FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
JULY, AUGUST, SEPTEMBER 1968

DECEMBER 1968

**AEC RESEARCH &
DEVELOPMENT REPORT**

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

operated by

BATTELLE MEMORIAL INSTITUTE

for the

UNITED STATES ATOMIC ENERGY COMMISSION UNDER CONTRACT AT(45-1)-1830

3 3679 00061 1998

BNWL-917
UC-80, Reactor
Technology

FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
JULY, AUGUST, SEPTEMBER 1968

By

E. R. Astley
Project Manager

FIRST
DISTRIBUTION JAN 3'69

December 1968

BATTELLE MEMORIAL INSTITUTE
PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON 99352

Printed in the United States of America
Available from
Clearinghouse for Federal Scientific and Technical Information
National Bureau of Standards, U.S. Department of Commerce
Springfield, Virginia 22141
Price: Printed Copy \$3.00; Microfiche \$0.65.

FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
JULY, AUGUST, SEPTEMBER 1968

ABSTRACT

This report was prepared by Pacific Northwest Laboratory (PNL) under Contract No. AT(45-1)-1830 for the Atomic Energy Commission, Division of Reactor Development and Technology. It is a report on progress for the Fast Flux Test Facility, during the period July, August, and September 1968.

CONTENTS

LIST OF FIGURES	vii
LIST OF TABLES	xi
INTRODUCTION	xiii
I. FFTF OVERALL PROGRAM PROGRESS	1.1
A. Selection of Reactor Concept	1.1
B. Physics	1.1
C. Metallurgy	1.2
D. Coolant Development	1.2
E. FFTF Fuels	1.3
1. Development.	1.3
2. Fuel Temperature Analyses	1.4
F. Reactor Core Studies	1.4
Control Rod Poison Materials	1.4
Core Meltdown	1.4
Inlet Model	1.5
Sensor Development	1.5
G. Mathematical Model of Heating and Ventilating System.	1.5
II. SYSTEMS.	2.1
A. Reactor Plant Design	2.1
1. Reactor Design.	2.1
2. Heat Transport and Irradiation Testing.	2.4
3. Reactor Plant Instrumentation.	2.6
B. General Plant Design	2.14
1. Structures, Utilities and Containment	2.14
2. General Plant Instrument and Electrical System	2.14
3. Other General Plant Design.	2.17
III. GENERAL TECHNOLOGY	3.1
A. Design Evaluation	3.1
1. Reactor Evaluation	3.1
2. Plant Evaluation	3.15

B.	Reactor and Systems Analysis	3.18
1.	Systems and Instrumentation Analysis	3.18
2.	Reactor Core Analysis	3.27
C.	Reactor Physics and Operations	3.36
1.	Nuclear Analysis.	3.36
2.	Experimental Physics	3.38
3.	Radiation and Shield Analysis	3.43
D.	Safety and Reliability Analysis	3.50
1.	Reliability Analysis	3.50
2.	Safety Analysis	3.52
IV.	DEVELOPMENT	4.1
A.	FFTF Fuels	4.1
1.	Fuel Fabrication Process Demonstration	4.1
2.	Fuel Process Development and Irradiation Test Fabrication.	4.2
3.	Irradiation Testing.	4.3
4.	Other Work in Progress.	4.11
B.	Engineering Development.	4.12
1.	Sodium Components Development	4.12
2.	Equipment Development	4.12
3.	Development Facilities.	4.16
4.	Other Work in Progress.	4.20
C.	Coolant and Process Development	4.21
1.	Corrosion and Chemistry	4.21
2.	Thermal Hydraulics	4.28
3.	Instrumentation Development Section	4.30
4.	Other Work in Progress.	4.37
D.	Metals and Materials Development.	4.38
1.	Materials Properties	4.38
2.	Fabrication Development	4.48
3.	Component Material Surveillance.	4.51

FIGURES

2.A.1	Flux/Flow Trip Circuit for Reactor Primary Sodium Loops	2.11
2.A.2	Closed Loop Primary Sodium Flux/Flow Trip Circuit	2.12
2.B.1	FFTF Ventilation Flow Diagram	2.15
3.A.1	Feature Identification for Concept I	3.3
3.A.2	Feature Identification for Concept II	3.4
3.A.3	Feature Identification for Concept III	3.5
3.A.4	Feature Identification for Concept IV	3.6
3.A.5	Feature Identification for Revised Concept I-A	3.7
3.A.6	Instrument Lance/Open Position Installation, Concept III	3.9
3.A.7	Safety Rod Drive Mechanism, Concept III	3.10
3.A.8	Tubesheet Seat Design	3.12
3.A.9	Proposed 61-Pin Test Assembly	3.13
3.A.10	Proposed Instrumentation and Fuel Pins	3.14
3.B.1	Reactor Power and Temperature Responses for 3-ft ² Pipe Rupture near Vessel Inlet, Advanced Carbide Core, 3-Loop, Full Power Operation, with Check Valve Closing in 3 Seconds	3.23
3.B.2	Primary Loop Flow Rates for 3-ft ² Pipe Rupture near Vessel Inlet, Advanced Carbide Core, 3-Loop, Full Power Operation, with Check Valve Closing in 3 Seconds	3.24
3.B.3	Sodium Levels, Break Pressure, and Integrated Break Flow for 3-ft ² Pipe Rupture near Vessel Inlet, Advanced Carbide Core, 3-Loop, Full Power Operation, with Check Valve Closing in 3 Seconds	3.25

3.B.4	Temperature Peaking as Represented by the Maximum Circumferential Temperature Difference of the Coolant as a Function of Axial Distance Along the Fueled Portion of the Hottest Pin with the Wire Wrap Located in the Gap	3.29
3.B.5	Circumferential Temperature Distribution in Cladding for Radius Fixed at Midcladding Value and for a Mixing Constant of $k = 5$	3.30
3.C.1	Phase B ZPR-III Assembly 51	3.40
3.C.2	Fuel Handling Scheme	3.42
3.D.1	FFTF Protection System	3.55
4.A.1	Neutron Radiographs of Lower Third of Capsules PNL 59-5 and 59-6	4.5
4.A.2	Neutron Radiographs of Center Third of Capsules PNL 59-5 and 59-6	4.5
4.A.3	Neutron Radiographs of Upper Third of Capsules PNL 59-5 and 59-6	4.5
4.A.4	GETR-TREAT Transient Analysis ARGUS Code with Transient No. 767 and Melt Area 70%	4.8
4.A.5	GETR-TREAT Transient Analysis ARGUS Code with Transient No. 926 and Melt Area 20%	4.9
4.B.1	Closed Loop Top Face Hardware Arrangement	4.14
4.B.2	Closed Loop for Concept I	4.15
4.C.1	Variation of Reaction Rates with Temperature	4.27
4.C.2	Gamma Effects on Relative Sensitivity of Low Level Neutron Detectors	4.32
4.C.3	Geometry of Gamma Test Facility	4.33
4.C.4	Tungsten Thermocouple Deviations Capsule BT-1 and BT-2	4.34
4.D.1	Arrhenius Plot of As-Fabricated Controls and Postirradiation Creep Tests on Type 304 Stainless Steel	4.41

4.D.2	Summary of Available Data on Fast Reactor-Induced Swelling	4.42
4.D.3	As-Irradiated Cavity Distribution of Type 304L Stainless Steel Irradiated in the EBR-II at 790 °F (421 °C) to a Fast Fluence of 4.8×10^{22} n/cm ² (E > 0.1 MeV) Measured in Foils of Thickness from 250 to 1200 Å	4.44
4.D.4	Creep Stress Versus Temperature-Compensated Rupture Life (Theta Parameter) for Type 304 Stainless Steel	4.46
4.D.5	Micrograph Showing As-Received B ₄ C Particles	4.47
4.D.6	Micrograph Showing Molten Variation in As-Received B ₄ C Particles	4.47
4.D.7	Micrograph Showing Cracks in B ₄ C Particles Exposed to 500 °C Sodium for 100 hours	4.49
4.D.8	Micrograph Showing Cracks in B ₄ C Particles Exposed to 500 °C Sodium 1000 hours	4.49
4.D.9	Hexagonal Tube Creep Change in Tube Dimensions with Time	4.50

TABLES

2.B.I	FFTF Inert Gas Selections	2.19
3.A.I	Comparison of Concept Features	3.2
3.B.I	Initial Core and Advanced Core Transient Studies	3.19
3.B.II	Temperatures for Most Severe Shutdown Conditions Allowable	3.26
3.B.III	Energy Release in Molten Fuel-Sodium Interactions	3.31
3.B.IV	Poison Requirements	3.32
3.B.V	Poison Rod Dimensions	3.33
3.B.VI	Heat Generation Rates in Poison Rods	3.33
3.B.VII	Maximum Poison Temperatures	3.34
3.C.I	Calculations and Reactivity Worths	3.38
3.C.II	NPTR Radiation Levels	3.44
3.C.III	Gamma Intensity Near Closed Loop Pipes in R/hr	3.48
3.D.I	Closed Loop Components Failure Rates and Mean-Time-to-Restore	3.51
4.A.I	GETR/TREAT Capsule Status	4.6
4.A.II	GETR Capsule Burnup Levels	4.6
4.B.I	Sodium Studies Facilities Status	4.20
4.C.I	Thermal Convection Loop Operating Parameters	4.23
4.C.II	Neutron Exposure in $nvt \times 10^{21}$ (E > 0.18 MeV) for Thermocouples in Each Designated Flux Region	4.35
4.D.I	Equivalent Irradiation Damage Fluence	4.53

FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
JULY, AUGUST, SEPTEMBER 1968

INTRODUCTION

This quarterly report is arranged in four chapters. Chapter I presents a brief description of FFTF overall program progress. Chapter II (Systems) describes systems management progress made by the FFTF Project Engineering Department during the reporting period. Chapter III (General Technology) reports progress made by the FFTF Reactor and Plant Technology Department; this chapter also includes results of work by all departments on reactor concepts. Chapter IV presents work by FFTF Development Department. Contributions from a number of off-site government organizations and industrial firms who are supporting PNL in the FFTF Program are also summarized in the various chapters. Prior progress is cited where appropriate for reader orientation.

I. FFTF OVERALL PROGRAM PROGRESS

This chapter presents a summation of FFTF Program Progress during July, August and September, 1968.

A. SELECTION OF REACTOR CONCEPT

Extensive studies directed toward selection of the reactor concept were performed with the assistance of Westinghouse Corporation. A decision was reached to concentrate all work on concepts which include:

- Vertical orientation of the fuel subassemblies
- Direct fuel withdrawal for refueling
- Use of a gas-cooled, fully shielded fuel handling machine
- Provision for contact maintenance in the area above the reactor head without reliance on space suits
- Proximity instrumentation for each driver fuel position for flow, temperature and failed fuel detection and location.

B. PHYSICS

A 1-dimensional diffusion theory program to compute resonance-shielded and collapsed-group cross sections, IDX, and a computer code to calculate group constants from currently available cross section data, ETOX, have been written for use in fast-reactor analyses.

Major sections of the multigroup 3-dimension synthesis diffusion analysis code, RSYN, have been completed and the computer strategy for solving the synthesis equations has been selected.

A completed study indicates that synthesis techniques are not appropriate for calculations of radiation dose rate at deep penetrations where the characteristic of interest, the radiation levels, are not averaged over the entire system.

C. METALLURGY

Neutron damage production in the FTR is being investigated both through an empirical damage function unfolding technique and by the calculation of gross defect production and vacancy clusters. The calculated values, derived empirically, provide an initial estimate for the damage function. Calculated values of gross defect production and vacancy clusters are presented for the FTR fuel cladding, grid plate and pressure vessel, and these values are presented relative to EBR-II, ETR and DFR. It is estimated that a fluence value 80 to 150 times larger is required at the FTR pressure vessel to accumulate a damage state equal to that at Row 2 in EBR-II.

An analysis of creep-rupture data for Type 304 SS showed that rupture lives during in-reactor testing and postirradiation testing are essentially the same for comparable conditions of stress and temperature. The effect of neutron irradiation, as reflected in both the in-reactor and postirradiation tests, is to reduce the rupture strength as compared to the unirradiated control data.

D. COOLANT DEVELOPMENT

The reaction mechanism of methane impurity in inert gas with sodium was determined to be the same in both argon and helium. Rate constants were established in both systems.

Tests were run to evaluate the inhibiting effect of oxygen on the nitriding of stainless steel to support the planned use of nitrogen cover gas in the cells. Nitriding was successfully inhibited at 1000 °F by 500 ppm oxygen concentration, and at 1400 °F by 5000 ppm concentration.

E. FFTF FUELS

1. Development

Product specifications and all other necessary documentation were submitted to RDT in preparation for issuing the bid package for the LMFBR Fuel Supplier Prequalification Program to potential bidders.

Preparation and installation of the FFTF Fabrication Demonstration Facility (the pin portion) is proceeding ahead of schedule.

An order of Type 316 SS tubing for developmental fuel pins was examined using ultrasonic and eddy current techniques. The portion of the tubing given special handling during fabrication had a significantly improved yield over the remainder that was handled in the normal manner.

Neutron radiography of two full-length fuel pins that completed the goal exposure of 10,000 MWd/tonne in the GETR showed the fuel pins to be in satisfactory condition except for some gaps observed between fuel pellets.

Three capsules were removed from MTR/ETR series, two after successfully reaching goal burnups of 50,000 MWd/tonne and one after experiencing a cladding failure at a burnup of 41,000 MWd/tonne.

Static and dynamic pressure creep tests of typical fuel duct sections were conducted. After 4300 hours at anticipated FTR operating pressure and temperature levels, the average flat-to-flat dimension of the static pressure creep specimen increased about 0.18%, with only slight changes in the average corner-to-corner dimension. The dynamic pressure creep specimen showed no significant change in either dimension after 3600 hours exposure.

2. Fuel Temperature Analyses

Reduction of severe coolant temperature gradients near the periphery of the fuel subassembly by means of flow suppressors was studied. Reduction of the maximum ΔT across the duct by 25 to 50% appears possible using any of three types of suppressors.

The temperature field in cladding and coolant associated with the hottest fuel pin was investigated using the MANTA code. The degree of mixing was varied and cases with and without wire wrap examined. Results indicate that mixing is important but not critical in determining the temperature distribution in the wire-cladding region.

F. REACTOR CORE STUDIES

Control Rod Poison Materials

Analyses of Ta and B_4C control poison materials were completed. It was concluded that Ta is preferred for shim rods residing in the core and that B_4C is better suited to in-core safety rods and peripheral shim rods.

Core Meltdown

The effects of core meltdown into a crucible below the core were studied assuming complete loss of coolant flow and with a sodium inventory restricted to that contained only in the inlet plenum. For full core meltdown, the study indicated that a 6-in. thick stainless steel crucible would be melted within 10 minutes after core meltdown. Assuming that meltdown of only seven subassemblies occurs, a quasi steady state heat transfer mode is achieved after about 15 minutes; sodium boiling occurs after eight hours with maximum crucible temperatures of about 2100 °F. Thus, it is concluded that melting of at least seven subassemblies can be accommodated in a stainless steel crucible below the inlet plenum gridplate.

Inlet Model

The inlet plenum feature model, a 1/3 scale hydraulic model of the FFTF piped inlet arrangement, was completed, and operation was started. The model will provide information which will aid in the evaluation and refinement of the inlet configurations of the reactor vessel system.

Sensor Development

Tests were run on candidate neutron detectors for FFTF to evaluate the high level gamma effects on the neutron sensitivity of both ^{10}B lined proportional counters and fission chambers. Under specific system configurations, both types provided useable short-term neutron sensitivity under high gamma background conditions.

W-5% Re/W-25% Re thermocouples had a -15% change in calibration at 1400 °F when subjected to neutron fluence levels of up to 3.4×10^{21} nvt. It was decided to give no further consideration of the use of these thermocouples in the FFTF.

G. MATHEMATICAL MODEL OF HEATING AND VENTILATING SYSTEM

The mathematical model for the heating and ventilation (H&V) system has been completed. This model was used for H&V system transient studies for loss of normal electrical power. Transient temperature responses were determined for six different H&V shutdown conditions plus an additional investigation of loss of normal electrical power during containment operation. These results will assist in determining the requirements of the H&V for an electrical power outage.

II. SYSTEMS

W. B. McDonald

A. REACTOR PLANT DESIGN1. Reactor Design

R. C. Walker

a. Reactor Vessel

R. C. Walker and O. W. Priebe

Babcock and Wilcox Co. has continued to assist PNL in evaluating alternative vessel concepts, including mechanical and thermal analysis of concept features, applicability of code requirements and manufacturability considerations. Although B&W has tentatively recommended a piped bottom inlet concept in preference to an annular or piped downcomer inlet concept, Westinghouse has recently suggested an alternative annular inlet concept which is currently under evaluation by both B&W and PNL. The annular inlet concept proposed by Westinghouse is aimed at reducing the primary vessel wall temperature well below the design bulk outlet temperature of 1200 °F and requires 5 to 6% bypass flow. Also, as a result of studies performed to date, it appears that the vessel head must be cooled to avoid temperature transient problems in that region.

b. Reactor Core

R. C. Walker and J. F. Wett

Westinghouse has completed a design study recommending a compliant grid spacer for further investigation and development as a backup to the reference wire-wrap spacing system. Three different types of spacers were examined: a tri-planar spacer, a rhombus cluster design, and a modification of a conventional PWR type spacer referred to as a honeycomb spacer.

As a result of the study, W has concluded, and PNL concurs, that, of the three generic types of spacers studied, the honeycomb spacer is the best alternative system for use in the FTR.

Of the many conceivable designs of a honeycomb type spacer, the one recommended by Westinghouse consists of a series of individually formed straps brazed together to form a honeycomb. Individual straps are 0.015-in. thick by 1.250-in. deep. Fuel pins are held in place by dimples and cantilever springs formed into the straps prior to brazing. The recommended strap material is Inconel 718 and the recommended braze material is Nicrobraz 50.

c. Backup Core Design

F. J. Kempf

General Electric Company investigated design of a regular vertical core as a backup to the reference core concepts being investigated by PNL and Westinghouse. For the backup core a reference control drive concept utilizing a 10-in. offset beam between the control drive extension and the control rods was established. A dash pot below the control rods decelerates the rods to a relatively slow speed following scram action. The energy of the rods at the end of the control rod stroke is transferred to springs located approximately at the center-of-gravity of the drive.

A core clamping system for the backup design was established using a torsion rod type of clamping mechanism similar to the type installed in the SEFOR reactor. In the clamping system, a clamping device would be installed within each one of a set of reflector assemblies which form a complete ring around the fuel assemblies. The ring of clamping mechanisms would be located radially beyond the outside diameter of the reflector plug. The clamping mechanisms are actuated

by a special tool inserted through access holes in the reactor outer head. This method of clamping the core has major advantages of not interfering with refueling or experiment handling since the clamping mechanisms are located outside the core. Also, the core can be tightened and verified before startup since no core temperature changes or hydraulic pressure is required for clamp actuation.

Significant design features of the vessel head included:

- (1) Three concentric piece construction - the head, the annular reflector plug, and the refueling plug.
- (2) DBA integrity requirements dominated the structural design and required 48 large holddown stretch bolts.
- (3) Cold structure design - the steel structure of each of the three head components will operate below 200 °F. The temperature gradient between the 1200 °F design temperature of the sodium and the structure is provided by a 30- to 40-in.-thick insulation region below the cooling coils on the lower edge of the structure.
- (4) The upper section of the reactor vessel and the flange are maintained cool by the head and external vessel cooling systems and thermal insulation of the vessel.
- (5) All reactor and experimental access, service and information systems penetrate the head.
- (6) Biological shielding (steel and lead) permits personnel access in the vicinity of the head when all plugs are in place. Additional lead shielding covers the closed loop branch piping and fuel leak detector system piping after it penetrates the head.

2. Heat Transport and Irradiation Testing

J. H. Westsik

a. Irradiation Test Facilities

M. A. Vogel

(1) Closed Loop System Analysis (M. K. Mahaffey).

Studies were completed on the decay heat removal capability of the in-reactor tube during test specimen removal. Results indicate that the present tube concept can dissipate approximately 54 kW to the reactor pool with pool sodium at 500 °F. A typical 6 MW test will decay to about 37 kW 24 hours after shutdown so that loop coolant flow can be turned off after this period. If the main pump flow fails, the reactor must be immediately shut down and emergency cooling supplied until the decay heat generation equals the tube heat loss at reasonable temperatures. Otherwise, coolant voiding and test fuel cladding melt will occur.

Computer models of the air blast heat exchanger (ABHX) and intermediate heat exchanger (IHX) have been used to determine related thermal capacitance between the primary and secondary loops and also to define the relative sizings of the IHX and ABHX. The results indicated that the compact IHX required for the system should be of counterflow design to achieve reasonable ABHX sizing. In order to minimize thermal stress problems secondary flow should equal primary flow.

A parametric computer study was initiated to determine the permissible range of testing considering loop power, temperature rise across the fuel, fuel outlet temperature, loop flow, and ABHX air flow. Initial results indicate a possible sodium freezing problem in the secondary coolant system for loop tests operating at less than 1.5 MW. This

is caused by a transfer of heat between the inlet and outlet flow channels in the tube, which requires the sodium entering the tube to be well below the selected test inlet temperature. Gas insulated gaps in the tube walls above the fueled region are being investigated as a solution to the problem.

Calculations have been made for test powers and flow rates associated with the vertical backup core using a lattice spacing of 4.45 in. and a test diameter of 2.6 inches. Since this is a reduction in test diameter compared to the split conical core, reduced flow rates and test power levels are necessary. Selected results indicate a maximum test power of about 4 MW in comparison to 6 MW for the split conical design.

(2) Short-Term Irradiation Facility (STIF)

(R. R. Derusseau). The final report on the STIF feasibility scoping design study of both rabbit and trail cable facilities has been completed by General Electric (Cincinnati). The final design for a rapid transfer rabbit system utilizes a liquid metal coolant (NaK) in both primary and secondary loops. As designed, this system is able to dissipate 0.5 MW of heat, operate between 150 °F and 1200 °F, and transfer the test specimen from the core reflector position to an inert hot cell for examination in approximately one minute. The specimens are basically uninstrumented. The transfer procedure involves propelling and cooling the rabbit specimen by both NaK and inert gas. The transfer between the two media is a complicated process and requires development to ensure workability.

The trail cable facility can be used for testing instrumented fuel bundles up to 1 MW power output using reactor core sodium as the coolant. Transfer times are longer than

for the rabbit, but larger test specimens can be handled. There may be transfer complications in taking the test specimen from the radioactive reactor cover gas to the non-radioactive transfer cell atmosphere while removing the decay heat load. This will be investigated further during conceptual design.

Relative costs outlined in the General Electric study are summarized as follows:

	<u>Rabbit Facility</u>	<u>Trail Cable Facility</u>
Development	\$ 650,000	\$290,000
Design	600,120	294,300
Equipment	1,843,490	225,650
Contingency	<u>309,000</u>	<u>81,000</u>
Total	\$3,402,610	\$890,950

3. Reactor Plant Instrumentation

E. M. Johnston and C. D. Swanson

a. Reactor Instrumentation and Control

R. R. Cone

(1) Reactor and Vessel Instrumentation Systems (W. Dalos).

This system provides the in-vessel sensors and associated electronics required for the surveillance and protection of the reactor. Because of the severe temperature and radiation environments, each type of sensor requires development or testing effort to ensure successful system performance. The status of these sensors is reviewed briefly.

(a) Coolant Temperature Measurement. Reactor and vessel temperatures will be measured with sheathed, magnesium oxide insulated chromel-alumel thermocouples. Design goals for reactor and vessel operation are for 1% accuracy, 2 sec response time and 20-yr mean-time-between-failures (MTBF).

Accuracy and response are expected to be within these limits at the end of the design exposure (one year at full reactor power). A thermocouple testing program is underway at EBR-II to establish the effects of high neutron fluence and thermal cycling on accuracy, response time and reliability. Thirty 1/8-in. sheathed thermocouples are currently being irradiated to 10^{23} nvt.

A need for the previously accepted response time of one second maximum has not been supported by analytical studies. Response times in the order of a few seconds are now considered adequate. This relaxation permits increased reliability since high reliability and fast response are somewhat incompatible. For example, higher reliability is obtained with 1/8-in. OD ungrounded thermocouples while faster response is obtained with 1/16-in. OD grounded thermocouples. The development effort will determine optimum choices among these conflicting alternatives. Most thermocouples for the thermal cycling tests will be Incoloy-sheathed but some thermocouples with other sheath materials will also be tested for comparative purposes.

(b) Coolant Pressure Measurement. Coolant pressure will be measured by a NaK-filled bellows with a capillary transmission line to an electromechanical transducer located outside of the coolant in a temperature of less than 500 °F. Initial test results on such an assembly equipped with a strain gage transducer showed acceptable calibration shifts with temperature variations. Other electromechanical transducers will also be evaluated.

(c) In-Core Coolant Flow Measurement. The in-core flow sensor for driver and open loop subassemblies is expected to be the full flow permanent magnet type being developed by ANL.

If there is insufficient space in the channel, a probe type eddy-current sensor as small as 3/4-in. OD could be used but accuracy is lower. Recent test results with an experimental eddy-current sensor operating in Wood's metal indicated an output of about 18 microvolts (peak-to-peak) per ft/sec per volt excitation. Assuming a coolant outlet velocity of 10 ft/sec and an excitation of 50 V, output is estimated at 4 millivolts (dc).

(d) Coolant Level. The operating level sensors will be induction probes similar to those used at Hallam. The drain-and-fill probe is a long assembly with about 18 thermocouples mounted at varying heights. Thermocouples immersed in the sodium indicate a higher temperature than those exposed to cover gas.

Scanning and comparing these outputs locates the liquid-gas interface. Experimental results reported in GEAP-5560 indicate that the temperature gradient in the cover gas immediately above the sodium pool will be large, possibly as high as 200 °F in the first inch.

(2) Fuel Failure Monitoring System (R. R. Cone). A review of the several possible methods for locating failed fuel pins yielded about ten potentially feasible methods. All methods involve sampling of channels to detect the fission or xenon tagging gas or the delayed neutron precursors emitted from a failed pin. Xenon gas tagging is a technique being tried at EBR-II and does not involve emitted fission products.

The basic philosophy currently being used to establish fuel failure detection and location requirements consists of the following:

- In the early stages of FTR operation, the reactor will not be operated intentionally with failed fuel.

- The fuel failure must be detected rapidly but the detection time has not been established.
- Fuel will be considered failed if the cladding integrity is violated at any location including the gas plenum region.
- Failed fuel location should be accomplished within the refueling time period for the reactor.

From studies made to date, the following tentative conclusions have been reached:

- Location of a failed subassembly will probably depend primarily on the detection of emitted fission gas (xenon and krypton). Delayed neutron (d.n.) detection is not favored because of: (a) unknown effects on detection sensitivity and reliability, (b) short half-life of d.n. precursors requiring rapid measurement, (c) dependency on power level (e.g., cannot be used when reactor is shut down), and (d) uncertainty as to their mode and rate of release of delayed neutrons.
- Some form of vortex generator will be required to separate the fission gas from the coolant, or supply a sodium stream enriched in fission gas. This separator must be located in the fuel subassembly duct or other location which can ensure that all of the subassembly coolant will pass through it.
- On a vertical core reactor the restricted space prohibits installation of the sample line traps, gamma shielding, detectors and other components required for each subassembly for "in-vessel" detection. Therefore, sodium samples from each subassembly must be delivered to an out-of-vessel location for detection purposes. Because the space for sodium vapor traps is inadequate and the uncertainties about gas line plugging, the sampling method must employ sodium enriched with fission gas.

(3) Flux Monitoring and Control (L. W. McClellan). A flux-flow comparator circuit has been incorporated into the Flux Monitoring and Control System to satisfy a design safety criterion for protection against thermal shock of the reactor and its heat transport loops. The criterion requires protection against low flow in the primary heat removal loops (including closed loops) during steady-state operation and during startup and shutdown when the normal low flow trip might be bypassed.

The flux-flow comparator circuit satisfies this requirement by providing a scram trip if flow decreases below the desired value.

Figures 2.A.1 and 2.A.2, Flux-Flow Trip Circuits, show the comparator circuit concept which uses a 2-out-of-4 flux-flow logic having four independent flux-flow primary trips.

b. Plant Protection System

G. H. Strong and J. P. Thomas

The Plant Protection System (PPS) includes all sensors, instruments and power supplies, logic elements and other actuating equipment required to prevent reactor accidents and to limit the consequences of an accident if one should occur. The PPS provides protection to the plant, personnel and public by initiating reactor scram, FFTF containment or emergency core cooling as required.

This system integrates and coordinates all of the instruments, devices, and components required to perform the overall protective function. These include instrument sensors, signal conditioning units, amplifiers, trip controllers, logic circuits, trip circuits and actuator devices. Overall timing coordination of all protective components is a function of the PPS. For example, the nucleonic part of the PPS is

FIGURE 2.A.1. Flux/Flow Trip Circuit for Reactor Primary Sodium Loops

B = BUFFERED SIGNAL

FIGURE 2.A.2. *Closed Loop Primary Sodium Flux/Flow Trip Circuit*

designed to integrate within an acceptable time period the entire sequence of events leading to reactor scram, beginning with an unsafe condition and followed by detection, amplification, logic-circuit action, actuator operation, rod drop, and reactor shutdown. Functions performed by the PPS are as follows:

- Initiate reactor scram whenever protective instrumentation detects an unsafe condition requiring emergency shutdown of the reactor to prevent fuel damage.
- Initiate the containment of the FFTF when the containment protective instrumentation detects an unsafe condition requiring closure of the containment vessel to prevent release of radioactivity to the atmosphere.
- Initiate emergency core cooling when the protective instrumentation detects an unsafe condition requiring emergency core cooling system action.
- Coordinate and integrate instrumentation, electrical, mechanical and control systems to ensure that the required overall protective functions are accomplished.

The main effort to date has been concentrated on the reactor safety circuit. It is planned that 2-out-of-4 (2/4) trip logic will be used for most trip inputs to reduce the number of spurious trips. The only input trips thus far identified which have less than 2/4 logic, are the low and intermediate neutron flux level and period trips which have 2/3 logic. Since the reactor will operate in the range of these instruments for only a very small portion of the total operating time, it is not expected that plant availability will suffer inordinately.

B. GENERAL PLANT DESIGN

E. M. Johnston

1. Structures, Utilities and Containment

F. H. Shadel

a. Heating and Ventilation System

B. O. Shaver

The conceptual design description for the Heating and Ventilation System was completed and sent to RDT for approval. This section summarizes the information presented. The Heating and Ventilation System provides the atmospheric conditions to ensure that plant objectives can be met in a safe and practical manner. The control of the atmospheric conditions includes the following major considerations:

- Control of the quantity and quality of the ventilation air;
- Control of airborne contamination by maintaining proper building pressure levels;
- Control the release of airborne contaminants to the environs;
- Clean up airborne contamination in the containment vessel under abnormal conditions;
- Control the containment vessel atmosphere within temperature limits under accident conditions.

A schematic arrangement of the system is shown in Figure 2.B.1.

2. General Plant Instrument and Electrical System

C. D. Swanson

a. Plant Instrumentation and Control

J. W. Mitchell

(1) Central Control and Data Handling Systems Design (G. L. Waldkoetter). Design study contracts initiated last quarter to obtain independent appraisals of the conceptual

FIGURE 2.B.1. FFTF Ventilation Flow Diagram

design of the Central Control and Data Handling System (CC&DH) were completed this quarter. The final reports from Autonetics Division of North American Rockwell Corporation and LTV Electronics, Inc. have been reviewed. The results of a third study, independently undertaken by Aerojet-General Corporation, have been discussed with the study team. The written report has not yet been received.

In general, all three contractors approved the PNL concept (described in BNWL-880). All contractors agreed that:

- Multiple computer configurations be used for real time data processing.
- Design and fabrication of hardware is within the state-of-the-art. No component development is required.
- Computer directed cathode ray tubes (CRT's) should be used for data display.
- Color CRT displays, though desirable from an information standpoint, are not recommended at this time because of cost, reliability, and noncolor blindness requirements imposed on the operator.
- Computer control of the sodium piping preheat system was particularly recommended as an efficient means for obtaining uniform heat distribution and reducing thermal stress on pipes and vessels.

b. Electrical

G. H. Strong

(1) Primary Electrical Power System (M. F. Wiitala). The Bonneville Power Administration (BPA) has proposed a concept for providing power to the FFTF. The concept was tentatively approved by PNL with the reservation that a study would have to be made to confirm that the concept meets the functions and requirements set forth in the Conceptual System Design Description (CSDD) of the Primary Electrical Power System.

The new concept redirects the existing 115 kV line between the Midway and Benton stations through the FFTF substation. Two series sectionalizing circuit breakers at the FFTF substation create two separate 115 kV lines to FFTF. A PNL study of the effects on FFTF of faults on either of the two new proposed sections of 115 kV line has been made.

Based upon the study, it is expected that it will not be necessary to scram the reactor upon any type of electrical disturbances except sustained faults on both feeder transmission lines (i.e. total loss of power). A disturbance of this type is not expected more than once in a lifetime of the reactor.

3. Other General Plant Design

a. Auxiliary Systems

R. V. Dulin

(1) Inert Gas Receiving and Processing System (C. J. Foley). The system CSDD was resubmitted for RDT approval. The revised CSDD reflected changes in gas selection caused by additional design studies and clarification of FFTF requirements with RDT and included the necessary concept

revisions to accommodate the changes in gas selections. The revised gas selections are summarized in Table 2.B.I. The principal changes occurred as a result of a decision not to include gas system provisions to accommodate vented fuel in the driver core in the conceptual and preliminary designs.

b. Fuel Examination Facilities

C. L. Boyd

(1) Fuel Examination Equipment (E. B. Ramsey).

(a) Neutron Radiography Equipment. General Electric (APO) is to provide facility requirements and a conceptual design of neutron radiography equipment for the FFTF Fuel Examination Facility which will also meet the requirements of ANL's Hot Fuel Examination Facility (HFEF).

A preliminary Phase I report (which includes literature and field survey reviews, recommendations, and proposed development areas) is being reviewed by PNL and ANL (HFEF) with the final Phase I report to be issued shortly.

(b) Disassembly-Reassembly Equipment. Union Carbide Corporation at Oak Ridge National Laboratory (ORNL) is to provide facility requirements and a conceptual design of disassembly-reassembly equipment for the FFTF Fuel Examination Facility.

To date, the work plan has been approved and a technology survey completed. A preliminary report is presently being prepared for PNL review and approval.

TABLE 2.B.I. *FFTF Inert Gas Selections*

<u>Application</u>	<u>Inert Gas Selection</u>
Reactor Primary Heat Transport System	Argon
Reactor Primary Sodium Receiving and Processing System	Argon
Closed Loop Primary Heat Transport System	Argon
Closed Loop Primary Sodium Receiving and Processing System	Argon
Reactor Secondary Heat Transport System	Argon
Reactor Secondary Sodium Receiving and Processing System	Argon
Closed Loop Secondary Heat Transport System	Argon
Closed Loop Secondary Sodium Receiving and Processing System	Argon
Fresh Sodium Storage	Argon
Contaminated Sodium Storage	Argon
Interim Fuel Decay Storage	Argon
Closed Loop Cell Atmosphere	Nitrogen
Reactor Primary Sodium Purification and Storage Cell Atmosphere	Nitrogen
Reactor Primary Heat Transport Equipment Cell Atmosphere	Nitrogen
Contaminated Sodium Processing Cell Atmosphere	Nitrogen
Radioactive Maintenance Cell	Nitrogen
Fuel Transfer and Examination Cell	Argon
Fuel Handling Machine (tentative)	Argon
Optional Atmosphere in Refueling Hot Cell	Nitrogen

III. GENERAL TECHNOLOGY

P. L. Hoffmann

A. DESIGN EVALUATION

L. M. FINCH

1. Reactor Evaluation

D. Marinos

a. Concept Identification

D. Marinos, G. R. Waymire, A. J. Anthony,
W. U. Jackson, W. H. Davenport and J. F. Lubeck

During the previous quarter, conceptual design task forces completed intensive studies to, (1) confirm the neutron flux deficiency of the (reference) split conical core for the FTR and identify a higher performance alternative core of conical geometry, and, (2) review and improve the (backup) vertical core design, particularly with respect to test loop size and driver fuel instrumentation. The results of these two task force efforts are summarized in the previous Quarterly Report⁽¹⁾ and are documented in informal reports.

Follow-on efforts to the task force work resulted in further review of identified viable concepts and the identification of two additional concepts for evaluation.

In addition to the conical dispersed core selected by the reference core task force effort a skewed version with features noted in Table 3.A.I was studied and designated as Concept I. The dispersed core design resulting from the earlier task force work was designated as Concept II.

The vertical (backup) task force recommended concept was designated as Concept III. An additional vertical design, designated as Concept IV, is also summarized in Table 3.A.I.

1. Fast Flux Test Facility Quarterly Progress Report April, May, June 1968, BNWL-880, Pacific Northwest Laboratory, Richland, Washington, October 1968.

TABLE 3.A.I. Comparison of Concept Features

Item	Revised Concept I-A	Concept I	Concept II	Concept III	Concept IV
Core Geometry	Vertical	Skew	Annular Cone	Vertical	Vertical
Fuel Handling Machine	Shielded- gas cooled	Wet Shielded	Snorkel un shielded	Under-sodium + snorkel for Closed & Open Tests	Under Head Pantograph
Service Area Atmosphere	Air	Air	Inert Hot Cell	Inert Hot Cell	Inert Hot Cell Air in Service Cell
Driver Instru- mentation	Individual package each assembly	Individual package each assembly	Individual package each assembly	Seven cluster lance	Seven finger lance ventilated for refueling.
Cover Penetra- tions	9-12 in. ø to ø Opening	12.5 in. ø to ø spacing all positions	9 in. ø to ø spacing all positions	19 lance positions for loops, controls & open tests + shield plug	19 lance positions + test positions
Access	Direct	Direct	Remote	Remote	Remote to loops Direct to top head mechanisms

The follow-on efforts not only aided in defining the optimum conceptual design features for the FTR, but also afforded an opportunity for the selected reactor plant designer, Westinghouse Electric Corporation, (Advanced Reactors Division), to become familiar with the basic criteria and designs being considered for the FTR. Westinghouse also provided valuable assistance by aiding in detailed evaluation of the four concepts.

Figures 3.A.1, 3.A.2, 3.A.3, and 3.A.4 show the basic features associated with Concepts I through IV, respectively. These concepts were presented to RDT in a combined PNL/RDT/ RPD meeting the last week of September. A generalized concept designated Revised Concept I-A evolved from this meeting. This concept utilizes a vertical core and requires that fuel will be removed by direct withdrawal. PNL and Westinghouse are presently in a study to optimize this concept. Figure 3.A.5 indicates the basic features of Revised

PERFORMANCE

- CORE CONFIGURATION - ANNULAR SKEW OR ANNULAR CORE
- LATTICE POSITIONS - 16% (SKEW)
- DRIVER SUBASSY - 68
- SAFETY RODS - 3
- CONTROL RODS - 3 + 24
- PINS PER SUBASSY - 217
- PIN DIAMETER - .235"
- PIN PITCH - .575"
- PLENUM LENGTH - 5'0"
- FUEL SUBASSY HEIGHT - 32'-6"
- CORE HEIGHT - 36' (CORE) 35' (SKEW)
- EQUIV. CORE DIA - 45' (CORE)
- CORE VOLUME - 928 L (SKEW)
- FLOW RATE - 30,000 GPM
- INLET TEMPERATURE - 550°F (FIRST CORE)
- OUTLET TEMPERATURE - 900°F (FIRST CORE)
- HOT CELL ATMOSPHERE - AIR

TESTING CAPABILITIES

- CLOSED LOOPS - 5
- OPEN TESTS - 8
- LOUVER DIAMETER - 2'-6"
- CLOSED LOOP FLUX - 0.70 MW (SKEW)
- CLOSED LOOP FLUX - 0.48-1.0 MW (CORE)

CONCEPT VARIATIONS

- CLOSED LOOP PIPING - OVERHEAD
- CLOSED LOOP PIPING - ON NOZZLE
- GAMMA SHIELD - PERFORATED
- GAMMA SHIELD - ROTATING PLUS

CONCEPTUAL

1/7/78 SK-14371-0010
1/7/78 SK-14371-0010

SK-3-14371

FIGURE 3.A.1. Feature Identification for Concept I

PERFORMANCE

CORE CONFIGURATION	ANNULAR CONE
LATTICE POSITIONS	168
DRIVER SUB-ASSEMBLIES	71
SAFETY RODS	3
CONTROL RODS	3 + 24
FUEL PINS/SUBASSEMBLY	217
PIN DIAMETER	.235
PIN PITCH	.275
PLENUM LENGTH	5 FT
FUEL SUBASSY HEIGHT	~19 FT
CORE HEIGHT	36 IN
EQUIVALENT CORE DIA	48.2 IN
CORE VOLUME	1064 LITERS
FLOW RATE	30,000 GPM
INLET TEMP (1ST CORE)	550 °F
OUTLET TEMP (1ST CORE)	900 °F
HOT CELL ATMOSPHERE	INERT

TESTING CAPABILITIES

CLOSED LOOPS	5
OPEN TESTS	8
USABLE DIA	2.6 IN.
CLOSED LOOP FLUX	0.7×10^{16} n/cm ² -sec

CONCEPT VARIATIONS

CLOSED LOOP PIPING	DUAL SET OVER NOZZLES OR EXTENDING ~8FT ABOVE COVER
GAMMA SHIELD	NONE REQUIRED
FUEL HANDLING TECHNIQUE	SNORKEL MACHINE FOR CLOSED LOOPS. UNDER-SODIUM FINNED POT FOR DRIVER FUEL.

CONCEPTUAL

1ST ISSUE 8/9/68

SK-3-14372

FIGURE 3.A.2. Feature Identification for Concept II

PERFORMANCE

LATTICE POSITIONS	127
DRIVER ASSEMBLIES	105
SAFETY RODS	3
CONTROL RODS	36
FUEL PIN/SUBASSEMBLY	127
PIN DIAMETER	.210
PIN PITCH	.270
PLENUM LENGTH	3 FT
FUEL SUBASSEMBLY HEIGHT	203.75 IN
CORE HEIGHT	36 IN
EQUIVALENT CORE DIAMETER	40 IN
CORE VOLUME	743.9 L
FLOW RATE	151×10^6 G/HR
INLET TEMPERATURE (FIRST CORE)	700° F
OUTLET TEMPERATURE (FIRST CORE)	1000° F
HOT CELL ATMOSPHERE	ARGON

TEST CAPABILITIES

CLOSED LOOPS	6 - 1/2" TUBE IN CENTER
OPEN LOOPS	12
USABLE DIAMETER	2.75 & 2.25 ON RE-ENTRANT
CLOSED LOOP FLUX	0.95×10^{16} WITHOUT FUEL SHUFFLE
	1.01×10^{16} WITH FUEL SHUFFLE

CONCEPT VARIATIONS

SUBASSEMBLY SIZE AND NUMBER

FUEL HANDLING TECHNIQUE

SNORKEL MACHINE FOR CLOSED LOOPS.
FINNED POT UNDER SODIUM FOR FUEL.

★ 63 ENRICHMENT ZONE #1
42 ENRICHMENT ZONE #2

CONCEPTUAL

SK-3-14373
7-2-60 ISSUE 2-13-64
1-1-61 ISSUE 2-13-64
1-1-61 ISSUE 2-13-64

SK-3-14373

FIGURE 3.A.3. Feature Identification for Concept III

CONCEPTUAL

147 ISSUE 8-8-68

SK-3-14374

FIGURE 3.A.4. Feature Identification for Concept IV

SK-3-14397

FIGURE 3.A.5. Feature Identification for Revised Concept I-A

Concept I-A. Revised Concept I-A is presently being optimized in two variations: (1) a dispersed head with individual tube access and (2) a multiple access nozzle concept.

b. Concept III Instrumentation

G. R. Waymire

In Concept III, instrumentation is provided to the fuel assemblies via instrumentation lances which are concentrically located at the shield plug penetrations around an open test position or control rod drive. The configuration of the lance is basically the same for each position except for the connections of the inner instrumentation. A lance assembly consists of a 6.0-in. OD, 0.150-in. thick wall pipe, approximately 20 ft long, which serves as a structural support for thermocouples, an eddy current flowmeter, and a FEDAL (failed element detection and location) sensing tube. Three tubes are routed down the ID of the lance and into the outlet region of each fuel assembly in a cluster of six. Two tubes contain the thermocouple and flowmeter leads. The third FEDAL tube is utilized for fission gas monitoring.

In addition to providing the conduit for routing instrumentation leads, the lance also serves as a backup holddown feature for the six subassemblies it serves. Figure 3.A.6 shows a typical lance design. Figure 3.A.7 shows a typical safety rod drive as related to a lance.

c. Subassembly Seat Configuration

G. R. Waymire

A feature of the FTR which appears to be relatively concept independent is the lower portion of the driver duct and tube-sheet interface. This region also appears to offer the best position for a fixed axial datum for the driver fuel and open test assemblies. A fixed datum position is required for all

3.9

BNWL-917

SK-3-14064

FIGURE 3.A.6. Instrument Lance/Open Position Installation, Concept III

3.10

BNWL-917

SK-3-14067

FIGURE 3.A.7. Safety Rod Drive Mechanism, Concept III

removable core components in order to predict the relative axial position of the fuel and hence the resultant reactivity change caused by thermal expansion. This datum was chosen because:

- Thermal expansion in an upward direction results in a negative reactivity feedback with a dispersed core.
- A seating action at this tubesheet offers the best method for concentrically locating the assemblies in the plenum extension tube.
- Thermal expansion between the tubesheet and the core is small and predictable since it is controlled by the coolant inlet temperature.

An initial configuration for the seat design is shown in Figure 3.A.8. The design intent is to seat the subassembly on the chamfered edge section with sufficient force to hold the assembly down but with enough bearing surface to minimize loading. By-pass flow is then allowed to pass through the radial slots. A seal is provided by the proper combination of clearance and length below the seat and sizing of the by-pass flow slots. This sealing combination is required to prevent high velocities through the slots thus minimizing cavitation and erosion.

This design with variations in material combinations and dimensions will be tested at LMED. Several design iterations based on test data and additional results analysis will be required.

d. Closed Loop Test Specimen

W. C. Chenault, W. H. Davenport

Conceptual design of a test specimen train was completed for a specimen that can be disassembled for interim examination and reassembled remotely. A 61-pin test bundle is shown on Figure 3.A.9 with 32 of the pins instrumented with thermocouples. The center fuel pin is omitted and used for inlet flow instrumentation. Figure 3.A.10 indicates a method of contact instrumentation for individual fuel pins including fuel pin centerline temperature and fuel cladding temperatures.

FIGURE 3.A.8. Tubesheet Seat Design

SK-3-14065

FIGURE 3.A.9. Proposed 61-Pin Test Assembly

CONCEPTUAL

0.015 DIA HOLES

SK-3-14066

FIGURE 3.A.10. Proposed Instrumentation and Fuel Pins

2. Plant Evaluation

L. M. Finch

a. Thermal Analysis - High Temperature Vessel Cover Design Alternate

D. P. Schivley

From a cursory analysis, it was concluded that a high temperature (>1000 °F) design for the reactor vessel cover is not attractive because (1) it offers no inherent compensation for the thermal transients associated with reactor scrams, (2) it may impose relatively high thermal stresses in temperature weakened structural material, and (3) it imposes an intolerable delay after reactor shutdown until the cover has cooled sufficiently to permit refueling or other servicing operations.

In the calculations it was assumed that the sodium pool temperature would undergo a step change from 1200 to 800 °F at time zero. The initial cover temperature was assumed to be 1200 °F throughout. The model assumed that the cover was perfectly insulated at its outer surface. Thus, cooling was only by heat transfer to the sodium pool. The computed cover temperatures at the inner and outer surfaces to 100 hours are shown below:

<u>Time, hr.</u>	<u>Cover Temperature, °F</u>		
	<u>Inner</u>	<u>Outer</u>	<u>ΔT</u>
0	1200	1200	0
10	1068	1195	131
25	988	1164	176
50	934	1092	157
75	904	1027	123
100	880	975	95

The results show that the cover is insulated extremely well from the sodium pool. Also, it is shown that the thermal resistance of the cover is high. The maximum gradient across the cover is about 44% of the step change in the sodium pool temperature. Since the system time constants are the same for a heat-up as a cool-down, the lag shown in the cool-down calculation also exists

in the heat-up and the dimensionless temperature transient is the same for both cases. Thus, the start-up temperature transient can be calculated from the data listed above.

b. Meltdown Crucible Calculations

D. P. Schivley

Calculations were made to determine the fraction of core meltdown which could be contained by the crucible without self-destruction. The stainless steel crucible can be destroyed if the heat generation in the molten fuel is greater than that which can be conducted into the steel or transferred to the sodium pool at the free surface. In addition, the crucible may be melted and destroyed if the heat flux (after the sodium pool becomes saturated) causes burnout and film type boiling on the bottom and free surfaces of the crucible.

Initial calculations examined the full core meltdown. The computations indicate that the crucible is completely molten after about 10 minutes. No further analysis was made for this case because it does not appear that a crucible can be made which can contain the entire core meltdown.

Similar calculations were then made assuming meltdown of the fuel in seven of the 68 ducts. These computations indicate that very little, if any, stainless steel becomes molten. The transient calculations indicate that quasi steady state is established after about 15 minutes and the maximum crucible temperature is about 2000 °F. Conservatively, the sodium does not reach boiling temperature until after eight hours. With seven ducts (average tube power) the average surface heat flux over the bottom and free surfaces of the crucible is about 10,000 B/hr-ft². Although film boiling data are not available for our geometry, it is believed that film boiling does not occur at the bottom of the crucible. However, if film boiling did occur the maximum steel temperature would still be only about

2100 °F as compared to a melting at ~2600 °F. Thus, it is concluded that the crucible can contain at least seven fuel bundles and perhaps more.

These calculations do not include effects of chemical or metallurgical combinations of molten core constituents with the meltdown crucible, or with sodium. Low melting point eutectics or reduced nuclear fuel can modify the results of these calculations significantly.

c. Advanced Core Analysis

R. B. Rothrock

Potential carbide core designs were developed for the conical and skewed core arrangements to verify that the dispersed reactor designs currently under study are capable of achieving future testing goals. Principal conclusions from this effort are:

- Both conical and skewed arrangements appear to be capable of achieving a peak flux of around 10^{16} n/cm²-sec with gas bonded mixed carbide fuel.
- In both cases, it is necessary to reduce the number of core subassemblies to approximately a 61-position core, based on the present dispersed core subassembly size of 4-1/2 in. across flat diameter.
- The use of BeO or other moderator may be required to achieve an adequate Doppler coefficient in future compact cores.
- The use of longer fission gas plenums in future cores emphasizes the advantages of enlarging the flow duct cross section as closely above the top of the fuel as the diverging geometry permits, particularly for high power density cores.

B. REACTOR AND SYSTEMS ANALYSIS

R. E. Peterson

1. Systems and Instrumentation Analysis

H. G. Johnson

a. Digital Simulation of the Reactor and Heat Transport System

C. D. Flowers

The digital reactor and primary heat removal system models have been used to calculate loss of BPA electrical power transients and primary loop pipe rupture transients. The studies are a part of the Systems Analysis effort to determine the design requirements of an emergency core cooling system that will maintain adequate core heat removal capability and will prevent coolant boiling when the main heat transport system becomes inoperative.

Loss of BPA transients for the initial oxide core operating at reduced power with two heat removal loops have been studied. (Initial core 3-Loop, full power loss of BPA transients was reported in the quarterly report BNWL 880.⁽¹⁾) Loss of BPA transients has been studied for an advanced mixed carbide core operating at full power (3-loops) and reduced power (2-loops). Steady-state conditions for the cases investigated are presented in the next section.

The loss of coolant transients that have been studied are for pipe ruptures downstream of the pump near the reactor vessel inlet nozzle. Ruptures from 0.05 ft² to 3 ft² have been investigated for both the initial oxide core and advanced mixed carbide core operating with three loops full power and two loops reduced power.

1. Fast Flux Test Facility Quarterly Progress Report, April, May, June, 1968, BNWL-880. Pacific Northwest Laboratory, Richland, Washington, October 1968.

System Initial Conditions. As stated, both initial core and advanced core transient studies have been made with both cores operating on three loops with full power and on two loops with reduced power conditions. The following steady-state conditions were assumed.

TABLE 3.B.I. *Initial Core and Advanced Core Transient Studies*

Initial Core - Oxide Fuel

3-Loop Operation	2-Loop Operation
400 MW Total Power	264 MW Total Power
300 °F Vessel ΔT	300 °F Vessel ΔT
900 °F Vessel Outlet	900 °F Vessel Outlet

Advanced Core - Mixed Carbide Fuel

3-Loop Operation	2-Loop Operation
450 MW Total Power	300 MW Total Power
400 °F Vessel ΔT	400 °F Vessel ΔT
1200 °F Vessel Outlet	1200 °F Vessel Outlet

The advanced FTR core that was studied contains a mixed carbide (PuC-UC) fuel with stainless steel cladding. The kinetics parameters and reactivity feedback coefficients are assumed to be the same as the initial oxide core values. The fission product heat generation parameters are also assumed to be the same.

The models used to calculate the system transients simulated an eddy-current clutch pump drive system with a 2-sec clutch time constant. A wound rotor motor drive with liquid reostat control is also a possible concept for pump drive. Additional transients using a wound rotor pump drive simulation will be studied at a later date. Reactor scram data for program input is based on a 1-g rod acceleration with 15\$ total reactivity insertion. This assumes four rods, each worth 5\$, with one rod sticking.

The sequence of events for the loss of BPA transients is as follows:

- (1) Loss of BPA power at 1 sec
- (2) Reactor scram at 1 3/4 sec, eddy-current clutch de-energized at scram
- (3) Clutch is re-energized and emergency pump speed controller takes over at measured pump speed of 300 rpm. Emergency controller set point is 100 rpm pump speed.
- (4) Emergency cooling provided by:
 - (a) Pony motors either engaged automatically at motor speed of 125 rpm, or 20 sec after scram, or
 - (b) Flow decay to natural circulation cooling.

The following sequence of events was assumed for the pipe rupture transients that were investigated:

- (1) Pipe rupture at 2 sec
- (2) 1-g, 15\$ reactor scram at 4 sec
- (3) Eddy-current clutch de-energized at 5 sec for flow coastdown
- (4) Clutch re-energized at 300 rpm pump speed, and
- (5) Emergency speed controller set point is 120 rpm, with the main motors driving the pumps.

Results & Conclusions

Loss of BPA Power Transients - Two-Loop Initial Core. For the transient conditions studied it appears that a separate emergency core cooling system is not required for loss of electrical power transients with the initial FTR core operating at reduced power (264 MW) with two coolant loops functioning. Pony motors or natural circulation will provide adequate core cooling to prevent coolant boiling in the reactor hot channel. The same results were obtained for the initial core operating with three loops at full power.

Loss of BPA Power Transients - Three- and Two-Loop Advanced Core. For the loss of BPA Power Transients studied, hot channel coolant boiling does not occur in the hypothetical advanced mixed carbide core. Pony motor operation or natural circulation is adequate to remove the core stored and decay heat.

Rates of average core outlet temperature change following the initial peak, the relative average fuel-coolant heat transfer rates per unit length, (UA') (ΔT) , and core coolant ΔT values for the two cores are compared below for the case where one pony motor fails.

	Avg. Core Outlet Max. Rate of Temp. Change $^{\circ}\text{F}/\text{sec}$	Avg. Core Outlet Total Max. Temp. Change $^{\circ}\text{F}$	Fuel-Coolant $UA' (\Delta T)$	Core ΔT
Advanced Carbide Core	~ 200 $^{\circ}\text{F}/\text{sec}$	~ 265 $^{\circ}\text{F}$	~ 800 $\frac{\text{Btu}}{\text{sec ft}}$	400 $^{\circ}\text{F}$
Initial Oxide Core	~ 30 $^{\circ}\text{F}/\text{sec}$	~ 125 $^{\circ}\text{F}$	~ 480 $\frac{\text{Btu}}{\text{sec ft}}$	300 $^{\circ}\text{F}$

Advanced FTR core transient results are applicable only to the assumed core design data which are listed in Table 3.B.I. This core design is purely conceptual and is only representative of future cores.

Pipe Rupture Transients - Break Near Reactor Inlet. For the transients studied it was found that the reactor hot channel coolant does not boil in either the advanced or the initial cores if the check valve closes in 3 sec after reverse flow begins. However, hot channel boiling was observed for the cases of breaks larger than 0.5 ft^2 where the check valve was assumed to stick open. As well, negative flows through the reactor were detected for breaks greater than 0.2 ft^2 when the check valve remained open. This is due to the fact that the flow path from the inlet plenum out the pipe break has a lower resistance than the flow

path from the inlet plenum through the reactor. Transient results for a 3 ft² break with the advanced core, 3-loop operation are presented in Figures 3.B.1, 3.B.2, and 3.B.3. This transient provided the most severe hot channel results for the cases where the check valve closed. Figure 3.B.1 presents the reactor power, coolant temperatures, and average fuel temperature. Primary coolant flow rates are presented in Figure 3.B.2 and sodium levels and break pressure are presented in 3.B.3. Twelve feet must be added to the vessel level to obtain the same absolute elevation as the pump barrel level. A reactor power controller is included in the simulation, and its action is seen in Figure 3.B.1 by observing the power curve. The main conclusion is that a reliable check valve is required to prevent sodium boiling. This check valve should be located as close to the reactor inlet plenum as is possible. This will increase the chance of the break being upstream of the check valve. Additional loss of coolant transients for ruptures upstream of the pump in the low pressure pipe legs are planned. Model development for these cases is underway. Ruptures between the check valve and the inlet plenum are also being studied.

b. Analog-Hybrid Simulation of the Reactor and Heat Transport System

R. D. Behnam and A. L. Gunby

(1) Model Development. Heat transfer coefficients for core, IHX and DHX were programed as appropriate functions of flow and temperature. The effectiveness of the variable coefficients was checked by performing simulated transients with constant and then variable heat transfer. For the range of flows (5 to 100%) and temperatures (400-1200 °F) currently of interest the following conclusions were drawn:

- (a) The core (fuel-to-coolant) heat transfer coefficient may be a constant without significant loss in accuracy.
- (b) The IHX heat transfer coefficient may be a constant without significant loss in accuracy.

FIGURE 3.B.1. Reactor Power and Temperature Responses for 3-ft² Pipe Rupture near Vessel Inlet, Advanced Carbide Core, 3-Loop, Full Power Operation, with Check Valve Closing in 3 Seconds

FIGURE 3.B.2. Primary Loop Flow Rates for 3-ft² Pipe Rupture near Vessel Inlet, Advanced Carbide Core, 3-Loop, Full Power Operation, with Check Valve Closing in 3 Seconds

FIGURE 3.B.3. Sodium Levels, Break Pressure, and Integrated Break Flow for 3-ft² Pipe Rupture near Vessel Inlet, Advanced Carbide Core, 3-Loop, Full Power Operation, with Check Valve Closing in 3 Seconds

(c) The DHX heat transfer coefficient must be a function of air flow and temperature.

Accordingly, the DHX heat transfer coefficient is a function of air flow and temperature while the others remain as constants.

c. Heating and Ventilation System Analysis

R. N. Madsen and S. H. Christensen

H&V System Model Development. The development of the mathematical model representing the Heating and Ventilation (H&V) System has been completed. The model of the H&V System is operational for the following areas:

- Containment Work Area - 100 °F allowed
- Refueling Hot Cell - 100 °F allowed
- Gas Systems Space - 120 °F allowed
- Fuel Handling and Examination Area - 100 °F allowed
- Secondary Sodium Heat Transport Building - 120 °F allowed

The transient temperature response in each space was obtained for each of six conditions representing the range of effects of power losses. Results are summarized in Table 3.B.II.

TABLE 3.B.II. *Temperatures for Most Severe Shutdown Conditions Allowable*

	1-Hour Transient Temp, °F	2-Hour Transient Temp, °F
Containment Work Area	87	90
Refueling Hot Cell	89	92
Fuel Handling & Exam. Area	85	89
Gas Systems Space	102	103
Secondary Sodium Heat Transport Building		
Pump Level	121	90
Piping Level	121	85
Cleanup Level	118	85
Auxiliary & Equip. Level	118	105

Loss of normal electrical power during containment mode does not appreciably increase the temperature in the Containment Work Area or the Refueling Hot Cell.

2. Reactor Core Analysis

P. D. Cohn

a. Engineering Hot Channel Factors

T. J. Bennett

An analysis was completed in which the engineering hot channel factor (HCF) for the FFTF core was determined. A fuel hot channel factor of 1.13, which corresponds to a maximum linear generation rate of 14.3 kW/ft was calculated using a quasi-statistical method of analysis. Fuel and inner cladding maximum variances were determined to be 341 °F and 158 °F, respectively.

b. Temperature Field in Cladding and Coolant Resulting from Asymmetrical Cooling

D. C. Kolesar

The temperature field in the cladding and coolant belonging to the hottest FFTF fuel pin was investigated using MANTA (PNL modified version). Particular attention was devoted to disturbances in the field resulting from effects of a wire wrap. A consequence of lower coolant flow rates in such regions where the flow area is more confined is a local hot spot or a "temperature peaking" phenomenon in the neighborhood of the gap between flow channels.

The pin and bundle characteristics employed in this analysis are those typical of the FFTF design.

MANTA employs a diffusional mixing model with a multiplying factor, k , inserted to permit correlation with experimental data. Here results are reported for no mixing ($k = 0$), for molecular diffusion ($k = 1$), and for conservative mixing ($\beta = 0.01$) employed in earlier COBRA code calculations ($k = 5$).

Figure 3.B.4 provides a comparison among the three mixing models in terms of maximum temperature difference in the circumferential direction in the coolant as a function of distance up the fueled portion of the fuel pin. This maximum temperature difference is between the hottest interfacial value at the wire-coolant boundary and the coolest coolant node. It is felt that mixing characterized by $k = 5$ describes the system in a conservative manner. Thereby, the highest temperature peaking is observed to occur at the center of the fueled portion of the pin and for $k = 5$ it possesses a value of about 47 °F above the bulk mean coolant temperature. The bulk mean value can be shown to be slightly hotter than the coolest coolant node.

Temperature distributions through the cladding are displayed in Figure 3.B.5 for various distances up the fueled portion of the fuel pin. A mixing constant of $k = 5$ is regarded as conservative for design purposes.

c. Maximum Energy Release Caused By Expansion of Molten Fuel-Sodium Mixture

A. Padilla

During a major nuclear excursion in a fast reactor where massive fuel melting has occurred, cladding failure could result in a violent molten fuel-sodium interaction. The mixture of molten fuel and sodium has a potential for doing work and the maximum potential can be determined using a thermodynamic approach.

The results of our analysis are given in Table 3.B.III as compared with data computed by the method of Hicks and Menzies⁽¹⁾ which assumes:

- Thermal equilibrium between fuel and sodium during the expansion

1. E. P. Hicks and D. C. Menzies. Theoretical Studies on the Fast Reactor Maximum Accident, ANL-7120. Argonne National Laboratory, Argonne, Illinois, October 1965.

- Equilibrium temperature below the sodium critical point.
- No phase change for the fuel is also shown for comparison.

FIGURE 3.B. 4. Temperature Peaking as Represented by the Maximum Circumferential Temperature Difference of the Coolant as a Function of Axial Distance Along the Fueled Portion of the Hottest Pin with the Wire Wrap Located in the Gap

FIGURE 3.B. 5. Circumferential Temperature Distribution in Cladding for Radius Fixed at Midcladding Value and for a Mixing Constant of $k = 5$

TABLE 3.B.III. *Energy Release in Molten Fuel-Sodium Interactions*

	Initial Fuel Temperature, °K	Energy, MW-sec
Adiabatic Expansion (no heat transfer with fuel during expansion)	3800	493
	4000	541
	4850	702
Heat transfer until fuel starts to solidify, adiabatic expansion thereafter	4000	978
Perfect heat transfer during entire expansion:		
Present method	4000	1288
Hicks and Menzies	4000	1513

d. Poison Rod Temperatures

A. Padilla

Conceptual poison rod configurations were selected based on previous neutronics and heat generation studies. The absorber material for the in-core shim rods is tantalum in a 7-rod array. The absorber material for the in-core safety rods is boron carbide, nominally 80% of theoretical density and enriched to 26.2 at.% B-10, in a 61-rod array. The absorber material for the peripheral shim rods is natural boron carbide, nominally 80% of theoretical density, in a 61-rod array.

Neutronics calculations have shown that the volume of tantalum must be approximately twice that of natural boron carbide to achieve the same reactivity worth in either the core or the reflector. Volume fractions required for tantalum and boron carbide are shown in Table 3.B.IV. The same calculations show that the burnup rate for tantalum is approximately

one-fourth that of boron. Therefore, the use of tantalum is attractive, if practical configurations (containing approximately 50% poison material) can be found.

TABLE 3.B.IV. Poison Requirements

<u>Parameter</u>	<u>In-Core Shim Rod</u>	<u>In-Core Safety Rod</u>	<u>Peripheral Shim Rod</u>
Ta, vol%	45	60	55
Natural B ₄ C, vol%	22	29	22

Table 3.B.V shows the various poison rod dimensions based on the required volume fractions in Table 3.B.IV. When the required volume fraction for tantalum is below 0.50, it is possible to utilize a 7-rod configuration. For the higher volume fractions, it becomes necessary to utilize a single rod which has a center hole to reduce the maximum internal temperatures. The tantalum and boron carbide multirod configurations are enclosed by an outer duct of the same dimensions as the driver fuel subassembly and an inner structure which avoids placing a direct load on the poison rods during a scram.

Heat generation rates used in this study have been calculated and are shown in Table 3.B.VI. The heating rates for the safety rods were assumed to be the same as those for the peripheral rods.

The maximum poison temperatures for the various rod configurations in Table 3.B.V were calculated assuming the heat generation rates in Table 3.B.VI. The results are shown in Table 3.B.VII. The thermal conductivity of 5.2 Btu/hr-ft-°F was assumed for the boron carbide corresponding to an 80% theoretically dense rammed and sintered pellet. The heat transfer from the pellet to the stainless steel cladding was assumed to occur by conduction across a 0.006-in. diametral helium gap. A thermal conductivity of 38 Btu/hr-ft-°F was assumed

TABLE 3.B.V. *Poison Rod Dimensions (in.)*

	<u>Rod OD</u>	<u>Rod ID</u>	<u>Cladding Thickness</u>	<u>Rod Spacing</u>
<u>In-Core Shim Rod</u>				
Tantalum (7 rods)	1.300	0	0.030	0.0625
Natural B ₄ C				
7 rods	1.300	0.576	0.080	0.0625
19 rods	0.773	0.205	0.070	0.0625
57 rods	0.530	0	0.050	0.071
61 rods	0.416	0	0.040	0.056
<u>In-Core Safety Rod</u>				
Tantalum (Annulus)	3.974	0.750	0.030	
Natural B ₄ C				
7 rods	1.300	0	0.080	0.0625
Enriched B ₄ C				
61 rods	0.416	0	0.040	0.056
<u>Peripheral Shim Rod</u>				
Tantalum (Annulus)		Same as In-Core Safety Rod		
Natural B ₄ C		Same as In-Core Safety Rod		

TABLE 3.B.VI. *Heat Generation Rates in Poison Rods*

	<u>In-Core Shim Rod</u>			<u>Peripheral Shim Rod</u>		
	<u>B₄C</u>	<u>Ta</u>	<u>Stainless Steel</u>	<u>B₄C</u>	<u>Ta</u>	<u>Stainless Steel</u>
Neutron Capture, W/g	48.8	11.4	0.9	26.6	6.2	0.5
Gamma Radiation, W/g	9.5	9.5	9.5	4.6	4.6	4.6
Neutron Scattering, W/g	10.4	1.7	2.2	5.0	0.8	1.1
Total, W/g	68.7	22.6	12.6	37.2	11.7	6.2

TABLE 3.B.VII. *Maximum Poison Temperatures (°F)*

	<u>Natural B₄C</u>	<u>Enriched B₄C</u>	<u>Tantalum</u>
<u>In-Core Shim Rod^(a)</u>			
7 rods	1811		1596
19 rods	1346		
37 rods	1158		
61 rods	1017		
<u>In-Core Safety Rod^(b)</u>			
Annular rod			
7 rods	1686		1997
61 rods		741	
<u>Peripheral Shim Rod^(a)</u>			
Annular rod			
7 rods	1271		2172
19 rods	1037		
37	943		
61	872		

(a) *Assumes a coolant temperature of 725 °F*

(b) *Assumes a coolant temperature of 550 °F*

for the tantalum with no resistance between the tantalum and the stainless steel cladding, corresponding to a metallurgical bond or a very thin sodium bond.

The choice of boron carbide in a 61-rod array for the in-core safety rod and the peripheral shim rod is based on the consideration that the maximum calculated temperatures of 741 and 872 °F represent a minor extrapolation of the irradiation data currently available. However, future experimental

data may show that much higher operating temperatures are possible, thus permitting a smaller number of larger rods. Tantalum is preferred over boron carbide for the in-core shim rod because it is judged at the present time that tantalum operating below 1600 °F represents less uncertainty than boron carbide above 1000 °F.

C. REACTOR PHYSICS AND OPERATIONS

R. E. Heineman

1. Nuclear Analysis

W. W. Little, Jr.

a. IDX, A One-Dimensional Diffusion Theory Cross Section Generating Code

R. W. Hardie and W. W. Little, Jr.

A 1-dimensional diffusion theory program (IDX) designed to compute resonance-shielded and collapsed-group cross sections, has been developed at PNL for use in fast-reactor analyses. Resonance-shielded cross sections are calculated using data (infinite dilution cross sections and resonance shielding factors) in the "Russian" format. Interpolation schemes are used to compute shielding factors applicable to specific compositions. IDX will also accept input cross sections in the "DTF" format (i.e., resonance-shielded cross sections). Collapsed-group cross sections by reactor zone are calculated using flux-weighting.

The spatial flux profiles and eigenvalue are computed by standard source-iteration techniques. Convergence is accelerated using fission source over-relaxation. Adjoint calculations are performed by transposing the scattering matrix and fission source and then inverting the group order of the input data.

Criticality searches can be performed on time absorption (α), material concentrations, region dimensions, and buckling. Alpha and k_{eff} can be used as parametric eigenvalues.

The format of the input data (e.g., cross sections, flux dumps, geometry and composition specifications) is compatible with the Los Alamos 1- and 2-dimensional transport codes DTF-IV and 2DF, the BNW 2-dimensional diffusion theory code 2DB, and the BNW perturbation code PERT-IV. All five codes use the same input module.

All programing is in FORTRAN-IV. Variable dimensioning is used to make maximum use of the available fast memory.

b. A Multigroup 3D Synthesis Code

H. S. Zwibel

RSYN is a multigroup 3D synthesis diffusion analysis code now in preparation. Full 2D diffusion analysis is provided by the diffusion analysis routines in 2DB. The flux distribution in the third dimension is synthesized from a 1D spacially-dependent linear combination of the 2-dimensional fluxes obtained by analyzing 2D sections with 2DB. Expected availability is February 1969. The major problem area is uncertainty about the success of the selected strategy for a large number of energy groups and a large number of 2D sectional cuts.

c. ETOX, A Code to Calculate Group Constants for Nuclear Reactor Calculations

R. E. Schenter, J. L. Baker, and R. B. Kidman

Computer code ETOX has been written to calculate group constants for fast reactor calculations from currently available cross section data. The code is specifically designed to use the Evaluated Nuclear Data File (ENDF/B)⁽¹⁾ as input for microscopic cross section values. Output from this code includes punched cards in the Russian (Bondarenko)⁽²⁾ format which can be used as input to the fast reactor codes FCC-IV⁽³⁾ and 1DX.

1. Honeck, Henry C., ENDF/B, Specification for an Evaluated Nuclear Data File for Reactor Applications, BNL-50066 (T-467) Brookhaven National Laboratory.
2. Bondarenko, I. I., Group Constants for Nuclear Reactor Calculations, Consultants Bureau, New York (1964).
3. Little, W. W., Jr., R. W. Hardie, FCC-IV-A Revised Version of the FCC Fundamental Mode Fast Reactor Code, BNWL-450, Pacific Northwest Laboratory, Richland, Washington. 1967.

The output includes infinite-dilute group cross sections, inelastic scattering probability matrices, and temperature dependent self-shielding factors for arbitrary values of σ_0 (total cross section per atom).

A document describing ETOX is being prepared.

2. Experimental Physics

R. A. Bennett

A. NPTR-PSAR Static Reactivity Calculations

S. L. DeMyer

Static reactivity calculations have been made in support of the Preliminary Safeguards Analysis Report (PSAR) for the Nuclear Proof Test Reactor (NPTR). The results of these calculations are presented.

For the purposes of these preliminary calculations, the Annular Conical Core midplane dimensions and loading pattern were assumed. General areas covered by these calculations are reactivity worths of loading errors and reactivity worths of the NPTR control systems. A summary of the calculated results is presented in Table 3.C.I.

TABLE 3.C.I. *Calculations and Reactivity Worths*

<u>Calculation</u>	<u>Worth - Δk</u>
Maximum in-core fuel loading error	0.0113 ($\sim 3.5\%$)
Maximum reflector loading error	0.0027 ($< 1.0\%$)
Total strength of reflector control system	0.0368
Total strength of in-core control system	> 0.0702
Total strength of tertiary control system	Design and composition dependent

b. Phase B FTR Critical Experiments Program

W. R. Young and J. V. Nelson

Experimental work on ZPR-III Assemblies 51 and 52 is complete. Assembly 51 was an irregular right cylinder whose driver zone and reflector compositions approximated those quantities in the split-core reference FTR.⁽¹⁾ Figure 3.C.1 shows a cross section of Assembly 51 in the ZPR-III. This assembly, which had a critical mass of 212 kg ^{239}Pu in a 307 liter volume, was the object of an extensive experimental program. Assemblies 52a through 52f had the same driver zone and reflector compositions as Assembly 51 but, in addition, these cores had simulated test zones of various shapes and volumes.^(2,3) The experimental program of Assemblies 52 did not provide for the detailed nucleonics measurements of Assembly 51 but concentrated on the effect of test zones on critical mass, the relative flux to power ratio, and the peak to average flux ratio.⁽²⁾

Reporting of experimental results from Assembly 51 is not complete at this time. A number of important results have been received from ANL and are being analyzed by PNL.

c. Radiation Levels Near NPTR Fuel

H. C. F. Ripfel

The NPTR will use fuel containing plutonium of a variety of isotopic compositions; hence, radiation dose levels are a matter of concern in the area of fuel handling where they

-
1. Reactor Development Program Progress Report, ANL-7419, January 1968, p. 11.
 2. Reactor Development Program Progress Report, ANL-7460, June 1968, p. 16.
 3. Reactor Development Program Progress Report, ANL-7478, July 1968, p. 11.

Driver Zone, 307 &

Critical Mass, 215 kg Fissile Material

1-4 Spiked Safety Drawers

5 Control Drawer

☒ "Lean" Drawers

FIGURE 3.C.1. Phase B ZPR-III Assembly 51

might put limitations on the operation of the facility. It is necessary to assess these levels to provide the designer with radiation exposure information upon which to base the design.

Areas which have a potential for significant radiation can readily be identified from the fuel handling scheme, as shown in Figure 3.C.2:

- (1) Single fuel pin handling - preferably manually - in the pin exchange hood.
- (2) Observation of the hood operations - preferably visually.
- (3) Handling of fully assembled modules by crane for transportation to and from the facility, the pin exchange hood, the storage vault, the cell. This includes insertion into and removal from the core-- preferably with close-up observation.

For present purposes, consideration is restricted to external exposure only, as the fuel is seal welded within several mils of steel. This cladding also provides enough shielding to eliminate radiation other than gamma rays and neutrons.

It is envisaged that the NPTR will have two types of fuel which can be characterized by the isotopic composition of the plutonium, i.e., one will have a Pu-240 content of ~10% (Material I), and another of ~23% (Material II).

Some realistic assumptions have been made in anticipation of already known or predictable design features. The pin exchange hood is designed as a glove box (to provide protection in the case of an accidental breaching of the fuel pin cladding). The gloves are made of Neoprene, 33 mils thick. The cladding of the fuel pins comprises 16 mils of stainless

steel. The observation window of the hood consists of plastic 3/8-in. thick. The fuel modules have cans of wall thickness of 150 mils of stainless steel. The operator is to wear gloves of the same kind as described above. The reactor loading device provides a concentric steel shielding around the modules of 0.5-in. wall thickness. An additional shielding of two inches of Boron-polyethylene is considered to reduce the neutron radiation level.

Radiation dose rates, for single pins and pin bundles of these materials, at the surface and at a distance of two feet (which is the approximate position of the operator relative to the material), are given in Table 3.C.II.

3. Radiation and Shield Analysis

W. L. Bunch

a. Application of Synthesis Techniques to Shielding Problems

H. S. Zwibel

Variational synthesis methods have been successfully applied to a variety of problems. With this approximation technique, a multivariable problem is reduced to one of lesser dimensionality. The reduction is carried out by calculus of variation techniques with emphasis on producing optimum accuracy in some integral quantity, referred to as the characteristic of interest.

In order to investigate the possible applicability of these methods to deep penetration problems, an angle-space synthesis approximation to the Boltzmann transport was applied to several problems.^(1,2) In each of these, the characteristics of interest (criticality constant for a homogenous slab

-
1. H. S. Zwibel, "An Angular Synthesis Approximation to Thin Critical Slabs," Trans. Amer. Nuc. Soc. 10, No. 1, June 1967.
 2. H. S. Zwibel, B. Bowes, "Space-Angle Synthesis," Trans. Amer. Nuc. Soc. 11, No. 1, June 1968.

TABLE 3.C.II. NPTR Radiation Levels

Material	Dose Rate, mrem/hr ^(a)			
	Neutrons	Gamma	Neutrons	Gamma
	Total		Total	
	I		II	
Single Pin:				
Surface	3.6	107	10	250
Through 16 mils of steel and 33 mils of Neoprene	111		260	
At 2-ft distance from side.	0	0.175	0.03	0.41
Through 16 mils of steel and 3/8-in. of Lucite	0.175		0.44	
217 Pin Subassembly:				
Surface	47	5	150	12
Through 166 mils of steel and 33 mils of Neoprene	52		162	
At 2-ft distance from side	1.6	-	4.4	0.01
Through 0.666 in. of steel and 2 in. of Boron-polyethylene	1.6		4.41	

a. Tolerance dose rates for continuously occupied areas: 0.2 mrem/hr. Higher dose rates require a timing of the operations.

reactor and total flux in a noncritical slab with a uniform source) was calculated with high accuracy.

With this encouragement a shielding problem was then attempted. A source was impressed at the left end of a uniform slab and the emergent flux at the right end was calculated, but in this case without success. That is to say, ordinary diffusion

theory was just about as accurate. The reason for this failure of the variational synthesis method was probably induced by the nature of the characteristic of interest. For the successful examples, the characteristic of interest was obtained by averaging the flux over the entire system. This exploits the full potential of the variational method in that errors in the flux are averaged out (see Reference 2 for a good example of this). The characteristic of interest for the shielding problems admitted no such cancellation. Because of this, synthesis methods as applied to such stringent problems are apparently not successful.

There are, however, areas in shielding for which one might profitably employ variational synthesis. For example, the decay heat in a shield is obtained by integrating the flux (times a cross section) over the entire shield. Such a characteristic of interest can probably be calculated by synthesis methods with good accuracy.

b. Hot Cell Shielding Considerations

W. L. Bunch, C. A. Mansius, and L. D. O'Dell

Consideration has been given to the use of a hot cell above the FTR in which refueling operations would be carried out. Such a hot cell would serve several purposes, including a radiation barrier. Three studies were made to consider the effectiveness of a hot cell shield: for sodium spills, from sodium vapors, and from fission products following a design basis accident. It is concluded that a hot cell designed to accommodate irradiated fuel would be satisfactory from these three aspects.

For this study, a sodium spill was considered to contain about a pound of sodium and form a disc about 2 cm thick and 18 cm in diameter. Specific activity of primary system sodium has been calculated at 0.05 Ci of ^{24}Na and 10 μCi of

^{22}Na per cm^3 . The gamma intensity of 100 cm from the disc source was calculated to be 50 R/hr and 3 mR/hr, respectively, from the two sodium isotopes. The ^{24}Na activity decays with a half life of 15 hours; therefore, the hot cell could probably be entered within a few days following a spill to permit contact cleanup. If a pound of sodium also contained one-millionth of the equilibrium fission product inventory in a fuel subassembly, these would generate a gamma field of about 5 R/hr at zero decay time and about 0.5 R/hr a week later. Cleanup of a sodium spill might be complicated by the presence of fission and/or corrosion products.

If a hot cell is used above a reactor concept in which the cover shield is removed to facilitate refueling, then sodium vapors would evaporate from the pool and deposit on the cell walls. Calculations indicate that radioactivity of the sodium deposit would not prevent personnel access into the hot cell, though long-lived fission or corrosion products might cause significant levels. This conclusion assumes that entry could be delayed sufficiently to permit the decay of ^{24}Na . Calculations were based on a pool temperature of 400 °F with an evaporation rate of 10^{-6} pounds of sodium per hour per square foot. Activity of the pool sodium was assumed to build up exponentially with a 2.6 year half life toward a saturated activity of 20 $\mu\text{Ci/g}$. Radioactive sodium was considered to be deposited during shutdowns when the cover was removed, and the residual deposit then decayed during succeeding operating periods. The equilibrium gamma field was estimated to approach about 0.1 mR/hr because of the ^{22}Na , after about ten years of operation. Depending on fuel failure history, fission products in the sodium might increase the gamma field by one or two orders of magnitude.

Following a design basis accident, it was assumed that all of the sodium and fission products would be vaporized and contained within a hot cell. The 30-day integrated exposure received by an operator remaining at a control position located 20 ft from the cell was estimated based on gamma intensities calculated using the computer code ISOSHL. A relatively insignificant exposure would be received from all sources through a hot cell wall containing 400 g/cm^2 . If, more realistically, it were assumed only the noble gases would remain suspended then a shield containing about 200 g/cm^2 would probably provide sufficient personnel protection. Actual shield requirements are dependent on the selected design basis accident together with tolerable radiation criteria yet to be established.

c. Gamma Intensity Near Closed Loop Pipes

C. A. Mansius

It will be necessary to shield personnel and equipment from the radioactive coolant, corrosion, and fission products contained in closed loop pipes both during operation of the loop and following shutdown. Shield requirements are important in some concepts because of the limited amount of space above the reactor cover. This region must accommodate not only the closed loop piping, but also control rod drives, electrical lines, and fuel handling equipment, yet be shielded to provide access for maintenance and repair work.

The gamma intensity as a function of distance and shield thickness was calculated for two different size pipes containing radioactive sodium with a specific activity of 0.01 Ci of ^{24}Na per cm^3 . Both pipes were assumed to be made of stainless steel, with the 8-in. diam pipe having 1/2-in. thick walls and the 3-in. diam pipe having 1/4-in. walls. Specific activity of the sodium is dependent on design of the closed loop, and could be a significant factor different than the assumed value.

From Table 3.C.III it can be seen that a little more than three inches of iron or about one inch of depleted uranium will reduce the gamma intensity by a factor of ten. For pipe diameters between the tabulated sizes, the gamma intensity will be proportional to the cross sectional area of the contained sodium. For distances in the range tabulated, the intensity is inversely proportional to the distance from the pipe surface.

TABLE 3.C.III. Gamma Intensity Near Closed Loop Pipes in R/hr

Pipe Diameter, in.	Distance From Pipe, ft	Shield Material and Thickness, in.			
		Air	Iron(2)	Iron (4)	Depleted Uranium(2)
8	1	1800	460	76	11.0
	5	370	100	24	2.9
	10	170	45	11	1.3
3	1	370	100	24.0	2.4
	5	64	17	4.6	0.53
	10	22	6.2	1.7	0.19

If it is assumed that a particular component or piece of equipment will deteriorate after exposure to about 10^7 or 10^8 rads, then about two to four inches of iron or equivalent will be required to protect such equipment located in the vicinity of the closed loop piping. If personnel access for maintenance work near the sodium-filled pipes is delayed for about a week or ten days following shutdown, then this amount of shielding would also be adequate for the residual ^{22}Na activity. However, if personnel access will be required during operation or immediately following shutdown, significantly thicker shields will be required.

These calculations provide realistic estimates of shield requirements for conceptual design purposes. More detailed calculations will be required for specific system arrangements as the design progresses.

d. Failed Fuel Detection

W. L. Bunch, C. A. Mansius, and L. D. O'Dell

It is desired that the FTR be instrumented to detect and locate failed fuel elements. A number of physics calculations were made to assist in the selection of a conceptual design for the failed fuel element detection and location (FFEDAL) system. Typical calculations include an estimate of the gamma field below the cover and the effect of burnup on tag material.

The gamma field below the cover is from the radioactive sodium as well as the reactor system. If the depth of the sodium pool above the reactor core exceeds about ten feet, then the gamma field results almost entirely from the radioactive sodium and is estimated to be greater than 10^5 R/hr. A field of this magnitude makes it impractical to locate FFEDAL detectors below the cover.

The three stable xenon isotopes 124, 126, and 129, have been suggested for use as tag materials whose isotopic ratio could be associated with a specific fuel subassembly. If burnup of the isotopes alters the initial ratio significantly prior to failure of the fuel cladding, then the tag material will not locate the failure precisely. Spectrum weighted, theoretical cross sections of 130, 61, and 320 millibarns for the three isotopes of xenon, respectively, were calculated by R. E. Schenter⁽¹⁾ using an optical potential model. At a goal exposure of about 3×10^{23} nvt, these cross sections indicate that tag increments of about 5% would be required to avoid ambiguity resulting from differential burnup of the tag material.

1. R. E. Schenter, Battelle-Northwest, unpublished data, Sept. 1968.

D. SAFETY AND RELIABILITY ANALYSIS

D. E. Simpson

1. Reliability Analysis

R. A. Harvey

a. Preliminary Reliability Analysis of the Closed Loop System

S. S. Hintze

The purpose of the analysis was to predict the reliability of the conceptual Closed Loop Test System and to determine its effect on reactor unavailability. It is anticipated that the information will be useful to the system designers in the choice and number of components and also in the number and configuration of the closed loops.

Because of the lack of data on components operated in 900 °F-1200 °F sodium, extrapolations were made to arrive at the failure rates; hence the failure rate information is considered preliminary.

An analysis of the total system for one loop was performed giving a system failure rate of 43.56×10^{-6} and a system mean-time-to-restore of 131 hours. The system availability was calculated to be 0.99352 and the associated reactor unavailability as 0.00625 or 6.25 hr/1000 hr per closed test loop.

Failure rates and mean-times-to-restore are given in Table 3.D.I for closed loop components.

b. Fuel Pin Reliability

H. C. Martin

To achieve a plant availability of 75%, the down time attributable to fuel failures must not exceed 10 hr of down time per 1000 hr of calendar time, or 87.66 hr/year. Since all fuel failures may not occur in fuel pins, 80 hr/year is assumed for the fuel pins. The time required to recover from

TABLE 3.D.I. *Closed Loop Components Failure Rates and Mean-Times-to-Restore*

<u>Component</u>	<u>λ, FAIL./ 10⁶ hr</u>	<u>MTR, hr</u>
Instrumentation 2 out of 3	2	25
Intermediate Heat Exchanger	10	200
Primary Pump	15	500
Check Valve	4	500
Heater	3	500
Strainer	1	200
Air Blast Heat Exchanger	7	100
Blower	3	100
Secondary Pump	15	500
Pump-Check Valve Combina- tion Primary	4.28	500
Pump-Check Valve Combina- tion Secondary	4.28	500
Mechanical Failures	5	800
Piping	5	800

a fuel failure ranges from nine days for some vertical core concepts to 1.5 days for the dispersed core. The failure rates dictated by the availability allocation of 80 hours per year and the time required to recover from a fuel failure are: one fuel failure in the reactor every 2.7 years if the recovery time is nine days, to 2.2 failures per year if the recovery time is 1.5 days.

The required full exposure reliability for one fuel pin (R_p) was calculated for both of the above recovery times for a peak exposure of 80,000 MWd/tonne, 14,224 fuel pins in the core, and an unavailability goal of 80 hr/yr. For a recovery time of nine days, $R_p = 0.999974$. For a recovery time of 1.5 days, $R_p = 0.999844$.

2. Safety Analysis

D. D. Stepnewski

a. NPTF Accident Analysis

A. R. Schade

A fault tree was constructed for the NPTF outlining the basic faults which could lead to a core disassembly. (The expression of a failure path does not necessarily mean that the event is credible.) The fault tree was used to determine what static and dynamic reactivity studies were necessary for the NPTF PSAR. The basic faults have been examined to identify design requirements, protective system requirements, operational procedures and administrative controls that are necessary for fault prevention.

Conclusions reached to date are:

- (1) Control rod motion by an object falling from the polar crane can be precluded by design requirements coupled with administrative controls and operational procedures.
- (2) Because of the low heat generation rates at normal operational conditions, no mechanism can be identified which is capable of generating sufficient energy to expel a control rod by internal force.
- (3) Introduction of a liquid moderator (H_2O or hydrocarbon, etc.) into the core voids and the flooding of one "dry" (sodium absent subassembly would give a reactivity input up to 12\$. Flooding the core voids only

would give ~9\$. The possibility of the accidental insertion of a liquid moderator into the core must be design limited. Quantity of moderating liquid must be limited to safe amounts with all potential moderating materials under administrative control.

- (4) A shutdown criticality accident must be precluded by the guaranteed subcriticality of the reactor.
- (5) Core compaction caused by external falling objects can be precluded by design requirements coupled with administrative controls and operational procedures.
- (6) Fuel melting and core compaction can be avoided since a power/heat imbalance at normal operational or shutdown condition is not possible. The maximum short term fission power of the NPTF is 2 kW. The inherent heat generation of the Pu fuel will be about 3 kW. This gives a maximum power of 5 kW.

An analysis (BNWL-766)⁽¹⁾ has shown that the reactor operating at a power level of 5 kW without cooling will result in a 6 °F temperature increase per hour. This is a fairly conservative estimate, as the calculational model assumed that there was very little radial heat flow. The balance of the heat was assumed to flow out of the ends of the subassemblies. This low temperature rise will allow the core to be unloaded in the event that there would be a complete loss of core cooling before fuel damage occurs.

b. Definition of Protection System

C. L. Fies

The separation of protection instrumentation from control instrumentation is an important safety consideration that involves complex interfacing problems. A clear-cut protection

1. Q. L. Baird and H. C. F. Ripfel. NPTF Heat Transfer Analysis, BNWL-766, April 1968.

system definition at the start of design is conducive to orderly solution of the design problems inherent in this area. The Protection System for the FFTF has been defined to include the functions provided by actuators, such as containment isolation and emergency cooling. The Protection System, therefore, provides the integrating function which ensures that performance requirements (response time, accuracy, etc.) are met for the total system from the sensor which detects the onset of unsafe conditions through the actuators which provide completion of the protection function. A pictorial representation of the functions provided by the protection system is shown in Figure 3.D.1. Some definitions which clarify the terms used in Figure 3.D.1 can be stated as follows:

Protection System: All electrical and mechanical devices involved in providing an irreversible safety shutdown of the reactor and those devices which, in the event of an accident, provide engineered safeguards.

Protection Instrumentation: All electrical and mechanical devices from the sensors to the actuation device input terminals which generate signals to actuate the Safety Shutdown and Engineered Safety Features.

Safety Shutdown (Scram) System: That part of the Protection System which initiates and performs a rapid shutdown of the reactor.

Engineered Safety Features: That part of the Protection System which acts in response to post-accident situations.

FIGURE 3.D.1. FFTF Protection System

IV. DEVELOPMENT

D. L. Condotta

A. FFTF FUELS

W. E. Roake

1. Fuel Fabrication Process Demonstration

C. A. Burgess

FFTF Fuels Development maintains the responsibility of developing and evaluating the capabilities of commercial fuel suppliers to fabricate FFTF fuel cores. The first step in this development is the LMFBR Fuel Supplier Prequalification Program in which potential FTR vendors will be evaluated according to their capability to handle plutonium-bearing fuel material and to fabricate a limited number of finished fuel pins. In addition, a Fuel Fabrication Demonstration Facility capable of fabricating FTR fuel assemblies and demonstrating an acceptable process will be built in support of the FTR.

a. LMFBR Fuel Supplier Prequalification Program

C. A. Strand, J. F. Williams and R. E. Bardsley

Product specifications for the FFTF driver fuel pin were approved and submitted to RDT for review and comment prior to issuing the bid package for the LMFBR Fuel Supplier Prequalification Program to potential bidders.

b. FFTF Fuel Fabrication Demonstration Facility

J. W. Thornton, M. D. Jackson, C. A. Strand,
J. E. Sammis and R. E. Bardsley

The renovation work for the pin portion of the Fuel Fabrication Demonstration Facility neared completion.

Preliminary performance tests of the pellet press were conducted to observe press operating characteristics, using

iron powder as feed material during the early tests. Uranium dioxide pellets produced by the press, and then sintered, showed good reproducibility of dimension and density over the range of parameters tested, which included pressures of 30,000, 40,000, and 50,000 psi and dwell times of one and five seconds.

c. Cladding

R. J. Lobsinger

Approximately 2700 ft of Type 316 SS tubing to be used for developmental fuel pins were examined using ultrasonic and eddy current techniques. Of this total amount, 1500 ft had received special handling during fabrication to reduce the occurrence of manufacturing defects. The results of the nondestructive examination at PNL confirmed this approach. The tubing that received special handling had only a 5% reject rate (with an additional 7% not testable because of high background noise); whereas the tubing with normal handling had a 15% reject rate (with an additional 23% not testable because of high background noise). The fact that the material given extra care during fabrication had significantly fewer defects demonstrates the necessity of specifying such a requirement for high quality tubing.

2. Fuel Process Development and Irradiation Test Fabrication

B. R. Hayward

The objectives of this work are to define optimized mixed oxide fuel fabrication unit operations; develop fabrication data and procedures for selecting a reproducible and reliable fuel fabrication process; develop feasible and reliable unit operations for manufacturing full scale FTR driver fuel pins and subassemblies; define product requirements for FTR driver

fuels, pins, and subassemblies through close interaction with FTR fuel design and analysis studies; and fabricate fuel pins required for irradiation tests and assemblies for mechanical, hydraulic, and sodium flow tests.

a. Mixed Oxide Fuel Process Development

H. G. Powers, D. A. Stranik, and M. J. Barr

Two statistical approaches are being used to investigate the process interrelationships. The first approach is a fractional-factorial experiment used to investigate the main effects of nine key variables and their more important interactions. The nine key variables are: powder morphology (surface activity, particle size, etc.); press behavior (pressure, dwell times, etc., as they affect green density); sintering temperature; sintering time; binder percentage; preslug pressure; binder type; recycle material characteristics; and the percentage of recycle material.

The second approach is a series of small experiments using complete statistical block designs as appropriate. These smaller experiments have been used to evaluate minor process variations, determine practical limits for variables in the fractional-factorial experiment, determine reproducibility of key unit operations, and determine more accurately the nature of process interrelationships.

3. Irradiation Testing

B. R. Hayward

Currently available irradiation test data on austenitic stainless steel-clad mixed oxide fuel does not permit an accurate prediction of the behavior of this fuel system at the fuel burnups and cladding fluence levels anticipated in the initial FTR driver core.

The development of analytical models for describing the synergistic behavior of fuel and cladding during long term irradiation has not progressed to the application stage. Consequently, an irradiation testing program focused on development of the initial FTR driver fuel is a necessary and logical means to providing a reliable fuel design.

- a. Thermal Irradiations of Prototypical Length Fuel Pins
G. E. Culley, R. L. Meyers, E. O. Ballard,
and J. E. Hanson

The objective of this work in the FTR driver fuel development program is to evaluate through steady-state and transient thermal flux irradiations those performance characteristics of prototypical FTR mixed oxide fuel pins that are not strongly dependent upon the flux spectrum. The irradiation tests being conducted are designed to provide data that will contribute to the assurance that safe operation of the fast reactor core using mixed oxide fuel can be achieved.

Capsules PNL-59-5 and 6 were irradiated in the GETR to a goal exposure of 10,000 MWd/tonne (capsule designations and descriptions were given in the previous quarterly report).⁽¹⁾ Neutron radiography (Figures 4.A.1, 4.A.2, and 4.A.3) show the fuel pins to be in apparently satisfactory condition except for the presence of gaps between the fuel pellets.

Table 4.A.I presents the status of the 12 PNL-59 series capsules through GETR cycle 99. In accordance with an AEC directive, the irradiation of capsules PNL-59-11 and 12 were terminated, and the capsules are presently stored in the GETR canal.

The steady-state irradiation data for the PNL-59 capsules were reduced and plots of peak pin power versus time were made. These plots were integrated and the fuel burnup calculated, and the results are shown in Table 4.A.II.

1. Fast Flux Test Facility Quarterly Progress Report, April, May, June 1968, BNWL-880. October 1968.

FIGURE 4.A.1. Neutron Radiographs of Lower Third of Capsules PNL 59-5 and 59-6

FIGURE 4.A.2. Neutron Radiographs of Center Third of Capsules PNL 59-5 and 59-6

FIGURE 4.A.3. Neutron Radiographs of Upper Third of Capsules PNL 59-5 and 59-6

TABLE 4.A.I. *GETR/TREAT Capsule Status (September 16, 1968)*

<u>Capsule</u>	<u>Approximate Burnup, MWd/tonne</u>	<u>Location</u>	<u>Next Action</u>
PNL-59-1	13,000	RML Pool at GE VNC	Transient Irradiation
2	13,000		
3	13,000		
4	13,000		
5	13,000	GETR Canal	Neutron Radiography and Destructive Exam
6	13,000		
7	22,000	GETR Pool (Irradiation)	Continue Irradiation
8	22,000		
9	13,000		
10	13,000		
11	4,000	GETR Canal (Storage)	Undetermined
12	4,000		

TABLE 4.A.II. *GETR Capsule Burnup Levels
(Burnup is axial average in MWd/tonne)*

GETR Cycle No.	Capsule Number, PNL 59-											
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>	<u>10</u>	<u>11</u>	<u>12</u>
94	3,872	3,998	4,200	4,136								
95	4,419	4,439	4,585	4,446	4,257	NA	4,284	4,248				
96	4,460	4,489	4,595	4,416	4,272	NA	4,308	4,352				
97	--	--	--	--	4,640	NA	4,617	4,676	4,592	4,542	NA	NA
Total	12,751	12,926	13,380	12,998	13,169	NA	13,209	13,276	4,592	4,542	NA	NA

NOTE: NA - Not Available

The transient response of the GETR-TREAT capsule with nonirradiated fuel was studied using the ARGUS code. Input data include the power transient and initial capsule temperature. Various combinations of data were used in order to find a combination which would produce the following desired results:

- (1) 70% area fuel melt with cladding ID temperature equal to 1550 °F.
 - (2) 20% melt with cladding ID temperature equal to 1300 °F.
- These results correspond to two levels of damage severity which are to be achieved by the transient testing program.

The best results obtained thus far are shown in Figures 4.A.4 and 4.A.5. The resulting temperature histories are shown superimposed over a plot of the power transient. Figure 4.A.4 shows the fuel center temperature, which reached and surpassed the melting temperature of 4900 °F. Successive nodal points reached and surpassed the melting temperature until and including the 19th node, which reached a maximum temperature of 5102 °F. The maximum temperature of the 20th node was 4816 °F. The radius within which 70% of the fuel cross-sectional area is contained lies between these two nodes. Figure 4.A.5 shows the fuel and cladding temperatures in the case where 20 percent of the fuel cross-sectional area would be molten.

(1) Postirradiation Evaluations. Calculations were made to evaluate the fuel pellet separations which appeared in the GETR test capsule neutron radiographs. (See Figures 4.A.1, 4.A.2, and 4.A.3.)

A model was developed from which total fuel expansion was calculated. This value compared favorably with the total expansion measured from the actual extensometer reading. The model predicts an extensometer movement in the hot condition

FIGURE 4.A.4. GETR-TREAT Transient Analysis ARGUS Code with Transient No. 767 and Melt Area 70%

FIGURE 4.A.5. GETR-TREAT Transient Analysis ARGUS Code with Transient No. 926 and Melt Area 20%

of 0.344 in. (the measured extensometer reading for five capsules averaged 0.356 in.). In developing the model it was assumed that after thermal expansion of the fuel pin the fuel pellets would become attached to the cladding wall and, when cooled to ambient temperature, would become separated by differential thermal expansion. The gap formed between each fuel pellet upon cooling was found to correspond to the fuel centerline temperature profile; i.e., the higher the temperature, the larger the pellet-to-pellet gap. This explains the general observation that the largest gaps lie in the lower one-third of the fuel columns.

b. Thermal Flux Irradiations of Hypostoichiometric
UO₂-PuO₂ Materials

L. A. Pember and R. D. Leggett

The objective of this task is to evaluate the high exposure performance of mechanically blended low-density solid pellet and high-density annular pellet UO₂-PuO₂ fuel as a function of stoichiometry. The O/M ratios being evaluated are 1.94, 1.96 and 2.00 at fuel exposures of 50,000 MWd/tonne and 100,000 MWd/tonne.

Capsule BNW-1-11 (annular pellets, O/M = 2.00) attained its goal burnup of 50,000 MWd/tonne and was discharged and shipped to LASL for postirradiation examination. Neutron radiography showed the fuel pin to be in apparently satisfactory condition.

Capsule BNW-1-4 (solid pellets, O/M = 1.96) was withdrawn at 41,000 MWd/tonne burnup after thermocouple readings became erratic indicating pin failure. Neutron radiography revealed that rupture of the fuel pin cladding had occurred. The capsule has been returned to PNL. Postirradiation examination will be conducted at LASL at a later date.

Capsule BNW-1-9 (annular pellets, O/M = 1.96), which was discharged from the reactor during the last quarter, was shipped to LASL for postirradiation examination. Neutron radiography showed the fuel pin to be in apparently satisfactory condition.

4. Other Work in Progress

Other fuel development work in progress is listed below.

- Fuel pellet characterization
- Subassembly process development
- Quality assurance program for fuel pins.

B. ENGINEERING DEVELOPMENT

J. R. Carrell

1. Sodium Components Development

T. W. Withers

a. Intermediate Heat Exchangers and Heat Dumps

L. A. Whinery

The industrial IHX design studies have been authorized on the basis of utilizing a single design study contractor. Based upon solicited proposal submittals, the Babcock and Wilcox Company has been selected to perform the design study activities.

A proposed proof-testing program for the intermediate heat exchanger and heat dump has been submitted to RDT for review and comment. It is PNL's recommendation that testing be performed at the LMEC Sodium Component Test Installation (SCTI).

b. Sodium Pumps

W. Babcock

Westinghouse has completed a draft report, Extrapolation of State-of-the-Art Sodium Pumps to the FFTF Primary Pump Reference Design Requirements. This report concludes that the Fermi pump concept represents the least departure from existing technology and recommends a Fermi type pump for the FFTF application.

2. Equipment Development

C. A. Munro

a. Control Mechanism Development

J. C. Noakes and P. K. Telford

The control mechanism development plan requires parallel effort on two reference designs and the backup design to ensure

highly reliable control rod drive mechanisms for the FTR. Design studies are being performed on the magnetic jack drive (reference design for the primary safety drive) by Westinghouse.

A request for proposal was prepared for the design study of the Roller Nut drive (reference design for the control rod drive).

The design of the modified Fermi drive (reference backup design for either the primary safety or control rod drive) was reported last quarter. Preparation for testing individual components of the modified Fermi drive has started for the accelerating spring and the latch assembly.

b. Closed Loop Tube and Hardware Development

J. W. Kolb and B. G. Smith

Continued development of the Closed Loop Top Face Hardware Arrangement, Figure 4.B.1, has resulted in a smaller diameter spool piece with one sodium seal backed up with a remote weld joint. This simplified arrangement assumes that a mechanical seal can be used in place of a remote internal weld for change-out of the spool piece. This change-out is necessary only in case of a ruptured or expanded test specimen. As a backup design, the creep-compensating type closure may be used in essentially the same spool piece as shown in Figure 4.B.2, Closed Loop for Concept I. Because of the decrease in size, either of the above methods can be adapted to most reactor concepts presently under consideration for the FTR.

c. Fuel Examination

Neutron radiography is a means of nondestructive examination of irradiated and unirradiated components and assemblies.

General Electric Vallecitos Nuclear Center (under contract to PNL) is making a survey of neutron technology and experience

DETAIL A

CONCEPTUAL 1: 3/8" ISSUE 10-20-68
2: 1/2" ISSUE 10-20-68

SK-3-14034

FIGURE 4.B.1. Closed Loop Top Face Hardware Arrangement

4.15

BNWL-917

SK-3-14031

FIGURE 4.B.2. Closed Loop for Concept I

as it applies to examination of irradiated materials. The survey makes the following points as technical input for a conceptual design of neutron radiography equipment to fit the needs of the FFTF and HFEF fuel examination programs.

- Neutron Source - A small, special-purpose 500-1000 kW reactor is best suited to satisfy the requirement for the neutron source.
- Beam Preparation - The divergent-beam collimator is recommended.
- Imaging - The most satisfactory method used for imaging highly radioactive objects is the transfer process.

For imaging unirradiated fuel and other nonradioactive objects, two systems are recommended. The first system is an image intensifier with a TV readout and provides direct, real-time imaging with fair resolution and contrast. A video tape recorder is recommended to provide a convenient and permanent record. The second system utilizes gadolinium enriched to greater than 80% Gadolinium-157 as an image converter.

- Imaging of Defects in Fast Fuels - Experience in neutron radiography of fast reactor fuels is insufficient to predict accurately the ultimate limitations in detecting fuel defects.
- Cold Neutron Radiography - Only theoretical work has been done to date with the cold neutron radiography of nuclear fuels.

3. Development Facilities

K. G. Toyoda

a. Hydraulic Core Mockup

H. Leigh

The objective of the Hydraulic Core Mockup Program is to provide development and testing support for the resolution of FTR vessel and core hydraulic and mechanical design problems.

RDT technical approval of the Project Proposal, Rev. 1, Hydraulic Core Mockup, and the Design Criteria, subject to incorporation of comments, was received. The Design Criteria has been revised to incorporate AEC comments.

(1) Hydraulic Balance and Tubesheet Leakage. The goal of this program is to evaluate hydraulic balance forces as affected by duct geometry and coolant flow rate, and to determine the duct-tubesheet leakage characteristics of alternative sealing and seating configurations. This testing is complete and the final report, BNWL-883,⁽¹⁾ is being readied for publication.

These tests have demonstrated that the resultant axial force on the FFTF fuel ducts can be controlled by the use of the hydraulic balance concept. The axial force on the fuel duct was found to be proportional to the coolant flow rate through the duct. All forces produced were in a downward direction with the exception of the test sequences where the leakage rate was high enough to cause a back pressure on the leakage collection chambers. The magnitude of the axial forces using a 2.890 in. diameter tube ranged from a low of 5 lb (4.3 lb with sodium at 750 °F) at a flow rate of 100 gal/min to a high of 47 lb (40 lb with sodium at 750 °F) at 350 gal/min flow with corresponding diametral clearances of 0.005 in. and 0.017 in. The forces expected on a vertically oriented end entry tube would be about 500 lb in the upward direction at 75 psi tubesheet differential pressure.

Tests were conducted over a range of duct outlet flow areas from 7.95 in.² to 43.1 in.². These tests demonstrated that increasing the flow area had only a small effect upon

1. P. M. Jackson. Hydraulic Balance Concept and Tubesheet Leakage, FFTF Hydraulic Feature Model.

the axial forces, provided that the shape of the discharge ports was not altered. When the outlet configuration was changed from round holes to vertical slots the downward axial forces were increased about 50%.

The tubesheet leakage rates were found to vary directly with increased annular clearance and as the square of the differential pressure across the tubesheet.

(2) Annular Inlet Feature Model. The purpose of the one-tenth scale model of the annular inlet concept is to assist in evaluating the mixing characteristics of the inlet streams as they flow within the annular zone formed between the main reactor vessel and the reactor core vessel. Previous testing with this model yielded information in vortex formation characteristics. Modification to the model by the addition of flow straighteners and inlet diffusers did not result in any noticeable change to the vortices which occur in the lower vessel region just outside the fuel duct inlet chamber. Additional tests were made with the deflectors tipped at 15 deg and 30 deg from the vertical. The 30 deg tests provided about a 360 deg rotation of the coolant stream, in the annular region, before it entered the lower plenum zone, with no vortices formed. The angular sweep of the coolant produced a higher differential pressure between the inlet lines and the plenum zone than had been observed in any previous test. This testing is complete unless the annular inlet is selected as the FFTF vessel concept.

(3) Inlet Plenum Feature Model. This is a one-third scale hydraulic model of the FFTF piped inlet arrangement. The purpose of this model is to provide information which will aid in the evaluation and refinement of the inlet configurations for the reactor vessel system.

Installation of the model was completed, duct flow devices calibrated, and testing performed in accordance with quality assurance acceptance test procedures for the feature model loop, pump and motor. Initial operation of the model is underway.

(4) HCM Instrumentation. Based on recent information from Atomics International, an improved sensor excitation and signal processing circuit has been designed which adds appreciable amplification (about 1000 times) to the sensor output, and tailors the system frequency response to those particular signal frequencies (5 to 30 Hz) that AI reports to be the most significant.

b. Sodium Core Development Facilities

K. D. Hayden

The objective of the sodium development facilities program is to establish the technical basis and requirements of sodium core development facilities responsive to the long- and short-range needs of the FFTF program, and to provide the necessary technical support for acquisition of facilities.

(1) FFTF Sodium Facilities Building, Project BAP-027. This 100 ft by 60 ft by 24 ft high building will be used to consolidate the location of several existing sodium loops and small sodium systems and will provide space for the location of new sodium facilities. The directive authorization to proceed with the design and construction of the project was issued on July 11, 1968, by the AEC Richland Operations Office. A revised design criteria was approved and issued on September 5, 1968, to relocate the offices outside the high bay area and to reduce the high bay area from 70 to 60 ft in width. The plot plan and low voltage switchgear procurement specifications have been completed and the

remainder of the building design by Hanford Engineering Services is proceeding satisfactorily with 50% of the design complete. Design completion is scheduled for November 26, 1968, and construction completion schedule for September 15, 1969.

c. Sodium Studies Facility Operation

J. D. Berg and L. J. Defferding

The Sodium Studies Facility provides a centralized laboratory for small test loops and sodium chemistry equipment. Current status is given in Table 4.B.I.

TABLE 4.B.I. *Sodium Studies Facilities Status*

<u>Loop or Facility</u>	<u>Temp, °F</u>	<u>Operating Hours</u>	
		<u>This Period</u>	<u>Total</u>
Small Component Evaluation	1500	2176	8728
Small Heat Transfer	1500	216	356
Na Purification	700	2184	5784
Radioisotope Transport	1400	754	5054
Fission Product Control	Being Installed		
Static Pots (4)	1200	912 (3)	1852

4. Other Work in Progress

The following other items of Equipment Development work are in progress.

- Single Duct Fuel Handling Demonstration Mockup.
- Design of Fast Reactor Thermal Engineering Facility.
- Performance of Engineering Testing Services.

C. COOLANT AND PROCESS DEVELOPMENT

B. M. Johnson

1. Corrosion and Chemistry

D. W. Shannon

a. Radioisotope Transport and Deposition

W. F. Brehm and E. A. Kovacevich

The purpose of this task is to define the effect of reactor operating parameters (temperature, flow rate, temperature and flow distributions, oxygen level in sodium, purification flow fraction) on the transport and deposition of ^{51}Cr , ^{58}Co , ^{60}Co , and ^{54}Mn . To provide a test means of observing the migration of radioisotopes released from irradiated source tabs in loops containing flowing sodium, the Radioisotope Transport Loop (RTL) and the Thermal Connection Loops (TCL) are used.

(1) RTL Loop Tests. Current testing was performed using a Type 304 SS specimen of 3/8-in. tubing that had been irradiated in the EBR-II as follows:

Exposed to reactor sodium on OD only for about 100 days
at 750 °F

No significant stress on tubing

Flux level 1.6×10^{15} n/cm²-sec, total fluence

1.4×10^{22} n/cm² (>0.18 MeV)

Removed from reactor December 1966.

The Radioisotope Transport Loop was operated for 351 hours under the following conditions with the specimen installed:

Specimen temperature, °F	1300
System Thermal Gradient, °F	400
Primary Sodium Flow, gal/min	1.7
Oxygen Content, ppm	<10

Following the scheduled shutdown, the loop piping was counted with a NaI crystal with a 1-in. collimator. Upon completion of the gamma scanning, the loop was prepared for another run in which the irradiated test specimen was removed. The loop was operated for two runs of 416 hours and 384 hours under the following conditions:

Specimen temperature, °F	1300
Systems Thermal Gradient, °F	400
Primary Sodium Flow, gal/min	1.7
Oxygen Content, ppm	<10

The purpose of this series of tests was to measure the rate of migration of the radioactive deposits previously deposited by mass transport from the irradiated specimen. The last run was terminated after 384 hours of operation when a sodium leak developed in the primary air-to-sodium heat exchanger. The loop was again gamma scanned to measure any further migration of the ^{54}Mn originally introduced from the EBR-II specimen. The ^{54}Mn in the loop showed a steady decrease throughout most of the primary system, particularly in the hot leg immediately downstream from the specimen. The activity increased from run to run in the cold trap. The large decrease in hot-leg activity is to be expected since it is the region of largest gross material removal. The peak in activity seen 32 inches downstream in the data taken after 800 hours operation without source probably represents deposition of part of the material released farther upstream near the specimen. This shows that part of the ^{54}Mn moves by isotopic exchange.

(2) Thermal Convection Loop Tests. Three TCL loops were set up and tests were run with operating parameters as shown in Table 4.C.I. Loops 3 and 4 were run to see whether specimen location affected deposition patterns. Loop 5 was

TABLE 4.C.I. Thermal Convection Loop Operating Parameters

Loop Number	Test Time	T Max	T Min	Velocity (est.)	Specimen Type and Location ^(a)
TCL 3	912 hr	1230 °F	1030 °F	0.4 ft/sec	24h K-irradiation and EBR-II stock at hot spot
TCL 4	912 hr	1300 °F	1180 °F	0.3 ft/sec	Same as loop (3) located 7 in. upstream from hot spot, specimen temp 1225 °F
TCL 5	750 hr	1400 °F	1000 °F	0.6 ft/sec	EBR-II stock, subsequently irradiated in C reactor 10h, at hot spot

a. EBR-II stock contains ^{54}Mn and small amounts of ^{58}Co . C and K irradiations produce ^{51}Cr , ^{59}Fe , and ^{60}Co . The Cr and Co levels were very low in these specimens. ^{59}Fe has never been found in more than trace levels in any loops.

run at a different temperature. Loop 3 was cut up and gamma counted using a Ge(Li) detector. Loops 4 and 5 were counted in place with the portable NaI detector.

The data showed ^{54}Mn only on the loop walls and in loop sodium. No ^{58}Co , ^{60}Co , or ^{51}Cr was found even in the diode count. Loop 3 showed the highest ^{54}Mn deposition at the minimum temperature location, but the activity was rather evenly distributed with only 40% variation between the maximum activity and the minimum near the specimen. Loop 4 had the greatest deposition, about 16 in. downstream from the sample and 7 in. downstream from the maximum temperature location. Again the activity was generally distributed with maximum deposition rate only 40% higher than the minimum. Loop 5, with a large ΔT , had maximum ^{54}Mn deposition again at the minimum temperature spot, but the counting rate there was a factor of five higher than at a spot near the specimen. The results of these runs show that in the absence of large temperature gradients in a loop, isotopic exchange of ^{54}Mn is

still dominant, and specimen location is important. With an increase in maximum temperature and total loop ΔT , the transport rate of ^{54}Mn increases.

(3) Significance of Test Results. These results confirm the previous data indicating that Cr and Co migrate slowly in loops, but that Mn spreads out quickly. The isotopic exchange is present and detectable with all three elements (as expected) but the Mn migration is much more affected by temperature and chemical parameters than the Cr and Co migration. The tendency for manganese to be deposited in the lower temperature regions suggests a classical solution and precipitation phenomenon in the TC loops. However, the RTL observations of steady decrease in concentration at most points of the loop except the cold trap suggests that the presence of oxygen affects the deposition tendency of ^{54}Mn .

Known amounts of material were removed from Loop 3 by electropolishing, and the solutions were counted for ^{54}Mn . Significant ^{54}Mn penetration was observed 0.2 mil deep into steel exposed for 912 hr at 1100 °F, indicating that some irreversible deposition of radioisotopes on pipe walls will occur.

b. Sodium Cover Gas Interactions

H. P. Maffei

Experimental studies to establish allowable impurity levels for FFTF cover gases were conducted on the reactions of cover gases and their impurities with molten sodium (by determining the equilibrium values and kinetic reaction rates between the cover gas impurities and molten sodium). The reaction of methane in argon with molten sodium was measured at 800 °F, 1000 °F, 1200 °F, and 1400 °F. All of the test capsules contained 100 cm³ of sodium at the particular test

temperatures. The sodium surface for each test was 2.47 in.². Two liters (760 mm pressure) of cover gas containing nominally 100 ppm of methane was recirculated over the sodium and the resultant decrease in methane concentration was measured by a gas chromatograph employing a flame ionization detector.

It was found that the reaction of methane in argon cover gas with sodium proceeds very similarly to the previously reported reactions of methane with sodium in helium cover gas. That is, the reactions follow the second order kinetic rate law which may be expressed as:

$$\frac{dx}{dt} = -k' x^2$$

where x is equal to the methane concentration at time t . If this equation is then integrated from zero time when $x = a$, the original methane concentration, to any time t , the following expression is derived:

$$\frac{a - x}{x} = a k' t$$

Straight line plots of $\frac{a - x}{ax}$ versus t verify that the reaction of methane in argon with sodium is a second order reaction similar to the reaction in helium cover gas. The measured rate constants for the four test temperatures for both argon and helium are:

Temperature	k'	
	Argon	Helium
800 °F	5.38×10^{-4}	3.45×10^{-4}
1000 °F	1.43×10^{-3}	4.02×10^{-3}
1200 °F	4.04×10^{-3}	8.53×10^{-3}
1400 °F	6.61×10^{-2}	5.2×10^{-2}

These results indicate an activation energy of 19.5 kcal/g mole as determined from a plot of the variation of reaction rates with temperature as shown in Figure 4.C.1.

It was demonstrated last quarter that the sodium-methane reaction in helium is controlled by the amount of sodium vapor present in the gas plenum (which is a function of the sodium pool temperature). Because of the similarity of the reaction rates shown above for both helium and argon, it was concluded that the same controlling variable holds for both cover gases.

(1) Detection of Oil Leaks into Sodium. Analyzing sodium cover gas for increases in the hydrogen and methane concentration has been found to be a rapid and sensitive method of detecting oil leakage into sodium.

Oil in the amount of 3.8 mg was introduced into a capsule containing 100 g of sodium (38 ppm oil) at 900 °F. Argon cover gas was purged through the capsule and analyzed by gas chromatography. Upon contact with the molten sodium, the oil was reduced to methane and hydrogen. After the introduction of the oil, the methane concentration increased from 25 ppm to 325 ppm. The hydrogen concentration also increased but was not quantitatively measured.

c. Nitrogen Cover Gas Studies

FFTF plans to use nitrogen cover gas in the equipment cells. At the present time it is not known if the nitrogen gas will nitride the stainless steel components in these cells. It is known that nitrogen will nitride stainless steel in the presence of a reducing atmosphere. Conversely, it has been reported that small quantities of oxygen or water vapor inhibit the nitriding of stainless steel. An investigation is presently being conducted to investigate

FIGURE 4.C.1. Variation of Reaction Rates with Temperature

further the inhibiting effect of oxygen on stainless steel nitriding leading to specifications of the oxygen content of FFTF equipment cell nitrogen cover gas.

Stainless steel samples (304 and 316) were exposed to nitrogen gas containing 500 ppm and 5000 ppm oxygen respectively for exposure of one month and three months at temperatures of 1000 °F and 1400 °F. Following the two exposures, duplicate samples were removed from the test for both chemical analyses and metallurgical examination. Although the metallurgical examination did not reveal any nitride formation, the chemical analyses showed an increase in nitrogen content for both the 304 and 316 SS specimens exposed to the 1400 °F, 500 ppm O₂ gas. Nitriding was successfully inhibited at 1000 °F by the 500 ppm oxygen concentration, and at 1400 °F by the 5000 ppm oxygen concentration, as shown below:

<u>Exposure Time</u>	<u>Oxygen Concentration</u>	<u>Change in N₂ Concentration of Stainless Steel Samples at Temperature of</u>	
		<u>1000 °F</u>	<u>1400 °F</u>
1 month	500 ppm	nil	160 ppm
	5000 ppm	nil	nil
3 months	500 ppm	nil	255 ppm
	5000 ppm	nil	nil

There was no excessive corrosion of the stainless steel coupons by the oxygen. The maximum corrosion rates measured averaged less than 0.1 mils/year for the samples exposed at 1400 °F.

2. Thermal Hydraulics

J. M. Yatabe

a. Heat Transfer Characteristics of FFTF Fuel Assemblies

V. R. Hill

The major purpose of this laboratory program is to determine the heat transfer characteristics of FFTF fuel assemblies

to provide data in support of the driver fuel design and the FTR safeguards analyses. Electrically heated and instrumented models of these fuel assemblies of different maximum heat flux and temperature capabilities will be required for various phases of the program.

Current program goals have been to develop heated pins for 7-pin bundle local cladding temperature tests. For these tests the heated pins must operate at heat fluxes of 500,000 Btu/hr-ft² at 800 °F sodium temperature and be instrumented to measure sheath temperature. Effort has been started to develop pins with Nichrome V central cores.

The first of these pins, Pin 650-65, was operated in the Small Heat Transfer Loop and successfully attained a heat flux of 500,000 Btu/hr-ft² at a sodium temperature of 800 °F. These conditions were sustained for 3-1/2 hours, at which time electric power surges from a faulty control device (subsequently replaced) increased the heat to approximately 150% of the original value of 30 kW and caused a failure in an electron beam seal weld (between pin sheath and end cap). The sodium leak caused immediate cessation of the test. Post-inspection showed that the pin itself had not failed and had the same core electrical resistance (at room temperature) as it had originally.

Electron bombardment heaters are being considered as a back-up concept for the indirect resistance heaters for the FFTF thermal hydraulic tests requiring combined high heat flux and high surface temperature operation (e.g., heat fluxes of 2,000,000 Btu/hr-ft² at 1500 °F surface temperatures). As a first step in this program, PNL has contracted EIMAC, a division of Varian Associates, to perform a study on the feasibility of constructing the high heat flux electron bombardment

heaters, modeling full sized FFTF fuel pins, instrumenting the pins to provide quantitative heat transfer results, and assembling the pins into sector models of FFTF fuel assemblies.

b. Gas Cooling of FFTF Fuel Assemblies During Inspection

D. R. Dickinson and T. C. Reihman

The objective of this program is to study the decay heat removal from the FFTF fuel assemblies by the forced circulation of an inert gas stream during fuel inspection, interim inspection, and disassembly. The cooling is performed by axial flow, transverse flow, or both. The experimental study will provide information to define the system flow requirements.

3. Instrumentation Development Section

M. R. Wood

a. Low Level Neutron Flux Instrumentation

C. N. Jackson and N. C. Hoitink

The objective of this task is to test and evaluate nuclear components of a counting system in an adverse gamma-to-neutron flux environment for the detection of delayed neutrons for failed fuel monitoring and for the reactor startup instrumentation channels. Measurements were made of high level gamma effects on the neutron sensitivity of both ^{10}B lined proportional counters and fission chambers. Under specific system configurations, both types provided usable short-term neutron sensitivity under high gamma background conditions. The current-sensitive preamplifier, which has a faster pulse-shaping capability, improved the gamma tolerance of both proportional counters and fission chambers. Either faster chamber collection or faster pulse-shaping by the electronic components, or both, improved the gamma tolerance of the system.

For both the fission chamber and the proportional counter, curves G and H of Figure 4.C.2 indicate that the best gamma tolerance (as a function of preamplifier type) involved the use of the current-sensitive preamplifier.

In proportional counters, increasing collection volumes and lengthening collection times significantly degrade gamma tolerance, as shown by curves A, C and D of Figure 4.C.2. The tolerance exhibited by a cluster of nine 0.25-in. diam units is portrayed by curve B of Figure 4.C.2. Unfortunately, paralleling for increased neutron sensitivity caused great degradation in gamma tolerance, as evidenced by direct comparison between curves B and D.

The test facility shown in Figure 4.C.3 provided a convenient gamma source for testing candidate neutron detectors for FFTF to gamma fluxes of 10^6 R/hr.

b. In-Reactor Coolant Temperature Sensors

D. C. Thompson, N. C. Hoitink, and J. L. Jackson

The objective of this task is the development and testing of thermocouples to provide reliable, commercially available sensors for measuring the coolant temperatures in the FTR thermal and nuclear environment. Completion of postirradiation testing of thermocouples contained in EBR-II Capsules BT-1 and BT-2 provided useful data for comparison of various types and also suggested possible improvements in calibration procedures. As expected, the tungsten thermocouples (W/W-26% Re and W-5% Re/W-25% Re) drifted excessively (see Figure 4.C.4) and will be unacceptable for use as FTR coolant thermocouples. Also of significance, none of the Chromel/Alumel sensors failed during the testing program, while a number of the other types did fail.

FIGURE 4.C.2. Gamma Effects on Relative Sensitivity of Low Level Neutron Detectors (Preliminary Data)

Neg 0683564

FIGURE 4.C.3. Geometry of Gamma Test Facility

FIGURE 4.C.4. Tungsten Thermocouple Deviations Capsule BT-1 and BT-2

It appears that insufficient data prevents legitimate predictions relating to calibration shifts of the Geminol thermocouples. Exposures of 2.4×10^{21} nvt caused a positive calibration shift up to 20 °F, being especially predominant at temperatures below 800 °F. At increased exposures up to 3.4×10^{21} nvt, the positive drift at lower temperatures reduces and a negative shift of up to 25 °F occurs at higher temperatures.

Tungsten thermocouple deviations (up to 15%) clearly indicate the inability of this type to withstand the environment. Wire resistance increases of 10% for the W/W-26% Re unit and 120% for the W-5% Re/W-25% Re unit were recorded as further evidence of changes in characteristics.

Irradiation of the capsules occurred in three different flux regions designated as Levels 1 through 3. The information in Table 4.C.II shows neutron exposures at the thermocouple junction and the maximum, minimum, and average exposures for sensors in each of the three regions.

TABLE 4.C.II. Neutron Exposure in nvt $\times 10^{21}$ ($E > 0.18$ MeV) for Thermocouples in Each Designated Flux Region

<u>Thermocouple Level</u>	<u>Bead</u>	<u>Maximum</u>	<u>Minimum</u>	<u>Average</u>
1	2.37	2.37	0.85	1.59
2	3.40	3.40	2.50	3.00
3	3.42	3.45	2.80	3.29

The corresponding data on Chromel/Alumel thermocouples were reported in the FFTF Quarterly Technical Report for April, May, June 1968, BNWL-880.

c. Reactor Environmental Effects on Signal Cables
G. J. Dau and J. L. Stringer

The prediction of the magnitude and polarity of the induced current and the measurement of the bulk conductivity

of a commercial insulator in its applied form are two of the goals in this task. In the area of induced current, the initial effort has been to develop a parametric model that gives a qualitative understanding of mechanisms influencing the generation of current. For determination of the insulator bulk conductivity in a radiation intensity, it was found that a simple extension of the current measuring technique would serve. This measuring technique has since been used to measure the combined radiation dose rate and temperature dependence for the bulk conductivity of MgO. The radiation dose rate was 9×10^{10} R/hr, and the maximum temperature attained was ~ 1200 °F for the measurement.

These values of induced current and insulator conductivity have made it possible (through a linear approximation) to evaluate the output error from a simple dc sensor circuit with the connecting cable in a high temperature and intense radiation environment. The error in the sensor output voltage at the meter can be expressed in terms of the sensor output voltage (V), the sensor output resistance (R_s), and the radiation induced current (I) as

$$\% \text{ error} = \left| \frac{IR_s}{V} \right| \times 100$$

In the derivation of the above equation, the dc sensor was connected to the meter through a 0.250-in. OD coaxial cable with a 0.012-in. thick stainless steel sheath, a 0.025-in. diam stainless steel center conductor and MgO insulation. Three feet of the cable was in a radiation dose rate of 5×10^9 R/hr, and 25 feet in a dose rate of 10^6 R/hr. The temperature of the total cable length was 1200 °F. Experimental exposure of cables of this size and composition have shown that a reasonable induced current value is $-1 \mu\text{A}$.

4. Other Work in Progress

The following other activities in coolant and process development are in progress.

- Selection and Testing of Cover Gas Instruments and Sampling Systems.
- Hydraulic Testing of FFTF Fuel Assembly and Components.

D. METALS AND MATERIALS DEVELOPMENT

J. C. Tobin

1. Materials Properties

J. E. Irvin

a. Recommendation of Material for FTR Vessel

R. A. Moen

Type 304 SS was recommended as the material of construction for the FTR pressure vessel.⁽¹⁾ The recommendation was based primarily on the following outstanding attributes of the material:

- Long-term stability
- Ease of fabrication
- Relative low cost.

The basic ground rules for this selection process were that the vessel be designed for maximum steady-state operating conditions of 1200 °F bulk outlet sodium coolant temperature and 900 °F bulk inlet sodium temperature. The vessel is to have a design life of 20 years at 75% operating efficiency. Vessel fluences received during the 20-year plant life were considered concept dependent. In most cases, fluence limits of about 10^{19} n/cm² ($E > 0.1$ MeV) were considered reasonable from the standpoint of design.

b. Tensile Testing

R. E. Robbins

The objectives of this program are to determine the combined effects of fast reactor irradiation and environment on the tensile properties of candidate stainless steel fast

1. Moen, R. A. and R. N. Johnson. Selection of Material for the FTR Pressure Vessel, BNWL-864. Pacific Northwest Laboratory, Richland, Washington, October 1968.

reactor vessel and core structural alloys and to evaluate the metallurgical stability of these materials after they were aged at elevated temperatures.

Tensile tests were conducted on AISI Type 316 SS specimens irradiated in the EBR-II at about 900 °F to a total fluence of 0.8×10^{22} n/cm². Increased yield strength resulting from irradiation was observed at all test temperatures (RT to 1400 °F).

c. Uniaxial Creep

A. J. Lovell and G. A. Dunigan

The objective of this program is to evaluate the effects of fast neutron irradiation and environment on the uniaxial creep-rupture properties of candidate vessel and core structural materials for the FTR. The information obtained will be used to establish the usable materials limits.

Data obtained on unirradiated Type 316 SS from 1100 and 1200 °F tests agree well with nationally published data.⁽¹⁾

The testing of Type 304 SS specimens irradiated in space-available pins in the EBR-II and in the hot water loop of the ETR, is continuing. Deformation (taken from pull-rod motion) versus time data for the 1000 °F tests indicate that irradiation in the EBR-II (to an estimated total fluence of 4×10^{21} n/cm²) has little effect on the postirradiation creep rate. In comparison, specimens that were irradiated to a fast fluence of $\sim 8 \times 10^{20}$ n/cm² (E > 1 MeV) at 540 °F in the ETR, then tested at 1000 °F, show reduced creep rates compared to as-fabricated controls. The stress versus rupture life data for Type 304 SS at 1000 °F show a decrease in rupture

1. Simmons, W. F. and Van Echo, J. A. *The Elevated Temperature Properties of Stainless Steels*, ASTM DS5-Si (1965).

strength for both irradiation conditions. The deformation versus time data indicate that irradiation at 900 °F in the EBR-II decreased the creep rate for postirradiation tests at 1250, 1300 and 1350 °F but possibly increased the creep rate at 1400 °F. An Arrhenius plot of the creep rate data is shown in Figure 4.D.1. The few data points available indicate an increased activation energy for the irradiated specimens over the as-fabricated controls.

d. Damage Analysis

J. J. Holmes and H. R. Brager

A study of the effects of fast reactor irradiations on volume changes of austenitic stainless steels is being made. Preliminary results suggest that this phenomenon will call for design allowances to accommodate the swelling.

A 304L stainless steel EBR-II control rod thimble (5C3) and a fuels irradiation capsule (X-011) were irradiated at temperatures ranging from 704 °F to 866 °F. The peak fast fluence for the thimble was 4.8×10^{22} n/cm² (E > 0.1 MeV) and that for the capsule was 2.3×10^{22} n/cm² (E > 0.1 MeV).

Volume change data for the capsule and thimble core are presented in Figure 4.D.2 together with all other available swelling data. Extrapolation of the raw data to target FFTF fuel cladding fluences of 1.3×10^{23} n/cm² indicates about a 20% volume increase. Until the mechanism of the swelling is better understood, the above extrapolation should be considered as tentative.

Neg 0683374-5

FIGURE 4.D.1. Arrhenius Plot of As-Fabricated Controls and Postirradiation Creep Tests on Type 304 Stainless Steel

FIGURE 4.D.2. Summary of Available Data on Fast Reactor-Induced Swelling

e. Transmission Electron Microscopy of Irradiated Austenitic Stainless Steels

H. R. Brager

Electron micrographs of Type 304L SS samples⁽¹⁾ cut from an EBR-II control rod thimble were analyzed. Cavity distributions of Type 304L SS irradiated to 5×10^{22} n/cm² at 790 °F (421 °C) were determined for foils of varying thicknesses, the determinations being dependent on the thickness of foil examined. The cavity distribution determined for foil thicknesses ranging from 250 to 1200 Å is indicated in Figure 4.D.3. The mean diameter of the cavity distribution tended toward larger values for thicker foils. These results appear to be consistent in the sense that, with the transmitted beam intensity being a function of the foil thickness, the very small cavities (~100 Å) would be difficult to observe in thick foils (≥1000 Å), and the measured distribution would be biased toward larger diameter cavities. On the other hand, the cavity distribution measured in thin foils (~200-300 Å) containing cavities whose size approaches the thickness of the foil would, as observed, be expected to be biased toward the smaller sizes.

f. In-Reactor Measurements

E. R. Gilbert and N. E. Harding

The in-reactor measurements program objectives are to determine the effects of irradiation on the mechanical properties of reactor structural materials, to establish property data suitable for design purposes, and to consider mechanisms by which irradiation affects macroscopic deformation.

1. 04 Program Quarterly Report, April, May, June, 1968, BNWL-846, p. 12.4. Pacific Northwest Laboratory, Richland, Washington.

Neg 0682654

FIGURE 4.D.3. As-Irradiated Cavity Distribution of Type 304L Stainless Steel Irradiated in the EBR-II at 790 °F (421 °C) to a Fast Fluence of 4.8×10^{22} n/cm² ($E > 0.1$ MeV) Measured in Foils of Thickness from 250 to 1200 Å

A creep stress versus temperature-compensated rupture-life (theta parameter) plot (Figure 4.D.4) was made for in-reactor, postirradiation, and unirradiated control creep tests using data by Bloom.⁽¹⁾ The conclusion that can be drawn from this plot is that the rupture life of Type 304 SS tested in-reactor during neutron irradiation or tested post-irradiation is essentially the same for comparable conditions of stress and temperature. The effect of neutron irradiation, as reflected in both in-reactor and postirradiation tests, is to reduce the rupture life or the rupture strength as compared to the unirradiated control values.

g. Chemical Compatibility of Shield and Control Rod Materials

L. R. Bunnell

Compatibility between B_4C and sodium and the transport of boron and carbon through sodium to stainless steel are being examined simultaneously in capsules designed for both purposes. The capsules contain a perforated B_4C carrier tube of stainless steel, physically separated from the outer stainless steel tubes by spacers. The carrier tube is filled with -10 +20 mesh B_4C . After being filled with sodium and welded shut, the capsules are heated to specified temperatures for given times, opened, and subjected to postmortem examination.

Two capsules, heated to 325 and 500 °C for 1000 hours, and B_4C from a 100-hour 500 °C capsule were examined. One of the most obvious results was the cracking of the B_4C particles. Figures 4.D.5 and 4.D.6 show the microstructure of two B_4C particles before reaction. Graphite is almost always present

1. E. E. Bloom. In-Reactor and Postirradiation Creep-Rupture Properties of Type 304 Stainless Steel at 650 °C, ORNL-TM-2130. Oak Ridge National Laboratory, (1968).

Neg 0682254-6

FIGURE 4.D.4. Creep Stress Versus Temperature-Compensated Rupture Life (Theta Parameter) for Type 304 Stainless Steel

FIGURE 4.D.5. Micrograph Showing As-Received B_4C Particles (100X). ["A" Is a Large Resin-Filled Pore, the Arrow Indicates a Large Graphite Flake.]

FIGURE 4.D.6. Micrograph Showing Molten Variation in As-Received B_4C Particles (100X). [The Dark Parallel Lines Are Caused by Graphite Precipitated from the Melt.]

as a minor phase, but its distribution varies widely. Figure 4.D.7 shows one of the most badly cracked particles. This particle had been in sodium 100 hours at 500 °C and its original microstructure was probably somewhat like Figure 4.D.5. Figure 4.D.8 shows a particle cracked by 500 °C-1000 hr sodium exposure. The 325 °C-1000 hr sample looked much the same. Although the sodium penetration in Figure 4.D.8 appears to be along the parallel graphite flakes, the mode of attack in the material containing less graphite is not so clear.

2. Fabrication Development

R. N. Johnson

Fuel Duct Evaluation

E. B. Schwenk

The effect of simulated reactor loading and temperature conditions on the creep behavior of Type 304 SS hexagonal fuel duct specimens is being determined experimentally because the complex geometry precludes a simple analytical analysis of creep behavior based on data from uniaxial specimens. Testing is being conducted under static and dynamic loading at representative FTR operating conditions. Changes in both the average corner-to-corner (ΔC) and average flat-to-flat dimension (ΔF) with exposure times are shown in Figure 4.D.9. After about 2500 hours exposure, the test temperature for the constant-pressure, constant-temperature test (static specimen) was increased to 1050 °F from a level of 1000 °F. The flat-to-flat dimension continued to increase in an apparent nonuniform manner to a value about 0.18% above its initial or pretest level after a total exposure of 4300 hours. The average corner-to-corner dimension continued to decrease slowly following an apparent maximum at 1500 hours exposure, until it about equaled its pretest value. The dynamic creep test specimen dimensions, on the other hand, showed no significant change throughout exposure at both 1000 °F and 1050 °F.

FIGURE 4.D.7. Micrograph Showing Cracks in B_4C Particles
Exposed to 500 °C Sodium for 100 hours (100X)

FIGURE 4.D.8. Micrograph Showing Cracks in B_4C Particles
Exposed to 500 °C Sodium 1000 hours (100X).
[Note Apparent Delamination near Graphite Flake
(arrow). "A" Is a Pull-Out.]

FIGURE 4.D.9. Hexagonal Tube Creep Change in Tube Dimensions with Time

3. Component Material Surveillance

J. W. Helm

The objectives of tasks in this section are threefold:

- (a) Surveillance and in-service inspection of critical items which have a direct influence on the reliable and safe operation of the FTR and which must be analyzed in depth for inclusion in the Safeguards Analyses Reports.
- (b) Surveillance and in-service inspection of critical items for early detection of abnormal or unexpected behavior.
- (c) Obtainment of information on materials and components to verify expected behavior of critical components in the primary system.

a. FFTF Dosimetry

J. L. Jackson

A PNL-ANL cooperative dosimetry experiment has been prepared and is scheduled for irradiation in the EBR-II reactor at the end of Cycle 31. The primary purposes of the experiment are to resolve the existing differences between calculated and measured fluxes and to provide accurate flux and fluence measurements in the EBR-II.

b. Calculation of Equivalent Irradiation Damage Fluences in the FTR and Other Test Facilities

G. E. Russcher

Most materials irradiations are conducted in reactor core test facilities, and data from these irradiations are used to forecast the damage to material in other types of reactors and in out-of-core locations such as the grid plate and pressure vessel. Application and interpretation of these data require an analysis of the damage effectiveness of the spectra in the reactor irradiation facilities and in the areas where the material will be used.

An initial analysis estimated the equivalent irradiation damage fluence in several environments of interest to fast reactor development and estimated comparative damage production in three critical components of the FTR core concepts. Damage effectiveness was evaluated using two criteria; one was based on gross defect production, and the other was based on the generation of clusters of 10 or more vacancies.

Irradiation damage effectiveness was calculated for three test facilities that have produced irradiation data of value to FFTF. These facilities are the EBR-II core - Row 2, the DFR core - Channel 9, and the ETR test loop G-7. Calculations were made for stainless steel in the following three locations of the FTR having quite different neutron spectra: the fuel cladding in the core center, the grid plate or tube sheet below the core, and the inner surface of the reactor vessel at midplane.

Average PKA energy $\bar{E}_{\text{PKA}}$ differs by roughly a factor of two among the various irradiation facilities. Such a difference is indicative of substantial neutron spectrum variations. An even larger change occurs away from the central core area since the average PKA energy drops by a factor of 10 between the FTR core and the FTR grid plate and reactor vessel areas.

The damage calculated for each of the irradiation environments (damage per unit volume of the specimen and per unit fluence) was used to develop an equivalent irradiation damage fluence, K_i . This is defined as the fluence in an environment which should produce the same damage as a unit fluence in EBR-II. Table 4.D.I shows the irradiation damage effectiveness of the various facilities and reactor locations. For example, based on the clustered vacancy model, the FTR pressure

vessel would incur the same damage at $1.5 \times 10^{20} \phi$ (total fluence) as the EBR-II would produce in a specimen irradiated in Row 2 to $1 \times 10^{18} \phi$. The fluences quoted are for neutrons of all energy.

TABLE 4.D.I. *Equivalent Irradiation Damage Fluence
(Normalized to the EBR-II Core - Row 2)*

	Primary Knock-On Atom Energy, avg	Based on a Clustered 10 Vacancy Model Equivalent Fluence K_i	Based on a Gross Displacement Model Equivalent Fluence K_i
	$\bar{E}_{PKA}(\text{keV})$	K_i^{total}	K_i^{total}
EBR-II Core	22	1.0	1.0
DFR Core	18	1.1	1.1
ETR - G7 Test Loop	8.1	9.6	9.3
FTR Cladding in Core Center	10	1.6	1.6
FTR Grid Plate	1.3	58	35
FTR Reactor Pressure Vessel at Midplane	0.84	150	81

NOTE: ϕ = Fluence (neutron/cm²)

D = Damage (defects/cm³ stainless steel)

$$K_i = \frac{[D/\phi]_{\text{EBR}}}{[D/\phi]_i} = \left[\frac{\phi_i}{\phi_{\text{EBR}}} \right]_D$$

The equivalent irradiation damage fluences K_i of Table 4.D.I suggest how data generated in test facilities can be applied to the design or evaluation of reactor materials at several locations in the FTR. If appropriate data from an irradiation test facility are available for the material, metallurgical history, temperature range, and engineering parameters, and assuming the engineering property change of interest is reasonably described by one of the damage models, then the equivalent fluence for the data source and the FTR application can be used to correlate the two. For example,

an engineering property change observed at 10^{19} total fluence in EBR-II would be produced in the FTR grid plate at 5.8×10^{20} n/cm². Similarly, data generated in the ETR-G7 at 10^{20} n/cm² would be produced in the FTR grid plate at $58/9.6 \times 10^{20}$ or 6×10^{20} n/cm².

Engineering property data can be correlated to three materials locations in the FTR: the reactor vessel at midplane, the grid plate, and the fuel cladding. The clustered vacancy or the gross displacement damage models can be applied simply by using the appropriate equivalent fluence values K_i . For example, in the FTR fuel cladding, an equivalent irradiation damage fluence of 1.6 produces the same amount of damage as unit fluence in the EBR-II mid core. Similarly the equivalent irradiation damage fluence is 60 at the FTR grid plate and 150 at the FTR pressure vessel, based on the clustered ≥ 10 vacancy model. However, using the gross displacement model, the equivalent fluences at the FTR fuel cladding, grid plate, and vessel are 1.6, 35, and 81, respectively. Equivalent fluence ratios such as these relating the FTR and EBR-II can be used to correlate irradiation data from test facilities to the materials locations in the FTR where design and evaluation data are needed.

DISTRIBUTIONNo. of
CopiesOFFSITE

1	<u>AEC Chicago Patent Group</u> G. H. Lee
30	<u>AEC Division of Reactor Development and Technology</u> M. Shaw, Director RDT Asst Dir for Nuclear Safety Analysis & Evaluation Br, RDT:NS Environmental & Sanitary Engrg Br, RDT:NS Research & Development Br, RDT:NS Asst Dir for Plant Engrg, RDT Applications & Facilities Br, RDT:PE Components Br, RDT:PE Instrumentation & Control Br, RDT:PE Liquid Metal Systems Br, RDT:PE Asst Dir for Program Analysis, RDT Asst Dir for Project Mgmt, RDT Liquid Metals Projects Br, RDT:PM FFTF Project Manager, RDT:PM (3) Asst Dir for Reactor Engrg Control Mechanisms Br, RDT:RE Core Design Br, RDT:RE (2) Fuel Fabrication Br, RDT:RE Fuel Handling Br, RDT:RE Reactor Vessels Br, RDT:RE Asst Dir for Reactor Tech Coolant Chemistry Br, RDT Fuel Recycle Br, RDT Fuels & Materials Br, RDT:RT Reactor Physics RDT:RT Special Technology Br, RDT:RT Asst Dir for Engrg Standards, RDT
258	<u>AEC Division of Technical Information Extension</u>
47	<u>AEC Division of Technical Information Extension</u> UK/U.S. Fast Reactor Exchange (12) Euratom/U.S. Fast Reactor Exchange (35)
1	<u>AEC Idaho Operations Office</u> C. W. Bills, Director

No. of
Copies

1	<u>AEC San Francisco Operations Office</u> Director, Reactor Division
4	<u>AEC Site Representatives</u> Argonne National Laboratory Atomics International Atomic Power Development Associates General Electric Co., Sunnyvale
6	<u>Argonne National Laboratory</u> R. A. Avery D. Meneghetti LMFBR Program Office (2) G. F. Popper R. A. Jaross
2	<u>Argonne National Laboratory</u> Idaho Falls, Idaho N. J. Swanson F. W. Thalgott
7	<u>Atomics International</u> D. J. Cockeran (5) J. J. Droher (2)
2	<u>Atomic Power Development Associates</u> Document Librarian
3	<u>Babcock & Wilcox Co.</u> S. H. Esleeck G. B. Garton P. J. Kovach
8	<u>Bechtel Corporation</u> J. J. Teachnor
1	<u>Combustion Engineering, Inc. (AEC)</u> W. P. Staker, Project Manager
2	<u>Gulf General Atomic Incorporated (AEC)</u> D. Coburn

No. of
Copies

5	<u>General Electric Co.,</u> <u>Advanced Products Operation</u> Karl Cohen (3) Bertram Wolfe Nuclear Systems Programs D. H. Ahmann
1	<u>Idaho Nuclear Corporation</u> D. R. deBoisblanc
1	<u>Los Alamos Scientific Laboratory</u> W. R. Wykoff
1	<u>NASA Lewis Research Center</u> R. A. Hildebrand
1	<u>Nuclear Materials and Equipment Corporation (AEC)</u> C. S. Caldwell
1	<u>Oak Ridge National Laboratory</u> H. G. Duggan
1	<u>Pacific Northwest Laboratory Representative</u> N. A. Hill (ZPR-111 Rep)
1	<u>Stanford University</u> R. Sher
1	<u>United Nuclear Corporation</u> <u>Research and Engineering Center Library</u> R. F. DeAngelis
11	<u>Westinghouse Electric Corp.</u> <u>Atomic Power Division</u> <u>Advanced Reactor Systems</u> E. C. Bishop J. C. R. Kelly (10)

No. of
Copies

ONSITE-HANFORD

1	<u>AEC Chicago Patent Group</u> R. K. Sharp (Richland)
4	<u>AEC Richland Operations Office</u> <u>FFTF Program</u> C. L. Robinson J. M. Shivley (3)
4	<u>AEC Site Representatives - PNL</u> P. G. Holsted
4	<u>Atlantic Richfield Hanford Company</u> L. H. Richards L. B. Christopher H. P. Shaw R. E. Tomlinson
8	<u>Battelle Memorial Institute</u> S. L. Fawcett (2) W. E. Kortier (3)
2	<u>Computer Sciences Corporation</u> G. L. Otterbein C. D. Thimsen
4	<u>Douglas United Nuclear Corporation</u> R. S. Bell C. D. Harrington C. W. Kuhlman O. C. Schroeder
2	<u>Hanford Engineering Services, Vitro, Incorporated</u> J. M. Frame G. Kligfield
5	<u>ITT/FSS</u> J. M. Heffner W. M. Hunt T. P. Leddy M. F. Rice C. W. Weeks

No. of
Copies

1	<u>Bechtel Corporation</u>	
	D. H. Weiss	
1	<u>Westinghouse Electric Corp.</u>	
	R. Strzelecki	
153	<u>Battelle-Northwest</u>	
	F. W. Albaugh	J. F. Erben
	W. G. Albert	E. A. Evans
	G. J. Alkire	L. M. Finch
	S. O. Arneson	J. C. Fox
	F. G. Arrotta	R. C. Free
	E. R. Astley	R. J. Gandy
	J. M. Batch	E. E. Garret
	T. Bauman	S. M. Gill
	A. L. Bement	J. P. Hale
	R. A. Bennett	W. L. Hampson
	C. L. Boyd	K. M. Harmon
	D. C. Boyd	H. Harty
	C. L. Brown	R. A. Harvey
	W. W. Brown	B. R. Hayward
	W. L. Bunch	R. E. Heineman
	C. A. Burgess	J. W. Helm
	C. P. Cabell	R. J. Hennig
	J. J. Cadwell	J. D. Herb
	A. C. Callen	G. M. Hesson
	C. M. Cantrell	P. L. Hofmann
	J. R. Carrell	J. E. Irvin
	W. E. Cawley	B. M. Johnson
	W. L. Chase	H. G. Johnson
	T. T. Claudson	R. N. Johnson
	P. D. Cohn	E. M. Johnston
	D. L. Condotta	J. N. Judy
	R. R. Cone	J. J. Kempf
	J. H. Cox	D. D. Lanning
	J. M. Davidson	G. A. Last
	C. T. Davis	D. C. Lehfeldt
	F. G. Dawson	H. D. Lenkersdorfer
	D. R. de Halas	W. R. Lewis
	V. A. Deliso	C. W. Lindenmeier
	A. W. de Mershand	H. E. Little
	G. E. Driver	W. W. Little
	K. Drumheller	C. E. Love
	R. V. Dulin	D. Marinos

Battelle-Northwest (contd)

W. B. McDonald	D. D. Stepnewski
M. H. Meuser	G. H. Strong
J. W. Mitchell	C. D. Swanson
R. A. Moen	J. C. Tobin
C. A. Munro	K. G. Toyoda
C. R. Nash	M. A. Vogel
R. E. Nightingale	R. C. Walker
D. J. Oakley	D. M. Walley
D. P. O'Keefe	J. H. Westsik
M. G. Patrick	J. Wett
C. L. Peckinpugh	R. G. Wheeler
L. T. Pederson	L. A. Whinery
J. A. Perry	R. D. Widrig
R. E. Peterson	T. W. Withers
O. W. Priebe	N. G. Wittenbrock
H. L. Pringle	M. R. Wood
W. D. Richmond	F. W. Woodfield
W. E. Roake	D. C. Worlton
R. L. Roske	J. M. Yatabe
J. M. Seehuus	H. H. Yoshikawa
F. H. Shadel	W. R. Young
D. W. Shannon	FFTF File (25)
D. E. Simpson	Technical Information (5)
C. R. F. Smith	Technical Publications (3)
R. J. Squires	