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BNWL-1274

FAST FLUX TEST FACILITY
MONTHLY INFORMAL
TECHNICAL PROGRESS REPORT
NOVEMBER 1969

December 1969

AEC RESEARCH &
DEVELOPMENT REPORT

PROJECT	FILE NO.	LOCATION	FILE ROUTE DATE

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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT
NOVEMBER 1969

E. R. Astley
Project Manager

C. P. Cabell

by C. P. Cabell

December 8, 1969

SUBMITTED TO AEC DEC 16'69

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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT
NOVEMBER 1969

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 November 1969.

PRIOR REPORTS IN THIS SERIES

BNWL-CC-479 R1	FFTF - Monthly Report	- January 1966
BNWL-CC-524 RL	FFTF - Monthly Report	- February 1966
BNWL-CC-589 R1	FFTF - Monthly Report	- March 1966
BNWL-CC-647 R1	FFTF - Monthly Report	- April 1966
BNWL-CC-683 R1	FFTF - Monthly Report	- May 1966
BNWL-CC-719 R1	FFTF - Monthly Report	- June 1966
BNWL-CC-762 R1	FFTF - Monthly Report	- July 1966
BNWL-CC-806 R1	FFTF - Monthly Report	- August 1966
BNWL-CC-848 R1	FFTF - Monthly Report	- September 1966
BNWL-CC-882 R1	FFTF - Monthly Report	- October
BNWL-CC-936 R1	FFTF - Monthly Report	- November 1966
BNWL-CC-973 R1	FFTF - Monthly Report	- December 1966
BNWL-CC-1030 R1	FFTF - Monthly Report	- January 1967
BNWL-CC-1081 R1	FFTF - Monthly Report	- February 1967
BNWL-CC-1093 R1	FFTF - Monthly Report	- March 1967
BNWL-CC-1174 R1	FFTF - Monthly Report	- April 1967
BNWL-CC-1020 R1	FFTF - Monthly Report	- May 1967
BNWL-488	FFTF - Monthly Report	- June 1967
BNWL-512	FFTF - Monthly Report	- July 1967
BNWL-549	FFTF - Monthly Report	- August 1967
BNWL-571	FFTF - Monthly Report	- September 1967
BNWL-582	FFTF - Development Report	- September 1967
BNWL-603	FFTF - Monthly Report	- October 1967
BNWL-615	FFTF - Development Report	- October 1967
BNWL-638	FFTF - Monthly Report	- November 1967
BNWL-639	FFTF - Development Report	- November 1967
BNWL-659	FFTF - Monthly Report	- December 1967
BNWL-666	FFTF - Development Report	- December 1967
BNWL-686	FFTF - Monthly Report	- January 1968
BNWL-694	FFTF - Development Report	- January 1968
BNWL-712	FFTF - Monthly Report	- February 1968
BNWL-718	FFTF - Development Report	- February 1968
BNWL-742	FFTF - Monthly Report	- March 1968
BNWL-747	FFTF - Development Report	- March 1968
BNWL-772	FFTF - Monthly Report	- April 1968
BNWL-783	FFTF - Development Report	- April 1968
BNWL-800	FFTF - Monthly Report	- May 1968
BNWL-803	FFTF - Development Report	- May 1968
BNWL-830	FFTF - Monthly Report	- June 1968
BNWL-863	FFTF - Monthly Report	- July 1968
BNWL-881	FFTF - Monthly Report	- August 1968
BNWL-900	FFTF - Monthly Report	- September 1968
BNWL-915	FFTF - Monthly Report	- October 1968
BNWL-937	FFTF - Monthly Report	- November 1968
BNWL-964	FFTF - Monthly Report	- December 1968
BNWL-990	FFTF - Monthly Report	- January 1969
BNWL-1006	FFTF - Monthly Report	- February 1969
BNWL-1043	FFTF - Monthly Report	- March 1969
BNWL-1062	FFTF - Monthly Report	- April 1969
BNWL-1097	FFTF - Monthly Report	- May 1969
BNWL-1133	FFTF - Monthly Report	- June 1969
BNWL-1164	FFTF - Monthly Report	- July 1969
BNWL-1173	FFTF - Monthly Report	- August 1969
BNWL-1217	FFTF - Monthly Report	- September 1969
BNWL-1246	FFTF - Monthly Report	- 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 format which identifies, for each item of work reported,

- (1) The applicable LMFBF 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 LMFBF 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.

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. These author codes are defined in Appendix "A" to this report.

CHAPTER I. PROGRESS SUMMARYA. SITE

The Site Investigation Program for the FFTF was initiated in October 1969. The investigations will be in five parts:

- Drilling and sampling of exploratory borings combined with excavating, sampling, and making tests in test pits.
- Laboratory testing of samples obtained from the drill holes and test pits.
- Seismic survey.
- Water well tests.
- Site aerial survey and topographic mapping.

B. COMPONENTS1. Fuel Ducts

Approximately 40,000 lb of vacuum melted Type 316 SS have been ordered for the fabrication development of the prototype fuel ducts.

Wear pads have been successfully attached to hexagonal duct sections by resistance spot welding followed by gas-tungsten arc seal welds around the edges. Some initial distortion of the duct was noted as a result of the gas-tungsten arc welds.

2. Heat Transport

A transient model simulating the most recent reactor core design and primary loop configuration has been developed and debugged for steady state conditions using the digital computer code DYNASAR. An identical model is near completion on the FFTF Hybrid computer.

Radiation levels in heat transport cells are projected as less than 100 mrem/hr. This level will permit limited access for inspection of maintenance.

C. INSTRUMENTATION AND CONTROL

Because of cost and complexity, decision was reached to discontinue development of the reference FEDAL concept. A new concept is being sought.

Decisions on FFTF Plant Control have been made. Functionally, the control of the plant will be analog-manual and automatic. A single computer and limited peripheral equipment will be used for limited recording, calculating, and display functions.

A commercial unit suitable for analyzing airborne ^{239}Pu has been found. This unit will discriminate against radon/thoron interference.

A tentative specification for detection of sodium-to-air leakage has been calculated as a requirement for the LMEC Sodium Leak Detection task. For the conditions assumed, this leak rate is 2.5×10^{-1} g/hr.

D. CORE DESIGN

It has been computed that differential swelling induced in the stainless steel ducts of FTR by neutron flux and temperature gradient is accompanied by reactivity changes. For 30,000 MWd/tonne exposure, this distortion is -0.29% on the midplane radius and +0.32% on the central axis. This change in reactivity is significant and amounts to about 10% of the value of the estimated burnup per fuel cycle.

The draft of document BNWL-1251, FFTF Closed Loop Test Characteristics, has been completed and is being circulated for comment. The results indicate that a 2.5-in. test diameter is suitable for pin diameters of interest for 37-pin and 19-pin bundles. A 61-pin bundle would be feasible only at unusually small pin diameters and low linear power.

E. FUELS AND MATERIALS

Postirradiation creep tests on Type 304 and 316 SS show a marked difference in the behavior of these two alloys. After irradiation to $\sim 3.5 \times 10^{22}$ n/cm² (total) at 1100 to 1150 °F, Type 304 specimens show a three- to four-fold decrease in rupture life; while Type 316 specimens show a fifty-fold-plus decrease in rupture life. This variation has significant implication in terms of LMFBR components that operate in the 1100 °F temperature range. Thus the Type 316 material, which exhibits a substantially longer rupture life in the unirradiated condition, can actually show a significantly lower rupture life than Type 304 SS after irradiation.

Characterization of the creep behavior of unirradiated weldments shows that rupture life and creep rates can be significantly different from the properties of Type 304 SS base metal. Ductility of weldments in creep can be substantially lower than the ductility of Type 304 SS base metal. Such property variations could be significant in affecting local deformation or ductility limits of welded structures subjected to creep conditions.

Evaluation of the elevated temperature fatigue crack growth rates of Type 304 SS is continuing. The effect of temperature level on fatigue crack growth rates is significant; at 1000 °F and low stress intensity factor levels crack growth rates are 10 to 12 times greater than at room temperature. Testing at 1200 °F has begun. The potential detrimental effects of loading frequency, load wave form and environments at elevated temperatures on crack growth rate are discussed.

F. FUELS RECYCLE

A computer has been installed in the FFTF Fuel Fabrication Demonstration Facility, Building 308. Primary functions of the computer will be to maintain a historical record of FFTF fuel

and each fuel component; to maintain an accountability record for the facility by station or other location; and to act as a secondary critical mass limit check on fuel transfers between stations.

CCL Mark II fuel assembly is being fabricated for testing to provide design verification and identification and evaluation of potential long term failure modes for a slightly modified FTR Conceptual Fuel Subassembly.

Pin fabrication was completed through the loading of the upper reflector. The lower grid components (layer spacing grid plates, layer locking grid, and element supports) were fabricated and are being inspected. An electron beam welding technique was developed for fabrication of the layer spacing grid plate assembly. This technique was then used to weld the round rail to the plates.

Fuel pellet fabrication for EBR-II subassembly PNL-6 was completed. The fuel is 30% enriched ^{235}U in the standard EBR-II test composition of 75% UO_2 -25% PuO_2 .

G. PHYSICS

A 3-dimensional diffusion-burnup code, 3DB, has been developed; a formal report is being typed. Another 3-dimensional code, RSYN, which employs synthesis methods, is being debugged.

To assist in planning future fuel cycle studies, analyses were performed to determine the number of energy groups, the number of axial and radial burnup zones, and the length of the burnup time steps that are required for accurate burnup calculations. The necessity of recalculating the cross section data after each burnup interval was also studied.

The ZPPR shielding experiments, which are to provide an evaluation basis for the FTR conceptual design, are completed.

H. SAFETY

The energy generated from molten fuel-sodium interaction has been re-evaluated using a more advanced model. The previous model assumed that the fuel and Na were in thermal equilibrium, whereas, the present model attempts to simulate the dynamic nature of the heat transfer process.

FFTF safety development requirements related to the fuel element failure propagation program have been identified. Those program requirements which lie outside current programs at BNW have been outlined and these needs have been transmitted to RDT.

The history of the fuel core and structural material after a meltdown is being estimated with special emphasis on the downward progress of the molten material and on the heat transfer into the concrete. With the formation of a crust of frozen material caused by radiation heat losses, a maximum pool radius of 23 ft was reached after 100 days. The crust thickness at this time was 14 ft and the pool temperature was 1100 °C. The last bit of liquid in the pool froze at a depth of 22.4 ft after 1.8 years.

I. FFTF BNWL REPORTS AND DOCUMENTS ISSUED - APRIL 26, 1969
THROUGH NOVEMBER 14, 1969

TOPICAL REPORTS (GREEN COVER)

<u>BNWL No.</u>	<u>Authors</u>	<u>Title</u>
760	D. E. Simpson, et al	Preliminary Analysis of Postulated Maximum Accidents for the FFTF
874	J. W. McLaughlin	Fault Tree Analysis
896		Monthly
954	R. W. Hardie W. W. Little	Fox Code
957	G. B. Barton	Instrumentation for Inert Gas Analysis
962	O'Dell, W. L. Bunch	Revised Fast Reactor Library for RIBD
967	R. A. Bennett, S. L. DeMeyer	Analysis of FFTF Phase "A" Control Rod Experiments
970	J. R. McBride B. C. Fryer	FFTF Modified Fermi Control Rod Drive Mechanism Conceptual Design Study
992	O'Dell, R. W. Hardie W. W. Little	Numerical Comparison Diffusion Transport Codes
993	D. L. Koreis C. E. Leach	Range of Allowable Operating Conditions for FFTF Closed Loops
1011	D. J. Oakley	Vessel Evaluation Summary
1025	J. M. Stringer	Signal Cable Read Out Errors
1035	E. B. Schwenk L. A. James	Estimate of the Fatigue Cracking Behavior of the Fast Test Reactor Primary Piping
1036	P. K. Telford	Design Survey and Conceptual Definition of a Reactivity Oscillator for the Fast Test Reactor
1052	J. L. DeMeyer	Centerless Grinder
1067	W. L. Little, Jr. L. L. Maas	Nuclear Parameters and Parametric Studies for the Fast Test Reactor (FTR)

BNWL No.	Authors	Title
1069	R. Kolowith	Sealing Mechanisms for FFTF Closed Loop and Open Test Position Closures
1070	R. E. Keyes	Reactor Flow and Pressure Drop Study
1075	M. K. Millhollen	Sodium Flow Test of Seven-Rod Cluster in Sodium Components Evaluation Loop (SCEL)
1093	G. E. Russcher	Calculated Damage Functions for Determining Irradiation Effectiveness
1105	R. G. Hoagland	Effect of Strain Rate on Mechanical Properties of Austenitic Stainless Steels
1130	J. J. McCown	Evaluation of Techniques for Controlled Addition of Impurities
1142	W. H. Sutherhand	A Finite Element Computer Code (AXICRP) for Creep Analysis
1162	R. W. Hardie W. W. Little	PERT-V, A Two-Dimensional Perturbation Code for Fast Reactor Analysis
1176	R. R. Studer	Removal of Superficial Blemishes from Type 316 Stainless Steel Fuel Cladding
1192	R. V. Bowersock	A Critical Review of Stress - Corrosion Cracking as Associated with the FFTF Driver Fuel Cladding
1197	D. R. Marr M. G. Zimmerman	FTR Shield Design Cross Sections -- A Potential Evaluation
1198	J. E. Sammis	Parametric Study of the UO ₂ Pellet Compacting Operation
1213	R. E. Bardsley	Computerized Preventative Maintenance System for FFTF Fuels Demonstration Facility
1230	J. R. McBride	Stepping Motor Control Rod Drive for FFTF Report on Development and Testing Program
1231	J. R. McBride	Control Rod Drive Mechanism Extension Rod Disconnect Study
1232	J. R. McBride	Contracting Ring Seal for FFTF Control Rod Literature Search and Industry Survey
1257	J. J. Regimbal	FTR Protection with In-Channel Instrumentation

TOPICAL DOCUMENTS (WHITE COVER)

<u>BNWL No.</u>	<u>Authors</u>	<u>Title</u>
1027	Staff, FFTF Fuel Department	FFTF Fuels Development Current Preliminary Data Report for February, 1969
1036	P. K. Telford	Design Survey and Conceptual Definition of a Reactivity Oscillator for the Fast Test Reactor
1047	W. Babcock	State of Technology of Sodium Pumps
1064	E. G. Stevens, et al	FFTF Fuel Pin and Subassembly Conceptual Design Methods and Data
1065	G. B. Barton	Gas Chromatograph System for FFTF Failed Fuel Detection and Location - State-of-the-Art Report
1068	J. C. Tobin	Current Preliminary Data Metals and Materials Development
1070	R. E. Keyes	Reactor Flow and Pressure Drop Study
1086	R. A. Moen	Fluence Limit for FTR Vessel, Core Barrel, and Core Support Structure
1092	D. C. Boyd	Number and Type of Operating Cycles for the FFTF
1093	G. E. Russcher	Calculated Damage Functions for Determining Irradiation Effectiveness
1094	J. R. Sheff	Anomaly Detection in the FTR by Means of Fluctuation Measurements
1102		UK Commercial Data from BNWL-1064
1103	E. O. Ballard C. A. Pember	Proposal to Irradiate Unencapsulated Fuel Pins in the EBR-II: Subassemblies PNL-6, PNL-7, and PNL-8
1105	R. G. Hoagland	Effect of Strain Rate on Mechanical Properties of Austenitic Stainless Steels
1108	A. F. Lillie P. D. Cohn	FFTF Fuel and Core Parameter Studies
1118	J. G. Bradley W. R. Lewis	FFTF Testing Capability
1123	D. L. Lessor B. H. Duane	Permanent Magnet and Pilot-Static Probe Flowmeters for Liquid Sodium

BNWL No.	Authors	Title
1126	J. C. Noakes	Selection of Concepts for FFTF Control and Safety Rod Drives
1127	W. F. Brehm	Rebuild of Radioisotope Transport Loop for Fission Product Control Tests -- Design Description
1136	J. M. Yatabe, et al	Fast Reactor Fuel Assembly Thermal Hydraulic Development Program - Sodium Heat Transfer
1138	W. R. Young R. A. Bennett	Analysis of FTR Phase B Critical Experiments Part 1 -- 2PR-III Assembly 51
1139	W. R. Young	Analysis Phase B, Critical Experiment Part II -- ZPR-III
1182	B. C. Fryer	Survey of Scram Assist Devices for Application in the ITR Vertical Concept.
1190	R. D. Leggett, et al	Studies of Past Irradiation Examination of EBR-II Driver Fuel
1193	T. C. Reihman	Gas Cooling of FFTF Fuel Assemblies During Examination and Disassembly Experimental Feasibility Study
1207	T. C. Reihman	An Experimental Study of Pressure Drop in Wire Wrapped FFTF Fuel Assemblies
1209	C. D. Swanson	FFTF Instrumentation and Control System Task Force Study
1211	J. R. McBride	Annular Orifice Control Rod Decelerator for the FFTF Report on Development and Testing Program
1225	C. R. F. Smith	An Evaluation of Xenon Tagging -- A Concept for Locating Failed Fuel in the Fast Test Reactor
<u>PERIODIC REPORTS (GREEN COVER)</u>		
567		FFTF Periodic Technical Report, July through December, 1967; May, 1969
896		FFTF Fuels Department Current Preliminary Data October, 1968 - February, 1969
941		FFTF Periodic Technical Report, October 1968 through February, 1969; May, 1969
1054		FFTF Fuels Department Current Preliminary Data, March, 1969

<u>BNWL No.</u>	<u>Authors</u>	<u>Title</u>
1062		FFTF Technical Progress Report, April, 1969
1090		FFTF Quarterly Report, March, April, and May, 1968; August, 1969
1097		FFTF Technical Progress Report, May, 1969
1114 FF1		Fuels Department Current Preliminary Data, May, 1969
1120 FF1		FFTF Fuels Department Current Preliminary Data Quarterly, March, April, May, 1969
1133		FFTF Technical Progress Report, June, 1969
1155 FF1		FFTF Fuels Department Current Preliminary Data, June, 1969
1164		FFTF Technical Progress Report, July, 1969
1173		FFTF Technical Progress Report, August, 1969
1189 FF1		FFTF Fuels Department Current Preliminary Data, July, 1969
1217		FFTF Technical Progress Report, September, 1969
1246		FFTF Technical Progress Report, October, 1969

CHAPTER II. PLANT DESIGNA. OVERALL PLANT DESIGN

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

TABLE 2.I. Baseline Design Data

Summary of Baseline Design Data as of 11/17/69 - W. M. Noble

<u>General Plant Data</u>	<u>Units</u>	<u>Values</u>
<u>Reactor Systems (31000)</u>		
Core arrangement		vertical
Design life	years	20
Total power, initial	MW _t	400
Reactor coolant		sodium
Reactor bulk inlet temperature, initial	°F	600
ultimate	°F	750
Reactor bulk outlet temperature initial core	°F	850
design maximum and ultimate	°F	1050
Core temperature rise average initial	°F	300
ultimate	°F	350
design maximum	°F	400
Reactor cover gas		argon
<u>Core</u>		
Number of core lattice positions		91
Direction of coolant flow		upward
<u>Driver Fuel</u>		
Cladding material		Type 316 SS
Fuel geometry		hexagonal pin cluster
<u>Reactor Vessel Material</u>		Type 304 SS
<u>Heat Transport System (32000)</u>		
<u>Primary Loops</u>		
- Number		3

TABLE 2.I. (cont'd)

<u>General Plant Data</u>	<u>Units</u>	<u>Values</u>
- Primary Loop Material		Type 304 SS
Primary Pumps		
- Number		3
- Design flow rate	gpm	14,500
- Design pump head	feet	500
- Available net positive suction head	feet	51
- Design temperature	°F	1050
- Speed control		wound rotor motor with liquid rheostat
Intermediate heat exchangers		
- Number		3
- Type		Vertical shell and tube
- LMTD		
Initial	°F	85
- Capacity		
Initial	MW	133
Ultimate	MW	157
<u>Irradiation Testing System (33000)</u>		
<u>Test Facilities</u>		
Closed loops		
- Test section outlet temperature	°F	1400 (bypass flow permitted)
- Test section diameter	in.	2.5-3.0
- Material (in-core tube)		Type 316 SS

B. FFTF SITE, STRUCTURES, CONTAINMENT, AND UTILITIES

1. FFTF Site Investigation

F. H. Shadel (941)

The Site Investigation Program for the FFTF was initiated in October 1969. The investigations will be in five parts:

- Drilling and sampling of exploratory borings combined with excavating, sampling, and making tests in test pits.
- Laboratory testing of samples obtained from the drill holes and test pits.
- Seismic survey.
- Water well tests.
- Site aerial survey and topographic mapping.

Technical specifications have been prepared for each part of the work and the following details describe the manner in which the work is being carried out:

a. Exploratory Drilling and Test Pit Excavation

One deep hole, extending about 50 feet into the Columbia River basalt, was drilled and sampled to determine the sequence of strata, and the location of the groundwater table at the BNW-selected general site area. Four additional holes, each about 200 feet deep extending into the Ringold formation, were drilled and sampled at locations adjacent to the deep hole. The locations of these were based on the requirements of the shear wave velocity determinations to be made in the seismic survey.

Soil samples have been taken in the borings. A drive sampler was used in sand and gravel. Standard penetration tests were made in sandy materials and undisturbed samples taken in clay soils.

Hatch Drilling Company has completed drilling the five test holes to the following depths:

<u>Hole No.</u>	<u>Depth Below Ground Level (ft)</u>	<u>Water Table Below Ground Level (ft)</u>
1	220	170
2	649*	185
3	206	170
4	205	180
5	210	178

*Boyles Brothers began core drilling Hole No. 2 November 8, 1969 at -566.6. Established basalt flow at El 594.5. Ground elevation is approximately 555 feet MSL.

b. Soil Testing

Bechtel Corporation has entered into a consulting agreement with Shannon and Wilson, Inc., a soil testing firm, to make field density determinations in test pits at the site and to perform laboratory soil tests in accordance with the technical provisions of a soil testing specification. The magnitude of the soil testing program was dependent upon the material encountered and the number of samples that could be obtained in the drilling.

Eleven test pits were excavated at locations selected by Bechtel Corporation. The purpose of these pits are threefold. First, field density determinations were made in them by Shannon and Wilson. These densities will later be compared with information obtained from the drilling and the seismic survey. Second, they permitted inspection of the uppermost portion of the overburden; and third, the overburden may be sampled and evaluated for determination of its suitability as backfill material.

Bechtel Corporation has entered into a consulting agreement with Professor H. B. Seed of the University of California for advice on the requirements for soil testing to determine the dynamic properties of the materials encountered, and also to analyze the liquefaction potential of the site soils when subjected to the operational and design basis earthquakes.

Bechtel Corporation will analyze all the field and laboratory test data and will prepare a soil report.

c. Seismic Survey

A seismic survey to allow correlation of geologic units between drill holes, and to define the compressional and shear wave velocities of the materials beneath the site, is being performed by Weston Geophysical Engineers, Inc., Weston, Massachusetts. These data are necessary for predicting the response of the foundation to seismic acceleration, and for investigating the soil liquefaction potential of the foundation. These data are especially valuable in evaluating the properties of coarse granular materials, which are very difficult to sample.

The seismic survey will consist of two seismic lines crossing the site. The lines will be at right angles and normally intersect at drill holes. This phase of the survey will be designed to profile the various geologic units down to and including bedrock (basalt), and to determine compressional and shear wave velocities. In addition, the shear wave velocities of the materials between ground surface and a depth of 150 ft will be determined in detail. These shear wave velocities must be measured in drill holes. The data obtained therefrom will allow the determination of the upper limit of the shear modules of the foundation. Finally, in order to evaluate the density of the foundation materials, some of the holes drilled for the soils investigation will, if possible, be logged by others with a nuclear density probe (neutron or gamma ray transmission).

The second phase of the geophysical exploration is currently planned for the week of December 1, and the hydraulic draw-down test for about the second week in December.

d. Water Well Tests

Field permeability tests will be performed in two of the drill holes at the site. The purpose of the tests will be to: (1) measure the permeabilities of the materials in order to evaluate the consequences of accidental discharge of radioactive material at the site, and (2) to evaluate the materials as a source of ground water supply.

e. Site Aerial Survey and Topographic Mapping

The site survey will be performed by aerial photography and ground control surveying. It will include preparing topographic maps for the FFTF site, its immediate vicinity and routes for the access road, railroad, and power transmission tie-line to the Midway-Benton power transmission line.

Aerial photographs will be obtained initially for an area large enough to permit additional contour mapping, if necessary, and specifically include the FFTF site area and routes for the access road, railroad, electrical power lines, communications, and water supply lines.

Although the indicated area to be photographed may appear to be large, it should be recognized that the cost of aerial photography is small in comparison with the cost of mapping. It is intended, therefore, to select the routing of road, railroad, power line and water pipeline from the aerial photographs and then designate the areas for mapping as required for the preliminary and final design.

CHAPTER III. COMPONENTSA. REACTOR COMPONENTS1. Fuel Ductsa. Material Selection

E. G. Stevens (913)

A WARD recommendation to change the reference fuel assembly duct material to Type 316 SS, 20% cold worked from Type 304 SS, solution annealed material has been accepted.

Potential benefits are as follows:

- Type 316 CW SS has less relative fluence-induced swelling than Type 304 SA material in the FTR duct operating temperature range. In reactor duct-to-duct interactions could therefore be expected to occur only at higher relative burnups with Type 316 CW material.
- Duct fabrication with Type 316 CW SS is feasible and the overall cost differential for the FTR core does not appear serious.

b. Duct Welding Development

W. F. Brown (AOO) R. N. Johnson (962)

(1) Automatic Welding System. The detail design for the automatic welding system is approximately 75% complete. The Idler collet framework and duct handling saddle for the welding fixture machine base is completed. This effort represents approximately 5% of the hardware fabrication. Work on the collet mechanisms has started.

Feasibility studies for the welding computer control has been prepared and submitted to the AEC for review.

(2) Wear Pad Attachment. A wear pad was attached to a 12-in. long Type 304 SS hexagonal tube. The wear pad was 0.060 in. thick and 4 in. wide. The hexagonal

tube had a wall thickness of 0.143 in. and a nominal ID across flats of 4.334 in. The wear pad consisted of two half hexagonals which were assembled around the center of the SS tube.

The pads were attached by resistance spot welding and manual gas tungsten-arc welding.

Dimensional characterization of the duct section before and after attachment of the wear pads revealed the following information:

- Spot welding of the pads to the duct caused no significant change in dimensions.
- Gas-tungsten-arc welding around the edges of the wear pads caused significant distortions, even though an internal back-up fixture was used. The inside dimensions of the duct decreased in varying amounts up to a maximum of 0.026 in.
- Thermal cycling of the ducts through a temperature range of 400 °F to 1100 °F caused no significant change in dimensions.

c. Duct Procurement Activities

R. N. Johnson, R. H. Todd (962)

Procurement was directed toward obtaining Type 316 SS in the 20% cold worked condition. Negotiations with the two selected vendors have established the feasibility of fabricating the ducts from this material and have determined the estimated effects on cost and schedule. The contract changes resulting from the change of material have delayed completion of contract negotiations. In order to minimize program slippage, BNW is ordering and will supply to the vendors the vacuum melted Type 316 SS needed for prototype development of the fuel ducts. In addition to the advantage in the schedule of having the stainless steel on order before the contracts have been signed, BNW's purchase of the steel means that both duct vendors will be using steel procured from the same source.

The revised duct specifications and statement of work, reflecting the change in material and changes identified during contract negotiations, have been submitted to RDT for review and comment.

2. Reactivity -- Control Drives

a. Control Rod Drive Conceptual Studies

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

Development and testing reports on the control decelerator, the stepping motor drive study, the extension rod disconnect study, and the control rod seal study were issued. The electro-mechanical latch study report is being reviewed for publication.

b. Roller Nut CRDM Testing at LMEC

J. C. Noakes, (914)

The N. S. Savannah replacement roller nut CRDM and associated equipment were received at LMEC and the receiving inspection completed. The components required for the facility modification are in the AI shop. Start of testing was scheduled for early December.

The contract modification with Todd Shipyards to perform wear testing of a roller nut CRDM in argon has been agreed to by Todd Shipyards and testing was scheduled to begin early in December at Galveston, Texas.

c. Prototype CRDM

P. K. Telford (914)

The third issue of the prototype CRDM procurement specifications was received from Westinghouse for BNW approval.

WARD-FRV-128, Control Rod Drive Mechanism Selection for FFTF, was received from Westinghouse. The report recommended the roller-nut CRDM, with a bellows seal for the reference CRDM for the FTR.

d. System No. 33 - Nuclear Controls

P. K. Telford (914)

Approval, with comment, was received from RDT on CCDD No. 33. In accordance with the comments, Section 1 of the CCDD is being revised.

3. Nuclear Control Components

H. C. F. Ripfel (811)

The control rod lifetime was defined⁽¹⁾ as a minimum of three effective full power months which corresponds to one fuel burnup cycle (about 100 full power days). There is little doubt that this goal can be reached with the reference design using enriched B_4C as a poison material and gas plena on the (poison) pins. However, from an economics point of view, it would be desirable if the duty time of the control rods could be increased to be consistent with the burnup limit (of 10% $\Delta k/k$). This is especially difficult to achieve with the operational control-secondary safety rods, because individual ones may be fully inserted while the reactor is at full power. The short lifetime is caused by the cutback in core component design length (12 ft) which results in reduced gas plena length (18 in.).

It was agreed upon to consider other designs (poison materials) as a fall-back position. Tantalum is a poor candidate for use on the primary system control rods and is a marginal candidate for the operational control - secondary safety system. Assuming a volumetric efficiency in excess of 70%, e.g., "strong-back" 19-pin design, the tantalum weight per rod is about 132 kg. This is to be compared to about 250 kg of tantalum required.⁽²⁾

1. WARD FRP-138, L. E. Strawbridge to W. B. McDonald, Poison Element Review Meeting, letter dated August 20, 1969.

2. This is based upon the Reference Concept, BNWL-955, where 3.0 to 3.1 kg ^{10}B per rod are necessary for the secondary system to meet total worth requirements.

The latter has been estimated based upon a worth ratio of 80 (kg Ta/kg ^{10}B) which also implies that no allowance needs to be made for burnup. Shuffling of the rods to higher worth positions plus an increased number of rods would be inevitable. Unless current design goals are changed it appears reasonable to concentrate on B_4C pins and assess the possibility of vented pins as a fall-back position if the tests show that the pins with gas plena perform unsatisfactorily.

B. HEAT TRANSPORT, COOLANT AND GAS COMPONENTS

1. Auxiliary Systems

R. V. Dulin (905)

a. System 81 - Sodium Receiving and Processing

Initial planning for design and development of system components indicates that such items as cold traps, EM pumps, and storage tanks may be designed and procured within normal Project schedules, and will not require special expediting.

b. System 82 - Inert Gas Receiving and Processing

Study of design considerations for radioactive gases from fuel elements has been reduced to consideration of provisions for operation with failed fuel. Vented driver fuel is no longer under consideration.

c. System 84 - Decontamination and Cleaning

The draft CSDD is undergoing continuing technical evaluation within the Project. Concern exists primarily with attempting to define a system on the basis of inadequate sodium systems decontamination technology.

2. Reactor and Heat Transport System Analysis

C. D. Flowers (822)

A transient model simulating the most recent reactor core design and primary loop configuration has been developed and

debugged for steady state conditions using the digital computer code DYNASAR. An identical model is near completion on the FFTF Hybrid computer. The model includes the vertical reactor core and two independent primary loop heat transport systems with hot leg primary pumps. One of the two simulated loops represents one HTS loop and the other simulated loop represents two HTS loops for a total of three loops.

The model is capable of calculating transients resulting from pipe ruptures in one of the three primary loops. The present model is programmed to calculate hole-type ruptures in the pipe at the reactor inlet nozzle. The model, however, can be modified easily for ruptures at any location in the primary loop. The model can be used also for nonrupture reactor and primary loop transients such as loss of BPA power, pump seizure, reactor scram, etc.

3. Dump Heat Exchanger Analysis

R. N. Madsen (822)

Transient results using the previously developed simple DHX simulations were sufficiently different from the reference TAP simulation that a cause for the variation was sought. Since the steady state agreement was excellent, it was felt that the transient results should be closer than those which were being obtained.

A detailed investigation resulted in the determination that fin efficiency was the cause of the discrepancy. Fin efficiency is used to simplify heat transfer calculations associated with extended surface finned tubes and is defined as the ratio of actual heat transferred from a given fin to the heat which would be transferred if all the fins were at a uniform temperature equal to the temperature at the base of the fin. As included in the simple simulations, fin efficiency, Ω , was a function only of the heat transfer coefficient, the

thermal conductivity and the geometry, which is constant for any given design. The difficulty was found to arise because of the mistaken assumption that this simple Ω was valid for both steady-state and transient calculations.

In order to improve this aspect of the DHX simulation, two approaches are simultaneously being developed. One method eliminates fin efficiency altogether and simulates the heat transfer by representing the fins with additional nodes. This method requires twice the number of integrations as previously used and thus increases the complexity of the simulation. The other method being investigated uses only three additional nodes. Work performed to date indicates that the first approach might be the most satisfactory, although the work is still in progress.

C. CORE COMPONENT AND FUEL HANDLING EQUIPMENT

1. Methods of Shortening Test Assemblies

R. C. Walker (912)

AI issued a Technical Data Record Report (AI-AEC-TDR No. 12891) which identified and evaluated methods for shortening closed loops and open test assemblies to the same length as a driver fuel assembly. Two concepts were recommended: (a) the full length closed loop would be removed from the reactor and transferred to an assembly and disassembly cell where it is shortened and/or reconstituted; (b) an open test assembly would be shortened in the reactor by shearing the instrumentation leads and would thereafter be handled like a driver fuel assembly. These recommendations will be reviewed by BNW and Westinghouse during the early part of December and the results will be integrated into the Task Force activities for recommendation to RDT.

CHAPTER IV. INSTRUMENTATION AND CONTROLA. FFTF I&C SYSTEMS1. Central Control and Data Handling System (System 91)

G. G. Thieme and G. L. Waldkoetter (933)

The following agreements were reached between RDT and BNW with respect to the design of System 91:

- Functionally, the control of the plant shall be analog-manual and automatic.
- A single computer and limited peripheral equipment shall be used in the following functional areas:
 - Recording - limited to data in Central Control.
 - Calculations - to include thermal power and reactivity balance.
 - Display - as auxiliary-to-analog indicators--using a limited number of CRT's on the console wings.

These agreements reject the proposition that a computer-based display and data recording system be used initially as a primary FFTF operations tool. A new draft of CSDD 91 is being prepared which reflects these agreements.

The performers of 189 Task IC-A, subtasks 2, 3 and 4, have been instructed to close out the work now in progress and to document the current status and progress made to date. The titles of these subtasks are:

- Subtask #2 - Non-Baseline Software
- Subtask #3 - Baseline Software
- Subtask #4 - Control Room Mockup.

Design of the software system will not proceed beyond items now being worked on and already near completion. The preparation of a software specification, however, will continue and is to be completed by 12/12/69. The software specification

will describe functional and operation features against which software proposals will be evaluated. In the event of a special software design contract, the spec will become the design requirement.

CRT display work in support of FFTF Control Room mockup is to cease once hardware repairs and modifications now in progress are completed. Documentation will provide up-to-date information on the display-computer interface hardware and software as well as any hardware modifications and current operational features.

2. Reactor and Vessel Instrumentation System (System 92) L. E. Fort (932)

A study was conducted to plan the requirements for a test program to evaluate the effect of thermal transients which may be imposed on the subassembly instrumentation. Two sources of transients were investigated.

- Reactor scram
- Sudden partial subassembly blockage.

Previously documented^(1,2) material from BNW and Westinghouse was used as a base. The following test conditions have been formulated for the above two events:

Fifty excursions from 1000 °F to 1500 °F at a rate of 600 °F/sec followed by a five-second soak at 1500 °F.

One additional thermal transient, not considered previously, has been identified, namely: If preheat procedures are not adopted, it will be possible to insert an instrument

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1. J. J. Regimbal. Technical Bases for FFTF Driver Fuel Instrumentation, BNWL-916. Battelle-Northwest, Richland, Washington, October 1968.
 2. Westinghouse letter Report FPC-57, "Summary of FFTF Reactor Plant Transient," dated May 28, 1969.

probe at ambient temperature (70 °F) into a 600 °F sodium pool in a very short time interval. Further test planning is in progress.

3. Plant Instrumentation System (System 93)

M. O. Rankin (933)

A tentative "acceptable" leak rate of nonradioactive sodium-to-air was determined to be 2.5 g/hr, based on a maximum level of one-tenth of the Threshold Limit Value for NaOH in the human body. Under these conditions, the maximum acceptable airborne concentration of Na_2O will be $2 \times 10^{-4} \text{ g/m}^3$.

A survey of commercially available smoke (Na_2O) detectors is being made. One detector, utilizing a light-scattering principle, is reported capable of detecting $1.5 \times 10^{-3} \text{ g Na}_2\text{O/m}^3$. Capabilities of other detectors will be reviewed.

4. System 94 - Fuel Failure Monitoring System

R. R. Derusseau (924)

As a result of Technical Review Board recommendations, design and development efforts related to scanning of a continuously flowing sodium sample from each fuel subassembly have been terminated.

Potential alternative concepts evaluated by the review board included gas tagging, flowmeter monitoring, pin pressure monitoring, noise monitoring (neutronic and signal), and in-vessel gas separation. Further engineering effort is required to define alternatives before selection of a new concept.

5. Flux Monitoring and Control System (System 95)

L. W. McClellan (932)

a. Conceptual System Design Description for System 95

No change in status from last month's report.

b. Neutron Detector Thimble Designs

Parallel studies are in progress at both BNW and WARD on the feasibility of cooling the in-vessel neutron detector thimbles. At this time BNW is studying the feasibility of cooling with NaK and with nitrogen gas; WARD is studying the feasibility of using a stream of argon gas.

A BNW analysis of a NaK cooling concept was completed by J. E. Krajicek of the FFTF Safety and Systems Analysis Sub-department. This analysis showed that NaK cooling of a BNW thimble design to 300 °F was technically feasible. The model used in this study is delineated on drawing SK-3-14662, In-Core Flux Monitor Assembly not included in this report.

c. Detector Location Studies

A Westinghouse report, "Location and Environment of the FTR Neutron Flux Monitor," (FPS No. 173) is being reviewed.

An effort is being made to correlate all BNW and Westinghouse information relative to the neutron detector locations. Uranium-235 reaction rates, gamma levels, effects of stored fuel, physical access, and compatibility with the core and vessel concept are being considered.

6. Radiation Monitoring System (System 96)

M. O. Rankin (933)

The methods of continuously monitoring for airborne plutonium were reviewed. One vendor has been found who will provide a small economical unit which can detect a continuous concentration sensitivity of one mpc* in a 4-hr period. The unit utilizes a silicon surface barrier detector, an amplifier and a single channel analyzer. The window of the analyzer is adjusted to view the 5.15 MeV energy peak of ^{239}Pu . Field

* Maximum permissible concentration

experience at Hanford Project indicates these monitors perform extremely well. The analyzer is very stable and does not require frequent adjustment. The surface barrier detector has very low response to gamma background and, therefore, does not require any lead shielding.

B. DEVELOPMENT OF SENSORS AND RELATED EQUIPMENT

1. Low-Level Neutron Flux Instrumentation

C. N. Jackson, N. C. Hoitink, N. S. Porter,
R. C. Weddle (M00) and D. C. Thompson (931)

As previously reported, a temperature exposure of 900 °F for 720 hr on a "cable only" portion of an assembly caused no significant deterioration in the "breakdown pulse" characteristics on a 9-ft coaxial cable. Subsequent short-term temperature tests of cable seals and connector terminations of this same type of cable indicated substantial degradation of "breakdown pulse" characteristics with time. These results (only partially developed at this time) indicate that a problem exists at the seal and connector terminations.

Additions of ^{60}Co to the BNW gamma irradiation facility increased the source strength to 6.1×10^5 Ci; thus, the available dose rate essentially doubled at the various test positions.

2. Pressure Sensors for Sodium Service

R. D. Crosier (C00), K. O. Creek (932) and M. O. Rankin (933)
189 Task 12169/ID-C, LMFBR Element No. 4

Approximately 4500 hr of testing at 625 to 875 °F have been accumulated on a transducer rated at 600 °F and 250 psia.

A total of 1406 hr testing at 300 to 800 °F have been accumulated on a pressure transducer rated at 30 psia and 100 °F.

A total of 2409 hr at temperatures 300 to 825 °F have been accumulated on a transducer rated at 900 °F and 30 psia.

3. Eddy Current Probe Development for In-Core Flowmeter*

J. Brewer (ANL) and R. L. Brown (MOO)

Tests continued to determine an optimum probe configuration and coil wire size. The 1/8-in. depth coils continue to exhibit excellent reproducibility and stability. Feasibility of using zirconium-copper alloy was investigated. Grain growth of the copper and high oxidation rate of zirconium do not encourage use of this material for high temperature coils.

4. In-Reactor Coolant Temperature Sensors

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

Hot cell postirradiation testing has been completed on about 30 1/6 in. OD thermocouples built to Hanford specifications, which have been irradiated in EBR-II to slightly above 10^{21} nvt fast and at 1050 °F. The time-response tests have not yet been analyzed, but there does not appear to be any significant change. The thermal calibration showed that the error band was from +6 °F to -12 °F over a temperature range of 200 to 1400 °F. MgO insulation resistance has decreased by four to six decades; however, this change will not be enough to cause calibration errors.

During this reporting period, all planned hot cell testing for thermocouples irradiated in EBR-II capsules BT-3 and BT-4 was completed.

* Reported by M. J. Reib (932)

The hot cell tests completed this month included thermal calibration, insulation resistance, time-response and wire resistance.

The considerable volume of obtained data will require several months of fairly intensive and careful effort to be evaluated fully. Details will be presented in reports during the coming months.

C. OTHER INSTRUMENTATION AND CONTROLS WORK

1. Heat Transport System Control Study

R. A. Harvey and S. A. Hunt (823)

A simplified hybrid-computer simulation of the Heat Transport System was prepared for use in studies of control system stability. A frequency response comparison with the more detailed hybrid-computer simulation indicated adequate agreement in response characteristics for gross control stability studies, but not for fine tuning of the controllers. Major control philosophy and controller adjustment trends can be determined from the simplified simulation. An alteration to the model is in progress which will change a heat exchanger from a single-node to a two-node model.

2. Test Channel Instrumentation Analysis

J. J. Regimbal (823)

Additional analyses have provided a better definition of the protective coverage furnished by in-channel sensors against the consequences of accidental blockage of flow in a high-heat-flux fuel assembly. Sodium-bonded, carbide fuel with a lineal heat rating of 43.5 kW/ft was chosen as the test model. To facilitate computing, the test assembly was assumed to have the same geometry and dimensions as an FTR driver assembly. The channel sodium flow rate was adjusted to produce the temperature

rise across a hottest driver pin ($\Delta T = 406$ °F). Of course, the analysis is significantly influenced by the resultant 3.5 factor increase in steady-state coolant velocity.

Results of the study appear in Figure 4.1. Blockage accidents are specified in terms of a steady flow reduction to a postulated final, stabilized level. Each curve traces the accident conditions which barely produce coolant boiling despite detection and protective scram. Flowmeters were set to trip at 55, 50, and 40% of full flow. Thermocouple trips were set at 30 °F above the steady-state outlet temperature of 1006 °F. Thermocouple time-constants of 0 and 2 seconds were examined.

It can be concluded that adequate protection is furnished against blockages of 70% or less which take place over about one second by a very fast thermocouple or by a flowmeter with a rather relaxed trip setting. The 2.0-second thermocouple furnishes about the same protection as is computed for the case of oxide fuel.⁽¹⁾

1. J. J. Regimbal. FTR Protection with In-Channel Instrumentation, BNWL-1257. Battelle-Northwest, Richland, Washington, October 1969.

FIGURE 4.1. Protection Against Boiling in a Carbide Fuel Test Channel

CHAPTER V. SODIUM TECHNOLOGYA. ANALYTICAL METHODS FOR SODIUM AND COVER GAS*

R. L. Dillon (A00)

1. Sampling and Analysis of Radioactive Sodiuma. High Level Analytical Facility Design

Analytical procedures are well established for the determination of impurity levels in nonradioactive sodium. In order to adapt these procedures to the analysis of highly radioactive sodium coolant a mockup facility is being constructed. This facility will consist of two adjacent portions, one containing an air atmosphere and the other a pure argon atmosphere. Since its purpose is to test each analytical operation and each piece of apparatus under conditions simulating actual hot cell operations, it will be equipped with master-slave manipulators, viewing windows, air locks and a purification system for the argon atmosphere.

The newly constructed mezzanine in EDL-146, 324 Building has been chosen as the location for this facility. Movement of the air-atmosphere portion of the mockup from EDL-102 to EDL-146 has been started and should be completed during the next report period. Site preparations necessary for installation of the inert-atmosphere portion will be completed within the next month. Construction of the inert-atmosphere mockup cell has begun. Current estimates forecast completion by mid-December.

b. Remote Sampler Development

The small components evaluation loop and bypass sampler were shut down October 10 as scheduled and moved to Building 335. Alterations are being made to the sampler while it is out of service for the move. The top sampler is being changed

*Reported by W. R. Wykoff (922)

to use an "L" shaped instead of a "U" shaped sample tube. This will make possible more uniform cooling of the sample and simplified access for crimp-sealing. Also a permanent attachment is being provided for leak checking the sample loop.

As part of the program to evaluate crimp sealing as a tool for remote sampling three pieces of sodium sample tubing from the samples taken with the bypass sampler were double crimped and are being stored in air. Later they will be checked for oxygen penetration through the crimps.

2. Selection and Testing of Cover Gas Instruments and Sampling Systems

Evaluation of mass spectrometric and gas chromatographic equipment suitable for monitoring impurities in FFTF sodium cover gas has been completed and the gas chromatographic approach selected as best suited. This decision has shifted emphasis in the development effort to the sampling area and the solution of the sodium vapor removal problem.

The laboratory phase of the program has been reactivated after a move from Building 3706 to Building 325. Problems encountered earlier in obtaining the desired temperature profile on the current vapor trap design have been at least partially overcome in the relocated equipment through a change in the design of the furnace which places the bottom of the packed section of the trap even with the upper end of the furnace. As a result, significant improvement in trap performance has been observed with a 19-in. packed column reducing the sodium vapor content of a $1000 \text{ cm}^3/\text{minute}$ stream from a 650°F pot to 59 ppm. This is still short of the design goal but represents a large step in the proper direction. Development and evaluation are continuing.

3. Sodium Studies Facility Service Analysis

a. Carbon Analysis

The helium diffusion cell has been installed and is being used to purify helium for the carbon analysis. Preliminary tests show the purity of the helium to be as good as that obtained from the No. 314 purification system.

b. New Analytical Methods

Virtually all of the experimental work has been completed for the new calcium and halogen (chloride) methods. A modest atomic absorption spectrophotometer has been set up in the Inorganic Analysis Section. About 0.1 ppm of calcium can be detected with this instrument. The halogen method is less sensitive as it can only detect about 0.5 ppm of chloride.

4. The Analyses of Particulates in Sodium

Scanning electron beam microscopy has been selected as one of the first methods to be scouted as a means of measuring particulate matter in sodium.

5. The Analysis of Carbon Species

Scouting experiments on the measurement of carbon species in sodium have been initiated using the aqueous hydrolysis approach. A gas chromatograph has been set up and calibrated for carbon dioxide. The detection limit is presently about 0.1 ppm.

6. Selection and Testing of In-Line Instruments

a. Carbon Meter

Test of the carbon meter continued this month. By watching the carbon meter response to three impurity injections into the sodium purification loop some insight into the behavior of graphite in sodium at 600 °F was gained.

The observations are as follows: (a) when graphite is put in pure sodium the carbon meter does not respond; and (b) when oxygen is present with the graphite, the meter responds. Furthermore, the extent of the carbon meter response depends at least qualitatively upon the oxygen level in the sodium and upon the amount of graphite exposed. The role played by the oxygen may be as a catalyst to speed the dissolution or sodium attack of the graphite, or perhaps the oxygen may combine into a Na-C-O species required to produce "high" carburizing potential from graphite.

The above observations at 600 °F are identical to 1400 °F phenomena. This is encouraging and it provides further evidence that measurements by the carbon meter probe at 1400 °F are usefully and uniquely related to carbon-sodium phenomena at other temperatures.

b. Oxygen Meter

Revisions made in the oxygen meter probe to eliminate the ambient temperature sensitivity were successful. The overhauled cell operates satisfactorily and was responsive to the oxygen impurity injection mentioned in Section 2.1 above. However the emf of this cell is lower and the sodium temperature coefficient is higher than is common.

B. COOLANT AND GAS TECHNOLOGY DEVELOPMENT

1. Examination of EBR-II Piping Specimen*

R. L. Dillon (A00)

Gamma scans of the EBR-II pipe described last month were started. Apparently a substantial fraction of all isotopes counted (^{137}Cs , ^{54}Mn , and ^{60}Co) was in the sodium phase, i.e., count rates for all isotopes were low after the sodium was removed from the pipe. These results for Mn and Co are quite

*Reported by W. R. Wykoff (922)

surprising and in conflict with results obtained with samples from the Radioisotopes Transport Loop. Evaluation is continuing.

2. Sodium Vapor Trap Design*

R. L. Dillon (A00)

An analysis was made of thermodynamic and kinetic phenomena affecting the physical state of sodium vapor in a trap to be used to remove the sodium from a vapor phase being sent to a gas chromatograph. Results showed that the design of a fog-free vapor trap, while possible, would be difficult. The trap would be subject to stringently controlled operating conditions. It appears that a trap incorporating a defogging device would be more practical.

3. Development of Nondestructive Decontamination Processes*

R. L. Dillon (A00)

Work was started to determine (on a qualitative basis) if a surface can be decontaminated simply by heating and exposing it to a vacuum. The concept is based on the premise that the radioactivity is adsorbed on the surface. If effective, it would be a process with many attractive features. No results are available as yet as assembly of the experimental apparatus has just begun.

4. Oxidation Decontamination*

R. L. Dillon (A00)

Decontamination tests continued with emphasis on increasing the decontamination factors per steam oxidation and oxide film removal cycle. Samples used were the tube samples from the RTL just upstream from the heat exchanger. The Type 304 SS loop specimens were exposed to flowing sodium at 1300 °F for 6000 hr prior to the decontamination tests. ⁶⁰Co and ⁵⁴Mn present in the loop sodium were deposited on the tube specimens and subsequently diffused into the material.

*Reported by W. R. Wykoff (922)

The samples were counted and then oxidized in a steam atmosphere at 100 psi and 550 °C for 96 hr. They were cycled through four alkaline permanganate-citrox decontamination cycles. The decontamination factors for ^{60}Co were increased significantly over previously reported values. This increase is attributed to the complete removal of the oxide film formed during the steam oxidation step. The decontamination factors for the ^{54}Mn were still low, being a maximum of 5.

5. Sodium Removal Studies for the FFTF Reactor Vessel*

R. L. Dillon (A00)

Tests are being conducted to estimate the potential for caustic stress corrosion cracking in the FFTF reactor vessel. This Type 304 SS vessel (which will be exposed to a high temperature 96% N_2 -4% O_2 atmosphere with traces of water vapor) is to be able to recover from four sodium spills without replacement. There are no caustic stress corrosion 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. As reported in the last monthly report, the unsensitized samples were exposed for 120 hr without failure. The sensitized samples completed a 120 hr test run.

C. SODIUM FACILITIES

1. Sodium Chemical Technology Facility*

R. L. Dillon (A00)

Functional requirements for the Prototype Applications Loop sodium sampler and cover gas system have been prepared. Many of these plans will be applied directly to SCTF sampling

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

and cover gas system design. Sodium sampling techniques for SCTF are: flow through tube, vacuum distillation, on-line particulate filtration and a modified plugging meter with a removable orifice plate. The primary instrument for monitoring impurities in the cover gas will be a chromatograph. The cover gas will be sampled using a recirculating method rather than a bleed-and-sample method.

2. Cover Gas Evaluation Loop*

R. L. Dillon (A00)

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

Final assembly and installation of equipment on the loop and modifications to the glovebox are proceeding with an estimated completion date in late November. Checkout, initial operational tests, and emergency procedures have been written. A new sodium sampling system will be used on CGEL to avoid air contamination of the loop when a sample tube is installed. The sample tubes will be prefilled with high purity sodium on the sodium purification loop before attaching to the CGEL.

3. Prototype Applications Loop*

R. L. Dillon (A00)

Detailed design work is continuing on PAL at a satisfactory pace. A rough draft Quality Assurance plan has been written and distributed for comment. Purchase orders have been prepared and issued for most of the materials and equipment that can be identified at this time.

The stress analysis of the PAL piping system has begun and the first run will be made on the computer next week.

*Reported by W. R. Wykoff (922)

4. Thermal Convection Loops*

R. L. Dillon (A00)

The thermal convection loops are being operated to determine the transport characteristics of corrosion product isotopes. Loop 7R (1400 °F max/940 °F min) has completed 3050 hr of its scheduled 5000 hr run. Loop 10 (925 °F/440 °F) has completed 1870 hr of a scheduled 2416 hr period. Both loops were restarted without incident following the recent power outage.

5. Small Component Evaluation Loop

J. D. Berg (954)

The system operated at 600 °F and 10 gpm to provide sodium for the remote sampler loop until 10-13-69. At this time, the sodium was drained to the dump tank in preparation for relocating these two facilities (SCEL and Sodium Sampler Loop) from Building 314 to the new Sodium Facilities Building (Building 335). Both facilities have been relocated into Building 335 and are being prepared for return to operational status.

6. Small Heat Transfer Loop

J. D. Berg (954)

The 7-pin, loose-wire wrap, fuel cluster was removed from this facility on 10-13-69 in preparation for a calibration check of the test section inlet and outlet temperature elements. During this operating period, the cluster accumulated 264 hr of exposure at 1150 °F, 19.5 gpm (≈ 31 fps). Following cleaning, the element was returned to the test sponsors for disassembly and examination. Meanwhile, the dummy test section which contains two platinum, platinum-10% rhodium (reported last month) Type "S" thermocouples was installed in the loop and data obtained for calibration of the in-place RDTs and type "K" (CA) thermocouples. For this test the loop

*Reported by W. R. Wykoff (922)

was placed in operation at 400 °F bulk sodium temperature and output of the above TCs recorded after each 100 °F increase to 900 °F. Following this effort, which was completed by 10-23-69, the loop was shut down for removal of this equipment and to complete preparations for the next seven-pin electrically heated simulated fuel test.

To reduce the noise problem on signals going to the data acquisition system, only shielded leads will be used. Installation of these shielded leads is now complete.

Performance of the new power meter was erratic during the last heat transfer test. A dummy electrical test section was installed to investigate the problem, and it has now been determined that the Hall Effect module is defective. The complete instrument is being returned to the manufacturer for repair.

7. Static Sodium Pots

J. D. Berg (954)

Testing of pressure transducers in static pots No. 1, No. 3, and No. 4 was continued until 10-19-69. At this time, each pot was drained and the system secured for preparation of the facility to move from Building 324 basement to Building 335. All three test transducers were removed and cleaned prior to storing for reinstallation after the facility is relocated.

8. Sodium Purification Loop

J. D. Berg (954)

The main loop of this facility was operated continually during this report period. The plugging meter loop (PM), however, was shut down for the direct reading O₂ probe replacement. Following this installation, the new probe was placed in operation at the same conditions (600 °F, 0.25 gpm) as the old probe. Meanwhile, the facility remains at test conditions and has now accumulated more than 14,000 hr of operation.

9. Fission Product Screening Test Loop

J. D. Berg (954)

Installation of the new sodium cold trap was completed the second week in October. A thermocouple and trace heater check was then conducted to ensure instrument integrity. A test procedure is being prepared to permit checkout of this equipment during operation.

10. Sodium Facilities Building (Building 335)

J. D. Berg (954)

This building has been completed and three of the four loops planned for this building have been relocated.

11. FFTF Sodium Facilities Building (Building 335)

K. D. Hayden (954)

The building is essentially complete and paperwork is being processed to close out the project.

12. LMFBR Core Segment Development Facility (Building 336)

K. D. Hayden (954)

Design work by Vitro is currently 85% complete, with 100% completion scheduled for December 15. Emphasis is being placed on completion of design for the Fixed Price Contract portion of the project (building and services) so this can proceed at an early date.

13. Fast Reactor Thermal Engineering Facility

K. D. Hayden (954)

The design of the Fast Reactor Thermal Engineering Facility is about 35% complete.

Bid review on the 1.5 MW dc power supply resulted in contract negotiations with the General Electric Company.

CHAPTER VI. CORE DESIGNA. CORE MECHANICS CONCEPTUAL DESIGN, DEVELOPMENT, AND TESTING1. Radial Restraint Design and Developmenta. Computer Code Development Support

L. 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, with the latter one under development.

Bar and creep tests under pure bending are in progress to demonstrate the degree of similarity between tension creep and compression creep. At the end of this report period, two tests had been completed.

b. BEAM CREEP Code Development*

W. H. Sutherland (COO)

The elastic portion of the BEAM CREEP Code has been completed and checked out. Production checkout on a typical flow duct has been started. The flow duct selected for checkout purposes was a Type 316 SS cold-worked driver duct located in Row 6. The environmental conditions at a time of 400 days were provided by using the computer code TANDBM.

The temperature and swelling loads result from the transverse gradients that exist and are computed from code input which consists of mid-point values of temperature and swelling strain gradients. The finite element idealization used is essentially the same as previously used in performing an analysis with the MIT STRESS. Three types of cross-sections

* Reported by P. D. Cohn (832)

were used to idealize the driver ducts and the input for these sections was made to specifications in WARD-FRA-25.

For two cases where the swelling and thermal effects are considered separately, the maximum displacement induced by swelling is about 94 mils whereas the maximum induced by temperature is about 7 mils. For unrestrained elastic flow duct displacements when the restraint mechanism is released, the maximum displacement caused by temperature is about 0.26 in. and the maximum caused by swelling is about 1.58 inches. The maximum reaction for the combined loads case occurs at the support point above the core and is equal to -2319 lb.

c. Core Restraint Methods

G. R. Waymire (811)

The recent core structures study group investigated postulated conditions which would occur as a result of differential stainless steel swelling when the core was released for refueling. Assumptions utilized included:

- All ducts except the closed loops have essentially equivalent stiffness.
- Frictional loads are disregarded.
- Closed loops were assumed to be rigid.
- Deformations are based on agreed-to swelling model calculations.

With these basic assumptions, two cases are being studied. One case assumes that a restraint was applied by the movement at the top of the core in the outer reflector row. The second case assumes zero restraint by the shielding rows.

The study provides a "first-cut" idea of the problems involved when the core is released for refueling. Problems with compaction of the core are also evident.

2. Control Rod Design

T. J. Bennett (913)

The W control rod design study (an attachment of W letter WARD-FR-98, November 3, Georges to McDonald) has been received by BNW. FFTF engineers are now reviewing this report.

B. CORE PROCESS TECHNOLOGY

1. Hydraulic Core Mockup Tests of Potential Inlet Plenum Arrangements

H. Leigh (954), P. M. Jackson, G. W. Riedeman, and J. Spalek (COO)

The one-third size hydraulic inlet plenum model testing has been performed with series Va through VIIla test configurations as shown in Figure 6.1. This testing was performed to evaluate the effects of various plenum height-to-diameter (H/D) ratios upon fluid mixing and distribution.

Series V through VIII tests are directed toward evaluation of plenum arrangements which the Reactor Plant Designer has under consideration. These arrangements include a liner to reduce the plenum diameter, a flat plate in the plenum bottom, and an expansion spool piece. All are used in various combinations to alter the height-to-diameter ratio of the plenum. This H/D ratio for each arrangement is shown on the drawing. These arrangements did not utilize the basket-strainer component.

The reduced diameter and the "no basket" condition (Series V to VII) has had the general effect of causing an unbalanced core flow distribution when the downcomers are rotated to the 10 degree test position. At the 0 degree position, flow distribution appears to be uniformly balanced and not significantly different from previous tests with the basket arrangement.

SK-3-15419

FIGURE 6.1. One-Third Size Hydraulic Models of Inlet Test Configuration

2. Fuel Assembly Design

E. G. Stevens (913)

The wire-wrap pin spacer system has been baselined and approved by RDT for use in the first core driver fuel assembly.

The final BNW issue of CCDD-35, Conceptual Component Design Description for the First Core Fuel Assembly Component, as approved by RDT (with comment) has been completed. Distribution of this document as BNWL-500, Volume 35, is in progress.

Fuel pin bundle compression tests with the CCTL Mark I reassembled 217-pin bundle were run by BNW in the new compression test rig. Results indicated that the apparent "spring travel" was approximately 30 mils rather than the previously extrapolated approximately 64 mils based on a crude, horizontal bundle test. The effect could be that less available area for fuel pin swelling exists within the duct resulting in an adjustment of the predicted maximum achievable burnup before bundle-to-duct interaction.

A revised upper retainer grid design and wire attachment method has been proposed by WARD. Supporting data have been requested to permit an in-depth evaluation.

Surveillance and in-service inspection requirements for the fuel assembly have been approved within BNW. A request for RDT approval to baseline these requirements is to be made shortly.

A WARD-recommended listing of specific design criteria for the fuel assembly has been received and is being evaluated. The purpose is to document all of the basic reference parameters upon which the design is based.

3. Fuel Assembly Testing

a. Sodium Flow Tests

T. J. Bennett (913)

The purpose of these tests is to provide design verification and identify long-term failure modes in flowing heated sodium.

1. CCTL Mark II. A full-scale model of the FFTF reference design will be tested 9000 hr at conditions of maximum reactor flow, maximum cladding surface temperature and proposed coolant purity conditions to provide verification of the mechanical design and evaluate the effect of potential long-term failure modes. Fabrication of the assembly was held up to permit the resolution of specific test features and design changes between BNW and the Reactor Plant Designer.

2. 7-Pin Loose Wire Wrap Assembly. A small assembly test to evaluate the potential for excessive mechanical wear in FTR fuel assemblies caused by the movement of broken and loose wire wrap spacers. The test has been suspended. The test had been performed for approximately 1800 hr in sodium (flow rate 30 ft/sec at 1150 °F).

b. Heat Transfer Characteristics of 7-Pin Bundle

J. M. Yatabe (913)

The purpose of the task in FY 1970 is to perform sodium heat transfer testing with multipin models of FFTF fuel assemblies to determine local cladding temperature profiles in normal and structurally distorted FFTF fuel assemblies. The preliminary 7-pin bundle test, to evaluate the temperature difference between edge and central coolant channels and the potential hot spot under a wire wrap spacer, was completed, and a retest performed. Post-inspection disassembly of the test section indicated that a wire wrap had rotated to change

the positions of the thermocouples junction from immediately adjacent to the pin sheath to a position directly in the coolant stream. This inspection result voids the preliminary test conclusion that no cladding hot spot exists directly under a wire wrap. Corrective measures to prevent this rotation will be made in the next test assembly. Other post-inspection results indicate that other rotational shifts in wire wraps occurred; no distortion of duct or pins from their original configuration occurred, and a cladding thermocouple failed at the electrical head by shorting. The next test assembly to evaluate the effects of increased duct-bundle spacing on temperature profiles is 75% completed and undergoing the final electron beam seal welding.

c. Coolant Mixing and Flow Distribution Test

J. M. Yatabe (913)

The objective of this subtask is to provide flow distribution of the FTR fuel assembly design. These tests will complement and extend the range of the heated bundle tests performed under Task TH-A, "Heat Transfer Characteristics of Fast Reactor Fuel Assemblies".

Fabrication of the first 7-pin mixing and flow using noise analysis and absolute resistance measurement technique distribution test assembly is 60% complete. The test assemblies will exactly model the first 7-pin heated bundle test configuration. Fabrication of the instrumented pin calibration facility is 75% completed.

d. Sodium Erosion Tests

D. J. Oakley (913) and A. B. Webb (954)

The sodium erosion test of the hydraulic balance and fuel assembly seat, TI-B, is now in operation at LMEC. TI-D is currently being fabricated. Initial data have been obtained from Test TI-B at 900 °F and 750 °F. Testing is continuing.

4. Poison Section Irradiation Development Parameters

T. J. Bennett (913)

BNW has transmitted the following recommended irradiation development parameters to the RPD for comment:

One 7-pin bundle of boron-carbide-containing pins will be irradiated in EBR-II. BNW plans to use the following irradiation parameters in this test:

<u>Number of Pins</u>	<u>Pin Average Heating Rate (watts/gram of B₄C)</u>	<u>Max. B₄C Temp. (°F)</u>
3	100	1230 °F
2	80	1100 °F
2	60	1030 °F

Basis for the above temperature versus heating rate correlation is the CSDD No. 33. The value of 100 watts/gram corresponds to that of the third row of the FTR, and 80 watts/gram corresponds to the fourth row. Recent BNW calculations together with results from the critical experiment program (BNWL-967) in support of the FFTF indicate that there is a significant increase in the reaction rate of poison material (B-10) in the reflector region. The maximum heating rate in the reflector region corresponds closely to the heating rate in the third row of the FTR, i.e., 100 watts/gram of B₄C.

5. Fuel Performance Analysis

a. Hybrid Computer Simulation

R. D. Leggett (750) and C. R. Cole (MOO)

The 2-zone and 3-zone sintering models were combined into a single program on the Hybrid computer so that either model can be selected at will. The program output now includes fuel structure (radial or axial), temperature profile (radial or axial), fission gas release (for a given axial segment as well as total), and internal gas pressure. The program will now

handle a radially varied specific power profile such as is required for thermal flux irradiations. Cladding swelling is also being programmed into the Hybrid.

A script was prepared to accompany a movie illustrating "Hybrid Simulation of FTR Fuel Pin Structures" that was prepared several months ago. The movie will be shown during the presentation of a paper "Hybrid Computer Simulation of Fuel Pin Performance" at the ANS Winter Meeting in San Francisco.

b. Combination of SINTR and EXPAND

D. C. Bullington and E. O. Ballard (750)

The computer programs SINTR and EXPAND were combined in order to predict axial fuel swelling profiles and to predict the burnup required for fuel-cladding interaction. The combined code will be known as SIEX.

The influence of initial fuel density was studied for initial densities of 87.5% and for the following conditions:

Fuel Surface Temperature Bottom	°F	946
Fuel Surface Temperature Top	°F	1032
Peak Linear Power	kW/ft	4.8
Pellet Diameter	in.	0.194

The decrease of 6.5% in initial pellet density reduced the radial expansion by 1.1%, representing a decrease in volumetric expansion of 3.3% after a burnup of 100,000 MWd/tonneM.

c. IRRADIATION TEST FACILITIES

1. Design of 2-MW Simple Unitized Modular Closed Loop System
M. A. Vogel (924)

Atomics International (under the Technical Direction of Westinghouse) completed the concept definition of the 2-MW Simple Unitized Modular Closed Loop System.

Normal system operation involves circulation of sodium by means of electromagnetic pumps through the in-reactor tube containing the irradiation test section. Heat produced in-core is transported by means of the sodium stream to the shell side of the intermediate heat exchanger where it is transferred through the tubes to the secondary nonradioactive coolant circuit. The secondary electromagnetic pump circulates sodium through the tube side of the intermediate heat exchanger and transports the heat absorbed to the air dump heat exchanger for dissipation to the environment.

Engineered safeguards to ensure cooling of a test after any single loop fault leading to a reactor scram include an auxiliary electromagnetic pump which continuously provides approximately 10% of the required test flow. The check valve arrangement on the discharge of the two pumps provides circulation for decay cooling from either pump should the other system fail. Backup for secondary system failure is provided by natural circulation from the intermediate to the dump heat exchanger and by the heat-loss capability of the primary system.

The basic component arrangement involves a modular stack which is unitized for initial installation. The modules are designed to permit remote removal by cutting should contamination levels require this. The reference normal mode of repair is by personnel contact with the equipment in place in the cells after a decay period and after draining the radioactive sodium.

2. Prototype Closed Loop

a. Design of In-Core Facilities

J. W. Kolb (914)

The objective of this task is to perform the design of the Prototype In-Core Closed Loop Tube and Top Head Jumper Piping. This includes performance of analysis and interface studies,

and definition of required development and testing to provide maximum assurance that the design is responsive to the final design of FFTF Closed Loops. It is intended that the prototype design developed from this work will be fabricated under a separate contract for prototype development testing which will provide the technical information necessary to finalize and verify design of FFTF Closed Loops.

A Work Plan covering this task has been prepared by Westinghouse, and is being reviewed by BNW. The new plan (WP 33000-72) will go into effect at the completion of conceptual design work plan 60-01 funded under BDR-362, and will be based on this conceptual work.

3. Closed Loop Test Characteristics

R. E. Keyes (812)

The draft of Document BNWL-1251, FFTF Closed Loop Test Characteristics, has been completed and is being circulated for comment. The characteristic curves were revised to suit the latest information on pressure drop through wire-wrap tube bundles. The results indicate that a 2.5-in. test diameter is suitable for pin diameters of interest for 37-pin and 19-pin bundles. A 61-pin bundle would be feasible only at unusually small pin diameters and low linear power.

4. Closed Loop Tube Assembly

a. Felted Metal Insulation Tests

K. R. Wheeler (A00) and S. M. Gill (962)

The design of the tubular sections of the Closed Loop Tube Assembly requires controlled thermal conductivity between the loop sodium and the reactor sodium, and between the two channels of the loop. To achieve this it is planned to use double-walled stainless steel tubing with a sandwich filling of felted stainless steel.

The purpose of this study is to determine the combination of density and bonding parameters for the composite wall structure which gives the required thermal conductivity of approximately 2 Btu/hr/ft/°F at 1200 °F. Tubular sections will be evaluated in air and flowing sodium.

Test pieces to evaluate the effects of insulation density and bonding on thermal conductivity are being manufactured off-site with delivery expected in three weeks. Flat sheets of varying density insulation to be used in additional test section fabrication will be delivered in mid-January 1970. The manufacturer has been unable to form the 60% density insulation sections into the required cylindrical shapes by means of cold-roll-forming. Other shaping methods will be explored.

D. OTHER CORE DESIGN STUDIES

1. Development Problems for Advanced Cores*

P. D. Cohn (832) and L. M. Finch (810)

A study was performed to define the general characteristics of second generation cores for the FTR and to identify the reactor components subject to replacement upon installation of these cores.

The second generation fuel for the FTR is expected to be first an advanced oxide followed by a carbide ceramic pin design.

A second generation fuel should have "advanced" characteristics when compared to the initial oxide core. These characteristics will advance the testing capability of the FTR by (a) improving neutron flux or temperature performance or by (b) improving the economics and availability of reactor operations.

* As reported in BNWL-1246, FFTF Technical Progress Report, October 1969, Battelle-Northwest, Richland, Washington.

The oxidized fueled core is composed of 91 active components within the fueled zone which is surrounded by 15 control rods and 111 reflector components. The core is an array of vertical, hexagonally shaped prismatic components with dimensions of 14 ft length and 4.48 in. across the flats of the hexagonal section. The core volume (900 liters) is less than the initial core volume, thus providing the desired increase in neutron flux to 0.9×10^{16} neutrons/cm²-sec. The closely arrayed pin design is dependent upon development of either radiation-induced swelling resistant materials or compensating operational techniques. The extended fission gas plena (compared to the initial core pin) will provide extended life for the fuel while reducing pressure-induced strain in the fuel pin cladding.

The carbide fueled core is composed of 61 active components within the fueled zone which is surrounded by 12 control rods and 96 reflector components. The core is an array of vertical, hexagonally shaped prismatic components with dimensions of 10 ft 10 in. length and 4.5 in. across the flats of the hexagonal section. The core volume (590 liters) is much less than the initial core volume providing a significant increase in neutron flux to 1.10×10^{16} neutrons/cm²-sec. The closely arrayed pin design is dependent upon development of either radiation-induced swelling resistant materials or compensating operational techniques. The vented fission gas design is a major development problem associated with this core design.

2. Upgraded Oxide Fuel Analysis

W. T. Wood (832)

The purpose of this study was to investigate an improved (increased thermal performance) oxide fuel and the effect that an alteration of the energy transport path would have on the

peak fuel temperature. The simplest mode of accomplishing this is the introduction of stainless steel wafers between each fuel pellet in the fuel column.

Preliminary results on the hottest pellet in the FFTF core are displayed in Figure 6.2. The mathematical model incorporated fuel sintering but excluded energy exchange with surrounding pellets. The results were obtained from the TIGER computer code in conjunction with external iterations.

FIGURE 6.2. Peak Fuel Temperature Versus Wafer Thickness

CHAPTER VII. FUELS AND MATERIALSA. FUELS AND MATERIALS EVALUATION1. Biaxial Stress - Rupture Properties of Type 316 Stainless Steel - R. W. Barker (722)

The objective of this program is to determine the effects of fast reactor irradiation on the biaxial stress-to-rupture properties of candidate fast reactor cladding materials.

a. Creep Tests

Biaxial stress-rupture tests on irradiated Type 316 SS are continuing with tests now on-line at 1000 °F. The biaxial creep tests are being interrupted periodically to obtain strain measurements. Preliminary evidence seems to indicate a discontinuous strain pattern. Little or no primary creep is observed.

b. Effect of NaOH

The need for interim examination of irradiated PNL fuel pins en route to long-term exposure has prompted a limited effort to determine if the sodium removal procedure to be used at the interim discharge will have deleterious effects on irradiated fuel pins while cooling. The procedure by which residual sodium will be removed from fuel pins requires the conversion of sodium to sodium hydroxide. Concern exists that the conditions favorable for caustic stress corrosion, such as temperature, stress, and strong caustics, may exist during the sodium removal cycle.

Simple tests were designed to determine if any effect could be observed on Type 304 SS from pseudo-typical sodium clean up conditions. Samples of sensitized and nonsensitized Lot "F", Type 304 SS were exposed to a hydroxide-saturated water solution for times to 100 hr. The corrosive solution was heated to estimated typical fuel pin cooling temperature,

~105 °C, during the exposure cycles. Optical examination revealed no evidence of attack on either sensitized or non-sensitized material.

Two sensitized Lot "F" biaxial stress-to-rupture test specimens were exposed to the heated corrosive solution for 100 hr. One specimen was mounted in a stress-rupture test rig and pressurized to achieve 10,000 psi maximum hoop stress during the exposure cycle. The second specimen was exposed nonstressed. After exposure, no damage to either specimen could be observed. After exposure to the heated caustic the specimens were rinsed in water, and dried. Both samples were later stressed-to-rupture at 24,500 psi and 1200 °F. Two similar sensitized but nonexposed (to heated caustic) specimens were stressed-to-rupture under like test conditions. Data from these tests are compared with data from like tests of as-received material. Little difference was observed in the results from the exposed and from the nonexposed sensitized specimens that does not fall within normally expected scatter.

It can be concluded that the sensitized test material was not degraded by the caustic exposure. Although the effect of caustic exposure on irradiated sensitized stainless steel is uncertain, it appears (from current work on biaxial stress properties of stainless steels) that the effects of irradiation are far more deleterious than are low temperature exposures to strong caustics. The possibility could exist, however, that fuel elements with higher surface temperatures may be susceptible to attack from caustic during the sodium removal cycle.

2. Mechanical Testing of Fuel Pin Cladding -

R. L. Fish and R. W. Barker (722)

The purpose of this program is to provide a basis for evaluating the combined effects of fuel-cladding interaction, fluence, and operating temperature on the post-irradiation burst and stress-rupture properties of fuel pin cladding.

Initial leak testing was performed on three specimens from PNL 1-18 which had been irradiated in the EBR-II reactor to a peak burnup of approximately 8500 MWd/tonne. Fissures were found in the cladding of a section taken from the uppermost 2 3/4 in. of the fuel column. Two other specimens from immediately above and below this specimen revealed no leaks in the cladding.

Replicas were made of the cladding surface where the leaks were detected. These replicas reveal cracking of the stainless steel along grain boundaries. Both transverse and longitudinal sections are being prepared from this specimen for metallography and electron microprobe work.

Tests were conducted to identify the agent responsible for the intergranular corrosion observed in fuel cladding from above midplane in PNL mixed oxide fuel pins.⁽¹⁾ The corrosive effect has been found to be a general condition (not intergranular in nature) having little effect on matrix properties. The corrosion product has not yet been identified.

3. Damage Analysis

J. L. Straalsund (722), H. R. Brager, R. L. Fish,
J. Bates (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.

1. R. L. Fish, FFTF Quarterly Development Progress Report, July, August, September, 1969. Section 2-e, Cladding Evaluation.

a. Swelling Behavior Experiment

An irradiation experiment is being designed to determine the swelling behavior of FFTF control rod and prototypical subassembly materials. Initial data from this experiment should be available by January 1971.

Three pins are being planned for this experiment. One pin will contain prototypical cladding, wire wrap and duct materials (20% C.W. Type 316 SS) to determine the effect of minor fabrication differences on the swelling behavior of these components. Current subassembly design assumes that swelling depends only on extent of cold work and is not a function of the method of producing the cold work. This pin is designed to test the validity of that assumption.

The second pin will contain candidate wear-pad material samples and bimetallic welded couples of pad and duct materials.

The third pin will contain tantalum control rod alloys. This pin will provide swelling data at temperatures and fluences of interest to the FFTF to aid in selection of control rod materials.

Progress on the design of this experiment includes formulation of the basic sample geometries, layout of the individual capsules used to hold the samples, and initial thermal analysis for capsule sizing.

b. Void Formation

Transmission electron microscopy studies of irradiated Type 316 and Type 304 SS are being continued to develop a phenomenological description of void formation. The essential

characteristics of the fluence and temperature dependence of void formation are illustrated in Figures 7.1 and 7.2. The void size increases nearly linearly with increasing irradiation temperature, but increases only slightly with fast fluence. Conversely, the void number density decreases with increasing irradiation temperature but increases with fluence. The decrease in void number density with increasing temperature appears to be more pronounced at low fluence than high fluence.

At high fluence, the void number density and size appear to be similar for AISI Types 304 and 316 SS but recent results suggest that the microstructures may be much different at low fluence. After irradiation to 6×10^{21} n/cm² ($E > 0.1$ MeV) at 700 °F, Type 304 SS contained a high concentration of voids with an average diameter of about 110 Å. There were however, essentially no voids observed in Type 316 SS after a similar exposure 2.5×10^{21} n/cm² ($E > 0.1$ MeV) at 700 °F.

4. EBR-II Irradiations

P. K. McDaniel (722)

The EBR-II irradiations program is to provide coordination and direction for the timely accomplishment of fast reactor irradiations of structural materials. EBR-II irradiations are in progress as scheduled. There were no subassembly discharges or changes. Three Mark B-n Type experiment pins are being constructed for insertion in EBR-II around January 1, 1970. These pins will be BNWL-B-41, BNWL-B-42, and BNWL-B-43. Their goal exposure is to be 3×10^{22} n/cm² ($E > 0.1$ MeV).

The objectives of these experimental pins are: (a) to provide differential swelling data on prototypical cladding wire wrap and duct; (b) to measure differential swelling rates of prototypical duct-wear pad couples, and (c) to investigate the swelling behavior of tantalum control rod alloys (Pin No. BNWL-B-43).

FIGURE 7.1. Void Size in Fast Reactor Irradiated Austenitic Stainless Steel as a Function of Fast Fluence and Irradiation Temperature

FIGURE 7.2. Void Number Density in Fast Reactor Irradiated Austenitic Stainless Steel as a Function of Fast Fluence and Irradiation Temperature

5. Subassembly PNL-2
L. A. Pember (750)

The PNL-2 (X032) subassembly was discharged from EBR-II at an estimated peak exposure of 47,000 MWd/tonneM. The capsules have been neutron radiographed and show no evidence of fuel pin failure. The capsules have been received at BNW. After the lot is gamma scanned, five capsules will be selected for shipment to LASL for destructive examination. Three capsules will be destructively examined at BNW.

6. Subassemblies PNL-9, 10 and 11
E. O. Ballard and L. A. Pember (750)

The data package for PNL-9, 10 and 11 is completed and is being typed. It is calculated that goal burnup of 89,000 MWd/tonneM peak can be achieved by all three subassemblies. Fuel-cladding interaction and pin-bundle-hexcan interaction is predicted to not occur for any of the pins in PNL-9, 10 and 11.

7. Subassemblies PNL-12 Series, EBR-II Run-To-Failure Tests
M. K. Millhollen (750)

The approval-in-principle for EBR-II run-to-failure tests was granted. BNW and ANL plan to conduct an integrated failure-testing program.

Preliminary planning at BNW has delineated a series of initial tests which involve an artificially defected pin surrounded by six non-defected pins. Pin failure could be controlled, relative to exposure, by notch defects placed on the internal surface of the cladding and by pressurizing the pin with helium during fabrication. Location of the artificial defect could be varied in each test in order to determine where on a fuel pin a defect during operation might cause the most serious consequences.

8. EBR-II/TREAT Testing

G. E. Culley and D. O. Sheppard (750)

The design of the EBR-II/TREAT capsule was modified to reflect design improvements achieved while using the two capsules fabricated for practice reencapsulations. The drawings are being updated prior to final approvals.

The EBR-II/TREAT hazards analysis is nearing completion with the effort during this report period being concentrated on the neutronics effect of the experiment on the TREAT reactor and the experiment response to the maximum accident condition.

9. Steady-State Irradiations

G. E. Culley (750)

Sections from the PNL-59-7 fuel pin that have been metallographically examined the fuel, which received an exposure of 69,000 MWd/tonneM at 13.5 kW/ft at section J-2, is in very good overall condition. The bottom section shows no chemical and mechanical interaction between the fuel and the insulator pellets. The most noticeable feature other than the cracked condition of the fuel is the dishing of the pellets. These pellets originally had flat ends but they acquired a "dish" of approximately 0.001 to 0.002 in. on each end.

Burnup affects the relationship between the test pin power and the reactor power by decreasing the amount of fissile material present. For unirradiated fuel the calculated value for the ratio is $0.81 \times 10^{-4} \text{ Watt}^2/\text{cm}^3$ which agrees reasonably well with the measured value of 0.9×10^{-4} . At a burnup of 18,000 MWd/tonneM, the calculated value decreases to 0.77×10^{-4} and at 69,000 MWd/tonneM it is 0.65×10^{-4} .

In order to perform transient tests with a reasonable degree of accuracy, it is important to know precisely the radial power profile and the ratio of pin power to reactor

power. The recent neutronics calculations have been giving good agreement with experimental results and it is believed that the analytical models and techniques have been sufficiently developed and understood to give the required results.

As reported previously, three of the five TREAT capsule head blocks developed leaks during hydrostatic testing. Arrangements for repairs on the heads are being made.

10. Fast Flux Pin Irradiation

a. 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 peak burnup at the end of EBR-II Run 38 (October 18, 1969).

<u>Subassembly</u>	<u>Burnup, MWd/tonneM</u>
PNL-3(X059)	12,800
PNL-4(X059)	16,600
PNL-5(X054)	28,000

The subassemblies were returned to the reactor following the 62.5 MW testing which took place during the first part of Run 38.

b. Subassemblies PNL-6, 7 and 8

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

Subassembly PNL-7 was charged into EBR-II at the beginning of Run 38B. The estimated peak burnup at the end of Run 38 is 1900 MWd/tonneM.

Pins for subassembly PNL-8 were shipped to the EBR-II for assembly and insertion. Goal exposure on these subassemblies is 80,000 MWd/tonneM peak.

Corrections and additions to the PNL-6, 7 and 8 data package were completed. Based upon the calculations made for fuel pin bundle and hex tube swelling it is predicted that PNL-7 and 8 will be able to reach goal burnup of 80,000 MWd/tonneM peak. Subassembly PNL-6 may be limited to 60,000 MWd/tonneM without interim examination since pin bundle - hex tube interference is predicted to occur at that burnup.

c. Instrumented EBR-II Subassembly (PNL-17)
M. K. Millhollen (750)

It was decided that weighted composite flux and power estimates based on the respective times of the mode of reactor operation would be used to predict the performance of the PNL-2 pins. These data are now being programmed into the combined SINTR-EXPAND code known as SIEX.

11. Irradiation of Hypostoichiometric $\text{UO}_2\text{-PuO}_2$ Fuel
J. W. Weber (750)

The status of the BNW-1 capsule series is summarized in Table 7.I. Capsules BNW 1-10 and 1-12 will be discharged on November 9 at the end of ETR Run 104. The capsules will be gamma scanned and then neutron radiographed at the MTR sometime in December. Results from these nondestructive examinations will determine whether irradiation of the capsules should be terminated or if they should be returned to the ETR for additional irradiation.

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

Issue Date: 11/1/69

Test Designation	Description(a) Fuel Form	O/M	Facility	Status			
				Goal Exposure MWd/tonneM	Actual Exposure MWd/tonneM	Discharge Date	
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) 92,200	(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) 96,500	(Est) 2-70	Discharged 2-69 for Interim Exam. Recharged 5-69

(a) 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

B. MATERIALS TECHNOLOGY

1. Uniaxial Creep*

A. J. Lovell, J. F. Bates (AOO)

The objectives of this program are to evaluate the effects of fast neutron irradiation and environment on the pre and post-irradiation uniaxial creep-rupture properties of candidate vessel and core structural materials for LMFBR and to establish the usable materials limits to ensure their safe and reliable performance.

The attempts to correlate the creep rate data to unirradiated Types 304 and 316 SS have resulted in the following expressions:

Type 304 SS

$$\ln \dot{\epsilon} = 49.70 + 6.87 \ln [\sinh(2.14 \times 10^{-5} \sigma)] - \frac{47,044}{T}$$

$$\dot{\epsilon} = (e^{49.7}) [\sinh(2.14 \times 10^{-5} \sigma)]^{6.87} (e^{-47,044/T})$$

Type 316 SS

$$\ln \dot{\epsilon} = 62.48 + 8.23 \ln [\sinh(1.79 \times 10^{-5} \sigma)] - \frac{58,870}{T}$$

$$\dot{\epsilon} = (e^{62.48}) [\sinh(1.79 \times 10^{-5} \sigma)]^{8.23} (e^{-58,870/T})$$

where

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

σ = applied engineering stress (ksi)

T = temperature of test (°K)

These expressions were obtained using the creep data available over a wide range of applied stress and test temperatures. More details on the data and method used will be forthcoming.

Tests were made on Type 304 SS, one at 1000 °F and two at 1100 °F, irradiated in Subassembly X018. The loss in rupture life for the 1000 °F test after an irradiation of

* Reported by J. C. Tobin (960).

$\sim 1.6 \times 10^{22}$ n/cm² (total) at 930 °F is more than a 30-fold reduction when compared to extrapolation of control data on solution-treated material. The strain during the test was very low, $\sim 0.3\%$. The indicated reduction in the minimum creep rate was about 3-fold. Tests at 1100 °F after irradiation at 1140 °F to 3.7×10^{22} n/cm² (total) do not show the marked rupture life losses with only three- and four-fold losses for the 30 ksi and 25 ksi tests.

Two 1100 °F tests were made on Type 316 SS irradiated at 1110 °F to 3.5×10^{22} n/cm² (total) in Subassembly X018. These two specimens show a marked reduction in rupture life somewhere in the 50-fold-plus range. The creep rates are increased, and the ductility, as indicated by the creep curves, is less than 1%. These results demonstrate a marked difference in irradiation response of these two alloys. In the unirradiated condition, the rupture life of Type 316 SS at 1100 °F is about five times that of Type 304 SS. However, after irradiation to approximately 3.5×10^{22} n/cm² (total) at 1100 °F, the rupture life of the Type 304 material is now about two to three times longer than that of the Type 316 material. This reversal in rupture life behavior further illustrates the significance of irradiation effects in stainless steels. Needless to say, the effects that can change the rupture properties so markedly are of interest in terms of fast reactor components, thus additional analyses are needed. Additional tests on both alloys are in progress. Also, post-test evaluations of the failed specimens will be initiated in the near future.

2. Tensile Studies*

A. L. Ward (A00)

The objective of this program is to provide mechanical property and metallurgical stability data on LMFBR vessel

* Reported by J. C. Tobin (960).

and core structural components, including weldments, in direct support of FFTF in order to define adequate lifetime performance under anticipated FTR service conditions.

The effects of irradiation and testing conditions on the tensile properties of AISI 316 and Type 304 SS are being evaluated. In general, the degree of irradiation hardening, in terms of increased yield strengths and reduced ductilities, increases to a maximum near 2.0×10^{22} n/cm² (total) and 960 ± 35 °F. At higher fluences, less hardening is observed due to the effect of the higher irradiation temperature on the rate of damage accumulation. The maximum yield strength of 75,900 psi at 800 °F constitutes nearly a 4-fold increase relative to unirradiated controls. No well-defined difference between Types 304 and 316 SS can be established in this analysis because of the irradiation temperature difference between the two groups of specimens; the Type 316 SS was located below midplane and the Type 304 SS above. Type 304 specimens show a peak strengthening at the same fluence level as the Type 316, but, possibly because of irradiation temperature sensitivity in this range, it shows a lower maximum yield strength.

3. Weldment Studies*

A. L. Ward, A. J. Lovell, L. D. Blackburn, J. F. Bates
(A00)

The objective of this program is to provide mechanical property and metallurgical stability data for candidate LMFBR vessel and core structural weldment materials in direct support of FFTF in order to define an adequate lifetime performance under anticipated FTR service conditions.

Four weld processes are being considered for the FTR vessel weld merit program.

* Reported by J. C. Tobin (960).

Characterization of the creep-rupture properties of as-received weldments is now complete.

Ductility, as measured by total elongation, of all weldments is lower than base metal for the conditions studied.

These results demonstrate that the creep behavior of welds in Type 304 SS can be significantly different from the creep behavior of base metal. Such property variations could be significant in affecting local deformation or ductility limits of welded structures under creep conditions.

4. Subcritical Crack Growth*

E. B. Schwenk (AOO), L. A. James (AOO)

The objective of this program is to determine the effect of temperature on the fatigue crack growth properties of alloys that are to be used in the FFTF.

Figures 7.3 and 7.4 show all the results to date on six crack-growth specimens RML-2, -3, RMT-4, -5, and F-8 all conducted at room temperature and RML-6 at 1000 °F. Figure 7.4 shows clearly the effect of temperature on crack-growth rate. It should be noted that for all the data shown, average net section stress levels were well below yield stress, the highest value occurring in RML-2.

The data on RML-3 and RMT-4 and -5 were discussed in a previous monthly report. The expanded ΔK scale of Figure 7.4 shows that specimen RML-2 load-cycled at 2 cpm and at a higher mean stress level displayed an increased crack growth rate (about double) in comparison to RML-3 and RMT-4 which were cycled at 400 cpm and a lower mean stress level. Additional tests under such conditions would be necessary to see if the increased da/dN values are consistent.

The effect of an environment such as humid air or salt water can also increase crack growth rates in other alloy

FIGURE 7.3. Fatigue Crack Growth Rate Versus Stress Intensity Factor Range (ΔK) for AISI 304 SS (annealed) at Room Temperature

FIGURE 7.4. Fatigue Crack Growth Rate Versus Stress Intensity Factor Range (ΔK) for AISI 304 SS (annealed) at Room Temperature and 1000 °F

systems and is apparently the result of a chemical deterioration occurring at the crack tip. Thus it is possible that the data in Figure 7.3 may reflect, in addition to a possible mean stress effect, a combined influence of cycle rate and laboratory humidity.

5. Radiation Effects on Tantalum Absorbing Materials for Control Rods*

A. L. Pitner (A00)

Tantalum capsule TA-4 was removed from the X042-A subassembly in EBR-II at the end of Run 38 and disassembled. When the inner capsule was removed from the outer cladding, extensive deterioration was evident. There were two breaks in the inner capsule cladding, and two other areas where reactions had obviously occurred. All the faults were located at meltwires and the Type 304 SS cladding. The condition of TA-4 caused concern for TA-5 which was scheduled for further irradiation, because both capsules were fabricated similarly and at the same time. Consequently TA-5 was also removed from the subassembly and neutron-radiographed. The results showed an offset in the inner capsule that could be interpreted as a cladding break. Plans for the disposition of TA-5 and the replacement tantalum capsule(s) for X042-B are presently being determined. A laboratory compatibility test of the melt wires in contact with tantalum and Type 304 SS at elevated temperatures is also underway.

6. B₄C - Sodium Compatibility*

L. R. Bunnell (A00)

Samples of B₄C pellets irradiated in a Hanford reactor to a burnup of about 4.5% have recently been examined. Observations are tabulated in Table 7.II.

In order to investigate further the cause of cracking in high density B₄C pellets exposed to sodium (previously attributed to residual fabrication stresses) two pellets, 85% and 96% TD, were heated to 2000 °C for 1 hr and slowly cooled. Examination

* Reported by J. W. Irvin (963)

TABLE 7.II. Observations on Hanford-Irradiated B₄C Pellets

<u>Sample Number</u>	<u>% TD</u>	¹⁰ B <u>Burnup</u>	<u>Temper- ature, °F</u>	<u>Observations</u>
1-15	99	4.71%	1130	Transverse section shows circumferential cracks extending inward a maximum of ~0.016 in. The crack separates burned-out material with almost complete grain boundary separation from material with as-fabricated appearance. No evidence of lithium-bearing compounds was visible.
1-13	99	4.67	865	Same comments as above, but a family of concentric cracks exists, with some transgranular cracking. Same crack depth.
1-12	90	4.64	1120	Transverse section shows cracks, both circumferential and radial, extending inward to a maximum distance of 0.071 in. No sharp delineation between surface burnout regions and core regions was visible. No cracking between grains was seen. No evidence of lithium-bearing compounds existed.
1-4	90	3.84	725	Same comments as above, circumferential cracks extending inward a maximum of 0.045 in. High magnification photos showed no connection between grains, which suggests that this is a non-typical sample, possibly hot-pressed at an unusually low temperature.
1-10	65	4.44	865	Radial and circumferential cracks extend inward about 0.073 in. maximum. No clear delineation showing surface burnout. No visible lithium-bearing compounds.

after exposure to sodium at 500 °C for 1000 hr disclosed that very little cracking had taken place. A slower cooldown after hot-pressing, or a further annealing step, may provide gains in pellet integrity and sodium compatibility.

CHAPTER VIII. FUELS RECYCLEA. FUEL TECHNOLOGY1. LMFBR Fuel Development

B. R. Hayward (723)

The NUMEC Task 7 irradiation test fabrication effort was slowed temporarily because higher priority was assigned to the FFTF fuel pin prequalification round and a reassessment of the task budget. NUMEC made sintering furnace equipment modifications to raise the O/M content of the pellets to meet specification. Sample batches were sintered but analytical results are not yet available. It is expected that this equipment change will also reduce the nitrogen contamination (~1200 ppm) reported last month. The updated NUMEC Engineering Test Plan was prepared and is now in the final approval phase. The data package is nearly complete but requires some component analysis results.

2. Analytical Chemistry

B. R. Hayward (723)

Additional analyses were received from GE and NUMEC on the Phase II analytical chemistry program. The compilation of results shows that a few scattered results are missing from GE and NUMEC. The ANS paper on results of this analytical program was drafted and is in final review by the authors.

3. Image Enhancement

B. R. Hayward (723)

Philco-Ford expects to complete within two weeks the processing of the five image enhancement specimens submitted last month. A topical report on image enhancement for FFTF technology is in the preliminary draft stage.

4. FFTF Fuel Procurement

J. F. Williams (712)

The status of the LMFBR fuel supplier prequalification contracts is as follows:

Babcock and Wilcox has completed cross blending of mixed oxides, green pressing, and sampling of green pellets. Four batches have been sintered; three of these have been released for loading, and the fourth is undergoing inspection. Nine fuel pins have been completed and are in final inspection. All metal components were released by QA and 30 first end welds are complete. Insulator pellets are complete. Initial shipment of pins is scheduled for early December.

NUMEC was released to cross blend PuO_2 and UO_2 and to press and sinter fuel pellets. Sintering trials were run and adjustments in the furnace atmosphere were made to obtain fuel of proper O/M. NUMEC's subcontractor for metal components received approval of procedures, and production of components has started.

United Nuclear completed three production sintering batches. The first two meet specification with grinding, the third meets specifications without grinding. Metal components are in the final stages of inspection. First end-cap welding will start as soon as inspection is finished and components are cleaned. Insulator pellets are completed, including final inspection.

Cladding tubing found defective by NUMEC and UNC was replaced by BNW. The defective material is being returned to BNW.

The Model 55 shipping container was approved by the Department of Transportation for increased Pu content. All information available was forwarded to the contractors.

5. Fuel Fabrication for Test Pins for Irradiation in EBR-II
W. E. Warden (733)

Fuel pellet fabrication for EBR-II subassembly PNL-6 was completed. The fuel is 30% enriched (^{235}U) in the standard EBR-II test composition of 75% UO_2 -25% PuO_2 . Six kilograms of fuel were sintered for the subassembly.

6. Effects of Oxygen-to-Metal Ratio on Mixed Oxide Fuel Pellets
H. T. Blair (731)

Calculations were made of the effect of changes in oxygen-to-metal ratio during sintering of fuel pellets on the plutonium content and the percent theoretical density as presently calculated. Assuming that stoichiometric starting powders

containing no moisture or other impurities are batched as 25 wt% PuO_2 - 75 wt% UO_2 , the plutonium content of the sintered pellet will vary inversely with the oxygen-to-metal ratio when calculated as a percentage of total oxide weight.

These calculations help explain the slightly greater plutonium content of previously prepared pellets which were prepared from powder having a 22.0 wt% plutonium content.

The sintered density of fuel pellets is presently reported as a percent of theoretical density. The theoretical density is a function of the oxygen-to-metal ratio. Therefore, the oxygen-to-metal ratio must be considered whenever calculating or reporting pellet densities in terms of percent theoretical density. The presently used value of 11.09 g/cm^3 is the theoretical density of stoichiometric mixed oxide. Prototypical FTR fuel pellets have an oxygen-to-metal ratio between 1.95 and 1.99. This range of oxygen-to-metal ratios and associated theoretical densities provides a range of 1.07 percent in the calculated percentage of theoretical density depending on what theoretical density is used. Reporting of sintered densities in terms of weight per unit volume rather than as a percentage of theoretical density is being considered.

7. Plutonium Dioxide Sinterability Testing

M. J. Barr (731)

Plutonium dioxide sinterability tests were continued. Test pellets made from PuO_2 originating from calcined plutonium nitrate and from burned plutonium metal were prepared for sintering. An observed decrease in PuO_2 surface area and water content, and an increase in green pellet density as the nitrate calcining temperature is increased is consistent with theory. The stability of the carbon impurity level from 450 to 1000 °C calcining temperature indicates that most of the carbon is in a relatively non-volatile form.

8. Pellet Diameter Control

J. D. Schaffer (740)

In view of the requirement to load CCTL Mark II pins with pellets having diameters larger than those loaded routinely, FQA devised a plan for sampling processed pellet lots which would detect, with a high degree of confidence, lots which include a small percentage of pellets having diameters outside of limits established by Fuel Element Development. These limits defined pellet diameters, which in the past horizontal loading efforts constituted a no-go (no-fit) condition. Applying the plan, it was discovered that finished-pellet lots contained enough out-of-limits pellets to cause concern and thus a recommendation for further fine-grinding. Rather than further process the pellets, it was decided to load vertically; a decision that rendered a minimum of pellet binding with strong indications that vertical loading should be developed for routine processing.

9. Criticality Safety Control by Computer*

C. L. Brown (COO)

A computer has been installed in the FFTF Fuel Fabrication Demonstration Facility, Building 308. Primary functions of the computer will be to maintain an historical record of FFTF fuel and each fuel component; to maintain an accountability record for the facility by station or other location; and to act as a secondary critical mass limit check on fuel transfers between stations. Limits and conversion factors to be applied to the computerized criticality safety program were submitted in Nuclear Criticality Safety Memorandum No. 47.

* Reported by W. W. Little (831)

B. CLADDING TECHNOLOGY

1. Fuel Cladding Procurement

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

A vendor devised a revised tube cleaning process in which the tubes are soaked in a commercial cleaner and the tubing interior scrubbed with gun brushes. Thorough scrubbing results in tubes that will not blacken gun patches wet with alcohol, acetone, or water and pushed through the tubes.

Eddy current inspection of the first lot of 0.250-in. diameter 20% cold-work tubes received from the vendor indicates that these tubes are the highest quality tubes received to date on the FFTF project. The only defects found are so small that they can be located visually only after removing the inclusions by etching in nitric-hydrofluoric acid, then observing the pits left behind with a high power microscope.

Bids were invited for production of 25,000 feet of cladding for fuel pin fabrication demonstration from a limited quantity (810 ft) of stainless steel bar owned by BNW. Invitations to bid were submitted to 25 companies with five responding. Of those responding only two can produce the required tubing from the available bar stock and only two can produce the tubing within the required time. Two of the bidders were asked to re-evaluate their bar stock requirements to reduce the amount of material needed to that available.

2. Vendor Development Program

R. C. Aungst, R. J. Lobsinger, R. R. Studer, and
J. C. Tverberg (721)

Carpenter Technology Corporation

The final draft of the Quality Control and Procedures Manual was reviewed and approved with several minor changes. Carpenter was informed of the changes and was authorized to begin fabrication of the tubes on Contract BDR-590.

Sandvik

Evaluation of the preproduction lot of Sandvik tubes continues.

Superior Tube Company

Discussions were held with Superior Tube Company concerning special handling procedures for fabricating LMFBR tubing.

Wolverine Tube Company

Approval was given to Wolverine to use approximately 1,000 lb of the off-chemistry steel for Phase I set-up portion of Contract BDR-586.

3. Parameter Studies

R. C. Aungst and R. R. Studer (721)

NDT Standards

Replicas of the notches produced by four different fabricators are being evaluated to determine industrial capability to produce NDT standards. These replicas were sectioned and photographed at a magnification of 250X. One vendor's work must be classified as very poor. The notches are all very broad and rounded on the corners. The second vendor's product

has narrower, more sharply-defined notches, but still very much rounded in the bottom. A third vendor produced erratic results with some notches being much better than others. The fourth vendor was the only one who produced consistently high quality notches with reasonably sharp, well-defined corners although his control of notch depth was off in a few instances.

4. LMFBR Fuel and Cladding Information Center

F. R. Shober and B. R. Hayward (723)

Preparation of a list of data fields for the fuel fabrication section was initiated using the appropriate fields from the interim fuel pin file. Additional fields will be added to provide space for the detailed characterization and description of reactor fuels as required.

The mechanical property file was updated by adding data from 46 tensile tests and 24 creep tests of austenitic stainless steel. This brings their respective totals to 2,365 and 54 tests of file.

Data retrieval and plotting activities consisted of:

- (a) a tabular printout of the tensile properties for Types 304, 316 and 348 SS; (b) preparation of 47 engineering stress versus engineering strain graphs for Types 304 and 316 SS; and
- (c) preparation of eight creep strain versus time curves from the creep data on Type 316 SS.

C. FUEL PIN TECHNOLOGY

1. Prototypical FTR Fuel Pins

C. A. Strand, J. E. Sammis, and H. T. Blair (731)

Preparations were begun to fabricate 300 fuel pins meeting the design requirements for the fuel supplier qualification program.

The test fabrication objectives are to:

- Obtain additional process development information
- Obtain more detailed process control data
- Evaluate the qualification fuel specifications and fuel design
- Obtain experience in handling full-length fuel pins
- Develop cost data for fuel fabrication.

The completed pins will be used to evaluate the radiation dose rates that will be encountered during the fabrication and assembly of a 217-pin subassembly.

2. Fabrication of Test Pins for Irradiation in EBR-II

R. M. Crawford (733)

PNL-6. Thirty-six fuel pins for subassembly PNL-6 are loaded with fuel, xenon tagged, and welded. Nine additional pins remain to be loaded to complete the forty-five pins required for PNL-6. Subassemblies PNL-6, 7, 8 provide for testing of 75% UO_2 - 25% PuO_2 fuel material in wire wrapped, 0.250 OD, 316 stainless steel cladding (annealed) with internal components arranged in a manner similar to the FFTF reference fuel pin design.

PNL-8. Thirty-seven fuel pins for EBR-II unencapsulated irradiation test subassembly PNL-8 were packaged and shipped to EBR-II.

PNL-17. The thermocouples for use on PNL-17, an instrumented subassembly, were received from the vendor and were given detailed inspection by both dye penetrant and radiography. These tests indicate that the thermocouples are not acceptable for use because of cracks in the transition weld between the thermocouple sheath and the solid wire extension used for the attachment to the bottom end cap.

All of the fuel pins for PNL-17 are complete with the exception of the 16 pins requiring thermocouples as wire wrap. These pins are complete with the exception of the thermocouple wire wrap.

A vacuum brazing technique was developed for attaching the thermocouples to the top end cap.

D. FUEL SUBASSEMBLY TECHNOLOGY

1. Project Status

M. D. Jackson (732)

Project BCP-034 (Fuel Subassembly Fabrication Demonstration Facility) provides for a facility to demonstrate fabrication methods for producing fast reactor fuel pin subassemblies. The overall project is 92% complete. The estimated completion date of this project has slipped from November 15 to December 31, because of a late delivery of a portion of the monorail system. The monorail system was to have been received on October 30. The expected date for receipt of this equipment has slipped to December 15, as a result of problems at the vendor's plant. Facilities Engineering personnel continue efforts to expedite delivery of this equipment to ensure that no further slippage occurs in the schedule.

2. Large Capital Equipment Status

a. Scanning Electron Microscope Hot Stage

D. E. Blahnik (732)

A hot stage system is being developed for heating mixed oxide pellets to 1600 °C for submicromorphology and sintering characteristic studies with scanning electron microscope. It is anticipated that a unit capable of this service will be available by January 15, 1970. The video tape recorder specification is being revised to allow procurement of the latest (most appropriate) equipment available on the market.

b. Real Time X-Ray Imaging Equipment

M. D. Jackson (732)

X-ray equipment is to be procured which will provide real time imaging and film radiography for FFTF fuel pins. The equipment will be used to inspect:

- (1) Proper placement of fuel pin internal components.
- (2) Length of fuel pin column ± 0.020 in.
- (3) Evidence of damage caused by mishandling.
- (4) Quality of end-cap welds.

An order was placed for this equipment for delivery by March 1, 1970.

3. Fabrication of CCTL Mark II Assembly

D. E. Blahnik and R. L. Myers (732)

A fuel assembly is being fabricated for tests in the CCTL to provide design verification as well as identification and evaluation of potential long-term failure modes for a FTR Conceptual Fuel Subassembly.

Pin fabrication was completed through the loading of the upper reflector. A total of 245 pins are temporarily stored until the upper end cap designs become available.

The lower grid components (layer spacing grid plates, layer locking grid, and element supports) were fabricated and are now being inspected.

An electron beam welding technique was developed for fabrication of the layer spacing grid plate assembly. This technique was then used to weld the round rail to the plates. The weld build-up was minimal and the lower end caps fitted without any additional preparation after welding.

4. Small Component Test Loop - Test Bundle Examination

R. L. Myers (732)

The SCTL loose wire test bundle was removed from the heat transfer loop after a pump failure occurred in the sodium components test loop. The bundle was cleaned and delivered to the Subassembly Development Section for inspection.

The pins had a somewhat dull appearance overall, but no significant wear or corrosion was visible.

The fretting marks observed under the wires during the first examination had become more apparent, especially where the wire was between the pin and the duct.

The pin internals could be heard sliding back and forth as the bundle was tipped from end to end. A possible explanation is that the springs have relaxed and no longer hold the pin internals in place.

CHAPTER IX. PHYSICSA. CORE PHYSICS1. 3-Dimensional Code Development

R. W. Hardie (831)

A three-dimensional (X-Y-Z, R- θ -Z, triangular-Z), multi-group diffusion theory depletion code has been developed for fast reactor criticality and burnup analysis. The code can be used to:

- Compute k_{eff} and perform criticality searches on time absorption, reactor composition, reactor dimensions, and buckling by means of either a flux or an adjoint model,
- Compute material burnup using a flexible material shuffling scheme,
- Compute flux distributions for an arbitrary extraneous source.

All programming is in FORTRAN-IV. Since variable dimensioning is employed, no simple restrictions or program complexity can be stated. Running time for a k_{eff} calculation on a UNIVAC 1108 is about 1/2 hour for a 20 x 20 x 20 mesh problem using two energy groups. Each successive burnup time step takes about 1/4 the above time.

2. FTR Stability Analysis

James R. Sheff (831)

The FTR stability analysis has now been done on two different programs and agreement has been obtained. The conclusions from the study are that the input to both programs is correct and that both programs are operating correctly. The latter conclusion is possible because the analytical methods of solution are different in the programs. Both programs will continue to contribute to the FTR stability analysis since both have major restrictions. One has zero flexibility for expansion of the basic analysis but has been partially

installed in a generalized calculating system⁽¹⁾ which has wide data manipulation capabilities. The other has greater flexibility for stability analysis, but has impossible input and output features. The output is not even labeled, and most input is one number per card. The comparison case required over 200 input cards.

The stability analysis is moving forward on three fronts:

- (1) The key parameters in the analysis are being changed to the latest reference values.
- (2) Subroutines are being written to generate the data needed for the stability analysis from the FTR reference data as it is usually available.
- (3) The new program will be inserted into the generalized calculating system to make its input and output flexible.
- (4) The more detailed capabilities of the new program will be brought into the stability analysis. The subroutines in item (2) will calculate from specific heats, reactor temperatures, thermal conductivities, and component dimensions, the coefficients now required as input to the programs. These coefficients are products, ratios, and sums of the basic data.

Integration of the new program will be slowed by numerical problems encountered in the comparison case. The matrix inversion routine failed for part of the calculations. This failure was eliminated by going to double precision arithmetic which is costly in terms of computer time. Secondly, failure of the inversion can be subtle rather than fatal. That the subtle failure will go unobserved is the danger.

The new results are being included in the justification document for the FTR oscillator.⁽²⁾

-
1. Sheff, James R., *User's Manual for NOISY1 - A Program for Calculation of Space Dependent Spectral Densities in Cubical Reactors*, BNWL-1260, 1969.
 2. *Sheff, James R., "Justification and Specification of an FTR Oscillator," Draft Issue, April 1969.*

3. Burnup Survey Calculations

L. D. O'Dell (831)

Analyses were performed in order to determine the number of energy groups, the number of radial and axial burnup zones, and the length of the burnup time steps which are required for accurate burnup calculations. The necessity of recalculating the cross section data after each burnup interval was also studied. The geometry used in all calculations was that of a 1 zone core operating at 400 MWt with a radius of 56.42 cm and a height of 100 cm. The core was surrounded both axially and radially by a 30 cm stainless steel reflector.

The survey for the number of energy groups was performed in R-Z geometry using 26, 18, 12, 8, 4, and 2 group cross section sets generated by 1DX from the DTX data tape. These cross section sets were used in the burnup code 2DB to make one-dimensional radial burnup calculations ($B^2 = 5.37 \times 10^{-4}$) out to 300 days (two equal burnup steps). The results of this survey indicate that only a few group cross section sets are required for burnup calculations.

The survey for the number of radial and axial burnup zones was performed using the 4-group cross section set generated in the first survey and the burnup code 2DB. The first set of calculations were made in R-Z geometry with 2DB operated in the 1-dimensional radial mode and the core divided into 1, 2, 4, and 8 equal width radial burnup zones. The results of these four calculations indicate that approximately four radial burnup zones are required to describe the actual burnup taking place. The second set of calculations were performed in X-Y geometry using 2DB in the 1-dimensional axial mode ($B^2 = 1.05 \times 10^{-3}$) with the core divided into 1, 2, 4, and 8 equal thickness axial burnup zones. These results indicate that approximately four axial burnup zones are also

required to represent the burnup accurately. All of the above burnup calculations were done for two burnup time steps of 150 days each.

The third survey was to find out how large the burnup time steps could be and still give accurate burnup results. The calculations were performed using a 4-group cross section set with 2DB in 1-dimensional R-Z geometry and two equal-width radial burnup zones. The time steps used were: one time step of 300 days, two time steps of 150 days, three time steps of 100 days and six time steps of 50 days. The results of these burnup calculations showed extremely good agreement, indicating that single time steps of 300 days are acceptable.

To check the necessity of recalculating the cross section data after each burnup interval a 26-group 2DB R-Z geometry burnup calculation was performed for a burnup time of 300 days. The atom densities at 300 days were then programmed into 1DX and a new 26-group cross section set calculated. The new cross section set and the 300-day atom densities were then used in 2DB to recalculate k_{eff} , breeding ratio, peak flux, and the peak-to-average power at 300 days. The results of these calculations are shown in Table 9.I and they indicate that it was not necessary to recalculate the cross data.

TABLE 9.I. Recalculation of Cross Section Data
After a 300-Day Burnup Interval

	<u>2DB with Burnup Time of 0 Days</u>	<u>2DB with Burnup Time of 300 Days</u>	<u>2DB with Recalculated Cross Sections at 300 Days</u>
k_{eff}	1.0653	1.0017	1.0014
Peak Flux	6.21×10^{15}	6.77×10^{15}	6.77×10^{15}
<u>Peak</u> <u>Average</u> Power	1.539	1.543	1.543
Breeding Ratio	0.576	0.624	0.625

4. RSYN Code Development*

K. B. Stewart (COO)

The computer program RSYN was modified to require that $D(x,y)$ grad $\Phi(x,y)$ be continuous across material boundaries. The mixing functions, for the cases compared, improved indicating that the RSYN generated fluxes were approaching the input trial functions calculated by 2DB. The major problem in implementing the continuity requirements was an error in logic introduced by the FR5 compiler's over-optimization in the sections used to calculate the partial derivatives around the edge. The compilation was corrected by setting the affected variables equal to themselves, thus creating an artificial definition point. The section of coding will be re-formulated in such a manner as to not require the introduced definition points.

The buckling option of RSYN was also tested and an error was detected and corrected.

5. Criticality Experiments with 25 wt% PuO₂-U(Nat)O₂ Pins in Water*

C. L. Brown and C. A. Rogers (COO)

Exponential and critical approach experiments with 25 wt% PuO₂-U(Nat)O₂ fuel pins in water were continued. These experiments are being carried out to support the nuclear criticality safety aspects of FFTF fuel handling, storage, and shipment.

The pins are clad in 0.230 in. OD x 28.7 in. long Type 316 SS tubing. The fuel column (pellets) in each pin is 0.195 in. diameter x 27.25 in. long and contains ~30.5 g Pu, with ~12% ²⁴⁰Pu.

In the experiments, the pins are arranged in a vertical, triangular pitch lattice with water reflection on all sides.

* Reported by W. W. Little (831)

Measurements at four lattice pitches have been completed; 0.45 in., 0.779 in., 0.90 in., and 1.35 in. at the 0.779 in. and 0.90 in. spacings. Excellent results were obtained at the 0.45 in. and 1.35 in. spacings, but the lattices were too far subcritical to predict accurately the critical number of pins; however, reasonably accurate exponential data were obtained. These data are currently being analyzed. Experiments are now in progress with pins at a 0.60 in. lattice pitch -- the pitch at which the maximum material buckling is expected to occur.

6. FTR - Critical Experiments Program Status

R. A. Bennett (833)

a. FTR-2 Experiments in ZPPR

FTR-2 shielding experiments were essentially completed on November 13, and the remaining experiments were completed on November 19.

b. FTR-3 Critical Experiments Program Status

ANL personnel are planning to transfer the FTR-3 Critical Experiments Program to ZPR-9 under the assumption that the plutonium approval for ZPR-9 will be timely and not produce significant delays in the program. This action could lead to significant delays in the program if the authorization is not received by December 20, 1969. RDT action is required.

BNW personnel have reviewed, revised, expanded, and returned ANL plans for the FTR-3 Critical Experiments Program. This new plan is being reviewed by both Westinghouse and ANL personnel. The initial steps of the plan have general concurrence of all parties and the lack of a final detailed plan should not impede progress.

Sodium void experiments, using ANL PuO_2 - UO_2 pins, are included in these FTR-3 plans. The use of ANL pins will provide good tests of calculational models of sodium voiding in pin systems even though their diameters are $\sim 3/8$ in. rather than

the $\sim 1/4$ in. for FTR pins. It is also expected that these pin experiments will yield an assessment of heterogeneity effects between pin and plate systems of nearly identical composition. Final details of the experiments are being developed by ANL personnel.

7. Analysis of Doppler Experiments

W. R. Young (833)

Reactivity changes accompanying the Doppler effects experiment at the central core location in ZPR-II Assembly 51 were calculated to within the experimental uncertainty. The reactivity change from Doppler effects when a 134.2 g UO_2 sample was heated from 300 to 1100 °K was measured to be -0.386 ± 0.048 Ih* and calculated as -0.348 Ih.

8. Reactivity Effect of Stainless Steel Swelling

S. L. DeMyer (833)

Differential swelling induced in the stainless steel ducts of FTR by neutron flux and temperature gradient is accompanied by reactivity changes. For 30,000 MWd/tonne exposure, this distortion is calculated to be -0.29% on the midplane radius and +0.32% on the central axis. The reactivity effect caused by pinching has been inferred from two multiplication calculations converged to 2×10^{-4} on k. Since two slightly different mesh structures were used the k-difference of +0.34% Δk is uncertain by $\pm 0.04\% \Delta k$.

Since this calculated effect is equivalent to a significant fraction of the estimated burnup per fuel cycle, e.g. approximately 10%, it should be included in the selection of excess reactivity requirements for FTR.

* $1\% \Delta k/k = 1036$ Ih

9. 3DB Code

R. W. Hardie and W. W. Little (831)

The three-dimensional diffusion-burnup code, 3DB, has been completed; a document is in preparation. 3DB will permit a detailed space-time simulation of the effects of axial discontinuities in hexagonal lattice structures. The code is needed for operational treatment of irradiation tests, control rods, and nonuniform axial burnup.

B. RADIATION AND SHIELDING

1. Decay Heat

D. R. Marr (813)

In the event of a primary coolant pumping failure and subsequent reactor scram, the requirements of the backup cooling system will be dictated primarily by the fission product decay heat. Calculations using the computer code RIBD and the latest nuclear data for ^{239}Pu fast-neutron induced fission yield significantly less decay heat for the first few minutes than in the case of ^{235}U thermal-neutron induced fission. These calculations neglected those short-lived isotopes that have never been identified experimentally, but whose formation is predicted theoretically in the fission process. Inclusion of these isotopes using reasonable decay energies and half lives was shown to increase the decay heat rate of ^{239}Pu fission products at shutdown by only about 12%. All of the nuclear data sets employed were checked for self consistency as to conservation of charge and mass-energy with reasonably favorable results. Statistical comparisons of the theoretical fission product yields indicate that more decay events per fission are expected in ^{235}U than in ^{239}Pu . These efforts are continuing in order to determine the decay heat rate that should be associated with ^{239}Pu fission products immediately following shutdown.

2. ZPPR Shielding Experiments

E. T. Boulette (813)

The ZPPR shielding experiments, which are to provide an evaluation basis for the FTR conceptual design, are completed. Although the necessary reduction and tabulation of data are incomplete, some of the results have been forwarded to BNW. In particular, the shield and reflector spectra were received and were compared to those obtained from 2-dimensional diffusion theory calculations. In Figure 9.1 is depicted the shield comparison. Since absolute experimental values were not obtained, the spectra were normalized in such a manner that the areas under the curves (bounded by the same energy limits) were equal. The overall agreement is quite good, however, a few regions, such as the Na resonance (25 keV) disagree by about a factor of two. A similar comparison was made for the reflector spectrum with about the same agreement.

Additional comparisons will be made as the experimental data are received.

3. Activity of ^{22}Na in FTR

C. A. Mansius (813)

The radiation emitted by the long-lived (2.58 year half life) isotope ^{22}Na will establish minimum radiation levels from components in the primary heat transport system during extended shutdown periods. This isotope is generated by an n,2n reaction with ^{23}Na , which has a threshold (neutron) energy of about 12.5 MeV. It is calculated that about 80% of the ^{22}Na will be generated in the core coolant, with the remaining 20% generated in the reflector and shield coolant. The calculated specific ^{22}Na activity is $0.35 \mu\text{Ci}/\text{cm}^3$ for a $10,000 \text{ ft}^3$ primary coolant system volume and $0.23 \mu\text{Ci}/\text{cm}^3$ for a $15,000 \text{ ft}^3$ system. These activity levels will result in radiation levels less than 100 mrem/hour in the heat transport cells, thereby permitting limited access for maintenance or inspection.

FIGURE 9.1. Comparison of Measured and Calculated Spectrum in the ZPPR Shield

CHAPTER X SAFETYA. PLANT SAFETY ANALYSISEvaluation of Plant Protection System Alternatives
for Unavailability

O. B. Monteith (821)

A study was initiated to investigate the unavailability of various configurations of the plant protection system. The term "unavailability" used herein is defined as, the probability that the system will not be ready to perform its intended function at some future point in time. This definition takes into account the periodic testing of the system, and random failures of the PPS components. It does not include the effects of systematic failures of the equipment, i.e., failures from environmental effects, human error, or unrecognized dependence on a common element.

The unavailability was found to range from 4×10^{-9} to 4.3×10^{-4} .

A weekly testing interval was employed in this study. This testing interval implies that all components are tested and returned to "as good as new condition" after each test. As a consequence, a test each week on all safety rods would be required to confirm the designed scram capability.

Data for the safety rods were based on a paper given at the October ANS meeting titled, "A Probabilistic Evaluation of a Nuclear Plant Protection System" by D. Fitzgerald, C. Feavyear, and G. Knighton. This data represents 72 months of operating experience on the SM-1A reactor at Ft. Greeley, Alaska. During this period there were noted 15 instances of safety rod hang-up, and 23 instances of shim rod hang-up.

B. ANALYSES OF HYPOTHETICAL ACCIDENTS

1. China Syndrome Analysis*

G. Jansen (A00)

The history of the fuel core and structural material after a meltdown is being estimated with special emphasis on the downward progress of the molten material and on the heat transfer into the concrete.

The size of the molten pool formed by dissolution of the concrete floor in contact with a 400 MW core was calculated as a function of time.

The fuel and cladding were initially at 3550 °C with a fuel volume of 10.5 ft³. When the volume of fuel and dissolved concrete reached 2550 °C and a volume of 15.2 ft³, the steel no longer floated on the melt. The heat content of the core and cladding created an initial pool with a 3.2 ft radius and saturated with UO₂ at 1720 °C. Heat generation and heat transfer from the pool were assumed to begin at this point. The molten steel froze when temperature was reduced to 1600 °C and the radius had increased to 3.5 ft.

With radiation direct from the molten pool to the reactor cavity and heat transfer downward into the concrete by conduction, a maximum radius of 4.6 ft was reached after 0.5 hr. The melt temperature at this time was 1330 °C.

With the formation of a crust of frozen material caused by radiation heat losses, a maximum pool radius of 23 ft was reached after 100 days. The crust thickness at this time was 14 ft and the pool temperature was 1100 °C. The pool radius and crust thickness were respectively 6.2 ft and 0.4 ft at 1 hr, 10 ft and 0.8 ft at 10 hr, 16.4 ft and 2.8 ft at 100 hr, and 22.6 ft and 9.2 ft at 1000 hr. The last bit of liquid in the pool froze at a depth of 22.4 ft after 1.8 years.

* D. D. Stepnewski (821)

A progress report summarizing the assumptions on which these calculations are based and the significance of the results is being prepared.

2. Sodium Spray Oxidation in Low Concentration Oxygen

P. R. Shire (821)

Chemical equilibrium calculations have been performed to determine the maximum theoretical adiabatic pressure and temperature versus sodium mass expelled for the range of oxygen concentrations at 0-1-2% in inert gas (nitrogen). The present application is for the determination of maximum accident conditions in FTR inner containment barrier equipment cells. The initial conditions are inert gas at 100 °F at one atmosphere and liquid sodium at the maximum primary system temperature 1050 °F.

It is assumed that sodium is ejected in a fine particle spray ($\sim 500 \mu$ diameter) and mixes perfectly with the cell gas. Also, the gas temperature rise caused by reaction heat is assumed to be adiabatic with respect to the cell walls. For the case of no oxygen the hot sodium and cell gas come to an equilibrium temperature which rises asymptotically to the sodium temperature, 1050 °F, as the quantity of sodium is increased, and this corresponds to a cell pressure of 25 psig. If sodium quantity exceeds $\sim 1 \text{ lb per ft}^3$ of cell volume, a compression effect induced by the added sodium volume is significant and causes the pressure to rise beyond 25 psig.

When oxygen is present the temperature and pressure rise rapidly with an increase in sodium quantity to a peak at a 4:1 sodium-oxygen molar ratio; this corresponds to 2.25 lb Na per 1000 ft^3 in 1% oxygen and 4.5 lb Na per 1000 ft^3 in 2% oxygen for 100 °F initial gas temperatures. With 1% oxygen, additional sodium continues to heat up the gas increasing the pressure asymptotically to 25 psig. With 2% oxygen, additional sodium cools the gas decreasing the pressure

asymptotically to 25 psig. An oxygen concentration of $\sim 1.5\%$ will generate a peak pressure of 25 psig at a 4:1 sodium-oxygen molar ratio and this will neither increase nor decrease when additional sodium is ejected.

When cell gas is initially at 500 °F a peak pressure of 17.5 psig is attained in 2% oxygen and a peak of 10 psig is attained in 1% oxygen at a 4:1 sodium-oxygen ratio. These pressures decrease asymptotically to 8.4 psig as sodium quantity is increased.

3. Sodium Spray Fire in Air - Parametric Study

P. R. Shire (821)

A parametric study of containment accident pressure versus mass of sodium expelled for various contained volumes has been performed with the computer code NAFIRE* utilizing the method of free-energy minimization. Initial conditions assumed liquid sodium at maximum operating temperature of 1050 °F expelled into air at 80 °F and atmospheric pressure. The following assumptions were made for the chemical equilibrium reaction:

- Liquid sodium ejected in fine particle spray dispersion ($\sim 500 \mu$ diameter) particles.
- Perfect mixing of sodium with entire air volume.
- Adiabatic pressurization of the air mass, i.e., no heat transfer to containment walls for duration of the reaction.

As shown in Figure 10.1 maximum pressures of about 123 PSIG were computed for the various configurations examined.

* Anderson, J. L. "Chemical Equilibrium Calculations" Program B-194 Argonne National Lab., (March 1966). (Converted to Univac 1108 Fortran V as NAFIRE, November 1968).

FIGURE 10.1. Adiabatic Containment Pressure for Sodium-air Reaction Pressure Versus Sodium Mass Expelled for Several Contained Volumes with No Heat Transfer to Containment Boundaries

4. Sodium Coolant Ejection Studies

G. L. Fox (821)

Some preliminary estimates have been made on the quantity of sodium which may be ejected above the cover during the DBA. Two mechanisms for limiting the sodium ejection have been considered:

First, after the slug impact, when the slug is still pressed against the cover by the blast pressure, the sodium falls by gravity along the vessel walls and a vapor bubble propagates upward in the center. This bubble rise rate is estimated to be about 8 ft/sec; with the present concept, the sodium ejection could continue for about three seconds.

The second mechanism for limiting sodium ejection is condensation of the sodium vapor in the core region which reduces the blast pressure. The Naval Ordnance Enrico Fermi model tests indicate that the blast vapor bubble was cooled, for a reactor with internal structure and filled with coolant, within Ω_2 seconds, where $\Omega_2 = 0.0333 R$, ft. The blast pressure decay was assumed to be approximated by

$$P = P_o e^{-t/\Omega_2} \left[\frac{U_i}{U_i + U} \right]$$

P = Blast pressure

P_o = Initial pressure

t = time

Ω_2 = time constant = 0.5 sec

U_i = reference volume = 1300 ft³

U = Volume of sodium expelled

R = Vessel inside radius

The sodium ejection was obtained by solving Newton's equations for a fluid column with friction and entrance losses. The cover opening was assumed to be annulus 1 in. wide around the 20-ft vessel (5.13 ft²) and 3 ft long.

The results of the two methods are shown in Figure 10.2. Note the model with condensation of sodium vapor ejects about 40% of the amount of sodium which is ejected by the other model. It will be noted that the data given in Figure 10.2 apply to an initial blast pressure (residual pressure in the vessel following impact of the sodium slug with the cover) of 200 psi. This parameter is related to DBA magnitude, and the 200 psi value corresponds a 300 MW-sec DBA for a 17 ft diameter vessel with ~ 0.5 ft gas gap. Thus, as an example, for the foregoing conditions, it is estimated that about 37,000 lb of sodium could be expelled from the reactor vessel.

5. Energy Generation from a Molten Fuel-Sodium Interaction

A. Padilla (832)

Equilibrium models have been used previously to determine the energy release from a severe disruptive accident. It is assumed in these models that molten fuel and sodium are mixed instantaneously and attain thermal equilibrium prior to any expansion. The energy available for destructive work is then calculated by expansion of the sodium to a lower pressure.

It may be overly conservative to assume that thermal equilibrium can be attained except for extremely small fuel particles intimately mixed with small amounts of sodium. A preliminary transient model has been developed to determine the fraction of thermal equilibrium attained prior to boiling and also to estimate the amount of acoustic (shock wave) energy generated.

For a fuel particle diameter of 1000 microns and a sodium/fuel mass ratio of 1.0, the pressure in the heated region intersected the saturation pressure wave when only 49% of thermal equilibrium was attained.

For a single driver fuel subassembly, the maximum acoustic energy is about 70% of the maximum inertial energy or about

FIGURE 10.2. Sodium Mass Ejection

40% of the total energy. However, significantly lower acoustic energies are obtained for fuel particle diameters of 1000 microns and larger.

C. OTHER SAFETY PROGRAMS

FFTF Safety Development Requirements

J. W. Hagan (825)

FFTF safety development requirements related to the fuel element failure propagation program have been identified. Those program requirements which lie outside current programs at BNW have been outlined and these needs have been transmitted to RDT. Further coordination is required to establish priorities and schedules after review of these needs with respect to ongoing LMFBR safety development programs. Effects upon design conservatism, if specific data are unavailable, must also be considered in this assessment.

Similar review of FFTF needs related to other safety development programs is now in process.

APPENDIX

AUTHOR CODES FOR FFTF PERIODIC TECHNICAL REPORTS

(Effective 10-20-69)

APPENDIX A

AUTHOR CODES FOR FFTF PERIODIC TECHNICAL REPORTS

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