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FAST FLUX TEST FACILITY
MONTHLY INFORMAL
TECHNICAL PROGRESS REPORT
DECEMBER 1969

AEC RESEARCH &
DEVELOPMENT REPORT

ROUTE TO		DATE	

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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT
DECEMBER 1969

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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT
DECEMBER 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 December 1969.

PRIOR FFTF MONTHLY REPORTS

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
BNWL-1274	FFTF - Monthly Report	- November 1969

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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.

A list of reports issued during the month is given in Appendix B.

CHAPTER I. PROGRESS SUMMARYA. COMPONENTS1. Operation with Defected Fuel

It was concluded that reactor operation with even one failed fuel pin would require either (a) maintenance of rates well below those experienced at Fermi, or (b) substantial continuous purging of the cover gas.

2. Vessel Fluence

A revised fluence limit document was prepared. The document, "Fluence Limit for FIR Vessel, Core Barrel, and Core Support Structure," (MMR-25) reflects application of new fast reactor data.

B. CORE DESIGN1. Radial Restraint

Mechanical restraint of core components may result in excessive loadings on the fuel ducts at the wear pad locations. Several methods to reduce the loadings are being examined, such as eliminating the below-core wear pad, providing clearance between the wear pads when the core is clamped, and reducing the duct wall thickness.

C. FUELS AND MATERIALS1. Fuel Clad Interface

Corrosion has been observed at the fuel clad interface in irradiated BNW mixed oxide fuel pins. The cause of the corrosion has not yet been fully defined; test results to date indicate that rubidium hydroxide attack produces corrosion strikingly similar to that observed in the irradiated fuel pins.

2. Materials Technology

Tensile testing is nearly complete for specimens representing the four weld processes included in the FTR vessel weldment program. Preliminary indications are that increases in yield strength and losses in ductility result from irradiation to $\sim 5 \times 10^{21}$ n/cm² (total) at 800 °F.

Evaluation of the elevated temperature fatigue crack growth behavior of Type 304 SS is continuing. The crack growth rate at 1000 °F is 10 to 12 times greater than at room temperature. Preliminary results indicate that cracking at 1200 °F may proceed at a considerably higher rate than at 1000 °F.

D. PHYSICS

Reflector and control loadings for FTR-3 were completed and fissile plutonium loadings began on December 11, 1969. This work is approximately on schedule and initial criticality is expected by mid-January 1970.

Nominal loadings and the physical configuration have been agreed upon as described in Steps 1 and 2 of the "Detailed Plans for FTK-3 Experiments" (ANL-69234).

E. SAFETY

The SOFIRE code was modified to include the effect of a 1/4 in. thick steel liner over the concrete cell floor and walls. New calculations show the floor liner is effective in reducing sodium temperature immediately, due to rapid heat absorption in the steel liner. As a result of this change, peak pressures are reduced about 2 psig.

CHAPTER II. PLANT DESIGNA. OVERALL PLANT DESIGN

This section is presented in the form of Table 2.1 as follows.

TABLE 2.1. Baseline Design Data

Summary of Baseline Design Data as of 12/16/69 H. D. Lenkersdorfer

<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>		
Primary Loops		
- Number		

TABLE 2.1. (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
- LMTU		
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. Plant Protection System (PPS), No. 99

The Conceptual System Design Description (CSDD) has been completed and approved by the BNW Configuration Control Board.

CHAPTER III. COMPONENTSA. REACTOR COMPONENTS1. Guard Vessel

R. C. Walker (7N91)

In response to a BNW request for a Guard Vessel material recommendation, WARD has recommended that 1 Cr, 1/2 Mo alloy still be used for this application.

2. Reactivity - Control Drivesa. Roller Nut CRDM Testing

J. C. Noakes and P. K. Telford (914)

Testing of the N. S. Savannah replacement roller nut drive mechanism will provide essential design data for the prototype control rod drive. The roller nut detailed inspection and assembly at LMEC was completed. The drive and control system checkout was successfully completed before installing the drive on the sodium vessel. Start of testing over sodium was delayed because of the late arrival of critical facility modification components.

3. Fuel Ductsa. Duct Welding Development

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

The detail design for the automatic welding system is approximately 90% complete. Machine shop work on the collet mechanism was started and is approximately 50% complete. Approximately 15% of the overall hardware fabrication of the welding head fixture and the welding fixture machine base has been done.

The checkout at the vendor's plant for Package I (welding power supply) and Package II (welding head and duct fixture components) is in progress. The electrode grinding machine

was received this month, and feasibility studies for the welding computer control were approved. A request for quotation for Package III (computer, operator pendant control, programming software and interfacing of Packages I, II, and III) was submitted to the vendor.

b. Wear Pad Attachment

W. F. Brown (A00) and R. N. Johnson (962)

Thermal cycling of test assembly tube No. 61 was continued this month. The fabrication and the results after 25 thermal cycles were reported last month. The assembly was thermal-cycled 47 more times during this month for a total of 72 cycles (Table 3.1). After the fourth series of thermal cycles, the Inconel 718 wear pad was reduced from 0.060 to 0.050 in. thick by grinding.

To increase the heating rate, the salt bath was held at 1470 °F during the sixth test run. When the part temperature approached 1200 °F, the part was removed manually and air-quenched. Since the heat-up rate was so rapid, it was difficult to achieve a uniform maximum temperature. The average recorded maximum temperature reached for the wear pad thermocouple was 1268 °F, for the tube thermocouple, it was 1364 °F. The range of the maximum temperatures recorded was 1130 to 1409 °F for the wear pad thermocouple and 1256 to 1418 °F for the tube thermocouple.

After each series of thermal cycling tests, the welds and wear pads were visually examined at 10X magnification and the dye penetrant was inspected. No indications were found, and ultrasonic inspection of the spot welds revealed no failure. Thermal cycling Tests 4 and 5 did not appear to cause significant changes in dimensions. Measurements of the tube after the sixth test have not been completed.

TABLE 3.1. Wear Pad Thermal Cycling Tests on Tube 61

Test	No. of Cycles	Sominal Temperature Range, °F	Furnace Environment	Heating Furnace Max. Temp., °F	Cooling Medium	Heating Rate, deg/min		Cooling Rate, deg/min	
						Pad T/C	Tube T/C	Pad T/C	Tube T/C
1	5	600-900	air	900	still air	38	--	55	--
2	10	500-1,000	air	1000	still air	26	--	55	--
3	10	400-1,100	air	1100	still air	47	--	47	--
4	10	400-1,200	salt bath	1200	air blast	267	--	137	--
5	10	400-1,200	salt bath	1200	air blast	300	406	140	156
6	27	400-1,300	salt bath	1470	air blast	1730	2000	165	197

Work was started on the fabrication and assembly of Haynes Stellite alloy 6B wear pads. The as-received sheet, 0.094 in. thick, was sheared into strips 4 1/8 in. wide. The strips were hot-rolled to 0.065 in. in four passes; shear-trimmed to size and annealed at 2225 °F for 15 min and then air-cooled. After cleaning by dry-honing, the strips were roll-straightened and the edges beveled by grinding. Attempts to press-brake form the material to fit a hex tube (two pieces for each complete wear pad) resulted in severe cracking. Two pieces fractured as if they were pieces of glass. The balance of the strips were heat-treated again at 2250 °F for 20 min, air cooled, roll-straightened, and mechanically cleaned. Press-brake forming on six of these pieces yielded four without cracks and two with small cracks on the tension side of the bends. The uncracked pieces are currently being processed for assembly and welding.

4. Control Rods

D. Marinos and H. C. F. Ripfel (811)

CCDD#33 has been modified to incorporate a majority of the comments submitted by RDT.

A design review of the latest Westinghouse control element design was held November 19-21, 1969. Agreements include the basic design for an irradiated test specimen of B₄C to be fabricated for insertion in EBR-II. Complete agreement on the

feasibility of a 12 ft core assembly was not reached although Westinghouse felt that all problems for the initial and advanced cores could be overcome.

B. HEAT TRANSPORT, COOLANT AND GAS COMPONENTS

1. Operation with Failed Fuel Pin

R. V. Dulin (905)

A study is underway to determine the design impact of operating the reactor with a limited number of failed fuel pins. A transient analysis was performed to determine the radiation dose rates resulting from leakage of contaminated cover gas into the work area after failure of a single fuel pin. The analysis assumed a 1200 ft³ reactor cover gas space with a cover gas temperature of 1050 °F, a work-area volume of 1,220,000 ft³ (per the task force), and an airflow of 25,000 ft³/min through the work area. Continuing reactor operation after the failure was assumed. Leakage rates and cover-gas purge rates were used as parameters for the study. Noble gas isotopes ^{83m}Kr, ^{85m}Kr, ⁸⁵Kr, ⁸⁷Kr, ⁸⁸Kr, ^{131m}Xe, ^{133m}Xe, ¹³³Xe, ¹³⁵Xe, ^{135m}Xe, ¹³⁷Xe, and ¹³⁸Xe were considered, by using Westinghouse steady release data and an assumed burst release of 50% of the inventory in the fuel pin. Figure 3.1 portrays some of the results of this transient analysis. It should be noted that this single fuel pin failure can result in significant radiation dose rates, both transient and steady state, unless cover gas leakage rates are held well below the 18 to 48 ft/day experienced* at Fermi.

*Refer to APDA-CFE-2; "Compilation of Current Technical Experience at Enrico Fermi Atomic Power Plant," September 1966.

FIGURE 3.1. Airborne Radioactivity in Work Area Following Failure of 1 FTR Fuel Pin

Figure 3.2 was plotted to show the relationship between cover-gas leakage and number of failed fuel pins necessary to hold the radiation dose rate in the work area at 2.5 mrem/hr during steady state conditions at full power reactor operation. It was concluded that reactor operation with even one failed fuel pin will not be possible unless cover-gas leakage rates are held well below those experienced at Fermi, or unless a very substantial cover-gas purge rate is held. This study is continuing.

2. Dump Heat Exchanger Analysis

R. N. Madsen (822)

The method of attempting to approximate the fin efficiency, Ω , during a transient temperature analysis⁽¹⁾ has been abandoned because of stability problems accompanying the approximation of both derivatives, $(dT/dr)_{r=r_1}$ and dT/dt in the integral term

$$\int_{r_1}^{r_2} r dT/dt \, dr.$$

The use of additional nodes to simulate the fins is not costly from the standpoint of computer time and will be used.

3. Hybrid Simulation of the Reactor and Heat Transport System

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

Simulation Development: Two new programs have been written and checked out: (1) BIAS, a program used to subtract steady-state or dc values in order to expand the scale on a transient variable; and (2) SIGEN, a routine which produces a predetermined step, ramp, or sinusoid function for input anywhere in the simulation.

1. FFTF Monthly Informal Technical Progress Report, October 1969, BNWL-1274, Battelle-Northwest, Richland, Washington. November, 1969.

FIGURE 3.2. Allowable Cover Gas Leakage Versus Number of Fuel Pins for Operations with Defected Fuel

A new vessel model was patched and checked out on the recently-acquired shielded patch board. This more complex model (3 nodes axially) will perform more accurately than the previous one and will be the basis for the next generation of the FFTF hybrid system simulation.

4. Comparison of IHX Studies

R. E. Keyes (812)

A brief review of Westinghouse and Foster Wheeler proposals for an IHX suitable for the Heat Transport System disclosed the following:

- Both proposed a straight tube with U-bend heat exchange tube bundle.
- Both estimated a primary side, shell side, pressure drop of about 6 psi, which is under the 10 psi maximum that was specified.
- Both recommended a partial mock-up of the shell and tubes to check out flow patterns and induced vibrations.
- Both considered the crevice-free tube-to-tubesheet weld to be costly and to require development.
- Both proposed that alternates be considered and investigated for the tube-to-tubesheet weld that would not have to be crevice-free.
- FW (specifically) proposed tests for determining the capability of the tube weld and U-bend to withstand vibrations and to find the fatigue limit.
- Westinghouse's analysis of the nozzle thermal transients and stress indicated the cycles expected were less than the limit. FW's analysis indicated the possibility of requiring thermal shielding at the nozzles.
- FW saw little need for the 30 MW model. Westinghouse considered it useful but assigned a second level priority to it.

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

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

An estimate of the process variables required for control and for information in the FFTF Central Control Room was prepared for us by the Operations Section. The bases for this estimate were operational requirements at the Central Control Room without regard for the method or hardware used to provide operational data.

To facilitate the presentation of operational data, the operating cycle was divided into 10 operating modes applicable to 6 process systems. The required operation data were then listed for each process system in each operating mode. The respective operating modes and process systems are listed below:

<u>Operating Modes</u>	<u>Process Systems</u>
Pre-startup	Reactor
Reactor Startup	H.T.S.
Reactor Ascent to Power	C.T.L.
Power Operation	Refueling
Reactor Shutdown	Containment
Decay Heat Removal	Peripheral Systems
Preparation for Refueling	
Refueling	
Recovery from Refueling	
Fuel Transfer	

A review draft of this work has been issued internally for comment.

This tabulation of required operational data will provide a basis for updating I&C variables lists; it will also provide a basis for evaluating the relative importance of the variables so that decisions can be made respecting sampling priority, recording, computer display, etc.

2. Reactor and Vessel Instrumentation System (System 92)

a. Thermal Transient Testing of Instruments

L. E. Fort (932)

A letter report was transmitted to KDT recommending that thermal transient tests be performed on typical subassembly outlet thermocouples and flowmeters. Thermal transients at all other in-vessel instruments are not considered severe enough to warrant testing.

The letter report recommended that thermal transients typifying (1) a scram condition, and (2) a 60 to 70% subassembly blockage (flow reduction) be simulated. Transients which satisfy these conditions are:

Scram Transient: AT = -350 °F from 1000 °F with maximum rate of -150 °F/sec and 10 sec duration.

Partial S/A Blockage: AT = +600 °F from 900 °F with maximum rate of +600 °F/sec and 1 sec duration.

Further, the letter report stipulates that these thermocouples and flowmeters are required to withstand repeated scram transients without significant degradation in performance; damage to these instruments resulting from a partial subassembly blockage transient is acceptable, however.

b. Reactor Core Parameter Probe

W. Dalos (932)

The sodium loop testing of the Reactor Core Parameter Probe (RCPP) has been completed at AI. The results, included

in a test report soon to be issued, were in agreement with the expectations for the test. They have been partially reported in the AI flowmeter task monthly report for November. The AI test report will indicate where design improvements could be made in future RCPP models, but these changes will not be made unless a new program is initiated.

3. Plant Instrumentation (System 93)

M. O. Rankin (933)

Currently, LMEC proposes to use a water-calibrated Venturi flowmeter as the standard against which large permanent magnet flowmeters will be calibrated. Tentatively, the FFTF accuracy requirement for primary HIS flowmeters is ± 1 to 2% upwards from 10% of design flow. (This is approximately the maximum flowrate provided by the pony motor driven pumps.) It remains to be determined whether water calibration of a Venturi flowmeter can be transposed to a sodium system within this accuracy requirement. If this is not feasible, BNW may request that LMEC provide some other means for calibration.

4. Failed Fuel Detection and Location (System 94)

a. Gas Disengagement Studies at ANL

E. Sowa (ANL) and M. J. Reib (BNW, 932)

The first series of Gas Disengagement Studies at ANL using a cascading type gas disengager was completed. These tests were run at a sodium temperature of 500 °F, and sodium flow rate of 2 ft/sec. Gas flow rates through the disengager varied from 0.75 to 4.25 ft³/min. Results show that at the low gas flow rates the disengaged gas is carried past the detector in slugs. At low flow rates several minutes are required to extract the disengaged gas and move it by the detectors. At high gas-flow rates, the disengaged gas reaches a peak at the detector 24 sec after it is released in the sodium. A second test series is to be run with the sodium temperature at

1000 °F, with all other conditions the same as during the 500 °F tests. It is expected that the gas disengagement work will be terminated upon completion of the second test series.

5. Flux Monitoring and Control System (System 95)

L. W. McClellan (932)

a. Detector Location Studies

A Westinghouse report, "Neutron Flux at Ex-Vessel Instrumentation Locations (FPS #283)," was received and has been reviewed. This report presents revised data on the effect of in-vessel stored fuel on ex-reactor neutron sensors. However, differences in RNW and Westinghouse calculations respecting gamma flux levels and materials used in the reactor cavity remain to be resolved.

b. Neutron Detector Thimble Cooling Studies

The present in-vessel neutron detector thimble design provides for cooling the detector with reactor sodium bypass flow at essentially inlet sodium temperatures (~600 °F).

Studies were initiated at BNW and at Westinghouse to determine the feasibility of cooling the neutron detectors to 300 °F or less by use of an external cooling system. The BNW studies were based on cooling by NaK and by N₂ gas while Westinghouse based its on cooling by Ar gas. In all cases, feasibility was established assuming reasonable flow rates and heat loads, but without special efforts to optimize all parameters. The Westinghouse study was reported this month in FTS-115.

Parameters of the NaK and N₂ cooling systems are given below:

<u>NaK</u>	
Inlet Temperature, °F	200
Outlet Temperature, °F	211
Flow Rate, lb/sec	10
Heat Load, Btu/hr	94,500

Nitrogen

Inlet Temperature, °F	150
Outlet Temperature, °F	350
Flow Rate, ft ³ /min	330
Heat Load, Btu/hr	56,000

Cost estimates were made for these systems. Three independent systems (one for each thimble) are required because the cooling system must meet the single failure criteria.

c. NaK Cooled Neutron Detector Concept

G. R. Waymire (811)

A conceptual design for a NaK cooled in-reactor neutron monitor has been completed. Such a design was required to provide a configuration for determining feasibility for cooling the in-reactor detector to 300 °F. The configuration (SK-3-14662) was also utilized for cost comparison between the various cooled and uncooled detectors under consideration.⁽¹⁾

Where possible, the drawing utilizes subcomponents already under development such as a modified roller nut drive, and a modified closed loop tube with reversed flow. The unit provides a 42 in. sensor stroke.

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 (MOO), and D. C. Thompson (931)

a. Coaxial Cable Noise Level Tests

A recently concluded 1200 hr test investigated "breakdown pulse noise" (BPN) characteristics of the connector and seal areas of cable junctions at 350 °F. A 31-in.-long section

(1) FFTF Monthly Informal Technical Progress Report, November 1969. BNWL-1274, p. 6.2. Battelle-Northwest, Richland, Washington.

of ERI cable was tested, as shown in Figure 4.1. Operation in this guarded configuration with 300 V applied to the detector yielded stable system performance throughout the test. Integral bias measurements with 200, 300, 400 and 600 V high voltage indicated no measurable amount of BPN.

FIGURE 4.1. Cable Thermal Test Setup

Concurrent measurements of system performance with the unguarded configuration indicated significant amounts of BPN. High voltages of 400 to 600 V caused BPN levels of up to 8.3 V magnitude.

This relatively short term test of the connector-seal zone of a cable indicates that the guarded, two-cable technique offers strong promise for possible resolution of the BPN problem.

b. Addendum to Low-Level Neutron Flux Instrumentation Bibliography

A literature search completed during this reporting period updates the bibliography published in BNWL-946, "FFTF State-of-the-Art Report for Low Level Neutron Flux Instrumentation." The resulting new bibliography, contains nearly 100 items, as opposed to only 21 in the original. Besides referring to current literature, the list includes earlier basic documents cross-referenced within the cited reports.

2. In-Core Flowmeter*

a. Permanent Magnet Probe Development

F. Verber (ANL)

Tests of the permanent magnet probe in CAMEL have shown no unusual results. Further analysis has shown that a smaller gap between the magnet and housing can be tolerated, allowing the magnet diameter to be increased about 0.03 in. and thus improving the probe sensitivity. The present probe housing is 6 in. long without end pieces--1 in. too long to fit in the CCTL test position. The feasibility of shortening the housing to 5 in. is being investigated.

b. Eddy Current Probe Development

J. Brewer (ANL)

An alumina-plasma sprayed bobbin with wet wound coils of nickel-clad silver wire was tested at 700 to 1200 °F. Results from these tests showed a very stable system, with no shift of signal detected and only one mv unbalance occurring between 700 and 1200 °F. The current probe configuration is one primary coil and two secondary coils 1/2 in. wide and 1/8 in. deep.

3. Surveillance Sensors

R. W. Smith, (MOO) J. H. Cox (964) and K. O. Creek (932)

The objective of this task is to develop sensors for instrumentation for FFTF surveillance and in-service inspection applications.

Although plans for the evaluation of a Sperry ultrasonic wheel transducer** (Model 50D467) for gamma radiation

* Reported by M. J. Reib (932)

** A candidate for inspecting vessel and piping welds.

tolerance and for operation at 500 °F were reported previously,^(1,2) such an evaluation at BNW has been discontinued.

The wheel transducer is 6 in. in diameter and consists of a piezoelectric crystal mounted on an axle and surrounded by a fluid contained in a flexible tire.

The internal fluid normally used in the unit (50% ethylene glycol and 50% water) cannot be used above the 300 °F boiling point. The tire material (K1305 Silicon rubber) was immersed at 500 °F in substitute fluids which remain fluid at 500 °F, such as Voranol, polypropylene glycol P400 and P1200, polyethylene glycol E-200, Dow 710 silicone fluid, Dowtherm A and ambiflo H-149. In each test the tire material was dissolved or severely weakened by the fluid after 20 hr. Other candidate tire materials such as Silicon rubber Silastic 52 and Silastic 82 were tested at 500 °F with the candidate fluids with similar results.

Tests have also shown that the ultrasonic pulses from the wheel cannot be transferred into a stainless steel plate even at room temperature, unless a coupling fluid is used between the wheel and the plate, and that the transducer output signal is extremely noisy because of reflecting ultrasonic energy inside the wheel.

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- (1) *FFTF Monthly Informal Technical Progress Report, BNWL-1173, August, 1969, Battelle-Northwest, Richland, Washington, p. 3.5*
- (2) *FFTF Quarterly Progress Report, June, July, August 1969, BNWL-1174, Battelle-Northwest, Richland, Washington October, 1969, p. 3.6*

C. OTHER INSTRUMENTATION AND CONTROLS WORK

1. Heat Transport System Control Study

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

A two node dynamic simulation model of the heat exchangers was programmed and tested on the Hybrid computer. The simulation is part of a simplified simulation of the Heat Transport System for use in automatic control system stability studies. The simplified model permits relatively high speed iterative computations (1000 times faster than real time) for use in defining general trends and interactions in controller adjustments. A performance measure based on the effectiveness of the system in attenuating transient effects due to disturbances is being developed for use in comparing different control schemes.

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

R. L. Dillon (A00)*

1. Carbon Meter

Tests continued this month to define the coolant chemistry conditions that cause a carbon meter response. An injection of NaOH and graphite into the SPL at 600 °F did not change the carbon or oxygen meter readings. This suggests that NaOH, which dissociates at 1400 °F to strongly increase the carburization of graphite, does not dissociate at 600 °F.

2. In-Line Hydrogen Detector for Argon Cover Gas

A detector that could measure the hydrogen concentration within the gas plenum and give a quantitative hydrogen measurement is being developed and tested at BNW. It employs the principle of hydrogen diffusion through a nickel membrane followed by oxidation of the hydrogen to water. A new detector has been designed, fabricated, and inserted into a sodium-containing capsule. The inside of the nickel membrane of the new detector is purged with argon gas which is, in turn, continuously monitored with a thermal conductivity detector.

Preliminary tests with the new detector have demonstrated the instrument to be fairly sensitive to hydrogen in the argon cover gas. With the nickel membrane heated to 1400 °F, 100 ppm of hydrogen in the sodium cover gas gave a reading of approximately 40 ppm of H₂ in the argon purge gas (purge gas 50 ml/min). As the hydrogen in the sodium cover gas was increased to 1000 ppm, the hydrogen concentration in the purge gas increased to a maximum of 400 ppm within 6 min.

*Reported by W. R. Wykoff (922)

Additional testing of the detector is planned during the coming period. Calibration curves for various hydrogen cover gas concentrations will be determined and optimum response times will also be investigated.

3. Calcium Analysis

The development of a method for calcium in metallic sodium has been completed, and further work on it will center on segregation, precision, external contamination and other nonchemical problems associated with real samples.

4. Chloride Analysis

The method for chloride in metallic sodium has been completed except for trying to improve the detection limit by the use of ultrapure reagents.

B. SODIUM FACILITIES

1. Sodium Facilities Building (335 Building)

J. D. Berg (954)

Installation of the Small Components Evaluation Loop (SCEL) and the Sodium Sampler Loop (SSL) in the 335 Building has been completed. Checkout of these facilities is in progress.

2. Small Heat Transfer Loop

J. D. Berg (954)

This facility was operated for training of new operations personnel and checkout of new variable speed blower motor. Following installation of the new seven-pin test assembly and calibration of instrumentation, the loop was returned to operation for power testing. The tests again consisted of isothermal runs before and after tests where power was applied to the assembly. The power tests included 22 runs at various flows and power settings up to 26 gpm, 200 kW, and a sodium temperature of 800 °F. In addition, for this series of tests, X-rays were taken of the test assembly while full power was

applied. The X-rays will provide proof of pin position at the above operating conditions.

3. Fission Product Screening Test Loop

J. D. Berg (954)

This loop was started up for checkout of a new sodium cold trap (reported last month). The checkout was required because the new trap design differs from any placed in operation previously. It was determined during this effort, that the trap will not reduce system oxide content without plugging off. Therefore, following the accumulation of additional data on the trap, the loop will be shut down while these difficulties are being evaluated.

4. Sodium Purification Loop

J. D. Berg (954)

Early this report period, we experienced a gradual reduction in sodium flow on this system. Since this condition generally indicates cold-trap plugging, attempts were made to redistribute the oxides in the trap by raising trap temperature to maximum attainable (-400 °F) with existing trace heaters and then slowly reducing internal temperature to normal operating conditions (-250 °F). Following the fourth heatup and subsequent temperature reduction, the trap began to operate normally and presently is continuing. The loop remains in operation at 600 °F and 0.5 gpm with cold trap, direct indicating oxygen and carbon probe in service.

5. Corrosion and Wear Surveillance

J. D. Berg (954)

A surveillance program has been initiated for the Sodium Facilities. An outline of surveillance items and schedules appropriate for each facility is being prepared.

Surfaces of piping samples from the Small Components Evaluation Loop are in good conditions, based on metallography

of piping cross-sections. Two piping samples, each about 3 in. long, incorporating a total of five welds, were removed during test section modifications. These samples include surfaces exposed to 600 °F sodium for more than 3400 hr. Surfaces adjacent to the welds show deposition of some materials at the grain boundaries to a depth of about 0.007 in. Several areas show slight surface roughening (<0.0015 in.), and inspection reveals a single area with some undercutting to a depth of about 0.0025 in. Weld metal surfaces show no deterioration. Nonweld areas are fine-grained and in good condition.

6. FFTF Sodium Facilities Building (Bldg. 335)

K. D. Hayden (954)

The building has been completed and the project closed out.

7. LMFBR Core Segment Development Facility (Bldg. 336)

K. D. Hayden (954)

Design work by Vitro is 98% complete with the design package for the fixed price portion of the project 100% complete. The bid package for the fixed price portion is in preparation by the AEC and will be sent out to interested bidders on December 15, 1969. This portion is estimated at \$225,000 to \$275,000.

The design of the sodium systems (not part of the fixed price portion) is about 95% complete and the specifications and drawings for the vessels are 100% complete and will be sent out shortly for bid.

8. Fast Reactor Thermal Engineering Facility

K. D. Hayden (954)

The design of the Fast Reactor Thermal Engineering Facility is about 56% complete. Preparation of valve and pump technical specifications for procurement is 90% complete. These specifications are being prepared in the RDT standards format.

Title II design work was assigned to Vitro/HES, excluding that associated with the portions of the sodium systems which will be designed by BNW.

9. High Temperature Sodium Facility

R. R. Wyer (954)

Revisions to the design criteria have been processed by the Architect-Engineer. The facility consists of a high-bay sodium laboratory, a service wing, and a two story office building. High bay consists of a building 75 ft wide $\times$ 175 ft long $\times$ 85 ft tall. In addition, high bay has a full 20 ft basement, a 52,000 gal sodium storage system, a 100,000 scf argon storage system, a 50,000 scf nitrogen storage system, a 50 ton building crane, and a below grade pit 25 ft in diameter by 60 ft deep. The service wing includes provisions for a machine shop, a pipe shop, an electric equipment shop and an instrument shop, storage for electrical and mechanical equipment, a change room, a lunch room, and a control room for experimental activities associated with high bay. The office building is of multi-wing design and will accommodate approximately 370 people.

Planned development facilities to be installed in the high bay wing include the Composite Reactor Components Test Facility, a prototype of the FFTF Closed Loop, and a facility for testing FFTF fuel exam equipment.

10. Composite Reactor Components Test Facility

R. R. Wyer (954)

The Composite Reactor Components Test Facility consists of a large sodium vessel (16 ft diam by 42 ft long) and the supporting equipment and facilities required to perform full scale static sodium testing on FFTF in-vessel or associated equipment. Equipment to be tested with the facility include the in-vessel handling machine, the ex-vessel handling machine, an instrument tree, in-vessel fuel storage, and other items such as the core restraint mechanisms.

Scope engineering for the facility has been completed by Vitro, final design criteria have been prepared, and an Engineering Activities Proposal has been completed. Title I design is scheduled to start in January 1970.

11. FFTF Hydraulic Core Mockup Facility (321 Bldg.)

H. Leigh (954)

Estimates are being obtained from Atkins & Merrill, Inc., on a 0.285 and a 0.425 scale model to be used for both hydraulic and vibration studies in the HCM facility. These estimates will aid in the final selection of scale for the model.

The design effort by Vitro has been deferred pending firm design concepts on the FTR piping, size, etc.

CHAPTER VI. CORE DESIGNA. CORE MECHANICS CONCEPTUAL DESIGN, DEVELOPMENT AND TESTING1. Radial Restraint

L. R. Besel (913)

Design effort is continuing to resolve the problem of applying excessive loads to the ducts as a result of core clamping or restraint of the core components during operation. Several methods such as eliminating the below core wear pad, providing clearance between the wear pads when the core is clamped, and reducing the duct wall thickness are being examined to reduce the loads on the ducts at the wear pad locations.

2. Control Rod Poison Section Design

T. J. Bennett (913)

The WARD control rod design study (WARD-FR-92) is being reviewed by FFTF engineers. WARD has recommended that a 61 pin bundle be considered as the reference design. WARD expects the pin outer diameter to be in the range of 0.400 to 0.475 in. and has recommended 0.435 in. OD for the EBR-II irradiation tests. This diameter range of 0.400 to 0.475 in. permits the "inner can" or "flexible" type design configuration, both of which were discussed in the WARD control rod design study.

B. CORE PROCESS TECHNOLOGY1. Fuel Assembly Testinga. Sodium Erosion Tests

A. B. Webb (954)

Sodium erosion testing, to explore sodium erosion effects at high velocity in the fuel duct tip/receptacle area, continued in the Large Component Testing Loop at LMEC. As of 8:00 a.m., December 15, 1969, Test T-1B had been run for

1083 hr at steady-state conditions of 750 °F bulk sodium temperature and 140 psi differential pressure across the test section. Figure 6.1 shows the flow history of the test module which has the largest flow and velocity of the five modules. This T-1-5 module has exhibited curious behavior while the remaining four modules have not changed their respective flows since steady-state conditions were established. The serious low of flow in this module in the earlier part of the test was partially recovered by the use of two operations: backflushing and application of vibration to the test assembly under flow conditions. The backflush operation caused some gas entrainment problems for the test loop system. Near the end of November, the flow in this module stabilized at 17.5 gal/min but has subsequently begun to increase over the past two weeks. The flow increased about 0.3 gal/min in two weeks so that the flow as of December 10, 1969, in the T-1-5 module is at 17.8 gal/min (highest flow established at initiation of test was 20.1 gal/min.) From the 10th to the 15th of December, this increasing trend is much less so that the flow is again tending to level off at 17.9 gal/min.

C. IRRADIATION TEST FACILITIES

1. Closed Loop Design

M. A. Vogel (924)

Studies by Atomics International considered the adequacy of a reduced cell size to 14 × 25 × 24 ft deep for accepting a near term 2 MW Simple Unitized Modular Closed Loop or a future 4 MW Special Purpose Closed Loop. It was determined that there would be adequate space for the 2 MW loop, but that some components of the 4 MW loop would have to be deleted or located outside the cell.

A second study is being performed to evaluate the special benefits derived from a "pool" closed loop concept which would

FIGURE 6.1. Test Module T-1-5 (0.025 in. diam)

involve placing the major primary system components in a single tank filled with sodium. Work is still in progress to determine maintainability, spare requirements and construction costs.

2. Closed Loop Mechanical Nozzle Closure Program

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

The objectives of this task are to

- Compile experimental test data verifying the principle of operation of a creep compensating connector assembly
- Provide experimental test data for each of six specified seal concepts being considered for FFTF closed loop nozzle closure application
- Establish a reference clamping device and seal(s) for subsequent reliability testing
- Provide a basis for the development of a remotely operable instrument connector, incorporating electrical service and tubing connections capable of surviving in the FFTF environment.

a. Seals and Seal Modules

Proof pressure and leakage tests have been completed on two vendor-designed seals.

The seals were inserted into the test module and assembled into the prototype Creep Compensated Connector and loaded above the minimum loading specified by the vendors (with the seal module pressurized). The proof pressure tests were carried out according to the test plan. The seals contained 375 psig helium, with no structural deformation apparent after the test.

Room temperature leak rate tests were conducted in accordance with the test plan and operating guide. After pressurizing the seal module with helium to 10 psi differential, the apparent helium leakage saturated the leak detector (calculated leakage greater than 3×10^{-4} atm cm³/sec, leak reading detectable by the leak detector). The seals, therefore, failed the leak rate test.

3. Closed Loop System - Design of In Core Tube and Top Head Piping

J. W. Kolb (914)

The objective of this task is to perform the design of the In Core Closed Loop Tube and Top Head Jumper Piping. It is intended that the design developed from this work will be fabricated under a separate contract for testing, which will provide the technical information necessary to finalize and verify design of FFTF Closed Loops.

The design now includes provision for "Pot" handling and shielding in the spool to protect control rod drives during refueling. These two provisions may result in a reduction of allowable test specimen size.

Steady-state temperature analysis of the closed loop spool and internals by Westinghouse indicates that the current Westinghouse design would require cooling for instrumentation. A design change to eliminate the problem is under study.

D. OTHER CORE DESIGN STUDIES

1. Simulation of Fuel Region for Digital and Hybrid Pipe Rupture Models

C. D. Flowers (822)

Simulation of the core active zone and calculation of fuel temperatures is carried out by using four equally spaced axial mesh points. Three radial mesh points are used within the fuel. Two schemes for dividing the fuel region radially have been investigated: (1) equal volume per unit length radial regions, and (2) equal A-radius radial regions.

The fuel temperatures at each radial node position will be higher for the equal A-radius model, thus resulting in a

higher core-average fuel temperature. The average steady-state fuel temperatures for the core conditions used in this study were found to be:

<u>Fuel Model</u>	<u>Core Avg Fuel Temp, °F</u>
Equal Volume/Unit Length	1883
Equal A-radius	1985

Also, it appears that the equal A-radius method provides a more representative nodal average fuel temperature for the central fuel node.

The two fuel models have also been compared for a combined flow and power transient, with the average core coolant outlet temperature response being the variable of interest. The transient conditions are:

- a Flow transient for a 1.41 ft² break at the reactor vessel inlet at 2 sec
- a 1 g (15%) scram at 4 sec
- Pump coastdown at 5 sec.

The average core outlet temperature for the equal A-radius model peaks about 5 °F higher than the equal volume/unit length model and remains higher following the reactor scram and pump coastdown (at 10 sec the average core outlet temperature is ~27 °F higher for the equal A-radius model). This will, therefore, result in higher hot channel coolant and clad temperatures for the equal A-radius model.

To further verify the core simulation method used in the digital and hybrid pipe rupture simulations, we have compared the equal A-radius fuel node geometry model with a much more complex model of the core using the MELT-II computer code.

MELT-II is a coupled neutronics-heat transfer code designed primarily for investigating fast reactor core behavior under major accident conditions. However, the code can also be used to investigate less severe transients.

Taking all of the differences into account between the simple models and the MELT-II model, we conclude that the simplified pipe rupture reactor model of the core with equal A-radius fuel node geometry, provides satisfactory agreement with the complex multi-node MELT-II core model. It will, therefore, be used in subsequent pipe rupture investigations.

2. Vessel Thermal Transients for Scram with Full Flow;
Comparison of Oxide and Carbide Fuel
A. L. Gunby (822)

To obtain a more realistic estimate for primary hot leg transients at ultimate conditions, the hybrid simulation HYSIM-2A was used for both the current oxide core and a hypothetical carbide core. A core scram with full flow was examined as a function of outlet plenum mixing (Figures 6.2 and 6.3).

Tube outlet temperature for a carbide core decreases significantly faster than for an oxide core at the same conditions. This effect is caused by the relatively high conductivity of the carbide fuel, resulting in a much lower fuel temperature. Because of the lower fuel temperature, less stored heat is available in the carbide fuel following core shutdown.

Figure 6.3 shows the effect at the vessel outlet. For a value of 700 ft³ mixing volume, the ratio of transient rates for carbide at ultimate conditions to oxide at initial conditions is 1.65. Since the factor differs significantly from a value of 1.25, proposed by Westinghouse, we recommend the use of a hypothetical advanced core in simulation of vessel transients for ultimate conditions, as contrasted to the present 1.25 factor,

FIGURE 6.2. Tube Outlet Transient Core Scram, Full Flow

FIGURE 6.3. Vessel Outlet Transient Core Scram, Full Flow

3. Hybrid Simulation of the Reactor and Heat Transport System
R. D. Benham (MOO) and A. L. Gunby (822)

Simulation Development

Two new programs have been written and checked out:

- (1) BIAS, a program used to subtract steady-state or dc values in order to expand the scale on a transient variable; and
- (2) SIGEN, a routine that produces a predetermined step, ramp, or sinusoid function for input anywhere in the simulation.

A new vessel model was patched and checked out on the recently acquired shielded patch board. This more complex model (3 nodes axially) will perform more accurately than the previous one and will be the basis for the next generation of the FFTF hybrid system simulation.

E. EXAMINATION FACILITY

1. Equipment Development
(ORNL)*

This equipment is required for the remote disassembly and reassembly of irradiated core components and test assemblies requiring postirradiation examination and evaluation. Documentation of the "Conceptual Study of Disassembly-Reassembly Equipment for Examination of FFTF Core Components" will be published in early 1970.

Recent changes in criteria for the subject equipment requires that an additional study be performed to augment the conceptual study. This additional work may be generally divided into three parts: (1) determining the effects upon the disassembly-reassembly equipment and facility related requirements when the lengths of FFTF core components and test assemblies are reduced to 12 ft, (2) preparing new cost and

* Submitted by C. L. Boyd (942)

scheduling data for future portions of this project to reflect new criteria, and (3) documenting the results of these additional studies in an ORNL report that will be an addendum to the Conceptual Study Report.

CHAPTER VII. FUELS AND MATERIALSA. FUELS AND MATERIALS EVALUATION1. Fuel-Clad Interface

R. W. Barker (722)

Further work was conducted toward identifying corrosive constituents at the fuel clad interface in irradiated BNW mixed oxide fuel pins.⁽¹⁾ Capsules containing potential corrosive agents were constructed and thermal aging begun.⁽²⁾

This experiment was removed from the furnace after 519 hr for an intermediate examination. Most capsules were found to be in good condition and were returned for further aging. However, Lot "F" Type 304 and Lot "G" Type 316 capsules containing anhydrous rubidium hydroxide had failed. The condition of the capsules can be observed in Figure 7.1.

Optical examination showed an extensive network of intergranular attack as can be observed in Figures 7.2 and 7.3. Whole grains are observed to have been dislodged. The similarity between micrographs of rubidium hydroxide attack and that observed in BNW mixed oxide fuel pins is striking.

Calculations indicate that the intergranular attack progressed under a stress of only about 200 psi, thermal stresses excepted. It is interesting to note the intergranular attack which occurred near the exterior surface, apparently after failure. The importance of this observation is that this type of attack may progress in the absence of significant stresses.

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1. FFTF Quarterly Technical Report June, July, August, 1969, BNWL-1174, Battelle-Northwest, Richland, Washington, (Section VII-A).
 2. Quarterly Technical Report October, November, December, 1969, BNWL-1275, Battelle-Northwest, Richland, Washington, (Section VII-A).

Neg 469-2734-35A 1X

Neg 469-2732-33A 1X

FIGURE 7.1. Ruptured Capsules of Lot 'F' 304 (left) and Lot 'G' 316 SS (right)
Charged with 1 g of Anhydrous Rubidium Hydroxide and Aged 519 hr
at 900 °F

Neg 469-2734B

100X

Neg 469-2734D

250X

FIGURE 7.2. Transverse Section of Lot 'F' Type 304 SS Charged with Rubidium Hydroxide and Aged 519 hr at 900 °F.

Neg 469-2732B

100X

Neg 469-2732F

250X

FIGURE 7.3. Transverse Section of Lot 'G' 316 SS Charged with Rubidium Hydroxide and Aged 519 hr at 900 °F.

2. Mechanical Testing of Fuel Pin Cladding

R. L. Fish (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 postirradiation burst and stress-rupture properties of fuel pin cladding.

Efforts were concentrated on determining the mechanism responsible for the previously reported low stress, low ductility, intergranular failures in particular PNL-1 fuel pin specimens. These fuel pins were clad with Type 304 SS (E-Lot) and irradiated in the EBR-II to a peak burnup of approximately 9000 MWd/tonne (5.0×10^{21} n/cm²; E > 0.1 MeV).

Aging of Type 304 SS (E-Lot) for times comparable with the number of equivalent full power days to which the PNL-1 fuel pins were irradiated (1370 hr), indicates that observable carbide precipitation along the grain boundaries will occur at as low as 850 °F. The extent of precipitation along the grain boundaries increases with aging temperature. The results from comparing relatively limited grain boundary precipitation seen in these specimens with the very heavy precipitation observed in the above-midplane cladding (operating at approximately 900 °F for the same length of time) suggests that irradiation may have enhanced the process.

3. Damage Analysis

J. L. Straalsund (722), H. R. Brager (A00),
J. F. Bates (722), R. L. Fish (722)

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

Density measurements were completed on a pin (#A4) that was used to contain tensile specimens for irradiation in Row VII (Subassembly X-010, Position 7F3) in the EBR-II. During irradiation, the inside of the pin was exposed to helium and the outside to coolant sodium. In the Figure 7.4, the density and estimated exposure are plotted as a function of distance from the midplane, with the error bands representing the spread of five independent measurements for each point.

Two aspects of the results totally unexpected were:

- The maximum swelling was significantly less than would be obtained from the swelling equation⁽¹⁾ for solution treated stainless steel. While the irradiation temperature is not well known, it is certain that the temperature at the midplane was no less than the inlet temperature, 700 °F. When 700 °F and a fluence of 3×10^{22} n/cm² (E > 0.1 MeV) is used, the swelling equation gives a value of 0.5%, compared to the value of about 0.3% that was observed.
- The axial gradient in swelling appears to be much less pronounced than expected. For example, the temperature may be expected to increase somewhat from the bottom of the pin to the midplane, but a minimum expected change in swelling can be estimated by assuming no change in temperature. By doing this and by using the estimated fluences in Figure 7.4 with the fluence exponent of 1.7 in the swelling equation, one can calculate a value of 2.6 for the minimum expected ratio of swelling at the midplane to swelling 8 in. below the midplane. As may be seen from Figure 7.4, however, the difference in the amount of swelling observed at the two positions is very little.

1. FFTF Monthly Informal Technical Progress Report September 1969, BNWL-2217, Battelle-Northwest, Richland, Washington.

FIGURE 7.4. Irradiation Dose and Swelling Profile for Pin A-4, Irradiated in Row VII in EBR-II

It is possible that the above features may be caused by a flux rate effect or, perhaps, a spectrum effect. It is noteworthy that this is the first set of data from EBR-II Row VII. For positions this far from the core center, the flux is much lower and the energy spectrum is considerably different. However, the irradiation conditions and material characterization need to be defined more completely before any meaningful interpretations can be made. Fortunately, the pin contained flux monitors at several locations, and these are presently being analyzed. ANL at Idaho Falls has agreed to provide estimates of the sodium coolant temperatures in the X-010 Subassembly; also, specimens have been taken from the pin for composition and helium analysis. Further analyses of the swelling behavior will await the results of these efforts.

4. EBR-II Irradiations

P. K. McDaniel (722)

The EBR-II irradiation program is designed to provide coordination and direction for the timely accomplishments of Fast Reactor Irradiations on Structural Materials. EBR-II irradiations are in progress as scheduled. There have been no subassembly discharges or changes.

An experimental subassembly consisting of full B-7 Type elements is being constructed for insertion in EBR-II by mid-March 1970, and should be discharged by February 1971. This will be BNWL's fifth full subassembly to be placed in the EBR-II. Pin numbers for this subassembly are BNWL B-44 to B-50. The goal exposure is to be 4×10^{22} n/cm², while the irradiation temperatures will range from 700 to 1200 °F.

The objective of this experiment is to evaluate the extent of the effect metallurgical variables have on the swelling of nonfissile FFTF core materials. Emphasis is to be placed on variables that represent small deviations from commercial alloy compositions and conventional fabrication treatment for Type 304 and 316 SS.

5. Subassemblies PNL-1, PNL-2, and PNL-X Pins

J. W. Weber and D. E. Christensen (750)

Metallographic examinations of PNL-1 and PNL-X pins have been completed at LASL and the negatives of the photomicrographs sent to BNW for printing and subsequent analysis.

Axial gamma scans of five PNL-2 (PNL-2-15, 16, 17, 18 and 19) pins show no unusual activity peaks.

6. Subassemblies PNL-3, PNL-4, and PNL-5

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

These unencapsulated 37-pin subassemblies attained no additional burnup as the EBR-II shutdown for annual maintenance continued.

7. Subassemblies PNL-6, PNL-7 and PNL-8

L. A. Pember (750) and R. M. Crawford (733)

Subassembly PNL-7 (37 unencapsulated pins) attained no additional burnup as the EBR-II shutdown for annual maintenance continued.

Subassemblies PNL-6 and 8 (37 unencapsulated pins) were assembled at the EBR-II site on December 5, 1969. The assembly was witnessed by a Fuels Evaluation Section staff member. During assembly of subassembly PNL-6, it was noted that the wire wrap on two pins, PNL-6-24 and PNL-6-25 had been installed 180° out of phase. This was duly recorded. The only effect is that orientation of the pin numbering flat in the subassembly is 180° out of phase with the other pins.

Flow testing of the assembled PNL-8 subassembly revealed a flow rate 13% less than anticipated. This is tentatively attributed to the fuel pin diameters being at the upper limit of the tolerance range. Re-evaluation of the calculated fuel pin temperatures under these conditions shows that this flow-rate is still satisfactory.

8. Subassemblies PNL-9, PNL-10 and PNL-11

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

Preparation of the Engineering Test Plan for these 61-pin subassemblies (0.230 in. diam pins, 20% CW 31655) was essentially completed. Final internal review and transmittal to the AEC is planned for next month.

Preparation of the first typed draft of the data package (Experiment Description and Safety Analysis) for these tests was completed and is being reviewed.

9. PNL-12 Series Subassemblies - Run to Failure Tests in EBR-II

M. K. Millhollen (750)

A tentative plan for the conduct of run-to-failure testing of mixed oxide pins in EBR-II was formulated. Two approaches

to producing in-reactor defects in fuel pins are being formulated. One approach is that of drilling 0.005 in. diam holes through the cladding wall which will then be filled with Silfos brazing material. The other approach is to thin the cladding wall at localized spots (notch defect) to permit rupture from internal pressure buildup.

10. Subassembly PNL-17 - Instrumented Subassembly

M. R. Millhollen (750)

The Experiment Description and Safety Analysis (data package) for this experiment was typed and is being internally reviewed.

11. EBR-II/TREAT Irradiations

D. O. Sheppard (750)

The hazards analysis for this series of experiments in the TREAT reactor was drafted and is being internally reviewed.

12. GETR/TREAT Irradiations

G. E. Culley (750)

The operating conditions for the fuel pins in the Mark-II GETR/TREAT and the extended test capsules were redefined based upon the present postulated FIR operating conditions.

Shipping, storage and contract arrangements have been made to relocate ten GETR/TREAT capsules from GF/VNC to BNW for remote re-encapsulation in preparation for transient testing.

Two TREAT capsules that have passed the 15,000 psi hydrostatic pressure qualification test are being shipped to BNW. Three remaining capsules must be reworked and the purchase order for this work is being negotiated with GE-VNC.

Metallographic preparation and photography of the fuel portion of the samples cut from PNL-59-7 was completed. Replicas for electron microscopy were obtained and the electron beam microprobe samples are ready for final preparation when the microprobe becomes available.

13. Fuel Performance Analysis and Prediction

R. D. Leggett, E. O. Ballard (750) and
C. R. Cole (MOO)

A paper entitled, "hybrid Computer Simulation of Fuel Pin Performance," was presented at the ANS winter meeting in San Francisco on December 3, 1969.

The axial and radial structure profiles of five pins from the PNL-2 subassembly were estimated through the use of the Hybrid Computer. Two pins should not contain columnar grains and, hence, no central hole while the other three pins should contain columnar grains and central holes that vary from 0.016 to 0.030 in. in diameter.

14. Irradiation of Hypostoichiometric $\text{UO}_2\text{-PuO}_2$

J. W. Weber (750)

BNW 1-10 and 1-12 capsules were discharged from the ETR on November 9 at the end of Cycle 104. Exposures are estimated to be 92,200 and 96,500 MWd/tonne for 1-10 and 1-12, respectively. Gross gamma scans were completed and compared with those made at 55,000 MWd/tonne (est) on both capsules. Significant movements of fission products were indicated during this last exposure interval.

B. MATERIALS TECHNOLOGY1. Uniaxial Creep^{*}

A. J. Lovell (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 LMFB and to establish the usable materials limits to assure their safe and reliable performance.

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

The testing of solution-treated and solution-treated plus aged controls is continuing. The emphasis is being placed on longer term tests.

One test at 1100 °F of a Type 304 SS weldment specimen irradiated in EBR-II has been completed, and others are in progress. The preliminary results indicate that the sub-arc weld specimen irradiated to $\sim 1 \times 10^{22}$ n/cm² (total) at about 1100 °F has a decreased creep rate, decreased ductility, and possibly a small increase in rupture life when compared to as-welded control data. The testing is continuing and will include a set of solution-treated specimens, as well as other weldments from the same irradiation.

The testing of Types 304 and 316 SS specimens irradiated in EBR-II, Row 2 (X018) is continuing; new results are tabulated in Table 7.1. The variation in response to irradiation by the two alloys is still evident. The creep rates of Type 304 SS specimens tested are generally decreased. The Type 316 SS material tested at 1000 and 1100 °F show increased creep rates. The increased creep rates and low ductilities in the Type 316 SS lead to very low rupture lives. A very striking illustration of the effect of irradiation on creep behavior is evident in tests on identical specimens [1-18-62 and 1-18-63 irradiated to 2.6×10^{22} n/cm² (total) at 1020 °F] at 30,000 psi at both 1000 and 1200 °F. In the unirradiated condition, creep rate at 1200 °F is over three orders magnitude higher than at 1000 °F; however, in the irradiated conditions, the creep rates are essentially the same at the two temperatures.

2. Weldment Studies*

A. L. Ward (A00)

The objective of this program is to provide mechanical property and metallurgical stability data for candidate LMFBR

*Reported by J. C. Tobin (960)

TABLE 7.1. Uniaxial Creep-Rupture Properties of AISI Types 304 and 316 SS
(Irradiated in the Solution-Treated Condition) (a)

Specimen No	Total Fluence (x 10 ⁻²² n/cm ²)	Irrad Temp, °F	Test Temp, °F	Stress, ksi	Minimum Creep Rate (in./in./hr) × 10 ⁵		Rupture Time, hr		% Elongation		
					Irrad.	Control	Irrad.	Control	Irrad.	Control	
Type 304											
1-9-43	1.6 ^(b)	930	800/ 1000 ^(d)	20.0/ 35.0 ^(d)	9.8	1.9	27.3	1506.1	~1.5 ^(c)	12	
1-9-4Z	1.6 ^(b)	930	1000	30.0	0.2	0.6	250.5	~9000.0	~0.3 ^(c)	--	
1-15-1	3.7 ^(b)	1140	1100	30.0	10.0	47.5	72.9	201.0	~0.5 ^(c)	20	
1-15-2	3.7 ^(b)	1140	800/ 1100 ^(d)	25.0	0.9	11.7	210.7	27.0	~0.5 ^(c)	15	
1-15-43	3.7 ^(b)	1140	1100	20.0	0.1	1.5	On Test	~8000.0	On Test	--	
Type 316											
1-18-62	2.6 ^(b)	1020	1000	30.0	4.4	0.02	339.6	~20,000	~1.2 ^(c)	--	
1-20-63	1.6 ^(b)	900	1000	30.0	1.2	0.02	104.2	~20,000	~1.3 ^(c)	--	
1-16-62	3.5 ^(b)	1110	1100	30.0	12.7	3.6	26.1	~1500.0	~0.7 ^(c)	--	
1-16-63	3.5 ^(b)	1110	1100	26.0	2.2	0.7	48.8	~5000.0	~0.6 ^(c)	--	
1-16-64	3.5 ^(b)	1110	1100	20.0	0.4	0.09	1417.3	~100,000	~1.5 ^(c)	--	
1-18-63	2.6 ^(b)	1020	1200	30.0	~10.5	50.0	On Test	~100	On Test	--	

(a) All tests in inert gas.

(b) Based on calculated fluence.

(c) Estimates from creep curve.

(d) Specimen held at the lower temperature for a period of time in search of a transient effect

vessel and core structural weldment materials in direct support of FFTF in order to define adequate lifetime performance under anticipated FTR service conditions.

Tensile testing of all-weld and base metal (Type 304 SS) specimens irradiated in the EBR-II to $\sim 5 \times 10^{20}$ and $\sim 5 \times 10^{21}$ n/cm² (total) at 700 and 800 °F is in progress. Initial test results suggested that the properties were not affected significantly by irradiation to the lower fluence ($\sim 5 \times 10^{20}$ n/cm² at 700 to 800 °F). Therefore, the testing program has subsequently placed emphasis on specimens representing the more severe irradiation condition ($\sim 5 \times 10^{21}$ n/cm² at 800 °F) where yield strength increases and ductility losses have been observed.

Analysis of the tensile data from this group ($\sim 5 \times 10^{21}$ n/cm² at 800 °F) is continuing as test results become available. The following preliminary observations are made on the basis of presently available data from tests conducted in the range from 600 to 900 °F.

Yield strength increases of 10,000 to 20,000 psi are shown by specimens from the TIG, MIG, submerged-arc, and stick-electrode weld processes. Ductility losses are indicated by 5 to 10% reductions in uniform and total elongations in all weldments, except the stick-electrode where no reductions are seen. The weldment materials tend to show somewhat less irradiation hardening than the base metal. Base metal specimens show 25,000 to 30,000 psi increases in yield strength and 5 to 15% reductions in total elongation. Specimens from the TIG, MIG, and submerged-arc weldments all show postirradiation total elongation values which are near the 10% level.

A study has been initiated to determine whether a larger-diameter specimen would improve the reproducibility of test results and also be more representative of the bulk weld material being studied. Specimens with diameters of 0.250 in.

(compared to 0.125 in. for the miniature specimen used in the weldment program) were fabricated from base metal (Type 304 SS) and weld metal from the four weld processes. Tensile tests are being conducted in the range from 600 to 900 °F. Preliminary indications are that the larger specimen provides no well defined advantage; the magnitude of the scatter band is essentially the same as that shown by the group of miniature specimens.

3. Fluence Limits for FIR Structural Materials

R. A. Moen (963)

Fluence limits for the FIR vessel, core barrel and core support structure were initially established in MMR-8 and later in BNWL-1086.⁽¹⁾ Since that time, a considerable amount of new fast reactor data have become available. With the new data and a better understanding of the various problems associated with fluence limits, a revision to the above two documents was prepared. Metals and Materials Recommendation No. 25 (MMR-25), entitled, "Fluence Limit for FIR Vessel, Core Barrel and Core Support Structure," now supercedes the MMR-8 and BNWL-1086 documentation on the subject of fluence limits.

4. Subcritical Crack Growth*

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

The objective of the Subcritical Crack Growth Studies is to determine the effect of temperature on the fatigue crack growth properties of alloys that are to be used in the FFTF. Knowledge of these properties are necessary for estimating the remaining service life or number of load cycles to grow a crack in a component from a finite size, below an NDT detection limit, to a size that can be detected.

1. R. A. Moen. Fluence Limit for FIR Vessel, Core Barrel, and Core Support Structure, BNWL-1086, Battelle-Northwest, Richland, Washington, Rev. May 1969.

* Reported by J. C. Tobin (960)

The test program centers on determining the fatigue crack growth behavior of the major system structural material Type 304 SS with respect to analysis by the crack propagation power law:

$$da/dn = c(\Delta K)^n$$

where

da/dn = crack growth rate (in. cycle)

ΔK = range of stress intensity factor (psi $\sqrt{\text{in.}}$)

c, n = constants for a given material metallurgical condition and test state (e.g. high temperature and sodium environment)

for temperature levels environments and material conditions that are expected to be characteristic of piping system operation. This information will provide a basis for reasonable estimation of the remaining fatigue or service life in piping components using fracture mechanics analysis techniques.

Fatigue Crack Growth Data

Figures 7.5 through 7.7 illustrate the fatigue crack growth behavior results obtained to date. Figure 7.5 shows the room temperature behavior for specimens RML-2, RML-3, RMT-4 and RMT-5. The open symbols show the behavior of cracks oriented in the "RW" direction (Figure 7.8) and the closed symbols are for the "WR" orientation. As would be expected for an annealed cubic material, there is no significant difference in the crack growth rates of the two orientations. Other orientations will be surveyed at a future date. It will be noted that at crack growth rates slightly in excess of 1×10^{-6} in./cycle there appears to be a change in slope. This behavior has been noted in other materials, and will be investigated further with specimen RML-5 which is currently in test. The least-squares regression curves shown were calculated by considering 104 individual data points.

FIGURE 7.5. Type 304 SS (annealed), Single-edge-notch Specimens. Cycled in air at 75 °F.

FIGURE 7.7. Specimen RML-4, 304 SS, Annealed, Single-edge-notch
Cycled at 4 cpm in Air at 1200 °F, RW Orientation

FIGURE 7.8. Crack Orientation Systems. "R" represents the working (longitudinal) direction, "W" the long transverse direction, and "T" the short transverse (thickness) direction.

The cracking behavior at 1000 °F (specimen RML-6) is shown in Figure 7.6. As has been previously reported,⁽¹⁾ cracking at this temperature occurs at 10 to 12 times the rate at room temperature.

Figure 7.7 shows the results obtained to date for specimen RML-4 which is being tested at 1200 °F. There are too few data points at present to draw any general conclusions, but the data does infer a higher slope to the da/dn vs AK line than was observed at 1000 °F. If this trend continues, it would indicate an acceleration of the cracking behavior at higher values of AK . Extrapolation downward of the least-squares regression line of Figure 7.6 would imply that cracking (in the region of $\Delta K = 10,000 \text{ psi } \sqrt{\text{in.}}$) occurs faster at 1000 °F than at 1200 °F. This does not seem reasonable, and it is entirely possible that a slope change occurs for the elevated temperature behavior in much the same fashion as is exhibited at room temperature. This will be investigated in subsequent tests at 1000 and 1200 °F.

5. Kadiation Effects on Absorbing Materials for Control Rods

A. L. Pitner (A00)*

The second subassembly in the 550 to 1200 °F irradiation of B_4C in a Hanford reactor has been discharged after an estimated ^{10}B burnup of 10%. It will be disassembled at the same time as the first subassembly in the 500 to 700 °F test that was discharged the previous month. All capsules will be punctured and the gas pressures measured to determine the amount of helium gas released from each sample.

1. FFTF Monthly Informal Technical Progress Report November 1969, BNWL-1274, Battelle-Northwest, Richland, Washington, December 8, 1969.

* Reported by J. C. Tobin (960)

CHAPTER VIII. FUELS RECYCLEA. FUEL TECHNOLOGY1. Pellet Process Development

M. J. Barr (731)

a. PuO₂ Sinterability Testing

Table 8.1 shows the relative sinterability of PuO₂ from the three sources currently being considered for fast reactor fuel. Of the three source powders, calcined oxalate PuO₂ was the most difficult to press as a single oxide. Pressing difficulty was evidenced by pellet diameter variation, pellet ejection pressure, and the increased dwell time required.

TABLE 8.1. Sinterability of Various Source PuO₂ Powders

PuO ₂ Source	Sintered Density, % TD (11.46 g/cm ³)	Density Range	Δ Diam Max-bfin, in.
Recalcined Oxalate			
Lot #28-3-7	92.30	91.0 - 94.3	0.005
Burned Metal			
Lot #26-9-5	97.00	96.5 - 97.5	0.002
Calcined Nitrate			
450 °C	92.38	91.7 - 92.8	0.0045
650 °C	91.73	90.7 - 93.0	0.003
800 °C	91.54	90.5 - 92.1	0.0035
1000 °C	91.17	90.4 - 91.8	0.003

b. Activation of 93.15% Enriched UO₂

In an effort to increase the sinterability of 93.15% enriched UO₂, an investigation was made of the effect of varying the activity of the starting powder through material treatments. Test pellets of enriched UO₂ were sintered in dry Ar - 8% H₂ for 4 hr at 1675 °C. It can be seen from Table 8.2 that the crystallographic change occurring during the transition from

persons responsible for fuel design. Fuel manufacturing process capabilities were also considered in the design of the sampling plans. The Operating Characteristic (OC) curve for each plan was calculated to show the degree of protection afforded by the plans against the acceptance of pellet lots of varied degrees of quality. The percentage of pellet lots expected to be rejected by the plans and the proportion of defective pellets in the accepted product were also estimated. Review, by personnel of QA, Fuel Element Development, and Fuels Evaluation, will be done to ensure the adequacy of protection and workability of the plans. Upon approval, the plans will be documented, together with necessary additional detail, and issued as Inspection Instructions for use by those Fuels Department personnel performing the inspection function.

3. Fuel Pellet Water and Off-Gas Studies

H. G. Powers (740)

Due to difficulties encountered in controlling fuel pellet water and gas content, a cooperative study was organized by the Special Products Fabrication Section, Analytical Chemistry and Fuels Quality Assurance. The objective of the study is to assess the reproducibility of the water and off-gas tests and to determine the effect of various handling and process operations upon fuel pellet water and gas content. Results of the study to date are summarized as follows:

- Low density pellets retain H_2 during the normal sintering process.
- Absorbed H_2 can be removed by heating for four hours at 800 °C.
- Low density pellets are stable at room temperature after degassing.
- Modification of water apparatus has resulted in valid H_2O determination; results are reproducible to within several ppm.

5. Offsite Fuel Development

B. R. Hayward (723)

The ANS paper on Phase I results of this program was completed and submitted. While the results have been encouraging (Table 8.3), the level of competence that is a goal of the LMFB program is not yet achievable by any participating laboratory. Significant needs emphasized by the results of Phase I include:

- Primary and working standards
- Standardized analytical methods
- Improved analytical methods
- Uniform sample size and sample pretreatment.

All four factors were included in the Phase II effort.

B. CLADDING TECHNOLOGY

1. Cladding Procurement and Development

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

High temperature tensile testing of the CarTech production material was transferred from Atlas Testing Laboratories to the CarTech Metallurgical laboratory in Reading, Pa. A comparison of the data between the two laboratories shows a statistical difference between the yield strength obtained by the two testing laboratories (Table 8.4). Work is underway to determine the reason for this disparity.

Histograms prepared from the eddy current data obtained on lots of tubing produced at CarTech by two different processes show the benefit of special handling during the fabrication of tubing. Figure 8.1 is a histogram of tubing produced by a conventional tube drawing process while Figure 8.2 illustrates a histogram produced by CarTech's "prime" process. This later process involves thorough cleaning between every fabrication pass and requires the use of fresh drawing lubricants for each operation. Future lots of tubing will be produced by this "prime" process.

TABLE 8.4. Comparison of High Temperature Testing Data At Two Different Laboratories

	<u>Atlas Testing Laboratory</u>	<u>CarTech Research Laboratory</u>
1000 °F UTS ksi	89.5	91.2
1000 °F YS ksi	74.5	65.9
1000 °F Elong % in 1 in.	11.0	10.5

PROCESS: 20 min BRIGHT ETCH, 2 BAR DRAWS ON GROUND
MANDREL, FINAL PLUG DRAW

NORMALIZED INTEGRATED AREA: 33

RATING: 6

FIGURE 8.1. Histogram of CarTech Tubing Produced by a Conventional Tube Drawing Process

2. Cladding Specification Revision

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

The first draft of Revision II, A-0086, "Product Description for FFTF Driver Fuel Cladding" was completed. The specification was completely revised to reflect comments from the other LMFBF contractors, from AEC-RDT, from the cladding vendors, and from BNW's experience in the procurement of FFTF quality fuel cladding. The major revisions to the specification included the following: a list of definitions; tightening of the chemical composition of the steel to minimize compositional variances and to give 5% ferrite in a weld bead on the steel; tightening the requirements for ultrasonic inspection; modifying mechanical testing requirements including loading rates during the various tests; eliminating belt sanding or surface grinding of the tubes; specifying a standard NDT notch standard for use in ultrasonic or eddy current inspection; restricting the methods used for halogen analysis; including scratch standards to establish rejection criteria; and numerous other changes that will give a uniform product from all vendors.

3. LMFBF Fuel and Cladding Information Center

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

The LMFBF Fuel and Cladding Information Center is operated by Battelle-Northwest for AEC-RDT. It is principally a data storage, retrieval and analysis center, and a data banking system, which stores experimental data and results. It does not rely completely on documented and referenced reports as its source of data and information. The system, when complete, will contain data relating to all LMFBF type fuel pins.

Major efforts are being continued in the following areas:

- Cladding fabrication file: data collection
- Fuel fabrication file: data field selection

be examined metallographically to determine the effect of the machining tolerances on the penetration and throat thickness.

All of the welding parameters were held constant during the test to get only the effect of the dimensional change on the finished weld.

FIGURE 8.3. Reference FFTF Fuel Pin Closure Weld

CHAPTER IX. PHYSICSA. CORE PHYSICS1. Sensitivity of Energy Release Calculations to Core Sodium Content

A. E. Waltar (831)

The energy release calculated for a major fast reactor excursion can be highly sensitive to the equation of state specified, particularly if most of the normal sodium inventory is present during the disassembly phase. Figure 9.1 illustrates the total energy, energy in molten fuel, and available work (assuming adiabatic expansion of the sodium following complete mixing with the molten fuel) computed as a function of a homogeneous volume percent of sodium in the core. The calculations are very sensitive to the exact sodium concentration near the base condition of 39 vol% because a very small change in the volume available for fuel expansion can cause dramatic changes in the pressure of the heated, confined liquid -- the predominant disassembly force for situations of this type. As more sodium is removed, however, fuel vapor pressure becomes significant; beyond a certain point, this mechanism is sufficient to dominate the disassembly process. Hence, it is crucial to correctly specify the core conditions for sodium-in disassembly calculations. It is important to note, however, that for these calculations, sodium was assumed to be homogeneously distributed throughout the core. For certain situations, gross heterogeneities associated with sodium voids may exist, and effects such as implosion could render quite different results. This effect is currently being assessed.

2. ETOX - R. B. Kidman (COO)

ETOX-1DX Assembly 52 multigroup constants were compared with MC² Assembly 52 multigroup constants.

The ²³⁹Pu fission cross section in Group 20 is given by ETOX-1DX as 8.284 b while MC² gives 7.5718 b. This discrepancy occurs because the two codes treat the cross section floor in the unresolved region with different numerical approximations. The current ENDF/B floor data for plutonium show large variations within energy groups 20 and 21. Consequently, the numerical approximation of a E⁻¹ flux currently used in ETOX for the floor calculation is inadequate. Although some compensation is present in the MC² scheme, this scheme is also suspect. This problem should be eliminated when new ENDF/B data with improved treatment of the floor become available as planned.

The original ²³⁹Pu fission cross sections in Group 21 were 9.329 b from ETOX-1DX, and 7.628 b from MC². The above effect also acts in this group and can be represented by increasing the MC² value to approximately 8.14 b. This is also an overlap group (resolved to unresolved), and the correction of an error in ETOX lowers its value to approximately 9.06 b.

The differences in the resolved region are now being considered.

3. FTR-Critical Experiments Program

R. A. Bennett (833)

a. FTR-2 Experiments in ZPR-9

Central reaction rate ratios were measured in ZPPR on November 21, 1969, which completed the experimental program on a one-zone, 1000 liter, simulation of FTR. Nickel reflector, B₄C control, and sodium coolant platelets were immediately prepared for transfer to ANL-Chicago for use in the FTR-3 program on ZPR-9.

content of FTR-3, necessitating the removal of the entire complement of peripheral controls to achieve criticality. Preanalyses of sodium voiding experiments in this modified version of FTR-3 are summarized in Table 9.1. Values were obtained from perturbation theory calculations using 26 group diffusion fluxes from an R-Z model of the core. Central axis and midplane spatial sodium worth curves for FTR-3 are similar in shape and magnitude to those computed for the FTR⁽¹⁾. Hence, it is expected that sodium void worths measured in this version of FTR-3 would be close to those for FTR.

TABLE 9.1. Calculated Worth of
Na Voiding, **FTR-3**

<u>Number of Central Matrix Tubes</u>	<u>Axial Dimension, in.</u>	<u>Worth, % $\Delta k/k$</u>
1	- 6 $\rightarrow$ + 6	0.0033
	-12 $\rightarrow$ +12	0.0048
	-18 $\rightarrow$ +18	0.0039
9 (3 $\times$ 3 array)	- 6 $\rightarrow$ + 6	0.0290
	-12 $\rightarrow$ +12	0.0426
	-18 $\rightarrow$ +18	0.0342
21 [(5 $\times$ 5 array) - 4 from corners]	-12 $\rightarrow$ +12	0.0954
69 [(9 $\times$ 9 array) - 12 from corners]	-12 $\rightarrow$ +12	0.2807
Inner core 145	- 12 $\rightarrow$ +12	0.4133

(1) W. W. Little, Jr., and L. L. Maas. Nuclear Parameters and Parametric Studies for the Fast Test Reactor (FTR), BNWL-1067, Battelle-Northwest, Richland, Washington, June 1969.

FIGURE 9.2. A Comparison of Calculated and Measured Radial Gamma Dose Distribution in the ZPPR/FTR2

CHAPTER X. SAFETYA. PLANT SAFETY ANALYSIS1. Startup Reactivity Transients

C. L. Fies (821)

The response of the FIR to reactivity insertions of 6 and 12¢/sec initiated during low power or startup operation has been investigated. Ramp rates of this magnitude are typical of those anticipated due to uncontrolled sod withdrawal caused by operator error or control system malfunction. No reactor trip was assumed in the analysis. Initial conditions assumed a prompt reactor power of 1 W and a decay heat power of 6 MW. Flow conditions were held constant during the transient at 10, 50, and 100% of full flow. The results of the analysis and typical power trace are shown in Table 10.1 and Figure 10.1, respectively. The following conclusions can be drawn based on these results:

- If reactivity is inserted under low flow conditions (10%), clad failure will occur before molten fuel conditions are reached. Cases 3, 5, and 6 show that hot spot clad temperature reaches 1600 °F while the fuel temperature is only 2200 to 2300 °F.
- At 50% flow conditions during a reactivity insertion molten fuel conditions are reached in the core at about the same time that clad failure conditions are approached (Case 2).
- Under 100% flow conditions molten fuel conditions are reached before clad temperatures become excessive. Cases 1 and 4 show that the fuel temperature reaches 4850 °F while the clad temperature is below 1200 °F.

- An increase in the Doppler coefficient from -0.004 to -0.0054 does not make a significant difference in the outcome of the accident. Comparison of Cases 3 and 5 (10% flow conditions) shows that the peak power is 65 MW less for the higher Doppler, but that clad temperatures of 1600 °F are reached only 2 sec later than for a Doppler of -0.004. Comparison of Cases 1 and 4 (100% flow) shows that the increased Doppler prolongs the time to fuel melting by 5.6 sec.
- Doubling the rate of reactivity insertion decreases the time to failure by a factor of two. Comparison of Cases 3 and 6 shows that increasing the ramp rate from 6 to 12¢/sec caused the time to clad failure (1600 °F) to decrease from 20 to 11 sec.
- Under low flow conditions (10%) the power range protection channel trip point must be lowered as the power levels off below full power conditions (Cases 3 and 5, and Figure 10.1).
- Low and intermediate protective channel level trips have sufficient time to stop the excursion before the failure point is reached. As shown on Figure 10.1, for example, the power rise becomes very steep at approximately 15 sec. A level trip placed between 10^{-3} and 10 MW would be exceeded before 16 sec. Rod insertion must terminate the excursion before 20 sec, indicating that 4 sec are available to initiate a reactor trip.

2. Containment Transients

R. D. Peak (821)

A parametric study was made of the initial temperature and pressure conditions in the reactor cavity and HTS cells' containment space following a "sodium-in" disruptive accident that might generate 390 MW-sec of work energy. The study

covered the containment range of 40,000 to 240,000 ft³. These initial conditions result from the assumed release of sodium from the core and upper plenum into the containment, the attendant mixing of this hot sodium with the 100 °F atmosphere, and the reaction with the 1% oxygen in the atmosphere. The study took cognizance of the space occupied by the expanding core sodium, liquid-vapor mixture and the associated compression of the atmosphere into the remaining space prior to the mixing of the sodium and atmosphere and attendant chemical reaction. For example, with the 40,000 ft³ containment, the core sodium might expand adiabatically to occupy a 10,000 ft³ space, thereby cooling to 1702 °F, while at the same pressure, 7.3 psig, the atmosphere would compress into the remaining 30,000 ft³ space, the compression heating the atmosphere to 168 °F.

The results of this study are summarized in Table 10.2. For large volumes of 200,000 ft³ and above, there was not enough core sodium to completely react with all of the oxygen available and so the stoichiometric amount of upper plenum sodium was added. Also, for the large volumes of 120,000 ft³ and above, the adiabatic results indicate atmosphere temperatures of 1040 °F and less, or less than the temperature of the sodium in the upper plenum of the reactor. With the assumption that abundant amounts of sodium escape from the upper plenum along with the sodium from the core, then the atmosphere temperature would approach the upper plenum temperature of 1050 °F, implying a containment pressure of 24.8 psig.

3. Sodium Pool Oxidation in Inerted Equipment Cells

P. R. Shire (821)

The SOFIRE code was modified to include the effect of a 1/4 in. thick steel liner over the concrete cell floor and walls. Formerly, the concrete floor was divided into a number of heat conducting slabs, all with the same properties, owing to the basic construction of the code. New calculations

APPENDIX A

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