78
1.5 Fuel Coolant Interaction	80
1.6 Large-Scale Test Capabilities	80
1.7 Fuel Dynamics	106
2. Core Debris Behavior	111
2.1 Molten Core Technology	111
2.2 Debris Bed Studies	122
2.3 Molten Fuel Pool Studies	129
2.4 Containment Analysis	142
3. Sodium Containment and Structural Integrity	145
3.1 Introduction	145
3.2 Preliminary Tests	146
3.3 Posttest Results	151
3.4 Facility Modifications	154
3.5 Safety	155
4. Research for Elevated Temperature Design Criteria	157
4.1 Creep-Fatigue Test Program	157
4.2 Creep Buckling Analysis	165
4.3 Cell Liner Analysis	165
4.4 General	167
5. Fuel Motion Detection	169
5.1 Introduction	169
5.2 In-Core Fuel Motion Detection	170
5.3 Electron Beam Flash X-Radiography	182
5.4 Gamma-Ray Coded Aperture Imaging in Many-Pin Bundles	183
6. ACPR Fuel Motion System	189
6.1 Additional Results of the Fuel Pin Imaging Experiment at SPR II	189
6.2 Collimation Test Module Experiment at ACPR	190
6.3 Detector Resolution Studies	194
6.4 Computer Reconstruction	195
7. Advanced Reactor Safety Research Assessment	199

ILLUSTRATIONS

<u>Figure</u>		<u>Page</u>
1	Axial Fission Energy Deposition in 14% and 20% Enriched UO_2 Irradiated in the PBE Experiment with 32-mm Thick Polyethylene Moderator	46
2	New Assembly for Prompt Burst Excursion Experiment	49
3	Drawing of Uranium Carbide Fuel Pin for PBE-SG Series of Experiments	53
4	Radially Averaged Energy Deposition versus Polyethylene Thickness for 15% Enriched UC During ACPR Maximum Pulse	57
5	Radial Fission Energy Deposition Distribution for 14% UO_2 and 15% UC with 1.25" Polyethylene Moderator During Maximum ACPR Pulse	58
6	Comparison of Dynamic Vaporization Data and Models for Uranium Dioxide	65
7	Diagram of the Vaporization Experiment Package Mounted in SPR-III	69
8	Central Rod Cladding Temperature ($t = 17$ ms) for Inter-Pin Power Ratio of 1.1	86
9	Central Rod Cladding Temperature ($t = 17$ ms) for Unity Inter-Pin Power Ratio	87
10	Cladding Hot Spot Temperature Across Bundle (Inter-Pin Power Ratio = 1.1)	89
11	Performance Capabilities of Candidate Fuels for a Gas-cooled Fast Reactor Test Facility	91
12	Effect of a Fe-Al Converter on Performance	93
13	Effect of a UO_2 Converter on Performance	94
14	Neutron Energy Losses in the Test Assembly	95
15	Effect of Converter on Performance, Water-cooled Concept	98
16	Effect of Fuel Loading on Performance, Water-Cooled Concept	99
17	Block Diagram of Gas Cooling System	102
18	Capsule for Fuel Failure Experiments Currently being Conducted in ACPR	108
19	Photograph of Fire and Pyrotechnics as Result of Contact between Melt and Concrete	113
20	In-Melt Temperature from Immersion Thermocouple in Test LSCRBR-2	116
21	Gas Phase Temperature 40.6 cm Above Melt in Test LSCRBR-1	117
22	Output from Displacement Gauges for Test LSCRBR-1	118
23	Cross-Sectional Map of Crucible Following Test LSCRBR-1	121
24	Matrix of Conditions Attained During the Experiment	127
25	Tracing of Thermocouple Signals	127

Illustrations (cont'd):

<u>Figure</u>		<u>Page</u>
26	Five-Element Ultrasonic Thermometer in UO ₂	133
27	Ultrasonic Thermometer Records for First In-Core Molten Pool Experiment	135
28	Temperature Profile Data in Fuel Near Incipient Melt	136
29	Calibration Curve for 2% Thoriated Tungsten Ultrasonic Thermometer	139
30	Response of Pressure Gauges of Temperature and Radiation	141
31	Sodium Pool Temperatures During Test No. 1	149
32	Sodium Pool Temperatures During Test No. 2	150
33	Posttest Configuration of Crucible P=2	153
34	Positron Annihilation (Doppler Broadening Method)	163
35	Schematic of PNC Elbow Creep Buckling Experiment	166
36	Isometric View of the Fuel Positioning System and Collet Locking Device	171
37	Detector Support Assembly Showing Detector Locations	172
38	Escaping Photons (Photons/SR-Primary Photon)	186
39	Escaping Photons (Photons/SR-Primary Photon), $E > 0.6$ MeV	187
40	Collimation Test Module	191
41	Modified Cylindrical Polyethylene Liner	193
42	Rough Scans of Spatial Resolution Values Obtained by Coded Aperture Imaging System Detectors	196

TABLES

<u>Number</u>		<u>Page</u>
I	Comparison of UC and UO ₂ Fuel Pins for PBE	55
II	PBE Experiments Through ACPR Shutdown	61
III	Some Input Parameters	83
IV	Bundle Characteristics	84
V	Fuel Specifications	96
VI	Cost Summary for Cooling System	105
VII	Gas Composition from Large-Scale Test	120
VIII	Comparison of Dominant Parameters of Out-of-Pile Experiments and D-Series Experiments	130
IX	Initial Conditions for Sodium-Concrete Interaction Tests	147
X	Unwelded 2-1/4 Cr - 1 Mo Steel	158
XI	Welded 2-1/4 Cr - 1 Mo Steel	159
XII	NDE (Ultrasonic) Tests of Unwelded, Creep-Fatigued 2-1/4 Cr - 1 Mo	161
XIII	Experiments with Metallic Paint Contacts	177
XIV	Experiments with Ion Implanted Contacts	179-180

ADVANCED REACTOR SAFETY RESEARCH PROGRAM QUARTERLY REPORT

INTRODUCTION

Sandia Laboratories' Advanced Reactor Safety Research Program, initiated in FY 1975,¹⁻¹⁰ is a comprehensive research activity conducted on behalf of the U. S. Nuclear Regulatory Commission (NRC) and is part of NRC's confirmatory research effort to assure that the necessary safety data and theoretical understanding exist to license and regulate the Liquid Metal Fast Breeder Reactor (LMFBR) or other advanced convertors and breeders which may be commercialized in the United States. The program includes a broad range of experiments to simulate accidental transient conditions in the LMFBR to provide the required data base to understand the controlling accident sequences and to serve as a basis for verification of the complex computer simulation models and codes used in accident analysis and licensing reviews. Such a program must include the development of analytical models, verified by experiment, which can be used to predict reactor performance under a broad variety of abnormal conditions. This work along with that of other U. S. and international researchers should provide the technology base on which licensing decisions can be made in confidence that the safety of the public is assured.

The early thrust of Sandia's program was designed to provide data associated with the hypothetical core disruptive accident, with emphasis on prompt burst (~ 1-ms period) energetics and the behavior of post-accident core debris. The scope of the program was expanded in FY's 1976 and 1977 to encompass other energetics and inherent retention concerns such as large-scale sodium containment and structural integrity; aerosol source studies; transition phase energetics; fuel failure and motion; and studies to quantify elevated temperature failure modes of critical component materials. A portion of the initial effort in the program was directed toward obtaining data to support the licensing review of the Clinch River Breeder Reactor (CRBR). Another important early thrust of the program is the development of the test facilities and techniques. In mid-FY77 three additional tasks were initiated, Fuel Failure, Containment Analysis, and Advanced Reactor Safety Research Assessment. For FY77, the program is organized in the following subtasks, progress on which is reported herein.

TASK 1 Accident Energetics

- Subtask 1.1 - Prompt Burst Energetics - Studies performed on single- and multi-pin geometries (oxides, carbides, and nitrides) with and without sodium to determine the consequences of fuel failure under conditions of prompt burst power excursions. The ACPR is operated in the pulse mode with initial periods as short as 1.3 ms.
- Subtask 1.2 - Equation-of-State (EOS) - The equation-of-state for high-temperature vapor pressure for fresh and irradiated oxide fuels and advanced fuels will be obtained. Sandia's electron beam and reactor facilities will be used for transient heating of reactor fuels to high temperatures (3,000 to 10,000 K) and pressures (1.5 to 300 MPa) to obtain the required EOS data.
- Subtask 1.3 - Transition Phase - Work in this task will be carried out to better define and resolve key uncertainties in the transition phase of LMFBR accident sequences. There will be two main areas of investigation: (a) support and experimental testing of SIMMER and other relevant transition phase computer codes and (b) a study of key phenomenological problems relevant to the transition phase. The latter work need not necessarily be directed toward the support of a specific computer code.
- Subtask 1.4 - Aerosol Source Normalization - Aerosol source normalization determined from in pile experiments will characterize physical properties of fuel particles resulting from the breakup of fuel pins subjected to simulated overpower excursions in the ACPR. The characteristics of these particles will be compared with those produced by out-of-pile techniques at ORNL. Subsequent in-pile studies will examine the interaction of particles with the supporting structure and with sodium.
- Subtask 1.5 - Large-Scale Test Capabilities - (a) Full-Length Capabilities, High Fluence Fast Pulse Reactor (HFFPR). A facility capable of performing large phenomenological experiments at short periods is under study. Such a facility would provide a considerable extension of capability beyond the phenomenological upgraded ACPR and would provide the basis of NRC's test program in the 1980's. In addition to LMFBR safety tests, the facility (due to the large area access to the core) would provide excellent capability for development of reactor safety diagnostics, particularly material motion detection systems, and for performance of

experiments which will observe the motion of core materials in simulated accidents. (b) Large-Scale Tests. One of the main problems facing the U. S. LMFBR Safety Program is a lack of adequate facilities to do large-scale testing. New facilities and upgrades of existing facilities have been proposed, but there is a possibility that they will not be available in time to support a 1986 commercialization decision. The Nation's Underground Nuclear Test Technology Program, which utilizes the facilities at the Nevada Test Site, has the potential for providing some of these needed data from 5 to 7 years before the proposed safety test facilities will be on-line. Studies are currently under way to determine the feasibility of such tests. Included in the study are source-driven, self-driven, and hybrid tests.

Subtask 1.6 - Fuel Dynamics Experiment - Fuel-coolant interaction studies will be conducted to determine the potential for the rapid transfer of energy from the fuel to the coolant. Initial efforts are concentrated on simulation tests (see Subtask 1.1). As results from these tests are obtained, phenomenological studies will be developed as appropriate. Emphasis in the future will be on the advanced fuels.

Subtask 1.7 - Fuel Failure - An integral part of the total analysis of postulated reactor accidents is the analysis of fuel failure. In this subtask, the mode and time of disruption of irradiated fuel pins and the rate of dispersal of fuel from disrupted pins will be investigated using the ACPR with high temporal and spatial resolution diagnostics. This effort will include analysis of results from FY77 experiments, reporting on the first set of irradiated oxide fuel experiments on time dependent fuel failure, design of follow-on experiments, resumption of oxide testing, and the commencement of tests on alternate and oxide fuels.

TASK 2 Core Debris Behavior

Subtask 2.1 - Molten Core Technology - Studies will be performed in the areas of molten UO_2 and steel interactions with steel liners and concrete for application to inherent retention concepts. The capability will be developed to include application to other core retention materials. Analysis of the accident sequences will be included in this task.

- Subtask 2.2 - PAHR Debris Bed - Post-accident heat removal (PAHR) debris bed studies will be undertaken to characterize the thermal behavior of internally heated debris beds to determine their melt-through potential in order to provide information needed for analysis of post-accident containment. The experiments will utilize fission heating and will be performed in-pile using the ACPR (Annular Core Pulse Reactor) in the steady-state mode.
- Subtask 2.3 - PAHR Molten Pool - PAHR molten pool studies will be conducted to characterize the thermal and kinetic behavior of internally heated molten pools of fuel materials, including fission product retention, and to determine the melt-through potential of such configurations to be used in analysis of post-accident containment. The experimental program will utilize fission heating of UO_2 fuel and will be performed in-core using the Annular Core Pulse Reactor in the steady-state mode.
- Subtask 2.4 - Containment Analysis - This subtask involves the development of a containment code for advanced reactor systems. The code will be capable of following a variety of possible accident sequences from the point of release from the primary containment to the development of the radiological release source term. The final model will be general enough to apply to all candidate advanced reactor systems, although the near term emphasis will be on sodium-cooled systems. The model will incorporate all significant state-of-the-art submodels and methods relevant to accident phenomenology and venting, recirculating flows, aerosol behavior, and other pertinent phenomena.

TASK 3 Sodium Containment Structural Integrity

This experimental study of large-scale sodium interactions with reactor materials is in support of the NRC confirmatory research program for sodium-cooled reactor systems. The immediate effort will be divided into two parts: sodium/concrete interactions and sodium/liner interactions. The sodium/concrete interactions focus on the exothermic reactions occurring when sodium is brought into rapid contact with concrete, with emphasis on the following four objectives: (a) quantitative measurement of the quantity and composition of the evolved gases; (b) quantitative measurement of the penetration rate of the sodium reaction zone into the concrete, including estimates of thermal cracking and exfoliation; (c) identification of the influence of special engineering features

such as the presence of cracks and steel reinforcements on the sodium penetration rate; and (d) development or confirmation of an analytic model of the process for use in licensing verification. These objectives will be supplemented by thermochemical measurements as appropriate. Generic Southeastern U. S. and other concrete types with possible LMFBR applications will be investigated. The liner interaction will be an analytic and experimental program to explore the consequences of hot sodium spills or sprays on steel liners over concrete substructures. This task will specifically address the situation in which a flaw in either one of the liners permits sodium to breach the liner and come in contact with the concrete substructure. The objective of this task is to establish the conditions for which the sodium/concrete reactions are sufficient to cause further damage to the liner and limit its effectiveness.

TASK 4 Research of Elevated Temperature Design Criteria

The primary objectives of the elevated temperature design criteria studies are: (a) to develop correlative NDE techniques that can be used to monitor progressive creep, fatigue, and combined creep-fatigue damage in LMFBR component materials; (b) to study the parameters that induce failure in steel liners following an accidental sodium spill in an equipment cell; (c) to evaluate the analytical methods and correlate with experimental data for creep buckling of LMFBR components operating at elevated temperatures; and (d) to design, fabricate, and operate a multiaxial creep-fatigue facility for the purpose of evaluation of creep-fatigue damage rules.

TASK 5 Fuel Motion Diagnostics

Subtask 5.1 - In-Core Systems - Facilities such as SLSF and EBR II do not have slots for high-resolution fuel motion detection devices. On large tests (subassembly and multi-assembly), planned for SAREF and other facilities, self-shielding may preclude using techniques such as coded aperture imaging, flash x-radiography, and the neutron hodoscope. The feasibility of in-core fuel motion detection will be investigated through analysis and experiments. The development of unfolding techniques and miniature radiation detectors, as well as experiments in SPR III utilizing a 37-pin bundle, are part of this task.

- Subtask 5.2 - Electron Beam Flash X-Radiography - Electron beam flash x-radiography will be investigated as a possible high-resolution, two-dimensional fuel motion detection device for up to subassembly size experiments. Point source as well as coded source techniques will be studied. Source requirements, resolution, detection systems, and ultimate system costs will be considered.
- Subtask 5.3 - Coded Aperture Imaging - During FY 1975 and FY 1976, Sandia performed studies to support the definition of a fuel motion detection system for the ACPR Upgrade. A system was defined and is now being developed under the program (see Task 6). These studies have the goal of defining an envelope of capability for full-length LMFBR fuel pins and to support system selection for SAREF. The high-resolution, three-dimensional capabilities of the technique will be emphasized.

TASK 6 ACPR Fuel Motion System

A fuel motion detection system is to be developed for the ACPR Upgrade. The design of the system will be based upon the experience gained in experiments with shielding, collimation, coded aperture imaging, and time-resolved pseudoholography at the ACPR, SPR II, and ^{252}Cf facilities. The system will be designed for high spatial and temporal resolution measurements of fuel motion in both transient and steady-state experiments, including 1- and 7-pin prompt burst excursion experiments, bed leveling and material separation studies in the post-accident heat removal experiments.

TASK 7 Advanced Reactor Safety Systems Assessment

The purpose of this task is to investigate the applicability of the methodology developed by the Reactor Safety Study (RSS) to a Liquid Metal Fast Breeder Reactor (LMFBR). The study will be based on a generic design for a three loop LMFBR with an intermediate sodium heat transport system. Engineering safety system and accident phenomena event trees will be constructed and probability ranges assigned in order to: (a) provide a demonstrated methodology for developing accident sequence diagrams for a LMFBR, (b) determine dominant accident sequences with corresponding radioisotope release categories, and (c) identify and systematize the key phenomena in the accident sequence.

TASK 8 ACPR Performance Improvement of the Annular Core Pulse Reactor

This task is reported in a separate quarterly report - see Annular Core Pulse Reactor Upgrade Quarterly Report April-June 1977, SAND77-1133 Sandia Laboratories, Albuquerque, NM, July 1977.

EXECUTIVE SUMMARY

1. Introduction

The Advanced Reactor Safety Research Program at Sandia Laboratories, Albuquerque, New Mexico is being conducted on behalf of the U. S. Nuclear Regulatory Commission (NRC). The program has as its overall objective the provision to NRC of a comprehensive data base essential to: (1) defining key safety issues, (2) understanding the controlling accident sequences, (3) verifying the complex computer models used in accident analysis and licensing reviews, and (4) assuring the public that advanced power reactor systems will not be licensed and placed in commercial service in the United States without appropriate consideration being given to their effects on health and safety.

The NRC program is carefully planned to complement the larger ERDA program, but at the same time to satisfy the NRC obligation of independent confirmatory research.

Together with other programs, the Sandia effort is directed at assuring the soundness of the technology base upon which licensing decisions are made, and includes experiments and model and code development.

Sandia Laboratories has been tasked to investigate seven major areas of interest which are intimately related to overall NRC needs. These are:

- a. Accident Energetics
- b. Core Debris Behavior
- c. Sodium Containment and Structural Integrity
- d. Research of Elevated Temperature Design Criteria
- e. Fuel Motion
- f. ACPR Fuel Motion System
- g. Advanced Reactor Safety Research Assessment

These major tasks are subdivided as necessary into subtasks to facilitate the organization of work and the interaction of subtask results into a body of coherent information which supports the requirements of the NRC.

2. Accident Energetics

2.1 Prompt Burst Energetics (PBE)

2.1.1 Introduction

The PBE Program is planned to identify and characterize phenomena associated with super-prompt critical core disruptive accidents. Experiments are designed to provide information of phenomena resulting in pressure generation and conversion of thermal energy to work in various reactor fuel-clad-coolant systems. Information on the energetic response, thermodynamic state, and spatial relocation of reactor materials during a super-prompt critical excursion is obtained. The experimental work is closely interfaced with the development of analytical models which uniquely describe the observed phenomena for incorporation into predictive accident analysis. The data being developed serve as initial conditions for post-accident heat removal studies.

In tests performed to date, single fuel pins have been pulse fission heated in the Annular Core Pulse Reactor (ACPR) to temperatures resulting in fuel vaporization. The pins, surrounded by helium or sodium, are contained in a rigid pressure vessel instrumented with thermocouples and pressure transducers and fitted with a movable piston at its upper end. Results of these experiments permit estimates to be made of the conversion of thermal energy to work. In addition, data histories are determined for a variety of fuel-coolant systems and initial conditions. A series of experiments using fresh uranium dioxide (UO_2) fuel and sodium coolant is in progress. A series using uranium carbide (UC) and sodium will follow later this year.

2.1.2 PBE Analysis

The fuel coolant interaction (FCI) code EPIC was debugged and became operational on the Sandia CDC system. During preliminary runs, it was obvious that the fuel vapor model in the code was inadequate resulting in inaccuracies in the way in which fuel was ejected from the pin into the sodium channel. Modifications to the fuel vapor model were designed and are being coded into EPIC.

Parametric studies on the effect of varying the sodium to fuel ratio in the PBE experiment yielded no noticeable effect although, in principle, a lower sodium/fuel ratio should enhance the possibility of a strong FCI. Computer results seem to indicate anomalies in the timing of an FCI which could be caused by previously described problems with the EPIC fuel vapor model.

Additional work has been done on EXPAND, a pin model used to determine the temperature-stress-strain state of the fuel and cladding in a pin prior to clad failure. The predictions obtained from EXPAND agreed closely with other methods of determining pin failure times in the PBE-5S experiment but met with less success in PBE-3S (multiple pulse) in which they indicated failure by plastic instability where none occurred experimentally. Several reasons for disagreements between computer predictions and experimental results are being evaluated.

The CSQII free piston hydrodynamic calculation was repeated, this time at 200-bar fuel vapor pressure. The bottom/top pressure ratio was 1.57 which agreed closely with the 1.52 obtained experimentally.

2.1.3 Dosimetry Experiments

A series of dosimetry experiments was performed to define the axial energy deposition profile in the PBE tests and to quantify the maximum radially-averaged energy deposition for the UO_2 pins used in previous tests. The energy deposited in irradiated fuel pellets was determined by fission product inventory techniques. The results were scaled by

the ratio of measured neutron fluences to yield estimates of the energy deposition in the PBE tests. The use of fluence values for scaling was accurate to within 0.5 percent and the uncertainty in the fission product inventory results was less than 5 percent.

2.1.4 New Pressure Capsule

A new pressure vessel has been designed and is being fabricated for use in PBE experiments. It is intended to permit installation of the fuel pin as the last step of assembly rather than one of the first and will facilitate future experiments.

2.1.5 Uranium Carbide Fuel Pins

An agreement was made with GFK, Karlsruhe, Federal Republic of Germany (FRG) to provide uranium carbide (UC) fuel pins for the PBE experiments and to provide UC pellets for EOS experiments. These fuel pins will be compatible with the new pressure capsule. The pins (except for wire wrap) will be delivered to Sandia Laboratories about August 10, 1977. The space wires will be installed at Los Alamos Scientific Laboratory.

2.1.6 Transient Rod Upgrade

To increase the energy deposition in PBE experiments during multiple pulse operations, work has been undertaken to increase the worth of the ACPR transient rod bank. This "mini-upgrade" will increase the transient rod bank worth from the present \$4.50 to an expected \$4.95. An experiment plan for the installation and testing of the "mini-upgrade" rods in the ACPR is under review by safety review committees.

2.1.7 Pressure Gage Calibration

In PBE experiments, pressure gages are a primary source of information for following the energy exchange between the vaporized fuel and the

sodium coolant. The purpose of this ongoing study is to estimate the relation between pressure gage response and source pressure, modified by the experiment geometry.

A second test was run in a study to explosively generate a shock pulse in water at least as long as a PBE pulse (~ 1 ms) and with a similar amplitude (~ 5000 psi). A 4 ms wide pressure pulse with 100 μ s, 5000 psi spike was produced 10 feet from the detonation. Pressure relief was probably due to tank rupture. A thicker walled reuseable tank is being installed to produce longer duration water shocks with less explosive. Gage calibration hardware has been assembled and checked and data from the last proof test is being examined.

2.1.8 Status

Eight PBE tests are planned during the next quarter. Components for these tests are being fabricated or purchased. The UC fuel pins are expected in early August with the wire-wrap to be completed before the end of August. The increased-worth transient rods are expected to be completed by mid-July. The internal safety committee reviews should be completed in mid-July.

2.2 Equation-of-State (EOS)

2.2.1 Present Status

Information on vaporizing UO_2 at temperatures and vaporization rates appropriate to advanced reactor safety analysis is now becoming available from various types of experiments. It has been commonly assumed in fast reactor analysis that the vaporization of fuels at high temperatures can be described by an equation of state for material in chemical and thermal equilibrium. However, comparisons made during the last quarter of dynamic experiments on oxide materials to equilibrium measurements show contradicting results.

High temperature information for UO_2 can be derived from several types of experiments: relatively long duration ones yielding equilibrium models, and more prototypic dynamic experiments, the most recent of which used the ACPR and the Sandia Relativistic Electron Beam Accelerator (REBA). Such experiments, detailed elsewhere in this report, directly describe the relation between energy and vapor pressure which is needed for fast transient calculations.

A dynamic equation-of-state based upon several different transient experiments can be defined. Significant differences between these results and equilibrium data are found which suggest the existence of a rate-dependent process in high temperature UO_2 . The magnitude of these differences emphasizes the importance of directly confirming this rate-dependent effect and measuring the appropriate rate constant.

2.2.2 Electron Beam Vaporization Studies

Recent progress in developing the long pulse mode of REBA is extremely encouraging and installation of experiments and preliminary diagnostic measurements began during the month of June.

2.2.3 Equation-of-State Analysis

Modification of the anharmonic heat capacity calculation has required improvement in an existing phonon code for fluorite structure materials. Completion of the calculation has, therefore, been somewhat delayed, but with the expectation of significant improvements in the accuracy of the resulting calculations.

The possibility has been examined of using various theories of non-equilibrium evaporation to explain fuel motion in the Sandia PBE tests. Such effects alone are not thought to totally account for observed voiding at the ends of fuel pins; ways to incorporate such effects into EPIC are being explored.

Investigations into ways of modeling rate-dependent effects in the heat capacity of UO_2 have continued with a view toward analyzing the results of a planned UO_2 resistive heating experiment.

2.2.4 ACPR, SPR-III Equation-of-State Measurements

In order to avoid any lapse in the equation-of-state program during ACPR shutdown, a proposed experimental procedure and plan for the initial phase of vapor pressure experiments for SPR-III was reviewed and approved by the Sandia SPR-III Experiment Review Committee and the Sandia Reactor Safeguards Committee without significant modification. In addition, several innovative concepts in test apparatus have been conceived in order to adapt the experiments to SPR-III.

2.2.5 New Experimental Directions

An experiment has been designed to measure the dynamic heat capacity of UO_2 . This experiment involves rapidly heating a UO_2 sample to temperatures above 1500 K at which a rapid increase in heat capacity has been noted. Once the final energy state is researched, changes in specific heat with time and/or heating rate can be detected from changes in the sample temperature. Circuit and hardware design for these experiments has been completed and the components are being fabricated.

2.3 Transition Phase

2.3.1 Introduction

Following the loss of initial core geometry in the loss-of-flow accident, and assuming that neither early shutdown nor rapid hydrodynamic disassembly takes place, the accident enters the transition phase. At this point, one or a number of subassemblies have melted down and radial propagation takes place as subassembly hexcans are melted through. The progression of the transition phase can strongly influence the severity of the accident. Therefore, detailed understanding of the mechanics of the accident stage is important. Answers to key

questions depend upon resolution of the uncertainties in transition phase phenomena. Sandia has recently begun studies on the mechanics of fuel blockage formation and boiling pool behavior, which involves both analysis and experimental studies.

2.3.2 Fuel Blockage Formation and Crust Stability-Analysis

Early efforts in this area were initially directed at evaluating previously proposed fuel freezing mechanisms. Resulting models, based upon the premise that fuel crusts on steel cannot be stable, predict fuel penetrations which are two to six times shorter than experimentally observed values. The mechanism which produces the necessary crust removal, and hence rapid freezing, is still unclear.

It has been proposed that crust breakup occurs when the molten steel thickness exceeds the depth of the laminar steel sublayer for the flowing fuel. The fuel crust is treated as a thin membrane which does not affect the radial velocity profile. This membrane is assumed to collapse when subjected to turbulence in the buffer layers. In most cases, this model predicts very rapid crust failure and thus does not appear to explain the discrepancy between the observed and predicted penetration distances. It has been observed that only for very high initial fuel temperatures does the fuel depth exceed the crust thickness at the instant of predicted membrane failure. This suggests that treating the crust as a thin membrane on molten steel may be questionable and that other crust removal mechanisms should be considered.

One such model is failure of the crust due to thermal stresses. This may have important implications not only in blockage formation but also in the ability of a molten core to remain subcritically dispersed by boiling steel. The existence of a stable fuel crust may be of key importance for the existence of a stable, subcritical core boilup.

The effect of thermal stress on the failure of mixed oxide crusts has been examined. If a crust is formed when molten steel contacts cold steel, the temperature drop across the crust will be roughly the difference in melting points between fuel and steel (~ 1350 K). If

the steel surface is solid, it can support the crust and may prevent thermal stress induced failure. However, if the steel surface is molten, the crust will partially fail. It is conservatively estimated that cracks from the cold side of the crust will penetrate about 3/4 of the crust thickness, leaving only 1/4 of the crust to withstand mechanical stress. The cracking of at least 3/4 of the oxide crust thickness should lead to a more rapid breakup of insulating crusts as fuel is ejected from the core. The effect of crust cracking is very important for a crust which is insulating a pool of boiling fuel and steel from the surrounding cold steel walls. Without crust insulation, a full core pool could lose too much heat by convection to remain boiled up, leading to a recriticality as the pool subsides. For a single subassembly pool at initial decay heat power levels the uncracked part of the crust would be about 1 mm thick. For single subassembly pools, the insulating crust may be stable at low power.

Along with modeling of the phenomena, it is essential to accurately predict the development, melt-through, and reformation of blockages. To this end, appropriate codes are being procured and one already received (SIMMER-I) is operative at Sandia for evaluation.

2.3.3. Fuel Blockage Formation and Crust Formation - Experimentation

In order to evaluate current models and to assist in the development of new phenomenological models, experiments with simulants are being planned. A short computer code has been written to evaluate candidate materials based upon available freezing and crust stability models. Early experiments will probably use frozen glycerine as the wall simulant and water as the simulant fuel. Initial fabrication of the glycerine channels is underway. In addition, planning has begun for adequate instrumentation.

Progress has continued on planning for the large-scale 19-pin freezing experiment using $\text{UO}_2\text{-ZrO}_2$ thermite or possibly molten UO_2 . This effort was reported in the January-March 1977 Quarterly.

2.3.4 Boiling Pool Phenomenology

Although the primary effort is directed at the mechanics of blockage formation, work on boiling pools is also being done. Experiments using a 1-kW microwave oven are planned utilizing water or possibly a combination of two immiscible liquids. Various experimental options may be included. These experiments should also provide valuable data for SIMMER model validation.

2.4 Aerosol Source Normalization - In-Pile Studies

2.4.1 Introduction

For those reactor accident scenarios in which the fuel vaporizes and ruptures the cladding, it is necessary that the state of the fuel before and after rupture be characterized by experiment. The first phase of this study conducted during FY 77 is measuring some of the physical properties of UO_2 fuel pins heated to vaporization by neutrons from a pulsed reactor. Succeeding work will study the interaction of neutron-vaporized fuels with the containing structure and with sodium.

Various debris properties of the UO_2 fuel as a function of time are being measured for both bare and clad fuel pins. These characterizations will: a) be used as a particle source term for aerosol codes; b) be sufficiently accurate and complete for comparison with hydrodynamic code predictions; and c) be compared with those particles sampled by similar methods in relevant out-of-pile experiments at Oak Ridge National Laboratory. This comparison activity will begin to establish limits of interest on the particle density and size spectra of the vaporized fuel.

A complete description of the experimental design was presented in the October-December 1976 quarterly.

2.4.2 Progress

Orders for all fabricated parts and hardware for the subsystems of the experiment were completed during this quarter. These components include the experimental canister containing the UO_2 fuel rod and sampling wheel, the passive pumping system, and the optical train for photographing the fuel disrupt event. In addition to experiment subsystems, the canister contains operating and measuring components for control of the experiment and the taking of data.

The experimental canisters have been assembled and function properly. Sampling surfaces in the wheels are being coated and the wheels are being assembled and aligned.

Next quarter, the complete system will be assembled in the ACPR and tests will commence.

2.5 Fuel-Coolant Interaction

Work on this portion of the LMFBR Safety Research Program is not scheduled to begin until FY 78. No activity is reported this quarter.

2.6 Large-Scale Test Capabilities

2.6.1 Optimum Nuclear Pin Bundle for Large-Scale Testing

Because in-pile testing with full LMFBR bundle size requires large and costly facilities, it is desirable to determine the minimum test scale satisfying prototypic requirements. An analytical tool capable of tracing the complete post-failure history for each pin in a true bundle geometry is not currently available. This study attempts to establish prototypic pre-failure thermal hydraulic conditions within the test vehicle, arguing in doing so that they have determinative effects on post-failure events. The conclusions of this study, based on single-phase hydrodynamics, have immediate applications to cases without pre-failure boiling, such as superprompt transients and slower transient overpower (TOP's) in irradiated fuel. For cases with pre-failure boiling, it is believed that achieving prototypic thermal-hydraulic conditions in the

bundle at the onset of pin failure is prerequisite to attaining a prototypic course of events after failure.

The computer program COBRA4I is in use and is being refined to adequately predict the phenomena associated with large-scale testing. The trend of results obtained shows that prototypic thermal hydraulic conditions at the onset of pin failure are at 61 pins with 37 pins probably an acceptable minimum. During fast transients, the cladding is still cold when the outer fuel surface is near melting. The difference in cladding hot spots between 7 and 91 pins is several hundred degrees Fahrenheit, and this difference virtually disappears at 61 pins. An important result is the shifting of cladding hot spots upward by 3-4 inches when the bundle size increases from 7 to 91 pins. A similar trend is observed for the outer-row pin, except for more noticeable irregularities in the axial temperature distributions, presumably due to boundary effects. Results indicate that although the fuel temperature is determined by preburst steady state, the cladding temperature appears to be more sensitive to heat transport between channels during transient. By manipulation of experiment parameters, a temperature profile across a small bundle can be made flatter, simulating the inner rows of the 217-pin bundle.

Calculations using the Sandia ACPR Upgrade axial power profile show a weak influence of this factor on the asymptotic trend discussed above, making the conclusions of this study more general.

2.6.2 New Facility Conceptual Design

As a part of the Sandia Fast Reactor Safety Research Program, the conceptual design of a facility capable of performing phenomenological experiments with full-length fuel pins has been initiated. One approach considers several test-facility design concepts for the High Fluence Fast Pulsed Reactor (HFFPR).

2.7 Fuel Dynamics Experiments

Following the initiating phase of an accident, redistribution of core materials is an important process which can have a dominant effect on the probability of a subsequent energetic excursion. Of paramount importance in determining how fuel material may redistribute is the manner in which the fuel initially disrupts.

The current Sandia fuel dynamics activity is designed to determine the mode of fuel disruption as a function of hypothetical LOF accident conditions and to develop a model which accurately predicts this behavior. These results will be used to improve the current accident analysis codes and to assess the safety of given reactor designs.

The current activity in this area is designed to investigate four aspects of fuel disruption in a loss-of-flow accident. These are:

- (1) mode of disruption
- (2) time of onset of fuel disruption
- (3) rate of fuel dispersal, if any
- (4) effect of occluded fission gas on fuel disruption

Initially a series of nine tests will be conducted in the ACPR in which single pellets of preirradiated fuel (about 5% burnup) in their clad will be heated to melt in order to study the failure process. The test pellets will have microstructures and gas content characteristics of fuel irradiated at local power levels of 4 to 12 kW/ft. The disruption process will be viewed with high-speed motion picture cameras.

This visual observation will produce the first direct evidence of the mode of fuel disruption. Previous evidence has been based only on indirect data such as hodoscope results and postmortem examinations. Recent results from the F1 and L5 tests produced evidence that swelling of solid fuel and frothing of liquid fuel may be the dominant disruption modes. This is in direct contradiction to the so-called "dust-cloud breakup" theory which is the basis for modeling of fuel disruption in

the SAS accident analysis code. Since the mode of fuel breakup can affect the dynamics and rate of subsequent fuel slumping or dispersal, direct observation of fuel disruption in these tests will substantially improve our ability to calculate reactivity effects of early fuel motion.

The time of onset and the rate of fission-gas driven fuel dispersal will be accurately measured in these tests for given ACPR multiple pulses. These are important parameters for advanced reactors which have substantial sodium void coefficients. This is because the outcome of a high-reactivity initiating phase can be very sensitive to small reactivity shifts. If fuel disperses immediately upon melting and at a rapid rate, effects such as a prompt critical excursion caused by FCI-driven sodium voiding (LOF-driven-TOP) may be avoided.

The pellets for the initial test series were cut from EBR II PNL-series pins, radiographed, and are being mounted in the test hardware. The test safety analysis has been completed and hardware assembled for the initial tests in late August. Models of fission gas bubble growth, coalescence, and buoyant motion in liquid fuel are being developed, and will be implemented for analysis of pellet frothing experiments. Later versions will include solid-state bubble dynamics and may explicitly consider the possibility of solid-state fuel breakup.

3. Core Debris Behavior

3.1 Molten Core Technology

3.1.1 Introduction

In the analysis of advanced core disruptive accidents, a number of scenarios lead to the release of molten core material. This molten material is primarily steel and fuel plus the radioactive by-products. The molten material enters the reactor cavity and either directly contacts the cavity boundary or, for sodium cooled systems, fragments upon contact with sodium which might be in the cavity. In the latter case, the debris bed formed could dry out, resulting in molten material contacting the cavity boundary.

The reactor cavity boundary can consist of anything between bare concrete and a fully designed core catcher. The study is currently aimed at cavity boundary designs important in the inherent retention concept. The program is designed to explore other concepts in the future utilizing the technology developed in the near term.

The inherent retention concept is based on a steel-lined concrete structure, possibly including a firebrick insulating material. Evaluations should consider first the retention capability of the concrete alone, followed by the interaction with the steel liner and insulating material. The phenomena of importance include penetration rates, gas evolution species and rates, and material solubility.

3.1.2 High-Temperature Melt/Liner Interactions

One test was performed with a high-temperature, thermitically generated, steel melt contacting a 3/8-inch thick steel plate with a 9 mm thick UO_2 layer. The object is to look at the effect of a UO_2 "crust" on melt penetration. Analysis of this and preceding experiments is underway.

3.1.3 Transient Large-Scale Melt-Concrete Interactions

The transient tests of the interaction between high-temperature steel melts and concrete generic to the Southeastern portion of the United States were conducted during this quarter. Contact between the melt and the concrete was accompanied by the fire and pyrotechnics observed in previous tests. These events were, however, less violent than those observed in the previous tests. Gas evolution vigorously agitated the melt, but did not expel great amounts of material. Little slag formation occurred and that which did form adhered tightly to the metal upon solidification. No concrete spallation was observed. Unlike previous tests, copious amounts of aerosol were formed. Other qualitative features of the tests were similar to those previously observed. These included cracking of the concrete and water flow from the cracks. Instrumentation and detailed results are given elsewhere in this report.

3.1.4 Sustained Large-Scale Melt/Concrete Interactions

Preliminary tests were conducted in the sustained test geometry except that solid iron was in the test cavity. The results demonstrated highly efficient coupling from the induction coils in the concrete to the iron, and also showed that the instrumentation contained within the coils requires additional work to adequately suppress the background noise.

3.2 Debris Bed Studies

3.2.1 Introduction

The objective of the experimental studies of LMFBR debris beds is to provide information on the behavior of reactor materials following a hypothetical core-disruptive accident (HCDA). Activity during this quarter was centered on preparation for, and performance of, the initial in-reactor experiment D-1.

3.2.2 Experiment

The out-of-pile systems test (OPST) was concluded March 25, 1977. The principal objective of the test was to verify the thermal performance of the debris bed experiment package and the helium cooling system prior to commitment to the in-reactor experiment in the ACPR. The test included 35 hours of system operation and training of personnel for round-the-clock operation.

Subsequently, systems were assembled and placed in position for the reactor compatibility experiment (RCE). The RCE was concluded on April 4, 1977.

Experiment D-1 was performed in the ACPR on May 2-3, 1977; 2.43 kg of enriched UO_2 particulate was fission heated, bed temperatures were measured, and the heat was transferred to overlying liquid sodium which was subcooled. Maximum specific power of approximately 1 kW/kg of UO_2 was developed in the bed while it was located in the central

irradiation space of the ACPR operating at a steady-state power of 425 kW. This experiment provided valuable information concerning experimental hardware and instrumentation and demonstrated that internally heated oxide fuel particulate can sustain power loadings and sodium temperatures at a bed loading of 300 kg/m^2 and remain stable. Heat transfer within the bed was primarily by conduction, but a local, single-phase convection cell also developed late in the experiment. Analysis is in progress.

3.2.3 Studies and Analysis

Initial studies have been completed of the degree of dynamic similitude which exists between out-of-pile experiments using materials, such as sand and water, and the LMFBR debris bed or D-series experiments.

3.2.4 Status

Analysis of experiment D-1 results is in progress and preparations for experiment D-2 are underway.

3.3 Molten Fuel Pool Studies

3.3.1 Introduction

The in-core molten fuel pool experiment program is investigating the heat flux distribution and containment ablation potential of molten LMFBR debris materials. Conduct of the studies involves use of real materials under typical temperature and heating conditions.

The major activity this quarter was the first in-core molten fuel pool experiments. Two successful runs were made at ACPR power levels of 150 kW and 600 kW. The ultrasonic thermometry indicated that incipient melting of the UO_2 occurred during the experiment at high power and also provided axial temperature profile data within the UO_2 during the experiments. Preparation of equipment and instrumentation preceded the experiments.

3.3.2 First In-Core Molten Fuel Pool Experiment

Following suitable preparation, the first molten fuel pool experiments were run this quarter. In general, pressure and temperature data were as predicted before the experiment. It was observed that the ultrasonic thermometers gave the only fuel temperature data obtained above 2000°C.

This first in-core experiment represents a major milestone in the molten fuel pool program. The capability for fission-heating UO_2 to the melting point in a controlled manner has been demonstrated. Containment and safety of the experiment under the severe temperature and radiation field conditions encountered have been proven. Subsequent experiments in the fuel-only series will melt a larger portion of the fuel, will further test the capabilities of ultrasonic thermometers, and will provide preliminary temperature gradient data from which to define the heat transfer characteristics of UO_2 at or near the melting point.

3.3.3 Diagnostic Development

The ultrasonic thermometry went through its final design and testing prior to the first in-core experiment. Electronics to provide acquisition and analysis of the signals from two ultrasonic thermometers, each having five sensor elements, were fabricated and tested.

Evaluation of two types of strain gages was accomplished with the result that the performance of one was demonstrably superior to the other in the temperature environments of interest.

3.3.4 Evaluation of Capsule Components

To satisfy a requirement of the Sandia Safeguards Committee, both sizes of tungsten crucibles were tested for cracks or deep flaws. Those failing the tests were disqualified for use. In addition, thermal stress calculations and pressure tests were performed on the inner steel containment vessel. No problems were uncovered.

3.3.5 Status

Two fuel-only experiments are scheduled for next quarter. Additional fuel will be melted and more accurate axial thermal profile data will be obtained using ultrasonic thermometry.

3.4 Containment Analysis

3.4.1 Introduction

This is the first report of a new subtask which will develop a total containment systems model which will examine a variety of possible accident sequences beginning at the point of release from the primary containment vessel and continuing through breach of the secondary containment vessel to development of the radiological release source term. The final model will be sufficiently general to apply to all candidate advanced reactor systems.

3.4.2 Progress

This quarter has been dedicated to activities necessary for initiation of this new subtask including evaluation of related work, review of available codes and models, preparation of work plans, assignment of personnel, hardware design, and fuel procurement.

4. Sodium Containment and Structural Integrity

4.1 Introduction

Any sodium cooled reactor system must consider the interaction of hot sodium with cell liners and given either a failed liner or a core disruptive accident (CDA), the interaction of hot sodium with concrete. The data base available for safety assessments involving these interactions is limited, especially for the concrete and failed liner interactions. This program has been specifically developed to provide the research required to verify the data base from a confirmatory point of view.

4.2 Preliminary Tests

A series of four preliminary tests was planned to verify the adequacy of the experimental facility and instrumentation and to gain experience in its use. The first two have been completed.

4.3 Posttest Results

The test chamber was opened approximately four days after each test and the crucible was removed. A very heavy white oxide layer, consisting of Na_2O dust, covered the entire floor of the chamber as well as the horizontal portion of the splash shield to a depth of 1/8 to 1/4 inch. This layer rapidly degraded (in a few hours) to sodium hydroxide when exposed to the air. Unreacted sodium was evident on the flat surface of the splash shield as well as in globules that had shot up and frozen to the ceiling of the test chamber.

The reaction debris in the crucible, a hard, brittle, rock-like material, was partially chipped away to determine the extent of penetration into the concrete and to analyze the debris.

4.4 Facility Modifications

The purpose of the four tests in the sodium-concrete interaction test program is, in part, to verify the adequacy of the experimental facility. For the first two tests, the facility worked quite well with only a few minor problems. These are being rectified.

4.5 Safety

A fire fighting school was conducted for the operating personnel and for representatives of the Kirtland Air Force Base fire department.

5. Research of Elevated Temperature Design Criteria

5.1 Creep-Fatigue

The testing of uniaxial specimens of 2-1/4 Cr - 1 Mo ferritic steel, for purposes of providing damage material for nondestructive examination,

is in its final stages. The two tests performed to date indicate that the fatigue damage is concentrated in the base metal portion of the straight-sided specimens, but that the strain range in the base metal is approximately twice the nominal strain range.

5.1.1 Nondestructive Examination

Only two of the creep-fatigue specimens have been examined to date. They were damaged to the point of crack initiation, where the tests were terminated. Virtually no change was seen in the longitudinal velocity but the shear velocities displayed the same type of behavior observed in the pure creep specimens.

5.1.2 Microstructural Analysis

Specimens from creep, fatigue, and creep-fatigue damaged samples are being thinned at present and will be microscopically examined shortly.

5.1.3 Position Annihilation

AISI 316 stainless steel specimens have been fatigued at room temperature, over a strain range of 0.9%. One specimen was tested to failure and the others were tested to various life fractions. Results are detailed elsewhere in this report.

5.1.4 Surveillance Test Loop

With both the ultrasonic and position annihilation nondestructive examination methods showing some degree of promise, some initial planning is underway for the surveillance test loop that will be designed to test both methods, and other possible methods, on prototypic hardware.

5.1.5 Multiaxial Creep-Fatigue Facility

Interaction with MTS, Minneapolis, MN, designers is continuing as is analysis of various wall thicknesses and loading combinations. Work on meaningful failure theory, based upon hydrostatic stress and deviatoric deformation, is proceeding.

5.1.6 Creep-Fatigue Damage Analysis

Work has been ongoing to develop a set of alternative rules for Class 1 nuclear components, subjected to creep-fatigue damage, evaluated with strain-range partitioning concepts.

5.2 Creep Buckling Analysis

The Japanese experiment on creep buckling of piping elbows, conducted by the Power Reactor and Nuclear Fuel Development Corporation (PNC), has been analyzed sufficiently, with the simplified elbow element in the MARC general purpose finite element stress analysis program. Sandia and Japanese analyses both overpredict deformation. Additional work is in progress.

5.3 Cell Liner Analysis

Parametric analysis of the steel liners of concrete equipment cells, when subjected to accidental sodium spills, is complete, except for a combined convection/conduction analysis of a jetting-type spill. A report on the cell liner analyses is in preparation.

5.4 General

Several papers and seminars have been presented by personnel associated with this program.

6. Fuel Motion Detection

6.1 Introduction

The purpose of this research is to study various material motion detection schemes, both experimentally and analytically, and to assess their potential as possibilities for existing and future safety test facilities.

In-core techniques that have application to medium- and large-bundle tests in facilities such as SLSF and SAREF are being studied. Sandia studies have shown that this approach may be competitive with high resolution systems from the standpoint of resolution on large-bundle tests.

Several out-of-core, line-of-sight diagnostic schemes are being investigated.

6.2 In-Core Fuel Motion Detection

The main objective of the in-core fuel motion detection (FMD) program at Sandia is to test the feasibility of using large arrays of in-core detectors to monitor fuel motion in medium- to large-scale tests. This objective will be accomplished by mechanically simulating fuel motion in a 37-pin test assembly placed in SPR-III. The program is divided into three tasks: the SPR-III FMD experiment, detector research and development, and the unfolding analysis.

6.2.1 SPR-III FMD Experiment

A purchase order for fabrication of the fuel pin positioning system was initiated. Prior to this, the design and operation was described to the Sandia SPR Committee which appeared to have no reservations about the safety of the design.

6.2.2 Detector Research and Development

Multi-directional efforts in the design and improvement of detectors and their operation have continued. These efforts are detailed elsewhere in this report.

6.2.3 Unfolding and Transport Calculations

The computer code called RHOPER, described in the previous Quarterly was modified to include angle integrations using higher order spherical

harmonic flux moments. Unfortunately, these changes did not remove the differences between the calculated detector response and the predicted response which is approximately two times greater. Thus there is still a discrepancy between the results from perturbation theory and those from direct flux calculations.

6.3 Electron Beam Flash X-Radiography

Fabrication began on the coded source flash x-radiography chamber and experiments are being planned. The evaluation of the effects of photon buildup upon spatial and areal density resolution capabilities in poor geometry situations continued this quarter.

An active detector system is being designed for use in the flash x-ray cinematography fuel motion detection program.

6.4 Gamma-Ray Coded Aperture Imaging in Many-Pin Bundles

Gamma-ray coded aperture imaging technology is progressing very satisfactorily as shown by recent experiments on the Sandia SPR-II and ACPR reactors. The feasibility of extending this technique to many-pin bundles involves the question of how the image quality will be affected by scattering and absorption of the gamma radiation in the fuel, clad, sodium and experiment containment, and how large the signal from the far side and center of the test region will be in comparison to the signal from the near side of the test fuel. A series of gamma-ray transport calculations began during this quarter to study these scattering and absorption effects.

7. ACPR Fuel Motion System

7.1 Additional Results of the Fuel Pin Imaging Experiment at SPR-II

The imaging of a fuel pin beside the SPR-II with an active detector system (which was discussed in the previous Quarterly) represented an important milestone in the ACPR (Upgrade) fuel motion system. Subsequent

measurements revealed that this accomplishment had greater significance than previously realized because the erroneous inclusion of boron in the polyethylene thermalizer near the pin had reduced the neutron fluence and hence the neutron induced pin signal by an order of magnitude. If pure polyethylene had been used the signal to background ratio would probably have been five times larger.

7.2 Collimation Test Module Experiment at ACPR

The collimation test module (CTM) experiment was conducted in May. The pseudoholograms obtained of a fuel pin viewed through a slot in the ACPR core clearly show that a properly designed collimator and shield structure will produce an adequate signal to background ratio. Fuel pin images have subsequently been obtained and the feasibility of the coded aperture system has been demonstrated.

7.3 Detector Resolution Studies

In an effort to better determine the spatial resolution values for the various detectors presently used in the coded aperture imaging system, a series of exposures were made using a sharp edged beam from the ^{252}Cf source. Tests showed that the 0.5 mm and 1.0 mm NaI scintillators have poor spatial resolutions. From discussions with the supplier and our own understanding of the possible causes of the poor resolution, a series of experiments to test the effect of surface polish parameters on resolution have been planned.

7.4 Computer Reconstruction

In an attempt to optimize the computer reconstructions of the SPR-II data, various schemes for dc subtraction prior to reconstruction were investigated. This work is not yet complete, but will continue in the future.

8. Advanced Reactor Safety Research Assessment

8.1 Introduction

This is the first report of a new task directed toward investigation of the applicability of the methodology developed by the Reactor Safety Study (RSS) to a liquid metal fast breeder reactor (LMFBR) or other advanced reactor plants. The RSS methodology will be adapted to construct, display, and evaluate accident sequence diagrams for advanced reactor systems. Analytical tools being considered for use in this assessment include event tree logic, analysis of existing reactor technology and engineered systems, and examination of accident phenomenology.

8.2 Progress

Characterization of the engineered safety systems is sufficiently complete to allow a preliminary list of core and ex-core accident initiating failures to be categorized. Other activities necessary for task initiation have commenced.

1. ACCIDENT ENERGETICS

1.1 Prompt Burst Energetics

(K. O. Reil, 5422; M. F. Young, 5425;
R. M. Elrick, 5422; K. Boldt, 5451)

1.1.1 Introduction

Prompt Burst Excursion (PBE) experiments are designed to provide information on the energetic response of various reactor fuel-clad-coolant systems to superprompt critical excursions. This program is directed at identification and characterization of phenomena resulting in pressure generation and the conversion of thermal energy to work. These experiments examine integral effects of fuel-clad-coolant interactions, fission gas release, and fuel and fission product vapor pressures during superprompt critical core disruptive accidents. The experimental work is closely interfaced with analytical modeling efforts to develop models which uniquely describe the observed phenomena for incorporation into predictive accident analysis. These experiments also provide information about the thermodynamic states and spatial distribution of fuel, clad, and coolant following a superprompt excursion. These data serve as initial conditions for postaccident heat removal studies.

In tests performed to date, single fuel pins have been pulse fission heated in the Annular Core Pulse Reactor (ACPR) to temperatures resulting in fuel vaporization. The pins, which have been surrounded by helium or sodium, are contained in a rigid pressure vessel which is instrumented with thermocouples and pressure transducers and which is fitted with a movable piston at its upper end. Estimates of the conversion of thermal

energy to work result from comparisons of the kinetic energy of the piston to the fission energy input to the fuel. Pressure, temperature, and piston displacement histories are determined for a variety of fuel-coolant systems and initial conditions. A series of experiments with fresh uranium dioxide fuel and sodium coolant is in progress. A series with uranium carbide and sodium will be performed later this year.

During this quarter activities were devoted to the analysis of previous experiments and preparation for upcoming uranium dioxide and uranium carbide experiments. Refinements were made in models in the EXPAND and EPIC computer codes to better describe observed phenomena (particularly, fuel-clad heat transfer and fuel motion resulting from local vapor pressure gradients). A series of dosimetry experiments was completed to quantify the energy deposition in oxide fueled PBE tests. These results of 2350 J/g and 2900 J/g, for 14% and 20% enriched, respectively, are somewhat lower than previously reported values of 2700 J/g and 3200 J/g. The uranium carbide fuel pins for PBE tests were designed and are being fabricated by GFK, Karlsruhe, Germany. The design of the new pressure vessel for the uranium carbide and future oxide tests was finalized and is being fabricated.

1.1.2 PBE Analysis

Activities this quarter have been devoted mainly to computer code debugging and development.

The FCI code EPIC is debugged and running on the Sandia CDC system. It became apparent while doing some preliminary fuel-vapor-driven voiding problems, such as might be encountered in a TOP-PBE, that the present fuel vapor model in the code is inadequate. Only small amounts of fuel

were ejected from the pin into the sodium channel (whatever mass of UO_2 was in the clad rupture nodes); further motion of fuel inside the cladding towards the break, as has been conjectured to occur in the PBE series from postmortem radiographs, was inhibited by the high ejection node pressure. It was felt that the ejection node pressure remains high because the fuel vapor model does not take account of the expansion and consequent evaporation of the fuel material into the sodium channel. This expansion and evaporation could lower the fuel temperature and vapor pressure at the ejection site, providing an axial pressure gradient in the pin that could cause fuel motion from the ends toward the break; the effect of this motion on reactor reactivity would depend, of course, on the axial location of the break. Other important factors determining the fuel motion are the axial power profile, the radial profile in the fuel pin, and the size of the clad rupture. Fuel motion in the pin could conceivably be away from the clad rupture under some conditions.

A fuel vapor model taking the heat of evaporation into account explicitly was formulated to investigate the abovementioned hypothesis; results from this model indicated that the hypothesized lowering of pin break pressure could indeed occur under PBE conditions. A model incorporating a modified version of a procedure due to D. Benson (4167, Sandia Laboratories) for calculating the deviation from saturation pressure, caused by adiabatic expansion, is presently being coded into EPIC.

Some parametric studies were done on the effect of varying the sodium to fuel ratio in the PBE experiment. No noticeable effect was seen although in principle a lower sodium/fuel ratio should enhance the possibility of a strong FCI. The computer results seem to indicate that any FCI would occur several milliseconds after the initial fuel vapor pressure events and would tend to be masked by the fuel vapor pressure.

These results are suspect due to the small amount of fuel ejected by the EPIC fuel vapor model as noted in the preceding paragraphs.

Further work has been done on EXPAND, a pin model used to determine the temperature-stress-strain state of the fuel and cladding in a pin prior to clad failure. The fuel-clad-heat transfer model was modified to include direct molten fuel-clad contact and the gas gap heat transfer model was refined. EXPAND predicts failure in the PBE-5S experiment at 2.8 ms after the ACPR pulse peak, according to the Larson-Miller parameter and life fraction integral, or 4.3 ms after the peak by plastic instability in the clad. The actual failure time inferred from experimental data was 2.8 ms; EXPAND is thus very close in predicting failure time for this type of situation, which has a strong pressure source inside the pin and monotonically increasing loading.

Runs for the multiple pulse case PBE-3S were not as successful; however, failure was predicted at 9 - 7 ms after the second peak by plastic instability, whereas in the experiment failure did not occur. This could indicate several possibilities: (a) code does not handle unloading of the clad correctly; (b) Tresca yield criteria in the code is not accurate for conditions of low internal pin pressure; and (c) other factors in the experiment, such as axial bedding of the pin and support of the clad by the capsule liner.

The multiple pulse case also results in low driving pressure (5 - 10 bar) and formation of a solid fuel crust on the inner clad surface; this seems to indicate that solid fuel particles would definitely be ejected into the channel under pressure for this case, whereas in the single pulse experiments there is a possibility of predominantly vapor being ejected.

The CSQII free piston hydrodynamic calculation reported in the previous quarterly (January - March 1977) was repeated using a 200-bar fuel vapor pressure rather than 63-bar. Results were qualitatively as reported for the lower pressure case except that the bottom/top pressure ratio was 1.57 rather than 1.7, in close agreement with the experimental value (1.52).

1.1.3 Dosimetry Experiments

A series of dosimetry experiments was performed to define the axial energy deposition profile in the PBE tests and to quantify the maximum radially-averaged energy deposition for the UO_2 pins used in previous tests. Fuel pins were irradiated during low-level ACPR pulses. The energy deposited in each pellet was determined by fission product inventory techniques. These results were then scaled by the ratio of measured neutron fluences to yield estimates of the energy deposition in the PBE tests. The validity of the scaling technique was verified by the use of uranium-loaded aluminum dosimetry wire which can be irradiated during maximum ACPR pulses. The use of fluence values for scaling was found to be accurate to within 0.5 percent. The uncertainty in the fission product inventory results is less than 5 percent.

The axial profiles determined by these experiments are shown in Figure 1. The maximum radially averaged energy depositions were found to be 2350 J/g and 2900 J/g for 14% and 20% enriched UO_2 , respectively. (In Figure 1 position 0.0 mm is the top of the fuel column.)

1.1.4 New Pressure Capsule

A new pressure vessel has been designed and is being fabricated for use in PBE experiments. The principal intent of the new design was

Figure 1. Axial Fission Energy Deposition in 14% and 20% Enriched UO_2
Irradiated in the PBE Experiment with 32-mm Thick Polyethylene Moderator

to permit installation of the fuel pin as the last step of assembly rather than one of the first. This is necessitated by future experiments which will involve preirradiated fuel, precluding handwork after installation of the fuel pin. Other minor changes were made to increase confidence in the experimental results. This pressure vessel will be used for the upcoming carbide series and future single-pin tests.

The new design is shown in Figure 2. It is very similar to the original design.² The geometry of the fuel pin test section with the fuel pin surrounded by a molybdenum spacer sleeve contained in an Inconel-718 pressure cell surrounded by beryllium heat sinks is identical in the two designs. Most other features are the same. The differences are:

- (a) Location of thermocouple penetrations.
- (b) Number and orientation of pressure transducers at the top and bottom of the test channel.
- (c) The rupture disc safety head geometry.
- (d) The bottom end closure of the capsule.
- (e) Constraint of the fuel pin.

In the new design the fuel pin is inserted through a port in the bottom of the fuel pin housing as the last step of assembly and is constrained only at the bottom by a spoke locked between the fuel pin housing and the bottom closure. This differs from the original design in which both ends of the pin were constrained. An additional pressure transducer was added at both the top and bottom of the channel to increase reliability of those measurements. The bottom transducers are now normal to instead of coaxial with the fuel pin. The rupture disc safety head was incorporated as an integral part of the piston housing,

Figure 2. New Assembly for Prompt Burst Excursion Experiment

reducing the amount of sodium in that plane and eliminating the possibility of trapping a gas bubble below the rupture disc.

To investigate the effects of different sodium-to-fuel ratios, cylindrical molybdenum spacer sleeves have been designed and ordered. The original fluted tube includes three times the sodium volume per unit height of fuel as will be present in an FFTF or CRBR subassembly.¹ A cylindrical tube with an inside diameter (ID) equal to the minimum ID of the fluted tube (0.350 inches) includes twice the prototypic amount of sodium and is compatible with existing fuel pin designs. A cylinder with a 0.3 inch ID includes approximately the prototypic sodium volume. Use of the smaller cylinder (0.3 inch ID) would require a smaller wire wrap on existing pins and may lead to test channel blockage. Therefore it is intended that only the 0.350 inch ID tubes be used in addition to the original fluted tubes in PBE tests.

1.1.5 Uranium Carbide Fuel Pins

During this quarter an agreement was made with GfK, Karlsruhe, Germany, to provide uranium carbide fuel pins for the PBE experiments and to provide UC pellets for EEOS experiments. The fuel pin, which is shown in Figure 3, was designed to be used in the new pressure capsule design, described above, and thus will be constrained only at the bottom end. The pins will be otherwise similar to the uranium dioxide fuel pins used in previous PBE experiments. The fuel is 15% enriched fresh uranium carbide. The pins will be assembled by GfK with Sandia-provided scratch gages and plenum springs nearly identical to those contained in the oxide pins. A comparison of the UC and UO_2 pins is shown in Table I. The pins (complete except for wire wrap) will be delivered to Sandia Laboratories about August 10, 1977. The spacer wires will be installed

COMPRESSION SPRING DATA

MATERIAL SIZE $\emptyset .64$ ($\emptyset .025$) INCONEL X
 OUTSIDE DIA $4.85 \pm .10$ ($.191 \pm .004$)
 WORKS IN 5.05 (.199) MIN DIA HOLE
 FREE LENGTH 101.6 (4.0)
 SOLID HEIGHT 39.37 (1.55)
 DIRECTION OF COIL OPTIONAL
 TYPE OF ENDS CLOSED & GROUND
 END BEARING 300°
 SQ. WITH AXIS 40°
 LOAD RATE, APPROX. 350 N/M (2 LB/IN)
 REMOVE SET YES
 TYPE OF LOAD STATIC
 TEMPERATURE 538° C (1000° F)
 WORKING LENGTH 44.7 TO 47.2 (1.76 to 1.86)

ITEM 6

NOTES:

1. SCRATCH GAGE AND SPRING TO BE AT TOP OF PIN.
2. SPACER WIRE, 1.34 DIA. (.053 DIA.) TO BE ON 304.8 PITCH (12 INCH PITCH) TENSION TO BE $35.6N \pm 2.2N$ (8 LBS $\pm .5$ LBS).
3. ALL MACHINE SURFACES TO BE $1.6 \mu m$ ($\sqrt{63}$) OR BETTER.
4. FUEL PIN ASSEMBLY, LESS WIRE WRAP AND SPOKE, ITEM 1, SHALL BE STRAIGHT WITHIN $.51$ (.02) OVER ENTIRE LENGTH WHEN LYING UNRESTRAINED ON A SURFACE TABLE AND ORIENTED TO GIVE MAXIMUM GAP BETWEEN TABLE AND FUEL PIN ASSEMBLY.

5. ITEMS 6 AND 7 TO BE PROVIDED BY SANDIA LABORATORIES.
6. EACH VACUUM ASSOCIATED WELD SHALL HAVE NO DETECTABLE LEAK WHEN TESTED WITH A MASS SPECTROMETER TYPE LEAK DETECTOR AT A SENSITIVITY OF 1×10^{-7} CC/SEC STP WITH A CONCENTRATION OF 100% HELIUM AROUND EACH WELD.
7. IDENTIFICATION OF EACH PIN (SLC 15.1, SLC 15.2, SLC 15.3, etc.) TO BE MARKED PER 9919100 CLASS C1 OR E1.
8. WELD AND INSPECT PER 9912119 CLASS II USING COVERED ELECTRODES PER ASTM A 298, CLASS E 309-15 OR E 309-16.

9. DIMENSIONS ARE IN MILLIMETERS WITH INCHES IN PARENTHESES.

10. WIRE WRAP TO BE DONE AT LASL.

11. SEALED PIN TO CONTAIN PURE HELIUM AT ATMOSPHERIC PRESSURE AT ROOM TEMPERATURE.

12. THE SCRATCH GAGE WILL BE LOADED AFTER WELDING THE BOTTOM END CAP AND SPOKE AND LOADING THE FUEL AND INSULATION PELLETS. THE INSTRUCTIONS ARE AS FOLLOWS:

1. Slide scratch gage into cladding until the stylus is contacted.
2. Orient scratch gage such that the stylus points in the same direction as the spoke on the bottom cap.

3. Depress stylus manually so that scratch gage may be further inserted into cladding. (Release stylus.)
4. Insert scratch gage using SLA-provided tool until insulator pellet is contacted (a scratch will occur on the clad during this operation).
5. Rotate the scratch gage 180° so that stylus points in opposite direction of insertion and remove insert tool.
6. Complete the assembly of the spring, top end cap and wire wrap.
7. Align slot in top end cap with spoke on bottom end cap before welding.

Figure 3. Drawing of Uranium Carbide Fuel Pin for PBL-SG Series of Experiments

TABLE I
Comparison of UC and UO₂ Fuel Pins for PBE

Fuel Material	UC	UO ₂
Uranium Enrichment (%)	15	14 + 20
Fuel Mass (g)	80	64
Pellet density (% + D)	84	92
Pellet diameter (mm)	4.96	4.93
Fuel length (mm)	362	344
Total pin length (mm)	556.3	498.3
Fuel-clad bond	Helium	Helium
Plenum length (mm)	90	90
Spring solid height (mm)	39.4	39.4
Cladding material	1.4970 SS	316 SS
Cladding outside diameter (mm)	6.00	5.83
Cladding thickness (mm)	.38	.38
Wire wrap diameter (mm)	1.35	1.39

at Los Alamos Scientific Laboratory. DTF-IV predictions of energy deposition in the carbide fuel pins as a function of polyethylene moderator thickness are shown in Figure 4. Radial fission energy distributions (from DTF-IV) for 15% enriched UC and 14% enriched UO_2 are shown in Figure 5. These results were normalized to yield the measured values for 14% and 20% enriched UO_2 with 1.25-inch thick polyethylene moderators.

1.1.6 Transient Rod Upgrade

To increase the energy deposition in PBE experiments during multiple pulse operations work has been undertaken to increase the worth of the ACPR transient rod bank.

The increased worth transient rods ("mini-upgrade" rods) for the ACPR have been ordered and their fabrication is nearly complete. The "mini-upgrade" rods are two fast transient rods that will replace the present fast transient rods and have a reactivity worth approximately 15 percent greater than the present rods. The "mini-upgrade" will increase the transient rod bank worth from the present \$4.50 to an expected \$4.95. The increase in the rod worth is accomplished by using 93% ^{10}B -enriched B_4C as the neutron poison material in the rods. A sufficient number of ^{10}B -enriched B_4C pellets for two transient rods have been hot pressed by Los Alamos Scientific Laboratory to densities of 93.0 percent theoretical density (or approximately 2.36 g/cc). The pellets are presently being clad to transient rod specifications by General Atomic. An experiment plan for the installation and testing of the "mini-upgrade" rods in the ACPR is currently being reviewed by safety review committees.

Figure 4. Radially Averaged Energy Deposition versus Polyethylene Thickness for 15% Enriched UC During ACPR Maximum Pulse

Figure 5. Radial Fission Energy Deposition Distribution for 14% UO_2 and 15% UC with 1.25" Polyethylene Moderator During Maximum ACPR Pulse

1.1.7 Pressure Gage Calibration

1.1.7.1 Introduction -

In the PBE experiment, pressure gages are a primary source of information for following the energy exchange between the vaporized fuel and the sodium coolant. The purpose of this study is to estimate the relation between pressure gage response and source pressure, modified by the geometry of the experiment.

Modification of the pressure pulse as it travels through the sodium filled annulus will be studied out-of-pile in the PBE geometry using a well-defined shock wave as the pressure source. The shock wave will be produced by detonating high-explosive in water to improve the impedance match into the PBE system.

This study is a simplification of events in the PBE experiment since here the pulse propagation is uncomplicated by sodium voiding caused by injection of the fuel vapor. This simplification should allow a well-defined source pressure to be unfolded from the measured pressure and permit some estimate of the motion of the effective mixing front.

Within the PBE system, the pressure pulse is measured at three positions along the annulus including those gage positions in the PBE experiments. At two of the positions, the response of the normally used eddy current gage will be compared with that of a piezoresistive gage.

1.1.7.2 Progress -

A second proof test was run in a study to generate a shock pulse in water at least as long as a PBE pulse (~ 1 ms) and

with a similar amplitude (~ 5000 psi). Using 12 kg equivalent of TNT, a 4 ms wide pressure pulse with a 5000 psi spike lasting 100 μ s was produced 10 feet from the detonation. Pressure relief in the initial spike was probably due to tank rupture.

A thicker-walled reusable tank is being installed upright in the ground to produce longer duration water shocks with smaller amounts of explosive. Adequate pulse amplitude should be obtained by placing the gage calibration hardware closer to the explosive. The 4-foot diameter reinforced pipe with 1-1/2 inch steel walls is being placed in an existing 10-foot diameter pipe facility. The armor plate base of the facility will be used as the new tank bottom. The space between the pipes will be reinforced with concrete.

The gage calibration hardware has been assembled and checked. Pressure data from the last proof test is being examined for Fourier analysis.

Next quarter, design and fabrication of a support for the calibration hardware will begin as soon as the test facility is ready for interface design. At least one series of shock propagation measurements using glycerine in the PBE system will be made in the next quarter.

1.1.8 Status

Eight PBE tests are planned during the next quarter. These are summarized in Table II. Components for these tests are being fabricated or purchased. The UC fuel pins are expected in early August with the wire-wrap to be completed before the end of August. The increased-worth

TABLE II
PBE Experiments Through ACPR Shutdown

Designation	Fuel Material and Enrichment	Pulse Mode	Moly Spacer Sleeve	Capsule	Target Date	Comments
PBE-8S	UC ₂ 20% (IASL)	Max. Multiple Pulse ^(a)	0.350" ID Cylinder ^(b)	Original Design	Aug. 1977	
PBE-9S	UC ₂ 20% (IASL)	Max. Multiple Pulse ^(a)	0.350" ID Cylinder ^(b)	Original Design	Aug. 1977	Repeat of PBE-8S
PBE-SG1	UC 15% (GfK)	Max. Single Pulse	Fluted ^(c)	New Design	Sept. 12, 1977	1500 J/g peak average reduced poly. thickness
PBE-SG2	UC 15% (GfK)	Max. Single Pulse	Fluted ^(c)	New Design	Sept. 14, 1977	2500-2700 J/g
PBE-SG3	UC 15% (GfK)	Max. Single Pulse	Fluted ^(c)	New Design	Sept. 16,	2500-2700 J/g
PBE-10S	UO ₂ 20% (IASL)	Max. Multiple Pulse ^(a)	Fluted ^(c)	Original Design	Aug. 1977	Repeat of PBE-8S with fluted tube
PBE-11S	UO ₂ 14% (HEDL)	Max. Single Pulse	0.350" ID Cylinder ^(b)	New Design	Aug. 1977	Repeat of PBE-5S with straight tube
PBE-12S	UO ₂ 14% (HEDL)	Max. Single Pulse	Fluted ^(c)	New Design	Aug. 1977	Repeat of PBE-5S with new capsule

(a) Depends on transient rod upgrade.

(b) ID equal to minimum ID of fluted tube (Na Vol. = 2 times prototypic).

(c) Na Vol. of fluted tube = 3 times prototypic.

TABLE II (Cont'd.)
FBE Experiments Through ACPR Shutdown.

Designation	Fuel Material and Enrichment	Pulse Mode	Moly Spacer Sleeve	Capsule	Target Date	Comments
PBE-8S	UO ₂ 20% (LASL)	Max. Multiple Pulse (a)	0.350" ID Cylinder (b)	Original Design	Aug, 1977	
PBE-9S	UO ₂ 20% (LASL)	Max. Multiple Pulse (a)	0.350" ID Cylinder (b)	Original Design	Aug, 1977	Repeat of PBE-8S
PBE-SG1	UC 15% (GfK)	Max. Single Pulse	Fluted (c)	New Design	Sept. 12, 1977	1500 J/g peak average reduced poly. thickness
PBE-SG2	UC 15% (GfK)	Max. Single Pulse	Fluted (c)	New Design	Sept. 14, 1977	2500-2700 J/g
PBE-SG3	UC 15% (GfK)	Max. Single Pulse	Fluted (c)	New Design	Sept. 15, 1977	2500-2700 J/g
PBE-10S	UO ₂ 20% (LASL)	Max. Multiple Pulse (a)	Fluted (c)	Original Design	Aug, 1977	Repeat of PBE-8S with fluted tube
PBE-11S	UO ₂ 14% (HEDL)	Max. Single Pulse	0.350" ID Cylinder (b)	New Design	Aug, 1977	Repeat of PBE-5S with straight tube
PBE-12S	UO ₂ 14% (HEDL)	Max. Single Pulse	Fluted (c)	New Design	Aug, 1977	Repeat of PBE-5S with new capsule

(a) Depends on transient rod upgrade.

(b) ID equal to minimum ID of fluted tube (Na Vol. = 2 times prototypic).

(c) Na Vol. of fluted tube = 3 time prototypic.

transient rods are expected to be completed by mid-July. The internal safety committee reviews should be completed in mid-July.

1.2 Equation of State

(D. A. Benson, 5167; E. Gorham-Bergeron, 5425;
K. O. Reil, 5422; J. E. Smaardyk, 5167; B. M. Butcher, 5167)

1.2.1 Present Status

Information for vaporizing UO_2 at temperatures and vaporization rates appropriate to fast reactor safety analysis is now becoming available from several different types of experiments. It has been commonly assumed in fast reactor analysis that the vaporization of fuels at high temperatures can be described by an equation of state for material in chemical and thermal equilibrium. However, comparisons made during the last quarter of dynamic experiments on oxide material to equilibrium measurements show contradicting results. These observations can be resolved by a rate dependent effect in the high temperature oxide.

High temperature information for UO_2 can be derived from several types of experiments. Traditionally, measurements of the heat capacity and heat of vaporization have been used to extrapolate high temperature behavior. Since these values are derived from long duration experiments, we refer to them as equilibrium models of the high temperature properties. Second, a series of dynamic experiments under more prototypic transient conditions are now available, the most recent of which are the ACPR experiments described in the last quarterly report. Such experiments directly describe the relation between internal energy and vapor pressure which is needed for fast transient calculations. These experiments will now be reviewed.

Two groups have reported UO_2 vapor measurements for pressures from a few atmospheres to 65 atmospheres using laser induced heating of sample surfaces.^{11,12} These data relate temperature and pressure. Heat capacity information is needed to place this information in a dynamic energy-dependent format. Vaporization experiments which use an electron beam to heat UO_2 have produced data¹³ directly in a dynamic format with heating times of the order of 1 microsecond. These data are shown in Fig. 6. Also shown are the results of Reil¹⁴ generated by fission heating of UO_2 over millisecond times. Uncertainties in thermal losses during the longer time interval used for his work have led to indicated bounds for the vapor saturation curve (dashed lines on Fig. 6).

A third type of dynamic experiment that contains saturation curve information is a full fuel pin test. In these experiments the pressure of a coolant channel surrounding the pin is measured as the cladding ruptures during a reactor heated transient. The pressure and peak energy density of the fuel can be determined and are assumed to be dominated by the fuel vapor pressure. Results from the 5S test of the PBE¹⁵ series and the TREAT¹⁶ (S-11) experiments with unirradiated UO_2 pins are shown in Fig. 6. The error bars on the data indicate the authors' quoted uncertainty in energy scale.

If limited thermal loss is assumed for the measurements of Ref. 14 the data from the dynamic experiment shows agreement to within the measurement uncertainty. Also, if one uses the measured heat capacity of Kerrisk and Clifton¹⁷ and Leibowitz¹⁸ and the vapor pressure curve described by Booth¹⁹, a curve may be constructed to compare with the dynamic experiments. This is labeled as Model 2 on the figure. Model 2 differs by more than a factor of two in pressure or nearly 15 percent in energy from the dynamic experiments. A difference this large apparently cannot be explained by the stated uncertainties in the heat capacity or by reasonable

Figure 6. Comparison of Dynamic Vaporization Data and Models for Uranium Dioxide

(The vapor pressure as a function of internal energy, relative to energy at room temperature conditions, is shown. As discussed in the text, the dynamic vapor data among the collection of experiments are consistent but differ significantly from the equilibrium properties of Model 2.)

variation of the heat of vaporization. However, a rate-dependent effect can explain the differences, since the heat capacity values are from long duration measurements while the other information is derived from fast transient measurements.

The large increase of the excess heat capacity near melt is proposed as a source of this rate-dependent effect. Bergeron²⁰ has proposed diffusion arguments which prevent the excess heat contribution to millisecond duration experiments. To illustrate the importance of this effect, Model 1 is calculated using a heat capacity (described in Ref. 13) which omits the peak in the heat capacity curve, together with the vapor pressure versus temperature curve from Model 2. Acceptable agreement is found between this model and the dynamic data. This heat capacity model also makes the laser heating experiments of Ref. 11 and 12 consistent with the dynamic data.

In summary, a dynamic equation of state based upon several different transient experiments can be defined. Significant differences between these results and equilibrium data are found which suggest the existence of a rate-dependent process in high temperature UO_2 . The magnitude of these differences emphasizes the importance of directly confirming this rate-dependent effect and measuring the appropriate rate constant.

1.2.2 Electron Beam Vaporization Studies

Recent progress in development of the long pulse mode of REBA is extremely encouraging and installation of experiments and preliminary diagnostic measurements began during the last month. Up to 700 cal/g operating energy is currently available, which should be sufficient for

additional experiments on pure unirradiated UO_2 and UO_2 -fission product simulant mixes.

1.2.3 Equation-of-State Analysis

Efforts this quarter have concentrated on three main areas of equation-of-state analysis.

The anharmonic heat capacity calculation described in the previous quarterly report has been modified to include an effective field calculation of UO_2 phonon modes. This modification has required that we improve an existing phonon code for fluorite structure materials. As a result some delays in the completion of the calculation have occurred, however, with the anticipated effect of significant improvements in the accuracy of the resulting calculations.

A second area of interest has been the possibility of using various theories of non-equilibrium evaporation (i.e., forced congruent evaporation and/or surface area limited evaporation) to explain fuel motion in the Sandia PBE tests. Although we now believe that such effects alone will not totally account for the observed voiding at the ends of the fuel pin, ways to incorporate such effects into EPIC are being explored.

Finally, investigations into ways of modeling rate-dependent effects in the heat capacity of UO_2 have continued with a view toward analyzing the results of a planned UO_2 resistive heating experiment.

1.2.4 ACPR, SPR-III Equation-of-State Measurements

The ACPR measurements on pure unirradiated UO_2 are now substantially complete. Extensive further measurements at this time are not

anticipated because of the heavy usage of ACPR prior to shutdown. However, hardware is under development for several measurements on a carbide fuel if the schedule permits.

In order to avoid any lapse in the equation-of-state program during the ACPR shutdown, we propose to transfer our experiments to the SPR-III reactor. During the last quarter a proposed experimental procedure and plan for the initial phase of vapor pressure experiments for SPR-III was reviewed and approved by the Sandia SPR-III Experiment Review Committee and the Sandia Reactor Safeguards Committee without significant modification. In this first phase, fully enriched uranium dioxide samples (not to exceed 1 g) will be pulse heated to give transient pressure excursions of up to 650 atmospheres (700 cal/g energy).

A diagram of the SPR-III vaporization experiment package as mounted in SPR-III is shown in Fig. 7. The apparatus consists of a pressure instrumented experimental vessel (at the center) thermally isolated from an outer container, which, in turn, is enclosed by a polyethylene moderator cylinder with end caps.

The pressure instrumented experimental vessel for SPR-III differs in several ways from the configuration used in ACPR. First, instead of measuring just the vapor pressure state associated with a fixed volume in each experiment, the SPR-III capsule has a piston gage below the sample as one of the sample boundaries. The motion of the gage will be monitored with respect to time by a linear velocity transformer. Thus, in addition to measurement of the initial state of the sample after pulse heating, continuous measurement of the expansion states of the material during the experiment will be attempted as the vapor drives the piston downward.

Figure 7. Diagram of the Vaporization Experiment Package Mounted in SPR-III

A new calorimeter concept which will measure more directly the energy absorbed by the sample is another innovation in the SPR-III experiment. These measurements will be cross checked with results from the fission product inventory technique applied to the sample and with uranium-aluminum fission wires mounted externally to the vessel.

As part of the initial feasibility study, a series of TWOTRAN neutron transport calculations was made for the experimental configuration. These results, coupled with dosimetry measurements on SPR-III using simulated capsules, suggest that more than enough energy is available for experiments on UO_2 . The first experiments are planned for late summer.

1.2.5 New Experimental Directions

An experiment has been designed to measure the dynamic heat capacity of UO_2 . We propose to rapidly heat a UO_2 sample resistively to temperatures above 1500 K, where standard long-time calorimetry methods have shown a rapid increase in the heat capacity. Special precautions will be taken to avoid instabilities due to the increase of the electrical conductivity of UO_2 with increasing temperature. Once the final energy state is reached, changes in specific heat with time and/or heating rate can be detected from changes in the sample temperature. Circuit and hardware design for these experiments has been completed and the components are being fabricated.

1.3 Transition Phase

(S. W. Eisenhower, 5422; R. W. Ostensen, 5425)

1.3.1 Introduction.

Following the loss of initial core geometry in a loss-of-flow accident, and assuming that neither early shutdown nor rapid hydrodynamic disassembly takes place, the accident enters the transition phase. At this point one or a number of subassemblies have melted down and radial propagation takes place as subassembly hexcans are melted through. The progression of the transition phase can strongly influence the severity of the accident, and it is therefore important to understand the mechanics of the accident stage in detail. In particular, several key questions need to be answered. First, in the early stages of the accident is the fuel dispersive, that is, how far can the molten-fuel and fission-gas mixture move axially before it freezes? Secondly, assuming that blockages form and the core is bottled up, what is its configuration - does it remain boiled up or does the pool collapse and a recriticality take place? Answers to these questions depend upon resolving the uncertainties in transition phase phenomenology. Work at Sandia has recently begun on the mechanics of fuel blockage formation and boiling pool behavior. Both analytic and experimental studies are included in this effort.

1.3.2 Fuel Blockage Formation and Crust Stability - Analysis

Early efforts in this area were initially directed at evaluating the previously proposed fuel freezing mechanisms: conduction,²¹ bulk²², and ablation-induced freezing²³. In conduction-controlled freezing, a frozen layer grows as $t^{1/2}$ until the entire flow channel is plugged. For the UO_2 -steel system, penetrations on the order of 10^4 cm are predicted. This is in contradiction to experimental data where penetrations on the order of 10 - 100 cm are observed.²⁴ The second two models are quite similar and are based upon the premise that fuel crusts on

molten steel cannot be stable. These models predict fuel penetrations which are two (bulk freezing) to six (ablation freezing) times shorter than the experimentally observed values. The mechanism which produces the necessary crust removal, and hence rapid freezing, is still unclear.

Epstein, et al.,²³ have proposed that crust breakup occurs when the molten steel thickness exceeds the depth of the laminar steel sublayer for the flowing fuel. The fuel crust is treated as a thin membrane which does not affect the radial velocity profile. This membrane is assumed to collapse when subjected to turbulence in the buffer layer. In most cases this model predicts very rapid crust failure. Hence it does not appear to explain the discrepancy between the observed and predicted penetration distances. For instance, for a fuel temperature of 3000°C , an initial wall temperature of 800°C and an initial fuel velocity of 1500 cm/s, crust breakup in 46 microseconds is predicted. At this point the molten steel depth is 2.8 microns while the fuel crust is 6.6 microns thick. In fact only for very high initial fuel temperatures does the steel depth exceed the crust thickness at the instant when membrane failure is predicted. This suggests that treating the crust as a thin membrane on molten steel may be questionable and that other crust removal mechanisms should be considered.

One such model is failure of the crust due to thermal stresses. This may have important implications not only in blockage formation but also in the ability of a molten core to remain subcritically dispersed by boiling steel. If oxide crusts on the surrounding steel walls are not stable, the driving temperature drop for convective heat transfer is the difference in temperatures between the pool and the melting point of steel. This temperature drop is so large (1400 K or more) that convective heat losses may be too large to permit sufficient vapor

generation to maintain subcritical boilup. Thus the existence of a stable fuel crust may be of key importance for the existence of a stable, subcritical core boilup.

The effect of thermal stress on the failure of mixed oxide crusts has been examined. Consider a thin crust which is infinite in the transverse direction. The hot side of the crust is at its melting point and is in contact with molten fuel. The cold side is in contact with molten steel. It is assumed that the crust is not free to relieve any thermal stress by bending. This is because such bending would force part of the crust away from the wall, which implies failure of the crust in order to permit molten fuel to flow into the space opened by the crust bending. The thermal stress, σ , may then be found from

$$\sigma = \alpha E(T - T_0)/(1 - \mu) ,$$

where α is the coefficient of thermal expansion, E is Young's modulus, μ is Poisson's ratio, and T and T_0 are the fuel temperature and a reference temperature, respectively. The constants α and μ are assumed to be temperature independent. If the temperature profile is assumed to be linear and E is a linear function of temperature, going to zero at the melting point, then the unknown reference temperature may be eliminated and the stress distribution found.

Since the crust is free to expand or contract freely in the transverse direction, there can be no net traction across the cross section of the crust, viz.,

$$\int_0^h \sigma(z) dz = 0 ,$$

where h is the crust thickness. The reference temperature, T_0 , is then given by

$$T_0 = T_M - 2\lambda h/3 ,$$

where T_M is the melting point of the fuel and λ is the temperature gradient (dT/dz) across the crust. The thermal stress distribution is then given by

$$\sigma = \frac{\left| \frac{dE}{dT} \right| \alpha (\delta T)^2}{1 - \mu} \left(\frac{2}{3} - \frac{z}{h} \right) \frac{z}{h} ,$$

where δT is the temperature drop across the crust.

With the hot crust boundary at the melting point, the maximum tensile stress occurs at the cold crust boundary. If this stress is equated to the failure stress, σ_u , of mixed oxide, a maximum temperature drop, δT , is found for which cracks will not propagate:

$$\delta T = \left[3(1 - \mu) \sigma_u / \left(\alpha \frac{dE}{dT} \right) \right]^{1/2} .$$

For a coefficient of expansion of $2 \times 10^{-5} \text{ K}^{-1}$, an ultimate strength of 206 MPa and a temperature derivative of Young's modulus for mixed oxide fuel of 150 MPa/K, the maximum uncracked crust thickness has a temperature drop of only 330 K. Crusts can exist with a larger temperature drop, but fracture of the crust should occur on the cold side with cracks propagating part of the way through the crust. The cracks should end in the region where the temperature is 330 K below the fuel melting point. Clearly, the strength of the crust will be determined by the thickness of the uncracked region.

If a crust is formed when molten fuel contacts cold steel, the temperature drop across the crust will be roughly the difference in melting points between fuel and steel (about 1350 K). If the steel surface is solid, it can support the crust and may prevent thermal stress induced failure. However, if the steel surface is molten, the crust will partially fail. Cracks from the cold side will penetrate

about 3/4 of the crust thickness, leaving only 1/4 of the crust to withstand mechanical stress. For two reasons, this is probably a low estimate of the degree of cracking. First, the very rapid formation of the crust should make it weaker than normal fuel. Second, the use of a Young's modulus which goes to zero at the melting point is an idealization used for extrapolating data from lower temperatures. Since very low values of the Young's modulus allow accommodation of strain with a low stress, a non-zero Young's modulus at the melting point would result in higher stress and therefore deeper cracking.

The cracking of at least 3/4 of the oxide crust thickness should lead to a more rapid breakup of insulating crusts as fuel is ejected from the core. Thus, in fuel freezing and blockage formation analyses, thermal stress cracking must be considered. In addition, care must be used in extrapolating from results based on simulant materials since these might not exhibit the same behavior under thermal stress as mixed oxides do.

The effect of crust cracking is very important for a crust which is insulating a pool of boiling fuel and steel from the surrounding cold steel walls. The thickness of this crust, L , is given by

$$L = (\sqrt{1 + 2k\delta TQ/q^2} - 1)q/Q ,$$

where k is the thermal conductivity (about 300 W/m² K), Q is the volumetric power in the crust, and q is the total heat flux to the steel. At full power in an FFTF subassembly, Q is about 1500 MW/m³ and q is about 15 MW/m². This leads to a crust thickness of 0.26 mm. Since 3/4 of this crust thickness will be cracked, the effective crust thickness for mechanical strength considerations is only 63 μ m. Such a weak crust could not survive in a highly turbulent boiling-pool environment. The

crust thickness would be about the same for a pool of 100 subassemblies at 10 percent power. Since this is not much greater than initial decay heat power (6% of nominal), stable crusts can probably not exist in full core pools. Without crust insulation, a full core pool could lose too much heat by convection to remain boiled up. This may lead to a recriticality as the pool subsides. At higher power levels, of course, the pool may boilup even with high convective heat losses. For a single subassembly pool at initial decay heat power levels the uncracked part of the crust would be about 1 mm thick. Thus, for single subassembly pools, the insulating crust may be stable at low power.

Along with phenomenological modeling, it is essential to accurately predict the development, meltthrough, and reformation of blockages. As a first step toward this goal, existing computer programs which are suitable for such calculations are being obtained and evaluated. The SIMMER-I code²⁵ is operative at Sandia and its use is being examined. Efforts here are being coordinated with the SIMMER verification effort. In addition the ANL fuel freezing code²⁶ will be available shortly and evaluation of this code is planned for the near future.

1.3.3 Fuel Blockage Formation and Crust Stability - Experimentation

In order to evaluate current models and to assist in the development of new phenomenological models, experiments with simulants are being planned. Selection of suitable wall-fluid combinations is complicated by the fact that not only thermal properties (such as thermal diffusivities, latent heats, Prandtl numbers, etc.) but also mechanical properties (such as modulus of elasticity and ultimate stress) must also be considered. A short computer code has been written to evaluate candidate materials based upon available freezing and crust stability

models. Early experiments may be done with ice as the wall simulant and cyclohexane as the simulant fuel.

These tests, and later, more advanced experiments, will be run using a variety of initial conditions. Test evaluation requires adequate instrumentation, and planning for this has begun. The models predict key times (onset of wall melting, crust failure and blockage formation, and rates), steel melt, crust formation, and ablation. To differentiate between the models, dynamic (time-dependent) measurements are required. That is, one needs to know temperature, pressure, and velocity as functions of time and position. It is also desirable to know the crust thickness, and, following crust removal, the wall ablation rate. Instrumentation will consist of thermocouples in the wall and fluid, at various radial and axial locations, pressure transducers and inlet and outlet flowmeters. When possible high speed motion pictures will be made. Measurements of ablation rate may be made if an electrically conductive wall is used. As the wall ablates the resistance across it changes, and when corrected for temperature effects a direct measurement of material loss is possible. By segmenting the wall, axial variations in ablation rate can be obtained.

Progress continues on planning for the large-scale 19-pin freezing experiment using $\text{UO}_2\text{-ZrO}_2$ thermite or possibly molten UO_2 . This effort was described in the last quarterly.¹⁰

1.3.4 Boiling Pool Phenomenology

Although the primary effort is directed at the mechanics of blockage formation, work on boiling pools is also being done. This includes the work on crust stability reported on above. Experiments using a

1-kW microwave oven are planned. Water or possibly a combination of two immiscible liquids will be used. The capability of suddenly changing the heat transfer rate at the wall, simulating fuel crust removal, may also be included. By varying the power level and observing the pool height and void distribution, the effects of flow regime can be investigated. These experiments should also provide valuable data for SIMMER model validation.

The possibility of performing in-pile, boiling pool experiments on ACPR was reported in the last quarterly.¹⁰ These studies are continuing.

1.4 Aerosol Source Normalization - In-Pile Studies

(R. M. Elrick, 5422)

1.4.1 Introduction

For those reactor accident scenarios in which the fuel vaporizes and ruptures the cladding, it is necessary that the state of the fuel before and after rupture be characterized by experiment. The first phase of this study conducted in FY 77 will measure some of the physical properties of UO_2 fuel pins heated to vaporization by neutrons from a pulsed reactor. Succeeding work will study the interaction of neutron-vaporized fuels with the containing structure and with sodium.

Debris properties of the UO_2 fuel, such as particle size, velocity, and density distributions as a function of time, will be measured for both bare and clad fuel pins. This characterization will: a) be used as a particle source term for aerosol codes; b) be sufficiently accurate and complete for comparing it with hydrodynamic code predictions; and c) be compared with those particles sampled by similar methods in

relevant out-of-pile experiments at Oak Ridge National Laboratory in which the fuel material is resistively heated. This comparison of in-pile to out-of-pile techniques will begin to establish limits of interest on the particle density and size spectra of the vaporized fuel.

A complete description of the experimental design was presented in the October-December 1976 quarterly.⁹

1.4.2 Progress

Orders for all fabricated parts and hardware were completed during this quarter for the subsystems of the experiment: a) the experimental canister containing the UO_2 fuel rod and sampling wheel; b) the passive pumping system to maintain a pressure of 10^{-3} torr in the canister; and c) the optical train for photographing the fuel disrupt event. In addition to the fuel and sampler, the canister also contains the sampler motor and speed monitoring transducer, a lamp and mirrors for illuminating and photographing the fuel, a thermocouple vacuum gage for monitoring the canister pressure, electrical connections to the fuel element to measure its resistance by a current-voltage measurement, and a thermocouple to measure fuel temperature before the experiment begins.

The experimental canisters have been assembled and function properly. They are in the process of being vacuum and pressure tested. Sampling surfaces in the wheels are being coated and the wheels are being assembled and aligned.

An experimental plan was prepared for presentation to the Sandia ACPR and Safeguards Committees on reactor safety.

Next quarter the completed system will be assembled in the ACPR for a series of tests to sample and photograph debris from bare UO_2 fuel enriched to 20% and 30%. The neutron energy dose to the fuel will be about 700 cal/g deposited in ~ 5 ms, FWHM.

1.5 Fuel Coolant Interaction

Work on this portion of the LMFBR Safety Research program is not scheduled to begin until FY 78. No activity is reported this quarter.

1.6 Large-Scale Test Capabilities

(H. C. Monteith, 5411; W. E. Nelson, SAI;
D. H. Nguyen, 5425; J. S. Philbin, 5452)

1.6.1 Optimum Nuclear Pin Bundle for Large-Scale Testing

Because in-pile testing with full LMFBR bundle size requires large and costly facilities, it is desirable to determine the minimum test scale satisfying prototypic requirements. An analytic tool capable of tracing the complete post-failure history for each pin in a true bundle geometry is not currently available. This study attempts to establish prototypic prefailure thermal hydraulic conditions within the test vehicle, arguing in doing so that they have determinative effects on post-failure events. These conditions include coolant hydrodynamics, wire-wrap effects on transverse momentum, hexcan edge effect and enthalpy transport between flow channels via turbulent and diversion crossflow and mixing. The conclusions of this study, based on single-phase hydrodynamics have immediate applications to cases without prefailure boiling, such as superprompt transients and slower TOP's in irradiated fuel.^{27,28} For cases with prefailure boiling, it is believed that achieving proto-

typic thermal-hydraulic conditions in the bundle at the onset of pin failure is prerequisite to attaining a prototypic course of events after failure.

The computer program COBRA4I,²⁹ the latest version of the COBRA series, is used in this study. New storage compaction techniques and an iterative solution procedure enable COBRA4I to handle the full 217-pin wire-wrapped asymmetric bundle. The code's hydrodynamics package takes into account the heat transport between channels via turbulence and wire-wrap diversion crossflow and mixing, as well as the edge streaming effect. Unfortunately, the code currently cannot obtain the solutions for the diversion crossflow with sodium boiling. It is believed, however, that little is gained with this capability, unless the code can also trace the post-boiling history for each pin, separately, including the determination of pin performance limits.

Several hydrodynamics and heat transfer empirical correlations are programmed into COBRA4I, the correlations' constants remaining as inputs. In particular, this study uses a "smooth" friction factor for all channels,³⁰ neglecting the friction multiplication factor:

$$f_s = 0.316 \text{ Re}^{-0.25} ,$$

where Re is the Reynolds number. The sodium film heat transfer coefficient is given by the relation³¹

$$h = \frac{k}{D} \left[0.015 \text{ Re}^{0.86} \text{ Pr}^{0.8644} + 4.98 \right] ,$$

where k is the thermal conductivity, D is the hydraulic diameter, and Pr is the sodium Prandtl number. The single phase turbulent mixing used in this analysis is of the form:

$$\omega' = \beta s \bar{G} ,$$

where ω' is the turbulent crossflow, β is the turbulent mixing input parameter, s is the subchannel gap spacing and $\bar{G}$ is the average mean flow rate in adjacent subchannels i and j :

$$\bar{G} = (m_i + m_j)/(A_i + A_j) .$$

The general pin characteristics, shown in Table III are those of CRBR, with clockwise wire-wrap and stainless steel cladding. Additional geometric inputs for each bundle size are shown in Table IV. The flat-to-flat distance a of the bundle is computed by the expression:

$$a = 2 \left[np \cos(30^\circ) + r + b \right] ,$$

where

p = rod triangular pitch

r = rod radius

b = gap between outer-row rods and can wall

($b = p - 2r$ in this work)

n = number of rows in the bundle (2 for 19 pins, 3 for 37 pins, etc.).

Other geometric details, such as wetted and heated perimeters, hydraulic diameters, relative locations of wire-wrap crossings, and rod-to-channel connection data are calculated by a separate program, GEOM, and the results input to COBRA4I.

Both prototypic CRBR³³ and non-prototypic Sandia ACPR Upgrade³⁴ axial power profiles are used to determine the effects of this power factor on the optimum bundle size. The inter-pin power ratio affects the temperature profile across the bundle. Most test reactors to date are conceived with an inter-pin ratio of ~ 1.1 . This study compares the results using this ratio and the prototypic ratio of unity. A

TABLE III

Some Input Parameters

Pin Characteristics

Pin outside diameter (in)	0.23
Pin Triangular pitch (in)	0.2877
S-S cladding thickness (in)	0.015
Wire wrap direction	Clockwise
Wire wrap pitch (in)	11.9
Wire wrap diameter (in)	0.056

Physical Properties

Fuel thermal conductivity (B/hrft ^{°F})	3.0
Fuel specific heat (B/lbm ^{°F})	0.08
Fuel density (lbm/ft ³)	610.0
Cladding thermal conductivity (B/hrft ^{°F})	14.5
Cladding specific heat (B/lbm ^{°F})	0.14
Cladding density (lbm/ft ³)	481.0
Effective wire pitch fraction for forcing flow	0.06
Coefficient for turbulent mixing correlation (β)	0.02
Turbulent cross-flow resistance factor	0.5
Transverse momentum factor	0.5

Initial Operating Conditions

System pressure (psia)	60.0
Coolant Inlet velocity (ft/sec)	20.3
Coolant Inlet temperature (^{°F})	600.0
Steady state average heat flux (B/hr ft ²)	0.1×10^6

TABLE IV
Bundle Characteristics

	Number of Pins				
	7	19	37	61	91
Flat-to-flat (in.)	0.8437	1.342	1.840	2.325	2.837
Number of channels	12	36	72	120	180
Number of gaps	18	54	108	180	270

prompt burst power pulse superimposed on a low steady-state power is considered. The power pulse peaks at 13400 MW and has a pulse width at half-maximum (PWHM) of about 5.3 ms.

The trend of several key variables has been determined as function of increasing test scale. The variables, selected on the basis of their effects on pin failure are: the fuel hot spot, the cladding and fuel axial temperature distributions, and the diversion crossflow between channels at each axial location. The pins under consideration are the central pin and an outer-row pin. The latter is also considered because it is a preferred test pin as far as fuel motion diagnosis is concerned. It is doubtful, however, that an outer-row pin is representative in post-failure events.

The trend of results obtained shows that prototypic thermal hydraulic conditions at the onset of pin failure are established at 61 pins, with 37 pins probably an acceptable minimum. Of special interest is the axial cladding temperature, shown in Fig. 8, for the inter-pin power ratio of 1.1. Because of the fast transient considered, the cladding is still cold when the outer fuel surface is near melting point. The difference in cladding hot spot temperature between 7 and 91 pins is of several hundred degrees ($^{\circ}\text{F}$), and this difference virtually disappears at 61 pins. An important result is the shifting of cladding hot spot upward by 3-4 inches when the bundle size increases from 7 to 91 pins. (The prototypic cladding hot spot is at the top.) A similar trend is observed for the outer-row pin, except for more noticeable irregularities in the axial temperature distributions, presumably due to boundary effects.

Figure 9 shows the central pin cladding temperature for the prototypic power ratio of unity. The trend again upholds the conclusion that a 61-pin bundle is most desirable. The fuel temperature, however,

Figure 8. Central Rod Cladding Temperature ($t = 17$ ms) for Inter-Pin Power Ratio of 1.1

Figure 9. Central Rod Cladding Temperature ($t = 17$ ms) for Unity Inter-Pin Power Ratio

depends weakly on the bundle size. These results indicate that although the fuel temperature is determined by preburst steady state, the cladding temperature appears to be more sensitive to heat transport between channels during transient.

The cladding temperature across the bundle are shown in Fig. 10 for an inter-pin power ratio of 1.1. This two-dimensional effect likely will influence post-failure events more, such as molten fuel crossing between channels, fuel freezing, plugging or sweep-out, and pin failure propagation. The difference in cladding hot spots between 61-pin and 91-pin bundles is reduced to a couple of degrees ($^{\circ}\text{F}$) for the central pin and less than 10 degrees ($^{\circ}\text{F}$) for the outer pins, indicating a fast rate of convergence beyond 61 pins. An inter-pin power ratio larger than unity tends to flatten the temperature distributions across the bundle. This fact can be used to the designer's advantage: by adjusting the inter-pin power ratio, a temperature profile across a small bundle can be made flatter, simulating thereby the inner rows of the 217-pin bundle.

Finally, calculations using the Sandia ACPR Upgrade axial power profile show a weak influence of this factor on the asymptotic trend discussed above, making the conclusions of this study more general.

1.6.2 New Facility Conceptual Design

As a part of the Sandia Fast Reactor Safety Research program, the conceptual design of a facility capable of performing phenomenological experiments with full-length fuel pins has been initiated. One approach is the High Fluence Fast Pulsed Reactor (HFFPR). Several test-facility design concepts are being evaluated for the HFFPR. This section

Figure 10. Cladding Hot Spot Temperature Across Bundle (Inter-Pin Power Ratio = 1.1)

describes preliminary results of neutronics calculations for both gas- and water-cooled concepts, cross section development efforts, and the design of the ex-reactor cooling system for the gas-cooled reactor concept.

The technology of high-temperature gas-cooled systems has been adequately demonstrated in the following instances:

- (a) The Peach Bottom Prototype HTR Power Plant.³⁵
- (b) The Dragon Reactor Experiment.³⁶
- (c) The Los Alamos ULTRA-High-Temperature-Reactor Experiment (UHTREX).³⁷

The characteristics of these reactors and their relationships to the design of the HFFPR are discussed elsewhere.³⁸

1.6.2.1 Neutronic Design Calculations for a Gas Cooled HFFPR -

A summary of the performance characteristics for three fuel types is shown in Figure 11. The energy depositions shown are the values at the edge of a prototypic, CRBR, equilibrium cycle sub-assembly with 30.16% Pu-239 heavy atom enrichment.

The data labeled E/Q_L on the figures indicate the edge-to-centerline ratios of the energy deposition across a 217-pin test bundle. These ratios are 1.13, 1.04, and less than 1.0 for the BeO/UO_2 , $\text{Al}_2\text{O}_3/\text{UO}_2$, and Nb-U fuels, respectively, in prototypic fuel bundles. However the E/Q_L ratios are considerably higher for fully enriched UO_2 test bundles of the same size; the E/Q_L is 1.34, for example, with the BeO/UO_2 driver fuel and no convertor. The energy deposition at the edge of the fully enriched bundle is 7726 j/g for the same example. The E/Q_L ratios can be reduced if

Figure 11. Performance Capabilities of Candidate Fuels for a Gas-cooled Fast Reactor Test Facility

(Energy Depositions shown are on the edge of a 217-pin, prototypic (BOEC) test subassembly.)

one is willing to accept a reduction in performance. This is accomplished by using various convertor materials with high volumetric heat capacity and poor moderating properties. The tradeoff is illustrated in Figure 11 and 12 for the BeO/UO₂ fuel with several thicknesses of an iron-aluminum convertor containing 12.5 weight percent of UO₂ (50% enriched). The results for a 3% enriched UO₂ convertor are shown in Figure 13.

The neutron losses through the test assembly for the Al₂O₃ driver (relatively hard spectrum) and the BeO driver (softer spectrum) are shown in Figure 14. These results were obtained by monitoring the fission activity in a uniformly dispersed U-235 tracer material.

All of the neutron transport results reported here were derived from one dimensional s_n calculation using the DTF-IV code³⁹ and the Hansen Roach 16-group cross section library.⁴⁰ The molecular ratios and enrichments of the BeO/UO₂ and Al₂O₃/UO₂ fuels were selected on the basis of an infinite media parametric study described in the last quarterly.¹⁰ Specifications for the gas-cooled fuels are given in Table V.

1.6.2.2 Neutronic Design Calculations for a Water-Cooled HFFPR -

Calculations on BeO/UO₂ water-cooled systems are being evaluated as an alternative to the gas-cooled systems. Molecular ratios of 35/1, 50/1, and 70/1 using 15-25% enriched uranium as the driver fuel have been examined. Because of the soft spectrum a convertor region has been incorporated to reduce the peaking in the test assembly. The convertor is a dry, gas-cooled design with a 15 percent cooling fraction and consists of 4 wt% fully enriched

Figure 12. Effect of a Fe-Al Converter on Performance

Figure 13. Effect of a UO_2 Convertor on Performance

Figure 14. Neutron Energy Losses in the Test Assembly

TABLE V
Fuel Specifications

BeO/UO₂ ceramic fuel

Molecular ratio, 20/1
Weight percent UO₂, 34.9%
UO₂ Enrichment, 40%
Fissile atom fraction, C_D, 0.0093
Composite fuel density, 3.998 g/cc
Peak operating temperature, 1500°C

AL₂O₃/UO₂ ceramic fuel

Molecular ratio, 9.4/1
Weight percent UO₂, 21.85%
UO₂ Enrichment, 50%
Fissile atom fraction, C_D, 0.01
Composite fuel density, 4.567 g/cc
Peak operating temperature, 1250°C

Nb-U metallic fuel

Weight percent Uranium, 30%
Uranium enrichment, 93.2%
Fissile atom fraction, 0.08
Composite fuel density, 9.62 g/cc
Peak operating temperature, 1250°C

Material compositions of the driver core - volume fractions

Fuel - 0.659 (some cases 0.709 as indicated)
Gas coolant - 0.25 (some cases 0.20)
Structure (SS304) - 0.055
Gap (gas) - 0.036

UO₂ in SS430 of 85 percent theoretical density. Other convertor materials may be evaluated as the need arises.

Figure 15 shows the effect of convertor thickness on the peak energy deposition (j/g) in the test bundle and also the effect on the edge to center ratio of the test bundle. The fissile atom fraction of the driver fuel and convertor are denoted as C_{df} and C_{dc} . The curve marked "Driver = 1200°C" shows the expected energy deposition in the test bundle if one has no regard for the integrity of the convertor. In order to insure the integrity of the convertor, one would have to restrict the limit of the peak temperature in the convertor to approximately 1200°C. This effect is shown by the curve marked "Convertor = 1200°C".

Figure 16 shows the effect of driver loading, C_{df} , on the peak energy deposition in the test bundle and also the effect on the edge to center ratio of the test bundle. The curves marked "Driver = 1200°C" and "Converter = 1200°C" have the same significance as in Figure 15.

As can be seen from Figures 15 and 16, the convertor is very efficient as far as reducing the edge to center ratio in the test bundle. However, one has to pay for this reduction in edge to center ratio with a reduction in performance. In order to achieve the 2500 j/g energy deposition goal in the prototypic test bundle, it may be necessary to allow an excessive edge to center ratio in the test bundle.

Efforts are continuing to raise the expected performance of the water-cooled system. There are several more parametric studies

Figure 15. Effect of Converter on Performance, Water-cooled Concept.

37 pin prototypic test fuel bundle
 Convertor: 4 wt % fully enriched
 UO_2 in 85% dense SS430
 1 inch thick
 $C_{dc} = 0.0128$
 15% cooling fraction
 $k_{eff} = 1.175$

Figure 16. Effect of Fuel Loading on Performance, Water-cooled Concept

of the water-cooled system to be made; the goal is to obtain reasonable operating temperatures in both the convertor and the driver simultaneously at an acceptable performance in the test fuel. Two of the parameters to be studied are the fissile loading of the convertor, C_{dc} , and the cooling fraction of the convertor. Other convertor materials may have to be evaluated.

1.6.2.3 Cross Section Development Efforts -

With regard to both the gas-cooled and water-cooled systems, temperature dependent cross section sets are being developed to augment the next phase of neutronic calculations. The modular code AMPX is being used to evaluate the raw ENDF/B data in order to catalog the necessary temperature dependent cross section sets. AMPX allows one to not only specify the temperature at which cross sections are evaluated, but also the spectrum that the cross sections are weighted against.

A master library of nuclide cross sections with a 123-group energy structure is being produced by the XLACS module. These in turn will be processed by the module NITAWL which in turn will be processed by XSDRNPM to produce cell weighted and collapsed cross section sets in ANISN format. These cross sections can in turn be utilized by the two dimensional transport code TWOTRAN.

1.6.2.4 Determination of the Nominal Design Parameters of the Gas-Cooling System -

The basic cooling system design consists of two heat exchangers:

- (a) Exchanger 1 - Heat transfer from primary helium loop to intermediate water loop.

- (b) Exchanger 2 - Heat transfer from the intermediate water loop to the cooling tower.

Since Exchanger 1 could possibly be contaminated, Exchanger 2 serves the purpose of isolating the cooling tower from Exchanger 1.

A linear mathematical analysis of the cooling system, as shown in the block diagram, has been performed in Ref. 38. The results of this analysis are presented in Figure 17.

Heat Exchangers and Pumps:

A thorough analysis of the heat exchangers has been carried out in Ref. 38. The helium to water heat exchanger is of the U-tube type because of its simplicity and ability to withstand thermal stresses. The structural design guide for the helium heat exchanger is code case 1592 (Class 1 components in elevated temperature service), Section III of the ASME Boiler and Pressure Vessel Code.

The analysis of the water heat exchanger has also been carried out in Ref. 38. The mass flow rates through both heat exchangers are given in Fig. 17. The results of the linear analysis, as well as the manufacturer's results, indicate that the size of both heat exchangers will be on the order of 3 feet in diameter and 10 feet in length.

Mechanical Technology Incorporated (MTI) has considerable experience in the manufacture of helium pumps. MTI manufactured the helium blowers for the UHTREX reactor and is now designing similar blowers for Union Carbide and ORNL.

Figure 17. Block Diagram of Gas Cooling System

The circulator can easily handle 26,000 lb/hr of helium at 150 psia and 600°F with a head rise between 2.5 and 5 psi. The approximate horsepower would be 110 at 2.5 psi head and 220 at 5 psi head.

Water pump 1, located between the two heat exchangers, will operate at a speed of 1750 RPM and will have about 7-1/2 horsepower. Water pump 2, serving the cooling tower, will be about 50 horsepower and run at 1750 RPM. Further details on the pumps are given in Ref. 38.

The Cooling Tower:

The cooling tower is a Marley series 15 forced draught tower. It is able to handle 450,000 lb/hr of water and drop the water temperature from 150°F to 100°F. It will cover an area of 20 feet by 16 feet and will stand 17 feet tall. The fan will be powered by a 20 horsepower motor.

The SERF Reactor Vessel as a Component of the HFFPR:

A preliminary survey was made of the SERF reactor vessel to determine if it might be used with the HFFPR. Analysis indicated³⁸ that the design stresses of this vessel will be greatly exceeded if it is subjected to temperatures on the order of 1200°F.

The vessel is locked in strong concrete with very little room for expansion. It should not be subjected to a temperature higher than 350°F. Even if a design scheme is devised whereby the vessel can be retained and not subjected to temperatures greater than 350°F, it should be cut off at the concrete so that there will be no overhang. Extreme pressures would be present in an overhang if it was present.

It may be possible to devise a scheme where the existing water cooling system for the vessel can be modified to keep its temperature below 350°F. If this is possible, then the vessel may be usable for the HFFPR.

HFFPR System Layout in the Sandia SERF Building:

It is suggested that the helium/water heat exchanger be placed in the Beam Room north of the irradiation cell and on the same level. Here it will be adequately shielded while being sufficiently close to the reactor.

The secondary heat exchanger may be placed in the pump house together with the two water pumps. The pump house is a new addition to the SERF building.

Finally, the helium purification system should be constructed as a modular unit and placed in lock 1, west of the irradiation cell and on the same level, after it has been provided with a gas-tight liner. Further details may be obtained from Ref. 38.

Cooling System Cost Estimates

In Table VI the estimated prices of all the major components in the cooling system are summarized.

TABLE VI
Cost Summary for Cooling System

Helium Heat Exchanger	\$100,000.00
Water Heat Exchanger	25,000.00
Helium Pump	450,000.00
Helium Pump Controller	75,000.00
Water Pump 1	1,700.00
Water Pump 2	3,000.00
Helium Purification	250,000.00
TOTAL	\$904,700.00

If duplicate pumps are to be utilized in case of pump failure, an additional \$455,000.00 must be added to the above.⁴¹ Adding an extra \$500,000.00 for other equipment, the entire cooling system could cost as much as \$2,000,000.00 for hardware alone.

1.7 Fuel Dynamics

(G. L. Cano, 5432)

An area of controversy in reactor safety revolves around the fuel behavior in a loss-of-flow (LOF) accident in fast reactors such as an LMFBFR. Redistribution of core materials, dependent on the mode of disruption, can have a dominant effect on the probability of a subsequent energetic excursion. If occluded fission gases in the fuel produce rapid fuel dispersal following heating and melting, the net effect can be full negative reactivity and safe reactor shutdown. In the absence of dispersal in such an LOF-driven transient overpower, a large number of pins could melt and produce recriticality in a slumped configuration or a molten pool of fuel. Disruption of the reactor vessel and other serious consequences could result. Thus it is necessary to study the phenomenology of such a postulated LMFBFR LOF accident under simulated conditions.

The current Sandia fuel disruption experiments are designed to study this phenomenology for a number of fuel parameters and power transients. In this initial set of tests, the objectives are:

- (1) to obtain evidence, photographically, of the fuel behavior.
- (2) to learn the mode and time of disruption of fuel pins and the rate of dispersal, if any, of fuel from disrupted pins.
- (3) to study effects of occluded fission gases on fuel disruption.

In this initial series of tests FD-1, each of nine mixed oxide fuel pellets of about 5% burnup, and of various power output values from

4 to 12 kW/ft will be internally fission-heated in an ACPR neutron multipulse. Fresh fuel pellets also will be tested for comparison of response to similar heating conditions with the pre-irradiated fuel. The multiple pulse is used to establish proper temperature profiles in the pellet to simulate LOF conditions.

In this quarter, the experiment canisters were designed and fabricated. A schematic is shown in Figure 18. The fuel is located at lower center of the canister between two mirrors. A lamp backlights the pellet which is imaged through two quartz windows and other relay optics to a telescope. The accompanying hardware has been obtained and assembly of the complete units is in progress.

Nine pins from the EBR-II PNL series of irradiated mixed oxide fuel pins were selected for this and a later series of experiments and were released by NRC for Sandia use. Further, three pins and the pellets within these pins have been selected for use in this series of tests. Radiographs of the three pins taken axially at two angles were very useful. Sectioning of the pins for pellet extraction and mounting of the test samples in the Sandia containers will be done by LASL. A transportainer for movement of the fuel loaded experiment canisters between LASL and Sandia has been designed; ten units are being fabricated.

The neutronics and energy distribution calculations for the pellets for given ACPR pulses are continuing. The optimal ACPR multipulse was defined, resulting, at worst, in a uniform temperature in the pellet and, at best, in a peaked center temperature.

Figure 18. Capsule for Fuel Failure Experiments Currently Being Conducted in ACPR (all dimensions are inches)

Finally, calculations of the canister pressure, temperature, and wall stresses, and fuel fission product inventory have been completed. Nevertheless, full credit for fission product and Pu containment by the dual canister has to be taken.

2. CORE DEBRIS BEHAVIOR

2.1 Molten Core Technology

(F. E. Arellano, 5831; D. A. Dahlgren, 5411;
N. R. Keltner, 9337; R. L. Knight, 5411; D. A. Powers, 5831;
A. F. Turbett, 5831; F. Zanner, 5833; T. Y. Chu, 9337)

2.1.1 Introduction

In the analysis of advanced reactor core disruptive accidents, a number of scenarios lead to the release of molten core material from the guard vessel. This molten material is primarily steel and fuel plus the radioactive by-products. The molten material enters the reactor cavity and either directly contacts the cavity boundary or, for sodium cooled systems, fragments upon contact with sodium which might be in the cavity. In the latter case, the debris bed formed could dry out, resulting in molten material contacting the cavity boundary.

The reactor cavity boundary can consist of anything between bare concrete and a fully designed core catcher. The study is currently aimed at cavity boundary designs important in the inherent retention concept. The program is designed to explore other concepts in the future utilizing the technology developed in the near term.

The inherent retention concept is based on a steel-lined concrete structure, possibly including a firebrick insulating material. The first evaluation for the inherent retention concept should consider the retention capability of the concrete alone. Following this, the interaction with the steel liner should be evaluated along with the insulating material. The phenomena of importance include penetration rates, gas evolution species and rates, and material solubility.

At the request of NRC/NRR, the initial program definition has been modified to include a large-scale transient molten steel/CRBR-concrete interaction and four 75-pound "thermitically" generated melt/concrete interactions.

2.1.2 High-Temperature Melt/Liner Interactions

One test was performed with a high-temperature, thermitically generated, steel melt contacting a 3/8-inch thick steel plate with a 9 mm thick UO_2 layer. The desire is to look at the effect of a UO_2 "crust" on melt penetration. Analysis of this and preceding experiments is underway.

2.1.3 Transient Large-Scale Melt-Concrete Interactions

Two transient tests of the interaction between high-temperature steel melts and concrete generic to the Southeastern portion of the United States were conducted during this report period. The experimental procedures used in the tests have been described elsewhere.³⁷ In essence, the first test (LSCRBR-1) consisted of pouring ~ 210 kg of furnace-prepared 30% stainless steel at 1740°C into the hemispherical cavity of a concrete crucible. The crucible was then cleaned and a second 210 kg pour of steel conducted (test LSCRBR-2). Concrete used to make the crucible used in the tests was similar to that used in previous tests^{42,43} except crushed limestone replaced sand in the concrete mix.

Contact between the melt and the concrete was accompanied by the fire and pyrotechnics observed in previous tests (Fig. 19). These events were, however, less violent than those observed in the previous tests. Gas evolution vigorously agitated the melt, but did not expel great amounts of material. Little slag formation occurred and what did

Figure 19. Photograph of Fire and Pyrotechnics as Result of Contact between Melt and Concrete

form adhered tightly to the metal upon solidification. No concrete spallation was observed. Copious amounts of aerosol were formed in these tests unlike in the previous experiments. Other qualitative features of the tests were similar to those previously observed. These features included cracking of the concrete and water flow from the cracks.

Instrumentation applied to the tests is described below:

- (a) Gas sampling apparatus: Twelve samples of the gases evolved during the melt-concrete interaction were extracted at approximately 4-second intervals.
- (b) Gas temperature monitors: Four thermocouples shielded to prevent radiant heat losses were located in the gas stream to monitor its temperature.
- (c) In-melt temperature sensors: Two shielded thermocouples were affixed to the interior of the crucible cavity to monitor the melt temperature.
- (d) Immersion temperature sensors: A hand-held immersion thermocouple assembly equipped with disposable elements was also used to monitor the melt temperature.
- (e) Embedded thermocouples: Sixteen Type K thermocouples were embedded in the concrete to monitor the rate of melt penetration and the thermal response of the concrete.
- (f) Displacement gauges: Two LVDT displacement gauges were mounted on the external face of the crucible to monitor the mechanical response of the concrete to the presence of the high-temperature melt.
- (g) Recording equipment: Events of the tests were recorded on videotape and photographically at 24 and ~ 400 frames per second.

The instrumentation performed satisfactorily and preliminary data drawn from these instruments are presented here. It is cautioned that these data are preliminary in nature and may merit correction after further analyses of the test results are available.

The temperature history of the transient melts has proven to be of central importance in formulating a mathematical model of the melt/concrete interaction. The immersion thermocouples used in test LSCRBR-2 yielded high-quality melt temperature data (Figure 20). The melt in this test cooled rapidly from 1743°C to 1490°C over the first 4 minutes of the test. An inflection, not an arrest, in the temperature then occurred as austenitic stainless steel was precipitated from the melt. The temperature dropped to $\sim 1430^{\circ}\text{C}$ during the 4 to 4 1/2 minutes required to complete solidification of the melt. The temperature then dropped rapidly again.

Gas temperature data produced by one of the sensors located 40.6 cm above the top surface of the melt are shown in Figure 20. The gas temperature rose to a broad maximum of $\sim 1150^{\circ}\text{C}$ at 30 to 70 seconds after the start of the test. The gas temperature then fell rapidly to 840°C two minutes after the start of the test and declined smoothly thereafter.

It would appear, after comparing Figures 20 and 21, that below 1620°C that volume of gas generated by the melt-concrete interaction abates. It is possible that this temperature marks the end of concrete melting.

Results from the displacement gauges on the concrete are shown in Figure 22. Cracks form soon after the melt is teemed into the crucible. The cracks apparently form at the top of the crucible and propagate

Figure 20. In-Melt Temperature from Immersion Thermocouple in Test LSCRBR-2

Figure 21. Gas Phase Temperature 40.6 cm Above Melt in Test LSCRB-1

a) Output from Top Displacement Gauge

b) Output from Lower Displacement Gauge

Figure 22. Output from Displacement Gauges for Test LSCRBR-1 (Top gauge was 12.7 cm above lower gauge.)

downwards. The arrest in the crack growth $\sim$ 150 seconds after the start of the test is not yet explicable. The overall cracking observed in tests with the generic Southeastern United States concrete was more extensive than that observed with other concrete types. Cracks appeared not only in the faces of the rectangular crucible block but also along the corners of the block. No horizontal cracking was observed.

Gas samples extracted from the gas stream evolved during test LSCRBR-1 proved to contain species similar to those observed in previous melt/concrete tests (see Table VII). The gases from this test were obviously well-reduced by their reaction with the melt. Early analysis of gas samples from test LSCRBR-2 indicate the presence of only carbon dioxide. Neither water, hydrogen, nor hydrocarbons have yet been detected in gas samples from this second test.

Extraction of the solidified melt from the crucible revealed a melt/concrete interface quite different from that observed in previous tests with other concrete types. Rather than a hard incipient melt region covering a powdery heat-effected zone, the interface consisted of two powdery zones. The zone nearest the melt was dark gray and easily removed from the crucible with a vacuum cleaner. Below this was a white powdery region which covered dehydrated but otherwise not greatly altered concrete.

A cross-sectional map of the crucible after test LSCRBR-1 showing these layers is shown in Figure 23. Erosion in test LSCRBR-1 was small compared to that observed in tests with other concrete types. The depth of the heat-effected zone was similar to that observed in other tests.

TABLE VII
Gas Composition from Large-Scale Test

Time (secs)	H ₂	CO (Volume %)	CH ₄	CO ₂	C ₂ H ₄
24	30.72	54.01	--	15.27	--
30	42.30	34.64	0.51	20.85	1.70
53	13.72	35.85	--	16.12	6.80
57	--	56.78	--	38.29	5.20
94	0.14	30.94	0.63	67.39	0.91
98	--	25.09	--	74.90	--

Figure 23. Cross-Sectional Map of Crucible Following Test LSCRBR-1

2.1.4 Sustained Large-Scale Melt/Concrete Interactions

Preliminary tests were conducted in the sustained test geometry except that solid iron was in the test cavity. The results demonstrated highly efficient coupling from the induction coils in the concrete to the iron. The results also showed that the instrumentation contained within the coils requires additional work to adequately suppress the background noise.

2.2 Debris Bed Studies

(J. B. Rivard, 5422; J. A. Brammer, 1136; A. R. Phillips, 5423;
D. N. Cox, 5422; H. G. Plein, 5422;
R. H. Nilson, 1262; D. O. Lee, 1262)

2.2.1 Introduction

The objective of the experimental studies of LMFBR debris beds is to provide information on the behavior of reactor materials following a hypothetical core-disruptive accident. Initially, it is desired to characterize the sodium dryout instability in a bed of frozen fuel particulate which may lead to fuel remelt and associated high temperatures in a typical LMFBR.

Activity during the report period was centered on preparation for, and performance of, the initial in-reactor experiment D-1.

2.2.2 Experiment

The out-of-pile systems test (OPST) was concluded March 25, 1977. The principal objective of this test was to verify the thermal performance of the debris bed experiment package and the helium cooling system prior to commitment to the in-reactor experiment in ACPR. During the test, the system was operated for a total of more than 35 hours with liquified sodium, and the thermal performance was established. Training of operating personnel for the round-the-clock operation was also conducted during this period.

Beginning on March 28, 1977, the helium cooling system, including auxiliary piping, was assembled to the unfueled experiment package after the latter was lowered into the ACPR experiment cavity in preparation for the reactor compatibility experiment (RCE). The objectives of the RCE were to insure compatibility between the experiment system and the ACPR, to determine nuclear parameters, and to rehearse experiment operations. Both the helium system and the reactor were operated and measurements were taken. Finally, the experiment package was removed and disassembly and movement into the special shielding cask was rehearsed. The RCE was concluded on April 4, 1977. The following paragraphs summarize the experiment.

Reference reactivities were determined before the central shield plug was removed from the ACPR. Then, the lower section and the intermediate section of the experiment package were mated and leak checked. The length was measured and the unit was then lowered with the 3-T crane into the central irradiation cavity of the ACPR until the handling

collar rested upon the work stand installed above the ACPR tube. Next, the top section of the experiment package was measured and hoisted with the 3-T crane into position above the intermediate section, and connections were made and the leak check performed.

The lower handling collar and work stand were removed and the upper section was lowered into the ACPR. At this point, it was noted that lateral clearance between the tube and the experiment allowed a small lateral movement of the top of the experiment. This was temporarily remedied using available handling equipment and "C" clamps. The extension of the experiment above the top of the ACPR tube was measured and compared with the length of the ACPR cavity and the previously determined length of the experiment to determine that the experiment was indeed seated on the lower reflector. This set of measurements agreed to within 1/8 inch.

Subsequently, the flexible pipe sections of the pipe module were connected to the top of the upper experiment section and leak checked. Upon completion of the leak checking, the system was connected to the large vacuum system and evacuated, then filled with helium. It was then again evacuated and refilled with helium to an operational pressure of 40 psia. Finally, two gamma detector systems were attached to the helium piping and their electronics activated, but signal outputs appeared to be incorrect. For the RCE, data on gamma levels in the helium flow were supplied by a RAM system supplied by the ACPR operations crew. A RAM system also monitored the ACPR cavity purge. At this point (March 31, 1977), after four days, the experiment system was declared operational and reactor measurements were begun.

Reactivity measurements showed that the experiment, containing everything except fuel and sodium, was worth $\$-2.51$ with respect to the "free field" configuration (with upper reflector), and $\$-2.19$ with respect to the "no upper reflector" configuration. The latter figure compares directly with the TWOTRAN calculation of $\$-2.61$ for the RCE.

For the survey of radiation fields above the experiment, the reactor was operated at 100, 300, and 600 kW. The observed levels were primarily neutrons and were linear with reactor power.

On April 4, disassembly of the experiment apparatus was begun. After removal of the piping connections and installation of the handling collar, the upper section was withdrawn from the ACPR. This was followed by the combined intermediate-lower section. Radiation levels were very low until the lower half of this combined section was removed at which time the lower section read about 6 R/hr on contact. (Activation of the shield in the intermediate section was negligible.)

On May 2 and 3, 1977, the initial experiment D-1 was performed in the ACPR. In this experiment, 2.43 kg of enriched UO_2 particulate was fission heated, bed temperatures and power were measured, and the heat was transferred to overlying liquid sodium which was subcooled. The experiment configuration, fuel, and operating conditions have been described in detail in previous quarterly reports.

X-ray measurements showed that the bed had an average depth of 58 mm. The particulate UO_2 used was irregular in shape and ranged in size from 100 to 1000 μm , which produced a porosity of 0.46. Measured permeabilities for similarly prepared UO_2 ranged from approximately 100 to 200 darcys depending on the degree of compaction. Maximum specific power of approximately 1 kW/kg of UO_2 was developed in the bed while it was located in the central irradiation space of the ACPR operating at a steady-state power of 425 kW.

The matrix of conditions attained during the experiment is shown in Fig. 24. The region enclosed by the dashed lines includes sodium conditions which characterize normal and off-normal operations of LMFBRs and power loadings which encompass a portion of a typical fission product decay curve.

Thermocouples at three elevations sensed the temperatures of the sodium; five thermocouples were located at two elevations within the particle bed. The sodium temperatures showed oscillations typical of the convective process. With one exception, temperatures within the particle bed (which ranged up to 1008 K), behaved in a manner characteristic of thermal conduction. Temperature differences measured between points located 33 mm apart vertically, when plotted versus bed power, are closely approximated by a line based upon the Fourier conduction equation for a heat-generating medium with an effective thermal conductivity equal to about 50 percent of that for liquid sodium.

Figure 24. Matrix of Conditions Attained During the Experiment (Bed Loading 300 kg/m²)

Figure 25. Tracing of Thermocouple Signals

(Shown is the transition of B2 temperature at bed bottom from conduction regime - B1, B3 - to convection regime paralleling sodium temperatures above bed - S2. Sodium was frozen and remelted during the gap indicated - gap not to scale.)

As the sodium temperature was being lowered from 773 to 673 K after 11 hours of operation of the experiment, the temperature signal from the thermocouple located in the lateral center at the bottom of the bed began a marked descent below the temperatures of adjacent bottom thermocouples. After 1.5 hr it began to parallel the behavior of the sodium temperatures, except at a slightly higher temperature (Fig. 25). Additional tests indicate that this region of the bed had experienced a transition to single-phase convection. The adjacent thermocouples remained unaffected throughout the remainder of the 24-hour experiment.

This experiment demonstrates that internally-heated oxide fuel particulate can sustain power loadings and sodium temperatures as shown in Fig. 24, at a bed loading of 300 kg/m^2 and remain stable (no dryout). Heat transfer within the bed was primarily by conduction, but a local, single-phase convection cell also developed late in the experiment. Analysis of the results of the experiment is continuing.

The following experiment, D-2, is scheduled to be performed in August. In this experiment, 4.86 kg of UO_2 will be utilized, which will result in a particle bed nearly twice as thick. Also, the experiment capsule will be fabricated from Inconel 617, rather than stainless steel, which will allow higher bed temperatures and higher specific power levels to be attained. These will produce local boiling of the sodium within the bed, which may in turn induce local dryout.

2.2.3 Studies and Analysis

Initial studies have been completed of the degree of dynamic similitude which exists between out-of-pile experiments using materials such as sand and water, and the LMFBR debris bed or D-series experiments.

The resulting scaling parameters have been obtained by nondimensionalizing the steady-state conservation equations which are applicable to the three regions of a two-phase debris bed -- the pool of liquid, the liquid saturated zone in the bed, and the two-phase boiling zone.

The comparison of the dominant parameters is given in Table VIII, together with comments on the similitude prospects and problems.

2.2.4 Status

The out-of-pile systems test (OPST) was completed in March, and the Reactor Compatibility Experiment (RCE) was completed in April. On May 2 and 3, the initial in-reactor debris bed simulation experiment, D-1, was performed. For the conditions of the experiment, the internally heated oxide fuel was cooled by the liquid sodium primarily in the conduction mode. Analysis of the experiment is continuing. The next experiment, D-2, is scheduled for August and will feature a thicker bed, higher specific power, and somewhat higher bed temperatures.

2.3 Molten Fuel Pool Studies

(H. G. Plein, 5422; G. A. Carlson, 5423;
W. H. Sullivan, 5423; M. R. Goebel, 1136;
A. R. Phillips, 5423)

2.3.1 Introduction

The in-core molten fuel pool experiment program will investigate the heat flux distribution and containment ablation potential of molten LMFBR debris materials. The studies will be conducted using real materials under typical temperature and heating conditions.

TABLE VIII

Comparison of Dominant Parameters of Out-of-Pile
Experiments and D-Series Experiments

Similitude Parameter	Comments
1. $Ra_1 = \frac{\rho_l g L K \beta}{\mu_l \alpha_B} \left(\frac{Q_v L^2}{k_B} \right)$	This Rayleigh number for the single phase region is based on the heat generation rate. For $Ra_1 \leq 32$ the single-phase region is conduction dominated.
2. $\Delta T^* = (T_s - T_p) \left(\frac{Q_v L^2}{k_B} \right)^{-1}$	The dimensionless temperature parameter is indicative of the subcooling effect of the pool on the bed.
3. $Ra_2 = \frac{\rho_{lv} g K L}{\mu_l \alpha_B}$	This is the Rayleigh number for the two-phase portion of the bed. For similitude this is probably the most important parameter to match.
4. $v = \frac{Q_v}{\rho_l h_{fg}} \frac{L^2}{\alpha_B}$	The volumetric rate of phase change is indicated in this two-phase parameter.
5. $T = (K\varepsilon)^{1/2} \frac{\gamma_{lv}}{\mu_l \alpha_B}$	This is the two-phase surface tension parameter which indicates the ability to resupply a two phase region through capillary action.

Nomenclature

c_l	Specific heat of liquid	α_B	Thermal diffusivity of the bed
h_{fg}	Latent heat of vaporization	μ_l	Viscosity of liquid
k_B	Bulk thermal conductivity in the liquid saturated region of the bed	β	Coefficient of thermal expansion
K	Permeability of the porous bed	ρ_l	Density of liquid
L	Overall height of the porous bed	ε	Porosity
Q'''	Heat generation per unit volume of bed	k	Relative permeability
T_p	Pool temperature	γ_{lv}	Interfacial surface tension between liquid and vapor phases
T_s	Saturation temperature of the boiling fluid	ρ_{lv}	Difference between liquid and vapor

The major activity during this quarter was the conduct of the first in-core molten fuel pool experiments. Two successful runs were made at Annular Core Pulse Reactor (ACPR) power levels of 150 kW and 600 kW. The ultrasonic thermometry indicated that incipient melting of the UO_2 occurred during the experiment at high power and also provided axial temperature profile data within the UO_2 during the experiments.

In preparation for the in-core experiments, the ultrasonic thermometers were calibrated and the ultrasonic sensing electronic modules were fabricated and assembled into the diagnostic console. Pressure and leak testing of the dual containment vessels as well as ultrasonic and dye penetration testing of the tungsten crucibles were accomplished.

2.3.2 First In-Core Molten Fuel Pool Experiment

The design of the experiment capsule for the initial in-core molten pool experiment was described in the previous quarterly report.¹⁰ This quarter, the containment vessels were fabricated and leak and pressure tested, the experiment capsule assembled, and the first in-core experiment run.

System diagnostics included pressure transducers, thermocouples, and ultrasonic thermometers. Tungsten-rhenium thermocouples were used to monitor fuel and crucible temperatures, while chromel-alumel thermocouples were used on the steel vessels. Two ThO_2 -sheathed tungsten ultrasonic thermometers were placed near the axial centerline of the fuel bed. Each ultrasonic sensor was designed to measure temperatures over 5 axial zones, each about 10 mm long.

All electronics for the diagnostics data acquisition were mounted in one double-wide rack. The most critical thermocouple data and the

pressure transducer data were recorded on continuous strip-chart recorders. Other thermocouple data were recorded on an intermittent strip chart recorder. Digital readouts for all thermocouples were also used. The ultrasonic thermometry raw acoustical signals were conditioned using electronics developed earlier in the program. After signal conditioning the data were sent to the Sandia Data Acquisition and Display System (DADS) computer for real-time conversion to temperatures and the temperatures displayed on a TV monitor. Ultrasonic temperature data were updated every 17 seconds during the experiment.

To provide confidence in the performance of the capsule and the diagnostics, a low-power (150 kW) reactor run was made a day before the main experiment. In this run it was determined that the bed specific power was 0.6 kW/kg. The reactor power for the main experiment was 600 kW, which gave a fuel bed specific power of about 2.4 kW/kg. Under these conditions, incipient fuel melting was reached in about 17 minutes, based on ultrasonic thermometry data. The reactor run was terminated at 17.7 minutes.

In general, the pressure and temperature data were as predicted before the experiment. The three tantalum-sheathed W5%Re/W26%Re thermocouples monitoring fuel temperature and inner crucible temperature gave spurious readings at temperatures above 1850-1950°C, presumably due to failure of the insulating properties of the BeO insulator. Other thermocouples gave satisfactory readings throughout the experiment.

The ultrasonic thermometers gave the only fuel temperature data obtained above 2000°C. Figure 26 shows schematically the position of the ultrasonic thermometers in the fuel bed. Both thermometers experienced some degree of inaccuracy during the run due to development of distorted acoustic reflections. Since these were not observed in

Figure 26. Five-Element Ultrasonic Thermometer in UO₂ (Schematic)

furnace tests, they are believed to be related to the in-core experimental configuration. In particular, it is believed that ZrO_2 powder from the low-density insulation had entered the space between the sensor and sheath, and that this ZrO_2 powder or a melt resulting from it was responsible for the signal errors observed. One ultrasonic sensor gave poor data above 2000°C , while the other sensor gave relatively good data with small ($< 100^\circ\text{C}$) systematic errors in some sensor elements.

Figure 27 shows data from the ultrasonic thermometer which functioned relatively satisfactorily throughout most of the experiment. As seen, the highest temperature sensor element is at the center, and is approaching the melting point of UO_2 (2860°C) around 17 minutes. The fluctuations in most sensor elements near 17 minutes are not indicative of temperature fluctuations, but probably represent perturbations of the sensor signals caused by movement of the fuel, sheathing or sensor rod, most likely accompanying partial fuel melting.

Figure 28 shows the experimental temperature profile, indicated in the fuel near incipient melt, compared to a temperature profile calculated for similar conditions. Although the actual temperatures are probably not quite correct for the reasons mentioned earlier, the power of this new diagnostic tool is clearly evident. Not only have temperatures nearly 1000°C higher than the failure point for tungsten-rhenium thermocouples been successfully measured, but also temperature profile data have been gathered simultaneously, using a single sensor rod with 5 sensing elements. By eliminating interference from ZrO_2 powder, it should be possible to obtain much better temperature data ($\sim \pm 15$ K reproducibility) in subsequent in-core experiments.

This first in-core experiment represents a major milestone in the molten fuel pool program. The capability for fission-heating UO_2 to

Figure 27. Ultrasonic Thermometer Records for First In-Core Molten Pool Experiment

Figure 28. Temperature Profile Data in Fuel Near Incipient Melt

the melting point in a controlled manner has been demonstrated. Containment and safety of the experiment under the severe temperature and radiation field conditions present have been proved. A novel diagnostic technique has been improved and successfully employed, measuring UO_2 temperatures and temperature profiles to the melting point for the first time in a reactor environment. Subsequent experiments in the fuel-only series will melt a larger portion of the fuel, will further test the capabilities of the ultrasonic thermometers, and will provide preliminary temperature gradient data from which to define the heat transfer characteristics of UO_2 at or near the melting point.

It is hoped that one fuel-steel experiment may be run before the ACPR shutdown, scheduled for October 1, 1977. This experiment will be useful in defining more sophisticated future experiments to study structural material ablation and fuel-structural material interactions.

2.3.3 Diagnostics Development

During this quarter, the ultrasonic thermometry went through its final design and testing prior to the first in-core experiment. Electronics to provide acquisition and analysis of the signals from two ultrasonic thermometers, each having five sensor elements, were fabricated and tested. The ultrasonic thermometer configuration giving the best results in furnace tests, and therefore chosen for use in the in-core experiments, is shown in Figure 26. The thermometer is fabricated of 0.5-mm diameter thoriated tungsten rod, with notches ground in the circumference at 10 mm intervals to provide the acoustic discontinuities which define the five sensor elements. The notch-to-rod-diameter ratio is approximately 0.6, giving reflection coefficients around 0.25. The sensor rod is surrounded by a ThO_2 sheath to separate it from the UO_2 fuel, which could distort the ultrasonic signals as it sinters and

melted. In earlier experiments, sensors with tungsten sheaths had shown excessive reflections developed due to sticking between the sensor and sheath. Use of a ThO_2 sheath appears to eliminate sticking reflections entirely, and constitutes a significant advance in the utilization of tungsten ultrasonic thermometers. However, condensation of ThO_2 vapor on the sensor rod during long periods (~ 60 minutes) at high temperatures ($\sim 2900^\circ\text{C}$) can cause distortion or decalibration of the sensor rod, representing a limitation of this concept. For the in-core molten pool experiments whose total duration is 20-30 minutes, with only 5-10 minutes at very high temperatures ($\sim 2860^\circ\text{C}$), decalibration due to the ThO_2 condensation is not a serious problem.

Prior to use in the in-core experiment, each thermometer was calibrated in a helium gas-filled furnace. A 20-mm diameter tungsten tube surrounding the sensor provided temperature uniformity, and also served as a blackbody enclosure for pyrometric temperature measurements. A calibration curve for one sensor is shown in Figure 29. Data from the initial temperature excursion, in which the properties of the tungsten rod were changing, are excluded. Once heat-treated, the properties remain relatively constant. The data shown were fit with a fifth-order polynomial which was then used in the in-core experiment to convert acoustical delay times for each sensor element into temperatures. The standard deviations of the data fit for the two sensors were 12.8°C and 8.2°C , from 1400°C to 2900°C .

Results of the ultrasonic thermometry in the first in-core experiment are described in Section 2.3.2.

Because earlier in-core qualification experiments on Kulite semiconductor strain gauges had shown a significant radiation sensitivity at elevated temperatures (~ 60 - 120°C), another experiment was run in

Figure 29. Calibration Curve for 2% Thoriated Tungsten Ultrasonic Thermometer

which a Statham metal strain gauge was compared with a Kulite semiconductor strain gauge, both with 340 kPa (50 psi) range. Figure 30 shows the results of a 120-minute run at 600 kW ACPR power, in which cooling air was used to maintain the pressure vessel at various temperature levels. The pressure in the vessel was cycled from 185 kPa to atmospheric (85 kPa) and back at ten-minute intervals, using an accurate dial gauge as the primary pressure monitor. Only the 185 kPa data, expressed as mV, are shown. As seen in the Figure, the Kulite gauge failed to perform satisfactorily above 60°C, in agreement with previous results, while the Statham gauge performed quite well to 160°C which is actually higher than its rated maximum use temperature (121°C). On the basis of this test, Statham transducers may be used in place of Kulite transducers in later in-core experiments.

2.3.4 Evaluation of Capsule Components

To satisfy a requirement of the Sandia Safeguards Committee, both sizes of tungsten crucibles were tested for cracks or deep flaws. The large tungsten crucibles, used as secondary containers, were checked with a dye penetrant as well as ultrasonically. A single crack was found in one of the six large crucibles which disqualified it for use. The small tungsten crucibles were too thin (0.5 mm) to allow ultrasonic testing, but were checked with a dye penetrant. Again one of those crucibles was found to have several serious flaws and discarded.

A thermal stress calculation was made for the inner steel containment vessel. The calculation used the code SASL⁴⁴ and a temperature profile generated by the code TAC2D⁴⁵. The containment vessel was shown to be capable of containing 3.1 MPa at the maximum expected temperatures. In addition to this calculation, both of the containment vessels were successfully pressure tested to 0.77 MPa which is much

Figure 30. Response of Pressure Gauges of Temperature and Radiation

greater than the maximum working differential pressure of 0.1 MPa.

During the assembly of the experiment, the inner containment vessel was leak tested at 0.21 MPa and the outer containment vessel was leak tested at 0.31 MPa for over an hour each with no detectable leaks. The vessels had earlier shown a leak rate of $< 10^{-6}$ std cc/s helium in a vacuum leak test.

2.3.5 Status

During the next quarter, another fuel-only experiment and the initial fuel-steel experiment are scheduled. For the next fuel-only experiment, additional fuel will be melted and more accurate axial thermal profile data will be obtained using ultrasonic thermometry. The revised experiment plan which details the fuel-steel experiments will be submitted to the ACFR Committee.

2.4 Containment Analysis

2.4.1 Introduction

This program will develop a total containment systems model which will examine a variety of possible accident sequences beginning at the point of release from the primary containment vessel and continuing through breach of the secondary containment vessel to development of the radiological release source term. The final model will be sufficiently general to apply to all candidate advanced reactor systems. The model will incorporate all significant state-of-the-art submodels and methods relevant to containment accident phenomenology. Consistent with the desire to develop a workable code as early as possible, the initial

program version will use existing phenomenological models and computer codes to a maximum degree and will concentrate on integrating them into a general systems program. This is the first report of this new subtask.

2.4.2 Progress

This quarter has been dedicated to activities necessary for initiation of this new subtask. Identification and evaluation of relevant work as accomplished previously and by others has been done. This work has included an in-depth review of available codes and models with the object of adapting as many of them as possible for use in this containment analysis study. Work plans have been under preparation concurrent with the selection and assignment of personnel. In addition, hardware design has been initiated and fuel procurement has commenced.

3. SODIUM CONTAINMENT AND STRUCTURAL INTEGRITY

(B. M. Butcher, 5167; D. A. Dahlgren, 5411; N. R. Keltner, 9337;
D. L. King, 9337; L. Kent, 9337; R. L. Knight, 5411;
R. A. Sallach, 5831; J. E. Smaardyk, 5167; H. J. Sutherland, 5167)

3.1 Introduction

Any sodium cooled reactor system must consider the interaction of hot sodium with cell liners and, given either a failed liner or a CDA, the interaction of hot sodium with concrete. The data base available for safety assessments involving these interactions is limited, especially for the concrete and failed liner interactions.

This program has been specifically developed to provide the research required to verify the data base from a confirmatory point of view. The initial sodium-concrete interaction emphasis concerns:

- (a) Depth of penetration of sodium-concrete reactions.
- (b) Effect of sodium spills and sprays at various temperatures.
- (c) Data on water release, hydrogen evolution, and release of other products.
- (d) Determination of the extent of thermal cracking and exfoliation.
- (e) Effect of steel reinforcement on sodium-concrete reactions.
- (f) Determination of the reaction products.

The sodium/steel liner interactions will be studied for both phenomenological and explicit liner designs. The primary purpose of these

experiments is to develop the consequences resulting from hot sodium spills or sprays on defective liners. The initial emphasis concerns:

- (a) The effect of defect size.
- (b) Determination of reaction products generated behind the liner plate, as well as gas pressures and temperatures.
- (c) Determination of the effectiveness of the vent system behind the liners in removing hydrogen and water vapor released from the concrete.

Liner distortions will be studied as well as characterizing failure mechanisms.

3.2 Preliminary Tests

A series of four preliminary tests were planned to verify the adequacy of the experimental facility and instrumentation and to gain experience in its use. The first two of these tests have been completed. Table IX lists the initial conditions for these two tests.

The first three preliminary tests were planned to contain no instrumentation. However, some instrumentation was employed to determine how much self-heating from exothermic reactivity and evolved gas could be expected. Three thermocouples, stainless steel sheathed Chromel-Alumel type, were placed in the crucible cavity supported by 1/4 inch diameter stainless steel tubes.

A strain gage pressure transducer was mounted on the test chamber (internal volume $\sim 500 \text{ ft}^3$) to monitor pressure. A rupture disk rated at 10 psi was also installed to prevent overpressurization of the chamber.

TABLE IX
Initial Conditions for Sodium-Concrete Interaction Tests

Test Number	1	2
<u>Crucible</u>		
Number	P-1-2	P-2
Material	Limestone (CRBRP) Concrete	Limestone (CRBRP) Concrete
Outside Dimensions	48 Inch Diameter x 19 Inch High	78 Inch Diameter x 22 Inch High
Cavity Dimensions	24 Inch Diameter x 7 Inch Deep	48 Inch Diameter x 7 Inch Deep
Length of Cure	261 Days	277 Days
<u>Sodium</u>		
Weight	43 lbs	220 lbs
Initial Depth in Crucible	3.4 Inches	3.4 Inches
Temperature (at dump pipe exit)	~ 450°C	~ 550°C
<u>Chamber Atmosphere</u>		
Composition	~ 50% Air, 50% N ₂	N ₂
Pressure	Atmospheric (12.1 psia)	Atmospheric (12.1 psia)

The chamber was vented through a 4 inch pipe ending in a differential pressure flowmeter. A mass spectrometer was connected to the pipe and thermocouples were placed in the gas stream on both sides of the flowmeter. A wet filter was attached to the exit of the vent pipe in an attempt to trap the sodium oxide particulate.

A movie camera (24 frames/second film speed) with color film was used to photograph the experiment through a quartz window in the chamber. Flood lights were located inside the chamber to illuminate the crucible. A video tape record was made of the outside of the test facility. For the second test, the video record included a sound track from a microphone placed under the test chamber.

The transfers of the hot sodium into the concrete crucibles were completed in less than 5 seconds. Because of the small quantity of sodium in the first test, considerable cooling took place as it passed through the dump pipe. The initial temperature of the sodium as it entered the crucible was about 450°C instead of the desired 550°C .

In both tests, the reaction was reasonably quiescent for about 5 minutes. During this time, the sodium pool cooled about 100°C . However, the sodium-concrete interaction was sufficiently exothermic to heat the reaction pool to about 800°C (compared to the boiling temperature of sodium at 12.1 psia atmospheric pressure of approximately 860°C). The heating phase was followed by a reasonably stable high-temperature phase. In the first test, cooling began about 22 minutes after the dump and cooled over a period of days to ambient temperature. In the second test, there is no well-defined transition from the active heating phase to the cooling phase. Figures 31 and 32 show the temperatures in the reaction product pool during Tests 1 and 2, respectively.

Figure 31. Sodium Pool Temperatures During Test No. 1

Figure 32. Sodium Pool Temperatures During Test No. 2

During the first test, loud reports were heard at irregular intervals, followed by the expulsion of large volumes of gas. In the second test, the reports were much more frequent, but less energetic. It is believed that these reports were due to the concrete spalling and the resulting gas from the reaction of the fresh concrete exposed to sodium.

The gas flowmeter registered very large flow rates, but precise measurements were erratic. In both tests, the flow restricting element became coated with sodium oxide (Na_2O) particulate. The mass spectrometer results were also inaccurate and sampling system design improvements will be required.

3.3 Posttest Results

The test chamber was opened approximately four days after each test and the crucible was removed. A very heavy white oxide layer, consisting of Na_2O dust, covered the entire floor of the chamber as well as the horizontal portion of the splash shield to a depth of $1/8$ to $1/4$ inch. This layer rapidly degraded (in a few hours) to sodium hydroxide when exposed to the air. Unreacted sodium was evident on the flat surface of the splash shield as well as in globules that had shot up and frozen to the ceiling of the test chamber.

The reaction debris in the crucible was a hard, brittle, rock-like material. This was partially chipped away to determine the extent of penetration into the concrete by the reaction. The maximum penetration depth was about 3 inches in the center of the crucible. The average downward penetration at the outer edge of the cavity was $\sim 1/4$ inch. Only slight penetration into the side wall was observed ($\sim 1/8$ to $1/4$ inch at the bottom corner) in the first test and none in the second test.

The reaction debris consisted of hard, irregularly surfaced material which was difficult to break. The debris appeared to consist of two phases: (a) a light gray material and (b) a dark material resembling fused slag. The light gray material was dispersed throughout the black "slag" in layers. Qualitative tests on the debris showed that it contains no free sodium, that it does not react with water, but that it does evolve gas readily when immersed in acids.

Samples of the two phases were analyzed separately by x-ray diffraction. The gray material consists mostly of calcium carbonate (CaCO_3). This analysis supports the conclusion that the gray material is unreacted concrete that spalled from the surface of the crucible.

The black material consists primarily of the compounds CaCO_3 and CaO with some Ca(OH)_2 , Na_2CO_3 , C , Na_2O , and MgO . Chemical analyses yields the following results: Na - 26%, Ca - 20%, Mg - 2%, C - 3%, and CO_2 - 16%. Based on the relative amounts of Na_2O dust and the black, slag-like reaction product, it appears that about 3/4 of the initial sodium reacted to form slag and the remaining 1/4 of the sodium went to form the Na_2O dust.

Thermal stresses in the crucible during the reaction were violent enough to cause radial cracking of the crucible. These cracks were independent of preexisting cracks. Slight water leakage from these cracks at the outside edge of the crucible was noted in the oxide layer on the test chamber floor.

In the second test, an annular region of broken-up concrete (without any traces of reaction products) was found underneath the layer of slag-like material as shown in Figure 33. Since there are no reaction

Figure 33. Posttest Configuration of Crucible P-2

products present, this concrete damage may result from unrelieved thermal stresses present after the sodium-concrete reaction was completed.

3.4 Facility Modifications

The purpose of the first four tests in the sodium-concrete interaction test program is, in part, to verify the adequacy of the experimental facility. For the first two tests, the facility worked quite well with only a few minor problems.

The differential pressure device used to measure gas flow rate has not worked properly due to the larger-than-expected amount of sodium oxide (Na_2O) particulate carryover. The first test used a laminar flow device as the impedance. This became plugged very quickly. For the second test, the impedance was a length of 1 inch diameter pipe. This became coated on the inside, so the calibration was changing continuously during measurements. For future tests, a large baffle assembly with filters will be placed upstream of the laminar flow device to filter out the particulate.

The early measurements with the mass spectrometer were inaccurate due to the design of the gas sampling pipe. This is being modified for future tests.

The load cells supporting the dump tank were expected to indicate the amount of sodium in the tank and permit determination of the dumping rate. However, it has been found that wind blowing against the tank causes sufficient fluctuations in the reading to make accurate measurements impossible. An electrical depth gage has been installed which can measure the initial sodium depth. By keeping the dump valve open

for a specified length of time, it is possible to ensure that all of the sodium has been dumped. The approximate rate can be determined from the movie taken during the dump.

The test plan originally included an immersion heater in the reaction pool to maintain temperatures in the event that the reaction was not sufficiently exothermic to do so on its own. However, in the two tests run so far, temperatures rose to about 800°C and stayed at that point until all of the sodium had been reacted without any external power addition. If this phenomenon continues in the next two tests, the immersion heater will not be included in the fully instrumented tests.

3.5 Safety

A fire fighting school was conducted for the operating personnel and for representatives of the base fire department. These people were instructed in the characteristics of sodium fires, in the use of proper clothing and protective equipment, and in the accepted methods of handling burning sodium.

4. RESEARCH FOR ELEVATED TEMPERATURE DESIGN CRITERIA

(R. E. Nickell, 5431; C. M. Stone, 5431; D. W. Lobitz, 5431;
C. H. Karnes, 5835; J. A. Van Den Avyle, 5835; O. J. Burchett, 9352;
J. H. Gieske, 9352; W. B. Gauster, 5111; S. W. Key, 1281)

4.1 Creep-Fatigue Test Program

The testing of uniaxial specimens of 2-1/4 Cr - 1 Mo ferritic steel, for the purposes of providing damage material for nondestructive examination, is in its final stages. Table X provides a description of the test matrix for unwelded 2-1/4 Cr - 1 Mo, and it can be seen that one test - pure fatigue, in argon, to crack initiation, but not to failure - remains. Note that fatigue or creep-fatigue tests that are taken beyond crack initiation and all the way to failure are not useful for NDE purposes. This is due to the hourglass shape of the specimen, which concentrates the damage at the minimum diameter cross section. Failure at that cross section deprives the NDE investigators of an integral volume with which to work. Table XI shows the test matrix for welded 2-1/4 Cr - 1 Mo. The two tests performed to date indicate that the fatigue damage is concentrated in the base metal portion of the straight-sided specimens. Since the strain is controlled by average axial measurement over a 1.25 cm gage length, and since the weld metal did not appear to deform significantly, the strain range in the base metal is approximately twice the nominal strain range.

The remaining test (#11) of unwelded 2-1/4 Cr - 1 Mo is to be a pure fatigue test, in argon, at 1% strain range, no hold period, and termination after crack initiation. Its object is to provide a pure-fatigue NDE sample. The remaining welded tests (#12-15) are to be pure fatigue tests, in air, with diametral strain control and extensometer located

TABLE X

Unwelded 2-1/4 Cr - 1 Mo Steel
(Creep-Fatigue; 593°C; Mar-Test, Inc.)

Specimen No.	Atmosphere	Strain Range, %	Held Period, Min.	Cycles		Notes
				N _I	N _F	
1	Air	0.5	0	--	20000	
2	Air	1.0	0	1691	2099	
3	Air	1.0	6, compression	658	1372	Test inadvertently carried out to failure.
4	Air	1.0	6, tension	452	--	Good NDE test.
5	Air	1.0	3, tension 3, compression	411	--	Good NDE test.
6	Argon	1.0	6, compression	2138	2968	Test inadvertently carried out to failure.
7	Argon	1.0	6, tension	986	1715	Test inadvertently carried out to failure.
8	Argon	1.0	0	1414	1810	Test inadvertently carried out to failure.
11	Argon	1.0	0	*	*	

*Test remaining to be carried out.

TABLE XI

Welded 2-1/4 Cr - 1 Mo Steel
(Creep-Fatigue; 593°C; Mar-Test, Inc.)

Specimen No.	Atmosphere	Strain Range, %**	Hold Period, Min.	Cycles		Notes
				N _I	N _F	
9	Air	1.0	0	-	343	Test Inadvertently carried out to failure.
10	Air	0.5	0	-	2911	Test inadvertently carried out to failure.
12	Air	0.5	0	*	*	Diametral control, axial base metal extensometry.
13	Air	0.5	0	*	*	Axial control, diametral base metal strain measurement.
14	Air	1.0	0	*	*	Diametral control, axial base metal strain measurement.
15	Air	1.0	0	*	*	Axial control, diametral base metal strain measurement.

*Test remaining to be carried out.

**The base metal portion of these straight-sided creep-fatigue specimens constitutes one-half of the gage length; the weld metal also constitutes one-half the gage length, but the strain is preponderantly accommodated by the base metal; therefore, the strain range in the base metal is a shade less than twice the strain range shown.

entirely in the base metal, or axial control with diametral strain measurement in the base metal.

Four creep tests are planned for welded 2 1/4 Cr - 1 Mo; two of these have been completed. Essentially, both specimens failed at the same time as their unwelded counterparts. This is not surprising since almost no creep strain was observed in the weld metal portion of the specimens. Necking and tensile instability were observed in the base metal. Therefore, it is uncertain whether additional testing is required.

Sixteen straight-sided creep-fatigue specimens of AISI 316 stainless steel have been shipped to Mar-Test, Inc. A test plan is being developed on the basis that all tests - creep, fatigue, and creep-fatigue - will be performed at Mar-Test or at Sandia.

4.1.1 Nondestructive Examination

Only two of the creep-fatigue specimens have been examined to date. Specimen #4 (six-minute hold in tension, 1.0% strain range) and specimen #5 (three-minute holds in both tension and compression, 1.0% strain range) were damaged to the point of crack initiation, where the tests were terminated. The specimens were machined flat on two parallel faces and scanned ultrasonically, using longitudinal waves and shear waves polarized axially or transversely. Since the damage in the specimen was of the fatigue-cracking type, rather than the large-strain tensile instability observed in creep rupture tests, the ultrasonic readings were felt to be representative of pure fatigue damage, as well. Table XII shows the results of these examinations. Virtually no change was seen in the longitudinal velocity (the ends of the specimen are assumed to be undamaged and are used for reference values). However, the shear

TABLE XII
NDE (Ultrasonic) Tests
of Unwelded, Creep-Fatigued 2-1/4 Cr - 1 Mo

<u>Specimen No.</u>	<u>Propagation Mode*</u>	<u>End</u>	<u>Location</u>	
			<u>Middle (Damaged)</u>	<u>End</u>
4	1	5.951**	5.961	5.944
	2	3.261	3.249	3.261
	3	3.261	3.269	3.261
5	1	5.959	5.966	5.959
	2	3.264	3.261	3.266
	3	3.264	3.277	3.266

*The number 1 refers to longitudinal wave propagation through the thickness.
The number 2 refers to an axially polarized shear wave propagating through the thickness.

The number 3 refers to a transversely polarized shear wave propagating through the thickness.

**All velocities are in mm/ μ sec.

velocities displayed the same type of behavior observed in the pure creep specimens -- namely, a decrease in shear wave velocity when axially polarized and an increase in shear wave velocity when transversely polarized.

4.1.2 Microstructural Analysis

At least two microstructural theories might serve to explain the observed ultrasonic phenomena. One might be called a lattice-distortion model and the other a grain-boundary distortion model. Specimens from creep, fatigue, and creep-fatigue damaged 2-1/4 Cr - 1 Mo samples are being thinned at present and will be microscopically examined shortly. This should provide some insight into the two theories.

4.1.3 Positron Annihilation

AISI 316 stainless steel specimens have been fatigued, at room temperature, over a strain range of 0.9%. One specimen was tested to failure and the others were tested to various life fractions (10%, 40%, and 63%). Figure 34 shows the Doppler broadening line shape parameter, as a function of the number of fatigue cycles. Note the saturation of position capture by defects at the 10% damage level, the slight decrease over a broad range, and the resaturation near the failure cycle level.

Additional specimens are to be fatigue-tested in order to refine the measurements in the 0-10% damage range, as well as to test at lower strain ranges.

Specimens and grips have been ordered for high-temperature (593°C) tests of 316 stainless, which should be underway during the next quarter. The room temperature matrix will be repeated.

Figure 34. Positron Annihilation (Doppler Broadening Method)

4.1.4 Surveillance Test Loop

With both the ultrasonic and positron annihilation nondestructive examination methods showing some degree of promise, some initial planning is underway for the surveillance test loop that will be designed to test both methods, and other possible methods, on prototypic hardware. Current thinking is that a U-tube specimen, of sufficient size to eliminate significant surface curvature effects, will be chosen. The ends of the U-tube will be fixtured so that either liquid sodium or gas can be circulated in order to control the thermal environment. In addition to thermal gradients, internal pressure, axial loading, and U-tube opening/closing lateral loads will be possible. High temperature instrumentation is now being evaluated.

4.1.5 Multiaxial Creep-Fatigue Facility

Interaction with MTS designers is continuing. The internal pressure gas system and the grips for a button-ended, closed cylindrical specimen, with about a 5-cm gage length and a 1.25-cm outside diameter, are of current concern. Analysis of various wall thicknesses and loading combinations (internal pressure, axial load) is continuing. Work on a meaningful failure theory, based upon hydrostatic stress and deviatoric deformation, is proceeding.

4.1.6 Creep-Fatigue Damage Analysis

D. W. Lobitz and R. E. Nickell have been working with J. M. Duke (Westinghouse, Tampa) and C. C. Schultz (Babcock and Wilcox, Alliance) to develop a set of alternative rules for Class 1 nuclear components, subjected to creep-fatigue damage, evaluated with strain-range partitioning concepts.

4.2 Creep Buckling Analysis

The Japanese experiment on creep buckling of piping elbows, conducted by the Power Reactor and Nuclear Fuel Development Corporation (PNC), has been analyzed sufficiently, with the simplified elbow element in the MARC general purpose finite element stress analysis program, to draw a number of conclusions. First, our analysis agrees with the original Japanese analysis; both overpredict deformation. This is due to the inherent limitation of the simplified elbow element (constant bending movement over an elbow sector). Since the Japanese experiment involved nonuniform bending around the elbow, the location of the primary buckling ovalization would normally occur at Section A-A' (see Fig. 35), where the bending movement is a maximum. However, the stiffening effect of the attached straight pipe section inhibits this ovalization. (If the bending were uniform, the primary buckling ovalization would take place at Section C-C'.) The result of these effects is to move the primary buckling ovalization away from A-A' to Section B-B', which has a relatively high elbow-closing bending movement and which is far enough removed from the straight pipe to preclude significant stiffening.

Because of these considerations, the simplified elbow element, which cannot take end stiffening into account in any rational way, is dramatically over-flexible. We are now pursuing the analysis with a doubly-curved shell element in order to accommodate end stiffening.

4.3 Cell Liner Analysis

Parametric analysis of the steel liners of concrete equipment cells, when subjected to accidental sodium spills, is complete, except for a combined convection/conduction analysis of a jetting-type spill. This

Figure 35. Schematic of PNC Elbow Creep Buckling Experiment

analysis is to simulate, as nearly as possible, the conditions of the HEDL LT-1 test. The Reynolds number is so high that computation is difficult. Nevertheless, the analysis has a reasonable chance of success.

Data from the literature on jetting flows involving gases at high Reynolds numbers indicate that a Nusselt number of about 40-50 correlates well with the LT-1 test conditions. Based on this data, our analyses with the largest heat transfer coefficient (about 6000 Btu/hr/ft² F or about 400 W/m² K) closely resemble the actual conditions.

A report on the cell liner analyses is in preparation.

4.4 General

Papers on creep buckling and creep-fatigue interaction are to be presented at the 4th International Conference on Structural Mechanics in Reactor Technology to be held in San Francisco, August 15-19, 1977. Speakers are C. M. Stone and D. W. Lobitz. A post-conference seminar on "Inelastic Analysis and Life Prediction in High Temperature Environment" will be held on August 22-23, 1977, also in San Francisco. D. W. Lobitz and R. E. Nickell are scheduled to speak on "Multiaxial Creep-Fatigue Damage".

A seminar was given on April 28, 1977, at Rensselaer Polytechnic Institute, Troy, New York, that described the U.S. NRC Elevated Temperature Design Criteria program. R. E. Nickell was the speaker.

S. W. Key, in place of R. E. Nickell, attended the 3rd International Conference on Pressure Vessel Technology in Tokyo, Japan during the

week of April 18-22, 1977. Key chaired a session at the conference. Due to illness Key was unable to accompany Dr. Thomas J. Walker (NRC) on a visit to several laboratories and industrial facilities the week following the conference. Details of the conference and visits are contained in a trip report by T. J. Walker.

5. FUEL MOTION DETECTION

(W. H. Buckalew, 5232; L. M. Choate, 5423; L. R. Edwards, 5423;
J. A. Halbleib, 5231; R. C. Heckman, 2151; F. R. Kroeger, 2151;
D. A. McArthur, 5423; J. E. Morel, 5231; L. D. Posey, 5452;
J. E. Powell, 5423; S. A. Wright, 5423)

5.1 Introduction

The current status of research in the area of material motion detection diagnostics for reactor safety experiments is reported in this section. The purpose of this research is to study various material motion detection schemes, both experimentally and analytically, and to assess their potential as possibilities for existing and future safety test facilities.

In-core techniques that have application to medium- and large-bundle tests in facilities such as SLSF and SAREF are also being studied. They have the advantage of not requiring a slot in the test-reactor core but have inherently lower spatial resolution than the out-of-core systems for small-bundle and few-pin tests. Our studies have shown that the two systems may be competitive from the standpoint of resolution on large-bundle tests.

Several out-of-core, line-of-sight schemes are being investigated, including gamma-ray pseudoholography involving coded aperture imaging, electron beam flash X-radiography, and dynamic fission track devices.

5.2 In-Core Fuel Motion Detection

The main objective of the in-core fuel motion detection (FMD) program at Sandia is to test the feasibility of using large arrays of in-core detectors to monitor fuel motion in medium- to large-scale tests. This objective will be accomplished by mechanically simulating fuel motion in a 37-pin test assembly placed in SPR-III. The program is divided into three tasks: the SPR-III FMD experiment, detector research and development, and the unfolding analysis. Activities for each of these tasks are reported.

5.2.1 SPR-III FMD Experiment

A purchase order for fabrication of the fuel pin positioning system was initiated. However, prior to sending out the purchase order, the design and operation was described to the Sandia SPR Committee to gain feedback information about the safety of the design. The Committee appeared to have no reservations about the safety of the design.

Figure 36 shows an isometric view of the fuel pin positioning system. With this design the experimenter may vertically displace any or all of the 37 pins by pulling cables which run to the experiment station. The unique feature of this design is a locking device which uses a set of collets to hold the pins in place. The collet release mechanism is under control of the reactor operator and thus prevents movement by the experimenter of the pins without the operator's approval.

A support grid for holding the fission couple detectors was also designed. Figure 37 shows a schematic of one assembly. The detectors are mounted on top of 1.5 mm stainless steel tubing. The stainless

Figure 36. Isometric View of the Fuel Positioning System and Collet Locking Device (upper assembly only)

- DETECTORS ABOVE GRID PLATE
- DETECTORS BELOW GRID PLATE
- ▽ DETECTORS AT GRID PLATE LEVEL
- + GROSS MOTION DETECTORS

Figure 37. Detector Support Assembly Showing Detector Locations

steel tubes are in turn mounted on a grid plate which can slide vertically such that the detectors on an adjacent grid plate will mesh with each other. In this manner, the detectors can be axially located either 25.4 or 12.7 mm apart. Radially, the detectors are located approximately 10 mm apart. This type of spacing agrees with the spatial resolution capabilities of the detectors.⁴⁶ Overall, there will be 180 detectors in the entire test assembly. One hundred and eight of them are located between the fuel pins, and it is expected that they will provide high fuel motion resolution capabilities. Another 72 detectors are located just outside the test assembly for coarse resolution.

In addition, calculations were performed to estimate the radioactivity of the test assembly after extended use. The dose to a man working on the test assembly was also calculated. Two days after 25 reactor pulses of maximum intensity (at 5 shots per day for 5 days), the activity of the test assembly is 4.5 curies. The dose to an unprotected man at one foot from the test assembly is 5 rad/hr. When the worker is protected with a 25.4 mm lead body shield, the dose is 0.4 rad/hr. After two weeks, the doses are 0.9 and 0.07 rad/hr, respectively. These calculations indicate that, after a reasonable waiting time, minor modifications and adjustments to the test assembly can be made without serious risk to the worker.

Further design work on the test assembly is now focusing on two areas. First, the design of a pellet restraint mechanism is being considered. Simple calculations indicate that the rapid heating of the U-10 Mo pellets can cause the pellets to move with velocities near 20 M/s if they are not restrained. If they are restrained, then care must be taken that the restraining materials do not exceed their yield points. In the design under consideration, the cladding will absorb

the kinetic energy of the pellets. The second design effort is concerned with choosing cables and connectors to remove the fission couple signals from the test assembly.

5.2.2 Detector Research and Development

5.2.2.1 Fission Couple Detectors -

Tungsten and uranium (U^{238} 10 w/o Mo) chips are needed to fabricate the thin film fission couples. These chips must be approximately 0.25 mm square or wafers 0.25 mm in diameter and 0.25 mm long. For bonding purposes, thin films of gold must be deposited onto one surface of the chip. Several attempts to fabricate these chips and deposit the gold were made during the last quarter.

Tungsten is being used in the initial fabrication tests to establish the most successful technique. Basically, two methods of construction are being tried. In the first and most successful technique, 0.25-mm diameter tungsten wires are placed in a glass-epoxy-glass sandwich which is supported by a ceramic jig. A thin diamond saw is used to cut 0.25-mm thick slices, one side of which is then deposited with a gold film using a sputtering device. After the gold is deposited, the epoxy is dissolved and the chips retrieved. It is expected that this technique will be equally suitable for U-10 Mo wire as well. One metre of U-10 Mo wire 0.25 mm in diameter was purchased from LASL. The dicing technique will be tried on it next quarter.

In the second technique disks 0.25 mm in diameter were punched through 0.18-mm thick tungsten foil. A foil of U-10 Mo is not

available; nevertheless, the toxic machine shop is trying to slice a 0.50-mm thick wafer from a U-10 Mo rod and grind the wafer down to 0.25 mm. This disk could then be coated with gold on one side and sawed into 0.25 mm cubes or punched in the same manner as the tungsten.

Work was also done on determining techniques for bonding tungsten and uranium chips and the lead wires to the chromel or alumel thin films. The most successful technique at present uses a gold preform (98% Au - 2% Si) 0.025-mm thick to act as "glue" between the chips (or lead wires) and the thin films. A disadvantage of the gold-silicon preform is that it melts at 370°C which is below the desired operating temperature of the detector (500°C). In the future, a pure gold preform will be used in the bonding process. Another disadvantage of the gold preform is that it is thermodynamically undesirable to have so much gold between the heat source (i.e., the chips) and the thermocouple junction. However, this method will probably be acceptable since thermodynamic modeling predicts the response time of the thermocouple to a step-temperature rise in the tungsten or uranium of approximately 200 μs .

In the past, some problems with making the chromel and alumel films stick to the glass substrate were encountered. It was found that the film adherence to the glass substrate improved considerably with the quality of the cleaning process prior to depositing the thin films. The most successful cleaning processes found to date use a standard ultrasonic/chemical cleaning method together with either air firing at $450\text{--}500^{\circ}\text{C}$ or exposure to short wave ultraviolet light.

5.2.2.2 Diamond Detectors -

This quarter the primary effort was again directed toward solving the charge injecting contact problem. Experimentation was conducted with different types of metallic paints and ion-implantation techniques. The most successful diamond detectors to date are those which were fabricated by phosphorus implantation. This is interesting in that phosphorus is an electron injector, while the Russian experience indicates diamond responds better to hole injection. This point will be discussed in more detail later.

Several detectors were previously fabricated that counted without polarization; however, these detectors exhibited no measurable energy resolution.^{9,10} These detectors were fabricated with an Ag-frite paint and were heat treated in a graphite crucible in air. (This work verified that the Russian technique of heat treatment in graphite did yield good mechanical bonds between metallic paints and the diamond surface.) It was speculated at the time that the poor energy resolution was due to the thickness of the diamonds being greater than the electronic Schuebweg.

In the current work, thinner diamonds were used along with different metallic paints and heat treatments. The results are summarized in Table XIII. Several facts have emerged, however, and these are listed as follows:

- (a) The method of heat treatment produces good mechanically bonded contacts and therefore it should be expected that electrical contact is being made to the diamond.

TABLE XIII
Experiments with Metallic Paint Contacts

Specimen Identity	Surface Prep.	Inj. Contact Prep. Material Heat Treat.	Polarization	Signal/ Noise	Bias Voltage	Resolution	Current/ Voltage	Comments
3.10A* 3020+	DD#	Ag 2500A* by evapo- ration	300-1000°C for 6 hrs. vacuum					Contact vanished during heat treat. Optical trans. down 10%, but recovered after cleaning. Diamond destroyed on 2nd experiment due to a furnace failure.
3.10B 3020	DD	Ag 7095 frit paint	600°C/6 hr. vacuum	bad 10-15 sec.	poor	bipolar none 400- 1000V.	ohmic	Contact mechanically sound. Counts only at high voltages. Diamond was cracked on a 2nd experiment.
3.10D 3020	DD	Pt-Ag 3709 fritless paste	875°C/4 hr. vacuum	bad 30 sec.	poor	bipolar none 400- 1000V.	ohmic	Contact shrunk in diameter, but was mechanically sound even after etching in acid.
3.10C 3020	DD	Pt-Ag 3709 fritless paste	875°C/15 min. vacuum	bad 203 min.	poor	bipolar none 400- 1000V.	ohmic	Contact did not shrink, was mechanically sound, but did not etch the diamond very much.
3.10E 3020	DD	Ag 7095 frit paint	600°C/3 hr. air Etched prior to putting on blocking contacts.	bad	poor	bipolar none 400- 1000V.	ohmic	Contact was mechanically sound. Etching prior to evaporating Au blocking contact offered no improvement.
3.10F 3020	DD Annealed 1000°C O ₂ etch	Ag 7095 frit paint	600°C/3 hr. air O ₂ etch at 900°C/5 min.	bad	poor	bipolar none 400- 1000V.	ohmic	The contact balled after the O ₂ etch, but left a highly etched ² region.

*Identification code, .3mm dia. x 0.1mm thick. +Doubledde gem identification number.
#Doubledde surface cleaning procedure.

- (b) All detectors exhibited bipolar counting characteristics and in some cases the detectors functioned better with negative polarity. According to the Russian work, diamond detectors are unipolar devices and, in the case of the Ag contacts, the bias should be positive.
- (c) The detector behavior appears to be independent of the diamond thickness in the range 0.1 to 0.25 mm. (This is the thickness range of the Russian detectors.)

The main conclusion of the work to date on the metallic paint is that these hole injecting contacts appear to be incompatible with our diamond samples.

The ion-implantation experimental results are summarized in Table XIV. Both boron (hole injector) and phosphorus (electron injection) were tried. Different implant concentration profiles and dosages were employed. The various boron implants were unsuccessful, and the observed behavior of these implants was very similar to that found for the metallic paint contacts. The common denominator that surfaces is that hole injection does not work on our diamonds.

The phosphorous implants, on the other hand, were quite encouraging. The first two diamonds that were phosphorus implanted appeared to be heavily damaged with considerable swelling of the implanted surface. (The fact that phosphorus causes considerable damage is not unexpected, since the phosphorous atom is larger and heavier than a carbon atom.) The inconsistency of some of the properties of these two implanted diamonds can probably be attributed to the structural damage. The last phosphorous implant was

TABLE XIV

Experiments with Ion Implanted Contacts

Specimen Identity	Surface Prep.	Inj. Contact Prep.	Polarization	Signal/ Noise	Bias Voltage	Resolution	Current/ Voltage	Comments
3.10G 3020	DD*	Boron, Ag paste+ 1 peak, 60 keV Dose, 5×10^{14} ions/cm ² Anneal, 1000°C/4 hr.	poor	poor	bipolar 1 kV	none	ohmic	Lost identity of implanted surface after anneal. Tried both sides but results the same. No surface conductivity, either side.
3.10A 3025	DD	Boron, Ag paste 1 peak, 60 keV Dose, 5×10^{14} ions/cm ² Anneal, 1000°C/4 hr.	fair	fair	pos. 1 kV	poor, slight structure	ohmic	Lost identity of implanted surface after anneal, no surface conductivity. Results are very questionable. These implants will be repeated.
3.10B 3025	DD	Boron, Ag paste 1 peak, 60 keV Dose, 5×10^{14} ions/cm ² Anneal, 1000°C/4 hr.	fair, levels out at a detectable rate	poor	pos. 1 kV (bipolar)	poor	ohmic	No surface resistance on implant. Counts with either polarity, but is better with <u>positive</u> bias.
3.10C 3025	DD	Boron, Ag paste 1 peak, 60 keV Dose, 5×10^{14} ions/cm ² Anneal, 1000°C/4 hr.	fair, levels out at a detectable rate	poor	neg. 1 kV (bipolar)	poor	ohmic	No surface resistance on implant. Counts with either polarity, but is better with <u>negative</u> bias.
3.10D 3025	DE	Phosphorous, 2 peak, 60/150 keV Dose, 1×10^{15} ions/cm ² Anneal, 1000°C/4 hr.	good	fair	neg. 60GV. (bipolar)	poor to fair, can see structure	ohmic	Surface resistance across implant. Implant area dark (damage). Counts at lower voltages. Works better at <u>negative</u> bias.
3.10E 3025	DE	Phosphorous, 2 peak, 60/150 keV Dose, 1×10^{15} ions/cm ² Anneal, 1000°C/4 hr.	good	fair	bipolar 1 kV	fair, 30% neg. bias 10% pos. bias	non-ohmic	Surface resistance across implant. Implant area dark (damage). Works better at <u>positive</u> bias, but should be better at <u>negative</u> bias. Could be due to damage.

*Doubledee surface cleaning procedure.

+Ag paste was used to make contact with the implanted layer.

TABLE XIV (continued)

Specimen Identity	Surface Prep.	Inj. Contact Prep.	Polarization	Signal/ Noise	Bias Voltage	Resolution	Current/ Voltage	Comments
3.10A 3028	DD	Boron, Ag Paste 2 peak, 25/60 keV Dose, 1×10^{15} ions/cm ² Anneal, 1000°C/4 hr.	poor	poor	bipolar 1 kV	none	ohmic	No surface resistance on implant. Could not retain a residual count rate with either bias direction.
3.10B 3028	DD	Boron, Ag Paste 2 peak, 25/60 keV Dose, 5×10^{14} ions/cm ²	poor	poor	bipolar 1 kV	none	ohmic	No surface resistance on implant. Could not retain a residual count rate with either bias direction.
3.10C 3028	DD	Boron, Ag Paste 3 peak, 25/50/80 keV Dose, 5×10^{14} ions/cm ² Anneal, 1000°C/4 hr.	poor	poor	bipolar 1 kV	none	ohmic	No surface resistance on implant. Could not retain a residual count rate with either bias direction.
3.10D 3028	DD	Phosphorous, Ag Paste 4 peak, 30/60/90/ 150 keV Dose, 5×10^{14} ions/cm ² Anneal, 1000°C/4 hr.	excellent	excell. even at 50V bias	neg. 20 to 300V	20% at -150V bias	ohmic	Surface conductivity on implant, implant area only slightly dis- colored. Operates only with negative bias, as it should. This is the best detector to date.

performed at a lower dosage, and this detector exhibited very encouraging properties. The detector did not polarize, operated a very low voltage (20 volts), and was unipolar with the correct polarity. The resolution was 20 percent for an Am^{241} alpha source. (The Russians claim 3 to 10 percent energy resolution.) The only inconsistent behavior (with some of the Russian results) of this detector was the current-voltage characteristics. The 20 percent resolution value may be somewhat high because of residual damage. This diamond was still somewhat discolored after implantation. Experiments are now in progress to optimize the phosphorous implant parameters, and to verify that the phosphorous results are not a fluke.

The big surprise in this study, thus far, is that electron injection works. Most of the Russian diamond detectors responded to hole injection, which is consistent with the fact that nitrogen (the major impurity) produces defect states in the upper half of the band gap in diamond. These are electron traps and, therefore, hole injection is needed. A possible explanation of our results, which indicate the need for electron injection, is that the bulk of the Russian diamonds came from Siberia, while ours came from South Africa, two different geological formations. It is quite plausible that our diamonds contain very dilute amounts of boron, which would produce defect states in the lower half of the band gap. These are hole traps and would require electron injection to obtain a nonpolarizing detector. Experiments are planned to determine if there is boron in our diamonds.

5.2.3 Unfolding and Transport Calculations

The computer code called RHOPER (for ρ perturbation) described in the previous quarterly report¹⁰ was modified to include angle integrations using higher order spherical harmonic flux moments. It was found that increasing the order of angle integration from P_0 to P_1 only changed the answers by about 10 percent. Unfortunately, these changes did not remove the differences between the calculated detector response and the predicted response which is approximately two times greater. In searching for this discrepancy, both Monte Carlo and discrete ordinate transport calculations were used with RHOPER to determine detector responses. Both methods give the same results. Consequently, there is still a discrepancy between the results from perturbation theory and those from direct flux calculations. Additional studies of differences will be carried out in the next quarter.

5.3 Electron Beam Flash X-Radiography

During this quarter, fabrication began on the coded source flash X-radiography chamber described in the last quarterly report. The major components have been received and some initial testing of component performance has begun. Experiments aimed at imaging a fuel pin using the coded source radiographic technique are planned for the next quarter on the Sandia HERMES-II accelerator. Other experiments are also planned with the purpose of imaging a small bundle of fuel pins with the Linatron 1500 using the conventional radiographic technique.

The evaluation of the effects of photon buildup upon spatial and areal density resolution capabilities in poor geometry situations continued this quarter. Experiments are being conducted on the Linatron 1500

using the conventional radiographic technique with x-ray film as the detector. Further experiments are planned using an active detector system. Most of these experiments should be completed during the next quarter.

An active detector system is being designed for use in the flash x-ray cinematography fuel motion detection program. The system will consist of a scintillator, an optical image intensifier, and a fast framing camera. Several different scintillator candidates are being considered, although NaI appears to be the most likely candidate at this time.

5.4 Gamma-Ray Coded Aperture Imaging in Many-Pin Bundles

Gamma-ray coded aperture imaging technology is progressing very satisfactorily, as shown by recent experiments on the Sandia SPR-II and ACPR reactors. The feasibility of extending this technique to many-pin bundles involves the questions of how the image quality will be affected by scattering and absorption of the gamma radiation in the fuel, clad, sodium and experiment containment, and how large the signal from the far side and center of the test region will be in comparison to the signal from the near side of the test fuel. It is likely that changes in the coded aperture design and the scintillator can be made to decrease the effects of scattering and absorption.

A series of gamma-ray transport calculations was therefore begun during this quarter to study these scattering and absorption effects. A 37-pin bundle was chosen as a reasonable scale-up from the seven-pin bundles to be studied in the ACPR experiments. The test geometry was approximated by a 5.5-cm thick homogeneous slab of material having the

same average composition as the CRBR. On the front and back of this slab were placed 2.54-cm thick slabs of stainless steel to approximate the containment and various walls in the coded aperture apparatus. Although a slab geometry probably overemphasizes scattering effects near the boundaries of the test fuel region, it is much faster in performing calculations and it should be reasonably accurate for the situation along the diameter of a large bundle.

The problem was decomposed into two parts: the calculation of the gamma ray spectra emerging from the test assembly (using the code TIGER),⁴⁶ and the calculation of the variations in energy deposition in the scintillator after the emergent gamma rays have passed through a cylindrically-symmetric coded aperture of arbitrary design (using the code CYLTRAN).⁴⁷ The calculation of the scintillator signal is lengthy and expensive, so each CYLTRAN calculation will use a point source of gamma rays with an "emergent" spectrum (either primary or scattered, as obtained from TIGER) corresponding to a particular depth in the test geometry.

The coded aperture imaging system images unscattered fission gamma rays at the position where they were emitted, and images scattered gamma rays at the position where they were last scattered. The unscattered gamma rays locate the position of the test fuel accurately, and the scattered gamma rays either further illuminate the test fuel, or can illuminate and contribute to the imaging of other components, such as the stainless steel and sodium of the test region. Therefore, it is important to distinguish between these two components of the emerging gamma ray spectrum in treating the test fuel region. The TIGER code was therefore modified to tally separately the spectra of fission gamma rays which originate within an arbitrary slab of test fuel and emerge from the test fuel region and its containment without further scattering

(the uncollided primary flux from the chosen slab). The modified code also tallies the spectra of gamma rays which last scattered in an arbitrary slab of test fuel or test containment and then emerged from the test fuel region and its containment without further scattering (the scattered flux from the chosen slab).

Figure 38 shows results of TIGER calculations for a 5.5-cm thick slab of core-average CRBR material surrounded on each side by 2.54-cm thick slabs of stainless steel. The CRBR material is divided into ten regions, and each steel slab into two regions, and the photons tallied in Fig. 38 cover the energy range of $0.01 \text{ MeV} \leq E \leq 6 \text{ MeV}$. Over this energy range, the total scattered photon flux is actually 63 percent larger than the total primary uncollided flux, largely because of the scattering in the front steel slab. The ratio of the uncollided flux from the fuel closest to the detector to that from the fuel farthest from the detector is 7.45.

Figure 39 shows the photon fluxes obtained if some means were used to remove the low-energy portion of the spectrum, so that the energy range is $0.60 \text{ MeV} \leq E \leq 6 \text{ MeV}$. The total scattered flux dropped by a factor of 2.5, so that it was only 76 percent of the uncollided flux, yet the uncollided flux decreased by only 13 percent. The ratio of the uncollided flux from the fuel closest to the detector to that from the fuel farthest from the detector also decreased to 5.8. These results indicate that a filter placed in the coded aperture system may tend to equalize signals from fuel regions at different depths, and also change the relative amount of signal from the steel containment substantially, with a small sacrifice in signal strength.

The CYLTRAN calculations of the scintillator signal strengths should permit realistic calculation of the effects of various filters and coded

Figure 38. Escaping Photons (Photons/SR-Primary Photon)

Figure 39. Escaping Photons (Photons/SR-Primary Photon), $E > 0.6$ MeV

aperture designs. Preliminary CYLTRAN calculations have been performed, but checkout of the details is still continuing. Some final CYLTRAN results should be obtained during the next quarter.

6. ACPR FUEL MOTION SYSTEM

(J. G. Kelly, 5423; K. T. Stalker, 2541)

6.1 Additional Results of the Fuel Pin Imaging Experiment at SPR II

In the previous quarterly report, details of the SPR II imaging experiment were presented. Images of 20% and 93% enriched fuel pins were obtained from pseudoholograms recorded on x-ray film and with an x-ray image intensifier. The pins were observed through a collimator mounted in the reactor shield wall. The reactor was placed just outside the collimator field of view next to the wall. The reconstructed images represent a major milestone in the fuel motion system development because they have demonstrated that fuel pins can be imaged with an active recording system in the presence of a reactor.

Subsequently, reactor time became available for careful measurements of the thermal neutron fluence which existed in the vicinity of the pin during the imaging experiments. The pins had been placed between two 5-cm thick plates of polyethylene for thermal neutron enhancement. The measurements showed that the thermal neutron fluence was 2×10^{11} n/cm² at the pins for a 300°C burst on the reactor. This was a great surprise to us because the estimated fluence had been expected to be 3×10^{13} n/cm². Investigations uncovered two errors. First, some earlier fluence measurements upon which the estimates had been based were incorrect and, second, the polyethylene blocks contained 5% by weight of boron. Measurements with pure polyethylene blocks indicated a fluence of 2×10^{12} n/cm². Thus, if the imaging experiments had been assembled properly, the neutron induced signal would have been ten times larger.

Our estimates now indicate that the largest component of signal came from reactor core γ rays scattered from the pin. Some neutron induced signal was present, however, because the 20% enriched pin image was weaker than that of the 93% enriched pin. The fact that the fuel pin was imaged despite the very small signal (much smaller than expected during a single frame in a fuel motion experiment at ACPR) shows that this fuel motion detection system has an even greater potential for high-speed fuel motion measurements than was initially expected.

6.2 Collimation Test Module Experiment at ACPR

The collimation test module experiment (CTM) was approved by the Sandia Safeguards Committee late in April and began May 11. The experiment was designed to determine whether an adequate signal-to-back-ground ratio could be obtained that would allow clear imaging of a fuel pin in the central test chamber. The CTM is illustrated in Figure 40. The x-ray film (protected in a lead housing) observed the pin through the coded aperture mounted in the collimator which penetrates the shielding and the reactor core.

Although the mechanical assembly procedures were very complicated because of the restricted access to the region where the CTM had to be placed, the setup of the experiment went smoothly. Gamma ray dose and neutron fluence measurements were first made with a thick lead plug in the collimator and no fuel pin or its canister in the central test chamber. During an 83-minute run at 20 kW (the fluence at the pin would provide 500 cal/g of fission energy), calibrated dosimeters measured about 15 rads of γ radiation at the detector plane, where the beam axis would be, and about 10 rads near the inside walls of the film chamber. This compares with an expected fuel pin signal of about 6 rads.

Figure 40. Collimation Test Module

The plug was then replaced by the collimator and coded aperture. When Type M x-ray film exposed to the radiation in the detector region was developed, a dark rectangular region correctly outlining the collimator field of view was seen. Indeed, some structure associated with the core fuel rods on the other side of the core was visible.

Next, a canister designed to hold a fuel pin for prompt burst excursion experiments was inserted. Along with some Al containment walls, this canister had a 3-cm thick polyethylene liner and a small diameter inconel tube to hold the pin. In order to reduce background radiations, the cylindrical polyethylene liner was modified as shown in Figure 41. The region behind the fuel pin was replaced by 3.8 cm of Pb and a viewing slot was cut on the observation side. Exposures without the pin itself in place showed that the Pb was very effective in reducing the background radiation coming through the collimator.

The 20% enriched fuel pin was then inserted, and after exposure the pseudohologram of a rod-shaped object was clearly seen on the x-ray film. The primary objective of the experiment was achieved by this result. The visibility of the pin pseudohologram structure indicates that information which is required (the signal) is sufficiently above backgrounds to allow determination of fuel motion in the simulated core disruptive accidents planned for ACPR. The engineering of the fuel motion system is now the principal remaining task. The feasibility of the approach has been demonstrated.

During the remaining days of the experiment, a number of other investigations were carried out to evaluate the properties of the system so that the final shield and collimation structures can be more easily optimized. Some of these were the following.

Figure 41. Modified Cylindrical Polyethylene Liner

- (a) Dosimetry revealed that the signal was indeed about 6 rads.
- (b) Pseudohologram formation was only slightly degraded by -
 - (1) the addition of 3 cm of polyethylene into the slot opening (Figure 41),
 - (2) the addition of 1.27 cm of extra steel into the collimated beam,
 - (3) the opening of a 2 cm gap in the lead shield (between the film chamber and the lead wall), and
 - (4) the substitution of water for the NaBF_4 solution.
- (c) Detection with Type AA film sandwiched between .013-cm thick Ta sheets produced a good pseudohologram during a 4 minute run at 10 kW power. This is the exposure which would be achieved during a 168 μs period in a burst experiment depositing a total of 500 cal/g in the test pin.
- (d) Although Type R film is a relatively slow film, it provided higher contrast shadowgrams and finer grain structure than the faster films.
- (e) Ta sheets next to the film both enhance the signal and attenuate low-energy background radiations.
- (f) A pinhole aperture (2mm x 5mm) was inserted into the collimator channel and a good pinhole picture of the pin was also obtained. This aperture produced lower contrast and required a somewhat longer recording period, however.

In the next quarterly we plan to include some of the reconstructed fuel pin images.

6.3 Detector Resolution Studies

In an effort to better determine the spatial resolution values

for the various detectors presently used in the coded aperture imaging system, a series of exposures were made using a sharp edged beam from the Cf^{252} source. The sharp edged beam was produced by shadowing the γ rays with a lead brick having a large radius of curvature. Exposures were made using Kodak AA Lead Pack film, the Varian X-Ray intensifier, and the EMI optical intensifier in conjunction with a 1.0-mm thick NaI(Tl) scintillator screen as the convertor (the discrete system). The output image from the X-Ray intensifier and discrete system were recorded on TR1-X film. The images were then digitized using a micro-densitometer. The results of the rough scans are shown below (Fig. 42). The density scales have been adjusted to normalize the step height and this plot gives only a rough comparison of the three detectors, but it is clear that the resolution of the system employing the 1.0-mm thick NaI(Tl) scintillator is on the order of 1.5 mm, while the Varian X-Ray intensifier and AA Lead Pack film are much superior. A visit was made to the supplier to discuss the possible reasons for this problem and the potential resolution available from NaI(Tl) scintillators. From these discussions and our own understanding of the possible causes of the poor resolution, a series of experiments to test the effect of surface polish parameters on resolution have been planned.

6.4 Computer Reconstruction

In an attempt to optimize the computer reconstructions of the SPR II data, various schemes for dc subtraction prior to reconstruction were investigated. Although the work is not complete, preliminary findings indicate that some advantage may be gained in using a line average (the average over a single reading pass across the film) as opposed to the whole picture average value previously used. Also, because of the assymetrical background present in the pseudoholograms recorded at

Figure 42. Rough Scans of Spatial Resolution Values Obtained by Coded Aperture Imaging System Detectors

SPR II due to the reactor being placed to one side of the fuel pin, it appears that some kind of local averaging would provide the best results. Several attempts at this type of averaging have been made using various moving window schemes, but a truly optimum result has not been obtained to date. The difficulty arises because of the changing periodicity of the recorded information as one scans across the pseudohologram because of the changing zone widths in the pseudohologram. This can be corrected by varying the window lengths as one proceeds across the pseudohologram, but an optimum method of implementing this algorithm without operator intervention has not yet been found. Further work along this line, as well as investigations of various reconstruction schemes which might produce better signal-to-noise ratios than the simulated Fresnel diffraction method of reconstruction, will be performed in the near future.

7. ADVANCED REACTOR SAFETY RESEARCH ASSESSMENT

(J. W. Hickman, 5412; M. Rios, Jr., 5412)

7.1 Introduction

This task is directed toward investigation of the applicability of the methodology developed by the Reactor Safety Study (RSS) to a liquid metal fast breeder reactor (LMFBR) or other advanced reactor plants. The RSS methodology will be adapted to construct, display, and evaluate accident sequence diagrams for advanced reactor systems. Event tree logic will be used to develop and display the accident sequences. Ongoing studies of advanced reactor plant characterization and engineered systems analysis, as well as those investigating accident phenomenology, will be reviewed to identify material applicable to this program. Further investigations will be performed as required to achieve the appropriate accident sequence logic. Suitable release categories will be developed for advanced reactors based on source term characteristics. Expected dominant accident sequences for each category will be identified. This is the first report of this new task.

7.2 Progress

Characterization of the engineered safety systems is sufficiently complete to allow a preliminary list of core and ex-core accident initiating failures to be categorized. A limited number of preliminary event trees have been constructed and tentative accident categories have been identified. In support of this task, personnel assignments have been made and work plans are under preparation. Further, related

work previously accomplished is being evaluated to determine the degree to which it can contribute to the present effort, and available codes and models are being reviewed to determine their adaptability to this program.

REFERENCES

1. Experimental Fast Reactor Safety Research Program (Combined) Quarterly Report July - December 1974, SAND75-0068, Simulation Sciences Research Department, Sandia Laboratories, Albuquerque, New Mexico, February 1975.
2. Experimental Fast Reactor Safety Research Program Quarterly Report January - March 1975, SAND75-0225, Simulation Sciences Research Department, Sandia Laboratories, Albuquerque, New Mexico, April 1975.
3. Experimental Fast Reactor Safety Research Program Quarterly Report April - June 1975, SAND75-0449, Simulation Sciences Research Department, Sandia Laboratories, Albuquerque, New Mexico, August 1975.
4. Experimental Fast Reactor Safety Research Program Quarterly Report July - September 1975, SAND75-0567, Reactor Research and Development Department, Sandia Laboratories, Albuquerque, New Mexico, December 1975.
5. Fast Reactor Safety Research Program Quarterly Report October - December 1975, SAND76-0157, Reactor Research and Development Department, Sandia Laboratories, Albuquerque, New Mexico, April 1976.
6. Fast Reactor Safety Research Program Quarterly Report January - March 1976, SAND76-0273, Reactor Research and Development Department, Sandia Laboratories, Albuquerque, New Mexico, July 1976.
7. Fast Reactor Safety Research Program Quarterly Report April - June 1976, SAND76-0372, Reactor Research and Development Department, Sandia Laboratories, Albuquerque, New Mexico, September 1976.
8. Fast Reactor Safety Research Program Quarterly Report July - September 1976, SAND76-0652, Reactor Research and Development Department, Sandia Laboratories, Albuquerque, New Mexico, January 1977.
9. Fast Reactor Safety Research Program Quarterly Report October - December 1976, SAND77-0145, Reactor Research and Development Department, Sandia Laboratories, Albuquerque, New Mexico, March 1977.
10. Fast Reactor Safety Research Program Quarterly Report January - March 1977, SAND77-0611, Advanced Reactor Research Department, Sandia Laboratories, Albuquerque, New Mexico, September 1977.
11. R. W. Ohse, P. G. Berrie, H. G. Bogensberger and E. A. Fisher, J. Nucl. Mat. 59, 112 (1976).
12. M. Bober, H. V. Karow, K. Schretzmann, Nuclear Technology 26, 237 (1975).
13. D. A. Benson, W. H. Buckalew, ANS Transaction 23, 325 (1976). See also D. A. Benson, SAND77-0429, Sandia Laboratories, Albuquerque, NM June 1977.

References (cont'd):

14. K. O. Reil, Effective Equation of State Measurements on Uranium Dioxide, Ph.D. Thesis, Univ. of New Mexico, Albuquerque, NM, May 1977.
15. T. R. Schmidt, LMFBR Prompt Burst Excursion Experiments in the Annular Core Pulsed Reactor, Proceedings of International Meeting on Fast Reactor Safety or Related Physics, October 5-8, 1976.
16. S. M. Zivi, M. Epstein, R. W. Wright et al, Nucl. Sci. and Eng. 56, 229, 240 (1975).
17. J. F. Kerrisk and D. G. Clifton, Nucl. Technology 16, 531 (1972).
18. L. Leibowitz et al, J. Nucl. Notes 39, 115 (1971).
19. D. L. Booth, UDAEA Report TRG 1759 (1968).
20. E. Gorham-Bergeron, Sandia Laboratories, personal communication.
21. P. Majumdar and D. J. Hsu, "A Computer Program for Transient Freezing Flow into a Channel and Its Application to LMFBR Safety Analysis," International Conference on Fast Reactor Safety and Related Physics, Chicago, IL, October 1976.
22. R. W. Ostensen, et al, "The Transition Phase in LMFBR Hypothetical Accidents," ibid.
23. M. Epstein, et al, "Transient Freezing of a Flowing Ceramic Fuel in a Steel Channel," Nuclear Science and Engineering 61, 310-323 (1976).
24. M. Epstein, et al, "Analytical and Experimental Studies of Transient Fuel Freezing," International Conference on Fast Reactor Safety and Related Physics, Chicago, IL, October 1976.
25. C. R. Bell, et al, SIMMER-I: An Sn, Implicit, Multifield, Multi-component, Eulerian, Recriticality Code for LMFBR Disrupted Core Analysis, LA-NUREG-6467-MS, Los Alamos Scientific Laboratory, Los Alamos, NM, 1977.
26. S. J. Hakim and R. J. Henninger, "A Hydrodynamic Model for Treating Transient Freezing in Pin Bundles," International Conference on Fast Reactor Safety and Related Physics, Chicago, IL, October, 1976.
27. Hartmut V. Wider, An Improved Analysis of Fuel Motion During an Overpower Excursion, Ph.D. Dissertation in Nucl. Eng., Northwestern University, 1974.
28. Supplement #5, Preliminary Safety Analysis Report, Clinch River Breeder Reactor, Project Management Corporation, October 1975.
29. C. L. Wheeler, et al, COBRA4I: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores, BNWL-1962, Battelle Pacific Northwest Laboratory, Richland, WA, March 1976.
30. William T. Sha, et al, "Boundary Value Thermal-Hydraulic Analysis of a Reactor Fuel Rod Bundle," Nucl. Sci. Eng. 59, 140-160 (1976).

References (cont'd)

31. C. L. Wheeler (BNWL), Private Communication (March 1977).
32. Preliminary Safety Analysis Report, Clinch River Breeder Reactor, Project Management Corporation, Chapter 4 (1975).
33. Ibid., p. 4-3-124.
34. Paul S. Pickard, Sandia Laboratories, private communication, February 1977.
35. W. G. Schuetzenduebel, "High-Temperature Gas Cooled Reactor Steam Generator Design," Nuclear Technology, Vol. 25, March 1976, pp. 315.
36. G. E. Lockett and R. A. Huddle, "The Dragon Reactor," Nuclear Power, February 1960, p. 112.
37. Ultra High Temperature Reactor Experiment (UHTREX) Facility Description and Safety Analysis Report, LA-3556 (Rev.), Los Alamos Scientific Laboratory, Los Alamos, NM,
38. Henry C. Monteith, Preliminary Design of the Cooling System for the High Fluence Fast Pulsed Reactor (HFFPR), Sandia Laboratories, Albuquerque, NM, SAND77-0963, to be published.
39. K. D. Lathrop, DTF IV, A Fortran-IV Program for Solving the Multi-group Transport Equation with Anisotropic Scattering, LA-3373, Los Alamos Scientific Laboratory, Los Alamos, NM, July 1965.
40. G. E. Hansen, W. H. Roach, G. I. Bell, J. J. Devaney, C. B. Mills, and L. C. Connolly, "Los Alamos Group-Averaged Cross Sections," LAMS 2941, Los Alamos Scientific Laboratory (September 1963).
41. Mechanical Technology Incorporated, 928 Albany-Shaker Road, Latham, New York, private communication.
42. Light Water Reactor Safety Research Program Quarterly Report January - March 1976, SAND76-0369, Nuclear Fuel Cycle Safety Research Department, Sandia Laboratories, Albuquerque, NM, September 1976.
43. Light Water Reactor Safety Research Program Quarterly Report April - June 1976, SAND76-0677, Nuclear Fuel Cycle Safety Research Department, Sandia Laboratories, Albuquerque, NM, February 1977.
44. M. W. Callabresi and S. T. Heidelberg, "SASL - A Finite Element Code for the Static Analysis of Axisymmetric and Plane Solids Subjected to Axisymmetric and Plane Loading," SCL-DR-720061, Sandia Laboratories, Albuquerque, NM, 1972.
45. T. F. Petersen, "TA02D - A General Purpose Two-Dimensional Heat Transfer Computer Code--User's Manual," GA-8008, Gulf General Atomics, 1969.
46. J. A. Halbleib, Sr., and W. H. Vandevender, Nucl. Sci. Eng. 57, 94 (1975).
47. J. A. Halbleib, Sr., and W. H. Vandevender, Nucl. Sci. Eng. 61, 288 (1976).

DISTRIBUTION:

U. S. Nuclear Regulatory Commission (244 copies for NRC-7)
Division of Document Control
Distribution Services Branch
7920 Norfolk Avenue
Bethesda, MD 20014

Division of Reactor Safety Research (8)
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555
Attn: C. N. Kelber, Assistant Director, Advanced Reactor Safety Research
M. Silberberg, Chief, Experimental Fast Reactor Safety Branch
R. W. Wright, Experimental Fast Reactor Safety Branch (6)

L. S. Rubenstein
Analysis Section
Program Support Branch
Office of Nuclear Reactor Regulation
Mail Station: Nickelsen Lane
Washington, D. C. 20555

Division of Military Applications (2)
Office of Safety and Facilities
USERDA
Washington, D. C. 20545
Attn: Cmdr. Frank Hughes
H. M. Busey

L. M. Groover
Program Analysis and Budget
Division of Military Applications
USERDA
Washington, D. C. 20545

Reactor Safety Research Coordination (4)
USERDA
Washington, D. C. 20545
Attn: R. W. Barber, Acting Director (3)
T. E. McSpadden, Project Manager

J. R. Roeder, Director
Operational Safety Division
USERDA/ALO
Albuquerque, New Mexico 87115

Special Program Division (2)
USERDA/ALO
Albuquerque, New Mexico 87115
Attn: C. B. Quinn
G. W. Johnson

A. DeVolpi
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

Distribution (cont'd):

R. A. Bari, Group Leader
Safety Evaluation Group
Fast Reactor Safety Division
Brookhaven National Laboratory
Upton, L.I., New York 11973

J. D. Orndorff
Los Alamos Scientific Laboratory
Group R-5, MS 560
Los Alamos, New Mexico 87545

University of California
Energy and Kinetics Department
5530 Boelter Hall
Los Angeles, California 90024
Attn: W. E. Kastenberg

Technical Library
Attn: James Hunter
Idaho National Engineering Laboratory
Idaho Falls, Idaho

A. W. Cronenberg
Department of Nuclear Engineering
University of New Mexico
Albuquerque, New Mexico 87131

Y. S. Tang
Westinghouse Corporation
Advanced Reactors Division
P. O. Box 158
Madison, Pennsylvania 15663

B. R. Sehgal
Electrical Power Research Institute
3412 Hillview Avenue
P. O. Box 10412
Palo Alto, California 94303

General Electric Corporation (5)
310 De Guigne Drive
Sunnyvale, California 94086
Attn: J. O. Bradfute, Manager, Dynamics and Safety
A. S. Gibson, Manager, Product Engineering
R. Greebler, Manager, Reactor Studies
J. E. Mott, Senior Engineer, Dynamics & Safety
M. I. Temme, Manager, Safety Analysis

W. E. Nyer
P. O. Box 1845
Idaho Falls, Idaho 83401

H. B. Piper
CRBR Project Office
P. O. Box U
Oak Ridge, TN 37830

Distribution (cont'd):

M. H. McTaggart
MOD (PE) AWRE
Aldermaston
Berkshire
England

J. R. Findley
Building 429
AERE
Harwell, Didcot
Berkshire
England

A. R. Baker
FRSD
UKAEA
Risley
Warrington
England

T. Ginsberg
Fast Reactor Safety
Brookhaven National Laboratory
Upton, Long Island, New York 11973

1000 G. A. Fowler
1100 C. D. Broyles
1120 G. E. Hansche
1126 G. L. Ogle
1130 H. E. Viney
1136 J. H. Davis
1136 J. A. Brammer
1136 H. C. Walling
1200 W. A. Gardner
1260 K. J. Touryan
1261 D. F. McVey
1262 H. C. Hardee
1350 R. G. Clem (2)
2151 F. R. Kroeger
2540 O. M. Stuetzer
2541 G. W. Gobel
2541 K. T. Stalker
3284 R. P. Tyler
5000 A. Narath
5100 J. K. Galt
5110 F. L. Vook
5120 G. J. Simmons
5130 G. A. Samara
5150 J. E. Schirber
5160 W. Herrmann
5167 B. M. Butcher
5231 J. H. Renken
5231 J. A. Halbleib
5231 P. J. McDaniel
5231 J. E. Morel
5400 A. W. Snyder (2)
5410 D. J. McCloskey

5411 D. H. Dahlgren
5412 J. F. Muir
5412 M. Rios, Jr.
5420 J. V. Walker (10)
5421 T. R. Schmidt
5422 R. L. Coats (10)
5422 H. G. Plein
5422 K. O. Reil
5422 J. B. Rivard
5422 D. J. Sasmor
5423 J. E. Powell (10)
5423 G. A. Carlson
5423 L. M. Choate
5423 L. R. Edwards
5423 J. G. Kelly
5423 D. A. McArthur
5423 A. R. Phillips
5423 H. L. Scott
5423 W. H. Sullivan
5423 S. A. Wright
5424 J. A. Reuscher
5424 P. S. Pickard
5425 M. F. Young
5425 D. C. Williams
5425 D. H. Nguyen
5425 R. W. Stensen
5430 R. M. Jefferson
5431 R. E. Nickell
5440 R. W. Lynch
5443 R. E. Luna
5452 L. D. Posey
5800 R. S. Claassen
5820 R. L. Schwoebel
5822 R. E. Whan
5830 M. J. Davis
5831 N. J. Magnani
5831 D. A. Powers
5832 R. W. Rohde
5847 C. H. Karnes
8266 E. A. Aas
8300 B. F. Murphey
8310 D. M. Schuster
9330 A. J. Clark, Jr.
9337 N. R. Keltner
3141 C. A. Pepmuller (Actg.) (5)
3151 W. L. Garner (3)
For ERDA/TIC, Unlimited Release
3172-3 R. P. Campbell (25)
For ERDA/TIC, Unlimited Release