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FAST FLUX TEST FACILITY
MONTHLY INFORMAL
TECHNICAL PROGRESS REPORT
JUNE 1969

AEC RESEARCH &
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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT
JUNE 1969

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July 8, 1969

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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT
June 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 June 1969.

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FAST FLUX TEST FACILITY
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June 1969

INTRODUCTION

This Informal Technical Progress Report is divided into five chapters. Chapter I presents a brief summarization of FFTF overall program progress during the reporting period and includes a "Design Definition Summary" which has been compiled by the editors from the report contents. This section summarizes significant design decisions and recommendations made, alternative designs proposed, requirements established, and technical design problems identified.

Chapter II, FFTF Design Data, presents a tabulation of principal concept characteristics.

Chapter III, IV, and V follow previous practice and describe the work of the various sections of the Engineering, Reactor, and Plant Technology and Fuels Departments respectively. Because of certain limitations of available printing processes used on this report significant amounts of detail may be lost in any photomicrographs presented in this report. AEC or Contractor personnel requiring glossy prints of the photomicrographs may obtain them by writing to the responsible FFTF Department Manager.

Damage fluence at the top of the receptacles and grid plate were calculated to be $\sim 2 \times 10^{14}$ and $\sim 7 \times 10^{12}$ respectively at the core centerline for concept VA. These fluences are below the damage criteria presently set for these components.

Reactor Nuclear Control. BNW recommended that the roller nut concept be used for the control and secondary safety systems; the magnetic jack concept be used for the primary safety system; and the BNW pneumatic drive be used as the alternative concept in the event that either of the other above drives fails to meet the design requirements.

50. Heat Transport System

Heat Transport System conceptual design modifications are being investigated in an effort to reduce space requirements and thereby to reduce associated size and costs of the containment vessel. A promising concept reduces the length of the primary piping by 33%.

As a result of Babcock and Wilcox Company work two interim IHX designs were selected as recommended bases for follow-on study. One concept features a helical coil tube bundle with floating lower head. The other concept utilizes a straight tube bundle with floating lower head. In both concepts the tube bundle is removable after cutting secondary inlet and outlet piping.

80. Sodium and Gas Processing and Receiving System

A new vapor trap design has been conceived and a prototype constructed. The new trap design is based on the premise that the bulk of the sodium which escapes conventional traps does so as an aerosol mist which can be trapped through re-vaporization followed by a second condensation. Initial tests show considerable promise.

B. SUPPORTING TECHNOLOGY

This section presents summaries of supporting technology items of wide general interest.

1. Safety Program

Arrangements have been made to ensure that the LMFBR safety technology at ANL is applied to its fullest extent in FFTF studies. The areas of immediate application include, for the PSAR and its supporting information, (a) Maximum accident studies which establish the DBA, and (b) Evaluation and specification of the fuel/cladding design bases and, correspondingly, damage criteria for the fuel.

2. Materials Technology

In connection with the potential problem of cracking in critical FFTF components, evaluation of the subcritical crack growth properties of Type 304 SS has been initiated. Initial tests are being performed for the purpose of determining the effect of cycle rate and crack orientation on crack growth rates.

It has been found that exposure to sodium vapor produces much more severe degradation in some stainless steels than does exposure to liquid sodium. The causes for this effect are incompletely understood; however, in the case of 201 stainless steel it was found that sodium vapor produces on accelerated surface attack.

3. Critical Assembly Experiments

FTR I and II Preanalysis

The critical mass of FTR-II has been computed using the as-built core compositions and assuming a 10 cm thick B_4C control ring. This value is 505 kg fissile mass.

Pin-Fueled Sodium Void Experiment

Further preplanning calculations have been performed for a pin-fueled sodium void experiment and a proposal for such an experiment has been drafted.

C. DEVELOPMENT OF FUELS AND THE FUELS SUBASSEMBLY

1. Fuel Process

Fabrication of the 300 FFTF type fuel pins for criticality studies is continuing in the FFTF fuel pilot line. A total of 98 fuel pins has been assembled. A total of 27 kg of mixed oxide powder has been processed and 14 kg of mixed oxide fuel pellets were debonded and sintered. The O/M results obtained from these sintering runs indicate that the O/M may be a function of the furnace load mass as well as the sintering temperature.

The capability for sintering fuel pellets to size through control of die size and forming pressure was demonstrated.

A completed draft of the Fuels Quality Assurance Manual was distributed within BNW for comment. This manual (in three volumes) covers the FFTF Fuels Quality Program and FFTF Fuels Quality Assurance Procedures.

Development of a computer method which can be used to select neutron spectra and flux at any radial or axial position in the EBR-II from calculated or measured data has been completed. This will enable rapid evaluation of fuel element behavior for any position in EBR-II.

2. Fuel Subassemblies

Bending of fuel ducts, as the result of differential swelling of stainless steel in the fuel subassemblies, has been identified as the major lifetime-limiting item for fuel.

Duct wear pad attachment studies have shown that simultaneous welding of the wear pads to opposite sides of the duct by resistance spot welding is practical with the proper fixtures. The method offers good possibilities for minimizing duct distortion during wear pad attachment.

To identify potential wear problems with wire wrapped fuel pins a 7-pin fuel assembly model (3 pins with wire stretched 0.020 in. and 3 pins with wire stretched 0.030 in.) is being tested in the Small Components Test Loop at 30 ft/sec and temperature of 1150 °F. An interim examination after 800 hr indicated no qualitative evidence of wear on cladding or wire wrap surfaces.

II. FFTF DESIGN DATA

This Chapter presents, as Table 2-I, a current summary of principal FFTF concept characteristics and data.

TABLE 2-I. *Summary of Principal Concept Characteristics and Data*

(Compiled by C. W. Higby - June 20, 1969)

	<u>Units</u>	<u>Values</u>	<u>Status^(a)</u>
<u>A. General Plant Data</u>			
1. Core arrangement		vertical	1
2. Design life	years	20	1
3. Peak flux	n/cm ² -sec	7.2×10^{15}	2
4. Total power	MW _t	400	1
5. Reactor coolant		sodium	1
6. Core volume	liters	1033	2
7. Core flow	lb/hr	1.5×10^7	2
8. Core pressure drop			
design maximum	psi	120	2
9. Reactor bulk inlet temperature			
initial	°F	600	2
design maximum	°F	900	1
10. Reactor bulk outlet temperature			
initial core	°F	900	2
design maximum	°F	1200	1
11. Core temperature rise average			
initial	°F	300	2
design maximum	°F	400	1
12. Reactor cover gas		argon	1

-
- (a) 1 - Firm requirement or definite concept characteristic.
 2 - Current reference design value.

TABLE 2-I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status^(a)</u>
<u>B. Reactor Vessel</u>			
1. Diameter	feet	17	2
2. Height	feet	50	2
3. Wall thickness	inches	2	2
4. Wall fluence, total (at 1200 °F)	nvt	1.0×10^{21}	2
5. Material		304 SS	1
<u>C. Core Design</u>			
1. Number of core lattice positions		91	1
2. Number of driver fuel subassemblies		76	2
3. Number of closed loops		6	2
4. Number of contact instrumented in-core open test positions		2	2
5. Number of proximity instrumented open test positions		1	2
6. Number of in-core safety rods		3	2
7. Number of in-core control rods		3	2
8. Number of peripheral control rods		15	2
9. Equivalent core diameter	inches	~48	2
10. Active core height	inches	36	2
11. Radial reflector material		nickel	2
12. Reflector thickness			
axial	inches	6	2
radial	rows	3	2
13. Fuel pin heat transfer area	ft ²	2980	2

TABLE 2-I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status</u> ^(a)
14. Core coolant velocity, maximum	ft/sec	30	2
15. Direction of coolant flow		upward	1
<u>D. Driver Fuel</u>			
1. Fuel composition		18 to 26 wt% PuO ₂ 74 to 82 wt% UO ₂	2
2. Cladding material		Type 316 SS	1
3. Linear heat generation rate, average (at 90% total power)	kW/ft	7.0	2
4. Overpower factor		1.20	2
5. Peak linear heat generation at over-power (at 90% total power)	kW/ft	14.6	2
6. Target burnup, average	MWd/tonneM	45,000	2
7. Cladding heat transfer coefficient	Btu/hr-°F-ft ²	37,500	2
8. Maximum fuel temperature (steady-state)	°F	4230	2
9. Fuel assembly duct length	feet	14	2
10. Fuel geometry		Hexagonal pin cluster	1
11. Pin diameter	inches	0.230	2
12. Spacer wire diameter	inches	0.056	2
13. Number of pins per assembly		217	2

TABLE 2-I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status^(a)</u>
14. Subassembly cross-section outside dimension (across flats)	inches	4.615	2
15. Lattice spacing	inches	4.715	2
16. Duct wall thickness	inches	0.140	2
<u>E. Physics Data</u>			
1. Delayed neutron fraction		0.003	2
2. Neutron lifetime	seconds	4.0×10^{-7}	2
3. Doppler	$T(\frac{dk}{dt})$	-0.004	2
4. Power density, peak	MW/liter	0.75	2
5. Power distribution (peak/average)			
radial		1.40	2
axial		1.24	2
total		1.74	2
<u>F. Test Facilities</u>			
1. Closed loops			
- number		6	2
- location		2 adjacent to core center	2
		1 at mid-radius	2
		3 at core periphery	2
- power handling capability	MW	3-5	2
- test flow rate	gal/min	200-350	2
- test section outlet temperature	°F	1400 (bypass flow permitted)	1

TABLE 2-I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status^(a)</u>
- test section length	inches	36	2
- test section diameter	inches	2.5-3.0	1
- pumping head - primary	lb/in. ²	250	
- test section pressure drop, maximum	lb/in. ²	90	2
- material (in-core tube)		316 SS	1
2. In-core open test positions			
- number with contact instrumentation		2	2
- with proximity instrumentation*		1	2
- power			
- coolant flow rate			
- test assembly length		Same as driver fuel subassembly	
- test assembly cross section			
- coolant			
3. Short term irradiation facility			
- type		trail cable	1
- number		1	2
- location		core periphery	2
- minimum irradiation time at constant flux	minutes	1	2
- sample length	inches	24	2
- sample cross section, maximum	inches	2.5	2

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

TABLE 2-I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status</u> ^(a)
<u>G. Heat Transport System</u>			
1. Primary loops			
- number		3	1
- primary loop material		304 SS	1
- primary loop flow (per loop)	gal/min	14,000	2
2. Primary pumps			
- number		3	1
- design pump head	feet	500	2
- available net positive suction head	feet	51	2
- design temperature	°F	1200	1
- speed control		wound rotor motor with liquid rheostat	1
- motor power			
brake	HP	1790	2
rated	HP	2500	2
3. Intermediate heat exchangers			
- number		3	1
- type		vertical shell and tube	1
- LMTD			
initial	°F	75	2
ultimate	°F	100	2
- capacity			
initial	MW	133	2
ultimate	MW	177	2
<u>H. Shielding</u>			
1. Within reactor vessel			
- material		stainless steel	2

TABLE 2-I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status</u> ^(a)
- configuration		hexagonal assemblies	2
- thickness	inches	21	2
2. Cover shield			
- material		low alloy steel	2
- thickness	feet	4	2
- coolant		argon, helium or Na	2

III. GENERAL TECHNOLOGY

P. L. Hofmann

A. REACTOR AND PLANT ENGINEERING

L. M. Finch

1. Reactor Engineering

D. Marinos

a. Nuclear Control Components

D. Marinos

The CCDD for Nuclear Control Components, CCDD-No. 33, has been approved by the BNW Configuration Control Board after final approval of some changes to the Design Safety Criteria on the same system. The safety criteria were revised to reflect the elimination of the stuck rod criterion from the secondary safety-operational control system and to insert a reliability criterion for the systems.

b. Core CSDD Preparation

G. R. Waymire

The reactor core CSDD has been approved by the BNW Configuration Control Board. The CCB approved version of the core CSDD was transmitted to RDT for review and approval on June 13.

c. Radial Restraint

G. R. Waymire

Discussions were conducted with Westinghouse to identify required analysis, design, and development effort which must be performed to complete a preliminary radial restraint design.

A preliminary list of design criteria was generated as well as an identification of analytical work required. An agreed upon development program has not been established pending completion of a detailed development plan from Westinghouse.

d. Movable Fuel Control

H. C. F. Ripfel

A feasibility study is being performed to evaluate the use of movable fuel as a control medium for the FTR. This study is to establish a firm position on this subject as requested recently by RDT. Preliminary results indicate that the reactivity worth of the fuel bundles in place of the existing poison rods are of decidedly less worth than poison as the position moves toward the center of the reactor. Safety rod positions located in the third row radially would only be worth about one third that of the poison rod. The in-core shim rod positions would be worth approximately one half that of a poison rod. Peripheral rod worths of movable fuel approach that of poison rods but the core must be enlarged or the control must replace existing driver element positions to achieve this worth.

The above worths are based on a fuel pin bundle of 169 pins since the outer row of pins must be eliminated from the 217 pin bundle to permit an inner movable structural duct to be placed inside a normal guide duct. Although increasing the number of fuel positions required to control the reactor does not degrade the core as would the poison rods, space for three times as many safety rods or twice as many in-core shim rods (from the standpoint of drive mechanisms) would result in a serious penalty insofar as availability and cost are concerned.

Scram requirements for movable fuel also seem to result in an added penalty because of the high flows required to cool the fuel rods. Insertion forces to propel a rod counter to the flow direction would be much higher than a poison rod where the flow is severely orificed as a result of lower heat generation rates.

Increased rod worths for movable fuel can be achieved by utilizing a poison follower above the fuel consisting of tantalum.

The gas plenum for the fuel pins must then be located on the bottom of the fuel thus increasing the subassembly length by up to 4 ft. This increase is then transmitted to all in-core assemblies, vessel clearance heights, fuel handling machine lengths and containment height requirements.

2. Plant Engineering

D. P. Schively

a. FFTF Heat Transport System 51

L. G. Henke

Various piping rearrangements of the basic (BNW, RDT and WARD agreed) FFTF Integral Primary and Emergency Core Coolant System were evaluated to reduce the overall size of the reactor containment below the 200 ft diam and to improve the heat transport system piping by eliminating all vertical expansion U-bends, and by a further reduction in the moments and forces on the vessel, pump, and IHX nozzles.

The most promising rearrangement reduced the length of the primary piping approximately one third, provided a nearly symmetrical piping layout for all three circuits, and made possible a slight reduction in the containment diameter. This is the Y-type equipment vault concept in which the centerling of all three/pump IHX vaults are spaced at 120° around the vessel cavity.

Further study of this unique concept, as depicted in Figure 3.A.1. (SK-3-14688), will be made to consider the architectural and structural aspects, and the location of ancillary equipment.

b. Structures and Containment

J. R. Samuel

Drawing SK-3-14677⁽¹⁾ was prepared showing one method of venting the machinery dome into the reactor cavity vault. The drawing shows the venting by means of a pipe attached outside the dome, near the operating floor, passing through the reactor

1. *SK drawing not shown in this report.*

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PRIMARY HEAT TRANSPORT SYSTEM
EQUIPMENT VAULT NO. 2

PRICAL CONTAINMENT VESSEL

support ring into the cavity wall then into the cavity, where a replaceable rupture disk is attached.

Other methods were considered but there are too many reactor oriented items installed directly above the reactor for a simple vent line to be installed directly in a vertical direction.

c. Reactor Flow and Pressure Drop Study

R. E. Keyes

Three tables were prepared to summarize FTR pressure drop studies as follows:

- Table 3.A.I., Reactor Pressure Drop, as based on the maximum-flow bundle coolant rate of 2.073×10^6 lb/hr with no orifice.
- Table 3.A.II., Reactor Power and Flow for high-flow and low-flow design arrangements and for 76 drivers plus three open loops and for the same plus six closed loops as drivers.
- Table 3.A.III., Available Pressure Drop for the four flow rates of the above table.

TABLE 3.A.I. Reactor Pressure Drop

Item	ΔP , psi	$\pm \Delta P$, psi	Ref. Drawings ⁽¹⁾ No., Issue, Date	Basis	Remarks
Inlet Plenum	—	—	SK-3-14585, 1st, 2/18/69	2100-1 in. dia. Holes, 2 in. Long	
Receptacle	1.1	0.2	SK-3-1440, 1st 10/22/68	Nose Pc Slotted to Match Receptacle slots; leak path P = 144; 0.02-in. diam C1.	Leakage = 2.1% depend on design
Fuel Bundle					
Friction Inlet	52.0	8.3	SK-3-14581, 2nd, 2/19/69	217-pins/0.23 in. diam pin/ 0.056-in. OD wire @ 12-in. pitch/93-in. Long	Temp effect with- in 1.3 psi band 500 to 900 °F
Inlet & Outlet & Shroud Tube	3.4	1.0			
Instrument Probe	15.0	2.0	SK-3-12896, 2nd, 3/14/69 SK-3-14581, Modified	4.12 in. diam x 0.60 in. diam C1 x 1-in. Long Leak Path; Na at 900 °F	Leakage 7.9 to 9.6% of flow
Shield at Inlet	6.7	1.1		12-in. Extension of Fuel Bundle	
Total	78.2	8.7			
Elev ΔP between nozzles	7.4	—	SK-3-14541, 1st	About 20.7 ft @ 700 °F	
Reactor	85.6	± 8.7			

(1) SK drawings are not included in this report.

TABLE 3.A.II. Reactor Power and Flow

Flow Path	Power MW	Tol $\pm$	ΔT $^{\circ}F$	High Flow		Low Flow	
				Flow $10^6 \times$ lb/hr	Remarks	Flow $10^6 \times$ lb/hr	Remarks
76 Drivers	365.8		91.4 301	13.577 80.28		13.577 84.92	
3 Open Loops	10.8		2.7 301	0.405 2.30	Assume 75% of driver	0.405 2.54	Same
Core	376.5	5.7		13.982 82.67		13.982 87.46	
Hold-down leakage	—			1.000 5.91	For study & test	1.000 6.26	Same
Control rods	3.5	1.5	0.9 200	0.303 1.79	ΔT for in-rods only	0.303 1.90	} Use nat. conv. to reactor pool
Reflectors	15.0	2.0	3.8 250	0.665 3.93		—	
Shields	4.0		1.0 250	0.177 1.04		—	
Grid Pl. Leakage	—			0.200 1.18		0.200 1.25	
Wall Cooling	—		100	0.500 2.95	2.5-in. liner	0.500 3.13	
In-vessel fuel	1.0			0.090 0.53		—	Nat. conv. in pool
Nominal Total	400.0	6.3	100.0 264	16.917 100.0	$Tol = \sqrt{\frac{1}{2}(\Delta T_n)^2}$	15.985 100.0	$\Delta T = 280^{\circ}F$
				=40,300 gal/min @ 900 $^{\circ}F$		=38,100 gal/min @ 900 $^{\circ}F$	
6 closed loops as drivers				1.046 6.17		1.046 6.54	
Maximum total	400.0		249	17.963 106.17		17.031 106.54	$\Delta T = 263^{\circ}F$
				=42,800 gal/min @ 900 $^{\circ}F$		=40,600 gal/min @ 900 $^{\circ}F$	

TABLE 3.A.III. Available Pressure Drop

Definition, Table 2	Flow		Normal HTS Delivery ΔP , psi	Available ΔP - Max Flow Driver psi
	10^6 lb/hr	gal/min		
Nominal-High	16.917	40,300	152	74
Nominal-Low	15.985	38,100	161	83
Maximum-High	17.963	42,800	142	64
Maximum-Low	17.031	40,600	151	73

These results are intended to serve as interim reference for design evaluation and will be included in a topical report. Revisions will be made when necessary as indicated by test results or design changes initiated during preliminary design.

3. Radiation and Shield Analysis

W. L. Bunch

a. Failed Fuel Element Detection by Xenon Tagging

C. A. Mansius

One method under consideration for the FEDAL system is to monitor the cover gas for the presence of xenon isotopes which had been precharged in the fission gas plenum of the fuel pins. ^{124}Xe , ^{126}Xe , and ^{129}Xe are the most promising isotopes for tag use because they originate infrequently or not at all as fission products and their burnout characteristics are favorable. A different ratio of these isotopes would be selected as specific tags for each subassembly. Only three different enrichments of these isotopes and natural xenon are available commercially, limiting the ratios that may be realized by mixing. Because the ratios change with exposure and because we require tag uniqueness independent of operating history, the number of specific ratios is limited. Table 3.A.IV shows the number of specific tags that may be made using ^{124}Xe and ^{129}Xe as a function of the differential burnout of the two isotopes. The best estimate of the

differential burnout at goal exposure is about 3 to 5%. However, there is a large uncertainty in the burnout calculation because of limited knowledge of the isotopic burnout cross-sections. A 10% differential burnout is believed to be an upper limit providing about 32 individual ^{124}Xe , ^{129}Xe ratios. Because of mixing constraints, simultaneous consideration of the ^{126}Xe , ^{124}Xe ratio has limited value; preliminary investigations show only a small increase in the number of individual tags may be possible.

TABLE 3.A.IV. *Estimated Number of Individual ^{124}Xe - ^{129}Xe Tags*

<u>Differential ^{124}Xe - ^{129}Xe Burnout, %</u>	<u>Number of Individual Tags(a)</u>
10	32
8	40
5	56
3	72

(a) *A small increase in the number of individual tags may be made by simultaneous consideration of ^{124}Xe , ^{126}Xe ratio.*

b. Axial Below-Core Shielding Calculations

E. T. Boulette

Radiation damage must be considered carefully in designing the below-core shield for the FTR. Two-dimensional diffusion theory calculations in R-Z geometry were made with the 2DBS code based on Concept V-A. Neutron intensities and spectra were calculated throughout the configuration. Total radial flux distributions at the grid plate and at the top of the receptacles are depicted in Figure 3.A.2. The neutron spectra at these elevations do not vary significantly with radial distance. The results indicate that the current total fluence radiation damage criterion at these locations are met by the Concept V-A below-core shield design.

c. In-Vessel Stored Fuel Heat Generation

E. T. Boulette

The latest conceptual design of the FTR includes in-vessel storage locations in the sodium annulus exterior to the radial shield. Preliminary calculations were made with 2DBS to estimate the fission heat generated in one stored fuel subassembly. In this calculation, the fuel subassembly and its 8-in. OD finned storage pot were homogenized into one zone. Based on these calculations, the fission power generated by one stored subassembly is about 85 kW during full reactor power. This rate is greater than desired; therefore, more detailed investigations will be made to determine methods of reducing the induced fission level in stored fuel.

FIGURE 3.A.2. Flux Distributions at the Top of the Receptacles and at the Top of the Grid Plate Versus Distance from Core Centerline

B. NUCLEAR AND PROCESS TECHNOLOGY

R. E. Heineman

1. Nuclear Analysis Section

W. W. Little, Jr.

a. Criticality Safety - FFTF Fuel Fabrication

D. R. Oden and C. L. Brown

Criticality safety technical limits were approved for the following:

- (1) Fabrication of $\leq 31\%$ $\text{PuO}_2\text{-U(93)O}_2$ fuel pins in Line 2, FFTF Fuel Fabrication Demonstration Facility (Specification 308-2.5 Rev. 1).
- (2) Shipment of irradiated FFTF fuel pins in BNW Cask 29 (NCS Memo 69-22).
- (3) Storage of ≤ 31 wt% $\text{PuO}_2\text{-UO}_2$ in plastic vials in Room 215, Bldg. 308 (NCS Memo 69-30).
- (4) Handling and Storage of Special 25 wt% $\text{PuO}_2\text{-U(N)O}_2$ pins in 305-B and 3731 Bldg. (NCS Memo 69-32).
- (5) Grinding mixed oxide fuel pellets in the FFTF Fuel Fabrication Demonstration Facility (Special Condition 308-3).
- (6) Shipment of mixed oxide fuel capsules to EBR-II (Shipping Clearance 78).
- (7) Shipment of 25 wt% $\text{PuO}_2\text{-U(N)O}_2$ FFTF fuel pins in the BNW Model 55 container (Request for Amendment 4, DOT SP 4915).

b. $\text{PuO}_2\text{-U(NAT)O}_2$ Fuel Pins in Air and Sodium

L. C. Davenport and C. L. Brown

Critical parameters have been completed for 31 wt% $\text{PuO}_2\text{-U(NAT)O}_2$ fuel pins containing no ^{240}Pu . Monte Carlo calculations have been used to determine the critical numbers of pins for hexagonal arrays in air and in sodium plus error limits for the cases: (1) pins close-spaced touching cladding-to-cladding, and (2) pins close-spaced touching wire-wrapped

pins. In all four cases, a 24-in.-thick concrete reflector completely surrounded the fuel bundle providing full reflection. Quoted error limits indicate the statistical precision obtained within the Monte Carlo tracking routines, but do not include any systematic error due to inaccuracies in the cross section set used. The calculated results are presented in the tabulations below:

*Critical Number of 31 wt% FFTF Fuel Pins
Fully Reflected, Hexagonal Geometry*

	Air Coolant (Pins Unmoderated)	Sodium Coolant Pins in $\rho = 0.971 \text{ gm/cm}^3 \text{ Na}$
Close Spaced, Touching Cladding-to-Cladding	1785 $\pm$ 55	1862 $\pm$ 78
Close Spaced, Touching Wire-Wrap to Wire-Wrap	2805 $\pm$ 218	2811 $\pm$ 167

Preliminary results were also obtained for pins in air, spaced 0.150-in. between nearest neighbors (no wire wrap). Rough extrapolation indicates the critical number to be approximately 4500 pins. The originally calculated results are presented below:

*Effective Multiplication Factor for 31 wt% FFTF Pins in Air
0.150-in. Between Cladding, Fully Reflected*

<u>Geometry</u>	<u>k_{eff}</u>
5941 Pins, Hexagonal Geometry	1.023 $\pm$ 0.037
6487 Pins, Hexagonal Geometry	1.032 $\pm$ 0.025

2. Core Analysis Section

P. D. Cohn

a. Poison Pin Temperatures

A. Padilla

Axial and radial temperature profiles for the safety, in-core, and reflector control rods have been calculated. Table 3.B.I. shows the parameters used in the analysis. Table 3.B.II. shows the maximum heat generation rates for the B_4C poison and the cladding which would occur at the outer row of poison pins. A chopped-cosine axial power distribution was assumed.

The results are shown in Figures 3.B.1. and 3.B.2. Figure 3.B.1. shows the axial temperature profiles for the coolant, cladding, and poison for the fully-inserted in-core and reflector control rods. The temperature profiles are almost identical with the reflector control rod except for having slightly higher temperatures at the pellet centerline. The maximum pellet temperatures of about 1180 °F is reached at a position approximately 6 inches above the core midplane.

Figure 3.B.2. shows the radial temperature profiles for the safety, in-core and reflector control rods at the position of maximum poison temperature. From Figures 3.B.1. and 3.B.2., it is seen that the major portion of the temperature gradient occurs in the helium gap between the B_4C pellet and the cladding. In actuality, swelling of the B_4C may close the gap and reduce the temperature drop significantly. However, the cold diametral gap was used in the analysis because of the lack of swelling data for B_4C .

b. Grid Analysis

C. L. Mohr

The grid spacer proposed for the FTR fuel has been analyzed to identify major operational problem areas. The results indicate that uncertainties of material swelling and in-reactor

creep will have a significant effect upon the success of this design.

Material swelling induced by neutron radiation will cause distortions in the cell lattice dimensions. This distortion would occur in addition to the general growth of the fuel pin and grid structure. This would require that additional space be left between the fuel pin and the inscribed circle within a given grid lattice position. Figure 3.B.3. shows the typical reduction or distortion of the inscribed circle within a grid lattice position. This figure shows the results if stress relaxation is disregarded.

The effect of radiation creep cannot be handled with the analysis methods presently available. An approximation of the effect of the upper limit can be made by assuming 100% relaxation. From in-reactor stress relaxation and creep data that are available it is apparent that there is certainly a potential for relaxation approaching a large percentage of the initial stress. Using this approach, the grid assembly can be analyzed as a plane truss structure. This allows each grid joint to rotate and would be equivalent to the complete reduction of bending stress in the individual grid segment members.

Figure 3.B.4. shows the results of this analysis for a typical cell. Cell distortion in localized areas of the grid, such as is in the extreme corners, would display more distortion than that shown in Figure 3.B.4.

The result of this analysis is to point out the potential effect of in-reactor creep and relative material swelling on the distortion of the grid cell. The resultant effect is that in order to prevent contact and seizure of the fuel pin by the grid support dimples, 0.003 in. must be added to the fuel pin clearance.

TABLE 3.B.I. Parameters for B_4C Poison Pin Thermal Analysis

Pin OD, inches	0.453
Cladding thickness, inches	0.040
Pellet-clad diametral gap, inches	0.005
Pellet density, % TD	88
B-10 enrichment, %	35
Coolant inlet temperature, °F	600
Coolant outlet temperature, °F	900
B_4C thermal conductivity	Variable (BNWL-891)
Cladding thermal conductivity	Variable (BNWL-891)
Helium gap thermal conductivity	Variable (LA-2316)

TABLE 3.B.II. Heat Generation Rates for Poison Pin Thermal Analysis

<u>Position</u>	<u>B_4C, W/cm³</u>	<u>Stainless Steel, W/g</u>
Fully-inserted in-core control rod at core midplane	223	12.6
Fully-inserted reflector control rod at core midplane	226	6.2
Safety rod cocked 10 cm above core	130	6.2

This means that early in life the fuel pin would then have a potential maximum clearance of 0.007 to 0.008 inch. Clearances of 0.007 to 0.008 in. between the fuel pin and grid would provide the potential for fuel pin cladding damage caused by vibration. Stress relaxation of residual bending stresses in the fuel pin would increase the potential for vibration and damage of the fuel pin.

FIGURE 3.B.1. Axial Temperature Profiles for Fully-Inserted Poison Pins

FIGURE 3.B.2. Radial Temperature Profiles for Poison Pins

FIGURE 3.B.3. Reduction in Cell Inscribed Diameter Induced by Swelling, Distortion, and 100% Stress Relaxation of Secondary Stresses

FIGURE 3.B.4. Reduction in Grid Cell Inscribed Circle Induced by Relative Swelling in the Grid Lattice

c. Impact of Swelling Model on FTR Core Design

R. J. Jackson

Interim guidance was provided to FTR driver fuel designers in the form of equations describing the relationship among metal swelling, fast neutron fluence and operating temperature. These equations are

for $\begin{cases} \text{AISI Type 304 SS, solution treated} \\ \text{AISI Type 316 SS, solution treated} \end{cases}$

$$\Delta V/V = 5.0 \times 10^{-36} (\phi t)^{1.66} \left[e^{-\frac{6800}{RT}} - (1.87 \times 10^4) e^{-\frac{27,000}{RT}} \right]$$

for AISI Type 316 SS, cold worked

$$\Delta V/V = 9.4 \times 10^{-33} (\phi t)^{1.48} \left[e^{-\frac{5600}{RT}} - (1.78 \times 10^5) e^{-\frac{30,000}{RT}} \right]$$

to a volume change $\leq 1\%$

and at $T \leq 500^\circ \text{C}$

Preliminary analyses were performed to determine the impact of the interim metal swelling equations, and the impact of current quantitative uncertainty of swelling on the FTR core design. The principally affected components are the fuel pin bundle with its associated spacer system, the subassembly flow ducts and the core radial restraint mechanism. The analyses considered duct bending and core packing. It is impractical to separate the effects of creep from analysis of duct bending so some creep effects are considered in that analysis. The results are summarized in Tables 3.B.III. and 3.B.IV. The conclusions of the analyses are these:

- (1). Differential swelling resulting in duct bending is the major fuel lifetime limiting item in the present design. More analysis of the duct bending problem is required. This must consider both the fuel shuffling schemes and a parametric study of the effects of creep, because no conservative assumption regarding creep rate is possible.

- (2). Potential limitations to fuel exposure include:
 - (a) duct bending caused by thermal camber if significant creep occurs in-reactor
 - (b) duct bending caused by differential volume increase if no creep occurs in-reactor
 - (c) core packing (occupying of all available core space by swollen ducts)
 - (d) duct packing (occupation of all available duct space by swollen fuel pins).
- (3) The uncertainty in extrapolating irradiation-induced swelling to goal burnup is very large and can change the predicted burnup limitation within the range of from 30,000 to 80,000 (or greater) MWd/tonneM, if duct packing becomes the limitation.

d. Pressure Drop
C. L. Wheeler

The pressure drop in the grid and wire-wrapped bundles were compared using both the COBRA computer code and standard flow area hydraulic diameter considerations. Koo's smooth-tube correlation for the friction factor was used in both cases.

For the current design of 19 grids for an 86.6-in. bundle the computed ratios of (grid/wire-wrap) pressure drop are 1.92 and 2.7. (See Figure 3.B.5.) For the maximum flow rate of 2.1×10^5 lb/hr the wire-wrapped bundle pressure drop is 50 ± 8 psi. Experimental data are not available for the grid design. However, the same bundle incorporating a grid spacer yields values of 96 psi (COBRA calculation) and 135 psi for the two methods of computation. These results point out the need for experimental data on the grid design and show that the grid bundle pressure drop could exceed the allowable pressure drop.

TABLE 3.B.III. Summary of Component Bending Reflection

Mechanism	Bending Source	Assumptions in Analysis	Restraint Condition	Burnup Limitation, 10 ³ MWd/tonneM
Flow Duct Bending	Thermal Camber	No creep	Fully restrained for operation	None
	Differential Swelling	No creep		~40
	Thermal Camber	100% stress relaxation	Uncertain	None
	Differential Swelling	100% stress relaxation		Uncertain
	Thermal Camber	No creep	Unlocked for refueling	None
	Differential Swelling	No creep		batch loading >10 1/2 in. free bow deflection Uncertain
Radial Re-Reflector Bending	Thermal Camber	100% stress relaxation	Unlocked for refueling	About 1 year life
	Differential Swelling	100% stress relaxation		
	Differential Swelling	100% stress relaxation		

TABLE 3.B.IV. Summary of Component Swelling in Radial Dimension

Condition	Swelling Model	Burnup Limitation Criterion	Burnup Limitation 10 ⁵ Mwd/tonneM
Core Packing	Mean Value +95% conf. on mean -95% conf. on mean	0.100 in. between ducts	62 to 75
Duct-Pin Bundle Interaction. Wire Wrap	Arrhenius type eq. Conversion to Sol. Treat Enhanced Swelling	Contact between pin bundle and duct wall	33 30
Duct-Grid Spacer Interaction	Arrhenius type eq. Conversion to Sol. Treat Enhanced Swelling	Grid swelling to contact duct wall.	~55 ~55

FIGURE 3.B.5. Comparison of Pressure Drop Between Wire-Wrapped and Grid Spaced Bundles

e. Local Temperature Peaking

D. C. Kolesar

The local reduction of the energy transfer rate induced by geometric asymmetries can result in local hot spots. For the wire spacer it is the region near the wire, particularly as the wire passes into the gap between pins, that constitutes a dangerous peaking problem. For the honeycomb grid it is the region adjacent to and downstream from points of grid-pin contact.

The local effect of 30-mil wire in an infinite array of pins with the wire always in the pin-pin gap has been considered. The method of solution involved a graphical approach created by Dwyer⁽¹⁾ and modified to incorporate a wire in a manner similar to a technique employed by Deissler and Taylor.⁽²⁾ A range of 25 to 60 °F was predicted in the cladding with the uncertainty a result of the unknown eddy diffusivity. The validity of the model is dependent upon the wire pitch being at least 12 in. A 56-mil wire will reduce the temperature peaking by perhaps 20%. Local peaking would no longer be accentuated by the presence of the wire in the gap.

The local effect of a grid is not expected to be too dissimilar from the wire-wrap. A 1-dimensional analysis which hypothesizes a step change in heat transfer coefficient provides a model for grid peaking. Figure 3.B.6. provides a parameterized representation of this analysis for various depressions of the heat transfer coefficient and disturbance sizes. The experimental work of Kattchee and Machewicz⁽³⁾ provide a means of identifying

(1) O. E. Dwyer, "Analytical Study of Heat Transfer to Liquid Metals Flowing In-Line Through Closely Packed Rod Bundles," *NUCLEAR SCIENCE AND ENGINEERING*, 24, 347, 1966.

(2) R. C. Deissler and M. F. Taylor, "Analysis of Fully Developed Heat Transfer and Flow in an Annulus with Various Eccentricities," *NACA Tech. Note 3451*, May 1955.

(3) N. Kattchee and W. V. Machewicz, "Effects of Boundary Layer Turbulence Promoters on the Local Film Coefficients of ML-1 Fuel Elements," *NUCLEAR SCIENCE AND ENGINEERING*, 16, 31, 1963.

FIGURE 3.B.6. Temperature Peaking Induced by a Local Disturbance of the Heat Transfer Coefficient

the heat transfer coefficient with the disturbance. A reduction of the heat transfer coefficient by a factor of 2 is reasonable in the wake of a protuberance. It is observed that no serious problem results for all disturbance size. However, if a bubble were to form or to be trapped behind a protuberance the problem could be severe. First, little mechanism is available to sweep the bubble away. Second, the resulting drop in heat transfer coefficient, here assumed to be $100 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$, influences local energy transfer seriously. Thus, the point of contact between grid and pin should be kept small to avoid or minimize the effects of bubbles.

f. Subassembly Enthalpy Peaking

C. L. Wheeler

The amount of enthalpy peaking in a subassembly is dependent upon the relative areas of the interior and peripheral channels and on the amount of energy and mass transport which occurs between channels. Wire wrap lowers the enthalpy peaking factor by increasing the intra-subassembly mass transport. (See Figure 3.B.7.) The Westinghouse spacers reduce enthalpy peaking by reducing the difference in peripheral and interior channel areas at each grid location.

The COBRA computer code was used to determine the amount of enthalpy peaking for both spacing methods. Values of 1.10 and 1.20 were determined for the loose and tight packed models of the wire-wrapped bundle. These values bracket the actual enthalpy peaking factor that will occur in the standard wire-wrapped bundle. The wire wrap enthalpy peaking factor can be reduced further by the use of peripheral flow suppressors. The computed enthalpy peaking factor for a grid spaced bundle with grids located on 4-in. increments in the active core region is 1.12.

It is difficult to place confidence limits on this value because of the limited amount of experimental data. However, this value is reasonable and it can be concluded that both spacer systems are comparable as far as enthalpy peaking is concerned.

If the current model of duct-bundle swelling is correct, the enthalpy peaking factor will remain essentially constant for grid spacers and decrease with burnup for the wire-wrapped bundle, but if the duct dimensions increase at a faster rate than the bundle dimensions the wrapped bundle enthalpy peaking factor will increase significantly.

FIGURE 3.B.7. Comparison of Enthalpy Peaking Factors Between Grid Spaced and Wire-Wrapped Bundles

3. Experimental Physics

R. A. Bennett

a. Status of Phase B Experiments

R. A. Bennett

Following the critical loading of ZPPR Assembly I, repeated measurements were made to improve statistics for the source determination. The core profile was reduced to the matrix profile of Assembly 56b, as built in ZPR III, to assess reactivity differences between ZPPR and ZPR-III due in part to the different void fractions. A full control ring has been added in the reflector of ZPPR Assembly I at a position anticipated to yield a critical configuration when the enclosed reflector drawers are replaced with fuel. This control ring is approximately two columns wide, and the loading is the same as the drawer loading used for measurements in the radial reflector of Assembly 56b in the ZPR-III, reported in the April 1969, ANL FFTF Informal Monthly Report. A subcritical measurement of k_{eff} has been made with the control ring in place. Presently, the reflector drawers enclosed by the ring are being exchanged for fuel to achieve criticality.

b. FTR I and II Pre-Analysis

J. V. Nelson

As reported last month, planning calculations are being performed for FTR I, II, and III, in the ZPPR. Tabulated in Table 3.B.V. are the dimensions and critical masses for FTR I and II. The results listed here were obtained using the two-dimensional diffusion code, 2DB in R-Z geometry. The worth of the control ring in FTR-II has been computed to be 5% $\Delta k/k$.

TABLE 3.B.V. *FTR I and II Dimensions and Masses**
(R-Z Calculation)

	FTR I	FTR II
Core height (cm)	91.4	91.4
Radial reflector thickness (cm)	30	20
Axial reflector thickness (cm)	30.5	30.5
B ₄ C control ring thickness (cm)	-	10
Calculated core radius (cm)	49.5	58.1
Calculated core volume (ℓ)	706	970
Calculated fissile mass (kg)	367	505
k _{eff} **	0.98	0.98

**The compositions of the core are those reported in May 1969, ANL FFTF Informal Monthly Report.*

***k_{eff} values have not been corrected for core heterogeneity or core gap. They are also subject to a transport theory correction. (Based upon ZPR-III Assembly 56b results the resultant fissile mass for this k_{eff} should be essentially the critical mass.)*

c. Pin-fueled Sodium Void Experiment

Q. L. Baird and J. V. Nelson

A proposal has been drafted for a pin-fueled sodium void experiment as discussed in the May Progress Report. It appears appropriate that the experiment be made a part of the FTR II sequence following the shielding experiment.

As a part of the planning for the experiment, a series of calculations have been done in R-Z geometry using 2DB code. The reactivity worth of converting a 26 cm radius central zone of FTR II to Modification A composition (cf May 1969, FFTF Monthly Informal Technical Progress Report, BNWL-1097) has been computed to be +1.01%. The $\Delta k/k$ for substitution of a 3 x 3 drawer array

of pin-fueled calandria at the center of this two-zone core is -0.14%. The reactivity worth of voiding a 42 cm central segment of the central drawer of this 3 x 3 array has also been computed and is nominally 5 inhours. Further calculations are underway to support this planning.

d. Analysis of ZPR-III Assembly 56b

S. L. DeMyer

Analysis of large substitution experiments, both at the core center and on the core periphery, have been completed. The central worths were calculated in both R-Z and x-y geometries in the diffusion theory code, 2DB. The peripheral rod worths were calculated in x-y geometry, only, in 2DB.

Table 3.B.VI. compares the measured and calculated worths of the central drawer exchanges. Table 3.B.VII. compares the calculated peripheral control rod worths with four measured values. The locations of the four counters in the assembly are given in the April 1969, ANL FFTF Informal Monthly Report. The measured worths derived from the data of counter 1 are thought to be the most accurate.

TABLE 3.B.VI. Central Drawer Exchange Worths

Experiment Discription	Experimental Worth, % $\Delta k/k$	Calculational Worth		C/E R-Z	C/E x-y
		R-Z; % $\Delta k/k$	x-y; % $\Delta k/k$		
Replace Fuel with Axial Reflector	-0.430	-0.578	-0.591	1.34	1.37
Replace Fuel with B ₄ C Control Rod	-1.516	-1.758	-1.748	1.16	1.15
Replace Axial Reflector with B ₄ C Control Rod	-1.086	-1.188	-1.163	1.09	1.07

TABLE 3.B.VII. Peripheral Drawer Exchange Worth

Experiment Description	Experimental Worth		Calculational Worth, % $\Delta k/k$	C/E
	Counter	% $\Delta k/k$		
Replace 4 Reflector Columns with B ₄ C Rods	1	-0.932	-0.928	0.996
	2	-0.890	-0.928	1.043
	3	-0.939	-0.928	0.988
	4a & b	-0.962	-0.928	0.965
Replace 16 Reflector Columns with B ₄ C Rods	1	-1.736	-1.749	1.007
	2	-1.700	-1.749	1.029
	3	-1.727	-1.749	1.013
	4a & b	-1.945	-1.749	0.899

e. Subcritical Reactivity Measurements

V. O. Uotinen

Experimental rod worths, inferred from measurements of reaction rates in subcritical assemblies, depend upon the location of the detector. For example, the worth of four large peripheral B₄C rods was measured in ZPR III Assembly 56b, and the worths translated by four detectors varied by as much as 14% (cf Section 3, above).

This difference is significant and suggests doubts about the validity of other measured large sample worths that have been reported. Such "measured" worths are being used by analysis groups to check calculational methods, although the measurements may, in fact, contain significant systematic errors.

The problem is also relevant to the FTR, since subcritical flux-monitoring will be used in the FTR to monitor the reactivity status, and to measure reactivity worths.

Calculations are being performed with the two-dimensional diffusion theory code 2DBS to investigate this problem.

f. Analysis of ZPR-III Assembly 51

W. R. Young

Small sample reactivity worths in Assembly 51 are over-calculated in absolute value for ²³⁹Pu and ²³⁸U. The values obtained from a 24-energy group PERT IV calculation with fluxes from 2DB core calculation are shown in the tabulation on page 3.33. It should be noted that when ²³⁹Pu and ²³⁸U are removed simultaneously, the effect is a partial cancellation of the calculated-experiment discrepancy because the worth of ²³⁹Pu is positive and ²³⁸U is negative. Integration of the worths along the axis of the core enhances the cancellation effect for mutual removal.

Small Sample Fuel Worths

	<u>Measured Small Central Sample Worth, lh/kg</u>	<u>c/e</u>	<u>Measured Small Sample Worth Integrated Along Core Axis, lh/kg</u>	<u>c/e</u>
^{239}Pu	580	1.31	392	1.28
^{238}U	-23	1.52	-12	1.72

The calculated worth for combined removal can be compared with two experiments in Assembly 51. In the first experiment 2.076 kg of ^{239}Pu , 5.978 kg of ^{238}U , and small amounts, negligible for this discussion, of other materials were removed from the central core drawer, P-16. The worth of this removal was evaluated by integrating the PERT IV results as well as by a pair of 2DB, R-Z reactor calculations. In the reactor calculations the cross sections for the removed materials were flux-volume weighted. These results are listed in the subsequent tabulation. Note that the whole reactor calculation, 2DB, gives a result more consistent with the measurement.

In the second experiment, the worth of replacing the entire contents of the central column with sodium cans was measured. These results are compared with calculations in the tabulation concerning Central Fuel Column Worth in Assembly 51. The calculations were both done with 2DB, using 8-energy groups. The x-y calculations used an axial buckling value of 6.82 M^{-2} for the core and radial reflector region and 4.255 M^{-2} for the sodium filled central column. These values were inferred from the results of the critical measurements on ZPR III Assemblies 52a, b, c, and f. A comparison was obtained of results from flux-volume weighted core cross sections and those from a calculation using homogeneous, non-flux-volume weighted cross sections. A c/e ratio of 1.26 resulted for the latter case.

These results are a partial explanation as to why critical mass calculations are much better than small sample fuel worth calculations. That is, the overcalculation of the negative contribution of ^{238}U , in part, cancels the large ^{239}Pu worth overcalculation. The magnitude of this effect on Assembly 51 is fairly clear and it will be necessary to consider its effect on the FTR design calculations.

Reactivity Change for ^{239}Pu and ^{238}U
Removal from Central Column of Assembly 51

<u>Worth of Fuel in P-16</u> <u>Inferred from measurement, Ih</u>	<u>c/e</u> <u>PERT IV</u>	<u>c/e</u> <u>2DB(RZ)</u>
741	1.24	1.15

Worth of Central Fuel Column in Assembly 51

<u>Measured Worth, Ih</u>	<u>c/e(2DB,R-Z)</u>	<u>c/e(2DB,x-y)</u>
763	1.17	1.17

C. SAFETY AND SYSTEMS ANALYSIS

R. E. Peterson

1. Safety Analysis

D. D. Stepnewski

a. Transient Temperature and Pressure Analysis of HTS Cell Following DBA

P. R. Shire

The DBA may involve a primary pipe rupture which results in the release of sodium coolant and core fission products to a heat exchanger cell. A Model of the HTS Cell and its environs was described by the TAP-3F code to evaluate temperature and pressure conditions following the accident.

The HTS Cell model, Figure 3.C.1, represented the nitrogen plus 2% oxygen at one atmosphere and 100 °F, with steel liner and protective insulation between the floor liner and concrete. Heat flow paths were upwards through the cell ceiling to the containment sphere air and then to the atmosphere. The paths through the walls and floor were to a smeared mass model of the remaining concrete, and then to earth as a sink.

The accident condition spilled 20 tons of primary sodium at 1200 °F on the HTS cell floor and this marked time zero for running the transient problem. It was postulated that one third of core fission products were released to the cell producing a decay heat load. Twenty-one percent of this decay heat load was entered to the cell gas node directly, representing gaseous fission products in the form of noble gases, halogens and volatiles. The remaining decay heat was entered at the sodium node along with ^{24}Na activity and chemical reaction energy from the pool burning of sodium in 2% oxygen.

FIGURE 3.C.1. HTS Cell Features Included in Model for Transient Temperature Analysis Following DBA + Loss-of-Coolant Accident

Output for the transient calculation is shown in Figure 3.C.2, temperature and pressure versus time. Cell temperature and pressure rose rapidly to its maximum 638 °F at 14.6 psig in less than 2 min, then dropped to 571 °F and 12.8 psig at 10 hours. From 10 to 100 hr cell temperature was decreasing at 1 °F per 15 hours. The steel floor liner in contact with sodium responded almost instantly to 1195 °F, then continued upward to its maximum of 1330 °F at 3/4 hr, then declined to 880 °F at 100 hours.

The concrete face temperature under the floor insulation responded slowly but surpassed the side wall temperature at 20 hr and 400 °F; at 100 hr it was 588 °F and still rising. The wall temperature 472 °F was still rising slowly at 100 hours.

FIGURE 3.C.2. Temperature and Pressure Versus Time Following DBA and Loss of Coolant Accident

to a sink at 418 °F using a film coefficient of 100 Btu/hr ft² °F between the bottom of the oxide insulating slab and the coolant. The radiation cooling case results in a limiting fuel layer thickness of 1.2 inches. In any event, the 1-in. fuel layer case was used in subsequent analyses.

A transient heat transfer analysis was performed for the geometry discussed above using NUTIGER. The 1 in. of fuel and 3 in. of oxide insulator were divided into 40 internal nodes of 0.1-in. thickness. The fuel nodes had internal heat generation which decreased from a maximum of 28 °/cm³ to 13.5 W/cm³ at 1 hr and to 7.5 W/cm³ at 10 hours. The temperature history of the system is shown in Figure 3.C.4. This figure shows that the insulator melts into a maximum depth and then resolidifies as the power level decreases. Figure 3.C.5 shows the heat flux variation from the upper fuel surface and lower oxide surface.

Three inches of oxide is shown to be a very effective barrier which limits the heat flux through the cavity floor. This means temperature gradients through the steel liner and cooling system may be kept to a reasonable level. The analysis given here disregards the effects of convection within the slab.

c. Machinery Dome - Reactor Cavity Venting

G. L. Fox

In one variation of the FFTF Containment Design Basis accident, the sodium-in case, it is postulated that the reactor vessel cover is lifted and releases vaporized core materials to the machinery dome. Previous calculations (P. R. Shire) show an initial peak pressure of 86 psia occurring in the dome. It is desirable to reduce this peak by venting to the reactor cavity and possibly to other cells within the first containment barrier of the plant system.

FIGURE 3.C.4. *Temperature History of Oxide-Fuel System (Upper surface cooled by boiling stainless steel layer, $T_{\text{boil}} = 4400$ °F; lower surface cooled by sink, $T_{\text{coolant}} = 500$ °F)*

FIGURE 3.C.5. *Surface Heat Flux Rates*
 (Upper surface cooled by boiling
 stainless steel layer,
 $T_{\text{boil}} = 4400^\circ\text{F}$; $T_{\text{coolant}} = 500^\circ\text{F}$;
 Fuel Thickness = 1 in.;
 Oxide Thickness = 3 in.)

A computer program was written to solve for transient pressures and temperatures when gas flows from one cell volume to another during a Design Basis Accident venting situation. The program is written in MIMIC, an analog simulation language, and will handle up to four cells connected in series by three flow ducts. Heat transfer within the cells and entrance and discharge coefficients are included. The following assumptions apply:

- (1) Fixed composition gas, i.e., constant isentropic exponent and gas constant.
 - (2) Quasi-static flow condition, (i.e., equilibrium equations apply at each instant).
 - (3) Friction and heat transfer within flow ducts is negligible.
 - (4) Sonic flow exists if pressure ratio between cells is greater than critical.
 - (5) Bernoulli's compressible flow equation applies for subsonic flow.
 - (6) Gas flow will be in same direction for all flow ducts.
- The program has been debugged and is ready for operation.

Sample calculations which represent venting from the machinery dome to the reactor cavity which is connected in turn to equipment cells is shown in Figure 3.C.6. The first case considers all flow duct areas to be 10 ft^2 . The initial pressure of 86 psia corresponds to the maximum machinery dome pressure, while the other cells begin at atmospheric pressure. Only the pressure in Cells 1 and 2, the machinery dome and reactor cavity, respectively, are shown for simplicity. From the figure it is apparent that pressure is equalizing at 30 psia in about one second for the 10 ft^2 venting area.

FIGURE 3.C.6. Pressure History During Cavity Venting

d. Transient Analysis of Reactor Cavity Response
to DBA

R. D. Peak

A digital computer program has been prepared for the determination of transient temperatures and pressures within the reactor cavity following the unlikely occurrence of a core melt-through accident. This work has involved the conversion of the digital computer program CONTEMPT. This program, described in IDO-17220, analyzes the energy-mass-pressure-temperature responses of a single containment which can be described by a vapor region (nitrogen and sodium vapor), a liquid region (sodium liquid), and up to 20 heat conducting regions (such as reactor vessel wall, guard vessel wall, radial shield, bottom shield, cavity wall, floor, and roof). This program was originally prepared to analyze the blowdown-loss of coolant accidents of water reactors in air containments. The main conversion was to substitute sodium and nitrogen properties for the water and air properties used by the program.

The converted code will be used to analyze conceptual designs of the reactor cavity cooling system as shown on SK-3-14636⁽¹⁾ and variations of this system.

e. Reliability Analysis of the Plant Protection
System (PPS)

O. B. Monteith

The analysis of the PPS is continuing. During this reporting period the rod scram release circuit design has been revised to include additional relays for the purpose of increasing the independence of the logic from failures in the relays. The revision presents a complicated arrangement of components for establishing its reliability diagram and math model.

1. *Not shown in this report.*

The long term availability of the revised rod scram release circuit for various testing intervals is shown in the accompanying table. These data will be used in the reliability analysis of the overall plant protection system. As a result of the circuit revisions, circuit unavailability has been reduced somewhat, as compared to the original results for comparable conditions.

ROD SCRAM LOGIC CIRCUIT UNAVAILABILITY

Failure Rate Range	<u>Hours Between Test</u>		
	8 hr	168 hr	1500 hr
Worst	1.2×10^{-6}	8×10^{-4}	0.06
Nominal	1.2×10^{-8}	5.4×10^{-6}	6×10^{-4}
Optimistic	1.2×10^{-12}	5.2×10^{-10}	4×10^{-8}

2. Systems Analysis

H. G. Johnson

a. Heat Transport System Analysis

(1) Dump Heat Exchanger Analysis. The DHX digital simulation using only six nodes each for the sodium, metal and air portions of the DHX is operational, and has been checked for steady-state operation. Transient operation is still being checked and compared to the reference model. This simulation predicts a steady-state heat transfer rate of 140.8 MW_t for the 258 tubes/pass DHX for flow and temperature conditions for which the TAP DHX-simulation calculates a rate of 135.5 MW_t and the ADOPT code, 133.3 MW_t . Comparison of temperature versus distance plots for the three cases shows the six-node model agrees very well for about the first third of the distance through the DHX for both the air and sodium fluids, then a slight but steady departure from the reference values occurs. This behavior, together with the long distance (25 ft) spanned by one node in the simple model, may indicate that the 4% error

here is entirely in the coarseness of the nodal mesh used. In order to check this hypothesis the model is currently being expanded to 12 nodes.

Both the reference TAP model and the six-node model were set up to compute coefficients and thermophysical properties on the basis of both local temperatures and on the arithmetic average of the end temperature conditions. A comparison of exactly the same case using both the TAP model and the six-node representation for both local and average conditions indicated that there were no significant differences in end conditions for steady-state calculations. (Node temperatures in the center of the exchanger differ in the two cases by a maximum of about 5 °F.) This feature still must be checked for transient runs, but indications are that the use of average properties is quite satisfactory. The great advantage to using average properties with the six-node model is that the computer run time is about halved over the case using local properties.

3. Control Analysis

R. A. Harvey

a. Driver Channel Instrumentation Analysis

J. J. Regimbal and G. H. Dickerson

The TIGER V code, used for studying the effectiveness of thermocouples and flowmeters in preventing sodium boiling, was modified to utilize previously computed information more effectively. The code change permits magnetic tape storage and retrieval of initializing data so that a coolant flow loss ramp can be extended beyond that of a previous run without the need to recompute the previous run. Alternative solutions, depending upon the mode of scram or power cutback, may also be computed from channel conditions stored at the time sensor trip settings are reached.

Limiting conditions for coolant boiling prevention by protective action were found for a variety of postulated flow loss occurrences. The flow is reduced steadily to an arbitrarily chosen final level, and temperature transients of the fuel, coolant, and thermocouples are computed. Safety system action occurs automatically when trip settings for flow or thermocouple temperatures are exceeded.

A typical solution is presented in Figure 3.C.7 for the case of flow reduction at a rate of 30%/sec to a level of 30% flow. Safety rod release occurred 100 msec after the thermocouple junction temperature reached 30 °F above the steady-state value. Disregarding effects of superheating and the sodium static head, boiling would have barely occurred at the core outlet despite protective action. If flow reductions to a level somewhat smaller or greater than 30% occurred at this same rate, the maximum SAFE thermocouple response time to prevent boiling is also smaller or greater, respectively (Figure 3.C.8).

4. Safety Technology and Practices

J. W. Hagan

ANL Participation in FFTF Safety Programs

Specific areas have been identified for more direct participation by ANL in the FFTF safety programs to ensure that the LMFBR safety technology at ANL is utilized to its fullest extent in the FFTF studies. The areas of immediate concern for the PSAR and its supporting information include: (a) Performance of the maximum accident studies which establish the DBA, and (b) Evaluation and specification of the fuel/cladding design bases and, correspondingly, damage criteria for the fuel.

The ANL SASIA accident analysis code is being incorporated in the FFTF studies to identify fuel failure thresholds and locations under hypothetical accident conditions. Adaptability

FIGURE 3.C.7. Core Outlet Temperature Due to 30%/sec Flow Reduction with Protective Action

FIGURE 3.C.8. Maximum Sensor Response in Time Permitted to Prevent Boiling at Core Outlet

of the code to an additional pin failure model is being considered an ANL prior to the actual runs for the PSAR input. The initial faulted conditions assumed for these runs have been tentatively established and are now being evaluated.

IV. ENGINEERING DEPARTMENT

D. L. Condotta

A. REACTOR SYSTEMS1. Reactor Design

S. O. Arneson and R. C. Walker

a. Reactor Core

J. F. Wett

BNW Configuration Control Board approval has been obtained and the following Design Description was transmitted for RDT review and approval:

- CCDD-35 - First Core Fuel Assembly Component

b. Reactor Vessel and Shield

R. C. Walker and O. W. Priebe

The Request for Proposal package for the reactor vessel, head, guard vessel and associated equipment was issued to Babcock, Wilcox, Combustion Engineering, and Chicago Bridge and Iron on May 27. Proposals are requested to be submitted to Westinghouse on July 7.

The scope of work for continuation of reactor vessel studies by Babcock and Wilcox was approved by RDT.

2. Reactor Equipment

S. O. Arneson and C. A. Munro

a. Control Mechanism Development

J. C. Noakes, P. K. Telford

(1) Roller Nut Testing. Agreement was received to use the replacement N. W. Savannah roller nut CRDM for the initial test at LMEC.

(2) CCDD No. 33 Support. A document is being issued recommending that the roller nut concept be used for the control

and secondary safety systems; the magnetic jack concept be used for the primary safety system; and the BNW pneumatic drive be used as the alternate concept in the event that either of the other drives fails to meet the design requirements.

b. Reactor Hardware

J. W. Kolb, B. G. Smith, L. R. Besel, R. F. Gilmore

Removable Driver Fuel Instrument Assembly. A Development Work Statement draft for verification of the removable driver fuel instrument assembly concept has been completed with internal FFTF comments incorporated. The purpose of this program is to determine, by testing, the feasibility of remotely removing and re-installing the removable instrument assembly through a curved guide tube during reactor operation. The draft is being forwarded to Westinghouse and the BNW Laboratory for comment and preliminary planning.

c. Closed Loop Nozzle Closure Program

R. F. Gilmore, J. W. Kolb

Welding and Cutting. The primary objective of this task is to evaluate methods to reliably separate the welded closed loop nozzle closures by remote control. The current design approach is the trepanning method using two single-point groove-cutting tools and two facing tools. The remote cutting machine is designed to separate completely the closed loop tube from the nozzle and prepare the joint for subsequent rewelding in a single cutting operation. Alternative methods to separate the welded nozzle closures are also being studied.

Shop fabrication drawings of the remote cutting machine are completed. Final tool bit design configurations have been established. Module testing of the remote cutter head assembly is being planned to verify positive chip control, confirm cutter speed and feed rates, and to determine actual cutter loads and the vacuum system necessary to exhaust the chips away from the cutting surface.

3. Core Development and Analysis

S. O. Arneson and J. R. Carrell

a. Sodium Flow, Hydraulic and Mechanical Testing of FFTF Fuel Assemblies (CFD-3) J. M. Yatabe

(1) Coolant Mixing and Flow Distribution. Continued evaluation of the absolute resistance technique for flow mixing determinations has identified a strong temperature sensitivity effect averaging 1.25% of measured resistance per degree Fahrenheit. Therefore, it has been concluded that the water temperature in the mixing tests must be controlled to ± 1 °F in order to produce adequate data. Using careful temperature control and other appropriate operation techniques, mixing data reproducible within $\pm 4\%$ was obtained in trial runs.

(2) Sodium Flow Testing. The Mark I assembly (a full-sized fabrication prototype) has now been tested to approximately 3600 hr at full flow and temperature conditions (400 gal/min, 1060 °F in the Core Component Test Loop at Argonne National Laboratory. The fuel assembly was removed from the facility, cleaned and returned to BNW for detailed inspection as previously reported. The assembly appeared to be in excellent condition, except for a slight loosening of the wire wrap spacers on the outer pins.

Seventy pins have now been removed from the test assembly, and the surface conditions of the pins appear to be excellent. However, by visual examination, the degree of wire wrap looseness appears to be decreased in the central pins of the assembly. Planned quantitative measurements of wire stretching on a number of individual pins are required before any conclusions can be derived on the relationship between spatial position and wire loosening.

(3) Loose Wire Wrap Effects. It has been postulated that wire wraps may loosen during reactor operation because of differential swelling. If movement of this slack wire could occur relative to the bundle, the coolant channel configuration could change, giving rise to hot spots. In addition, increased vibration amplitudes in the assembly could result, leading to more rapid wear rates.

To identify potential wear problems, a 7-pin fuel assembly model has been fabricated (three pins with wire stretched 0.020 in. and three pins with wire stretched 0.030 in.) and tested in the Small Components Test Loop. The conditions for the life test are steady sodium flow velocities of 30 ft/sec and temperature of 1150 °F. Sodium purity was maintained at below 10 ppm oxygen and 20 ppm carbon as measured by analytical chemistry performed on samples. The assembly has been exposed to these conditions for approximately 800 hr before removal for an interim examination.

The interim examination results are qualitative since no destructive examination could be permitted on the test assembly. Photographic and direct visual examination indicated no evidence of wear on cladding or wire wrap surfaces. The assembly was re-inserted into the loop and life test conditions re-established. The duration of the final phase of the test will be 3,000 hours.

b. Heat Transfer Characteristics of FFTF Fuel Assemblies
J. M. Yatabe

The Small Heat Transfer Loop modification to permit 7-pin bundle testing was completed and checkout tests demonstrated that all added components (e.g., pump, power supply, and heat dump addition) are performing properly. During the modification, a test on a single pin in water was run to give greater validity to the quality assurance being applied to the pins to be used for 7-pin bundle testing. A heat flux of

1,700,000 Btu/hr-ft² (water temperature 300 °F) was achieved without failure. However, the internal thermocouple was found to have shorted to the cladding. This situation will be rectified in the 7-pin bundles by having the vendor pre-oxidize the bare thermocouple wires to provide additional insulation.

c. Development Facilities Management

K. G. Toyoda

(1) Hydraulic Core Mockup Test Program (H. Leigh, J. Muraoka, D. L. Ballard, D. M. Jackson, L. R. Sweetin, J. Spalek, G. W. Riedeman). The objective of the HCM test program is to provide development and testing support for resolution of the FTR vessel and core hydraulic and mechanical design problems. The inlet feature model is a part of this program. Currently the original inlet model is being modified to more closely simulate the vertical core concept V-a. The complete modification will be made in steps and testing will proceed after each step. The initial model modifications include the following:

- The inlet end of all flow ducts were cut off, leaving an open end entry
- The ducts were anchored to the upper tubesheet to simulate a single tubesheet configuration
- The original inlet basket was reinstalled and used until the concept V-a basket is available
- The new steel elliptical head and a safety support were installed.

(2) Sodium Core Development Facilities (CM-3) (K. D. Hayden, R. R. Wyrer). Construction work is progressing on the FFTF Sodium Facilities Building (Bldg. 335). The contractor has placed all underground piping and conduit and has completed the pouring of building footings. The two 750 VA surplus

electrical transformers and 440-V motor control center from the deactivated reactor areas have been transferred to J. A. Jones warehouse for overhaul and storage. Vitro design work has been progressing on the LMFBR Core Segment Development Facility.

Preliminary design criteria and conceptual flow diagrams were prepared for the HTSF Sodium Receiving and Processing System and the HTSF Inert Gas Systems. The sodium system provides storage and cold trap purification for approximately 30,000 gal of sodium. The inert gas system provides storage for approximately 45,000 SCF of gaseous nitrogen and 90,000 SCF of gaseous argon at 750 psi.

(3) Sodium Facilities Operation (J. D. Berg, L. J. Defferding). The objectives of this section and description of equipment were given in BNWL-915.⁽¹⁾ The status of the sodium facilities is given in the following:

- Small Component Evaluation Loop: The FFTF fuel subassembly was removed, cleaned, examined and reinstalled in the loop for an additional 600 hr run. During the downtime a new temperature-power controller was installed on the main 12 kW heater.
- Small Heat Transfer Loop: Following the loop upgrading, checkout, and operator training, operation was initiated (utilizing a dummy test section) to clean up the sodium. A single pin test assembly was installed 5-19-69 and testing initiated on 5-22-69. For this series of tests, the simulated fuel pin power and loop temperature were increased simultaneously from zero power at 400 °F to about 30 kW at 800 °F with the system flow rate at 20 gpm. The flow was then reduced to establish the prototypical temperature rise across the test pin. The pin power and test section

1. FFTF Monthly Informal Technical Progress Report, BNWL-915. Battelle-Northwest, Richland, Washington, October 1968.

discharge temperature were cycled four times. The test section was removed from the loop and steam-cleaned according to standard procedures. After cleaning, the test section was leak checked and reinstalled on the loop for the second phase of this test. The second phase of the test included the determination of the performance limits for this test pin. During this phase of the test, a sodium leak and fire started in the test section enclosure when a break occurred in the end regions of the test assembly. The sodium ignited immediately on contact with the air in the metallic test section safety enclosure; the Dowtherm in the system used to cool the electrical leads also contributed to the fire. The fire was completely restrained within the test section safety enclosure. No delay to the program resulted.

- Static Sodium Pots: Testing (cycling pressure and temperature) of the pressure transducer in Pot No. 3 continued. Pots 1, 2 and 4 remained shut down, however, a pressure transducer is being readied for installation in Pot No. 4.
- Sodium Purification Loop: The cold trap inlet line trace heater was replaced (failure reported last month) by 5-2-69 and loop started up on 5-3-69. This system has remained in operation at 600 °F bulk temperature with direct reading O_2 and carbon probes in service.
- Fission Product Screening Loop: A new primary pump throat (failure reported last month) was fabricated and delivered 5-22-69. Installation of the pump throat and flow meter tube holders is in process.
- Mechanical Properties Loop: Idle this month, test section being fabricated.
- FFTF Sodium Facilities Building: A new 60 × 100 ft. building is under construction (completion about 12-15-69) to

house the Small Component Evaluation Loop, Small Heat Transfer Loop, and the Static Sodium Pots. In addition, spare parts and equipment storage space is provided along with a maintenance shop area.

4. Refueling and Radioactive Maintenance Systems

S. O. Arneson and W. E. Cawley

a. Conceptual System Design Descriptions

E. J. Ruane

System 42 - Nonirradiated Fuel Handling (E. D. Grazzini).

The system 42 CSDD has been revised to include additional requirements for pre-irradiation testing equipment and has been resubmitted to the BNW Configuration Control Board for review and approval.

B. FLUID SYSTEMS

B. M. Johnson

1. Heat Removal System

T. W. Withers

a. Sodium Pumps

W. Babcock

The Westinghouse Design Study is on schedule. Contractual limitations of time and money did not permit pursuit of the following problems in this study identified as follows:

- An uncertainty exists regarding the impeller capability to resist creep at temperatures over 900 °F.
- An inner tank liner is subject to cracking in the combination of high temperatures and transients.
- Shaft dynamic problems indicate that a second sodium bearing may be required for damping. At 500 rpm, the shaft deflection exceeds the bearing clearance. The torsional critical speed is too close to operating speed.
- The NPSH requirements have not been achieved.
- Very little substantive work was accomplished on vortexing problems.

The revised pump program and testing plan was completed and is being prepared for internal comment before submittal to RDT.

The revised state-of-the-art report on sodium pumps was published and distributed.⁽¹⁾

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1. W. B. Babcock. State of Technology Study -- Pumps, Experience With High Temperature Sodium Pumps in Nuclear Reactor Service and Their Application to FFTF, BNWL 1049 (OUO). Battelle-Northwest, Richland, Washington, June 1969.

b. Intermediate Heat Exchangers

L. A. Whinery

Babcock and Wilcox Company work activity on the FFTF IHX design study contract (BNW Contract No. BDR-571) was devoted primarily to development and evaluation of alternative IHX design layouts. This work progress is reported in a preliminary draft of Babcock and Wilcox Report No. BAW-BDR-571-1, FFTF IHX Exploratory Design Studies - Interim Report - Concept Selection. Twenty-two alternative IHX concepts were qualitatively evaluated by Babcock and Wilcox and two interim designs were selected as recommended bases for follow-on study. Table 4.B.I. summarizes major design parameters.

Concept I features a helical coil tube bundle with floating lower head. Secondary sodium inlet and outlet piping are routed concentrically through an overhead shield plug and must be cut prior to tube bundle removal. Tubes are attached to tubesheets using a crevice-free and backside weld capable of being inspected. Other than flow impingement baffles at the primary inlet, no shellside flow distribution baffles are proposed, thus leading to a low-pressure drop unit.

Concept II utilizes a straight tube bundle with floating lower head. The tube bundle is removable after cutting secondary inlet and outlet piping. Also, a rotatable upper shield plug is provided with index holes permitting access for in-place tube plugging using an explosive welding technique being developed by Babcock and Wilcox. Tube-to-tubesheet welds are of the partial penetration, J-groove type.

In addition to the on-going Babcock and Wilcox design studies, NYO has also negotiated work scopes with Westinghouse and Foster-Wheeler for performing LMFBR Program IHX design studies of a nature generally supporting the FFTF needs.

TABLE 4.B.I. *IHX Design Parameters Resulting from Preliminary Calculations*

Material (Type 304 SS), operating conditions, and design conditions for both Concept I and Concept II are identical in the following areas.

<u>Operating Conditions</u>		<u>Design Conditions</u>	
Heat Load	175 MW	Primary Sodium Pressure	200 psig
Primary Sodium Flow Rate	4.9×10^6 lb/hr	Temperature	1200 °F
Inlet Temperature	1200 °F	Secondary Sodium Pressure	250 psig
Outlet Temperature	800 °F		
Pressure	180 psig	Temperature	1200 °F
Secondary Sodium Flow Rate	4.9×10^6 lb/hr	<u>Heat Transfer Results</u>	
Inlet Temperature	700 °F	Log Mean Temperature Difference	100
Outlet Temperature	1100 °F		
Pressure	215 psig		

Certain dimensions and other values identified with the two concepts differ in the following areas.

	CONCEPT I SK-3581-700 ^a	CONCEPT II SK-3581-100 Rev. 1 ^a
<u>Tube Bundle Dimensions</u>		
Bundle Outer Diam, in.	80	55
Bundle Inner Diam, in.	42	26
<u>Tube Size and No.</u>		
Quantity	300	1500
Diameter, OD in.	1.5	0.75
Pitch, in.	3.5 (on tubesheet)	1.25
Minimum Wall Thickness, in.	0.060	0.035
<u>Heat Transfer Results</u>		
Overall Heat Transfer Coefficient	1050 Btu/hr-ft ² -°F	1440 Btu/hr-ft ² -°F
Heat Transfer Surface (min.)	6300 ft ²	4800 ft ²
<u>Pressure Drop Calculations</u>		
Total Primary Pressure Drop, psi	3.5	5
Total Secondary Pressure Drop, psi	6	8
<u>Major Dimensions</u>		
Shell Length, ft	36.5	32.5
Shell OD, in.	93.5	62.5
<u>Sodium Inventory</u>		
Shellside	33,000 lb	13,000 lb
Tubeside	30,000 lb	16,000 lb
<u>Removable Bundle Weight</u>	137,000 lb	73,000 lb

a. These SK drawings not shown in this report.

2. Auxiliary Systems

R. V. Dulin

a. System 81 - Sodium Receiving and Processing

The CSDD has been approved and issued for this system. The document is being revised currently to incorporate the approved Design Safety Criteria. The architect-engineer (AE) has completed review of the conceptual design and is proceeding with design studies on mixing tees, freeze traps, and sodium pumps. LASL is in the process of analyzing design and performance characteristics of various cold trap configurations in support of FFTF.

BNW is providing study results on the following studies:

Contaminated Sodium Disposal Method

Cross Contamination Potential

Material Selection for Sodium Services

Material Surveillance Methods

Impurity Removal and Control Methods

Controlled Addition of Impurities

b. System 82 - Inert Gas Receiving and Processing

The CSDD for this system has been approved and final issuance is awaiting incorporation of the approved Design Safety Criteria. The AE has completed review of the conceptual design and is proceeding with design studies on sodium vapor traps.

BNW is directing a design study by GE on the impact of failed and vented fuel operation on plant design.

c. System 83 - Auxiliary Heating and Cooling

The AE is preparing the CSDD for this system, which is to include those process cooling functions requiring liquid metal as the heat transfer medium as well as the heating function for all sodium-containing equipment. The CSDD is in early draft status.

d. System 84 - Decontamination and Cleaning

This system will include primarily the decontamination and cleaning of installed sodium piping systems and large components removed for maintenance. The equipment and processes required for such cleaning and decontamination present a number of technical problems, and a CSDD is considered the best way to bring these technical problems into project focus. The CSDD is in preparation.

e. System 85 - Chemical Analysis Facility

This facility is to include all of the radioactive and non-radioactive analysis required at the FFTF site. The AE has completed a final issue of the CSDD to be submitted for BNW comment, approval, and submission for RDT approval.

Since the Auxiliary Systems are primarily service systems for the major FFTF systems, detailed concepts are affected by concepts for those major systems. Thus, it will probably be necessary to make a final "round" of Auxiliary Systems concept revisions when major system concepts are finally established.

3. Irradiation Facilities

B. M. Johnson and J. R. Carrell

a. Short-Term Irradiation Facility - (STIF)

P. F. Shaw and M. K. Mahaffey

The STIF CSDD was approved by CCB. A justification for the facility was prepared for transmittal to RDT.

C. PLANT SYSTEMS

E. M. Johnston

1. Structure and Utilities

F. A. Shadel

a. Reactor Containment System (System No. 27)

C. C. Steele and N. Vieli

Representatives of BNW, Bechtel, Westinghouse, Atomics International, Aerojet-General, and RDT met at AEC Headquarters in May to review the conceptual design of the FFTF containment system described in CSDD No. 27. The following essential features of the containment system were agreed upon:

- A two-barrier containment arrangement.
- A spherical outer containment vessel.
- Closed loop equipment cells located within the reactor outer containment vessel.
- Primary sodium systems located within the reactor containment vessel.
- An effectively flat operating deck, with deviations from this objective individually justified.

CSDD No. 27 for the Reactor Containment System was approved subject to RDT comments.

b. Seismic Consultant Agreement

F. H. Shadel

BNW has retained John A. Blume and Associates, San Francisco, as the FFTF Consultant for seismic design and earthquake engineering.

2. Electrical Systems

G. H. Strong

a. Primary Electric Power System (System No. 11)

M. F. Wiitala

The Bonneville Power Administration submitted a drawing showing proposed protective relaying for the 115 kV system. The relaying scheme is based on information contained in CSDD-11 and Bechtel Engineering Report No. 24, CSDD-11. The scheme utilizes carrier current transfer tripping on the circuit breakers at the Midway-Benton tieline terminals and the FFTF sectionalizing circuit breakers to obtain circuit breaker reclosing in approximately 35 cycles (0.58 sec) after a transient fault. A review of the past history of the frequency of occurrence of transient faults on the tieline and an analysis of the effects of such faults on the operation of FFTF indicated that the fast reclosing was not necessary. BPA has been requested to revise their protective relaying diagram on the basis of a 1-sec reclosure time. The new diagram will be used by Bechtel as a basis for design.

3. Fuel Examination System

C. L. Boyd

Recommended Methods of Gas Cooling Driver and Test Fuel Assemblies

E. B. Ramsey

BNW submitted the following recommended methods to RDT for concurrence on the gas cooling of driver and test assemblies:

- Axial suction recommended for gas cooling during transfer, storage, examination and disassembly-reassembly operations with the fuel assembly flow duct in place.
- Transverse suction recommended for gas cooling during examination and disassembly-reassembly operations with the fuel assembly flow duct removed.

- A combination of axial and transverse suction recommended for transition gas cooling during removal and replacement of the fuel assembly flow duct.

The major advantages of the recommended methods are better control of sodium vapor, contamination, and hot exhaust gas; and easier monitoring of cooling gas outlet temperatures.

An additional consideration in the selection of the transition gas cooling method is the need for similarity of axial and transverse cooling methods; that is, a combination of either axial and transverse pressurized cooling or axial and transverse suction cooling. This requirement is based on data from transition cooling tests reported in BNWL-1062,⁽¹⁾ where a combination of pressurized axial flow and transverse suction flow resulted in sharp cladding temperature peaks indicative of poor flow distribution. Preliminary analysis showed that changing of the axial flow to suction would significantly reduce the temperature peaks occurring during simulated liner removal. Both GE and ORNL, who are engaged, respectively, in conceptual studies for Axial Gas Cooling Hardware and Disassembly-Reassembly Equipment are in agreement with the recommendations.

1. FFTF Monthly Informal Technical Progress Report, BNWL-1062, Battelle-Northwest, Richland, Washington, April 1969.

D. INSTRUMENTATION AND CONTROL

C. D. Swanson

1. Plant Instrumentation

J. W. Mitchell

a. Central Control and Data Handling - System 91

G. L. Waldkoetter, D. W. Hartmann

Bechtel's estimates run as high as 2800 preheat thermocouples as compared with the estimate of 1500 in CSDD 91. Until piping layout and pipe heating design are firmer, the higher number will be used for sizing the system.

A brief study on the feasibility of using two computers instead of the planned three for the CC&DH was completed. In the study, it was concluded that the identified system requirements can be met with either a 24 or 32-bit word size, 2-computer configuration. However, since the expected increase in computer usage will be in the area of calculations, the 32-bit word size computer appeared to be the better choice. Further study of this subject will continue during preliminary design.

b. Process Data Displays

B. E. Dozer

The 20-channel, field-effect transistor multiplexer has been received from Interstate Electronics and testing is nearly completed. In general, most of the tests meet the manufacturer's specifications.

2. Instrumentation Development

M. R. Wood

a. Low-Level Neutron Flux Instrumentation

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

This task encompasses the developmental test and evaluation of sensors, cables, and other components for high-sensitivity, low-level neutron counting systems. Both the

reactor startup and subcriticality instrumentation channels will experience severe environmental conditions of up to 10^6 R/hr gamma at temperatures (initially) of 900 °F and finally of 1200 °F.

Engineering developmental tests were conducted during the reporting period to determine the performance of high-temperature coaxial cables at temperatures up to 1200 °F. Measurements employed the integral bias curves technique and involved the components and instrumentation previously used for the high level gamma and temperature tests of the sensors. For direct comparison, spurious noise signals were measured using the same technique.

Transmission of the high voltage and signal through the heated zone of the cable encountered no operational problems over the full range of test temperatures. Preliminary short-term measurements, using short cable sections, revealed a cable noise problem at temperatures of about 600 °F and higher. As observed, lower detector voltages reduced the magnitude of the noise problem. Further characterization of the noise generation problems and methods for minimization or elimination will require further investigation.

Future plans include testing the detector and a greater length of cable (approx. 20 ft) to both temperature and gamma radiation fields. The silica-insulated cable will be placed in a furnace for up to one month to determine whether a time-dependent noise component is present.

b. Pressure Δ P Flow and Level Sensors for Sodium Service
R. D. Crosier, K. O. Creek

The objective of this task is to develop pressure and level instrumentation for FFTF applications. Pressure testing of the Consolidated Electrodynamics Corp. transducer 4-361-0001 SN-2260 (an unbonded strain-gage type rated at

600 °F and 250 psi) has been completed successfully at temperatures of 625 °F and 650 °F in the sodium pots being tested at 675 °F.

c. Driver Test Position Connector Concepts

R. D. Crosier, S. K. Meyers and K. O. Creek

A specification for an improved prototype closed loop connector including quality assurance requirements was sent to seven connector manufacturers to obtain their proposals, but no bids were received. The vendors are being questioned to determine their objections and to work out a suitable specification.

d. Surveillance Sensors

R. L. Brown and K. O. Creek

The objective of this task is to evaluate the Sperry wheel ultrasonic inspection unit for temperature and radiation tolerance and for application in inspecting FFTF vessel and piping welds.

A Sperry wheel unit has been procured, and initial evaluation of the components and materials in the wheel has been completed. A list of materials expected to fail because of radiation or temperature damage has been prepared. As originally delivered, the wheel transducer could not withstand the projected FFTF environments. The unit was intended to inspect a piping surface at 500 °F while the remainder of the wheel unit was maintained at a much lower temperature by artificial cooling.

Alternative materials for the components subject to damage are being investigated in order to recommend replacements.

e. Subassembly Instrumentation Testing in Sodium
M. J. Rieb

The task objective is to perform tests in sodium on full-size driver and open test position developmental instrumentation at reactor core subassembly flow and temperature conditions. Authorization to start negotiations with Westinghouse will start design of the test section as soon as contract negotiations are completed. This new loop is now filled with sodium and is undergoing the initial startup process.

E. SODIUM AND GAS TECHNOLOGY

B. M. Johnson

1. Corrosion and Chemistry

D. W. Shannon

a. Evaluation of Radioisotope Transport and Deposition

W. F. Brehm, E. A. Kovacevich, J. J. McCown

Sections of the dismantled Radioisotope Transport Loop were submitted for study of composition gradients near the metal-sodium interface. Preliminary results indicate a decrease in Cr, Ni, and Mn concentration near the metal surface of hot-leg samples. This is expected. The study is still underway.

Thermal convection loop TC-8 was shut down after 816 hr operation at 1300 °F maximum temperature, with 900 °F minimum. The distribution patterns for ^{51}Cr , ^{54}Mn , and ^{60}Co are being measured using a germanium diode detector. This is the third of the five different maximum temperature runs planned for TC loops.

b. High Level Analytical Facility Design

J. F. Jarosch, L. F. Lust

A study of the pros and cons of making the inert atmosphere portion of the hot cell mockup evacuable has been completed. It was concluded that the costs and inconvenience of an evacuable cell were not warranted. Design of a non-evacuatable cell is continuing and purchase of associated equipment has been initiated.

Laboratory tests of cleaning methods for sample tubes are continuing. Preliminary data indicate that mechanical abrading increases the oxide contribution of the tube rather than decreasing it as had been supposed.

A paper entitled "Techniques for Determining Impurities in Sodium Coolant" authored by R. F. Keough, K. C. Knoll, L. F. Lust and E. L. Osborne was presented at the Northwest Regional Meeting of the American Chemical Society on June 12.

c. Selection and Testing of FFTF Sodium-Sampling Methods

K. C. Knoll

Discovery and elimination of "bugs" in the prototype bypass loop sampler continues. Considerable difficulty has been encountered in maintaining the sample tube furnace at a uniform temperature. Several minor modifications have been made to improve temperature control and other performance characteristics. Plans are being made for the installation of smoke detector actuated solenoid operated valves and other necessary safety features to allow unattended operation of the sampler.

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

G. B. Barton

Gas Sampling System Development. A new vapor trap design has been conceived and a prototype constructed. The new trap design is based on the premise that the bulk of the sodium which escapes conventional traps does so as an aerosol mist. The new trap design breaks the aerosol through a re-vaporization followed by a second condensation. The prototype is designed for three re-vaporization thermal cycles. Initial tests show considerable promise.

Theoretical calculations leading to scale factor recommendations for packed and baffle traps are underway and should be completed early in the next report period.

e. Selection and Testing of In-Line Sodium Analysis Instruments

J. J. McCown

(1) Oxygen Meter. The UNC oxygen meter continues to operate satisfactorily on the sodium purification loop. Several attempts to add carbon compounds to the loop had no effect on the oxygen meter. Chemical analysis indicated no change in the oxygen or carbon content of the loop

(2) Carbon Meter. A spectrographic grade carbon rod was placed in a bypass addition tube on the sodium purification loop and contacted with 600 °F sodium for 40 hr. The carbon meter did not show any change in output signal during the test. Upon removal from the loop the carbon rod was found to be intact with no sign of erosion, etc. Chemical analysis showed no change in the total carbon content of the sodium (5 ± 2 ppm).

f. Decontamination

G. R. Bloom

Chemical cleaning procedures used for decontaminating water reactors are being tested for the decontamination of sodium reactors. One proposed procedure would (1) oxidize a layer of stainless steel about 0.1 mils deep, and (2) removed the oxidized layer with a standard water reactor decontamination process. The activation products diffused into the stainless steel would be oxidized along with the stainless steel. They would then be removed in the second step of the decontamination process along with the oxide film.

Samples from the dismantled Radioisotope Transport Loop were processed with a one cycle steam and citrate method and the results reported last month. One of the samples

was recycled through the oxidation and the decontamination step, and recounted. Both the new and the old data are given below to give a comparison.

	<u>Decontamination Factors</u>	
	1 Oxidation Decontamination	2 Oxidation Decontamination
	<u>Cycle</u>	<u>Cycles</u>
Co ⁶⁰	2.4	2.1
Mn ⁵⁴	1.1	1.9
Cr ⁵¹	1.6	2.4

The Co⁶⁰ decontamination factor did not change, the apparent decrease was attributed to counting error. The Mn⁵⁴ and Cr⁵¹ decontamination factors increased as another layer of metal was removed during the second oxidation-decontamination cycle. Calculated uniform penetrations from weight loss data were 0.3 mils for the first cycle and 0.6 mils for two cycles.

F. METALS, MATERIALS AND CODES

J. C. Tobin

1. Materials Applications

J. E. Irvin

a. Weldment Studies

A. L. Ward, A. J. Lovell

The objective of this study is to describe the effects of fast-reactor irradiation and environment on the tensile and creep-rupture properties of weldment materials. Information gained in this study will be used to specify welding materials and procedures to provide optimum compatibility

with the projected service conditions for the FTR vessel and core structural components.

Tensile testing of the as-fabricated Type 304 SS weldments has been completed to produce a testing matrix which includes four different welding processes (TIG, Submerged-Arc, Stick Electrode, and MIG), three specimen types (weld, fusion line, and base metal), and four testing temperatures. This matrix constitutes a base of tensile property information to be used in support of the weldment irradiation effects experiment.

After tensile testing, each specimen was macro-etched to distinguish weld metal from base metal. It was observed that, of the approximately 35 specimens containing base metal in the gage length, 75% fractured in the base metal. From a statistical view, it would appear that the base metal properties are limiting the test-to-failure behavior of the composite (weld, fusion line, HAZ, and base) weldment material. Of those facturing in weld metal, most had been tested at 900 °F or previously annealed at 1200 °F, thus suggesting the possible effect of some transformation of delta ferrite to sigma phase in the weld metal. Metallographic examinations are in progress to confirm to confirm the presence of sigma phase in those specimens exposed to temperatures in the range of 800 to 1600 °F.

A cooperative study with ORNL is continuing to characterize the effect of delta ferrite in weld metals and also to develop reliable methods of measuring ferrite contents. The results of independent ferrite determinations on the TIG weldment by the QTM (ORNL) method and point count analysis (BNW) are in sufficient agreement to be consistent with nationally published error bands.

A limited number of specimens are being tensile tested at 1100 °F to support the creep-rupture portion of this work. All four weld processes will be tested in addition to the base metal test. In order to facilitate comparisons with literature data, a few tensile tests are being conducted at room temperature.

The design of a new tensile specimen with a 0.250 in. gage diameter and 1.125 in. gage length has been completed. Weldment specimens will be fabricated to this design and tested to confirm the properties data obtained from the sub-sized specimens.

Creep-rupture tests on Submerged-Arc specimens at 1100 °F are in progress. The preliminary results from two short term tests indicate lower rupture lives by a factor of 15 when compared to Type 304 SS rod stock specimens in the 1950 °F solution annealed condition. Additional creep-rupture tests are in progress to evaluate this gross difference in rupture properties.

b. Subcritical Crack Growth

E. B. Schwenk, L. A. James

The objective of this program is to determine the slow crack extension behavior (i.e. subcritical crack growth) of the Fast Test Reactor Primary Piping system material. Such data will provide a basis for estimating the life expectancy of flaw-containing piping components for simulated environment and service conditions.

Initial work in this program centers on determining the fatigue crack growth behavior of Type 304 SS with respect to analysis by the crack propagation power law for temperatures and environments expected to be characteristic of piping system operation. This information will provide a basis for reasonable estimation of the remaining fatigue life in piping system components using fracture mechanics analysis techniques.

Tests now in progress are being conducted on 1/2 inch thick Type 304 SS characterized under the Irradiation Effects to Reactor Structural Materials Program.

Figure 4.F.1 shows a plot of average crack length, a , (average of measurements on both sides of specimen) versus cycles, N , to failure and Figure 4.F.2 is the corresponding ΔK versus da/dn curve. The data shown in Figure 4.F.2 have not yet been corrected for plasticity effects. Thus the magnitude of ΔK for crack growth rates greater than about 10^{-4} in./cycle probably does not reflect the actual crack tip stress state.

c. Radiation Effects on Absorbing Materials for Control Rods

A. L. Pitner

Profilometry tests were performed on the cladding of the boron carbide capsules BC-1 and BC-4 which were irradiated in EBR-II. Measurements were made on three indexed axes 120° apart, just as in the pre-irradiation measurement. No change in the cladding profile of either capsule was detectable. Following the profilometry, the capsules were each sectioned into five equal segments approximately 1.2 in. long. The compacted powder was removed with considerable effort from each section using a vibra-tool. The B_4C appeared to be sintered in the cladding, but once the sample was removed it again had the appearance of a fine powder. A sieve analysis will determine whether any sintering did occur.

The first assembly of the Hanford B_4C irradiation, designed to operate at sample temperatures of 1100 and 1300 $^\circ F$, increased in temperature so that it is now operating at 600 to 1200 $^\circ F$. The second assembly, designed for 700 and 900 $^\circ F$ operation, was running at temperatures from 500 to 700 $^\circ F$ just after startup was accomplished. Behavior of the first assembly, however, would indicate that these temperatures will increase

FIGURE 4.F.1.1. Average Crack Length Versus Load Cycles for 304 SS Specimen

Specimen No. :	RML-2	Stress Ratio :	~0
Temperature :	83°	Width :	4.91
Maximum Stress:	18,650 psi*	Specimen Type:	SEN
Thickness :	0.491	Frequency :	2 cpm
Material :	304 SS Annealed	Orientation :	RW
Environment :	Air	Date Tested :	5,6/69

* Dynamic Loads Calibration
Being Rechecked with Load Cells

FIGURE 4.F.2. Crack Growth Rate of 304 SS Plotted Versus Stress Intensity Factor

gradually with time. It is felt that the lower-than-design temperatures are probably resulting from the additional flux depression evident in the irradiation assemblies. Considerably more B_4C "poison" is present in the irradiation assemblies than was in the Snout-capsule heat transfer test performed last year.

2. Fabrication Development

R. N. Johnson

The objective of this program is to develop and evaluate fabrication processes required to produce components for the FTR. Development, evaluation, and qualification of commercial sources of supply is a program goal; and where components such as fuel ducts and test loops must be routinely replaced, the establishment of two or more competitive and continuing sources of supply is an additional objective.

a. Duct Welding Development

W. F. Brown

The automatic tube welding machine was inspected by BNW representatives at the Vendor's plant on May 22-23, 1969. The machine did not perform in accordance with the purchase specifications.

In order to determine the feasibility of making simultaneous spot welds on opposite faces of the hex duct, a series of simultaneous welds were made between pairs of unaged Inconel 718 and Type 304 SS with 4-in.-thick solid copper back-up and current conducting bar. Peel tests and tensile-shear tests were used for preliminary evaluation to establish a starting weld schedule. The results indicated that simultaneous welds can be made with the proper fixtures. For further tests, an adjustable copper back-up and current conducting fixture was made. This fixture will be inserted inside the hex tubes to conduct welding current to opposite faces and to prevent tube distortion.

b. Duct Gaging and Characterization

R. N. Johnson, R. H. Todd

A Request for Proposal for the duct gaging and characterization system was sent to twenty potential vendors on May 16, 1969. Nine of the 20 vendors so far have declined to submit proposals and one has replied that he will definitely submit a proposal. No replies have yet been received from the other 10 vendors.

c. Duct Procurement Activities

R. N. Johnson

A Request for Proposal for the fabrication of the prototype fuel ducts was sent to 12 potential vendors on May 26, 1969.

d. Closed Loops - Procurement Activities

S. M. Gill

Present plans call for the off-site manufacture of two prototypical closed loop tube assemblies, with delivery to BNW by July 1971 and installation in the Prototypical Closed Loop Test Installation by January 1972. Activity is currently underway on preparation of a Request for Proposal with the intent of contracting with a supplier for these assemblies within three to four months.

Preliminary effort has involved vendor evaluation to determine interest and to ensure adequate and competitive proposals. As a result of specific contacts and advertising, a list of eight probable qualified firms is available.

At the present time specifications and drawings for the RFP are being finalized and reviewed. The proposed design has been discussed with WARD personnel and their concurrence received.

3. Component Surveillance

J. W. Helm

a. Inservice Inspection Requirements

J. H. Cox

The report Surveillance and Inservice Inspection Requirements for the FFTF has been issued to the project for comments.

The purpose of this report is to recommend specific surveillance and inspection requirements, methods to perform examinations to meet the requirements, design requirements to permit the inspections, and the necessary development of the methods. The report is based on the ASME (N-45) Code for Inservice Inspection,⁽¹⁾ and on requirements studies by Westinghouse Advanced Reactors Division⁽²⁾ and Southwest Research Institute.⁽³⁾ In some cases, discussions from the latter two reports are used without modification.

b. FFTF Periscope

W. J. Coleman, J. D. Jensen

The conceptual design of an optical periscope suitable for use in inspection of the FFTF reactor vessel is continuing. The periscope assembly will consist of a head which contains the viewing prism and collecting optics, an objective section, an extendor section, and the eye piece section. Conceptual design of the periscope head has been completed.

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- (1) ASME, USA Standard Code for Inservice Inspection of Nuclear Reactor Coolant Systems, October 1968.
- (2) Buggy, J. J., D. L. and Paxson, E., Surveillance and Inservice Inspection Recommendations for the FTR: Techniques, Program Development, Westinghouse Electric Corp. Advance Reactors Division Report, WARD-3977-1, February 1969.
- (3) Wylie, R. D., et al, Criteria for Surveillance and Inspection of the Fast Flux Test Facility, Southwest Research Institute, Report SWRT 17-2552-01-A, May 1969.

V. FFTF FUELS DEVELOPMENT

E. A. Evans

A. FUEL ELEMENT DEVELOPMENT

C. A. Burgess

1. Subassembly Development Section

J. W. Thornton

a. Project Status

M. D. Jackson

(1) BCP-034 - Subassembly Assembly. BNW received a directive from AEC-RL during the week of May 18, to proceed with the construction phase of this project. The design work of the facility is 100% complete. The demolition work in Room 138 is 90% complete while the overall project is 8% complete.

The cleaning tanks to be installed under this project have been designed and are being fabricated in the Technical Shops.

(2) BAP-022 - High Bay Addition. This project was approved at AEC Headquarters in May. A directive was received by BNW from AEC-RL to proceed with the design phase of this project.

2. Process Development and Demonstration

R. E. Bardsley

a. Fabrication of 300 Fuel Pins for Criticality Studies

C. A. Strand, G. E. Sammis, H. T. Blair

Fabrication of the 300 FFTF type fuel pins for criticality studies is continuing in the FFTF fuel pilot line. A total of 98 fuel pins has been assembled. A total of 27 kg of mixed oxide powder has been processed and 14 kg of mixed oxide fuel pellets were debound and sintered. The O/M results obtained

from these sintering runs indicate that the O/M may be a function of the furnace load mass as well as the sintering temperature. An average of 4.1 wt% of the pressed green pellets is lost as volatiles in the debinding process. Another 0.9 wt% is lost as volatiles during sintering.

The pellet grinding, pellet gauging, and fuel pin loading hoods and their associated equipment became operational during the month.

The capability for sintering fuel pellets to size through control of die size and forming pressure was demonstrated. Two different process lots involved in this demonstration were debonded and sintered in separate runs. Debonding and sintering conditions were held constant for the two runs involved. Table 5.A.I shows some of the pertinent data concerning each of the batches. Figure 5.A.1 is a frequency distribution of the sintered pellet diameters measured at both ends of the pellets. The minor difference between batches A-6-S18 and B-2-S19 is probably because they originated as two separate processing lots.

TABLE 5.A.I. *Fuel Pellet Debonding and Sintering Run Data*

	<u>Batch A-6-S18</u>	<u>Batch B-2-S19</u>
Number of Pellets in Batch	2430	~1780
Number of Pellets in Sample	30	30
Die Size, in.	0.2402	0.2402
Forming Pressure, psi	30,000	30,000
Green Diameter, in.	0.2420	0.2421
Green Density, average (% of TD)	51.43	50.96
Sintering Run No.	18	19
Total Mass Sintered During This Run, (grams)	3917	3712
Average Diameter, large end, in.	0.19536	0.19425
Average Diameter, small end, in.	0.19432	0.19307
Average Taper, (total taper on pellet)	0.00101	0.00118
Maximum Diameter Measured at Large End, in.	0.1958	0.1946
Minimum Diameter Measured at Small End, in.	0.1938	0.1929
Average Sintering Density, (% of TD)	93.368	94.502
Maximum Density, (% of TD)	93.813	96.547
Minimum Density, (% of TD)	92.800	94.088

FIGURE 5.A.1. Frequency Distribution of Sintered Pellet Diameters

b. Process Development

M. J. Barr

(1) Fractional Factorial Experiment. Approximately 10 sintering runs remain to be completed. Current priorities permit one run per week with possibilities of two runs per week after July 1, 1969.

(2) PuO₂ Sinterability Tests. Additional oxalate precipitated and calcined plutonia (ARHCO Lot 28-10-31) was obtained for evaluation. Preliminary data indicate sintered densities similar to those previously obtained with ARHCO Lots 28-3-7, 28-9-8, and 28-9-26 were produced. Test data were obtained on as-received powder for ARHCO-calcined to 425 °C and PNL-calcined to 625 °C. Powders were pressed without binder or additional processing and were sintered in dry Ar-8% H₂ atmosphere to 1640 °C for 4 hours. All densities are given in percent theoretical. Data for Powder Lot 28-10-31 are as follows:

<u>Calcined Temperature, °C</u>	<u>Green Density, % TD</u>	<u>As-Sintered Density, % TD</u>	<u>O/M Ratio</u>	<u>Sintered Density After Oxidation, % TD</u>
425	50.47	77.59	1.83	79.15
625	57.15	77.78	1.83	79.41

The as-sintered density was calculated using a theoretical density of 11.46 g/cm³. When lattice parameters at the O/M ratio of 1.83 are considered, the actual theoretical density of the as-sintered pellets is approximately 82.5%. The discrepancy between the theoretical 82.5% and observed 79.3% (average after oxidation to an O/M of 2.00) is being investigated.

3. Special Products Fabrication Section

E. T. Weber

a. Assembly of Test Pins for Irradiation in EBR-II

R. M. Crawford

Forty-five fuel pins for PNL-7 were completed and are ready for final visual inspection and shipping. This number constitutes the total requirement of 37-pins for immediate assembly, plus eight spares for PNL-7.

The xenon tagging of the first 20 pins for PNL-7 was witnessed by a representative of ANL. All pins in this sub-assembly were xenon tagged.

Fifty percent of the fuel pellets for PNL-8 were characterized and are ready for loading. Fifty percent of the cladding tubes for PNL-8 were air gaged, bottom closure welds completed, radiographed, and made ready for loading.

b. Fuel Fabrication for Test Pins for Irradiation in EBR-II

W. E. Warden

Fuel fabrication for EBR-II subassembly PNL-6 is in process. This fuel consists of 30% enriched (^{235}U) in the EBR-II test composition of 75% UO_2 -25% PuO_2 . Five kg of PNL-6 mixed oxide fuel pellets have been sintered to the required density range ($93 \pm 2\%$ TD). Two additional kg of fuel pellets have been pressed and are in the sintering process.

Fuel fabrication for EBR-II subassembly PNL-17 is in process in Line No. 2 of the FFTF Fuel Process Demonstration Facility. This fuel consists of 65% ^{235}U enrichment in the EBR-II test composition of 75% UO_2 -25% PuO_2 . Seven kg of PNL-17 mixed oxide were processed through the powder preparation operations.

c. Fabrication of Fuel and Special Test Pins
for Irradiation Tests Under Task FP-3

W. E. Warden

Fuel pellet fabrication for one density range (94% TD) has been completed. This fuel is normal UO_2 and oxalate PuO_2 of composition 75% UO_2 -25% PuO_2 . Three kg of this fuel were sintered to the required density in producing 0.320-in. diam fuel pellets.

Seven of the eight fuel pins requested for 2-pin calibration test capsules were completed. Eight fuel pins for capsule FP-33 (Gap Closure Experiment) were assembled. The two fuel pins requiring a helium atmosphere were completed and the six fuel pins requiring various helium-xenon mixtures were processed to the xenon tagging operation.

B. FUELS QUALITY ASSURANCE

N. G. Powers

1. Development and Control of Inspection and Testing

a. Ultrasonic Cladding Tester

J. Ryden Jr., E. F. Perrizo, A. C. Callen

The laboratory ultrasonic cladding tester has been set up to inspect 0.230-in. OD tubing. On a recently-fabricated standard made of 20% cold worked TP316 cladding, EDM notches as small as 0.7 mil in depth were detected.

b. Fuel Pin Tubing Thickness Measurement

J. L. Thompson, F. M. Coffman, A. C. Callen

A Branson Vidigage was examined to determine whether it could resolve wall thickness of 0.230-in. OD by nominal 0.015-in. wall tubing to 0.0001-in. A special transducer and oscillator were procured, as recommended by Branson Instruments, and evaluation performed on a piece of tubing with a measured ring of wall varying from 0.01470 to 0.01505 in. No difficulty

was encountered in recording the thickness to 2×10^{-5} in. resolution, the approximate accuracy of the precision mechanical measurement of the tube wall thickness. Thermal drift of the Vidigage is noticeable over a long period (i.e., tens of minutes) but has not yet been accurately measured for determination of test statistics.

In the actual tubing test to be used early in the month of June, the tubing will be rotated at 1450 rpm. The Vidigage capability for measuring at a rate of 1800 per minute will provide thickness readings every 0.1 in. on the circumference of the tube. The resulting change of thickness occurring each 16 msec requires a circuit to hold each value for the maximum possible time to permit even a high speed recorder to indicate each value. Such a circuit has been built to do this and will be incorporated into the overall test.

c. Particle Size

M. W. Urie, W. L. Delvin

The cooperative work between Urie and Daniel, reported last month, was completed. The electron micrographs obtained by Daniel of the PuO_2 dispersions made by Urie showed reasonable correlation with the distribution curves obtained by Urie using the MSA particle size analyzer. The data show that the higher PuO_2 is fired and the longer the dispersion time, the finer the particles will be.

d. X-Ray Fluorescence

D. S. Skeie, W. L. Delvin

Initial results to determine the U/Pu in solid, mixed oxide were both encouraging and discouraging. Good straight-line calibration curves were obtained using standards prepared in the laboratory. Those curves were reproducible to within 5%. However, when a sintered pellet was crushed and measured, the U/Pu obtained was 5 to 1 instead of the actual 3 to 1

make-up value confirmed by laboratory analysis. Also, the unsintered, mixed oxide powder from which the pellet was made showed a 5 to 1 ratio. Plans call for assaying the standards by the laboratory to check their make-up.

e. Emission Spectrography

A. S. Wilson, K. M. Maloney, R. R. Rhodes,
R. Ko, W. L. Delvin

A UO_2 sample was analyzed by the emission spectrograph and by the spark source mass spectrograph. The following tabulation compares the results. Only those elements detected by one or both of the instruments are shown. The results are good to within a factor of 2 and 3 for the emission spectrograph and spark source, respectively.

Element	Emission Spec, ppm	Spark Source, ppm	Detection Limit ^(a) Emission Spec, ppm
Al	--- (b)	3	5
B	<0.1	---	0.1
Ca	<2	1	2
Cr	5	5	<5
Cu	0.5	---	0.2
Fe	100	50	50
Mg	100	30	<5
Mn	5	0.2	1
K	---	1	20
Na	---	4	2
Ni	<5	0.5	5
Sb	<10	---	10
Si	<10	---	10
V	<50	---	50

a. *Detection limits have not been established yet in a UO_2 matrix for spark source.*

b. *Not detected.*

An estimate of detection limits for six elements in UO_2 by spark source was made. Estimates of these elements, not determinable by emission spectrograph, are as follows:

<u>Element</u>	<u>Estimated Detection Limit, ppm</u>
W	2
Cl	2
S	10
F	1
N	1
C	1

No work has been done yet to check the validity of those estimates.

2. Quality Assurance Systems and Procedures

a. Follower and Data Card System

D. A. Stranik

A complete description of the card system and the cards has been written along with a flow diagram showing the flow of the data. The cards are at present being upgraded and finalized. The system description has been incorporated into the draft of the Fuels Quality Assurance Manual.

b. Fuels Quality Assurance Manual

J. D. Schaffer, D. A. Stranik, W. F. Stevenson

A completed draft of the Fuels Quality Assurance Manual was distributed for comment. This manual (in three volumes) adheres to the following outline:

FFTF Fuels Quality Program

Introduction

Scope

Objectives

Organization and Administration

Quality Organization and Interfaces

Supporting Documents/Procedures/Systems

FFTF Fuels Quality System Plan
 Fuel Pin Process Flow Diagram
 Fuel Pin Process Quality System

FFTF Fuels Quality Assurance Procedures (Book I)

Statistical Quality Program
 Calibration and Standards
 Qualification and Certification
 Design Reviews
 Control of Suppliers
 Inspection and Test
 Inspection and Test Methods (See Book II)
 Nonconforming Material
 Data Card System
 Handling, Packing, Storing, Shipping
 Audits

FFTF Fuels Quality Assurance Procedures (Book II)

Inspection and Test Methods
 Analytical Chemistry Methods
 Nondestructive Test Procedures
 Helium Leak Testing
 Sinterability Performance Test - UO_2
 Sinterability Performance Test - PuO_2
 Reference Standards

A program of comments collection, accuracy upgrading, and trial application is underway.

C. CLADDING DEVELOPMENT

T. T. Claudson

1. Cladding Development and Processing

J. C. Tverberg

a. Procurement Activities

R. J. Lobsinger

(1) Plenum Springs. Inconel X-750 plenum springs were ordered from an alternative vendor after the original vendor failed to produce acceptable springs. These springs will be used in EBR-II subassemblies PNL-9, 10, and 11, as well as the prequalification subassemblies.

(2) FFTF Fuel Cladding. Carpenter Technology Corporation, while producing tubing conforming to the ultrasonic inspection specification, has indicated a desire to produce fuel clad tubing far exceeding the specification requirements. To this end, the mandrels will be polished to an extremely fine surface finish and all cutting will be done with an electrolytic cut-off saw to avoid introducing metal chips into the system.

b. Vendor Development

J. C. Tverberg

Work has begun with three contractors on the Vendor Development Program. Tube drawing is well underway at Sandvik Steel, Incorporated. Wolverine Tube has placed an order for a special 4000-lb heat of 316 stainless steel with Allegheny Ludlum, delivery of which has been promised for August 15, 1969 at the latest. Carpenter Technology has made the basic vacuum induction heat of 8000-lb of 316 stainless steel and is awaiting analyses before making the final consumable arc remelt ingot.

c. Parameter Studies

R. C. Aungst, R. J. Lobsinger, R. R. Studer

General Electric, at Cincinnati, Ohio is on schedule in preparing a series of ingots to study the effect of variations in chemistry and melting practice on tubing quality. Four master heats of 400-lb each have been cast from which smaller test heats will be made by remelting and the chemistry adjusted.

The basic fabrication parameter study conducted by ORNL has yielded the instrumented extrusion of six billets into tube hollows. The next portion of the work will include melting of ingots in which the nitrogen and carbon will be varied within specified limits and the instrumented reduction of tube hollows on the special tube reducer and draw bench will be provided.

Initial results with the nondestructive cold work-grain size device were very encouraging. Specimens with known cold work and known grain size were distinguishable. Work is now starting using tube specimens with known levels of cold work.

Use of X-ray diffraction to distinguish different levels of cold work also have been quite successful. It is possible to obtain a better degree of resolution on varying degrees of cold work than earlier thought. Work is now underway to determine whether a texture difference exists between tube reduced and drawn material.

2. Cladding Evaluation

J. J. Holmes

a. Biaxial Stress-Rupture Studies

R. W. Barker

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 alloys.

Biaxial stress-rupture tests on AISI Type 316 SS irradiated in EBR-II to 0.18×10^{22} n/cm² ($E > 0.18$ MeV) were initiated during the May, 1969 reporting period. Tests at short rupture times and 1200 °F were completed and data reported. The rupture strain for one test was omitted due to the timing of the test failure with respect to the reporting period. The measured strain for this specimen (test conditions were 30,000 psi hoop stress and 1200 °F) was 2.4% as shown in Table 5.C.I.

The irradiation temperature for the biaxial stress-rupture specimens irradiated in EBR-II subassembly X-022, previously estimated to be about 1000 °F based on X-rays of fusible metal alloys in stainless steel capsules after irradiation, has been re-examined. On the basis of Oak Ridge National Laboratory gamma heat experiments in EBR-II, Row 7, the irradiation temperature is calculated to be 815 °F ±20.

TABLE 5.C.I. *Biaxial Stress-Rupture Tests on AISI Type 316 SS Irradiated to $\sim 0.8 \times 10^{22}$ n/cm² (E > 0.1 MeV) in EBR-II. Test Temperature was 1200 °F. The Irradiation Temperature was 815 °F $\pm$ 80.*

Unirradiated Data ^(a)			Irradiated Test No.	Hoop Stress, psi	Rupture Time, hr	Rupture Strain, %
Hoop Stress, psi	Rupture Time, hr	Rupture Strain, %				
42,000	0.1	15				
36,000	1.0	10	12	36,000	2.1	1.9
30,000	10.0	7	13	30,000	32	2.4
24,500	100.0	5				

a. *Stress and strain values were taken from unirradiated test curves at rupture times shown.*

The test rigs are developing leaks through repeated use, and operation has become difficult. Especially difficult has been maintenance of high pressures for long periods of time. Similar problems are also common at other laboratories.

In order to alleviate the problem of gas leakage, a new group of test rigs is being constructed. The rig design will be modified slightly by utilizing Aminco high pressure fittings wherever possible, rather than the Swagelok brand fitting, to obtain greater reliability.

Anomalies in test specimen behavior were observed in 316 specimens at a test temperature of 1400 °F. Three short term biaxial stress tests were initiated on irradiated material at this temperature and all three failed at the lower plug cladding-weld interface. It is not completely understood whether this phenomena is a function of the postirradiation condition or the welding process. In either case, specimens are being mounted with mechanical seals only. Biaxial stress-rupture testing using the mechanically sealed specimens is continuing.

b. Damage Analysis

J. L. Straalsund and H. R. Brager

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 using both microscopic modeling and empirical approaches.

Density measurements on sections from an EBR-II safety rod thimble were reported with the corresponding exposures in a previous monthly report. Further consideration of the data available indicates that the specimens were at a different position with respect to the mid-plane than had been reported. The corrected data are given in Table 5.C.II.

TABLE 5.C.II. *Irradiation Induced Swelling in EBR-II Safety Rod Thimble*

Distance from Mid-Plane, in.	Fluence $\times 10^{-22}$ n/cm ² , E > 0.1 MeV	Temperature, °F	Swelling, $\Delta\rho/\rho$, %
20.8	0.36	872	0.14
15.0	1.28	852	0.41
10.0	2.62	826	0.93
8.0	3.46	810	1.55
5.5	4.80	786	2.44
3.0	5.89	756	3.09
0.5	6.40	732	2.41
-2.0	6.14	718	1.53
-4.0	5.57	710	0.87
-9.0	2.62	700	0.47
-14.8	1.34	700	0.27
-19.8	0.601	700	0.11

In the previous monthly report, a least mean square fit was made to selected swelling data for annealed 304 stainless steel. The equation used in developing a swelling equation for design guidance was based partly on the safety rod thimble data, and the analysis was repeated using the corrected data. Although the revised equation indicates slightly more fluence and temperature dependence, the differences are quite small in comparison with the uncertainties involved and change of the interim design equation reported last month does not presently appear necessary.

The classical theories of homogeneous nucleation have been reviewed and applied to the formation of voids in the presence of an irradiation produced vacancy supersaturation to compare predicted with experimentally observed trends.

The variation in nucleation rate with flux predicted with classical nucleation theory is as follows:

$$I = A'(\phi)^{3-6}$$

where

I = nucleation rate

A = constant

ϕ = flux

The observed void production rate, however, varies in an approximately linear fashion with flux. Thus, the classical nucleation theory appears to predict a much higher flux dependency for the nucleation rate than is observed where typical experimental values for the pertinent parameters are used.

c. Mechanical Testing of Fuel Pin Cladding

R. L. Fish

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 the cladding. Specimens cut from irradiated PNL fueled subassembly pins are mechanically sealed and pressurized in a totally enclosed test vessel in the hot cell facilities of the 324 Building.

Three specimens have been successfully burst tested thus far. The history and test results for these specimens are tabulated in Table 5.C.III. Additional burst tests on specimens taken from the PNL-1 subassembly are planned for temperatures of 1000 and 1200 °F. These tests will be used for determining subsequent stress-rupture test pressure values for the other PNL-1 subassembly specimens.

TABLE 5.C.III. *Specimen History and Burst Test Results*

<u>Specimen</u>	<u>Fluence E > 0.1 MeV</u>	<u>Operating Temp, °F</u>	<u>Test Temp, °F</u>	<u>Hoop Stress at Inside Diam, psi</u>	<u>$\frac{\Delta D}{D}$, %</u>
Unirradiated Lot "E" Cladding	---	---	900	57,700 psi	19.2
PNL 1-6 Plenum Region	$\sim 1.2 \times 10^{21}$	~ 890	900	59,800 psi	18.4
PNL 1-6 Fueled Specimen	$\sim 3.2 \times 10^{21}$	840-850	899	73,100 psi	9.6

D. FUELS EVALUATION

J. E. Hanson

1. Task FP-2

a. Thermal Irradiation of $\text{UO}_2 + \text{PuO}_2$ Fuel Pins

G. E. Culley, E. O. Ballard, S. H. Christensen
and N. D. Mills

A section was cut from the fuel pin at the position of the lower axial fuel motion restrictor, 10.3 in. from the bottom of the fuel column. In addition to the fuel restrictor interface, this section shows what appeared on the neutron radiographs as a pellet separation that had been bridged with fuel. The fuel in this region operated at approximately 12.7 kW/ft and shows a well formed central void, columnar grains, and apparently a lesser degree of cracking than the lower-powered sections. The perturbation in the heat transfer path caused by the presence of the floating type axial fuel motion restrictor is readily apparent. It is also apparent that the pellet separation has been bridged by fuel. The bridge occurs only in the grain growth region and, by the shape of the central void, the mass transfer to provide the bridge material was from both the upper and lower pellets. The beta-gamma autoradiograph, showing the typical outline of the grain-growth region, indicates an accumulation of fission products in this region.

The next section taken was the peak power (13.7 kW/ft) transverse section. It is characterized by a slightly off center central void, columnar grain growth, and fuel in contact with the cladding. The fuel is not as badly cracked as the lower power sections, although some fallout did occur in this section during preparation. The beta-gamma autoradiograph also shows the ring outlining the grain growth region.

The last section to be taken was cut at the upper central void terminus. The micrograph shows the terminus of the central void, although the grinding of the sample was slightly skewed and some additional central void is evidenced by the beta-gamma autoradiograph. Aside from the normal grain structure and central void, the most predominant feature of this sample is the presence of a rather large pellet separation through which the cladding can be seen. This separation did not appear on the neutron radiographs. However, the gamma scan taken of the capsule and the one taken of the fuel pin gave conflicting indications of pellet separations in this area. Judging from the appearance of the cut ends, it appears that the bottom end of the fuel had moved with respect to the cladding after the cutting operation. This is significant in that, although the fuel-cladding gap appears to have been completely closed, the fuel pellets did move.

When all the photomicrographs are available, more detailed analyses will be made to correlate predictions and observations.

(2) Transient Irradiations. Permission to perform the TREAT calibration test has been received from ANL-TREAT. The calibration capsule was inserted into a TREAT capsule and preirradiation testing performed. The capsule is presently awaiting shipment to TREAT. Arrangements have been made to perform radiochemical burnup analysis on the fuel after irradiation in TREAT.

2. Task FP-6

a. Thermal Flux Irradiation of Hypostoichiometric UO_2 - PuO_2

R. D. Leggett, L. A. Pember, and J. W. Weber

Pins BNW-1-1, 2, 3 and 5 were recovered from their respective capsules in the BNW hot cells, cleaned in Dowanol

and water, visually examined and photographed. The four pins appear to be in excellent condition with no indication of a breach in the cladding. Negative results of sampling the inner capsules for fission gas supports the indication that all pins are intact. All pins were tightly bound inside this inner capsule. Thus, it was necessary to slit the inner capsules longitudinally in order to remove the pins without damaging them. The pins are now measured with a profilometer. Preliminary results from BNW-1-2 indicate that a maximum average diameter increase of about 0.008 in. occurred on the bottom 2 in. of the fuel column. This increase corresponds to about 3% ΔD . Some ovality was also noted. Following this, the pins will be gamma scanned, punctured for plenum gas collection and analysis, and sectioned for ceramography.

b. Electron Microprobe Examination

R. D. Leggett and E. D. Jenson

A section from pin BNW-1-8 (failed) is being studied to determine the location of fission products, the concentration profiles of uranium and plutonium, and the extent and nature of fuel-cladding interaction. Preliminary results indicate a very large depletion (factor of 10 to 20) of plutonium and a buildup of uranium (approximately 20%) at the fuel surface.

The plutonium depletion is thought to be largely the result of burnup effects in the thermal flux, while the uranium buildup is attributed to migration. Very few fission products appeared to be present in this section.

3. Task FP-7

a. Fuel Pin Irradiations in EBR-II

L. A. Pember, J. W. Weber, E. O. Ballard, and
S. H. Christensen

Five PNL-X pins are undergoing postirradiation examination in the LASL hot cells. Two of these pins PNL-X-6 and X-7,

operated at an estimated 10.6 kW/ft to a burnup of 8500 $\frac{\text{MWD}}{\text{MTM}}$, while the other 3 PNL-X-1, X-3, and X-4, operated at an estimated 6 kW/ft to 4800 $\frac{\text{MWD}}{\text{MTM}}$.

On opening, PNL-X-1 showed some peculiar markings in the cladding in the plenum region which, although not completely explained, appear to have had no influence on pin performance. The markings consisted of a network of dark lines appearing reddish-brown when viewed in air at low magnification. A qualitative analysis of some of the fine particulate brushed from these markings showed Fe, Cr, and Ni as the major constituents. The marks on the cladding are probably sites of preferential oxidation at light scratches placed on the surface of the pin during fabrication. Analysis of the Na from the PNL-X-1 and X-6 capsules showed slightly higher concentration of Fe, Ni, and Cr in X-1. PNL-X-6 exhibited none of the reddish-brown markings.

Examination of the PNL-X pins through profilometry and collection and analysis of the internal pin gas has been completed at LASL. Results from analysis of the profilometer data are shown in Table 5.D.I. Comparison of the maximum diameter in the fuel section with the average diameter through the gas plenum indicates a maximum diameter increase of 0.0005 mil on the higher power, higher exposure fuel pins. Ovality of the cladding OD was limited to 0.0004 in. or less.

Results from the plenum gas measurements and analysis are shown in Tables 5.D.II and 5.D.III. The volume of fission gas formed and the percent released is based on the estimated burnups. A comparison of the PNL-X-6 and X-7 percent gas release with results previously obtained from the PNL-1-19 fuel pins with similar design parameters, and irradiated under similar conditions, indicates the exposure estimate to be reasonable.

TABLE 5.D.I. PNL-X Cladding Diameters

<u>Pin Number</u>	<u>Maximum</u> <u>Cladding OD</u> ⁽¹⁾	<u>Maximum</u> <u>Diameter Increase</u> <u>Compared to Plenum</u>	
	<u>in.</u>	<u>in.</u>	<u>%</u>
PNL-X-1	0.2502	+0.0002	< 0.1
X-3	0.2500	+0.0003	0.1
X-4	0.2502	+0.0003	0.1
X-6	0.2503	+0.0003	0.15
X-7	0.2504	+0.0005	0.2

(1) Maximum from four diameter measurements around circumference at 0°, 45°, 90°, and 135° made at 1-in. increments along longitudinal axis.

TABLE 5.D.II. Gas Measurements on PNL-X Pins

<u>Pin Number</u>	<u>Total Fission</u> <u>Gas, cm³ @ STP</u>	<u>Est. Burnup,</u> <u>MWD/MTM</u>	<u>Fission Gas</u> <u>Formed,</u>	<u>Fission Gas</u> <u>Release, %</u>
			<u>cm³ @ STP</u>	
PNL-X-1	0.0272	4800	8.9	0.31
PNL-X-3	0.0251	4800	8.9	0.28
PNL-X-4	0.01185	4800	8.9	0.13
PNL-X-6	3.592	8500	15.68	22.91
PNL-X-7	3.386	8500	15.81	21.42

TABLE 5.D.III. Gas Analysis on PNL-X Pins

<u>Pin Number</u>	<u>% Xe</u>	<u>% Kr</u>	<u>% He</u>	<u>% N₂</u>	<u>% O₂</u>	<u>% Ar</u>
PNL-X-1	0.327	0.033	99.5	--	--	0.10
PNL-X-3	0.307	0.029	99.6	--	--	0.09
PNL-X-4	0.146	0.011	99.6	0.03	0.01	0.20
PNL-X-6	28.32	3.61	67.6	--	--	0.50
PNL-X-7	27.04	3.50	68.9	--	--	0.50

b. Electron Microprobe Examination of Irradiated
Fuels

E. D. Jenson

Fuel pin sample PNL-1-6F was examined in the shielded electron microprobe for variation in plutonium and uranium across the pin and identification of metallic-appearing ingots. This pin, irradiated to 8140 MWd/T with PNL-1-3, has exposure characteristics similar to those reported last month for PNL-1-3.

All ingots found near the central void of PNL-1-6 showed the presence of the fission product molybdenum. None of those examined showed the presence of iron. In contrast, all particles in the outer third of the pin showed the presence of iron, but no molybdenum. Of four particles located about halfway between central void and cladding, two showed the presence of iron and molybdenum, as well as an element tentatively identified as phosphorus and a probable trace of tungsten, but no technetium nor ruthenium.

c. EBR-II Instrumented Subassembly (PNL-17)

M. K. Millhollen

Components for the fuel pins are now being fabricated from the approved drawings under the direction of the Special Products Fabrication Section. Since it is not possible to ascertain when EBR-II will be raised to the 62.5 MW power level, the decision has been made to fabricate fuel pellets in which the uranium enrichment is reduced from 93 to 65%. If the reactor power is raised before the test is installed, the maximum linear heat generation rate will not exceed 12 kW/ft. If the power is not raised, the pin power generation will be approximately 25% less.

d. Transient Fuel Pin Analysis

F. E. Bard and D. S. Dutt

As was stated in the month of February,⁽¹⁾ the codes PECT-I and INEQ did not provide a dilatation creep and plasticity model. During the past four months, the dilatation element has been formulated and programmed. The code INEQ was then used to analyze a TREAT experiment. The results of the computer run showed INEQ to require too much computing time to be of any use. The code INEQ consists of 269 first order simultaneous differential equations with respect to time. Also for each time point, a matrix equation consisting of 22 simultaneous algebraic variables must be solved.

e. Dosimetry and Damage Analysis for Fuels

R. E. Dahl, Jr. and R. D. Bourquin

Development of a computer method suitable for use in selecting neutron spectra and flux at any radial or axial position in the EBR-II from calculated or measured data has been completed. This effort will enable rapid evaluation of fuel element behavior for any position in EBR-II within the accuracy of the spectral information (from reactor analyses and/or experimental measurements). For example, behavior of the highest average and the lowest power pins in a subassembly or the peak-to-average values for any pin can be calculated.

Documentation of the method of analysis used to relate cladding damage and fuel element burnup during the course of extended irradiations in fast reactors has been partially completed. This documentation includes a description of the methods and use of BURN BABY BURN.

(1) *FFTF Monthly Report, BNWL-1006, February 1969.*

4. Task CFD-3

a. CCTL Flow Test Examination

M. K. Millhollen

During the removal of the 217 pin flow test element from the CCTL, a steel cable previously exposed to sodium vapor during the test, was found to have a much lower breaking strength than was anticipated. Samples of the exposed cable and of new, unused cable were prepared metallographically and studied to determine the reason for the low strength.

There was also an opportunity to examine samples of 201 SS (1) unexposed to sodium, (2) submerged in liquid sodium, and (3) exposed to sodium vapor. The liquid sodium was at a temperature of 1060 °F for at least 3600 hr and at lower temperatures for an additional 1000 hr. The sodium vapor was that in the test vessel above the liquid level. Evidence of some limited surface attack was disclosed by a 500X magnification of the sample immersed in liquid sodium. Evidence of an accelerated surface attack in the sample exposed to sodium vapor was very obvious, and indicates that 201 SS cannot be regarded as a satisfactory material for extended periods of use in sodium atmospheres.

E. OFFSITE FUEL PROGRAMS

G. A. Last

LMFBR Fuel Development

B. R. Hayward and F. M. Smith

Additional work scope was added to the NUMEC and UNC process development contracts to determine the weight per unit length of the FTR fuel pellets and fuel columns. The purpose of this work is to evaluate alternate methods of specifying the fuel loading characteristics required in the FTR fuel pin design.

The design of two NUMEC 61-pin subassemblies for test in EBR-II will be based on 85% smear density. BNW has recommended the use of 68% enriched UO_2 in these tests to minimize the handling of these subassemblies at the time EBR-II goes to 62.5 MW operating level. Additional fuel pin hardware and certifications were forwarded to NUMEC, including fuel tubes and 0.250-in. diam end cap stock.

The total effort on Phase II of the Analytical Chemistry Program was about 70% completed by the first week of June. Four sites, AI, B&W, BNW, and UNC will have completed their analytical work before June 30. Two sites, GE and NUMEC, may have to carry the program beyond June 30 because of equipment difficulties. Late arrival and necessary checkout of the C.P. coulometer will delay the uranium and plutonium assays from GE. NUMEC is having difficulty in getting their gas evolution apparatus operable.

Kerr-McGee was offered an opportunity to join the Analytical Chemistry Program under a no-cost-to-FFTF contract. All data will be evaluated on the same basis as the other participants.

Burnout of the sintering furnace in the plutonium process line has compelled AI to abandon their efforts to complete the Task C-2 Scrap Recycle program. BNW has given AI permission to retain about 3 kg of the pellets and powdered material from the scrap recycle task for use in making some reactor physics measurements on a non-FFTF project. Any costs associated with returning this retained material after June 30 will be at no cost to BNW.

General Electric's proposal to extend Task F-2 work to improve homogeneity determination methods through alpha-autoradiography and neutron radioagraphy has been accepted. The contract has been signed and the work started.

Extension of the Analytical Chemistry Program, Task G-2, to source or raw materials required for FFTF fuel fabrication is being initiated. The formulation of detailed plans and scheduling of discussions with Dr. C. F. Metz of LASL and other interested personnel in BNW, AEC and ARHCO to lay out an optimum course of action are under way. Shipper-receiver differences in analysis of nuclear raw materials used for FFTF fuel fabrication, along with product quality and types of products have pointed out the need for standardized analyses, sampling procedures, and specifications. Thus far, analytical methods to define the quality of the raw materials are somewhat uncertain. The recent work done in Phases I and II of the Analytical Chemistry Program will be extended to raw materials to ensure comparable standardization of methods and evaluation of precision and accuracy of the methods.

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