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FAST FLUX TEST FACILITY
MONTHLY INFORMAL
TECHNICAL PROGRESS REPORT
OCTOBER 1969

AEC RESEARCH &
DEVELOPMENT REPORT

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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT
OCTOBER 1969

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Project Manager

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November 7, 1969

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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT
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ABSTRACT

This report was prepared by Battelle-Northwest under Contract No. AT(45-1)-1830 for the Atomic Energy Commission, Division of Reactor Development and Technology, to summarize technical progress made in the Fast Flux Test Facility Program during October 1969.

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FAST FLUX TEST FACILITY
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INTRODUCTION

Contents of this report are arranged and presented in a new format recently developed for FFTF periodic technical reports. The new format identifies, for each item of work reported,

- (1) The applicable LMFBR program plan element,
- (2) The FFTF system supported by the work, and
- (3) The BNW organization responsible for performance and/or management of the work.

The titles of the nine LMFBR program plan elements are used as the titles for the nine chapters in the body of the report. Each chapter is further subdivided to show specific FFTF systems and programs. Chapter II (Plant Design) presents design data for all systems, and covers progress reporting for the electrical, structures and site systems. Chapter III (Components) gives progress reporting for the reactor vessel, nuclear controls, fuel handling, heat transport and sodium, and gas systems. Chapter IV (Instrumentation and Control) covers the corresponding FFTF systems. Chapter V (Sodium Technology) presents technical information to support design of the heat removal, irradiation testing, and the sodium and gas systems. Chapter VI (Core Design) covers the core and irradiation testing systems. Chapter VII (Fuels and Materials) gives technical information to support design of process equipment throughout the plant. Chapter VIII (Fuel Recycle) covers fuel development and the fuel examination system. Chapter IX (Physics) and Chapter X (Safety) provide technical information supporting the design of the entire FFTF.

To identify the BNW organization responsible for the performance and/or management of the work, a special 3-digit "Author Code" is listed immediately after the name of each author of each item in the report.

CHAPTER I. PROGRESS SUMMARYA. STRUCTURE, SITE, UTILITIES

Drilling at the FFTF site to investigate load-bearing properties of the soil has been completed. Drilling required for seismic investigations of the underlying basaltic bedrock is continuing.

B. COMPONENTS

Westinghouse was directed to increase the reactor vessel diameter from 17 to 20 ft to permit use of a rotating plug fixed arm in-vessel fuel handling machine and separated instrument tree. The procurement documents for the reactor vessel, head, and guard vessel are being revised to reflect the change.

The fuel duct material has been selected as 20% cold worked Type 316 SS. Contract negotiations for the fabrication of prototype fuel ducts are in progress.

Progress review meetings for IHX design study contracts with Babcock & Wilcox, FW and WARD have revealed no insurmountable problems associated with the IHX concepts being developed for FFTF service conditions.

C. INSTRUMENTATION AND CONTROLS

An alpha-particle counting system, presently in use on the Hanford Atomic Project, has been chosen to monitor airborne plutonium collected on filters. This system discriminates plutonium in the presence of ^{220}Rn and ^{222}Rn by using a coincidence counting circuit.

DRDT has requested that the reference design for System 95 provide in-core neutron detector thimbles capable of being cooled to 300 °F. A study is in progress to meet this requirement.

Resistance measurements on chromel-alumel thermocouples which had been exposed to $\sim 3 \times 10^{21}$ nvt (fast) at EBR-II showed a negligible change in resistance in the thermal element leads.

D. SODIUM TECHNOLOGY

Installation of the remote flowthrough sampler was completed and sodium flow tests started. Four samples were taken successfully.

Significant amounts of ^{54}Mn were found by gamma scanning an irradiated pipe from EBR-II. These results confirm previous BNW radioisotope transport loop predictions.

E. CORE DESIGN

The formulation for the BEAM Creep Code has been completed, documented, and is being published. The general solutions developed will provide the required behavior prediction for FFTF fuel ducts.

A Study Group was established at BNW to evaluate the effect on core structures of the total reactor environment including thermal, mechanical, and structural effects from neutron induced swelling, in-reactor creep, and stress relaxation. The results of the study are being assembled into a report.

Studies are continuing to evaluate the independence requirements for the safety systems, and to define core radial restraint requirements. Fabrication of the CCTL Mark II Subassembly is in progress.

A preliminary study of an advance core for the FFTF was completed. Two fuel compositions - oxide and carbide - were described for full performance cores.

Testing requirements for the 2-MW modular closed loops were defined to include 1400 °F maximum sodium temperature

at test specimen; 1200 °F maximum sodium hot leg temperature; 400 °F minimum cold leg temperature; 400 °F maximum hot leg to cold leg ΔT ; and a maximum flow rate of ~290 gpm sodium.

F. FUELS AND MATERIALS

Creep-rupture tests were made at 1100 °F on solution treated (1950 °F/1 hr) Type 304 and 316 SS. For Type 304 SS irradiation reduces both the rupture life and minimum creep rate. For Type 316 SS irradiation reduces the rupture life and increases the creep rate.

Nondestructive examination of EBR-II capsules showed negligible swelling in B_4C and a small amount of swelling in Ta.

A remote re-encapsulation of a mock-up PNL-1 fuel pin was successfully accomplished in the 324 Building Shielded Materials Facility. This work is in preparation for re-encapsulation of a PNL-1 fuel pin previously irradiated in EBR-II.

Intergranular failure of cladding was NOT observed in a fuel pin which had been irradiated in GETR to a burnup of 50,000 MWd/tonneM average.

Subassembly PNL-2 (X032) which was discharged from EBR-II at an estimated burnup of 47,000 MWd/tonneM was successfully disassembled and the capsules neutron radiographed. The neutron radiographs indicate cladding integrity has been retained in all 19 fuel pins. One pin, deliberately made with thin (0.010 in.) cladding, appeared to be bowed to an appreciable extent.

G. FUELS RECYCLE

A method has been developed for quantitative analysis of fuel pellet homogeneity through electron microprobe scanning. Availability of the new analysis method has in

turn led to the development of an improved homogeneity specification for fuel pellets. Typical mixed oxide pellets fabricated by the BNW process readily meet the new homogeneity specification.

Approximately 5200 ft of high quality cladding tubing has been received from one vendor who has been authorized to begin production on the balance of the 30,000-ft order of tubing.

The criticality control program is completed and is in functional operation in the Fuel Fabrication Demonstration Facility. An extensive training program is underway to indoctrinate technicians in the use of the system.

H. PHYSICS

It was concluded that count rates in the 2-in. diameter ^{235}U chambers, located in the FTR reflector, will be sufficient at full shutdown ($k = 0.9$) to make adequate subcritical reactivity assessments. Also, it was concluded that safety rod movements from $k = 0.9$ will be an adequate scheme for establishing the neutron sensitivity of the sensors. Hence, an extraneous source in FTR is not a necessary requirement.

The predicted completion data for the FTR-2 Critical Experiments program is November 19, 1969.

The FTR-3 Critical Experiment program will follow the completion of the FTR-2 program.

I. SAFETY CONTAINMENT RESPONSE

A DBA energy release of 4000 MW-sec and a fuel vapor release of 200 kg was compared to a DBA energy release of 8000 MW-sec and 790 kg of fuel vapor reported previously. The reduced spray of fuel vapor into the cavity atmosphere reduced the initial pressure peak to 24 psig as compared to 50 psig.

The reduced value of excursion energy about doubled the time delay for initiation of boiling of the sodium pool on the floor of the cavity. The secondary pressure peak during the boil off of sodium from the cavity and its condensation in the cells remains the same, about 18 psig.

The consequences of a hypothetical pool fire in the main containment vessel were calculated for the case of a severe excursion accident with no machinery dome. Sphere internal pressure rose rapidly to a peak of 24.0 psig in 8.5 min; air temperature was 1060 °F while sphere steel temperature was 120 °F. An ultimate maximum temperature of about 650 °F for the sphere steel was indicated.

A draft of the PSAR was completed in October for submission to the CCB. However the Commission requested a 2 months delay in portions of the PSAR for re-evaluation of the basis for containment.

CHAPTER II. PLANT DESIGN

(See Appendix A for List of Author Codes Used in this Report)

A. OVERALL PLANT DESIGN

This section is presented in the form of Table 2.I as follows.

TABLE 2.I. *Current Design Data*Summary of Principal Concept Characteristics
and Data as of 10/16/69 - W. M. Noble (952)

Reactor Systems (31000) <u>General Plant Data</u>		<u>Units</u>	<u>Values</u>	<u>Status(a)</u>
1.	Core Arrangement		vertical	1
2.	Design life	years	20	1
3.	Peak flux	n/cm ² -sec	7.2×10^{15}	2
4.	Total power, initial	MW _t	400	1
5.	Reactor coolant		sodium	1
6.	Core volume	liters	1033	2
7.	Total reactor flow	gal/min	42,240 initial 43,290	2
8.	Reactor pressure drop design maximum	psi	145	2
9.	Reactor bulk inlet temperature initial	°F	600	1
	ultimate		750	1
10.	Reactor bulk outlet temperature initial core	°F	850	1
	design maximum and ultimate	°F	1050	1
11.	Core temperature rise average initial	°F	300	1
	ultimate	°F	350	1
	design maximum	°F	400	1
12.	Reactor cover gas		argon	1

(a) 1 - Baseline data.

2 - Current accepted reference design value.

3 - Contractor assumed design values.

TABLE 2.I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status^(a)</u>
<u>Physics Data</u>			
1. Delayed neutron fraction		0.003	2
2. Neutron lifetime	seconds	4.0×10^{-7}	2
3. Doppler	T(dk/dt)	-0.004	2
4. Power density, peak	MW/liter	0.75	2
5. Power distribution (peak/average)			
radial		1.40	2
axial		1.24	2
total		1.74	2
<u>Core (3103)</u>			
1. Number of core lattice positions		91	1
2. Number of driver fuel subassemblies		76	2
3. Number of in-core safety rods		3	2
4. Number of in-core control rods		3	2
5. Number of peripheral control rods		15	2
6. Equivalent core diameter	inches	~48	2
7. Active core height	inches	36	2

TABLE 2.I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status (a)</u>
8. Radial reflector material		nickel	2
9. Reflector thickness axial	inches	6	2
radial	rows	3	2
10. Fuel pin heat transfer area	ft ²	2980	2
11. Core coolant velocity, maximum	ft/sec	30	2
12. Direction of coolant flow		upward	1
13. Radial in-reactor shield			
- Material		stainless steel	2
- Configuration		hexagonal assemblies	2
- Thickness	rows	5	2
14. In-core open test positions			
- Number with contact instrumentation		2	2
- With proximity instrumentation*		1	2
- Power			
- Coolant flow rate			
- Test assembly length		Same as driver fuel assembly	

---

* One open test position will have superior accessibility. Driver fuel positions can be used as open test positions with limited accessibility.

TABLE 2.I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status(a)</u>
- coolant		Same as driver fuel assembly	
- test assembly cross section		Same as driver fuel assembly	
<u>Driver Fuel (31037)</u>			
1. Fuel composition		18 to 26 wt% PuO ₂ 74 to 82 wt% UO ₂	2
2. Cladding material		Type 316 SS	1
3. Linear heat generation rate, average (at 90% total power)	kW/ft	7.0	2
4. Overpower factor		1.20	2
5. Peak linear heat generation at over-power (at 90% total power)	kW/ft	14.6	2
6. Target burnup, average	MWd/tonneM	45,000	2
7. Cladding heat transfer coefficient	Btu/hr-°F-ft ²	37,500	2
8. Maximum fuel temperature (steady state)	°F	4230	2
9. Fuel assembly length	feet	12	2
10. Fuel geometry		Hexagonal pin cluster	1

TABLE 2.I (contd)

	<u>Units</u>	<u>Values</u>	<u>Status(a)</u>
11. Pin diameter	inches	0.230	2
12. Spacer wire diameter	inches	0.056	2
13. Number of pins per assembly		217	2
14. Subassembly cross-section outside dimension (across flats)	inches	4.615	2
15. Lattice spacing	inches	4.715	2
16. Duct wall thickness	inches	0.140	2

Vessels and Shields (3104)1. Reactor Vessel (31045)

- Diameter	feet	20	2
- Height	feet	50	2
- Wall thickness	inches	2	2
- Wall fluence, total (at 1000 °F)	nvt	2×10^{22}	2
- Material		304 SS	1

2. Cover Shield

- Material		low alloy steel	2
- Thickness	feet of stainless steel	4	2

TABLE 2.I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status^(a)</u>
-	feet of thermal baffling	1.5	2
	feet of cellular stainless steel	1.5	2
- Coolant		argon, helium or Na	2
<u>Heat Transport System (32000)</u>			
1. Primary loops			
- Number		3	1
- Primary loop material		304 SS	1
- Primary loop flow (per loop) (initial) gal/min		14,080	2
- Pipe size OD × wall thickness inches		20 × 0.375	2
2. Primary pumps			
- Number		3	1
- Design flow rate	gal/min	14,500	1
- Design pump head	feet	500	1
- Available net positive suction head	feet	51	1
- design temperature	°F	1050	1
- Speed control		wound rotor motor with liquid rheostat	1

TABLE 2.I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status(a)</u>
- Motor power			
brake	HP	1800	2
rated	HP	2500	2
3. Intermediate heat exchangers			
- Number		3	1
- Type		vertical shell and tube	1
- LMTD			
initial	°F	85	1
ultimate	°F	99.1	2
- Capacity			
initial	MW	133	1
ultimate	MW	157	1

Irradiation Testing
System (33000)

Test Facilities

1. Closed loops (33030)

- Number of positions	6	2
- Location	2 adjacent to core center	2
	1 at mid- radius	2
	3 at core periphery	2

TABLE 2.I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status (a)</u>
- Power handling capability	MW	4	2
- Test section outlet temperature	°F	1400 (bypass flow permitted)	1
- Number of loops installed initially		2 general purpose loops of 2 MW capacity and up to loops provided by users	2
		3 at core periphery	2
- Test section length	inches	36	2
- Test section diameter	inches	2.5-3.0	1
- Material (in-core tube)		316 SS	1
<u>Auxiliary Systems (44000)</u>			
Inert Gas Receiving & Processing (44040)			
1. Cover gas volumes			
- Primary cover gas			
reactor	ft ³	195	2
pumps (3)	ft ³	515	2
gas lines	ft ³	40	3
surge tank	ft ³	450	3
<u>Sodium Receiving and Processing (44030)</u>			
1. Sodium storage tankage for vessel drainage	ft ³	14,850	3

B. STRUCTURES, UTILITIES, AND CONTAINMENT

1. Structures and Containment

a. FFTF Site Investigation

E. M. Johnston (940)

The FFTF site investigation is in progress; Hatch Drilling Company is employing two drilling rigs to expedite the work. Drilling to ~220 ft (required to investigate the load-bearing properties of the site) has been completed. Drilling to bedrock at ~500 ft is continuing in order to make a seismic investigation.

b. Containment

The revised RFP for the containment vessel has been rewritten by Bechtel, and the revised document is being distributed for final review.

2. Utilities

E. M. Johnston (940)

Bechtel was requested to prepare a plan of action for prospective FFTF use of a Savannah River Plant Transformer, four Hanford 13.8 to 2.4 kV transformers, as well as 230 kV transmission line material and 13.8 kV switch gear also available at Hanford.

CHAPTER III. COMPONENTS

(See Appendix A for Author Codes Used in this Report)

A. REACTOR COMPONENTS1. Fuel Ducts

W. F. Brown (A00)
[Work sponsored by J. C. Tobin (960)]

a. Duct Welding Development

The conceptual design for the automatic duct welding system has been completed. Detail design is approximately 50% complete. Hardware fabrication for the duct fixture has been started.

b. Wear Pad Attachment

Studies were continued on the resistance welding of aged Inconel 718 to stainless steel. Two series of welds were made on aged Inconel 718, 0.060 in. thick, to annealed Type 304 SS, 0.143 in. thick. One series was directed to the study of the effects of current, time, weld pressure, forge pressure and time of forge application on cracking in welds with fused nuggets. The second series was directed to the study of the effects of these same variables on the tensile shear strengths of spot weld diffusion bonds. Preliminary evaluation of the results indicates that the extent of cracking was reduced in the second series, but that cracking still occurs when nuggets are of sufficient size to tear out of the Inconel 718 during tensile shear testing. The tensile shear strength for samples which show nugget tear was 6,000 lb, while the highest tensile shear strength for the diffusion bonded samples was 5,000 lb.

Two 12 in. long hexagonal duct test sections have wear pads assembled and are ready for welding. Since it appears that spot diffusion bonds can be achieved with satisfactory strengths with no cracking and other defects, this approach will be used on the test sections. The test sections will then be subjected to thermal cycling and other tests to determine the feasibility of this attachment procedure.

c. Duct Procurement Activities

R. N. Johnson (962)

Duct procurement contract negotiations are substantially complete with the two vendors selected for the fabrication of prototype fuel ducts. The costs and schedules established were based upon fabricating the ducts from Type 304 SS in the solution-treated condition. The choice of material and condition has been re-evaluated. However in the light of new stainless steel swelling information available, the choice of the duct material has been changed to Type 316 SS in the 20% cold worked condition. The vendors have indicated that the change may involve a slight increase in cost and schedule.

2. Reactor Vessel

a. Reactor Vessel Design

O. W. Priebe (912)

Westinghouse revised the equipment specifications for the reactor vessel, head, guard vessel and associated equipment to conform to BNW requirements prior to proceeding with design efforts by the vessel supplier. Subsequently, the vessel diameter was changed from the reference 17-ft diameter to 20-ft diameter. This change will require revisions to the specifications and drawings and approval of BNW and the AEC.

Westinghouse submitted an evaluation of several potential problem areas associated with the use of Type 304 SS for the reactor vessel and established a plan of action. A reactor cavity work plan was also presented to conform with establishing cavity requirements for Bechtel Corporation.

A summary of engineering work performed by WARD was also received of the efforts on the reactor vessel, head, guard vessel and associated equipment since configuration selection in February of 1969.

Babcock and Wilcox are devoting their conceptual study work on novel reactor vessel head designs and vessel support systems. Babcock and Wilcox plans to have potential candidate head designs available for review by November 1.

b. Caustic Stress Corrosion

G. R. Bloom (A00)*

Tests are being conducted to estimate the potential for caustic stress corrosion cracking in the FFTF reactor guard vessel. This Type 304 SS vessel, which will be exposed to a high temperature 96% N₂-4% O₂ atmosphere with traces of water vapor, must be capable of recovering from four sodium spills without replacement. There are no caustic cracking data for these service conditions.

Preliminary tests used Type 304 SS fuel cladding samples in room air to approximate the actual conditions. Sensitized and unsensitized samples were painted with a 75 wt% NaOH solution, pressurized internally to 800 psi with argon, and heated to 600, 800, 1000, and 1200 °F in air. The unsensitized samples were exposed for 120 hr without failures. The sensitized samples have undergone 100 hr of exposure without failure and will be exposed for an additional 20 hr. After exposure, the samples will be examined visually for oxidation, burst tested, and examined metallographically.

3. Nuclear Control Components

H. C. F. Ripfel (811)

a. CCDD

RDT approval for the Nuclear Control Components Conceptual Design Description was received. Resolution of RDT comments is in progress.

---

* Reported by W. R. Wykoff

b. Control Rod Driver Conceptual Studies

J. C. Noakes (914); W. S. Kelly, and
J. R. McBride (COO)

Results of survey of scram assist devices were issued. Final drafts of decelerator studies, drive line disconnects, stepping motor drives, and seal study were submitted for publication. Preparation of the electromechanical latch report is in progress.

c. Roller Nut CRDM Testing at LMEC

J. C. Noakes (914)

The test plan was approved and the drive was shipped to LMEC. Wear testing of the drive mechanism in argon gas was approved and preparations are underway to modify the contract with Todd Shipyards to perform the test at Galveston, Texas.

d. CRDM Prototype Procurement

P. K. Telford (914)

The draft issue of the technical input for the prototype roller nut drive procurement was received from Westinghouse, reviewed, and discussed to achieve a compatible 6-part format.

4. Other Reactor Components

a. Reactor Hardware Alignment Study

B. G. Smith (914); F. R. Reich,
and W. J. Coleman (MOO)

The final state-of-technology report describing the basic measurements and typical instruments required in the alignment and placement of reactor hardware in the FFTF and planned facilities has been completed. Review of the document will be performed during the month of October.

b. Removable Driver Fuel Instrument Assembly

R. F. Gilmore (914); and J. R. McBride (COO)

A description of the simulated test hardware, the test set up, and testing results were reported last month. The tests that were performed with the guide tube mounted in the vertical position showed that it was possible to insert the simulated instrument assembly unassisted into the guide tube and withdraw it with a force only slightly in excess of the assembly's weight.

Additional tests have been performed with the guide tube mounted in the horizontal plane. These tests gave results similar to those of the vertical tests.

c. IVHM Concept

E. J. Ruane (912)

A project decision was made that the In-Vessel Handling Machine (IVHM) will be separated in function from that of the Instrument Tree. The IVHM will be a rigid arm rotating plug design. Westinghouse is proceeding on this basis for the conceptual design, development planning, and testing for these components.

B. HEAT TRANSPORT COMPONENTS

1. Concept

a. CSDD Status

J. P. Petrek (923)

CSDD-51 is still under review by Westinghouse.

b. Hybrid Simulation of the Reactor and Heat Transport System

R. D. Benham (MOO); and A. L. Gunby (822)

The simulation of the reactor and primary hydraulics for pipe rupture studies has been completed on the Hybrid-2 machine. This program is identical to that employed on the

UNIVAC 1108 machine with the time-consuming, DYNASAR general purpose code. The hybrid simulation operates well and will be compared in detail with the slower, more expensive code during the next report period. Large cost savings in pipe rupture computations are expected.

2. Pumps

W. Babcock (923)

The final report for the Westinghouse Phase II Low Capacity Pump Study was received and is being evaluated.

The project proposal for pumps was completed and is being circulated for approval. Technical documents required for the RFP are completed.

3. Piping

M. W. Cook (923)

The first draft of the C. F. Braun Piping Design Guide has been received and is currently being reviewed by BNW and Bechtel. Comments will be assembled and submitted to LMEC early in November.

Bechtel is proceeding with a piping layout and design. Results to date indicate that the primary pump inlet can possibly use a 28-in. pipe, but since this is marginal, a 30-in. pipe may be required.

A Bechtel study indicates an HTS piping cost reduction of only \$37,500 resulting from the temperature reduction from 1200 to 1050 °F. However, this is based only on the reduced piping length. Further savings are probable.

4. Sodium Valves

C. R. Nash (923)

The report describing Procedures and Results for Selection of Prospective Exhibit-C Primary-Valve Concept for Preliminary Design has been received from Aerojet-General Corporation and

is being reviewed. By means of rating sheets and through discussions in a formal coordination and selection meeting, AGC selected three concepts of isolation valves and two concepts of check valves considered most promising for the preliminary design phase of the FFTF main heat transport system. They note that no single valve concept studies met all of the requirements set for the evaluation. The isolation valve concepts selected are the Tapered rotating-lift-plug type, the wedge-gate type, and the coaxial poppet type. The check valve concepts selected are the swinging gate type and the wedge gate type (power-operated and system-pressure-sequenced). The concepts were selected by rating each of the proposed concepts on its relative capability to meet the work statement requirements.

The RDT Standard, Isolation Valves for Liquid Sodium Service, and an accompanying commentary were received and are being reviewed.

5. Intermediate Heat Exchangers

a. Design

L. A. Whinery (923)

The Work Scope for the Westinghouse (WARD) LMFBR low-capacity IHX design study has been modified to: (a) delete superfluous work of a nature that overlaps with the parallel Babcock and Wilcox and Foster-Wheeler (FW) design studies, (b) redirect individual work tasks for more direct responsiveness to FFTF needs, and (c) improve the completion schedule data. It is presently estimated that a draft of the final WARD report will be available about November 15.

Progress review meetings on IHX design study contracts have been held during the past month with Babcock and Wilcox, FW, and WARD. Progress of all three contractors is proceeding essentially in accordance with Work Plan commitments. Final reports are scheduled for issuance on the following dates:

	<u>Issuance of Final Report</u>
Babcock and Wilcox	11-15-69
Foster-Wheeler	12-31-69
Westinghouse	1-15-70

Results of work performed to date by the three design study contractors have revealed no insurmountable problems associated with the IHX concepts being developed for the FFTF service conditions.

An IHX standard specification, together with FFTF ordering data, is being assembled by W (RPD) and is scheduled for submittal in draft form by November 1, 1969. A BNW/LMEC/RDT/W (RPD) task force review of the procurement specification documents is tentatively scheduled for the week of November 3.

b. Simulation of FFTF IHX

A. L. Gunby (822)

A study of IHX simulation approaches has been completed, extending the work reported previously.¹ The objective was to determine the limitations of current hybrid computer modeling for expected flow and temperature transients. As before, detailed models of the IHX design of March, 1969 (1660 tubes) were studied using the DYNASAR code. The following areas were studied: (1) the effect of using average temperatures for material properties, (2) the effect of using constant

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(1) *Monthly Informal Technical Progress Report, July, 1969, BNWL-1164, p. 8.1.*

values for material properties, (3) the effect of lumping metal with sodium, and (4) the effect of varying the number of axial nodes.

Average Temperatures. Results for outlet temperature responses showed that there was negligible difference between the two approaches.

Constant Properties. Again, results for outlet temperature responses showed negligible difference (less than 1 °F at all points in time following transient initiation).

Lumped Metal. When compared to the detailed model, outlet temperature transients were again shown to be negligibly different.

Number of Nodes. Using the detailed model, the number of axial segments was varied to 4, 6, and 8. Each model was subject to the temperature and flow input transients, with outlet results as expected. Accuracy increases and thermal transients become steeper with increased number of nodes. However, for the purposes of hybrid simulation (control studies, operator training, and system thermal transient check), a 4-node model with lumped metal and constant properties is satisfactory.

6. Coolant and Gas Systems

R. V. Dulin (925)

a. System 81 - Sodium Receiving and Processing

Bechtel is revising System 81 Schematics to reflect reduced reactor HTS secondary storage capability and improvements resulting from conceptual study efforts.

b. System 82 - Inert Gas Receiving and Processing

Harvard Air Cleaning Laboratory is performing a conceptual study of a bed for holdup of gases from failed and vented fuels.

Information is also being prepared by Bechtel and Westinghouse to support a decision on plant design provisions for failed and vented fuel.

c. System 83 - Auxiliary Heating and Cooling

Further study is proceeding by Bechtel and Westinghouse to determine reactor cavity cooling requirements for this system.

d. System 84 - Decontamination and Cleaning

A conceptual description of a system to be used for decontamination of systems and components has been completed and is undergoing technical review.

e. System 85 - Chemical Analysis Facility

Guidelines for completion of a conceptual design for this facility have been provided to Bechtel. Basically, the facility would consist of a remote sampling cave in containment, and pneumatic transfer of samples to an analytical laboratory outside containment. This laboratory would be equipped with either remote shielded coves or shielded glove-boxes for radioactive sodium analysis.

CHAPTER IV. INSTRUMENTATION AND CONTROLS

(See Appendix A for List of Author Codes Used in this Report.)

A. FFTF I&C SYSTEMS

(PBS 50,000)

1. System 91 - Central Control and Data Handling Systema. Control Room Mockup*

M. D. Erickson (MOO)

The Philco-Ford D-30 Display System has been installed. Work is proceeding on its interface with the Digital Equipment Corporation PDP/9 computer. At present the PDP/9 computer will write correctly on the D-30 Display System in both the high and low speed modes. But it is not yet possible to transmit information on the D-30 screen back to the computer. This problem is being corrected.

During this month, two memoranda were issued, namely:

- "FFTF Multiplexer Evaluation," Computer Systems Development Memo 69-12.
- "Diagram Generation on the Philco-Ford D-30 Color Display," Computer Systems Development Memo 69-11.

b. System Support*

G. W. Main (MOO)

Work is continuing on the development of the real-time system software in many areas. A Task compiler/assembler program is being developed which will permit use of the real-time system by personnel who are not familiar with the details of system operation.

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* Reported by J. W. Mitchell (933).

2. System 92 - Reactor and Vessel Instrument System

M. O. Rankin (933)

The CSDD for this system is awaiting DRDT approval.

3. System 93 - Plant Instrumentation System

M. O. Rankin (933)

Studies are in progress to reduce the number of wire runs from the transducers to the control room.

4. System 94 - Fuel Failure Monitoring System

a. Design Study - Fission Gas Detector Evaluation

W. Dalos (932)

A draft of a BNW design study on the comparative gamma detection sensitivity of GM tubes, gamma energy spectroscopy, and charged-wire detectors was sent to RDT on October 8. Before a final choice of the gamma detection method can be made, the coolant fissile contamination rate and the assumed fission gas rate of release from a ruptured fuel pin must be defined. These two values determine the background and the signal strengths.

b. Gas Disengagement Studies at ANL

E. Sowa (ANL) and M. J. Rieb (932)

The modification of the Fuel Failure Development Loop (FFDL) has progressed on schedule. The lower section of the out-of-vessel gas disengager was received from the shops and welded into the loop in October, and the remaining parts were delivered. Capsules containing ^{85}Kr were ordered from the Chemical Engineering Section.

c. Concept Evaluation

R. R. Derusseau (924)

A task force was assembled to evaluate the fuel failure location subsystem concept presented in CSDD-94. The task

force has completed this evaluation, considered alternative concepts, and presented their conclusions and proposals to department management. An early decision is expected to be made on whether to proceed with the reference sampling line concept or with alternative concepts. The status of the CSDD will be dependent upon this decision.

d. Fission Gas Sampling Development

R. R. Derusseau (924)

The sample line development task, 121FS-A, has received RDT approval to build a water mockup of the sample line. The initial Work Plan for both the water mockup and the sodium prototype loops was completed and sent to RDT for information. Westinghouse has submitted the design sketches to be used in constructing the water mockup and in designing the sodium prototype sampling line.

The development task 121FS-C proceeded with testing a modified model of the fission gas concentrator (vortex generator). The original W model was successful in concentrating the gas near the generator surface, but dispersed it again before it reached the sample pickup point. The model was then modified to remove a water-gas sample from the generator surface. Early results indicate that the removal efficiency was significantly increased over the test results prior to the modification.

5. System 95 - Flux Monitoring and Control System

a. CSDD Revision

L. W. McClellan (932)

Work has started on revision of CSDD-95.

b. Neutron Detector Thimble Design

L. W. McClellan (932)

DRDT has requested that the reference design for System 95 provide in-core neutron detector thimbles capable of being cooled to 300 °F. An FFTF Project analyses of neutron detector thimble cooling design is being performed jointly by the Engineering Department and by the Reactor and Plant Technology Department. These analyses will evaluate the feasibility of thimble cooling and will establish the requirements and methods for cooling the assembly (including sensors and cables) to 300 °F or less.

6. System 96 - Radiation Monitoring System

a. Plutonium Monitoring

M. O. Rankin (933)

An alpha-particle counting system, presently in use on the Hanford Atomic Project, has been chosen for monitoring airborne plutonium collected on filters. This system discriminates plutonium in the presence of ^{220}Rn and ^{222}Rn by using a coincidence counting circuit.

7. System 99 - Plant Protection System

G. H. Strong (943)

a. CSDD

The CSDD for this system has been reviewed by the Technical Evaluation Board and submitted to the FFTF Configuration Control Board.

b. Plant Protection System Reliability Analysis

O. B. Monteith (821)

Some preliminary evaluations have been made on the probability of spurious responses (false scrams) of the PPS resulting from equipment failures. An initial result for a typical

instrumentation subsystem is about 0.0004 false trips per year. Based on the present nominal number of 50 such subsystems, the expected spurious scram rate for the PPS instrumentation is about 0.02 per year. These early results do not include the effect of failures in the ac or dc power supplies, the safety circuits, nor the rod release circuits.

8. Control Analysis

R. A. Harvey (823)

A simplified dynamic model of the FFTF was simulated on the hybrid computer for control studies. A simplified transfer function determined from measurements on a detailed hybrid simulation was used to represent each portion of the dynamic operations of the facility. Various types of control schemes will be studied. The model (SSYM) offers two advantages over the detailed simulation; it leaves more amplifiers free for simulating controllers and it can be operated in a faster time scale.

The multivariable-representation of the FFTF has been determined so that monitored and controlled parameters appear as discrete values. Also, the blocks were determined to yield the maximum accuracy with the simplest transforms.

SSYM has been programmed on the analog computer. The frequency responses of the individual blocks in SSYM have been compared with those of the detailed simulation and good agreement has been achieved. The frequency response of the overall systems will be compared to ensure that SSYM is an accurate representation of the FFTF.

B. DEVELOPMENT OF SENSORS AND RELATED EQUIPMENT

1. In-Reactor Thermocouples

a. Effect of Irradiation on Resistance

N. C. Hoitink (MOO), N. S. Porter (MOO), R. C. Weddle (MOO), J. L. Jackson (AOO), and D. C. Thompson (931)

Resistance measurements on eleven 1/16 in. OD 30-in. long chromel-alumel thermocouples which had been exposed to $\sim 3 \times 10^{21}$ nvt (fast) at EBR-II showed a maximum change in resistance in the thermal element leads of less than 2%. This change in resistance is considered within the experimental error in the measurement.

2. Pressure Sensors for Sodium Service

R. D. Crosier (COO), K. O. Creek (932), and M. O. Rankin (933)

a. Sodium Pot Tests

Temperature testing of the following pressure transducers in sodium pots has continued during this reporting period:

- Consolidated Electrodynamics Corporation, Model 4-361-001 S/N 2260 (rated at 600 °F and 250 psia) is being tested at 875 °F.
- Kaman Nuclear, Model K-1900 S/N 1099 (rated at 1000 °F and 30 psia) is being tested at 725 °F.
- Consolidated Controls Corporation, Model 41M7130A S/N 1442 (rated at 900 °F and 30 psia) is being tested at 825 °F.

Data from these tests have not yet been reduced for evaluation.

3. Low-Level Neutron Flux Instrumentation

a. Coaxial Cable Noise Level Tests

C. N. Jackson (MOO), N. C. Hoitink (MOO), N. S. Porter (MOO), R. C. Weddle (MOO), and D. C. Thompson (931)

Low-level signals from high voltage neutron detectors are frequently obscured by "breakdown pulse noise" when coaxial

cables supplying high voltage to the detectors are heated. Cable noise tests were made on a neutron detector wherein 19 feet of two coaxial cables were heated to 900 °F while polarized with 400 volts. (One coaxial cable is used for high voltage supply, while the other is used for signal.) Compared to the conventional single cable configuration, noise generation in the double cable configuration was reduced by a factor of 10^4 . In these tests the detector was not heated. Temperature-induced noise in single cable chambers has been a significant deterrent to high temperature operation.

4. Reactor Environmental Effects on Signal Cables*

J. L. Stringer (MOO)

Experimental work on this task has been completed. A final BNWL report will be issued on all work that has not been previously reported. This report entitled Induced Current and Electrical Conductivity Measurement Techniques by J. L. Stringer is expected to be issued in November as BNWL-1148.

5. In-Core Flowmeter

a. Technical Support of ANL Flowmeter Development

R. L. Brown (MOO) and M. J. Rieb (932)

This task is directed toward providing in-depth technical support for the primary flowmeter development effort at ANL.

During this reporting period specific procedures were proposed to ANL for the construction of several copper-wound developmental probes to accelerate the selection of the reference flowmeter probe geometry. The use of copper wire, which is readily available in a wide range of sizes, should make possible the early selection of wire size and coil form geometry. However, it appears that neither the copper wire nor the nickel-clad silver wire presently being used at ANL for probe

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* Reported by D. C. Thompson (931).

development purposes will be suitable for the final design. Information is being compiled on other conductor materials in search of nonmagnetic materials with low resistivity and an adequate high-temperature lifetime.

6. AI Sodium Temperature and Flow Rate Probe
W. Dalos (932)

The FFTF Project will support a short term effort on the development of the AI Reactor Core Parameter Probe (RCPP). The RCPP is a device which can be inserted into a fuel duct to measure sodium temperatures and flow rates. At present, a memo purchase order (MPO) to support this work is being processed through the BNW Purchasing Office.

CHAPTER V. SODIUM TECHNOLOGY

(See Appendix A for List of Author Codes Used in this Report.)

A. ANALYTICAL METHODS FOR SODIUM AND COVER GAS*

R. L. Dillon (A00)

1. Remote Sample Development

The objective of this activity is to develop and proof test remotely operable samplers for the highly radioactive sodium from the FFTF.

The first attempt to sample the Small Components Evaluation Loop with the remotely operable bypass sampler was made. Before a full flow was established, a sodium leak was detected at the outlet Conoseal. The flow was immediately stopped and no fire or damage resulted. Possible causes for the leak were investigated thoroughly before a second sampling attempt was made. This investigation revealed the most probable source of the difficulty was in the Conoseal clamping mechanism. For succeeding samples, standard Conoseal nuts were used to fasten the sample section to the sampler. With this change the next four samples have been taken without incident. About 1 1/2 hr are necessary to leak check the sampler and preheat the lines. Flow can be easily controlled with the bypass valve. A standard operating procedure and operating check list have been prepared and used.

2. Carbon Analysis

Testing of the Manco guillotine as an extruder for carbon sample preparation has been completed and the device has been placed in routine use. Evaluation of the component concentrator for the trapping of traces of CO₂ has been completed and the device placed in routine use. The carbon sample load has

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* Reported by W. R. Wykoff (922)

been high during the report period. Plans have been made to attach an Electron Technology Inc. helium diffusion purifier to the train.

3. Laboratory for the Analyses of Alpha Contaminated Sodium

Final drawings of the air atmosphere glovebox have been completed and a cost estimate obtained with delivery scheduled for November. Room 312 is being vacated by its present occupants and room modification plans are under development.

4. New Analytical Methods

A method for measuring calcium in sodium is being evaluated and proof tested. It involves co-precipitation of calcium with strontium oxalate followed by atomic absorption analysis of the redissolved precipitate.

A new method for measuring halogens (exclusive of F) in metallic sodium is also being evaluated. It employs a mercury complexometric titration after isolating the halogens by an amalgamation technique similar to that which is used for oxygen determinations.

5. The Analysis of Carbon Species

A precision gas chromatograph is in place and scouting experiments on carbon species measurements have started. Initial experiments have been directed toward isolating elemental carbon. Carbon spiked sodium samples have been dissolved in either methanol or water inside an enclosed glass vessel under an inert atmosphere or vacuum. The resulting solutions containing carbon particles were filtered through Gelman Glass (no binder) micron filters, and then combined with oxygen in a standard carbon apparatus.

The blank associated with the 1.0-in. Gelman filters were found to be equivalent to about 5 micrograms of carbon. Degasing methods for lowering the blank are being considered.

The reaction $C_{(s)} + 2 OH^- + H_2O \rightarrow CO_3^{=2} + 2 H_2(g)$ reportedly interferes in aqueous dissolutions. No evidence of this has been observed in the present experiments.

B. COOLANT AND GAS TECHNOLOGY DEVELOPMENT

1. Selection and Testing of In-Line Instruments*

R. L. Dillon (A00)

The objectives of this task are to conduct field tests of currently available in-line sodium impurity instruments and to develop an in-line instrument system for the analysis of sodium in FFTF primary sodium systems.

a. Carbon Meter

Tests on the United Nuclear carbon meter continued this month. The carbon meter response to Na_2C_2 injections is being evaluated.

A calculated 10 ppm C equivalent Na_2C_2 was placed upstream of a fine screen in a flowthrough injection tube on the BNW Na purification loop (SPL) and produced a 10-fold increase in the carbon meter flux. It is clear that all this Na_2C_2 did not dissolve at once for the meter was still producing a high flux after one week. Yet the flux fell rapidly each time the source was valved off. The cause of this falloff is uncertain and is being investigated.

b. Oxygen Meter

The United Nuclear oxygen meter on the Sodium Purification Loop produced a steady voltage (subject to the previously reported annoying corrections for ambient and sodium temperatures) which did not change during the carbide addition mentioned above.

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* Reported by W. R. Wykoff (922).

A trip was made to Brookhaven National Laboratory to discuss the changes in the reference electrode construction being worked on by BNL. BNL proposes replacing the Cu/CuO electrode with either sodium/sodium oxide or air. They prefer to operate cells containing these electrodes at 400 °C rather than 315 °C as used for the Cu/CuO.

2. Decontamination*

R. L. Dillon (A00)

The objectives of this task are to screen possible decontamination solutions for effectiveness. A piece of pipe from the dismantled Radioisotopes Transport Loop (3000 hr of service at 1300 °F) was subjected to a solution containing 0.8 M oxalic acid, 0.2 M HF and 0.015 M H₂O₂ in the acid decontamination loop. Decontamination and weight loss data were obtained as a function of time. After 8.5 hr of exposure to the solution at about 95 °C, the decontamination factor was 12 for ⁶⁰Co and 6.7 for ⁵⁴Mn; calculated uniform penetration of the pipe was 0.31 mils.

These decontamination factors are well below the values sought (DF of 100 to 200). The test is continuing to determine if longer exposure will improve the DF.

3. Sodium Erosion Tests

A. B. Webb (954)

The T-1 test series, which includes T-1 B (now at LMEC for testing) and T-1 C & D (now being fabricated), is intended to explore sodium erosion effects at high velocity in the fuel duct tip/receptacle area. The T-1 B test apparatus consists of five modules having sodium velocities ranging from 30 to 70 ft/sec; the T-1 D test apparatus is a follow-on test consisting of two modules having sodium velocities of 90 and

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* Reported by W. R. Wykoff (922).

100 ft/sec; the T-1 C assembly consists of testing one of the T-1 D modules in water for hydraulic similitude purposes prior to the sodium test. In the series, the annular space between the fuel duct tip and the mating receptacle on the lower tube-sheet will be tested as well as the fuel duct/receptacle seat. High velocity sodium has been found by other investigators to erode stainless steels under certain conditions.

Fabrication of the T-1 C and D test apparatus was begun in September but was suspended pending configuration changes. The work has been resumed with most machining operations completed and piping preparation beginning in early October.

During preparatory activities at LMEC, metallic particles were discovered in the T-1 B Test Apparatus header piping. The headers were examined carefully and subsequent cleaning which included vibration, vacuuming, and swabbing was successfully accomplished on October 9, 1969. LMEC was authorized to resume preparatory activities, with T-1 B testing to be initiated on October 24, 1969.

4. Development of Power Controllers*

R. L. Dillon (A00)

The development of low cost power controllers is continuing. Presently the controllers are rated at 3.6 kVA and required about 16 in.² in panel area and are equipped with an ammeter. The objectives are to improve the controllers by:

- Reducing the area to ~5 in.² and maintain the 3.6 kVA rating and amperage indication
- To add a temperature-indicating function and maintain the 16 in.² area, the 3.6 kVA rating, and the amperage indication, and

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* Reported by W. R. Wykoff (922).

- To increase the kVA rating to 22 kVA and maintain the area >20 in.², and provide amperage and temperature indications.

The necessary components have been ordered and detailed design is underway. The assembly and evaluation will begin as soon as possible. It is believed that these controllers may have a place on reactor systems, in addition to testing facilities.

5. Chemistry of Large Sodium Systems*

R. L. Dillon (AOO)

The objective of the Chemistry of Large Sodium Systems program is to develop methods and equipment for the control of sodium quality in FFTF. This program consists of development tasks for the definition of problems and development of solutions in the areas of in-line instrumentation, cover gas sampling and analysis, sodium purification, and sodium sampling. These tasks will result in prototypes of reactor equipment which are to be developed utilizing the Prototype Applications Loop (PAL) for component testing and later the Sodium Chemical Technology Facility (SCTF) for large scale development and testing.

a. Prototype Applications Loop

The funding for the fabrication of the PAL has been approved and the initial preparations have begun. The PAL is the first major sodium system to be fabricated under the new Q/A guidelines set forth in BNWL-MA-64; therefore, there will be considerable emphasis on Q/A procedures. The PAL will be fabricated to the Level 3 standards specified in BNWL-MA-64. A CPM network describing the design and fabrication tasks and the interaction with Q/A has been prepared.

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* Reported by W. R. Wykoff (922).

b. Project BCE-042 Sodium Chemical Technology Facility

The SCTF has been conceived to provide the basic isothermal forced convection sodium loop required for the study of the chemistry of large sodium systems and for evaluation and development of FFTF and LMFBR purification, analytical, and sampling equipment. The basic loop includes a 1500 gal capacity sodium reservoir and a 2-in. piping system flowing sodium up to 200 gpm and 1200 °F.

The SCTF is to be funded as a capital equipment project, which is currently in the proposal stage. Rewriting and resubmission of the project proposal has been necessitated by an AEC ruling that the original site, the proposed 190 FA Building, was unacceptable. A feasibility study of the 335 Building showed that SCTF occupancy is compatible with other building functions if the building is modified by providing a pit below and a mezzanine area above floor level to create a 3-level working space.

Accordingly, the SCTF design criteria, preliminary project proposal, and the cost estimate have been rewritten and revised to reflect 335 Building occupancy. A rough draft of the design criteria has been circulated for comments, returned comments incorporated, and the criteria document is being prepared for circulation and approval within BNW.

c. OPERATION OF SODIUM FACILITIES

1. Small Component Evaluation Loop (SCEL)

J. D. Berg (954)

A decision was made to replace the failed pump (reported last month) in this facility with the EM pump that was removed from the Small Heat Transport Loop during upgrading in April. Although the replacement pump (MSA style VI) is smaller than the failed pump, it does have the capacity to supply sodium to

the remote sampler loop facility which is now connected to the SCEL. Following this installation, the SCEL and remote sampler loop sodium transport lines were preheated and charged with sodium and flow established in the SCEL and through the sampler loop bypass line. An attempt to establish flow through one leg of the sampler was aborted when sodium leaked from one of the mechanical seals in the sample leg. The sampler loop has been frozen off while the sample tube connectors are being modified to prevent leakage. During the interim, the SCEL is operating at 10 gpm and 600 °F bulk sodium temperature.

2. Small Heat Transfer Loop

J. D. Berg (954)

The 7-pin loose wire wrap fuel cluster was installed in this loop for continuation of the fretting corrosion test. Following system preheat and sodium purification, the loop was increased to test conditions of 1150 °F, 19.5 gpm (≈ 31 fps) by 9-7-69. This test assembly has now accumulated 1530 hr of exposure, of which 530 were accumulated during this report period.

The 7-pin electrically heated simulated fuel assembly, which was removed from this loop on 8-29-69, was cleaned and returned to test sponsors for disassembly and examination. For this cleaning effort, a new procedure was used to minimize the potential of damaging the delicate thermocouples within the test section duct. The new procedure employed hot oil, Dowanol, and water cleaning without submerging the test assembly. This method of cleaning proved to be very satisfactory.

The activities below are being conducted prior to the next 7-pin, electrically heated, simulated fuel assembly test to ensure reliable data:

a. Thermocouple and RTD Calibration

In-place calibration of the test section inlet and outlet thermocouples and resistance temperature detectors (RTDs) is required. To accomplish this, a dummy test section has been fabricated. This test section contains inlet and outlet thermocouple wells which will house "S" (Platinum, Platinum--10% Rhodium) type thermocouples for calibration of the in-place RTDs and type "K" (Chromel--Alumel) thermocouples. The "S" type thermocouples have been calibrated against freeze point standards by the BNW Standards Laboratory.

b. Shielding of Signal Leads

To reduce the noise problem on signals going to the data acquisition system only shielded leads will be used. Shielding of the present leads to the data acquisition system is 70% complete.

3. Static Sodium Pots

J. D. Berg (954)

Pressure transducers in static pots 1, 3, and 4 continued without difficulty throughout this report period. It is significant to report that the transducer in pot 3 has now accumulated ≈ 6000 hr of exposure. The Taylor pressure transducer in static pot 2 was removed, cleaned, and returned to the test sponsor.

4. Sodium Purification Loop

J. D. Berg (954)

This loop has continued operation without interruption with direct reading O_2 and carbon probes in service. Loop conditions are being maintained at 600 °F and 0.5 gpm.

5. Modifications to Fission Product Control Screening Loop*

R. L. Dillon (AOO)

A cold trap has been fabricated and installed in the Fission Product Control Screening Loop. The Pressure Piping Completion Record has been approved. The manufacturer supplied the wrong type of variable speed motor for cooling the cold trap. Arrangements have been made with Purchasing to correct this error and the replacement motor will be installed as soon as it arrives.

6. Facilities Additionsa. Project BAP-027 FFTF Sodium Facilities Building

K. D. Hayden (954)

Construction work has nearly been completed on the Sodium Facilities Building (335 Building). The 100 ft by 60 ft building and the personnel area were accepted from the Fixed Price Contractor on 9-19-69 with exceptions.

J.A. Jones personnel have installed two motor control centers in the building and have completed the outdoor switchgear and transformer installation. The relocation of the Small Components Test Loop from the 314 Building to the 335 Building has been initiated.

b. Project BCP-035 Building Mezzanine Addition

This project is being carried out to provide space for location of sodium development facilities. Completion is expected prior to the end of calendar year 1969.

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* Reported by W. R. Wykoff (922).

c. Project BAP-032 LMFBR Core Segment Development Facility

K. D. Hayden (954)

Design effort by Vitro is continuing on the LMFBR Core Segment Development Facility. The architectural and structural drawings are essentially complete. Continued effort is being applied to incorporate appropriate stress analysis into the design of the sodium vessels and piping.

It is intended to proceed on a separate bid package for the pit and steel-lined caisson. This bid award is scheduled to be awarded about the middle of November, 1969.

d. Project BCE-037 Fast Reactor Thermal Engineering Facility

K. D. Hayden (954)

The design of the Fast Reactor Thermal Engineering Facility is about 30% complete. Design Criteria Revisions 1 and 2, providing for substituting an EM or centrifugal free surface pump in the primary loop instead of the excess CANEL pump and for relocating the loop and test section outside Building 3718, were prepared and all approvals have been received.

BNW Quality Assurance Requirement 4-10 for "Construction Quality System Requirement" and Quality Assurance clauses have been prepared for application by the construction contractor during the FRTEF construction phase.

In compliance with BNWL-MA-64, BNW Manual for Design Construction of Systems and Equipment Containing Liquid Alkali Metal, Paragraph 5.6, Physics and Engineering Division has established a design safety review committee. A preliminary review meeting was held on October 14, 1969.

e. Cover Gas Evaluation Loop (CGEL)*

R. L. Dillon et al. (A00)

The Cover Gas Evaluation Loop is an isothermal sodium loop of about 40 gallon capacity for the study of sodium cover gas interactions and the testing of cover gas and sodium analysis instruments.

Construction of the Cover Gas Evaluation Loop has been completed and the system was successfully pressure tested and leak checked. The Pressure Piping Completion Record has been approved. However, extensive modifications have been requested to meet up-dated experimental requirements. The functional requirements are being prepared and detailed design for the modifications will begin shortly after receipt of the functional requirements.

Initial startup of the CGEL is planned for late November.

The first model of an in-line instrument package system is being designed for the CGEL and will contain an oxygen, carbon, and plugging meter. New bypass loops for two oxygen meters, one carbon meter and an oscillating plugging meter, with the necessary temperature and flow controls will be built for the CGEL package.

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* Reported by W. R. Wykoff (922).

CHAPTER VI. CORE DESIGN

(See Appendix A for List of Author Codes Used in This Report.)

A. CORE MECHANICS CONCEPTUAL DESIGN, DEVELOPMENT, TESTING1. Reactor Core Hardware Development Plan

J. F. Wett (913)

a. Reactor Hardware Test Program

L. R. Besel (913)

The objective of this work is to identify the areas of the current FFTF reactor core design which warrant developmental testing prior to their incorporation into the reactor plant and to prescribe such tests along with the recommended test facilities.

A discussion of the potential problem areas where testing is warranted has been prepared for all core subassemblies. Future plans are to group these problems according to test facility requirements and to identify the test priority which should be followed. Overall test procedures will then be developed for each facility.

b. Mechanical Core Mockup

L. R. Besel (913)

A mechanical core mockup has been suggested as a component which will be needed for FFTF hardware development. A study has been started to define more precisely the type of mockup needed for each of the components which would be assembled to form the Mechanical Core Mockup. The determinations are being made on the basis of the tests which would be performed with each of the components. The interaction of the tests proposed for the various components is also being examined to arrive at a logical sequence for assembling the complete mockup, since much testing can and should be done during the assembly phase.

2. Core Support Structure

D. J. Oakley (913)

Design analysis on the core support structure was redirected from a 17-ft internal diameter vessel to a 20-ft internal diameter. The initial drawings show that the core support structure is 20-ft in diameter, and approximately 4-ft thick.

Design criteria are being developed for the design of the core support structure. The limitation of core plate motion is established, since it implies a movement of the control rods with respect to the core. The following limits are established:

<u>Condition</u>	<u>Reactivity Limit, ¢</u>	<u>Core Plate Motion Limit, in.</u>
Slow Oscillations (frequency <1 cps)	1	±0.020
Operating Transients (e.g., startup)	5	±0.100
Accident Conditions (no fuel melting)	50	±1.0
Shock Wave (e.g., check valve slam)	50	±1.0

Stress determination will be based upon this ASME Code, Section 3 properties and intent.

Design drawings of the core support structure have been started, but are not complete. Based upon the structure shown, a steady state deflection of approximately 0.015 in. is calculated.

The sodium erosion tests of the hydraulic balance and fuel assembly seat, T-1B and T-1D, are not in operation. (See also Chapter VB paragraph 3.) Test T-1B is at LMEC and has been delayed by loop operational difficulties. The test section

had seal caps on the ends of the headers, which prior to loop installation had to be cut off. Upon removal of the seal caps, metal chips were found inside the headers. A procedure for additional inspection and cleaning the test section has been worked out and is being conducted.

Test T-1D is in fabrication. However, the design of the fuel assembly seat may have to be modified to match the design that is being developed and shown on the core support drawings.

3. Radial Restraint Design and Development

a. Computer Code Development Support

L. R. Besel (913)

The object of this effort is to provide experimental data for evaluation of two finite element computer codes for swelling and creep analysis of FTR driver fuel ducts. The two codes involved are the AXICRP computer code and the CRASID computer code. The latter one is under development.

The material characterization experiment, involving creep bending of a square bar of stainless steel at a temperature of 1200 °F, is underway. Calibration tests and bend tests to check out and calibrate the instrumentation were completed. Strain gages have been affixed along the length of the bar to monitor possible strain variations with temperature. A minimum test period of 1000 hr is planned.

Scheduled testing of the hexagonal duct (the pure bend creep test) depends on the timely delivery of the required oven and installation of additional electrical power to the test site.

b. Core Restraint Development

L. R. Besel (913)

The core restraint development and test program is to provide experimental data for design input, and development information necessary to demonstrate that the core restraint devices and systems are capable of meeting the design requirements within the constraint of the design and safety criteria. This effort is to include interfacing with such other core components as the instrument tree and IVHM.

As agreed with Westinghouse in a meeting on September 9, the simulated core model is to be constructed to evaluate experimentally kinematic and structural design problems related to core restraint, duct insertion and withdrawal, and preferred fuel management schemes. A project proposal for the simulated core model is being prepared so that detail design and procurement of materials can be initiated. Scope drawings are presently in progress to define the model and associated systems more precisely.

c. Simulated Core Model

G. R. Waymire, W. U. Jackson (811)

Restraint Development. Based on the agreements reached with Westinghouse concerning development testing in support of the radial restraint as noted in last month's report, conceptual drawings of a full core simulator have been completed as supporting information for the Simulated Core Model (SCM). The design utilizes simulated ducts fabricated from tubing with simulated stiffness. The contact pad area will be mocked-up with hexagonal stock attached to the tubes. Loading is afforded by a system of hydraulic jacks at the three loading elevations. Core support will utilize a tubesheet with prototypical seat configuration. Plan and elevation views are shown in Figures 6.A.1 and 6.A.2, respectively.

FIGURE 6.A.1. Simulated Core Model Plan

SK-3-14762

SK-3-14763

FIGURE 6.A.2. Simulated Core Model SCM Evaluation

4. Core Structural Analysis

R. J. Jackson (832)

The Core Structures Study Group has completed its definition of the reactor environment and identification of the major core component design problems facing the RPD. The definition of the reactor environment included detailed maps of the temperatures, fluences, and metal swelling throughout the core. Resultant structural changes caused by metal swelling and thermal distortion were calculated. Ducts, closed loops, and control assemblies lengthening, dilating and bowing, and interactions were predicted by use of elastic theory. A brief summary of the results are presented in Table 6.A.I. The structures report is being prepared.

TABLE 6.A.I. *Calculated Results of Thermal Distortions*

<u>Component</u>	<u>Impact of Swelling on Performance</u>
Driver Ducts	• Dilation leading to contact between Rows 1, 2, and 3 for Type 304 SS after about 220 days. • Duct bending is largest in Rows 5 and 6. • Duct top deflection during refueling is almost entirely restrained by the reflectors in Rows 7, 8, and 9. • Length increase can be accommodated by design changes in the instrument tree.
Control Rods	• Large diametral clearances are required between the guide tube and inner duct for adequate control rod life.
Closed Loops	• Closed loop dilatation and bending causes contact between it and adjacent ducts at about 320 days in Rows 5 and 6 (approximately 40,000 MWd/tonneM burnup in these rows) using the preferred CW Type 316 SS. • A crushing interference fit from differential swelling is predicted between the flow tube and the flow tube line.

5. Creep Code Development

W. H. Sutherland (COO)*

A general finite element analysis procedure has been defined which will permit the creep analysis of FFTF fuel assembly ducts, or any other statically indeterminate beam with arbitrary cross-section. This formulation is sufficiently general to include all of the RPD code specifications (WARD-FRA-25) with the exception of the plasticity capability. Furthermore, the incremental procedure employed for creep analysis has the potential to be generalized to include plasticity.

The creep formulation and documentation has been completed. Generalization of the incremental procedure to include plasticity is projected completion early in the fourth quarter of the fiscal year.

6. Core Concept

a. Reactor Core CSDD

G. R. Waymire (811)

Formal comments received from RDT concerning the Reactor Core CSDD have been reviewed. As a result of this review, most of the requested changes are being incorporated through the BNW approval chain. Several comments concerned the nonbaseline sections of the document and will be changed by Westinghouse at a later date.

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* Reported by L. M. Finch (810)

b. Concept Evaluation

D. Marinos, G. R. Waymire, and A. J. Anthony (811)

The Westinghouse concept evaluation report is complete and has been transmitted internally for comments. The report compares the reference concept and the Westinghouse concept, with variations on both. Specific comparisons include

- Cost, and
- Availability.

Potential problem areas are also identified.

B. CORE PROCESS TECHNOLOGY

1. Hydraulic Core Mockup - Studies

H. Leigh (954), and D. L. Ballard, P. M. Jackson, J. Spalek, G. W. Riedeman, L. R. Sweetin (COO)

a. Dispersion of Gas Bubbles

The one-third size hydraulic model of the FFTF piped inlet arrangement was operated to study the effects of large gas volumes entering the inlet chamber. Colored movies and black and white still photographs were taken to record observations on the efficiency of the plenum and basket assembly in dispersing gas bubbles. A slug of air (approximately 630 in.³) was dumped through one downcomer, with the inlet elbows at 22 1/2 degrees from axial position. This volume was well dispersed in the inlet plenum. The gas bubbles were broken up (diameter sized between 1/8 and 1/4 in.) as they passed into the basket and were uniformly distributed to all core flow channels. A second test condition involved bleeding a small but unmeasured air flow into one downcomer. Again, bubbles were quite uniformly distributed over the core.

b. Ultrasonic Measurements of Relative Air Content

The first on-line ultrasonic measurements of relative air content in the outlet model outlet pipe have been performed.

These measurements were done before on-line void fraction calibration data were available. However, the initial on-line measurements yielded the following important information:

- The success of the ultrasonic measurements of air content (in bubble form) in the 10-in. pipe of the outlet model, at flow rates up to 3600 gpm, confirmed experimentally the applicability of the ultrasonic pulse-transmission technique for on-line gas entrainment measurements.
- Hydraulically induced mechanical vibrations in the pipe holding the transducers have no adverse effect on the ultrasonic measurements.

Observed changes in the velocity of sound in water induced by changes in water temperature, also have no effect on the gas content measurements.

- Also, under the normal operating conditions with small and gradual temperature changes, the different thermal expansions of the Lucite and stainless steel transducer mounts induce no adverse results. However, in a system with fast, large temperature changes, thermal expansion would misalign the transducers so that remounting and recalibrating would be necessary.

c. Scale Model of Hydraulic Core Mockup

H. Leigh (954)

The contract for the HCM Scale Model with Atkins and Merrill, Inc., is in place and the initial meeting with them has been held. The contract for the model was based on a one-half scale model of a 17-ft diameter vessel, and the piping system and vessel support design are essentially complete with the required pipe on order. Because of schedule and cost constraints, the decision has been made to retain the model size of 8.5 ft diameter after the recent FTR concept diameter change from 17 to 20 ft, with the result that the model scale will be 0.425 to 1.0 rather than 0.5 to 1. The model designer

will proceed on this basis after BNW receives sufficient information from Westinghouse to proceed productively on the model design.

Overall design for the HCM facility is 80% complete and construction is 60% complete.

2. Sodium Flow Testing

a. Mark I Sodium Flow Test

T. J. Bennett (913)

The rough draft of the final report for the Mark I sodium flow test has been written. The purpose of this test was to evaluate the effect of sodium flow through a candidate (dispersed core) fuel bundle for the FFTF driver fuel. This bundle consisted of 217 pins, each 0.25 in. in diameter, and wrapped with 0.030-in. wire wrap on a 12-in. pitch. Both pins and cladding were made with Type 304 SS.

b. Mark II Sodium Flow Test

(1) Engineering Test Plan [T. J. Bennett (913)]. The Engineering Test Plan for the Mark II sodium flow test has been approved by BNW management and is being sent to Argonne National Laboratory for concurrence. The Mark II bundle simulates the driver fuel for the vertical core concept. The test plan calls for the 217-pin bundle (0.230-in. diameter pins wrapped on a 12-in. pitch with 0.056-in. wire wrap) to be tested for 9000 hr at 1100 to 1150 °F with a sodium flow rate of 525 gpm. Cladding and wrap are constructed of Type 316 SS.

(2) Subassembly Fabrication [D. E. Blahnik, R. B. Baker, and R. L. Myers (732)]. Development of a welding procedure for lower end cap closure welds was completed with satisfactory results for the modified lower end cap design. The operating procedure for welding the lower end cap was written. Approximately 75% of the lower end caps were welded.

An operating procedure for loading the pins was written and loading started. One hundred and fifty pins were loaded through the upper reflector, including the 40 pins to be fully characterized. Five pins were rejected because of the difficulty (pellet jamming) of using the specified small pellet-to-cladding gap.

A decision was made to expand the duct 0.054-in. beyond the theoretically tight bundle size. Using actual pin dimensions, the theoretically tight bundle measures 4.303-in. across flats. Therefore, the duct is being expanded to an inside flat to flat measurement of 4.357 in. Measurements of the wire and cladding OD were made to verify the cross flats bundle size. Duct nose-piece sections must be modified slightly for welding.

The lower spacing grid was redesigned to increase its strength for the projected test conditions. The new design requires fabrication of a new set of grid plates.

The design of the upper grid and upper end cap was not resolved.

c. Loose Wire Wrap 7-Pin Assembly

T. J. Bennett (913)

After a pump failure in the SCTL loop (Small Components Test Loop-BNW), the 7-pin bundle containing loose and broken wire wraps (three with 0.020-in. looseness, three with 0.030-in. looseness, and one broken wire wrap at the inlet end of the assembly) was transferred to the Heat Transfer Loop No. 1. To date this assembly has been exposed to about 1600 hr of 1150 °F sodium. The sodium flow velocity through this assembly is 31 feet per second. This 7-pin bundle will be removed from the HTL-1 loop after 2000 hr of testing, inspected, and reinstalled in the SCTL loop.

3. Bypass Flow Cooling of Driver Assemblies

D. C. Kolesar (832)

A modified version of the COBRA II code was used to predict the extent of driver duct cooling induced by bypass flow. The bypass flow is through the gaps between assemblies. The ducts were modeled by a slit approximation.⁽¹⁾ The power distribution was characteristic of assemblies located at the outer periphery of the core. The hydrodynamic effects of the wire spacers were not considered. The core flow rate and therefore core thermal balance was allowed to change slightly as the bypass flow was increased.

It was found that increasing the bypass flow rate may cause the duct temperatures to be reduced significantly. However, the temperature gradient through the wall is insensitive to this flow.

4. Fuel Performance Analysis

a. Dimensional Analyses of Typical FTR Fuel Pins

R. D. Leggett (750), E. O. Ballard (750),
D. C. Bullington (750), F. E. Bard (750), and
K. R. Merckx (A00)

An analysis of the expected deformations in 0.230 in. OD FTR fuel pins was completed using the most recent equations for cladding swelling for cold worked Type 316 SS, the latest estimates of pin powers and burnups in the FTR and the computer codes, SINTR, EXPAND, DEFORM, PECT-I and CYGRO. It was assumed that a 91% dense mixed oxide fuel pellet, 0.194 in. in diameter operated with an initial (cold) 6-mil diametral fuel-cladding gap at the following reference power levels:

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1. C. L. Wheeler, FFTF Monthly Informal Technical Progress Report, BNWL-990, Battelle-Northwest, Richland, Washington, January 1969.

- 4.8 kW/ft (minimum)
- 8.7 kW/ft (average)
- 12.2 kW/ft (maximum)
- 14.6 kW/ft (20% overpower)

An arbitrary incubation period of 4% BU was assumed before the fuel and cladding interacted and, the fuel was assumed to swell as a function of burnup rate and temperature. Maximum diameter increases of about 5.4% and 3.7% were predicted for the 20% overpower pin and low power pin, respectively, at 80,000 MWd/tonneM.

b. Hybrid Computer Simulation

R. D. Leggett (750), R. L. Fish (722),
and C. R. Cole (MOO)

Equations describing fission gas release, pressure buildup, 2-zone and 3-zone sintering and thermal flux radial power profiles were programmed into the hybrid computer and are in the final stages of debugging.

c. Additions to the SINTR Computer Code

E. O. Ballard (750)

SINTR (a simple fuel restructuring, one dimensional, steady-state heat transfer computer code) was revised to include calculations such as fission gas generation as a function of axial position, cladding fluence, and swelling.

C. IRRADIATION TEST FACILITIES

1. Closed Loops [4 MW Closed Loop Cell Size Assessment (AI)]

M. A. Vogel (924)

Atomics International under the Technical Direction of Westinghouse completed an assessment of the cell size required for a 4-MW special purpose loop. It was established that the Primary Heat Transport System cell should be 14 × 18 × 42 ft deep and the purification cell 14 × 14 × 12 ft deep. The

bases for this study were the requirements outlined in BNWL-500, Vol. 61, Conceptual System Design Description for the Closed Loops, except that the components were derated from 6 to 4 MW_t. The above cell dimensions were determined by three principal considerations:

- Number and size of components
- Space to accommodate bends to relieve pipe stresses
- Space to provide both completely remote and contact maintenance.

The study also recommended that a shield wall between the heat transport and purification cell is not required from the consideration of maintenance.

The above information was transmitted to Bechtel to serve as design bases for the closed loop structures within containment.

2. 2-MW Simple Unitized Modular Closed Loop (AI)

M. A. Vogel (924)

BNW provided testing requirements to Westinghouse to serve as a basis for developing the 2-MW loop concept. These are given in Table 6.A.II. Atomics International under the Technical Direction of Westinghouse completed an interim concept definition for a 2-MW simple unitized modular closed loop.

The loop concept definition is not yet complete. Simplification was attained by deleting some components, by relying on contact maintenance and by redefining components on the basis of the 2-MW heat dissipation requirement. The loop concept was unitized to the extent that the entire primary heat transport system package can be preassembled and installed as a unit. Modularization has been provided to permit remote removal of any one of five primary system modules from the cell in the event of major, nonremovable contamination.

TABLE 6.A.II. 2-MW Closed Loop Testing Requirements

<u>Test Conditions in Primary System</u>	<u>Value or Criteria</u>	<u>Remarks</u>
Maximum Na temperature leaving test	1400 °F	Achieved by internal bypassing and reducing to heat generation of test compatible with maximum flow capability
Maximum heat generation (test + structural)	2 MW	
Maximum Na hot leg temperature	1200 °F	
Maximum Na cold leg temperature	1150 °F	
Maximum Na flow rate	200 °F ΔT minimum at 2 MW (~ 290 gpm)	
Minimum Na flow rate	10% of maximum	To define control range
Maximum ΔT (hot leg versus cold leg)	400 °F	
Maximum test ΔP	100 psi at 200 °F ΔT (290 gpm)	To define design pump head higher head will be available at lower flow
Minimum cold leg temperature	600 °F at 2 MW and 400 °F ΔT (145 gpm)	Lower temperature permitted at less than 2 MW

3. Prototype Loop Test Planning

M. A. Vogel (924), and J. C. Moore (COO)

Test plans were prepared for the Prototype Closed Loop. These plans included objective, scope, justification, method and expected results for the following type of tests:

- Preoperational testing
- Steady state testing
- Transient testing
- Remote maintenance testing
- Accelerated life tests
- Decontamination testing

These test plans were based on the former 6-MW loop concept. Plans will be revised to reflect the 2-MW loop concept after concept definition is completed.

4. Technical Program Management

BNW-Westinghouse negotiations relative to scope of work and schedule for performing Prototype Closed Loop design under Contract BDR-619 are continuing.

BNW representatives participated in a Westinghouse review of Atomics International work on closed loop rescoping studies.

5. Closed Loop Mechanical Seal Closure Program

R. Kolowith (914), and M. R. Kreiter, E. F. Antal, S. K. Meyers, J. V. Thompson (COO)

The objectives of this task are to; (a) provide experimental test data verifying the principles of operation of a creep compensating connector assembly, (b) provide experimental test data for each of six specified seal concepts being considered for FFTF closed loop nozzle closure application, (c) establish a reference clamping device design and seal(s)

for subsequent reliability testing, and (d) provide technical guidance and evaluation in the development of a commercially available instrumentation connector.

a. Test on NDA 4-Point Seal

The first seal performance data has been recorded on this program. Proof pressure and leakage tests have been completed on one NDA 4-point seal. This seal concept was developed by Nuclear Development Associates and described in their report, Design Development Tests of Some Components for the 10 MW_e SDR, (Doc. No. NDA-84-21, May 25, 1969). Three seals with a cross section similar to that described by NDA, but larger in diameter were fabricated from Armco Ingot Iron. The load versus deflection curve at 18,000 lb loading indicated that the seal seated properly and that the spool was securely latched in the nozzle. The seal contained 375 psig helium with no structural deformation evident after the test.

A room temperature leak rate test was conducted. A leakage rate of 3.2×10^{-4} atm cm³/sec was measured at a pressure differential of 250 psi across the seal. There was no indication of clamp failure at any time during the test. Visual examination of the connector assembly after disassembly revealed no indications of connector failure.

A NDA report gives leak rates of 2×10^{-6} atm cm³/sec and 2×10^{-7} atm cm³/sec at 15 psi differential for two different seals. Extrapolation of BNW data gives a leak rate of 4×10^{-7} atm cm³/sec at that pressure.

b. Pressure Transient Test

Available funding forces indefinite suspension of preparation for the pressure transient test.

6. Closed Loop Tube Hardware Weld Closure Studies

J. W. Kolb (914), L. J. Rousseau (COO)

The design for the proof of principle prototype remote welder has been completed. The procurement of critical items for the assembly is 90% complete. Fabrication of the structural portion of the head assembly is in progress. Final modifications to the environmental test chamber have been completed. The chamber will be relocated to the BNW weld laboratory where proof of principle testing will be performed.

The automatic weld power supply and programmer has passed acceptance testing and is being prepared for weld testing.

7. Closed Loop Tube Remote Cutter

R. F. Gilmore (914)

An evaluation of development test results and reliability study has been completed for the selection of the proof of principle cutter test machine. The Side Mill Cutter remote machine has been chosen for fabrication, while development work on the Multiple Tooth tool concept will continue as a ready backup.

The design for the proof-of-principle machine is 65% complete. The procurement of critical long lead components is 80% complete. Demonstration tests of the remote cutter are scheduled for December 1969.

8. Closed Loop Meltdown

R. E. Keyes (812)

An analysis was made of the feasibility of retaining the melt of a 4-MW test in the bottom of the in-reactor test section. The results were included in Section 17 of the PSAR. The results revealed that for an 0.7-in. thick tantalum cup at the bottom of the test section, and an external flow of ~20 gpm, melt through did not occur.

D. OTHER CORE DESIGN

1. Advanced Core Studies

L. M. Finch, G. R. Waymire, and A. J. Anthony (810)

In conjunction with the core analysis section, an advanced core study has been completed. The report appraises the likely candidates for the second generation core of the FTR and defines the reactor components that are replaced during a second generation core installation.

Although a large number of potential fuels were examined as possible candidates, the prime incentives for advanced cores and state of the art ruled out all but three candidates for this initial study. Primary incentives included:

- Improved neutron flux
- Improved thermal performance
- Improved economics
- Improved availability.

Table 6.D.I. outlines the advanced core characteristics, while Table 6.D.II assesses the required component changeouts needed for each candidate fuel.

The characteristics of the full performance oxide and carbide cores are delineated in Tables 6.D.III and 6.D.IV, respectively. The fuel assemblies and tentative core maps for these concepts are shown conceptually in Figures 6.D.1 and 6.D.2.

The changes in the internal component dimensions which promulgate the changeouts noted in Table 6.D.II are shown in Figure 6.D.3.

TABLE 6.D.1.
ADVANCED FFTF CORE CHARACTERISTICS

CANDIDATE ADVANCED FUEL	INCENTIVE FOR USE	CONTRIBUTING CHARACTERISTIC CORE OR FUEL	ADVANCED DESIGN FEATURE	ADVANTAGE TO LMFBR PROGRAM
MIXED OXIDE (U-P)O ₂	GREATER NEUTRON FLUX FOR TESTS [0.9 x 10 ¹⁶ NEUTRON/ cm ² -SEC]	- HIGHER LINEAR POWER RATING-FUEL PIN - HIGHER CORE POWER DENSITY	- DECREASE LENGTH OF FUEL REGION - DECREASE LATTICE SIZE - DECREASE NUMBER OF FUEL ASSEMBLIES - INCREASED COOLANT VELOCITY - HIGH TEMPERATURE FUEL CLAD MATERIAL - IMPROVED FUEL CONDUCTIVITY - REDUCED OVERPOWER FACTOR - IMPROVED CORE INSTRUMENTATION	- IMPROVED BREEDING RATIO - HIGH POWER DENSITY
	SAFETY	- ASSURED CORE SUPPORT INTEGRITY - IMPROVED CORE INSTRUMENTATION	- SMALLER FUEL ASSEMBLIES - IMPROVED CORE SUPPORT - INCREASED SHIELDING FOR CORE SUPPORT STRUCTURE - IMPROVED INSTRUMENT "TREE" DESIGN - RADIATION RESISTANT MATERIALS - IMPROVED CONTROL ARRANGEMENT	IMPROVED SAFETY
	TEST ACCESSIBILITY	REDUCED HEAD AREA CONGESTION	- IMPROVED TEST LOOP ARRANGEMENT - REDUCED TEST LOOP SIZE	IMPROVED TEST DATA
	FUEL ECONOMY	IMPROVED FUEL BURNUP	- RADIATION DAMAGE RESISTANT CLAD AND STRUCTURE - REDUCED TEMP. AND STRAIN GRADIENTS - REDUCED METAL SWELLING - INCREASE LENGTH ABOVE CORE FOR EXTENDED FISSION GAS PLENA	IMPROVED FUEL CYCLE COST
	PLANT AVAILABILITY	- IMPROVED FUEL LIFE - HIGH INTEGRITY CORE COMPONENTS - IMPROVED CONTROL MATERIAL LIFE	- SEE ABOVE - INCREASED NUMBER OF REPLACEABLE COMPONENTS TO INCLUDE CORE SUPPORT AND RESTRAINT	- PLANT AVAILABILITY - ACCELERATED ACCUMULATION OF LMFBF DATA
	LMFBR PROTOTYPICALITY	EVOLUTIONARY IMPROVEMENT OVER INITIAL CORE FOR FFTF	HIGHER POWER DENSITY FUEL	PROTOTYPE OR DEMONSTRATION UNIT FUEL
MIXED CARBIDE (U-P)C	GREATER NEUTRON FLUX FOR TESTS [1.25 x 10 ¹⁶ NEUTRON/ cm ² -SEC]	- HIGHER LINEAR POWER RATING-FUEL PIN - HIGHER CORE POWER DENSITY - IMPROVED FUEL THERMAL CONDUCTIVITY	- DECREASE LENGTH OF FUEL REGION - DECREASE LATTICE SIZE - DECREASE NUMBER OF FUEL ASSEMBLIES - INCREASE COOLANT VELOCITY - HIGH TEMPERATURE FUEL CLAD MATERIAL - IMPROVED FUEL CONDUCTIVITY - IMPROVED CORE INSTRUMENTATION - REDUCED IN-CORE CONTROL - REDUCED COOLANT PRESSURE LOSS - VENTED FISSION GAS FUEL PIN DESIGN	- IMPROVED BREEDING RATIO - HIGH POWER DENSITY
	HIGHER COOLANT OUTLET TEMPERATURE (1100 °F)		- IMPROVED HIGH TEMPERATURE MATERIAL - IMPROVED CORE RESTRAINT - IMPROVED INSTRUMENTATION	HIGHER THERMODYNAMIC EFFICIENCY
	SAFETY	- ASSURED CORE SUPPORT INTEGRITY - IMPROVED CORE INSTRUMENTATION	- SMALLER FUEL ASSEMBLIES - IMPROVED CORE SUPPORT - INCREASED SHIELDING FOR CORE SUPPORT STRUCTURE - IMPROVED INSTRUMENT "TREE" DESIGN - RADIATION RESISTANT MATERIALS - IMPROVED CONTROL ARRANGEMENT	IMPROVED SAFETY
	TEST ACCESSIBILITY	REDUCED HEAD AREA CONGESTION	- REDUCED IN-CORE CONTROL - IMPROVED TEST LOOP ARRANGEMENT - REDUCED TEST LOOP SIZE	IMPROVED TEST DATA
	FUEL ECONOMY	IMPROVED FUEL BURNUP	- RADIATION DAMAGE RESISTANT CLAD AND STRUCTURE - REDUCED TEMPERATURE AND STRAIN GRADIENTS - REDUCED METAL SWELLING	IMPROVED FUEL CYCLE COST
	PLANT AVAILABILITY	- IMPROVED FUEL LIFE - HIGH INTEGRITY CORE COMPONENTS - IMPROVED CONTROL MATERIAL LIFE	- SEE ABOVE - INCREASED NUMBER OF REPLACEABLE COMPONENTS TO INCLUDE CORE SUPPORT AND RESTRAINT	- FUEL LIFE - LOWER COST FOR FULL CYCLE - PLANT AVAILABILITY - ACCELERATED ACCUMULATION OF LMFBF DATA
	LMFBR PROTOTYPICALITY	ALTERNATE FUEL CHOICE FOR LMFBF	HIGHER POWER DENSITY FUEL	PROTOTYPE OR DEMONSTRATION UNIT FUEL
LMFBR PROTOTYPE	FIND SOLUTION TO FUEL OR CORE PROBLEM	EXAMPLES: - POWER INSTABILITY - VIBRATION - COMPONENT DISTORTION - INSTRUMENT RESPONSE		
	DEMONSTRATE SATISFACTORY SOLUTION TO PROTOTYPE PROBLEM	EXAMPLES: - TRANSIENT POWER RESPONSE - MELTDOWN (SAFETY) - COOLANT BOILING (SAFETY) - THERMAL DISTORTION		

TABLE 6.D.II.
REACTOR CHANGES REQUIRED FOR ADVANCED CORES

ADVANCED DESIGN FEATURES	TYPICAL FOR ADVANCED CORE		REQUIRED CHANGE OF ORIGINAL REACTOR COMPONENTS									
	OXIDE	CARBIDE	FUEL	TEST LOOPS	CONTROL RODS	REFLECTORS	SHIELDS	CORE SUPPORT	CORE BARREL & RESTRAINT	INSTRUMENT TREE & PLUG	IVHM AND PLUG	REACTOR COVER
1. CHANGE CORE LENGTH	X	X	X	X	X	X	X	-	X	X	X	-
BELOW FUEL REGION	INCREASE	INCREASE										
FUEL REGION	DECREASE	DECREASE										
ABOVE FUEL REGION	INCREASE	DECREASE										
2. CHANGE CORE LATTICE	X	X	X	X	X	X	X	X	X	X	-	X
SIZE	DECREASE	DECREASE										
NUMBER	DECREASE	DECREASE										
3. CHANGE TEST LOOPS	X	X	-	X	-	-	-	X	-	X	-	X
ARRANGEMENT	X	X										
NUMBER	DECREASE	DECREASE										
SIZE	DECREASE	DECREASE										
KIND	X	X										
4. CHANGE NUCLEAR CONTROL	X	X	-	-	X	X	-	X	X	X	-	X
ARRANGEMENT	X	X										
NO. IN-CORE	DECREASE	DECREASE										
NO. EX-CORE												
SIZE	DECREASE	DECREASE										
KIND	X	X										
5. CHANGE CORE INSTR.	X	X	-	-	-	-	-	-	-	X	-	-
FLOWMETER	X	X										
TEMPERATURE	X	X										
FEDAL	X	X										
6. CHANGE CORE SUPPORT TECHNIQUE	X	X	-	-	-	X	-	X	X	X	-	X
RADIAL	X	X										
AXIAL	X	X										
MECHANISM	X	X										

TABLE 6.D.III.
FULL PERFORMANCE OXIDE FUELED CORE

<u>PARAMETER</u>	
FUEL TYPE	(Pu-U) ₂ O ₃ IMPROVED CONDUCTIVITY
BOND (FUEL/CLADDING)	GAS BOND
FISSION GAS DISPOSAL	5 ft PLENUM
TOTAL NUMBER LATTICE POSITIONS	91
CLOSED LOOP LOCATIONS	6
OPEN TESTS (CONTACT INSTRUMENTED)	2
OPEN TESTS (PROXIMITY INSTRUMENTED)	1
SAFETY RODS	3
IN-CORE CONTROL RODS	3
PERIPHERAL CONTROL RODS	15
DRIVERS	76
CORE VOLUME	900 LITERS
EFFECTIVE NUMBER OF DRIVERS (COUNTING OPEN LOOPS AT 75%)	78.75
BULK OUTLET TEMPERATURE	950 °F
HTS OPERATING POWER LEVEL	470 MW
PEAK FLUX	$9 \times 10^{15} \frac{n}{cm^2 SEC}$
PEAK ASSEMBLY POWER RATING	8.1 MW
CORE OUTLET TEMPERATURE	950 °F
ASSEMBLY BULK COOLANT ΔT, ORIFICED	350 °F
NUMBER FUEL PINS PER ASSEMBLY	217
FUEL PIN OD	0.236 in.
SPACER WIRE DIAMETER	0.040 in.
COOLANT VELOCITY IN HOT DUCT, ORIFICED	~35 FT/SEC
SUBASSEMBLY ΔP, INCLUDING ORIFICE	129 PSI
FLOW DUCT INSIDE DIMENSION	4.18 in.
FLOW DUCT OUTSIDE DIMENSION	4.48 in.
FUEL LENGTH	33.2 in.
PIN LENGTH	9 ft
PEAK LINEAR POWER INCLUDING 1.14 ENGINEERING FACTOR	18.4 kW/ft.
FUEL PIN ID	0.206 in.
PEAK HEAT FLUX	$1.1 \times 10^6 \frac{BTU}{HR-FT^2}$
PEAK CLAD TEMPERATURE (ORIFICED)	1200 °F
PEAK FUEL Q TEMPERATURE	4200 °F
CLADDING	TYPE 316 OR 348 SS MODIFIED FOR SWELLING RESISTANCE

TABLE 6.D.IV.
FULL PERFORMANCE CARBIDE FUELED CORE

<u>PARAMETER</u>	
FUEL TYPE	(Pu-U)C
BOND (FUEL/CLADDING)	GAS
FISSION GAS DISPOSAL	VENTED (DIVING BELL)
TOTAL NUMBER LATTICE POSITIONS	61
CLOSED LOOP LOCATIONS	5
OPEN TESTS (CONTACT INSTRUMENTED)	1
OPEN TESTS (PROXIMITY INSTRUMENTED)	3
SAFETY RODS	3
PERIPHERAL CONTROL RODS	12
DRIVERS	49
CORE VOLUME	590 LITERS
EFFECTIVE NUMBER OF DRIVERS (COUNTING OPEN LOOPS AT 75%)	51
HTS OPERATING POWER LEVEL	470 MW
PEAK FLUX	$1.25 \times 10^{16} \frac{n}{cm^2 SEC}$
PEAK ASSEMBLY POWER RATING	~11MW
CORE OUTLET TEMPERATURE	1100 °F
ASSEMBLY BULK COOLANT ΔT, ORIFICED	350 °F
ASSEMBLY BULK COOLANT ΔT, UNORIFICED	460 °F
NUMBER FUEL PINS PER ASSEMBLY	169
FUEL PIN OD	0.262 in.
SPACER WIRE DIAMETER	0.046 in.
COOLANT VELOCITY IN HOT DUCT, ORIFICED	48 FT/SEC
COOLANT VELOCITY IN HOT DUCT, UNORIFICED	37 FT/SEC
ASSEMBLY ΔP, INC. ORIFICED, UNORIFICED	110 PSI 80 PSI
FLOW DUCT INSIDE DIMENSION	4.120 in.
FLOW DUCT OUTSIDE DIMENSION	4.500 in.
FUEL LENGTH	32 in.
PIN LENGTH (VENTED PLENUM)	52 in.
OVERALL ASSEMBLY LENGTH	10 ft 10 in.
PEAK LINEAR POWER (91% DEPOSITION, 1.14 ENGINEERING FACTOR) AT OPERATING POWER	37 kW/ft
FUEL PIN ID	0.228 in.
PEAK HEAT FLUX	$2.2 \times 10^6 \frac{BTU}{HR-FT^2}$
PEAK CLADDING TEMPERATURE:	
1100 BULK OUTLET (ORIFICED)	~1300 °F
1100 BULK OUTLET (UNORIFICED)	~1450 °F
PEAK FUEL Q TEMPERATURE:	
1100 BULK OUTLET (ORIFICED)	3270 °F
1100 BULK OUTLET (UNORIFICED)	3360 °F

FIGURE 6.D.1. Advanced Core Study Oxide Fuel

SK-3-14750

FIGURE 6.D.2. Advanced Core Study Carbide Fuel (Gas Bonded, Vented)

SK-3-14751

FIGURE 6.D.3. Advanced Core Study Required Reactor Changes

SK-3-14752

CHAPTER VII. FUELS AND MATERIALS EVALUATION

(See Appendix A for List of Author Codes Used in This Report.)

A. FUELS AND MATERIALS EVALUATION1. BNW-1 Series Capsule Irradiationsa. Status

J. E. Hanson (750)

The status of the BNW-1 capsule series is summarized in Table 7.A.I. Two capsules continue their irradiation toward a goal exposure of 100,000 MWd/tonneM.

b. Compilation of Fabrication and Irradiation Histories

J. W. Weber and D. W. Dragnich (750)

Axial power distributions were calculated based on burnup analyses and gross, or $^{95}\text{Zr}/^{95}\text{Nb}$ axial gamma scans for BNW-1-4, 1-6, 1-9 and 1-11. These data are being compared to axial power distributions derived from thermocouple data. This comparison showed that the power derived from the thermocouple data is from 5 to 20% less than the power calculated from burnup data. Work is being continued in the reduction of thermocouple data and axial gamma scans to obtain a better correlation.

c. Gamma Scanning of Irradiated Pins

J. W. Weber (750), D. E. Christensen (COO), and
C. F. Beyer (COO)

Axial gamma scans for gross activity and for $^{95}\text{Zr}/^{95}\text{Nb}$ have shown several activity peaks on BNW-1-1, 1-2, 1-3 and 1-5. Spectra data collected along the length of the fuel pins, including several directly on the activity peaks, are being analyzed with a computer program GSSLRN (Gaussian - Learn) to determine the major gamma-emitting isotopes present.

TABLE 7.A.I. Summary Status of Irradiation Experiments Conducted Under Task FP-6

Issue Date: 10/8/69

Test Designation	Description ⁽¹⁾ Fuel Form	O/M	Facility	Goal Exposure MWd/tonneM	Status		
					Actual Exposure MWd/tonneM	Discharge Date	Status
BNW 1-1	Solid	1.93	MTR	100,000	(Est) 120,000	2-69	Being Exam. at BNW
BNW 1-2	Solid	1.93	ETR	50,000	(Est) 70,000	2-69	Being Exam. at BNW
BNW 1-3	Solid	1.96	MTR	100,000	(Est) 150,000	2-69	Being Exam. at BNW
BNW 1-4	Solid	1.96	MTR	50,000	71,000	6-68	Exam. Completed at BNW
BNW 1-5	Solid	2.00	MTR	100,000	(Est) 160,000	2-69	Being Exam. at BNW
BNW 1-6	Solid	2.00	MTR	50,000	75,200	10-68	Exam. Completed at BNW
BNW 1-7	Annular	1.93	MTR	50,000	<1,000		Lead Tube Failed. Exam. at BNW
BNW 1-8	Annular	1.93	MTR	100,000	29,000	1-68	Failed. Exam. at BNW
BNW 1-9	Annular	1.96	MTR	50,000	81,800	6-68	Exam. Completed at LASL
BNW 1-10	Annular	1.96	ETR	100,000	(Est) 86,000	(Est) 2-70	Discharged 2-69 for Interim Exam. Recharged 5/69
BNW 1-11	Annular	2.00	MTR	50,000	87,400	8-68	Exam. Completed at LASL
BNW 1-12	Annular	2.00	ETR	100,000	(Est) 91,000	(Est) 2-70	Discharged 2-69 for Interim Exam. Recharged 5/69

1. Fuel

25% PuO₂-75% UO₂
 Pellets 0.212 in. OD (0.052 in. ID for annular pellets)
 Smeared density - 87.5% TD
 Pellet density - solid pellet 93% TD, annular pellet 95% TD
 Fuel column length 13.5 in.

Cladding

OD - 0.250 in.
 Wall thickness - 0.016 in.
 Type 304 SS

Analysis of the BNW-1-1 spectra data by the computer program indicates that the fission product ruthenium is the major contributor to the activity peaks except for one peak at the top of the fuel column which is produced by cesium.

These analyses are being continued.

d. Postirradiation Examination of Fuel Pins

J. W. Weber (750), D. E. Christensen (COO), and
C. F. Beyer (COO)

BNW-1-1, 1-2, 1-3 and 1-5 were drilled in the plenum sections for measurement of total plenum gas. Samples of this gas were analyzed for composition. Results of the gas measurements and analyses are shown in Table 7.A.II. Calculated fractional fission gas releases are included. Calculation of the theoretically generated fission gas is based on the estimated burnups. There is reason to believe that the estimated burnup of BNW-1-3 is in error in that it is too high. Accurate burnup measurements will be obtained by chemical analysis.

2. BNW Series Fuel Pins

P. K. McDaniel (700)

On October 2, 1969 subassembly X-018B containing BNW Pin No. BNW B-40 was loaded into position 4E2 of EBR-II reactor grid. This pin contained specimens to evaluate the effects of metallurgical variables on irradiation induced swelling. The estimated discharge data for BNW Pin No. BNW B-40 is October, 1970.

The four BNW Pins (35, 36, 37 and 38) from subassembly X-067 were disassembled. The specimens are being tested. Flux monitors are at the counting laboratory at Idaho Nuclear Corporation waiting to be counted.

TABLE 7.A.II. *Measurement and Analysis of Fission Gas from BNW-1-1, 1-2, 1-3 and 1-5*

BNW	Volume of Plenum Gas Collected (ml-STP)	Analysis		Volume of Fission Gas Collected (ml-STP)	Burnup (est.) MWd/tonneM	Volume of Fission Gas Theoretically Generated	% of Fission Gas Released
		%Xe	%Kr				
1-1	184	91.18	5.28	177.5	120,000	241	73.6
1-2	138	90.60	5.25	132.3	70,000	142	93.0
1-3	166	90.80	5.43	159.7	150,000	316	50.5
1-5	208	91.34	5.12	200.6	160,000	326	63.0

3. PNL-1 Series Fuel Pins

a. Mechanical Testing

R. L. Fish (722)

The PNL-1 series pins were irradiated in the EBR-II reactor to a peak burnup of approximately 9000 MWd/tonne. These pins were clad with annealed Type 304 SS (E-1ot).

As reported earlier, the burst pressures on specimens taken from the upper portion of PNL 1-3 and 1-6 fuel pins were much less than that of the controls, plenum, or sections taken from the lower portion of the same pins. The corresponding ductility was also reduced greatly for the upper sections. The reduced strength and ductility was the result of severe intergranular attack by some constituent of the irradiated fuel activated by the applied stress.

b. Shielded Electron Microprobe Examination

E. D. Jenson (A00)

Efforts to identify the corrosive agent causing the failure by using a shielded electron microprobe have produced inconclusive results.

A section of fuel pin PNL 1-6-F was re-examined in the shielded electron microprobe. The section was prepared using a hydrogenated terephenyl liquid phase with a Freon rinse between each grinding and polishing step.

In an area of cladding-fuel interface the elements U, Pu, Cs, I, Ba, and Te were found in a layer adhering to the cladding. Cladding elements were found as metallic deposits in the fuel matrix and along the walls of a large crack. The composition of the particles in the crack is plotted against distance from the cladding in Figure 7.A.1. Chromium is seen to decrease, while nickel increases with distance from the cladding. Iron is shown as two plots, both decreasing with distance from the cladding. (The two separate segments probably are caused by the electron beam penetrating the smaller particles.)

To eliminate beam penetration effects, the ratio of iron to nickel, and also of chromium to nickel, were obtained as shown in Figure 7.A.2; both plots are fitted reasonably well by a straight line. These plots can be interpreted as indicating a decrease in the rate of deposition of iron and chromium with distance from the cladding (increasing temperature) and an increase in the rate of nickel deposition with increasing temperature (distance). Other deposits, not in visible cracks, were found and their composition and distance from cladding is given in Table 7.A.III.

4. PNL-59 Series Fuel Pins

G. E. Culley (750)

a. Mechanical Tests

R. L. Fish (722)

The PNL-59 series fuel pins were irradiated in the GETR thermal reactor to a peak burnup of approximately 69,000 MWd per tonne. The pins were clad with annealed Type 304 SS

FIGURE 7.A.1. Iron, Nickel, and Chromium Composition of Deposits in Fuel Crack Versus Distance from Cladding

FIGURE 7.A.2. Iron-to-Nickel and Chromium-to-Nickel Ratios in Deposits in Fuel Crack Versus Distance from Cladding

TABLE 7.A.III. *Iron, Nickel, and Chromium Content of Metallic Deposits Not in Cracks in Fuel Pin PNL 1-6-F.*

Particle Number	Distance from Clad. (microns)	Percent Composition		
		Fe	Cr	Ni
1	660	29.4	0.7	2.2
2	300	63.5	0.8	20.8
2	475	43.4	1.1	22.8
2	650	42.1	1.1	24.8
2	800	38.3	1.0	22.9
3	875	32.7	0.4	25.8

(F-1ot). The total peak fluence was 2.8×10^{20} n/cm², (E > 0.18 MeV). Temperature and fluence values for two specimens burst tested from the PNL 59-7 pin at 900 and 1450 °F are given in Table 7.A.IV. Post-test strain measurements are also included in this table. As reported previously, these burst pressures are comparable to that of the control specimen at 900 °F but somewhat lower than that of the control at 1450 °F.

TABLE 7.A.IV. *Irradiation History and Postirradiation Properties of Specimens from PNL-59 Series Fuel Pins.*

Specimen Identification	Mean Cladding Temperature	Cladding Fluence	Test Temp	Burst Pressure	Hoop Stress	Strain %ΔD/D
59-7: L	730 °F	2.7×10^{20} n/cm ² (E > 0.18 MeV)	1450 °F	3580 psi	26,280 psi	12.4%
59-7: E	640 °F	2.35×10^{20} n/cm ² (E > 0.18 MeV)	900 °F	8920 psi	65,470 psi	11.8%

b. Fuel Pin Power

G. E. Culley (750)

Analysis of the thermocouple data collected during the irradiations of the twelve PNL-59 capsules is in progress. Fuel pin power as a function of time and accumulated burnup was obtained. Table 7.A.V presents the results.

TABLE 7.A.V. *GETR Capsule Calculated Performance*

<u>Capsule</u>	<u>Axial Peak Power (kW/ft)</u>	<u>Burnup (MWd/tonneM)</u>	
		<u>Axial Peak</u>	<u>Axial Avg.</u>
PNL-59-1	13.87	18,617	13,810
PNL-59-2	14.07	18,537	13,750
PNL-59-3	14.56	19,277	14,298
PNL-59-4	14.16	18,647	13,832
PNL-59-5	13.64	18,505	13,653
PNL-59-6	14.21	19,544	14,423
PNL-59-7	13.52	68,803	51,376
PNL-59-8	13.67	69,440	51,811
PNL-59-9	13.99	72,040	54,117
PNL-59-10	13.65	69,823	52,432
PNL-59-11	14.73	68,184	20,904
PNL-59-12	13.26	25,992	19,412

5. Subassembly PNL-2

L. A. Pember (750)

The PNL-2 (X032) subassembly was discharged from the EBR-II August 31, 1969, at the end of Run 37. The estimated exposure on this subassembly at discharge is 47,000 MWd/tonneM. The subassembly was disassembled at EBR-II without incident. Neutron radiography performed at TREAT indicated that all pins were intact. The only unusual items visible on the neutron radiographs were what appears to be a small amount of fuel powder in the gas annulus around the insulator pellets on two pins and one of the pins with 0.010 in. thick cladding showed appreciable bowing. The pins from this subassembly will be shipped to BNW for postirradiation examination.

6. Subassemblies PNL-3, 4 and 5

L. A. Pember and E. O. Ballard (750)

Three 37 unencapsulated pin experimental subassemblies (25 wt% PuO_2 -75 wt% UO_2 , Type 304 SS cladding, 0.250 in. OD) in EBR-II are operating well. These subassemblies had attained the following estimated burnup at the end of EBR-II Run 37 (August 31, 1969).

<u>Subassembly</u>	<u>Burnup, MWd/tonneM</u>
PNL-3 (X051)	11,700
PNL-4 (X059)	14,700
PNL-5 (X054)	25,200

The subassemblies were removed from the reactor core during the first part of EBR-II Run 38 by reason of the increased power testing which took place during this period. Following the 62.5 MW test run the subassemblies were returned to their original core positions for further irradiation.

7. Subassemblies PNL-6, 7 and 8

L. A. Pember and E. O. Ballard (750)

Subassembly PNL-7 was assembled for insertion into EBR-II at the end of Run 38 A. This subassembly contains 37 unencapsulated pins clad in 0.250 in. OD solution treated Type 316 SS. The fuel is 25 wt% PuO_2 -75 wt% UO_2 . The UO_2 is 45% enriched in ^{235}U . Following assembly the subassembly was neutron radiographed in three orientations at TREAT to provide a record of the assembled condition and to demonstrate the feasibility of neutron radiography of a 37-pin subassembly. The cladding which was used for the fabrication of PNL-6, 7 and 8 was all on the high side of the tolerance (0.250 ± 0.001 in.) This stacking of tolerance caused the pin bundles to be too large to fit a standard Mark H-37 flow tube and still allow the nominal 0.010-in. gap between the assembly and the flow duct. However, a slightly oversized hex tube was found and used for the PNL-7

experiment. Additional, but slightly less oversized hex tubes were also found for the PNL-6 and 8 subassemblies. In these cases the 8-mil clearance will not have an affect on the goal burnup of the subassemblies from a pin bundle-hex can interference standpoint.

8. (PNL-17) Instrumented Subassembly

M. K. Millhollen (750)

ANL forwarded to BNW complete sets of drawings for the conduit pin, the assembly grid, and the entire subassembly. The prints were checked in detail and marked prints were returned to ANL. Only one major problem, length of thermocouple above the fuel pin, was found. This was brought to ANL's attention and they have agreed that the dimension they specified can be changed.

The flow test on the dummy 37-pin PNL-17 subassembly was completed by ANL. ANL reported a pressure drop of 8 psi across the pin bundle at a flow of 53 gpm which compares favorably with the BNW-calculated value of 7.9 psi.

9. Transient Irradiations

a. GE/VNC Capsules

G. E. Culley (750)

The five TREAT capsules fabricated by GE/VNC underwent qualification testing. Three of the five capsules failed the 15,000 psi hydrostatic test. The leak appeared at the epoxy seal in the head block and indications are that the leaks are in the epoxy resin rather than at the bond between the epoxy resin and the head block. A cost estimate for the necessary repairs is being generated by GE/VNC.

b. Computation Methods

G. E. Culley (750)

Significant progress was made in the capability to perform neutronics calculations of the fuel pins to be transient tested. These fuels contain a large portion of fissile material and a significant portion of the power is produced from nonthermal neutron-induced fissions. Thus, in addition to determining the spatial power distribution resulting from thermal neutron induced fissions, the ratio of nonthermal to thermal neutron induced fissions must be determined. The computer codes HRG and BATTELLE-REVISED-THERMOS, are being used to compute spectrum average cross sections for use in the DTF-IV code which calculates the spatial power distribution in the fuel. The net result is a much flatter radial power profile than is calculated when only the thermal fissions are considered.

c. Remote Re-encapsulation

D. O. Sheppard (750)

A remote re-encapsulation of a dummy EBR-II type fuel pin was successfully performed, demonstrating several untried systems and procedures.

10. Biaxial Stress-Rupture Studies for Type 316 SS Cladding

R. W. Barker (722)

Biaxial stress-rupture tests on 0.208 in. diameter x 0.008 in. wall, Type 316 SS fuel cladding in the as-received condition were begun at 900 °F. At this temperature the burst stress was 55,000 psi. A biaxial stress-rupture test was begun at 900 °F with a hoop stress of 52,000 psi. This test has now accumulated more than 1400 hr without specimen failure. It seems apparent that creep mechanisms are not operable or rates are extremely low in this temperature region (ϵ 0.46 T_m).

Therefore additional stress-rupture tests on as-received Type 316 SS at 900 °F will not be started at this time. However, the present 900 °F test is functioning well and will be continued.

It was determined that the burst stress at 900 °F for Type 316 SS irradiated to 1×10^{22} n/cm² ($E > 0.1$ MeV) at 800 °F is 60,000 psi. Biaxial stress-rupture tests of irradiated Type 316 SS will be performed at 900 °F in sufficient quantity to determine if creep mechanisms will be operable at stresses higher than that required to burst as-received material.

11. Damage Analysis

J. L. Straalsund (722) and H. R. Brager (A00)

The objectives of this effort are to establish the irradiation-induced swelling characteristics of FFTF alloys, to relate fast reactor induced submicrostructural changes to corresponding changes in mechanical properties, and to develop means for data extrapolation and interpolation, using both microscopic and empirical approaches.

Three dimensional graphs were constructed to relate void size and number density to fluence and temperature for nominally unstressed solution treated Type 304 and 316 SS. The three dimensional plots provide a better perspective of the trends in microstructural features with fluence and temperature. Tentatively, these plots show that:

- The number of voids produced increases with increasing fluence and decreasing temperature.
- The void size increases with increasing irradiation temperature, but for the temperature range between 370 and 450 °C and for fluences between 0.5 and 6×10^{22} n/cm² the void size appears to be independent of fluence.

The above conclusions suggest that the average void size is quickly stabilized at some limiting size which depends only on temperature. These trends need to be substantiated and clarified by obtaining more data in the high temperature and low fluence ranges. The data suggest that the fluence dependency of swelling is caused primarily by the fluence dependency of nucleation and that the temperature dependence of swelling is associated with both nucleation and growth. Efforts will be made during the next month to put these comparisons on a more quantitative basis by fitting equations to the void size and density data and comparing these with equations describing the swelling or total void volume.

In addition to aiding in the identification of trends in the microscopy data, the three dimensional plots indicate which irradiation conditions should be investigated to substantiate or clarify the indicated relationships. Thus emphasis was shifted to include examination of samples irradiated in the lower range of fast fluence, from 1×10^{21} to 1×10^{22} n/cm² (E > 0.1 MeV) and from 370 to 620 °C.

B. MATERIALS TECHNOLOGY

1. Uniaxial Creep-Rupture Studies*

J. F. Bates and A. J. Lovell (A00)

Preliminary attempts to correlate creep-rupture data of unirradiated solution treated (1950 °F/1 hr) Type 304 and 316 SS have shown that the expression $\dot{\epsilon} = A[\sinh(\alpha\sigma)]^n e^{-Q/RT}$ where A, α , n, and R are constant can be used.

$\dot{\epsilon}$ = minimum creep rate (in./in./hr)

σ = applied engineering stress (psi)

T = temperature of test (°K)

Q = activation energy (cal/mole).

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* Reported by J. C. Tobin (960)

Studies are now in progress to evaluate in more detail the significance of this relationship. A more detailed report will be forthcoming upon conclusion of these studies.

Data on the effect of irradiation on Types 304 and 316 solution treated (1950 °F/1 hr) SS specimens show that the reduction in rupture life and decrease in creep rate for the Type 304 irradiated specimens, as compared to the solution treated control, is roughly the same for total fluence of both 1×10^{22} and 4.2×10^{22} n/cm². However, the Type 316 specimens tested thus far show a reduction in rupture life and an increase in creep rate that is apparently strongly dependent upon the irradiation conditions. Further tests, which should aid in the understanding of this observation, are in progress.

Weldment specimens have been irradiated in EBR-II to a fluence of $\sim 9 \times 10^{21}$ n/cm² and are being prepared for creep-rupture testing in the hot cell in 321A Building. Specimens of the following types of weld processes are being prepared:

(1) Sub-Arc, (2) Stick Electrode, (3) TIG, and (4) MIG.

2. High Strain Rate Effects*

J. M. Steichen (A00)

Several high strain rate tests on 304 SS have been completed. These tests were performed at a strain rate of 10 in./in./sec at test temperatures from room temperature to 1400 °F. The preliminary data gave the following results:

- A comparison of strain rate effects on strength of Type 304 SS at temperature shows that ultimate strength is affected more than yield strength. At test temperatures of 800 °F and above, the highest values for the ultimate strength are obtained at a strain rate of 0.1 sec^{-1} . As the strain rate is increased to 1.0 and 10.0 sec^{-1} , a drop is observed in the ultimate strength.

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* Reported by J. C. Tobin (960)

- Uniform and total elongation of Type 304 SS show little sensitivity to strain rate change other than to suggest slight ductility decreases with increasing rates.
- If strength values are plotted against strain rate, the room temperature ultimate and yield strengths are observed to increase as strain rate increases. The ultimate strength appears to increase about 1500 psi for each order of magnitude increase in strain rate, while the yield strength appears to increase about 2500 psi for each order of magnitude increase in strain rate. At 1000 °F, the ultimate strength increases as strain rate is increased to 0.1 sec^{-1} . At rates above 0.1 sec^{-1} the ultimate strength decreases significantly. The yield strength at 1000 °F decreases only slightly at the higher rates.

3. Radiation Effects on Absorbing Materials for Control Rods*

A. L. Pitner (A00)

Post-irradiation analysis is continuing on the boron carbide samples from the first thermal reactor subassembly discharge. Five forms of B_4C are being investigated: 60 and 80% TD powder, and 65, 90, and 99% TD pellets. Burnup values were obtained by mixing each powder sample thoroughly (because of the high surface reaction rate), and performing mass spectrographic analyses of boron isotopic ratios. Temperatures were assigned to the fifteen samples in accordance with those inferred from five thermocoupled capsules.

In general, the results show that gas release increases with increasing irradiation temperature. The 99% TD pellets appear to retain the helium gas more than the other sample forms.

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* Reported by J. C. Tobin (960)

The visual appearance of the pellet samples showed significant deterioration from the irradiation. This was particularly true for the 99% TD pellets. These pellets swelled during irradiation and had to be forced out of their holders. Most of the pellets had broken into smaller pieces. Corners were rounded and cracks were prominent. The surface of the 99% TD pellets was quite powdery, while the surfaces of the lower density pellet, chipped easily. Densities of several of the pellet particles were measured by immersion weighing. After irradiation, the pellets all showed apparent densities near the theoretical value (2.52 g/cm^3), indicating that the porosity and cracks generated during irradiation are easily accessible to water.

Two more subassemblies are continuing under irradiation at similar temperatures, and will be discharged to yield data at ~ 10 and $15\% \text{ B}^{10}$ burnup. In addition, another capsule assembly operating at temperatures from 400 to 700°F is being irradiated to similar burnup levels.

The X042-A subassembly was removed from the EBR-II after $\sim 3\% \text{ B}^{10}$ burnup and $\sim 15 \times 10^{21}$ nvt tantalum exposure. These values are approximately double that attained before the first interim discharge. Neutron-radiography measurements showed that the poison column lengths of the two capsules BC-2 and BC-5 have returned to their original unirradiated values after showing apparent growths of 0.51% and 0.26% respectively, at the time of the first interim discharge. Capsules BC-3 and BC-8, which were charged after the first interim examination, showed column length changes of $+0.52\%$ and -0.51% . Accuracy of the measurements is considered to be $\pm 0.26\%$ ($1/64$ inch). Neutron-radiography measurements of a tantalum sample that has been irradiated since the original test insertion showed a length change of $+0.25\%$. This is approximately double the growth seen at the first interim discharge.

4. B₄C - Sodium Compatibility*

L. R. Bunnell (AOO)

Analysis by spark-source mass spectroscopy of Type 316 SS sodium-bonded to B₄C pellets and heat treated at 300 °C and 500 °C for 3450 hr showed decreasing boron and carbon transport with increased pellet density, as would be expected simply on the basis of surface area considerations. Little or no temperature dependence is evident in the data.

B₄C pellets of three densities irradiated in a thermal reactor to a nominal burnup of 3% have recently been received. The pellets swelled inside the stainless steel container and had to be pressed out. One pellet was left in place and an attempt will be made to polish the sample and thus observe a possible reaction product microstructure at the interface between B₄C and the Type 304 SS. Metallography was performed on the as-received B₄C pellets to provide a comparison for later work on the irradiated pellets. The two lower-density pellets had a uniform microstructure and it is evident that a coarse hot-press feed was used. Metallic particles observed contained iron and chromium, and are probably stainless steel introduced during grinding. Open and closed porosity was obtained by a water impregnation method.

Metallography is now being performed on the irradiated pellets. X-ray analysis of some of the "crust" of high burnup material on external surfaces showed B₄C peaks, plus weak peaks which could not be ascribed to lithium boride or lithium carbide.

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* Reported by J. C. Tobin (960)

CHAPTER VIII. FUELS RECYCLE

(See Appendix A for List of Author Codes Used in this Report.)

A. FUEL TECHNOLOGY1. FFTF Fuel Procurement

J. F. Williams (712)

The progress of the LMFBR fuel supplier prequalification contractors is as follows:

Babcock and Wilcox completed the blending of mixed oxides into five process batches and added binder to all batches. The first batch of pellets was pressed and started through sintering.

NUMEC processed the PuO_2 to the point where they are ready to start mixing the two oxides. Release to blend a small batch of mixed oxides for sintering tests was granted, based on a preliminary review of procedures submitted to date.

United Nuclear Corporation was released to blend the UO_2 and PuO_2 . This is currently in progress. Fabrication of metal components through a subcontractor was started. UO_2 insulator pellets were completed and are undergoing final inspection.

NUMEC, UNC, and Babcock and Wilcox reported some quality problems with tubing. NUMEC indicated a problem with burrs on the end of the tubing plus the presence of some unidentified foreign matter in the bore. UNC reported burrs on the end of tubing, and some internal contamination. Babcock and Wilcox reported some contamination on the interior surface of tubes. Babcock and Wilcox accepted sufficient tubes to complete the required number of pins. Replacement tubing will be supplied to NUMEC and UNC when BNW inspection is complete.

2. LMFBR Fuel Development - Offsite Technical Direction

B. R. Hayward (711)

Test sintering batches of mixed oxide powders under NUMEC Task 7 yielded pellets with low oxygen-to-metal ratio and high nitrogen (1.93 and 1200 ppm, respectively). NUMEC subsequently modified the sintering equipment to provide a controlled moisture addition.

NUMEC is using methylcellulose additions for pellet density control. The test batches for density control yielded erratic results. Larger test batches are planned in an attempt to solve this problem.

Subassembly X012 containing 19 NUMEC mixed oxide encapsulated pins was removed from EBR-II and disassembled for interim examination. The average burnup was 87,000 MWd/tonne (9.3 at.%). Neutron radiographs of the pins indicated no unusual conditions or failed pins. Minor evidence of cladding ridging was observed. Copies from these neutron radiographs will be submitted to Philco-Ford for image enhancement evaluation. It is intended that some of these pins will be reinserted for burnup to 100,000 MWd/tonne or, possibly, 150,000 MWd/tonne.

The Phase II analytical chemistry results from AI, Babcock and Wilcox, United Nuclear, General Electric Company, BNW, and NUMEC are now being compiled and evaluated at both BNW and LASL. NUMEC submitted only partial results because of analytical equipment problems. LASL has completed the work but has not yet submitted the report. The compilation of results should prove interesting since a cursory examination indicates discrepancies among certain analyses.

3. Development and Control of Pellet Homogeneity - Inspection and Testing

D. A. Stranik (740)

A method for quantitative measurement of fuel pellet homogeneity has been developed which in turn has resulted in the development of an improved homogeneity specification.

Quantitative characterization of the fuel pellet matrix was accomplished by scanning the pellets with the electron microprobe for PuO_2 concentration, sizes of PuO_2 enriched or depleted zones, and separation distance of such zones. This technique enabled a mathematical model to be developed relating fuel pellet characteristics to a figure of merit which describes the homogeneity of mixed oxide fuel in terms of the degree of uniformity of energy deposition (or temperature rise) under transient conditions in the UO_2 content of the fuel.

A mean value of the figure of merit, $\bar{M}$, equal to or greater than 0.8 was established by FFTF Project Nuclear Analysis Section as adequate to meet FTR fuel homogeneity requirements.

The microprobe data from three $(\text{Pu}_{.25}\text{U}_{.75})\text{O}_2$ fuel pellets with three different degrees of inhomogeneity were analyzed according to the homogeneity method model given above. Pellet 8131 was a typical irradiation test pellet used in PNL-3 fuel pins. Pellet 956 was a pellet with less ballmilling of the powder and was used to break in new processing equipment. Pellet 8235 was a pellet in which the powder was "V" blended only.

The 99% confidence limit for $\bar{M}$ for the three pellets indicates that all three are significantly greater than $\bar{M} = 0.8$ which has been established as the lower limit of acceptance.

B. CLADDING TECHNOLOGY

1. Vendor Qualification

R. C. Aungst and J. C. Tverberg (721)

All comments from the various LMFBR Contractors on the Fuel Cladding Product Description A-0086 were received and evaluated. Perhaps the most persistent criticism was directed at the inclusion of product, process, and test method specifications within a single document. All of these comments will be considered when the revised product description is rewritten and issued as the controlling document for the Vendor Qualification Program.

2. LMFBR Fuel and Cladding Center

F. R. Shober (723)

The LMFBR Fuel and Cladding Information Center is designed for computer storage and retrieval of data pertaining to the complete material, fabrication, irradiation, and postirradiation histories of LMFBR type fuel pins. The data fields frequently in place or being programmed are cladding fabrication, fuel fabrication, fuel pin fabrication, fuel subassembly fabrication, irradiation history, and postirradiation examination. Under consideration are two complementary sections to the system, one pertaining to document references and another related to materials and process specifications. The reference section will provide the necessary documentation for summary reports to be published by the center, and the specifications section will provide historical continuity between manufacturing processes and product performance. The latter will be tied closely with quality assurance activities.

The information and fields for fuel pin data were arranged in an hierarchical order within their respective sections to maintain compatibility in the system and to permit selective

accessibility and storage of data. The program format and data input forms are also being prepared for the fuel pin section.

Entries made in the LMFBR file include data from 107 tensile tests in the austenitic stainless steel file and data for 18 tensile tests from N-1 lot of tubing in the cladding characterization file. This brings their respective totals to 2,323 and 231. Data from 30 creep tests were added to the creep data file. Plots made of the creep data to test the system indicated the need for only minor changes in the program. Further check of the program will be made by retrieving the data from these 30 creep tests and plotting their strain versus time graphs. The files now contain the results of 6541 tests, with most of them being tensile tests.

C. FUEL PIN TECHNOLOGY

1. NDT Systems Development - Ultrasonic Test for Fuel Pin Welds J. L. Thompson (MOO), J. Ryden (MOO), R. M. Crawford (733), and A. C. Callen (740)

Five sample fuel pin end closure welds were previously made to achieve some degree of lack of penetration. The initial ultrasonic examination indicated several possible areas with less than full weld penetration. Radiographic examination of the sample welds indicates that the ultrasonic test detected tiny pockets at the tube ID-cap interface. Metallography of the samples will be made to confirm the nature and size of the indicated defects.

2. Cladding Test Round Robin W. F. Stevenson (740)

Twelve cladding tubes of Type 316 SS 0.230 in. OD, 0.015 in. wall thickness by 5 ft long have been shipped to United Nuclear Corporation for the purpose of qualifying a subcontractor for performing ultrasonic testing of cladding

tubes on subsequent contracts. These same tubes are to be tested by those other organizations participating in the pre-qualification program. Data from these tests will be analyzed to determine precision and accuracy.

3. Irradiation Test Pin Fabrication

R. M. Crawford (733)

Twenty fuel pins for EBR-II subassembly PNL-6 were loaded, xenon tagged and welded. Sixteen additional pins are loaded and ready for tagging and welding.

4. Fuel Pin End Closure Development

R. M. Crawford (733)

To date, approximately 2100 fuel pin end closures using the reference FTR weld design have been welded and radiographed. About 5% of these welds were examined metallographically. Metallographic examination revealed no evidence of any condition that would cause the weld to be rejected in any of the 81 welds examined.

Radiographs of all the pin closure welds for irradiation in EBR-II were examined and accepted by the quality assurance group at EBR-II.

Approximately 50% of the welds produced to date have had a build-up across the diameter of 0.001-in. or less and 95% of the welds have had a build-up on the diameter of 0.0025-in. or less. While 37 of the 2100 welds made were rejected for incomplete (less than 100%) penetration based on radiography, none of the 81 welds submitted for metallography showed incomplete penetration.

The following is a summary of the welds made to date:

Irradiation Test Pins for PNL-3, 4, 5 - 0.250 in. OD -
Type 304 SS

Total Welds - 376
Metallography - 19
Weld Rejects - 6 unable to close, 5 incomplete penetra-
tion, 20 Pu contaminated greater than
500 d/m - Total 31.

CCTL Test Pins - 0.250 in. OD - Type 304 SS

Total Welds - 677
Metallography - 5
Weld Rejects - 2 unable to close, 4 incomplete penetra-
tion - Total 6.

Irradiation Test Pins for PNL-7, 8 - 0.250 in. OD - Type 316 SS

Total Welds - 216
Metallography - 12
Weld Rejects - 0

Irradiation Test Pins for PNL-17 - 0.230 in. OD - Type 316 SS
20% Cold Worked Cladding

Total Welds - 97
Metallography - 5
Weld Rejects - 1 incomplete penetration

Physics Test Pins - 0.230 in. OD - Type 316 SS 20% Cold Worked
Cladding

Total Welds - 727
Metallography - 40
Weld Rejects - 27 incomplete penetration, 13 tungsten
inclusions, 2 porosity, 6 concavity,
4 other - Total 52.

It was necessary to increase the welding current approximately 14% to achieve complete penetration when the change was made from commercially available Type 316 SS to vacuum melted Type 316 SS end cap material. The vacuum melted end cap material which is the reference material for FFTF was used in the end caps for PNL-17 and the physics test pins.

The emphasis on the welding development program in the immediate future will be on: (a) the effect of end cap machining tolerances and end cap-to-tube fit on the completed weld and (b) an intensive physical testing program on the completed welds. This testing program will be similar to the program currently in progress for the fuel pin cladding and will include tensile tests, stress rupture tests, and burst tests.

D. FUEL SUBASSEMBLY TECHNOLOGY

1. Fabrication Technology

a. Project Status

M. D. Jackson (732)

Project BCP-034 (Fuel Subassembly Fabrication Demonstration Facility) provides for a facility for demonstrating fabrication methods to assemble fast reactor fuel pin subassemblies.

Work in Room 231, Building 308, progressed well. The cleaning tanks and air dryer are being connected to the building utilities. The installation of the fuel bundle storage cylinders in Room 138 is approximately 75% complete. The doorways connecting Rooms 132 and 138 are complete. The overall project construction is 80% complete and on schedule.

Project BAP-022 (Plutonium Fuel Laboratory Addition) provides for special facilities required for initial storage, handling, testing, assembly, and instrumentation of nonirradiated fuel.

The design work of this project is 50% complete and on schedule. Completion of the design phase of this project is scheduled for November 15, 1969.

b. Wire-End Cap Termination Weld (3/32 in. fillet type)
Evaluation

R. L. Myers (732)

Oxalic acid etch tests on six wire-to-end cap weld samples were completed. The samples included two each with the weld parameters set for minimum acceptable penetration, mid-range values, and maximum acceptable penetration for a total of six samples. The results show a definite step structure. Thus, the welding technique does not increase susceptibility to intergranular attack. Good corrosion resistance is therefore maintained.

2. Fuel Examination - System Group 70

a. CFDD No. 71

E. B. Ramsey (942)

Changes to the base section of the published Inert Gas Cell Examination Facility CFDD No. 71 were approved by RDT subject to resolution of comments. RDT comments were incorporated and revised pages are being prepared for distribution.

b. Disassembly-Reassembly Equipment

W. S. Chenault and E. B. Ramsey (942)

The requirement for this equipment is to make provisions for the remote disassembly and reassembly of those irradiated core components and test assemblies requiring examination and evaluation.

Work has been initiated at ORNL to identify the impact of the proposed test assembly length reduction on examination facility related requirements for disassembly-reassembly of FFTF tests. The documentation of this work will be published

as an addendum to the "Conceptual Study of Disassembly-Reassembly Equipment for Examination of FFTF Core Components."

c. Fuel Examination Gas Cooling Tests

J. M. Yatabe (913), D. R. Dickinson, T. C. Reihman,
and R. A. Hildner (COO)

The objective of this program is to determine the requirements for gas cooling of fuel assemblies during removal from the reactor and during postirradiation examination. Work done last fiscal year on this task showed that axial and transverse flow interactions allowed hot spots to exist in the fuel assembly (an electrically heated 169-pin bundle -- 0.250-in. pins wrapped on a 12-in. pitch with 0.030-in. wire -- the heated section is 36 in. long) using positive displacement of the gas.

It has been proposed that the Fuel Examination facility be modified to allow axial gas flow in suction rather than by positive displacement. Suction flow should prevent hot spots which resulted with positive displacement flow, and should also control foreign contaminants in the gas.

CHAPTER IX. PHYSICS

(See Appendix A for List of Author Codes Used in This Report.)

A. CORE PHYSICS1. FTR Critical Experimentsa. FTR-2 Experiments in ZPPR

R. A. Bennett (833)

Shielding and core experiments being carried out in the FTR-2 critical assembly in ANL's ZPPR are on a schedule with a predicted completion date of November 19, 1969. No problems have been identified that will interfere with the scheduled completion.

b. FTR-3 Experiments

R. A. Bennett (833)

Preliminary plans for the conduct of two-zone core experiments in the FTR-3 Critical Experiment program have been prepared by ANL personnel and are being reviewed by BNW personnel.

If authorization for plutonium is not received early in November, it is expected that the FTR-3 program will be carried out in ZPPR.

c. Analysis of FTR-2 Experiments on the ZPPR

J. V. Nelson (833)

The FTR-2 reference loading (1-70) contains 527.04 kg fissile material and is 102.5 Ih supercritical with all fuel control rods fully inserted. The multiplication constant of this loading was calculated to be 0.98713, using 2DBS in X-Y mode and a 13-group cross section set. After correcting to S_{12} transport theory and for core heterogeneity, the calculated k_{eff} was 1.0036 ($c/e = 1.0026$). Similar results were obtained for loading 1-52 which was measured to be 83¢ subcritical. The final corrected 2DB, X-Y k_{eff} , was 1.0003 and the corresponding c/e value was 1.0029.

The edge worth of fuel relative to radial reflector material, keeping the number of B_4C control ring drawers constant, was calculated for two drawers and compared to the measured values. The results are shown in Table 9.A.I.

The radial reaction rates for ^{239}Pu , ^{238}U , and ^{10}B , were measured in FTR-2 through Row 137, 3.036 in. above the midplane.

TABLE 9.A.I. Fuel Edge Worths, FTR-2 on ZPPR

Matrix Position	Fissile Mass, kg	Exchanges	Worth of Exchange, 1h		c/e
			Experimental	Calculated	
130,29	1.0116	Core for C.R.	+27.25	+30	1.09
129,28		C.R. for R.R.			
128,32	0.5088	C.R. for Core	-24.06	-23	0.96
127,31		R.R. for C.R.			

Both the measured and calculated activities are plotted in Figures 9.A.1, 2, and 3. The calculated reaction rates, using core averaged cross sections, and the measured data are normalized to the same value at the core center in all three plots. The calculational results were obtained from the 2DBS, 13 group, X-Y calculation of loading 1-70 mentioned above. In general, the discrepancy between measured and calculated reaction rates near the core reflector interface, which occurred in the analysis of previous cores [ZPR-3 Assemblies 48a, 51, and 56b⁽¹⁻³⁾], is not present in FTR-2. However, neither did

1. "2DBS, A Modification of 2DB," FFTF Monthly Informal Technical Progress Report, BNWL-990, Battelle-Northwest, Richland, Washington, January 1969, p. 3.54.
2. R. A. Bennett and S. L. DeMyer, Analysis of FTR Phase A Control Rod Experiments in ZPR-III, Assemblies 48 and 48a, BNWL-967, Battelle-Northwest, Richland, Washington, April 1969.
3. W. R. Young and R. A. Bennett, Analysis of FTR Phase B Critical Experiments, Part 1, ZPR-III, Assembly 51, BNWL-1138, Battelle-Northwest, Richland, Washington, August 1969.

FIGURE 9.A.1. FTR-2 Radial Reactor Rates, ^{238}Pu

FIGURE 9.A.2. FTR-2 Radial Reaction Rates, ^{238}U

FIGURE 9.A.3. Radial Reaction Rates, ^{10}B

the discrepancy occur in Assembly 48a when traverses were taken through a peripheral control rod. Both the calculated ^{239}Pu fission rates and the ^{10}B (n, α) reaction rates are considerably lower than the measured values throughout the reflector. One possible source of error, being investigated presently, is that the effect of the matrix structure outside the reflector is not accounted for in the calculational model.

The worth of the control ring was calculated in 2DB X-Y and R-Z geometries. The worth computed in R-Z mode was $-4.84\% \Delta k/k$ while the X-Y result was $-4.83\% \Delta k/k$.

d. ZRP-3 Peak-to-Average ^{239}Pu Fission Rate
V. O. Uotinen (COO)

The ratio of peak-to-average ^{239}Pu fission rate has been determined from measured and calculated fission rate traverses in ZPR-3 Assemblies 48a, 51, and 56b. For these assemblies, the peak-to-average ratio is consistently overestimated by the calculation. The average calculated radial peaking factor is 6.5 to 9.2% high; the average calculated axial peaking factor is 3.5% high; the overall peaking factor is 10 to 13% high.

e. FTR-3 Preanalysis
S. L. DeMyer (833)

Material compositions and the reference reactor configuration for the FTR-3 Critical Experiments program are tentatively approved.^(1,2) This will be a 2-zone core approximating a homogenized version of the FTR reference design. Preanalysis of this assembly is nearing completion.

1. R. A. Bennett. Unpublished Data. Battelle-Northwest, Richland, Washington, September 4, 1969. (Personal Communication to A. Travelli, "Required FTR-3 Experiments, Suppl. No. 1," BNW-FFTF-692557.)
2. A. Travelli. Unpublished Data. Argonne National Laboratory, Argonne, Illinois, September 29, 1969. (Personal Communication to R. A. Bennett, "Detailed Plans for FTR-3 Experiments," ANL-69-188.)

Some calculated characteristics are given in Table 9.A.II.

TABLE 9.A.II. *Calculated Characteristics
and Values of Material Compositions*

1. Fissile Mass, kg	539
2. Core Volume, liters	1039
3. Effective Multiplication Constant	1.015
4. Controls: 16 - 4 × 4 in. 32 vol% Natural B ₄ C, worth, % Δk/k	4.2
5. Controls: 5 cm thick ring, 32 vol% Natural B ₄ C worth, % Δk/k	6.1
6. Controls: 16 - 4 × 4 in. 48 vol% Natural B ₄ C worth, % Δk/k	4.8
7. A 16% increase in ²³⁸ U content of outer zone, worth, % Δk/k	-0.7
8. A 2 vol% reduction (22 liters) in core volume at the outer edge of outer zone, worth, % Δk/k	-0.3
9. Redistribution of 16 - 4 × 4 in. control zones in an array of 8 sets of two closely spaced zones, worth, % Δk/k	-0.4

f. ETOX

R. B. Kidman (COO)

Both ETOX (IDX) and MC² have generated multigroup cross sections for ZPR-III Assembly 52. A comparison shows good general agreement but a few differences exist, and those occurring in the resolved and unresolved region of ²³⁹Pu are being investigated.

The fission cross sections for Groups 13 (15 to 24.8 keV) and 20 (454 to 749 eV) were calculated in two ways by ETOX: (1) Using the ETOX unresolved approximations. (2) Using the MC² unresolved approximations. Since the resulting cross sections were insignificantly different, it appears that the different approximations will not account for the present ETOX - MC² differences in ²³⁹Pu.

The effect of varying the number of fine intervals for any coarse group had to be examined since ETOX and MC^2 vary in this respect. Changing the fine interval size did not change the ETOX value for Group 13. However, for Group 20, the ETOX value changed by approximately 2%. Group 13 has smooth floor values while Group 20 has discontinuous floor values which explains the above results. Fine interval size cannot explain the ETOX - MC^2 differences for Group 13, but, depending on how MC^2 handles the discontinuities, it may explain some of the Group 20 differences.

Very often, the floor discontinuities occur at the energies separating the resolved, unresolved and smooth cross section regions. ETOX was reprogrammed so that interpolation on the floor cross sections does not occur across any of these energies. This improved the ETOX - MC^2 comparison in the groups containing those energies.

The effect of ^{239}Pu - ^{238}U resonance overlap in Group 20 was investigated by running ETOX in the following ways: (1) Using a constant effective total cross section for ^{238}U . (2) Using a model that specifically accounted for the ^{238}U resonances. The similarity of the resulting cross sections suggests that ^{239}Pu - ^{238}U resonance overlap cannot explain the Group 20 ETOX - MC^2 differences.

Group 22 (167 to 275 eV) is a resolved region for both ^{239}Pu and ^{238}U . Resonances overlap was investigated in this group by using the BNL values for σ_t^{238} and σ_t^{239} , and the nuclide concentrations for Assembly 52. Preliminary results indicate that the effective group total cross sections for ^{238}U - ^{239}Pu , as calculated in the 1DX manner, do not vary more than 1% from the values which are calculated in an exact manner (i.e., by following the exact energy variation of σ_t^{238} and σ_t^{239}).

2. Uncertainties in FTR Design Parameters

R. A. Bennett (833)

Based upon the FTR Critical Experiments conducted to date, a preliminary assessment of uncertainties in some FTR design parameters has been made. These values are summarized in Table 9.A.III. Included in the table are reasonable corresponding design allowances that are to be included in the final specification of design parameters. Statistical confidence levels are included where possible. The major exception is the FTR Doppler coefficient uncertainty which is an intuitive estimate based upon experiment-theory correlations on small sample Doppler experiments conducted in the FTR Critical Experiments program.

TABLE 9.A.III. *Summary of Uncertainties
in FTR Design Parameters*

<u>Parameter</u>	<u>Value</u>	<u>Uncertainty</u>	<u>Design Allowance</u>
Critical Mass (CM), kg	Nominally 530 ²³⁹ Pu	±1.3% of CM (95% CL)	+1.3% of the Calculated Critical Mass
Peak-to-Average Power	Calculated Value	±3% (95% CL)	+3%
Sodium Void Coefficient			
Whole Core	Calculated Value	±7%	+7%
Central	Calculated Value	Factor of 2	+100%
Control Rod Worths			
Central	Calculated Value	±20% (95% CL)	+20%
Peripheral	Calculated Value	±6% (95% CL)	+6%
Doppler Coefficient	Calculated Value	±~50% ~95% CL)	-50%

3. Count Rates in the Shutdown FTR

V. O. Uotinen (COO)

Subcritical source calculations have been performed to determine reaction rates at in-vessel sensor locations in the FTR. These calculations indicate that the expected count rate at 96 cm, using a commercially available 2-in. ^{235}U fission chamber, is ~ 50 counts/second when operating in a gamma-field of 10^6 R/hour. According to stated design requirements,⁽¹⁾ a count rate as low as 10 counts/second would be acceptable. Hence, the expected count rate is more than adequate, and a startup source is not a requirement for the FTR.

Immediately following shutdown, while the gamma-field is higher than 10^6 R/hour, it should be possible to verify that the chamber is responding to neutrons by simply removing a safety rod or two.

B. RADIATION AND SHIELDING

1. Core Support Structure Shielding

D. R. Marr (813)

Location of core support structural members less than 5 ft from the core will require that efficient shielding material be incorporated to reduce radiation damage to tolerable rates. Calculations were made to determine the effect of sodium penetrations, which might exist below control rods or closed loops, through a stainless steel-sodium shield. Two-dimension diffusion theory calculations indicate localized total neutron flux leakage through such a penetration would be about a factor of five greater than through the shielded region. In order to substantiate the diffusion theory calculation, both a two-dimension discrete ordinates and a Monte Carlo calculation were made. Since these substantiated the

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1. CSDD for System 95, p. 1.1-18, March 18, 1969.

diffusion theory calculation, it is concluded that sodium-filled penetrations the size of a flow channel are not tolerable through the lower axial shield region if the core support structure is to be protected adequately.

2. Gamma Distributions

D. R. Marr (813)

Calculation of the gamma flux distribution throughout the FTR system is required to define heat generation rates and the radiation environment. In most regions external to the core, the gamma intensity is dominated by photons resulting from neutron capture. Thus, to obtain the gamma distribution, it is first necessary to calculate the neutron distribution and the corresponding spatial gamma production rates, followed by the gamma transport calculation. The necessary libraries and linking codes have now been developed to automate this process, providing a discrete ordinates oriented gamma capability.

3. Stored Fuel Effects on FTR Ex-Vessel Monitoring

E. T. Boulette (813)

The conceptual design of the FTR includes in-vessel storage of fuel subassemblies in the Na annulus around the radial shield. Since ex-vessel monitors are to be used for full-power monitoring, the effect of the stored fuel on the monitor signal was investigated.

Two-dimensional diffusion theory calculations in R- θ geometry were made based on Concept V-A. A 120 degree sector with reflective boundary conditions at $\theta = 0$ degrees and $\theta = 120$ degrees was employed. Two large homogeneous stored fuel zones were positioned in the Na annulus. The larger zone, containing 11 homogenized fuel subassemblies, was

bounded azimuthally by angles $\theta = 75$ degrees and $\theta = 93$ degrees; whereas the smaller zone, with 6 subassemblies, was bounded by the angles $\theta = 36$ degrees and $\theta = 51$ degrees.

Two calculations were made, one with the stored fuel in position, the other with the stored fuel replaced by sodium. The reaction rate distributions for ^{239}Pu (n,f) were calculated and a comparison was made along $\theta = 4.5$ degrees. The results of the comparison indicate that only about 6.5% of the signal received at the ex-vessel monitor is induced by flux from the stored fuel when the reactor is operating at 400 MW.

4. ZPPR FTR-2 Reaction Rate Calculations

E. T. Boulette (813)

A series of shield experiments are in progress on ZPPR to provide an evaluation basis for the FTR conceptual design. Two-dimensional diffusion theory calculations were made to provide guidance for the experiments. In addition to flux distributions, reaction rate distribution for $\text{B}(n,\alpha)$, $\text{Mn}(n,\gamma)$, $\text{Na}(n,\gamma)$, $^{235}\text{U}(n,f)$, $^{238}\text{U}(n,f)$ and $^{239}\text{Pu}(n,f)$ were calculated. In the case of ^{235}U and ^{239}Pu , reaction rate distributions calculated from core-spectrum and shield-spectrum weighted reaction cross sections were obtained and compared. Reaction rate comparisons indicate very good agreement (<1%) in the core region, with differences no larger than about 10% exterior to the core. In the case of ^{235}U , the core-spectrum weighted reaction rate was higher in the shield; whereas for ^{239}Pu , the shield-spectrum weighted reaction rate was higher in the shield.

5. Dose Rates at the FTR Operating Deck

C. A. Mansius and E. T. Boulette (813)

The current personnel protection criterion ascribed at the deck location above the FTR cover is 0.2 mrem/hr. Calculations were made to determine the necessary effective thickness

of stainless steel required to satisfy this limit. Two primary sources of radiation exist. One is the neutron flux emanating from the core and the other is the gamma flux from the ^{24}Na in the sodium pool. Based on a 14.5 ft pool, 39 in. of steel (or its equivalent) is required to reduce the neutron dose rate at the deck to about 0.1 mrem/hr. This amount of steel is more than adequate for the pool gammas, giving a total <0.2 mrem/hr.

Another source of radiation that was considered is the closed loop piping. Calculations indicate that 20 in. of steel (or equivalent) is required to reduce the gamma intensity from four 4-in. closed loop pipes (2 inlet, 2 outlet) to below 0.2 mR/hr in the region above the work deck.

CHAPTER X. SAFETY

(See Appendix A for List of Author Codes Used in this Report.)

A. PLANT SAFETY ANALYSIS1. Fault Tree Simulation Test

R. J. Schroder (821)

The fault trees for several protective functions in the plant protective system were evaluated using the fault tree simulation program, MFTS. These runs were for comparison with results calculated using standard reliability techniques. In general, the simulation results tend to be smaller than the hand calculated results. The differences were created primarily by a tendency for the program to underestimate unreliabilities. However, the downward bias is somewhat less than that indicated by the hand calculations since the latter were conservative.

The following cases were run using the indicated test intervals and failure rates.

<u>Protective Function</u>	<u>168 hr Test</u>		<u>1500 hr Test</u>	
	<u>Hand Calc.</u>	<u>MFTS</u>	<u>Hand Calc.</u>	<u>MFTS</u>
Low Level Neutron Flux Level	2×10^{-4}	1×10^{-4}	2×10^{-2}	2×10^{-2}
Low Level Neutron Flux Period	3×10^{-4}	1×10^{-4}	2×10^{-2}	2×10^{-2}
Intermed. Level Flux Level	2×10^{-4}	1×10^{-4}	1×10^{-2}	1×10^{-2}
Intermed. Level Flux Period	(Same as for Low Level Flux Level)			
Power Level Neutron Flux	3×10^{-6}	1×10^{-6}	2×10^{-3}	0.9×10^{-3}
Power Level Neutron Rate-of-Change	4×10^{-6}	2×10^{-6}	2×10^{-3}	1×10^{-3}
Bulk Outlet Temperature	9×10^{-7}	4×10^{-7}	7×10^{-5}	5×10^{-5}
Primary Loop IHX Outlet Temperature	1×10^{-6}	0.6×10^{-6}	3×10^{-4}	2×10^{-4}
Reactor Flux/Flow	2×10^{-5}	2×10^{-5}	5×10^{-3}	8×10^{-3}
Reactor Vessel Coolant Level	(Same as for Primary IHX Temperature)			

2. Multiple Protective Function Failure

R. J. Schroder (821)

Several possible accidents are subject to detection by more than one protective function. The probability of failure of two or more protective functions for two cases was investigated using MFTS. Nominal failure rates and a test interval of 168 hr were assumed in each case. Only independent component failures were considered. The results should be viewed accordingly. The following unavailabilities were found.

Low Level and Intermediate Level Neutron Flux 1×10^{-7}

Power Level and Intermediate Level Neutron Flux 1×10^{-9}

The above results can be used to conclude that if only independent component failures are considered the unavailabilities for any other pair of protective function (except those with common components) are about 10^{-9} or less. The unavailability of a pair of protective functions having common elements (e.g., power level flux level and flux-flow) is about equal to, but a little better (lower) than, the unavailability of the more reliable of the two.

The only conclusion that should be reached is that component failures will not make a significant contribution to the unavailability of multiple protective functions unless common components are involved.

B. ANALYSIS OF HYPOTHETICAL ACCIDENTS

1. Containment Transients

R. D. Peak (821)

The CAVCEL digital computer code (formerly called the CONSDM or CONRAD code) was used to analyze the responses of the 2-volume containment to a DBA energy release of 4000 MW-sec and a fuel vapor release of 200 kg. The results can be compared to a DBA energy release of 8000 MW-sec and 790 kg of fuel

vapor reported previously.⁽¹⁾ The 2-volume containment includes the reactor cavity and the three heat transport cells. The reduced spray of fuel vapor into the cavity atmosphere reduced the initial pressure peak to only 24 psig (compared to 50 psig previously). The reduced value of excursion energy about doubled the time delay for initiation of boiling of the sodium pool on the floor of the cavity. The secondary pressure peak during the boil off of sodium from the cavity and its condensation in the cells remains the same as before, about 18 psig.

2. Sodium Pool Fire in Containment Sphere, DBA with no Machinery Dome
P. R. Shire (821)

The consequences of a hypothetical pool fire in the main containment vessel have been calculated for the case of a severe excursion accident with no machinery dome. Containment sphere initial conditions are 80 °F and 14.7 psia. Approximately 75% of the vessel inventory or a 130 ton mass of 1050 °F liquid sodium is assumed to be expelled from the reactor vessel. Reactor head lift following the accident accommodates the large mass expulsion. With no machinery dome, liquid sodium is assumed to come to rest in a pool covering the entire operating floor of the reactor building. Uniform mixing of core debris with vessel sodium prior to expulsion was assumed pessimistically. Therefore, 75% of core decay heat from full power (400 MW_t) is carried with the sodium resulting in an initial heat input of 800 Btu/hr/ft^2 of pool surface. Heat losses from the sodium pool were to the sphere steel and environment and to the concrete operating floor. Calculation was performed with the SOFIRE code.

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1. MITPR, August 1969.

Sphere internal pressure rose rapidly to a peak of 24.0 psig in 8.5 minutes; air temperature was 1060 °F while sphere steel temperature was 120 °F. The run terminated at 2.8 hr because of low oxygen concentration (<0.05%) sphere internal pressure and temperature decreased to 8.3 psig and 680 °F while sphere steel temperature was at 627 °F and rising. The rising steel temperature and decreasing depleted air temperature indicated a limiting steel temperature of 650 °F.

3. Cold Sodium Insertion Analysis

C. L. Fies (821)

The insertion of cold sodium into the reactor core when the reactor is critical will result in a reactivity increase caused by the negative temperature coefficients inherent in the design. A cold sodium accident can only occur if the reactor is operating at a power level somewhat less than full power where an idle heat transport loop contains sodium at a lower temperature. The most severe cold sodium accident would occur if the reactor is placed in a condition where the average core temperature were a maximum and the heat transport system were in a condition where the maximum amount of cold sodium were available for insertion into the core. This condition, while unlikely, could occur if the reactor heat transport system were operating with only one loop to remove the decay heat from the reactor and the flow through the reactor was controlled in such a manner that a delta T was present across the reactor core. A similar condition could occur if the reactor were operating under natural circulation conditions with the use of one heat transport loop. With the above considerations in mind, analyses were performed based on the following assumptions:

- The reactor is at a low prompt power (1 Watt) and decay heat level is 10 MW.

- Decay heat is being removed with one heat transport loop operating at a flow in such a manner that the full core ΔT (600 °F to 900 °F) is maintained.
- The other two loops are isothermal at 350 °F.
- At time zero, the two cold loops begin pouring cold sodium into the reactor inlet plenum increasing the primary flow.

The reactor inlet plenum mixing creates an inlet temperature transient to the reactor core.

The flow and temperature transients were then combined with the following reactivity feedback terms:

Doppler Coefficient	$T \frac{dk}{dt}$	-0.004
Sodium Temperature Coefficient	$\Delta k/^{\circ}\text{F}$	-1.5×10^{-6}
Clad Expansion Coefficient	$\Delta k/^{\circ}\text{F}$	-2.5×10^{-6}
Radial Expansion Coefficient	$\Delta k/^{\circ}\text{F}$	-3.0×10^{-6}

The reactor ΔT is collapsed during the first five seconds of the transient due to the decrease in inlet temperature combined with the increase in reactor primary flow. Both the Doppler and expansion coefficients insert positive reactivity during the first 5.4 sec of the transient. At approximately 5.4 sec the net positive reactivity insertion approaches prompt critical and the reactor power increases. The reactor power is turned around by the Doppler and expansion feedback but then increases to approximately 250 MW in 20 seconds. No fuel or cladding damage will occur since a reactor trip occurs near 5.4 sec as a result of the power spike.

APPENDIX A

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