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FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
SEPTEMBER, OCTOBER, NOVEMBER 1969

AEC RESEARCH &
DEVELOPMENT REPORT

TABLE NO.	DATE	REV. NO.	REV. DATE

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UC-80, Reactor
Technology

FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
SEPTEMBER, OCTOBER, NOVEMBER 1969

By

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Project Manager

Compiled by C. P. Cabell

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ABSTRACT

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

PRIOR REPORTS IN THIS SERIES

BNWL-501	FFTF Quarterly Technical Report	1-1-67 to 3-31-67
BNWL-541	FFTF Quarterly Technical Report	4-1-67 to 6-30-67
BNWL-567	FFTF Periodic Technical Report	7-1-67 to 12-31-67
BNWL-660	FFTF Quarterly Technical Report	1-1-68 to 3-31-68
BNWL-880	FFTF Quarterly Technical Report	4-1-68 to 6-30-68
BNWL-917	FFTF Quarterly Technical Report	7-1-68 to 9-30-68
BNWL-941	FFTF Periodic Technical Report	10-1-68 to 2-28-69
BNWL-1090	FFTF Quarterly Technical Report	3-1-69 to 5-31-69
BNWL-1174	FFTF Quarterly Technical Report	6-1-69 to 8-31-69

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INTRODUCTION

Battelle-Northwest, as FFTF Project Manager, is discharging dual responsibilities. One major responsibility is management of the end-product oriented design and construction program to ensure conformance to technical requirements and to established costs and schedules. Progress in this area is reported monthly through the FFTF Monthly Informal Technical Report.

The second major responsibility discharged by BNW is management of the discipline-oriented technological program required for ensuring technical adequacy of the FFTF design. Progress in this highly technical area is the subject of this and other FFTF Quarterly Technical Reports.

An overall summary of progress is given in Chapter I. Progress in discipline-oriented technological work is presented in Chapters II through X. These chapters generally correspond to the LMFBR Program Plan Elements. Subdivisions of the chapters, however, are based upon specific FFTF components, systems, facilities, and programs.

Within Chapters II through X reports on specific topics show in their titles the code of the responsible FFTF organization. These codes are listed in Appendix A.

CHAPTER I. PROGRAM TECHNOLOGY HIGHLIGHTS

- The FFTF fuel assembly duct material has been changed to Type 316 SS, 20% cold-worked. This change was made to exploit the resistance of the material to irradiation induced swelling.
- Tests are in progress to estimate the potential for caustic stress corrosion cracking in the 304 SS FFTF reactor guard vessel.
- A transient model simulating the most recent reactor core design and primary loop configuration has been developed and debugged for steady state conditions using the digital computer code DYNASAR. An identical model is near completion on the FFTF Hybrid computer.
- Radiation levels in heat transport cells are projected as less than 100 mrem/hr. This level will permit limited access for inspection of maintenance.
- An evaluation was completed on the xenon gas tagging concept for location of failed fuel in the FTR. It was concluded that xenon tagging is a feasible concept for locating failed fuel in the FTR. It is simple in design and operation and requires very little in-vessel equipment.
- An alpha-particle counting system, presently in use on the Hanford Atomic Project, has been chosen to monitor airborne plutonium collected on filters. This system discriminates plutonium in the presence of ^{220}Rn and ^{222}Rn by using a coincidence counting circuit.
- A method has been developed for quantitative analysis of fuel pellet homogeneity through electron microprobe scanning. Availability of the new analysis method has in turn led to the development of an improved homogeneity specification for fuel pellets. Typical mixed oxide pellets fabricated by the BNW process readily meet the new homogeneity specification.

- A description is given of critical approach and exponential experiments to measure material bucklings, extrapolation distances, and critical masses as a function of lattice spacing for 25 wt% PuO_2 - UO_2 fuel pins in water.
- The results of a neutronics study of the adequacy of peripheral control rod worth measurements in ZPR-III Assembly 56B are discussed. The initial measurements were affected by spatial flux distortion.

CHAPTER II PLANT DESIGNA. OVERALL PLANT DESIGN

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

TABLE 2.I. Baseline Design Data

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

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

TABLE 2.I. (cont'd)

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

B. FFTF SITE, STRUCTURES, CONTAINMENT, AND UTILITIES

1. FFTF Site Investigation

F. H. Shadel (941)

a. Seismic Survey

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

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

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

CHAPTER III COMPONENTSA. REACTOR COMPONENTS1. Fuel Ductsa. Duct Welding Development

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

The conceptual design for the automatic welding system for the assembly welding of fuel duct components has been completed and the detail design is approximately 75% complete. Hardware fabrication on fixtures has been started.

Gas tungsten-arc welding and resistance welding procedures were established so that Inconel 718 wear pads could be attached to Type 304 SS hexagonal duct sections for testing and evaluation. The resistance welding studies indicated that cracking usually occurred in the nugget areas of conventional spot welds. Solid state bonding using spot welding techniques produced welds without defects. Tensile-shear values of over 5,000 lb per spot were obtained by using these techniques.

An aged Inconel 718 wear pad was attached to a 12-in.-long Type 304 SS hexagonal tube. The wear pad was 0.060 in. thick and 4 in. wide. The hexagonal tube had a wall thickness of 0.143 in. and a nominal ID across flats of 4.334 in. The wear pad consisted of two half hexagonals assembled around the center of the stainless tube.

The pads were attached by solid state spot welding four spots on each flat, followed by manual gas-tungsten arc welding around the edges as a seal weld. An internal, expanding copper back-up and current carrying fixture were used so that two spot welds could be made simultaneously on opposite faces. The two longitudinal butt joints and the two circumferential seal welds were made with Hastelloy Alloy W. Welding conditions are outlined in Tables 3.I and 3.II.

TABLE 3.I. Solid State Spot Welding Conditions
for Wear Pad Attachment on Tube 61
Spot Welding

Electrodes	- Class III Copper, 10 in. Radius on Electrode Face
Peak Welding Current	- 20,500 A
Welding Force	- 4,500 lb
Forging Force	- 6,000 lb
Heat Cycles	- 6
Cool Time Cycles	- 1.5
Impulses	- 6
Type Welding Machine	- Three Phase with Frequency Conversion
Back and Current Conducting Fixture	- Class III Copper

TABLE 3.II. Manual Gas Tungsten Arc Welding
Conditions for Wear Pad Attachment
on Tube 61

1. Butt Joint

Joint Design	- Single Vee butt joint with 70 deg included angle, 0.020 in. root face, 1/32 in. root opening
Filler Wire and Size	- Hastelloy Alloy W, 1/16 in. diam
Current	- 60 A
Electrode	- 3/32 in. diam, 2% thoriated tungsten
Shielding Gas	- Argon, flow rate of 20 cfh.

2. Seal Weld

Joint Design	- Lap joint with fillet weld, 0.060 in. wear pad sheet was beveled 15 deg to provide an edge 0.030 - 0.035 in.
Filler Wire and Size	- Hastelloy Alloy W, 0.045 in. diam
Current	- 50 A
Shielding Gas	- Argon, Flow rate of 15 cfh.

Radiography and dye penetrant examinations of the welds and wear pads revealed no indications of cracks or failures in the weld or base metals.

The test assembly was subjected to three series of thermal cycling tests performed before the wear pads had been ground to final size. The conditions for these tests are outlined in Table 3.III. In each series, a thermocouple was attached to the wear pad, and the heating and cooling rates were recorded. Two adjacent furnaces were used for these tests. When the tube was stabilized at the higher temperature, it was removed and cooled in air until it reached the lower temperature. After each series of thermal cycling tests, the welds and wear pads were visually examined at 10X magnification and dye penetrant inspected. No indications of any cracking or failure were found. Ultrasonic inspection showed that the spot welds had remained intact through all tests.

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

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

TABLE 3.III. Wear Pad Thermal Cycling Tests on Tube 61

<u>Test</u>	<u>Number of Cycles</u>	<u>Temperature Range, °F</u>	<u>Rate of Heating, Deg/min.</u>	<u>Rate of Cooling Deg/min.</u>
1	5	600-900	38	55
2	10	500-1000	26	55
3	10	400-1100	47	47

b. Duct Procurement Activities

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

Contract negotiations have continued with the two vendors selected on the basis of their proposals for the fabrication of prototype fuel ducts. Substantial progress has been made in achieving cost reductions over the original estimates. Costs and schedules were established based upon fabricating the ducts from Type 304 SS in the solution-treated condition. The choice of material and condition has been re-evaluated in the light of the latest stainless steel swelling information available; on this basis the choice of the duct material has been changed to Type 316 SS in the 20% cold-worked condition. The change may involve an increase in cost and schedule.

Because of the delays experienced in completing contract negotiations, material (approximately 40,000 lb of vacuum-melted 316 SS) required for the development of the fuel ducts has been placed on order by BNW. There is a considerable lead time involved in the procurement of any stainless steel under present economic conditions, and it is hoped that the duct development program can be held to the proposed time schedule by this action. The steel will be furnished to the development vendors in the form of machined billets of the proper size for their individual programs.

c. Material Selection

E. G. Stevens (913)

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

Potential benefits are as follows:

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

2. Reactor Guard Vessel

a. Guard Vessel Material Candidates

D. P. O'Keefe (961)

An evaluation on the basis of serviceability and cost led to the conclusion that the reactor guard vessel should be fabricated from ASME SA 387 B-E alloys. The significant requirements for materials in this application are stability at 900 °F, retention of mechanical properties after 10^{20} nvt, compatibility with nitrogen, and low cost.

The ASME Code was interpreted to prefer the following alloys for 900 °F service:

SA 387 B	1 Cr — 1/2 Mo
SA 387 C	1 1/4 Cr — 1/2 Mo — Si
SA 387 D	2 1/4 Cr — 1 Mo
SA 387 E	3 Cr — 1 Mo
SA 357	5 Cr — 1/2 Mo
SA 240	Austenitic stainless steel

For all of the above alloys, the potential for nitriding exists in the nitrogen environment of the guard vessel. However, it is not expected to be a significant problem and is not a basis for favoring any one alloy. The material and construction costs for the 5 Cr and stainless steel alloys is significantly higher than for SA 387. The SA 387 alloys will undergo an increase in nil ductility transition temperature with a fluence of 10^{20} , but existing data indicate the NDT will remain below 250 °F--the lowest operating temperature at which the guard vessel will receive the DBE impulse load.

It was concluded that the ferritic alloy steels bearing the ASME Code designation SA 387 B-E meet the service requirements that have so far been identified for the guard vessel and permit significantly lower construction cost than stainless steel.

3. Reactivity Control Driver

J. C. Noakes (914)

Westinghouse concurred with BNW's recommendation to utilize the Roller Nut Driver Concept for nuclear control of the FFTF. To provide early input to preliminary design, two experiments have been approved that use a roller nut driver from the N. S. Savannah replacement control rod drive program. One drive (and associated equipment) was shipped to LMEC for testing over liquid sodium to evaluate the drive performance. Receiving inspection has been completed, and testing is scheduled to start early next quarter. A second drive will be tested in an argon atmosphere at Todd Shipyards, Galveston, Texas, to determine the wear rate of critical components. The contract has been approved, and testing is scheduled to start early next quarter.

B. HEAT TRANSPORT COMPONENTS

1. Reactor and Heat Transport System Analysis

C. D. Flowers (822)

A transient model simulating the most recent reactor core design and primary loop configuration has been developed and debugged for steady state conditions using the digital computer code DYNASAR. The model includes the vertical reactor core and two independent primary loop heat transport systems with hot leg primary pumps. One of the two simulated loops represents one HTS loop and the other simulated loop represents two HTS loops for a total of three loops.

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

A sketch of the components simulated is shown in Figure 3.1 and a more detailed list follows.

Items Included in Primary Loop

Pipe Rupture Simulation

(Note: Model also calculates nonrupture transients)

I. Reactor Model

- A. Power calculation -- sum of neutron power and fission product power
- B. Inlet and outlet mixing plena
- C. Three parallel flow paths simulated
 - 1. Average driver channel

NOTE:

W_{RA} = DRIVER FUEL FLOW CHANNEL

W_{RB} = RADIAL REFLECTOR FLOW CHANNEL

W_{RC} = BY-PASS FLOW CHANNEL

FIGURE 3.1. Reactor and Primary Loops Configuration-Single Inlet Line with Rupture
(NOTE: Before the Break $W_{P1} = W'_{P1} = W''_{P1}$)

- 2. Average radial reflector channel
 - 3. Bypass channel
 - D. Fuel, clad, and coolant temperatures calculated in core zone with three radial and four axial mesh points taken in the fuel region
 - E. Reactivity feedbacks
 - 1. Doppler feedback - based on volume averaged fuel temperature
 - 2. Sodium density feedback - spatial weighting based on estimated sodium worth curve for average channel
 - F. Hot channel core outlet and maximum clad temperatures are calculated based on most recent hot channel factors
- II. Primary Loop and Reactor Hydraulics
- A. Calculates loop and reactor flows between the following known pressures for nonrupture transients and before the break for rupture transients
 - 1. Reactor vessel exit pressure - determined by vessel level which is calculated in the program
 - 2. Pump barrel pressures (both loops) - determined by pump barrel levels which are calculated in the program
 - B. Calculates loop and reactor flows between following known pressures after the break for rupture transients
 - 1. Reactor vessel exit pressure
 - 2. Pump barrel pressures
 - 3. Break pressure - pressure at break determined by assuming orifice loss to known cell pressure
 - C. Losses include
 - 1. Friction losses
 - 2. Fitting losses (valves, bends, elbows, etc.)
 - 3. Static losses
- III. Intermediate Heat Exchanger, IHX
- A. Primary and secondary inlet out outlet plena

- B. Five axial mesh points simulated to calculate primary and secondary sodium temperatures
- C. Tube wall metal lumped half-and-half with primary and secondary sodium
- D. Heat transfer coefficient and properties based on average primary and secondary side temperatures
- E. Primary and secondary isothermal bypass included
- F. Secondary flows and cold leg temperatures are forced inputs

IV. Primary Pumps and Flow Controllers

- A. Wound rotor motors with liquid rheostat control simulated (Hot leg pump location)
- B. General pump characteristics based on modified Donsky⁽¹⁾ curves assumed

- V. Transport delays between components - pipe heating neglected in present model (may be incorporated later).

2. Hybrid Simulation of the Reactor and Heat Transport System

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

The simulation of the reactor and primary hydraulics for pipe rupture studies has been completed on the Hybrid-2 machine. This program is identical to that employed on the UNIVAC 1108 machine with the time-consuming, DYNASAR general purpose code. The hybrid simulation operates well and will be compared in detail to the slower, more expensive code during the next report period. Large cost savings in pipe rupture computations are expected.

1. Benjamin Donsky. Complete Pump Characteristics and the Effects of Specific Speeds on Hydraulic Transients, ASME Paper No. 61-HYD-3, January 19, 1961.

3. Dump Heat Exchanger Analysis

R. N. Madsen (822)

Evaluation of the predicted DHX behavior using the four node simulation⁽¹⁾ as compared with the behavior predicted by the reference TAP simulation⁽²⁾ indicated that while the steady-state results were nearly identical, the transient response of the four node model was too rapid. Steady-state end points of the transient runs were again very close, however.

An extended re-investigation of the heat transfer calculations at length disclosed that the difficulty lay in simulation of the heat transfer in the extended surface fins. Fin efficiency is used to simplify heat transfer calculations involving extended surfaces, and is defined as the ratio of actual heat transferred from a given fin to the heat which would be transferred if all the fins were at a uniform temperature equal to the temperature at the base of the fin. The simplified 4-node simulation included the use of standard fin efficiencies for rectangular fins as given by Kern.⁽³⁾ As presented therein, fin efficiency, Ω , is a function only of the heat transfer coefficient, h , the thermal conductivity, k , and the geometry of the fin, which is a constant for any given DHX design. The difficulty was found to arise because of the mistaken assumption that the simple representation $\Omega = f(h,k)$ was valid for both steady-state and transient calculations.

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1. Fast Flux Test Facility Quarterly Technical Progress Report, BNWL-1174, p. 3.2, Battelle-Northwest, Richland, Washington.
 2. Fast Flux Test Facility Quarterly Technical Progress Report, BNWL-1090, p. 3.55, Battelle-Northwest, Richland, Washington.
 3. D. Q. Kern. Process Heat Transfer, p. 542, McGraw-Hill Book Co., Inc., New York, 1950.

Beginning with the previously given definition of Ω , which is:

$$\Omega = \frac{Q_{\text{actual}}}{Q_o}$$

where $Q_o = h A_f (T_f - T_a)$

and in which A_f = surface area of the fin

T_f = temperature of base of fin

T_a = air temperature associated with this fin,

and also the equation for the temperature at any point in a fin of constant thickness,

$$\frac{d}{dr} \left[r \frac{dT}{dr} \right] = \frac{h}{kY_b} rT + \frac{\rho c}{k} r \frac{dT}{dt}$$

in which $T = T_r - T_a$ at any radius r

$Y_b = 1/2$ fin thickness

ρ, c, k = thermophysical properties of the fin metal

t = time,

the time dependent expression for fin efficiency was derived. This expression is:

$$\Omega = - \frac{4\pi r_1 k Y_b}{h A_f (T_f - T_a)} \left(\frac{dT}{dr} \right)_{r=r_1} - \frac{4\pi}{h A_f (T_f - T_a)} \int_{r_1}^{r_2} \rho c Y_b r \frac{dT}{dt} dr$$

in which r_1 = radius at base of fin and all other symbols have been previously defined.

For steady-state $\frac{dT}{dt} = 0$ and $\frac{1}{(T_f - T_a)} \left(\frac{dT}{dr} \right)_{r=r_1} = \text{constant}$.

Thus, at steady-state, Ω is the simple function which was originally included, but Ω changes significantly when $\frac{dT}{dt} \neq 0$.

During a transient, in addition to the contribution of the second term, the $\left(\frac{dT}{dr}\right)_{r=r_1}$ factor in the first term changes as the temperature distribution in the fin changes during a transient.

Two different methods of correcting this difficulty have been investigated: the use of the actual expression for Ω derived above; and the abandonment of fin efficiency in lieu of simulating the fins by using additional nodes.

The use of the entire expression for fin efficiency necessitated the addition of only three integrations, but required some complex numerical approximations and logic in order to determine the value of $\left(\frac{dT}{dr}\right)_{r=r_1} = f(t)$ and $\left(\frac{dT}{dt}\right)$. This method was extremely susceptible to stability problems but was developed so that fairly good results were obtained with the exception of one unnatural oscillation at the beginning of a transient.

The other alternative takes the direct approach and simulates the fins. The disadvantage of this approach is that it doubles the required integrations. However, current investigation indicates that this may only require about 15% more computation time than by using the fin efficiency method. It also eliminates the stability problem and agrees well with the reference simulation. Consequently, this expanded model with four longitudinal and six radial nodes is being perfected for use as the DHX simulation.

4. Simulation of FFTF IHX

A. L. Gunby (822)

A study of IHX simulation approaches has been completed. The objective was to determine the limitations of current hybrid computer modeling for expected flow and temperature

transients. As before, detailed models of the IHX design of March, 1969 (1660 tubes) were studied by use of the DYNASAR code. The following areas were studied:

- The effect of using average temperatures for material properties
- The effect of using constant values for material properties
- The effect of lumping metal with sodium
- The effect of varying the number of axial nodes.

In all of the cases, the "mixed difference" scheme was used (alternating central difference with three-point backward difference from node to node), with flow and temperature transients input in different combinations.

Average temperatures: For these cases, material properties (density, specific heat, conductivity) and heat transfer coefficients for both coolant streams and the tube wall were computed based on an arithmetic average of the computed temperatures at the ends of the tube bundle. Flow decays and combined flow and inlet temperature transients were performed and compared to the detailed 8-node model, in which properties and heat transfer coefficients were computed at each axial mesh point. Results for outlet temperature responses showed that there was negligible difference between the two approaches.

Constant properties: For these cases, material properties were assumed to remain at their steady-state, initial values, while heat transfer coefficients were allowed to vary based on average coolant stream temperatures. Primary, secondary and combined flow decays were studied, and compared to the highly detailed model. Again, results for outlet temperature responses showed negligible difference (less than 1 °F at all points in time following transient initiation).

Lumped metal: The same three flow decays were applied to a model which lumped one-half the stored heat in the IHX tube metal with each coolant stream. Thus, separate metal temperatures at each mesh point were not computed. Average temperatures of each coolant stream were used to compute properties and heat transfer coefficients. When compared to the detailed model, outlet temperature transients were again shown to be negligibly different.

Number of nodes: Using the detailed model, the number of axial segments was varied to 4, 6, and 8. Each model was subject to the temperature and flow input transients, the outlet results were as expected: accuracy increases and thermal transients become steeper with more nodes. Figure 3.2 shows a comparison of primary outlet responses to a simulated secondary flow loss. However, for the purposes of hybrid simulation (control studies, operator training, and system thermal transient checks), it is concluded that a 4-node model with lumped metal and constant properties is satisfactory.

FIGURE 3.2. IHX Study FFTF; Design of March 1969, Constant Inlets, Secondary Flow to 10%

5. Heat Transport Control System Stability

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

A study was initiated to analyze the main heat transport control system stability in support of safety and system management. Transfer function measurements made from the hybrid computer simulation were used to develop a simplified model of the plant characteristics pertinent to the control stability problem. The use of the simplified simulation will permit high-speed computation to determine gross effects on system stability in relation to control configuration and controller adjustments. The validity of the simplified simulation has been verified by comparing its frequency response characteristics with those of the more detailed hybrid computer simulation.

CHAPTER IV. INSTRUMENTATION AND CONTROLSA. FFTF I&C SYSTEMS

1. Central Control and Data Handling System (System 91)
G. G. Thieme and G. L. Waldkoetter (933)

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

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

2. Reactor and Vessel Instrumentation System (System 92)

No report this quarter.

3. Plant Instrumentation System (System 93)
M. O. Rankin (933)

A tentative "acceptable" leak rate of nonradioactive sodium-to-air was determined to be 2.5 g/hr, based on a maximum level of one-tenth of the Threshold Limit Value for NaOH in the human body. Under these conditions, the maximum acceptable airborne concentration of H_2O will be $2 \times 10^{-4} \text{ g/m}^3$.

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

4. Fuel Failure Monitoring System (System 94)

a. FEDAL Monitor

R. R. Derusseau (924)

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

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

b. Failed Fuel Location by Xenon Gas Tagging

C. R. F. Smith (931), C. A. Mansius (N00) and D. L. Lessor (C00)

An evaluation was completed on the Xenon Gas Tagging concept for location of failed fuel in the FTR. (This concept is currently under development at ANL-Idaho for use in the EBR-II Reactor.) In this concept about 1 cm^3 of mixed isotopes of Xe is implanted in each fuel pin along with the helium fill-gas. Xenon tags of identical isotopic composition are implanted in each pin within a given subassembly. The tag associated with each subassembly must be distinguishably different from that in any other subassembly in the reactor. Some independent means (such as fission gas detection) is required to annunciate fuel failure. When a fuel pin fails, part or all of the xenon tag is released to the sodium coolant stream whence it finds its way into the reactor cover gas plenum. When fuel failure is annunciated, a fraction of the cover gas is passed through a cooled charcoal bed designed to absorb the xenon constituent. Subsequently, the xenon is desorbed from the charcoal and directed to a mass spectrometer which measures the isotopic

composition of the gas. The isotopic ratios of certain light xenon nuclides identify the subassembly containing the failed fuel pin.

Following are the major conclusions resulting from the evaluation of this concept:

- Xenon tagging is a feasible concept for locating failed fuel in the FTR. However, it will not infallibly locate all failed fuel under all circumstances. For example:
 - a. In the event of two or more nearly-simultaneous failures in different subassemblies, xenon tagging cannot be used to identify the failed subassemblies.
 - b. Failed fuel location by xenon tagging is not feasible where the reactor is deliberately operated with failed or vented fuel.
 - c. Xenon tagging is not a reliable concept for locating failures if fuel swelling inhibits the diffusion of tag gas throughout the fuel pin. Further analyses and experiments are required to establish the importance of this mechanism.

On balance, however, the xenon tagging concept is attractive because it is simple in design and operation, and it requires negligible in-vessel plumbing. It is also low in capital cost because the cost of the tag and the cost of implanting the tag are not capital cost items.

- The xenon isotopes which qualify for tagging are ^{124}Xe , ^{126}Xe and ^{129}Xe . The fission yield of ^{128}Xe in the FTR fuel precludes its use as a tagging isotope. It is recommended that ratios of $^{129}\text{Xe}/^{124}\text{Xe}$ and of $^{126}\text{Xe}/^{124}\text{Xe}$ be used as the basis for tags. Should it be necessary to implant a given tag ratio in more than one subassembly, then the measurement of the $^{128}\text{Xe}/^{124}\text{Xe}$ ratio can be used to identify the age (cumulative exposure) of a failed pin. Thus, ^{128}Xe can be used to extend the number of useful tags.

- Isotopic burnup is the principal determinant in establishing the number of unique tags which can be synthesized from available xenon mixtures.
- The maximum number of unique tags which can be synthesized (for FTR conditions) by utilizing only the $^{129}\text{Xe}/^{124}\text{Xe}$ ratio is estimated to range from 37 to 200. A closer estimate requires a more precise calculation (or measurement) of spectrum averaged cross sections of the xenon isotopes.
- The Xenon Tagging Concept can be used to locate a failed fuel before an appreciable fission product inventory is generated in a fuel pin providing the reactor cover gas is analyzed routinely for xenon isotopes following a fuel loading. Of all fuel failure location concepts, Xenon Gas Tagging is unique in this respect.
- The presence of xenon contamination in the cover gas may tend to mask the identity of a released tag unless both the xenon isotopic ratio and the xenon concentration in the cover gas is accurately measured both before and after the fuel failure event. It is not known at this time whether xenon contamination might prove to be a real problem in actual practice.

5. Flux Monitoring and Control System (System 95)

L. W. McClellan (932)

Neutron Detector Thimble Designs

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

A BNW analysis of a NaK cooling concept was completed by J. E. Krajicek of the FFTF Safety and Systems Analysis Sub-department. This analysis showed that NaK cooling of a BNW

thimble design to 300 °F was technically feasible. The model used in this study is delineated on drawing SK-3-14662, In-Core Flux Monitor Assembly not included in this report.

6. Radiation Monitoring System (System 96)

M. O. Rankin (933)

The methods of continuously monitoring for airborne plutonium were reviewed. One vendor has been found who will provide a small economical unit which can detect a continuous concentration sensitivity of one MPC* in a 4-hr period. The unit utilizes a silicon surface barrier detector, an amplifier and a single channel analyzer. The window of the analyzer is adjusted to view the 5.15 MeV energy peak of ^{239}Pu . Field experience at Hanford Project indicates these monitors perform extremely well. The analyzer is very stable and does not require frequent adjustment. The surface barrier detector has very low response to gamma background and, therefore, does not require any lead shielding.

B. DEVELOPMENT OF SENSORS AND RELATED EQUIPMENT

1. Low Level Neutron Flux Instrumentation

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

a. Thermal "Noise" Induced in Cables and Connectors

Low-level signals from high voltage neutron detectors are frequently obscured by "breakdown pulse noise" when coaxial cables supplying high voltage to the detectors are heated. Cable noise tests were made on a neutron detector wherein 19 feet of two coaxial cables were heated to 900 °F while polarized with 400 volts. (One coaxial cable is used for high voltage supply, while the other is used for signal.)

* *Maximum permissible concentration.*

Compared to the conventional single cable configuration, noise generation in the double cable configuration was reduced by a factor of 10^4 . In these tests the detector was not heated. Temperature-induced noise in single cable chambers has been a significant deterrent to high temperature operation.

Short-term temperature tests (to about 390 °F) of cable seals and connectors using this same type of cable have also been performed. These tests indicate that the two-cable, guarded-detector configuration virtually eliminates breakdown pulse noise arising from heated connectors.

b. Gamma Facility Modifications and Recalibration

Additions of ^{60}Co to the BNW gamma irradiation facility increased its source strength to 6.1×10^5 Ci; the available dose rate is now approximately doubled at the test positions. When current fabrication work is complete, this facility will be able to test at >300 °F and 1×10^6 R/hr.

c. Multiple Input Preamplifier Tests

A developmental model of a commercial three-input preamplifier is being evaluated. This preamplifier contains three separate channels of amplification, and provides pulse-shaping and discrimination prior to summation of neutron-generated signal pulses. Initial testing of this preamplifier using clustered detectors exposed to gamma fluxes of 10^5 R/hr showed improved tolerance to gamma flux as compared to single-input preamplifiers.

2. Pressure Sensors for Sodium Service

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

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

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

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

3. In-Reactor Coolant Temperature Sensors

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

Hot cell postirradiation testing has been completed on 30 of the 54-1/16-in. OD chromel-alumel thermocouples built to Hanford specifications, which have been irradiated in EBR-II to slightly more than 1×10^{21} nvt (fast) and at 1050 °F.

The following observations are based on a preliminary evaluation of test data:

- Within the accuracy of the measurement system ($\pm 1\%$), change in wire resistance is negligible.
- Time response measurements agree closely with preirradiation measurements.
- Postirradiation calibration shows a negative deviation in temperature (emf) compared to both the Pt/Pt-10% Rh standard thermocouple and to the nonirradiated (control) thermocouples. This negative deviation is annealed out of the irradiated thermocouples to some extent when they are exposed to temperatures higher than the EBR-II irradiation temperatures.
- Insulation resistance values, for the irradiated sensors, decreased by as much as nine decades; however, there is evidence that this is caused mainly by seal failure rather than by degradation of the insulation from irradiation.

C. OTHER INSTRUMENTATION AND CONTROLS WORK

1. Heat Transport System Control Study

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

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

2. Effects of Ionizing Radiation on Stabilized Photo Optics*

J. H. Kleinpeter (AOO), R. C. Hoffman (DUN) and
W. J. Clare (Wollensak, Inc.)

Several nonbrowning lenses were tested for useful life in a radiation field. Included in the test were a 30-150 mm variable focal-length or "zoom" lens, fixed focal-length lenses in 1/2-in. f/2.5, 1-in. f/2.5 and 4-in. f/2.5 sizes, and suitable controls of conventional glass. All the nonbrowning lenses were useful beyond 10^9 rad total integrated dose, and the 1-in. f/2.5 was usable after 2×10^{10} rad.

From these and associated data it was concluded that (1) nonbrowning lenses should be used in monitoring cameras located in radiation fields, (2) no lens lubrication of any type should be used, and (3) no plastic parts should be used.

* Reported by G. H. Strong (943)

CHAPTER V SODIUM TECHNOLOGY

Progress in the Sodium Technology Program for the FFTF is summarized in the Quarterly Progress Report, Sodium Chemistry Subdepartment, July - September 1969, BNWL-1200-1, now in publication.

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

L. Besel (913)

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

The first series of tests to be performed involved creep bending of a square bar of 304 SS. The purpose of this experiment is to characterize the creep behavior of the material in bending. Bar creep tests under pure bending are expected to demonstrate the degree of similarity between tension creep and compression creep. At the end of this report period, three bar tests had been conducted. The first bar was soaked at a temperature of 1200 °F under a no-load condition to check out the equipment. After this initial check-out, the bar was subjected to a steadily increasing load for one minute which was calculated to produce a maximum of 15,000 psi in the outer fiber stress. The deflection was constantly monitored with Linearly Variable Displacement Transducers (LVDT). The external transducers disclosed that the bar had not been plastically deformed, so loading was increased to produce 20,000 psi stress in the outer fibers of the bar. Upon loading, the bar showed a very slight permanent set. This permanent set was indicated by the LVDT's as the 20,000 psi stress level was reached.

The second bar was tested at 1200 °F and the imposed loading caused an outer fiber stress of 16,000 psi. Before

completion of the projected 100 hr test, a shorted condenser in the power supply caused an unknown overload on the specimen destroying the experiment. The third bar was tested for about 80 hr at a temperature of 1200 °F and a stress of 16,000 psi in the outer bar fibers. Mechanical measurements of the fiducial marks on the tension and compression faces which were in the central furnace region showed the following preliminary results:

Tension Creep + 0.18%

Compression Creep - 0.23%

It is expected that analysis of in-test and recheck of fiducial marks in the optical bench (where the initial values were measured) will be completed by mid-month. The next bar will be tested under the same nominal conditions to help establish the credibility of the results. Verification of the initial results shown above will permit selection of stress, time and temperature for the last test bar, and will indicate what redirection of the remaining experimental program is needed.

B. CORE PROCESS TECHNOLOGY

1. Gas Entrainment Measurement in the Outlet Region Model

H. Leigh (954) and J. Spalek (COO)

Possible reactor problems resulting from gas entrainment in the sodium coolant of the FTR requires that the potential for such occurrence be evaluated, and eliminated if possible. To provide the test data from the Hydraulic Core Mockup (HCM), on-line ultrasonic measurement of relative air content in the simulated coolant (water) is used. This technique has been successfully demonstrated in the HCM outlet region model. Initial on-line measurements of air content at the outlet nozzle pipe of the model yielded the following important information:

- The success of the ultrasonic measurements of air content (in bubble form) in the 10 in. pipe of the Outlet Model, at flow rates up to 3600 gpm, experimentally confirmed the applicability of the ultrasonic pulse-transmission technique for on-line gas entrainment measurements.
- The air content level recorded by a strip chart trace in the 10 in. outlet pipe of the Outlet Model is low, and very steady (uniform) in time, (except at the startup or during changes in flow rate). Thus, any changes in the air content level (or carry-under) can be easily and immediately detected by a glance at the strip chart trace. (That is, no statistical averages of areas under the strip chart trace need be done to detect the changes in air content level. However, this is true only for the very low, normally present air content level in the model's outlet pipe at high pool height.)
- Hydraulically induced mechanical vibrations in the pipe holding the transducers have no adverse effect on the ultrasonic measurements.
- Thermal Effects.
 - (a) Observed changes in the velocity of sound in water due to changes in water temperature, also have no effect on the gas content measurements. This is because it is the attenuation of sound (= change in "pulse-height") that is measured, and not the velocity ("location" in time) of the transmitted pulse.
 - (b) Also, under the normal operating conditions (small and gradual temperature changes), the different thermal expansions of the lucite and stainless steel transducer mounts have no adverse effect results. However, in a system with fast, large temperature changes, thermal expansion would misalign the transducers so that remounting and recalibrating would be necessary.

Calibration of the equipment for absolute void fraction measurements consists of injecting air into the outlet pipe at known rates. The absolute void fractions are then computed from the air and water flow rates, and plotted against the strip chart trace deflections. The points are then least-squares fitted to generate the calibration curves. This calibration process is repeated for various pool height and flow rate conditions, to eliminate any uncertainty in extrapolating the calibrated data to different conditions. Figures 6.1 and 6.2 give a pictorial description of the gas entrainment measuring equipment and technique. Figure 6.1 gives the overall layout. Figure 6.2 is a schematic of the apparatus, physical geometry, and illustrates the measuring techniques and theory.

FIGURE 6.1. Geometry for Ultrasonic Air Contents Measurements

FIGURE 6.2. Schematic of Apparatus, Geometry, Measuring Technique and Theory

2. Fuel Assembly Design

E. G. Stevens (913)

The wire wrap pin spacer system and a 12 ft maximum fuel assembly length have been baselined and approved by RDT for use in the first core driver fuel assembly.

A WARD recommendation to change the reference fuel assembly duct material to 316 SS, 20% cold-worked from 304 SS, solution annealed was reviewed and approved by BNW.

A BNW-WARD meeting to evaluate the metal swelling and creep equations to be used in FTR fuel assembly design was held on September 3-5, 1969. The previously agreed-upon swelling equations were recommended for continued use, namely:

A. Solution annealed 304 and 316 SS

$$100 \Delta V/V = 4.9 \times 10^{-49} (\phi t)^{1.71} \\ \times 10 (1.55 \times 10^4/T) - (5.99 \times 10^6/T^2)$$

B. 20% cold-worked 316 SS

$$100 \Delta V/V = 1.01 \times 10^{-36} (\phi t)^{1.69} \left(e^{-7800/RT} - 5480 e^{-25300/RT} \right)$$

The format of the equation for irradiation induced creep recommended is:

$$E_t^i = A \sigma (1 - e^{-t/B}) + C \sigma \phi^n t^m$$

Evaluation of the constants is in progress.

3. Fuel Subassembly Tests

a. CCTL Work I

E. G. Stevens (913)

Fuel pin bundle compression tests with the CCTL Mark I reassembled 217 pin bundle were run by BNW in a recently

completed compression test rig. Results indicated that the apparent "spring travel" was less than previously extrapolated, based on a crude, horizontal bundle test. The effect could be that less available area for fuel pin swelling exists within the duct resulting in an adjustment of the predicted maximum achievable burnup before bundle-to-duct interaction.

b. CCTL Mark II

E. G. Stevens (913)

The CCTL-Mark II is a full-scale model of the FFTF reference design to be tested for 9000 hr at conditions of maximum reactor flow, maximum clad surface temperature and proposed coolant purity conditions to provide verification of the mechanical design and evaluate the effect of potential long-term failure modes. Fabrication of the assembly is approximately 70% completed. The assembly contains a number of recent design changes, including reduced fuel pellet-clad gap, reduction of the number of zones of wire wrap looseness to a single zone, reinforcement of the lower grid to operate at the elevated test temperature provisions for four static pressure taps and segmentation of the upper and lower reflectors for greater pin flexibility.

c. Heat Transfer Characteristics of FFTF Fuel Assemblies

J. M. Yatabe (913)

The purpose of this work is to perform sodium heat transfer testing with multipin models of FFTF fuel assemblies to determine local clad temperature profiles in normal and structurally distorted FFTF fuel assemblies.

A retest was made of the 7-pin bundle normal geometry tests reported in the previous quarterly report. The measured temperature profiles in this verified the results of the first test and extended them to much lower flow rates.

Following this test the 7-pin heated bundle was disassembled and inspected. A photograph of the test assembly is shown in Figure 6.3. The major conclusions from this inspection were:

- A hollow wire wrap had rotated changing the original position of the thermocouple junction adjacent to the pin clad. This inspection results obviates the preliminary test conclusion that no hot spots appeared to exist at the contact between pin and wire wrap spacer. Preventive measures to prevent the reoccurrence of this problem have been taken.
- A clad temperature thermocouple failed due to shorting at the test assembly power lead-in. This problem can be corrected by a change in insulation design.
- No structural distortion of duct or pins occurred.

The major conclusions from this series of two tests and the post-inspection of the assembly are:

- The test assembly operated without fault for over 88 hours (40 hr at full power) clearly demonstrating the feasibility of the simulation of the fuel assembly thermal hydraulic conditions.
- The major problems occurred in the temperature instrumentation, especially the clad temperature thermocouples under applied power condition. Calibration methods for these thermocouples under power condition are being developed.

The second test series to evaluate the effect of increased bundle to duct spacing on fuel assembly temperature profiles are 90% complete, and the tests will be performed early in the next reporting period.

C. IRRADIATION TEST FACILITIES

1. Closed Loops (4 MW Closed Loop Cell Size Assessments) [A1]

Atomics International under the Technical Direction of Westinghouse completed an assessment of the cell size

Neg 0694105-3

FIGURE 6.3. Seven Pin Heated Model of FFTF Fuel Assembly
for Sodium Heat Transfer Testing

required for a 4 MW special purpose loop. It was established that the Primary Heat Transport System cell should be 14 × 18 × 42 ft deep and the purification cell 14 × 14 × 12 ft deep. The bases for this study were the requirements outlined in BNWL-500, Vol. 61, "Conceptual System Design Description for the Closed Loops," except that the components were derated from 6 to 4 MW. The above cell dimensions were determined by three principal considerations:

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

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

M. A. Vogel (924)

BNW provided testing requirements to Westinghouse to serve as a basis for developing the 2 MW loop concept. (Table 6.I.) Using these requirements, AI--under the Technical Direction of Westinghouse--completed the concept definition of the 2 MW Simple Unitized Modular Closed Loop System.

3. Closed Loop Tube Nozzle Closure Development

C. A. Munro (914)

a. Scope and Objectives

One of the major problems in assuring individual removal and replacement of test specimens in the operation of a test reactor such as the FFTF is the selection of methods of sealing reactor top face piping connections. The closed loops require a connection between the in-reactor tube and the ex-reactor heat transport system. This connection, which is made at the top portion of the tube (called nozzle), is difficult to achieve because of the small space envelope required and the high radiation and temperature levels.

The objective of this development program is to investigate both the mechanical sealing concept and the welding-cutting concept for making this connection.

TABLE 6.I. 2-MWt Closed Loop Testing Requirements

<u>Testing Conditions</u>	<u>Values</u>	
Maximum Na temperature from test section	1400 °F	Achieved by internal bypassing and reducing total heat generation of test compatible with maximum flow capability.
Maximum Na hot leg temperature - primary	1200 °F	
Maximum Na cold leg temperature - primary	1150 °F	Nearly isothermal test
Maximum Na flow rate	1.14×10^5 lb/hr	Determined by 200 °F ΔT at 2 MW
Minimum Na flow rate	10% of maximum	For control range
Maximum ΔT in primary	400 °F at 2 MW	
ΔP across test section	100 psi at $\Delta T = 200$ °F	
Minimum Na cold leg temperature - primary	600 °F at 2 MW	For lower temperature, accept lower power
Maximum heat generation of test section	2 MW	
Minimum cold leg temperature - secondary	500 °F at 200 °F ΔT and 2 MW	For IHX LMTD of 100 °F

b. Closed Loop Nozzle Mechanical Closure Studies

R. Kolowith (914) and M. R. Kreiter (COO).

The objective of this work is to develop a reliable closed loop nozzle mechanical closure mechanism amenable to remote operation.

The seal must perform effectively in a high radiation field at high temperatures to contain the flowing liquid sodium or the sodium saturated cover gas. The remotely operable and compact closure required for this application is not within the current state of technology.

Six metallic seals are being tested to determine compatibility with a creep compensating connector concept as shown in Drawing SK-3-15132 (Sheet 1 and 2) and Figure 6.4. This closure concept was designed to function remotely in the space required at high temperature and radiation levels. The seals and test joint modules are being supplied by seal vendors.

The connector has been successfully latched and unlatched at room ambient temperature conditions. The actuation was initially achieved with a manually operated hydraulic cylinder, also shown in Figure 6.4.

The first seal performance data has been recorded on this test program. Proof pressure and leakage tests have been completed on one NDA 4-point seal. This seal concept was developed by Nuclear Development Associates and is described in their report.⁽¹⁾ Three seals with a cross section similar to that described by NDA, but larger in diameter, were fabricated from Armco Ingot Iron. The first of these seals was assembled with the prototype creep compensating connector and loaded to approximately 18,000 lb. The load versus deflection curve indicated that the seal seated properly and that the spool was securely latched in the nozzle. The seal was pressurized to a proof pressure of 375 psig helium with no structural deformation evident after the test.

A room temperature leak rate test was conducted with the seal pressurized to 250 psig. Before the test, all parts of the vacuum system were flooded with helium and the leak detector output was monitored to locate any possible leaks. No leaks were found. A leakage rate of 3.2×10^{-4} atm cm³/sec was

1. Design Development Tests of Some Components for the 10 MW_e
Doc. No. NDA-84-21, May 25, 1969

QTY	MAT	DESCRIPTION	MATL	RBA
1	ASSEMBLY			SW#1
1	2	UPPER SEAT HOLDER		
1	3	PRESSURE PLATE		
1	4	LATCH CONNECTOR		
3	5	LATCH FINGER		SW#2
1	6	FINGER HOLDER		
1	7	BELLEVILLE WASHER		
1	8	BODY		
1	9	NUT		
1	10	CONNECTOR BLOCK		BY CUST
1	11	CONNECTOR BLOCK		
1	12	SPACER		
AR	13	BELLEVILLE WASHER		
3	14	ROLL PIN #19-048-250-850		LSNA
1	15	PACKING GLAND #HREL-125 DIA		CONAX
1	16	SPACER		BY CUST
AR	17	TUBING 1/2 OD x .065 WALL	304 SST	
4	18	SET SCR #10-24 UNC-2A x 1/2 LONG, SOC HD	18-8 SST	CUP POINT
3	19	SET SCR #11 UNC-2A x 1/2 LONG, SOC HD		CUP POINT
4	20	CAP SCR #10-24 UNC-2A x 1/2 LONG, SOC HD		
1	21	INSERT		SW#2
12	22	MACH SCR #10-24 UNC-2A x 1/2 LONG, FLND	18-8 SST

- GENERAL NOTES
(UNLESS OTHERWISE SPECIFIED)
1. TOLERANCES: FRACTIONAL $\pm .005$, DECIMAL $\pm .005$, ANGULAR $\pm 30'$
 2. ALL MACHINED SURFACES $\sqrt{}$ SURFACE TEXTURE SHALL BE IN ACCORDANCE WITH: FEEDS: MAX = 1 (LATEST REV.)
 3. BREAK ALL SHARP EDGES AND REMOVE ALL BURRS
 4. ALL WELD TO BE AS SPECIFIED OR APPROVED EQUAL QUALITY
 5. USE SILVER GOOD CHALL TND AT ASSY. (REF: CRAWFORD FITTING CO.)
 6. GEOMETRIC TOLERANCES IN ACCORDANCE WITH: USA STD DRAFTING PRACTICES: USA S1 Y14.5-1966.

U. S. ATOMIC ENERGY COMMISSION	
RESEARCH OPERATIONS OFFICE	
PACIFIC NORTHWEST LABORATORY	
OPERATED BY BATTELLE MEMORIAL INSTITUTE	
CREEP COMPENSATING CONNECTOR ASSEMBLY & DETAILS	
FAST FLUX TEST FACILITY	
DESIGNED BY: J. E. HANLEY	DATE: 3/6/64
CHECKED BY: J. E. HANLEY	DATE: 3/6/64
APPROVED BY: J. E. HANLEY	DATE: 3/6/64
REVISIONS:	REVISIONS:
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DETAIL 'X'

NOTE
SEE SHEET NO 1 FOR MATERIAL LIST AND GENERAL NOTES

		BY DATE		ISS	BY DATE	ISS	DESCRIPTION	
		APPROVED FOR CHANGING		DATE	BY	DATE		
		BY DESIGN (S)		DATE	BY	DATE		
DWG NO.		REV NO.		DRAWING TITLE (or NUMBER)		REVISIONS		
						CHECK PART ISSUE NO.		
REFERENCE DRAWINGS		COMMENT PART ISSUE NO.				DRAWING STATUS		
NEXT USED ON								

[illegible]

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FIGURE 6.4. Test Hardware for Creep Compensating Closure Concept

measured at a pressure differential of 250 psi across the seal. There was no indication of clamp failure at any time during the test. Visual examination of the connector assembly after disassembly revealed no indications of connector failure.

Extrapolation of our data gives a leak rate of 4×10^{-7} atm cm³/sec at 15 psi differential.

Test work is continuing.

C. Closed Loop Nozzle Weld Closure Studies

R. F. Gilmore (914) and L. J. Rousseau (C00)

The objectives of this program are to demonstrate the feasibility of remotely welding the closed loop nozzle plug and reactor top face spool piece.

The welded joint must be leak tight, provide pull down compression loading for mechanical sealing, and be structurally capable of withstanding extreme temperatures, and high gas pressure transient loads. The welding method must be highly reliable, capable of repeated operations and be performed completely by remote control. The assessment of design requirements, commensurate with known welding technology, immediately centered development work toward a controlled mechanically driven tungsten arc welding head and various filler ring configurations. Early experiments to achieve the optimum weld section began with 304 SS rings fused into different joint shapes, using various electrode adjustments. The present filler ring shape and joint profile is shown on the Side Mill Cutter Test Assembly illustration, Figure 6.7, p. 6.20. Development testing conducted in controlled inert gas environments with molten sodium below the joint has proven the selected

methods to be feasible. The optimum designs derived from feature testing have been incorporated into a new weld head assembly. This machine is currently being prepared for verification testing later this year. These tests will be performed in a new environmental chamber that will be capable of closely simulating reactor operating conditions. The chamber is equipped with a hoist system and support apparatus to also accommodate the remote cutting operations.

d. Closed Loop Nozzle Weld Cutter Studies

R. F. Gilmore (914)

The objective of this program is to develop reliable mechanical cutting methods to separate the welded top closure of the closed loop tube. The methods must be amenable to remote controlled operations, capable of repeated operations in high temperature sodium-vapor environments and in no way contaminate the sodium coolant.

Design studies and feature testing of various mechanical cutters have centered around the development of tool configurations that will repeatedly produce small metal chips capable of being completely removed through a vacuum system. Due to the design constraints of the weld joint configuration, only two concepts have been incorporated into cutter test sub-assemblies.

Figures 6.5 and 6.6 show the Side Mill Cutter Test Assembly, and Figure 6.7 is the Multiple Tooth Trepanning Test Assembly. Test performance evaluations and reliability considerations have resulted in the selection of the Side Mill Cutter concept to be built for the remote proof-of-principle machine and the development of the trepanning technique for the alternate method.

FIGURE 6.5. Side Mill Cutter Test Assembly

Neg 50518-3

FIGURE 6.6. Side Mill Cutter Test Assembly

Neg 50518-1

FIGURE 6.7. Multiple Tooth Trepanning Test Assembly

Procurement orders have been placed for the proof-of-principle cutter. Preparations are underway to construct a test stand and assemble test equipment for demonstration testing yet this year. The machine is expected to demonstrate that all operations performed on a heated weld-joint sample can be controlled from a remote control panel. The machine will be representative of the reactor equipment except for the single drive motor, gear box arrangement, and sodium vapor sealing. These features will be incorporated for sodium environmental testing.

The trepanning test assembly (Figure 6.7) which comprises 4 single point tool bits, has produced small chip formations when the assembly was allowed to chatter in the shop machine (radial drill press) setup. However, when the machine support structure was stiffened, and the vibration frequencies minimized, irregular length chips were formed and clogged the vacuum system employed. The test module has recently been modified to mechanically control this chatter motion concept of cutting. The mechanism consists of two ratchet type disks designed to produce short scallops 0.004 in. deep into the cutting material, very similar to the cut configuration obtained during the natural frequency test setup. The application of a hydraulic vibrator is also being considered to impart a pulsed feed to the cutter tools. Test results of this arrangement will appear in the next periodic report.

D. EXAMINATION FACILITY

1. Neutron Radiography

W. S. Chenault, D. J. Meyers and E. B. Ramsey (942)

An evaluation of neutron radiography was completed. It was decided to defer neutron radiography and make design provisions for possible future installation of a neutron radiography reactor.

2. Axial Gas Cooling Equipment

W. S. Chenault (942)

The conceptual design study of Axial Gas Cooling Hardware for removal of decay heat from irradiated driver and test fuel assemblies was completed and documented⁽¹⁾ by General Electric/BRDO.

RDT concurrence was obtained on the BNW recommendation that gas flow by suction be used for both axial and transverse cooling of driver and test fuel assemblies.

1. *Published as GEAP-10058 dated July, 1969.*

CHAPTER VII. FUELS AND MATERIALSA. FUELS AND MATERIALS EVALUATION1. Irradiation of Hypostoichiometric $\text{UO}_2\text{-PuO}_2$ -Fuel

R. D. Leggett, J. W. Weber, L. A. Pember, D. Christensen,
D. W. Dragnich (750) and E. D. Jenson (A00)

This task includes irradiation testing of mixed oxide fuel specimens of different stoichiometries to high burnups in a thermal flux. The apparent effective thermal conductivity during irradiation of hypostoichiometric mixed oxide fuel, relative to that of stoichiometric fuel, offers the potential for reactor operation at higher power densities and for lower fuel temperatures. The swelling and gas release characteristics of $\text{UO}_2\text{-PuO}_2$ fuel as a function of stoichiometry and fuel form must be determined at high burnup before a maximum performance fuel can be selected.

The general status of the fuel pins being irradiated and/or examined are summarized in Table 7.I. The reported burnup values were derived from ^{148}Nd analyses of sections removed from the center of the fuel pins. The burnup values for those pins not yet examined were estimated from the reported fluence, assuming a linear relationship between fluence and burnup. BNW 1-10 and 1-12 were discharged on November 9 at the end of ETR Cycle 104. These two capsules will be gamma scanned at the ETR and then when the MTR resumes operation in early December they will be neutron radiographed. A decision will be made then whether to return the capsules to the ETR for further irradiation or to terminate the irradiation and start examinations.

The other ten capsules have either completed their post-irradiation examination or are in the process of being examined.

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

Issue Date: 11/1/69

Test Designation	Description ⁽¹⁾			Status			
	Fuel Form	O/M	Facility	Goal Exposure MWd/tonneM	Actual Exposure MWd/tonneM	Discharge Date	Status
BNW 1-1	Solid	1.93	MTR	100,000	(Est)120,000	2-69	Being Exam. at BNW
BNW 1-2	Solid	1.93	ETR	50,000	(Est) 70,000	2-69	Being Exam. at BNW
BNW 1-3	Solid	1.96	MTR	100,000	(Est)150,000	2-69	Being Exam. at BNW
BNW 1-4	Solid	1.96	MTR	50,000	71,000	6-68	Exam. Completed at BNW
BNW 1-5	Solid	2.00	MTR	100,000	(Est)160,000	2-69	Being Exam. at BNW
BNW 1-6	Solid	2.00	MTR	50,000	75,200	10-68	Being Exam. at BNW
BNW 1-7	Annular	1.93	MTR	50,000	<1,000		Lead Tube Failed. Exam at BNW
BNW 1-8	Annular	1.93	MTR	100,000	29,000	1-68	Failed. Exam. at BNW
BNW 1-9	Annular	1.96	MTR	50,000	81,800	6-68	Being Exam. at LASL
BNW 1-10	Annular	1.96	ETR	100,000	(Est) 92,200	(Est)2-70	Discharged 11-69 for Interim Exam.
BNW 1-11	Annular	2.00	MTR	50,000	87,400	8-68	Being Exam. at LASL
BNW 1-12	Annular	2.00	ETR	100,000	(Est) 96,500	(Est)2-70	Discharged 11-69 for Interim Exam.

1. Fuel

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

Cladding

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

Axial power distribution has been calculated based on burnup analysis and gross or $^{95}\text{Zr}/^{95}\text{Nb}$ axial gamma scans for BNW-1-4, 1-6, 1-9 and 1-11. These data are being compared to axial power distributions derived from thermocouple data. The thermocouple-derived power is based on ΔT calculations assuming a single, time-average temperature for the entire period of operation of the experiment. This comparison showed that the power derived from the thermocouple data is from 5 to 20% less than the power calculated from burnup data. Work is being continued in the reduction of thermocouple data and axial gamma scans to obtain a better correlation. ETR and MTR thermal flux maps from core locations near the experiments are being obtained to compare with axial gamma scan data.

2. Fuel Pin Steady State and Transient Performance Limits

G. E. Culley, D. O. Sheppard, B. A. Crea (750) and
U. P. Jenquin (COO)

The objective of this program is to evaluate the performance characteristics of FTR prototypic fuel pins under steady state and transient irradiation conditions.

The first series of thermal flux steady state irradiations in support of this task were completed. Six capsules attained a nominal goal burnup of 10,000 MWd/tonneM and four capsules attained a goal burnup of 50,000 MWd/tonneM. Two capsules, PNL-59-11 and 12 were removed from the reactor at a burnup of 20,000 MWd/tonneM (goal of 50,000) due to a lack of sufficient irradiation units during FY-1969. Transient testing of the capsules at this intermediate burnup level is being planned.

Analysis of the thermocouple data collected during the irradiation of the twelve PNL-59 capsules was completed. Fuel pin power as a function of time and accumulated burnup was obtained. Table 7.II presents the results. These data compare very well with the radio chemical analyses conducted on

two of the fuel pins. The analytical values for the peak and average burnup on PNL-59-5 are 18,290 and 13,777 respectively. For capsule PNL-59-7 the peak and average values are 69,763 and 51,563 respectively. The neodymium fission product was used in the burnup analyses.

TABLE 7.II. GETR Capsule Calculated Performance

<u>Capsule</u>	<u>Axial Peak Power, kW/ft</u>	<u>Burnup, MWd/tonneM</u>	
		<u>Axial Peak</u>	<u>Axial Avg.</u>
PNL-59-1	13.87	18,617	13,810
-2	14.07	18,537	13,750
-3	14.56	19,277	14,298
-4	14.16	18,647	13,832
-5	13.64	18,505	13,653
-6	14.21	19,544	14,423
-7	13.52	68,803	51,376
-8	13.67	69,440	51,811
-9	13.99	72,040	54,117
10	13.65	69,823	52,432
-11	14.73	28,184	20,904
-12	13.26	25,992	19,412

The design work continues on two types of capsules to be steady state irradiated during FY-1970.

The sections cut from the PNL-59-7 capsule fuel pin are being examined. The burnup specimens which were cut from the axial peak and average positions as determined by the fuel pin gamma scan were radio chemically analyzed. Using the fission product, neodymium-148 the peak and average burnup values were determined to 69,763 and 51,563 MWd/tonneM respectively.

3. Fast Flux Pin Irradiation

a. Subassemblies PNL-3, 4 and 5

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

Three, 37-unencapsulated-pin, experimental subassemblies (25 wt% PuO_2 -75 wt% UO_2 , 304 SS cladding, 0.250 in. OD) in EBR-II are operating well. These subassemblies had attained the following estimated peak burnup at the end of EBR-II Run 38B (October 18, 1969).

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

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

b. Subassemblies PNL-6, 7 and 8

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

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

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

4. Fuel Performance Prediction and Analysis

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

This work is directed toward the development of empirical and theoretical formulations for predicting mixed oxide fuel behavior during irradiation. Detailed analyses of irradiation data will be employed to "calibrate" the various mathematical

models that exist and to select those models that best describe fuel performance. The ultimate goal of this task is to develop a comprehensive analytical procedure that can be used to predict fuel performance from considerations of fabrication parameters, irradiation conditions, and basic material properties of the fuel and cladding. Both steady-state and transient conditions will be considered.

a. Hybrid Computer Simulation

The Hybrid Computer has been found useful in correlating predictions of the behavior of mixed oxide, stainless steel clad fuel pins with postirradiation examination results. The Hybrid greatly simplifies the work of correlation and provides a visual display of the model predictions. Input variables that may be uncertain can easily be adjusted electronically to provide an immediate change of the visible output.

The equations describing fission gas release, pressure buildup, two-zone and three-zone sintering and radial thermal flux power profiles have been programmed into the Hybrid Computer and are in the final stages of debugging. The equations describing cladding swelling will also be incorporated into the program.

The analog portion of the computer has been programmed for a visual display of both radial and axial fuel cross sections for either the two or three-zone model. This simulation of the axial hole, columnar, equiaxed, and nonrestructured regions of the fuel allows very convenient comparisons and matching of fuel pin microstructures with model predictions.

The generation of both axial and radial temperature profiles in the form of temperature versus position plots is being incorporated as part of the program output. This will result in a significant time savings in the assimilation of fuel pin temperature data.

A limited amount of structure matching was done with PNL-1-6 (mixed oxide irradiated in EBR-II to 7850 MWd/tonneM at 8.3 kW/ft) by placing half a reduced mosaic on the output cathode ray tube of the Hybrid. It was discovered that this technique is unsatisfactory due to the difficulty in matching from one Hybrid setting to another.

A clear plastic overlay representing the fuel structure was found to improve the ease of matching but small differences were still found to exist due to the finite dimensions of the lines on the overlay and the circles generated by the Hybrid. Refinement of the system continues.

5. Subassembly PNL-2 Irradiation

D. C. Bullington, R. D. Leggett, and H. C. Bowen (750)

The PNL-2(X032) subassembly was discharged from the EBR-II August 31, 1969, at the end of Run 37. The estimated peak burnup on this subassembly at discharge was 47,000 MWd/tonneM. The subassembly was disassembled at EBR-II without incident. Neutron radiography performed at TREAT indicated that all pins were intact. The only unusual items visible on the neutron radiographs were what appears to be a small amount of fuel powder in the annulus around the insulator pellets on two pins and one of the pins with 0.010 in. thick cladding showed appreciable bowing. The capsules have been received at BNW. Following gamma scanning five capsules will be selected for shipment to LASL for destructive examination. Three capsules will be destructively examined at BNW.

6. Subassemblies PNL-9, 10 and 11 Irradiation

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

The experiment description and safety analysis report for the irradiation of these 61-pin* subassemblies in EBR-II has

* 0.230 in. OD $\times$ 0.015 in. thick, 20% cold-worked 316 SS cladding. Eighty-five and one-half percent TD smeared density, 25% PuO₂-75% UO₂ fuel. Pellets have nominal 3-mil deep dish on each end.

been completed and the first draft is being typed. It is calculated that the goal burnup of 80,000 MWd/tonneM (peak) can be achieved for all subassemblies, without cladding deformation due to fuel-cladding interaction. Additionally, fuel-pin-bundle-hex can interaction is not predicted to occur at the goal burnup.

7. EBR-II Irradiations

P. K. McDaniel (722)

The EBR-II Irradiations program is to provide coordination and direction for the timely accomplishments of fast reactor irradiations of structural materials.

Structural material irradiation experiments are in various stages of design, construction and irradiation in EBR-II. Currently an experiment consisting of three Mark B-7 Type experimental pins is being fabricated for changing into the EBR-II core about January 1, 1970. The goal exposure for this experiment is to be 3×10^{22} n/cm² (E > 0.1 MeV). The purpose of this experiment is:

- To provide differential swelling data on prototypic cladding, wire wrap and duct
- To measure differential swelling rates of prototypic duct-wear pad couples
- To investigate the swelling behavior of tantalum control rod alloys.

EBR-II operating power was increased on September 13, 1969 from 50 MW to 62.5 MW for Run 38A only. During this test, subassemblies X021B and X057 were removed from the reactor.

Two subassemblies, X067 and X019, were discharged from EBR-II on 8/5/69. Both subassemblies have been disassembled. Specimens are being tested and flux monitors are being counted.

Subassembly X018B containing pin no. BNWL B-40 was loaded into position 4E2 of EBR-II reactor grid on October 2, 1969.

An updated summary of experiments in EBR-II is given in Table 7.III. The accumulative exposure values given are total exposure to the center of the subassembly and acceptance limit of $\pm 15\%$ is placed on these values.

8. Stainless Steel Swelling

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

The objectives of this effort are to establish the radiation-induced swelling characteristics of FFTF alloys, relate fast-reactor induced substructural changes in microstructure to corresponding changes in mechanical properties and to provide a basis for extrapolation and interpolation of data using both microscopic modeling and empirical approaches.

Recent experiments concerning the effects of fast-reactor irradiation on stainless steels have demonstrated that a considerable volume change occurs during neutron bombardment.⁽¹⁾ Unlike volume changes in fuel materials, swelling in structural alloys and pure metals is due to the condensation of irradiation-produced excess vacancies to form three dimensional cavities. United Kingdom investigators have found as much as 6.8% volume change in 316 SS irradiated to 7.8×10^{22} n/cm² total fluence. Volume changes of this magnitude, if not taken into account, are sufficient to cause distortion in reactor components.

1. C. Cawthorne and E. J. Fulton. "Voids in Irradiated Stainless Steel"; Nature, vol. 216 p. 575, 1967.

TABLE 7.III. Summary of Experiments in EBR-II

Subassembly Designation	Reactor Position	Date Charged	Estimated Discharge Date	Type of Specimens	Material	Accumulative ^(b) Exposure n/cm ²	Goal Exposure n/cm
X018B	4E2	10/2/69	Oct. 70	Density	304, 316 SS	0.23×10^{22}	1.0×10^{23}
X020	6B5	1-13-67	July 70	28 Tensile 8 Uniaxial Creep	304 SS 316 SS	4.65×10^{22}	5.6×10^{22}
X021B ^(a)	2D1	2-27-67	Aug. 70	501 Tensile or Uniaxial Creep 65 Biaxial Creep	304 SS, 316 SS, 348 SS, 321 SS ^(c) Ni Base Alloys	7.10×10^{22} $(2.99 \times 10^{22})_B$	8.8×10^{22}
X027	4B3	11-21-67	Aug. 70	5 Tensile 3 Uniaxial Creep	304 SS	5.5×10^{22}	5×10^{22}
X041	7C4	7-24-68	May 70	713 Tensile or Uniaxial Creep 74 Biaxial Creep	304 SS, 316 SS, 321 SS, 348 SS	2.43×10^{22}	3×10^{22}
X057	2B1	2-24-69	March 71	577 Tensile or Uniaxial Creep 164 Biaxial Creep 58 Density	304 SS, 316 SS, ^(c) 321 SS, 348 SS, Ni Base Alloys	3.04×10^{22}	6.0×10^{22}

(a) Original subassembly with two single pin interchanges.

(b) Accumulative to end of Run 38B.

(c) Variety of other alloys.

Maximum target fluences for FFTF are as high as 10^{24} n/cm². Since such a high fluence cannot be attained in a reasonable time using currently available reactors, design data for high exposures must be based on extrapolation using physically realistic swelling models. The approach taken in this study is to develop swelling data at fluence levels that can be obtained in reasonable times and to use this data to develop the required swelling models.

A new swelling equation for solution treated AISI Type 316 and 304 SS has been developed:

$$100 \frac{\Delta V}{V} = 4.9 \times 10^{-49} (\phi t)^{1.71} 10^{\left(\frac{1.55 \times 10^4}{T} - \frac{5.99 \times 10^6}{T^2} \right)}$$

where

$$100 \frac{\Delta V}{V} = \% \text{ volume change}$$

$$\phi t = \text{fluence, neutrons/cm}^2 \text{ (E > 0.1 MeV)} \quad (1)$$

$$T = \text{temperature, } ^\circ\text{K}$$

Equation (1) is based on density measurements of irradiated solution-treated Type 304 EBR-II components and the results for solution-treated Type 316 tensile specimens reported in a previous quarterly report.⁽¹⁾ This equation is recommended for design purposes.

No changes have been made in the recommended equation for swelling in Type 316 SS with 20% cold work. The current equation is as follows:

$$100 \frac{\Delta V}{V} = 1.01 \times 10^{-36} (\phi t)^{1.69} \left(e^{-7800/RT} - 5480 e^{-25,300/RT} \right) \quad (2)$$

1. FFTF Quarterly Technical Report, June, July, August 1969.
BNWL-1174, Battelle-Northwest, Richland, Washington,
October 1969.

The temperature dependence of swelling predicted by these equations at a fluence of 5×10^{22} n/cm² ($E > 0.1$ MeV) is shown in Figure 7.1. The data used to develop the equations were normalized to 5×10^{22} n/cm² ($E > 0.1$ MeV), using the fluence dependency indicated by the equations, and plotted on the figure for reference.

Transmission electron microscopy studies of irradiated 316 and 304 stainless steels are being continued to develop a phenomenological description of void formation. The essential characteristics of the fluence and temperature dependence of void formation are suggested in Figures 7.2 and 7.3. It must be stressed that these figures are tentative. The void size increases nearly linearly with increasing irradiation temperature but increases only slightly with fast fluence. Conversely, the void number density decreases with increasing irradiation temperature but increases with fluence. The decrease in void number density with increasing temperature appears to be more pronounced at low fluences than at high.

B. MATERIALS TECHNOLOGY

1. Uniaxial Creep

A. J. Lovell (AOO)

Attempts to correlate the creep rate data of unirradiated 304 and 316 SS have resulted in the following expressions:

304 SS

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

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

316 SS

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

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

where

FIGURE 7.1. Temperature Dependencies Predicted by Swelling Equations for Solution Treated Type 316 and 304 SS and for Type 316 SS with 20% Cold Worked

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

FIGURE 7.3. Void Size in Fast Reactor Irradiated Austenitic Stainless Steel as a Function of Fast Fluence and Irradiation Temperature

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

σ = applied engineering stress, ksi

T = temperature of test, °K

These expressions were obtained using the creep data available over a wide range of applied stress and test temperatures.

The testing of 304 and 316 from Subassembly X018 is continuing. One test on 304 at 1000 °F and two at 1100 °F have been completed. The loss in rupture life for the 1000 °F test after an irradiation of $\sim 1.6 \times 10^{22}$ n/cm² (total) at 930 °F is more than a thirty-fold reduction when compared to extrapolation of solution treated control data. The strain during the test was very low, $\sim 0.3\%$. The indicated reduction in minimum creep rate was about three-fold. Tests at 1100 °F after irradiation at 1140 °F to 3.7×10^{22} n/cm² (total) do not show the marked rupture life losses with only three- and four-fold losses for the 30 and 25 ksi tests.

The data for the two 1100 °F tests on 316 SS irradiated at 1110 °F to 3.5×10^{22} show marked reduction in rupture life somewhere in the fifty-fold-plus range. Increased creep rates are observed, and the ductility (as indicated by the creep curves) is less than 1%. This rather divergent behavior between 304 and 316 has been observed earlier and the present specimens tend to confirm it. It is not expected that the variation in irradiation conditions can explain the differences and additional analyses are needed.

2. Tensile Studies*

A. L. Ward (AOO)

Tensile test results have been reported from AISI Types 304 and 316 SS irradiated in the EBR-II Subassembly X018. This experiment includes specimens irradiated to total neutron

* Reported by J. C. Tobin (960)

fluences ranging from a peak of 3.7×10^{22} n/cm² to 0.7×10^{22} n/cm² (3.2×10^{22} to 0.31×10^{22} n/cm², $E > 0.1$ MeV) at temperatures from 700 ± 10 °F to 1140 ± 50 °F.

In general, the degree of irradiation hardening, in terms of increased yield strengths and reduced ductilities, increases to a maximum near 2.0×10^{22} n/cm² (total) and 960 ± 35 °F in Type 316 and 2.0×10^{22} n/cm² and 1010 ± 30 °F in Type 304.

The maximum yield strengths in both alloys constitute a three-to four-fold increase relative to unirradiated controls. No well defined difference between 304 and 316 can be established in this analysis because of the irradiation temperature difference between the two groups of specimens; the 316 was located below midplane and the 304 above. Type 304 specimens show a peak strengthening at the same fluence level as the Type 316, but, possibly because of irradiation temperature effects in this range, it shows a lower maximum yield strength.

The effect of test temperature is indicated in those cases where several specimens represent common irradiation conditions. Type 316 specimens irradiated to 2.0×10^{22} n/cm² at 960 ± 35 °F and tested at 800, 900, 1000, 1100, and 1200 °F show yield strengths greater than 70,000 psi up to 1000 °F and 50,600 psi at 1200 °F. This behavior is consistent with the results of previous studies which resulted in attributing hardening up to ~1000 °F to Frank dislocation loops and polyhedral voids and increasingly to voids alone at higher temperatures where the loops anneal out.

The ductility behavior is also consistent with previous work. It is generally accepted that below about 1000 °F the observed ductility losses are associated with the hardening of the matrix, while at higher temperatures further losses are

added by grain boundary weakening and helium embrittlement. Transmission electron microscopy is in progress, and preliminary indications are that the observed properties of both alloys in this study can be related to the density and size of homogeneously distributed loops and voids.

3. Weldment Studies*

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

A series of specimens taken from a submerged-arc weldment in one-inch thick plate of AISI Type 304 stainless steel using Type 308 filler metal have been tensile tested at 700 °F following annealing at 1000, 1200, 1400, 1600, and 1800 °F for periods of 1 and 10 hr. Due to the large degree of data scatter inherent in the weldment specimen results, definitive conclusions regarding the effects of annealing temperature and time are difficult. There appears to be a reduction in yield strength with increasing annealing temperature, but any corresponding ductility improvement is obscured by the data scatter.

Room-temperature and 1100 °F tests were performed for as-fabricated specimens representing four welding processes: TIG, MIG, Submerged-Arc, and Stick-Electrode. The room temperature tests were performed in order to facilitate comparison with the literature and 1100 °F tests to support the creep-rupture portion of the weldment characterization program.

Tensile testing of irradiated weldment specimens (EBR-II Subassembly X067) has begun. Preliminary results indicate that the tensile properties are not significantly affected by irradiation to $\sim 5 \times 10^{20}$ n/cm² at 700 or 800 °F. Some irradiation

* Reported by J. C. Tobin (1960).

hardening is observed after irradiation to $\sim 5 \times 10^{21}$ n/cm² at 800 °F, but with the possible exception of the TIG weld case, ductilities losses to the critical 10% level have not been observed.

4. Subcritical Crack Growth*

E. B. Schwenk (A00) and L. A. James (A00)

Fatigue crack growth data now available are shown in Figures 7.4 and 7.5. The data in Figure 7.4 show a possible effect of mean stress level and/or a load cycle rate-environment influence on crack growth rate da/dN . Specimen RML-2⁽¹⁾ shows an increased da/dN in comparison to specimens (RML-3, RMT-4, F-8) that were load cycled at a higher rate (400 cpm) and lower mean stress level (~ 9300 psi). Environmental effects can increase crack growth rate in other alloy systems. Thus it is possible that the data in Figure 1 may reflect, in addition to a mean stress level effect, a combined influence of cycle rate and ambient laboratory conditions.

The effect of elevated temperature (1000 °F) on crack growth rate, as shown in Figure 7.5, is significant especially at lower stress intensity factor levels but less so at higher ΔK levels. For a ΔK level of 10,000 psi $\sqrt{\text{in.}}$ (extrapolated at 1000 °F), the crack growth rate is 10 to 12 times faster than at room temperature. For higher ΔK levels and, hence, for higher stress levels and/or longer cracks, which would most likely be detected, the crack growth rate at 1000 °F diminishes to a point where it is only four to five times greater than at room temperature for a $\Delta K = 35,000$ psi $\sqrt{\text{in.}}$

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

1. FFTF Quarterly Technical Report June, July, August, 1969.
BNWL-1174, Battelle-Northwest, Richland, Washington,
October 1969.

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

FIGURE 7.5. Fatigue Crack Growth Rate Versus Stress Intensity Factor Range (ΔK) for AISI 304 SS (Annealed) at Room Temperature and 1000 °F

5. High Strain Rate Effects*

J. M. Steichen (A00)

The results of high strain rate tests, presented in Figures 7.6, 7.7, and 7.8, indicate that the ultimate strength of 304 SS is the mechanical property which is most sensitive to strain rate at elevated temperatures. Little effect is observed (Figure 7.6) in the ultimate strength at a test temperature of 600 °F, but significant effects are observed at temperatures of 1000, 1200, and 1400 °F. It is interesting to note that the ultimate strength appears to decrease at these elevated temperatures as the strain rate increases to 10 sec^{-1} .

FIGURE 7.6. The Effect of Strain Rate and Test Temperature on the Strength of 304 SS

* Reported by J. C. Tobin (960)

FIGURE 7.7. The Effect of Strain Rate and Test Temperature on the Ductility of 304 SS

FIGURE 7.8. The Effect of Strain Rate and Test Temperature on the Reduction of Area of 304 SS

FIGURE 7.9. The Effect of Strain Rate on the Strength of 304 SS at R.T. and 1000 °F

The effect of strain rate on the 0.2% yield strength appears to follow a pattern similar to that of the ultimate strength, but the effect is less pronounced.

Total elongation (Figure 7.7) does not appear to be greatly influenced by strain rate but the results indicate that ductility slightly increases with strain rate up to a test temperature of about 800 °F. At a test temperature of 1400 °F, the total elongation appears to decrease with increasing strain rate. The uniform elongation also appears to decrease as strain rate increases at temperatures up to 1400 °F. At 1400 °F, uniform elongation for the three strain rates is essentially the same.

Values obtained for the reduction of area (Figure 7.8) indicate that the greatest reduction in area is observed at a strain rate of 1.0 in./in./sec at all test temperatures except for 1000 °F. The difference in values obtained at 1000 °F, however, are not considered significant.

If room temperature strength values are plotted against strain rate (Figure 7.9), the ultimate and 0.2% yield strength increase as strain rate increases. The ultimate strength appears to increase about 1500 psi for each order of magnitude increase in strain rate, while the yield strength appears to increase about 3000 psi for each order of magnitude increase in strain rate. At 1000 °F and at rates to 0.1 sec^{-1} , the ultimate strength increases. At rates above 0.1 sec^{-1} at this temperature the ultimate strength decreases significantly. The yield strength at 1000 °F decreases only slightly at the higher rates.

6. Chemical Compatibility of Control Rod Materials*

L. R. Bunnell (A00)

a. B₄C Sodium Compatibility and Component Transport

Boron carbide pellets of three densities (75, 85, and 96% TD) were placed in compatibility - transport capsules filled with reactor-grade sodium containing 3 ± 2 ppm oxygen and heated for 3450 hr at 300 and 500 °C. Previous short-term (100 hr) results showed low-density (75% TD) pellets performing well, with high-density pellets exhibiting peripheral cracking. The trend was reversed in the long-term experiments in that low-density pellets were severely attacked and cracked during exposure. Low-density pellets had lost pieces amounting to about 10% of their mass; weight losses of high-density pellets were negligible.

* Reported by J. C. Tobin (960)

Photomicrographs (Figures 7.10 through 7.12) show the B_4C pellets exposed at 500 °C. Results at 300 °C were similar but not quite so extensive. The 75% TD pellet contained large localized density fluctuations and the attack seems to have been worst in these locations. Note also that the high-density pellets were cracked at outer surfaces. This effect had been observed to a lesser extent on pellets tested 100 hr. To evaluate the possibility that stresses introduced during fabrication cause this cracking, 85 and 96% TD pellets were heated to 2000 °C and slowly cooled. The pellets were examined after exposure to sodium for 1000 hr at 500 °C and were found to be essentially free from cracks (Figures 7.13 and 7.14).

FIGURE 7.10. 75% TD B_4C Pellet Exposed to Sodium at 500 °C for 3450 hr. The edge of the sample was eliminated by cracking. Note large porous regions, where B_4C was apparently removed by sodium. 80X

FIGURE 7.11. 85% TD B₄C Pellet Exposed to Sodium at 500 °C for 3450 hr. Note cracks near edge. 80X

FIGURE 7.12. 96% TD B₄C Pellet Exposed to Sodium at 500 °C for 3450 hr. Note cracks near edge, generally more severe than 85% pellet. 80X

FIGURE 7.13. 85% TD B₄C Pellet, Annealed at 2000 °C, Exposed to Sodium for 1000 hr at 500 °C. The one crack visible was the only one seen. Note magnification change (160X) when comparing with Figure 7.11.

FIGURE 7.14. 96% TD B₄C Pellet Annealed at 2000 °C, Exposed to Sodium for 1000 hr, 500 °C. Only two or three cracks were visible in this section. 160X

FIGURE 7.11. 85% TD B₄C Pellet Exposed to Sodium at 500 °C for 3450 hr. Note cracks near edge. 80X

FIGURE 7.12. 96% TD B₄C Pellet Exposed to Sodium at 500 °C for 3450 hr. Note cracks near edge, generally more severe than 85% pellet. 80X

FIGURE 7.13. 85% TD B₄C Pellet, Annealed at 2000 °C, Exposed to Sodium for 1000 hr at 500 °C. The one crack visible was the only one seen. Note magnification change (160X) when comparing with Figure 7.11.

FIGURE 7.14. 96% TD B₄C Pellet Annealed at 2000 °C, Exposed to Sodium for 1000 hr, 500 °C. Only two or three cracks were visible in this section. 160X

The 316 SS from the long-term exposure capsules was examined metallographically and found to be free from carburization (Figure 7.15). Spark-source mass spectroscopy produced the B and C analyses shown in Table 7.IV.

FIGURE 7.15. Type 316 SS from Compatibility-Transport Capsule, Exposed 3450 hr at 500 °C, As-Etched. 250X. Small pits due to etchant used.

TABLE 7.IV. B and C Analyses of 316 SS.

<u>Condition</u>	<u>B, ppm</u>	<u>C, ppm</u>
300 °C, 75% TD pellet, 3450 hr	11.7	708
300 °C, 85% TD pellet, 3450 hr	10.5	678
300 °C, 96% TD pellet, 3450 hr	3.3	627
500 °C, 75% TD pellet, 3450 hr	27.6	861
500 °C, 85% TD pellet, 3450 hr	7.2	240
500 °C, 96% TD pellet, 3450 hr	17.1	654
Standard	3.3	220

The factor-of-three accuracies claimed for the method put all of the data in the same scatter band, but there seems an upward trend in material transfer as a function of decreasing density.

b. Dissolution - Deposition Transport of B_4C

Since the capsule work performed thus far has involved isothermal tests, a simple experiment was performed to assess the possibility of temperature - gradient boron and/or carbon transport. A capsule was constructed in which a B_4C pellet (75% TD for maximum surface area) could be confined to one end with a retainer and the capsule half-filled with sodium. The following cycle was used: heating to 500 °C, inversion, cooling to 150 °C, inversion, and heating to 500 °C. This process was repeated 20 times to give a total of 551 hr at 500 °C and 82 hr at 150 °C. Examination of the B_4C revealed negligible weight loss (0.125%), with no deposited layer in the cooler portion of the capsule. The above experiment, although simple, leads to the inference that B and/or C transport from B_4C to, and deposition within, cold legs of a reactor loop would not be of major importance.

c. Examination of Irradiated B_4C Pellets

Samples of B_4C pellets irradiated in a Hanford reactor to a burnup of about 4.5% were recently examined. Most of the pellets were removed from their capsules in the Radio-metallurgy facility; pellets not used for metallography are being retained in the inert atmosphere facility to prevent hydration and loss of lithium or lithium compounds which may be present. Samples from the pellets were successfully prepared by using nonaqueous polishing lubricants and were compared with the previously characterized unirradiated counterparts described above. Observations are tabulated in Table 7.V. An attempt was made to characterize a high and low temperature pellet of each density; the series is complete except for a high temperature 65% TD pellet (the capsule leaked). Figures 16a, 17a, and 18a show representative pellets

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

Sample Number	% TD	B ¹⁰ Burnup, %	Temperature, °F	Observations
1-15	99	4.71	1130	Transverse section shows circumferential cracks extending inward a maximum of ~0.016 in. The crack separates burned-out material with almost complete grain boundary separation from material with as-fabricated appearance. No evidence of lithium-bearing compounds was visible.
1-13	99	4.67	865	Same comments as above, but a family of concentric cracks exists, with some transgranular cracking. Same crack depth.
1-12	90	4.64	1120	Transverse section shows cracks, both circumferential and radial, extending inward to a maximum distance of 0.071 in. No sharp delineation between surface burnout regions and core regions was visible. No cracking between grains was seen. No evidence of lithium-bearing compounds.
1-4	90	3.84	725	Same comments as above, circumferential cracks extending inward a maximum of 0.045 in. High magnification photos showed no connection between grains, which suggests that this is a nontypical sample, possibly hot-pressed at an unusually low temperature.
1-10	65	4.4	865	Radial and circumferential cracks extend inward about 0.073 in. maximum. No clear delineation showing surface burnout. No visible lithium-bearing compounds.

(a) Transverse Macrophotograph of 96% TD B_4C Pellet Irradiated to $\sim 4.5\%$ Burnup at $610^\circ C$. Note large spalling-type cracks. 13.5X.

(b) Micrograph Spanning a Spalling Crack, Supposedly Separating Burned-Out B_4C from Essentially Virgin Material. Note the extensive grain boundary cracking in the upper zone. 80X.

FIGURE 7.16. 96% TD B_4C Pellet with Spalling

(a) Transverse Macrophotograph of 90% TD B_4C Pellet Irradiated to $\sim 4.5\%$ Burnup at $610^\circ C$. Note combination of radial and circumferential cracking. 13.5X.

(b) Micrograph Showing an Area of the 90% TD Pellet Analogous to Figure 7.16b. Note that no clear delineation between burned-out and virgin material is visible. 80X.

FIGURE 7.17. 90% TD B_4C Pellet

(a) Transverse Macrophotograph of 65% TD B_4C Pellet Irradiated to $\sim 4.5\%$ Burnup at $460^\circ C$. The gray area in the center is potting compound filling a large area which dropped away. Note radial and circumferential cracks. 13.5X.

(b) Photomicrograph Showing an Area of the 65% TD Pellet Analogous to Figure 7.16b. As with Figure 11b, note lack of delineation. 80X.

FIGURE 7.18. 65% TD B_4C Pellet

at low magnification; Figures 16b, 17b, and 18b show an area approximately scanning burned-out and nonburned-out regions at a higher magnification.

The damage in the high-density pellets is concentrated near the pellet surfaces due to self-shielding effects. The behavior of the less-dense pellets is unexplained as yet. X-ray diffraction performed on scrapings from the high-density pellets yielded no peaks except B_4C , which may have resulted from the large amount of grain boundary cracking and resultant high volume of unaffected B_4C in these samples. Thus, fragmentation during sampling could introduce virgin B_4C into the X-ray samples.

7. Radiation Effects on Absorbing Materials for Control Rods*
A. L. Pitner (A00)

The first subassembly of boron carbide being irradiated in a Hanford reactor was discharged after an estimated 5% B^{10} burnup and disassembled. Samples were in the form of 60 and 80% TD powder, and 65, 90 and 99% TD pellets. Irradiation temperatures ranged from 550 to 1130 °F. In general, a high percentage of gas release was noted, and pellet degradation was very evident, particularly in the 99% TD samples.

The subassembly was composed of fifteen capsules, each capsule containing a particular sample form. Five of the capsules were thermocoupled, and the remaining capsule temperatures were inferred from these readings. Table 7.VI summarizes the loading of the subassembly and the irradiation conditions.

The capsules were punctured and the internal gas pressure was measured after irradiation. The table also lists the percentage of helium gas released from the $B^{10} (n, \alpha) Li^7$ reactions assuming 3% B^{10} burnup. Burnup analysis is not yet complete,

* Reported by J. C. Tobin (960)

TABLE 7.VI. Summary of Loading and Conditions

<u>Sample</u>	<u>Form, % TD</u>	<u>Temperature, °F</u>	<u>Gas Release*, %</u>
1-1	60-powder	550	24.2
1-6	60-powder	700	40.0
1-7	80-powder	600	21.4
1-8	80-powder	605	
1-2	80-powder	655	35.7
1-9	80-powder	715	37.3
1-10	65-pellet	865	88.5
1-3	65-pellet	Leaker	
1-4	90-pellet	725	35.5
1-11	90-pellet	860	
1-12	90-pellet	1120	43.0
1-13	99-pellet	865	18.1
1-14	99-pellet	870	
1-5	99-pellet	1035	27.1
1-15	99-pellet	1130	

* Assumes 3% B¹⁰ Burnup.

however, and it must be emphasized that the gas release values quoted are very dependent on the burnup level. In any case, the relative values for the different sample forms should give an indication of comparative sample behavior. The results appear to be consistent within themselves. Gas release for similar materials generally increases with irradiation temperature. The gas release in the pellets is inversely proportional to the sample density, but apparently independent of density in the powders. The 88.5% gas release in the 65% TD pellet is unexpectedly high. Unfortunately, the only other capsule containing a 65% TD pellet leaked water during irradiation, and consequently, no check on the high gas-release figure could be made.

Although a 0.010 in. gap was allowed for expansion of the 0.240 in. diam pellet during irradiation, some of the pellets were stuck in their holders when opened, and had to be forced out. This was particularly true for the 99% TD pellets. All pellets showed considerable degradation from the irradiation. Corners were rounded, cracks were prominent, and the pellets were mostly broken into smaller particles. The surface of the 99% TD pellets was quite powdery, while the lower density pellet's surfaces chipped easily. The three pellets in Capsule 1-3, which leaked water during irradiation, were fused together, and the surface appeared very eroded.

The densities of several of the particles were measured by immersion in distilled water, and the results are shown in Table 7.VII. After irradiation, the pellets all showed apparent densities near the theoretical value (2.52 g/cm^3), indicating that the porosity and cracks generated during irradiation are easily accessible to water.

TABLE 7.VII. Pre- and Post-Irradiation Density Results

<u>Sample</u>	<u>% TD</u>	<u>Preirradiation Density, g/cm^3</u>	<u>Postirradiation Density, g/cm^3</u>
1-10	65	1.700	2.482
1-4	90	2.006	2.391
1-12	90	2.000	2.303
1-5	95	2.440	2.498
1-13	95	2.520	2.466
1-14	95	2.530	2.490
1-15	95	2.530	2.480

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CHAPTER VIII. FUELS RECYCLEA. FUEL TECHNOLOGY1. Analytical Chemistry

B. R. Hayward (711)

Results from the analytical chemistry round robin Phase II were received from AI, GE, B&W, UNC, NUMEC and BNW. LASL has completed their work and the report will be issued as soon as possible. The data from GE and NUMEC not available last quarter were received late this quarter. All data are now being compiled and evaluated at BNW and LASL. A cursory review of the results indicate wide scatter exists in some of the determinations. A statistical review and evaluation of methods will be required to determine the significance and extent of variations in precision and accuracy of the analytical results. It is apparent that the analytical program should be continued through both the prequal and fabrication rounds. It will be a key factor to the success of the fuel pin fabrication effort.

2. Pellet Homogeneity

Excellent progress has been made in the previously weak area of fuel pellet homogeneity measurement. Both BNW and GE made major contributions. The latter completed a small task initiated last quarter. The report was issued this quarter.⁽¹⁾ The homogeneity was defined in a quantitative manner and methods were described that could give a quantitative assessment of the degree of homogeneity. The track etch, alpha autoradiograph technique offers the most promise. The basic GE scheme uses an original chemical assay as a primary standard followed by an electronprobe as a secondary standard and finally an autoradiograph or equivalent as a working standard. The Battelle-Northwest work was primarily covered with the development of a

1. *NEDC-12026 Measuring Pu Homogeneity in Fast Reactor Fuels* by E. A. Aitken, et al.

mathematical model* and further electron microprobe work on Pu concentration zones. (see details under paragraph 3 below). The overall results show that fabrication of pellets to meet the homogeneity specification is easily attained. With the homogeneity definition essentially established, the current activities are on the development of the working standard. Additional homogeneity work will be required to define and develop methods for the determination of few large zones of Pu. Anticipated occurrence of this type of inhomogeneity is statistically very low and would probably require a 100% inspection method to detect.

3. Homogeneity Measurement

D. A. Stranik (740)

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

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

* A mathematical model was developed which related fuel pellet characteristics to a figure of merit (M). This describes the homogeneity in terms of the uniformity of energy deposition under transient conditions in the UO_2 and represents the limiting thermal response of $\text{UO}_2 + \text{PuO}_2$. A mean value of M of 0.8 was established as meeting FTR requirements.

that point as compared to the deposition rate which would exist in a homogeneous fuel of average composition. Since the model used in the development of the figure of merit does not allow for diffusion of energy, the figure of merit represents the limiting thermal response of mixed oxide fuel under transient conditions. The model defines a term (D_x) as the product of the calculated local heat generation rate and local UO_2 concentration. A figure of merit (M) is obtained by dividing D_x by an equivalent term $\overline{D}_x$ which is calculated for a theoretically homogeneous fuel.

$$D_x \text{ is defined as: } D_x = K_1 \times E_x \times C_{UO_2} \quad (1)$$

where

E_x = Energy deposition rate at point X.

The rate of temperature rise at point X is assumed to be directly proportional to E_x ; i.e., the heat capacity of the fuel is constant.

Also,

$$D_x = \left(0.026 + 0.948 \overline{C}_{PuO_2_x} \right) \left(1 - C_{PuO_2_x} \right) \quad (2)$$

where

$\overline{C}_{PuO_2_x}$ = average PuO_2 content along a diameter of a 20 μ sphere centered at point x.

Thus for homogeneous fuel,

$$\overline{D}_x = \left(0.026 + 0.0948 \overline{C}_{PuO_2} \right) \left(1 - \overline{C}_{PuO_2} \right) \quad (3)$$

where $\overline{C}_{PuO_2}$ is the PuO_2 content of the homogeneous mixed oxide fuel.

The relationship between $\overline{C}_{Pu}$ and $\overline{D}_x$ is shown in Figure 8.1

FIGURE 8.1. Variation of $\bar{D}_x$ with Plutonium Concentration in $\text{UO}_2\text{-PuO}_2$ Fuel Pellets

By definition,

$$M_x = \frac{D_x}{\overline{D_x}} \quad (4)$$

where M is the local figure of merit at point x

and

$$\overline{M} = \frac{1}{N} \sum_N \frac{D_x}{\overline{D_x}} \quad (5)$$

where $\overline{M}$ is the average figure of merit for N determinations.

The value of $\overline{C_{PuO_2}_x}$ calculated for any point x on the microprobe trace will be conservative in terms of using Equation (2) to establish a distribution curve for D_x values since the "observed" value of $\overline{C_{PuO_2}_x}$ will statistically vary about the true mean value of $\overline{C_{PuO_2}_x}$ for a hypothetical 20 μ diam sphere centered at point x .

This statistical variation will result in an increased spread in the calculated M_x values used to establish a distribution curve.

FIGURE 8.2. Hypothetical Distribution Curve for Figure of Merit M_x in a Heterogeneous UO_2 - PuO_2 Fuel Pellet.

Figure 8.2 illustrates the type of M_x distribution curve to be expected from a heterogeneous pellet. Two properties of the curve can be used to describe homogeneity requirements: (1) the value of $\bar{M}$, and (2) the fraction of M_x values which falls within a specified limit about $\bar{M}$.

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

The microprobe data from three $(\text{Pu}_{0.25}\text{U}_{0.75})\text{O}_2$ fuel pellets with three different degrees of inhomogeneity were analyzed according to the homogeneity model given above. The microprobe measurements were obtained using a $1\ \mu$ beam diameter and a scanning speed sufficiently slow to eliminate error due to instrument response time. The narrow beam was used to reduce the averaging effect of larger size beams and to better resolve concentration gradients. The instrument was calibrated by using mounted and polished microspheres of pure PuO_2 . Pellet 8131 was a typical irradiation test pellet used in PNL-3 fuel pins. Pellet 956 was a pellet with less ballmilling of the powder (which was used to break in new processing equipment). Pellet 8235 was a pellet in which the powder was "V" blended only. The microprobe traces were analyzed for the average PuO_2 content (C_{PuO_2}) along $20\ \mu$ increments centered at $3.3\ \mu$ intervals, and the actual PuO_2 content ($C_{\text{PuO}_2}_x$) at the center of each $20\ \mu$ increment.

The pellets were identified as 8235, 956, and 8131, and ranged from least homogeneous to most homogeneous, respectively. Analyses of the microprobe traces yielded the following data for each pellet:

<u>Pellet Identification</u>	<u>$\overline{C}_{PuO_2x}$</u>	<u>C_{PuO_2x}</u>	<u>D_x</u>	<u>M_x</u>	
<u>8235</u>	n = 488				
Mean	0.2978	0.3007	0.2027	0.9364	$\overline{M} = 0.9364$
Standard Dev.	0.1132	0.1370	0.0458	0.2117	$\sigma_{\overline{M}} = 0.0095$
<u>956</u>	n = 507				
Mean	0.2552	0.2562	0.1955	0.9783	$\overline{M} = 0.9783$
Standard Dev.	0.0572	0.0908	0.0297	0.1486	$\sigma_{\overline{M}} = 0.0066$
<u>8131</u>	n = 340				
Mean	0.2636	0.2675	0.2010	0.9890	$\overline{M} = 0.9890$
Standard Dev.	0.0321	0.0475	0.0171	0.0841	$\sigma_{\overline{M}} = 0.0046$

$\overline{M}$ for the pellets are all greater than 0.80. The M_x frequency distribution histograms for each pellet are shown in Figures 8.3, 8.4 and 8.5.

Confidence intervals can be calculated for each $\overline{M}$, and the end point of the interval (confidence limit) will be $\overline{M} - \sigma_{\overline{M}} k$

where $\sigma_{\overline{M}} = \frac{\sigma_{M_x}}{\sqrt{n}}$ and k for the 99% confidence interval is 2.586

when n = 500 and 2.590 and when n = 340.

The lower 99% confidence limit for each of the three pellets is:

$$\begin{array}{rcl}
 \frac{8235}{0.9364 - 0.025} & = & \underline{0.911} \\
 \frac{956}{0.9783 - 0.017} & = & \underline{0.961} \\
 \frac{8131}{0.9890 - 0.012} & = & \underline{0.977}
 \end{array}$$

FIGURE 8.3. Figure of Merit (M_x) Distribution for Pellet 8131

FIGURE 8.4. Figure of Merit (M_x) Distribution for Pellet 956

FIGURE 8.5. Figure of Merit (M_x) Distribution for Pellet 8235

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

4. Fuel Manufacturing Development Program

G. A. Last (710), J. F. Williams and F. M. Smith (712)

During May of this year three contracts were placed with industry for prequalification as LMFBR fuel suppliers (Babcock and Wilcox Co., Nuclear Materials & Equipment Corp., and United Nuclear Corp.). Each contractor is fabricating 125 prototypic FTR fuel pins plus additional pellets equivalent to 75 additional pins. Complete documentation of process and quality assurance procedures was required of each contractor for BNW review and comment prior to use by the contractor.

All three contractors completed the documentation requirements and started production of the 25 wt% PuO_2 mixed oxide fuel. Two suppliers completed fabrication of initial batches of sintered fuel pellets that meet specifications.

Nine finished fuel pins were produced by one supplier. No significant plutonium contamination was found in the final end cap welding operation and the pins are undergoing final inspection.

Residues from draw lubricants, burrs, and slight deformations incurred during machining were cause for the rejection of a number of the fuel pin tubes supplied to the contractors. The rejected tubing was replaced by BNW.

All fabrication and quality assurance work by the three fuel suppliers is expected to be completed during the next quarter. Receiver inspection and product evaluation will begin at BNW as soon as finished pins and fuel pellets are received.

B. CLADDING TECHNOLOGY

1. NDT Standards for Cladding

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

Inquiries were submitted to seven fabricators of NDT notch standards to provide standards to our specification. Of these seven, four vendors responded and orders for standards were placed with them. Each standard was to contain 12 notches covering a range of three depths: 0.0005, 0.0010 and 0.0015 in. deep. The notches were to be in two directions: circumferential and axially on both the inside and outside surface. (Figure 8.6) The vendors were each provided with a 15 in. length of cladding that was virtually free of defects, and each was required to provide replicas of all notches. These notches were sectioned at midlength and photographed at 250X to show the shape of the notch. One requirement of the specification was that the corners of the notches should be as square as possible within the limitations of the vendor's capabilities. The photographs presented in Figure 8.7 graphically illustrate the degree to which each vendor met this requirement. The four vendors have been designated as Vendor A, B, C, and D. Vendor B has produced by far the best and most consistent notches from the standpoint of shape. The next evaluation will be to subject the notches to ultrasonic and eddy current testing and compare them on the basis of response to the electronic signals. When all tests are completed a full topical report will be issued covering the study.

FIGURE 8.6. NDT Standard Used for Ultrasonic and Eddy Current Testing of FFTF Fuel Cladding

Neg 0693424-8

VENDOR A

VENDOR B

VENDOR C

VENDOR D

FIGURE 8.7. Comparison of Typical 0.0015 in. Deep Notch from Each of Four Vendors Produced on NDT Artificial Notch Standards

2. Clad Vendor Development Program

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

A summary of the fabrication processes to be used by the different vendors is presented in Figure 8.8.

Carpenter Technology Corporation, Sandvik Jern Verkes, Superior Tube Company and Wolverine Tube Company are engaged in the tubing development program.

3. LMFBR Fuel and Cladding Information Center

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

The data-input forms for the cladding fabrication section were completed and copies distributed to probable users. Data

Neg 0693994-20

FIGURE 8.8. Manufacturing Processes Used in Vendor Development Program

collection and storing Lots E, F, G, and H of austenitic stainless steel purchased by Battelle-Northwest are continuing. Storage of mechanical properties data, which are more accessible to information center personnel, is proceeding more rapidly than is the melting and primary fabrication data.

A rough draft of the data-input forms for the interim pin file was completed and copies made to implement data collection. The format for the storage program is complete and will provide for data storage directly to magnetic tape. Initially, data will be collected and stored for several BNW fuel pin experiments and a "debugging run" of the program made. At the same time the data-input forms will be finalized, approved and forwarded to the printer by forms control. The finalized forms will be distributed in about four weeks.

Programs and formats have been prepared to create a creep data file and a mechanical properties evaluation file. The programs have been edited and "debugging runs" made. Generalized plotting and statistical data plotting programs were provided for use with all data files of the system. These have greatly enhanced the data analysis capabilities of the information center.

Work was initiated on the listing of data fields for the fuel fabrication file, starting with the appropriate fields from the interim pin file. The additional fields to be added will provide data storage capabilities for the detailed characterization and description of reactor fuels. Considerable interfacing with onsite, offsite experimenters and fuel fabricators is required to properly meet their data storage needs.

The mechanical property files were updated with addition of data from 168 tensile tests and 54 creep tests on austenitic stainless steel. Their respective totals are 2,365 and 54 tests on file.

Data retrieval and plotting activities consisted of a tabular printout of the tensile properties for Types 304, 316 and 348 SS; preparation of 47 engineering-stress versus engineering-strain graphs for 304 and 316 SS; and preparation of eight creep-strain versus time-curves from the creep data on Type 316.

C. FUEL PIN TECHNOLOGY

1. Fuel Pin End Closure Development

R. M. Crawford (733)

To date, approximately 2,100 fuel pin end closures using the reference FTR weld design (Figure 8.9) have been welded and radiographed. About 4% of these welds were examined metallographically (Figure 8.10).

Metallographic examination of 81 welds indicated no evidence of any condition that would cause the weld to be rejected. Radiographs of all the pin closure welds for irradiation in EBR-II were examined and accepted by the quality assurance group at EBR-II.

Approximately 50% of the welds produced to date had an as-welded build-up across the diameter of up to 0.001 in., and 45% of the welds had a build-up on the diameter of between 0.001 to 0.0025 in. Approximately 5% of the welds had build-up exceeding 0.0025 in. While 37 of the 2,100 welds made were rejected for incomplete (less than 100% penetration) penetration based on radiography, none of the 81 welds submitted for metallography showed incomplete penetration.

FIGURE 8.9. Reference FFTF Fuel Pin Closure Weld

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

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

Total Welds - 376

Metallography - 19

Weld Rejects - 6 unable to close, 5 incomplete penetration, 20
Pu contaminated greater than 500 d/m

Total Rejects - 31

CCTL Test Pins - 0.250 in. OD - 304 SS (annealed)

Total Welds - 677

Metallography - 5

Weld Rejects - 2 unable to close, 4 incomplete penetration

Total Rejects - 6

469-1295A S-222
End Closure Weld 15X

469-1295B S-222
End Closure Weld 75X

FIGURE 8.10. Typical Fuel Pin Closure Weld 0.230 OD 316 SS,
20% Cold-Worked

Irradiation Test Pins for PNL-7, 8 - 0.250 OD - 316 SS (annealed)

Total Welds - 216
 Metallography - 12
 Weld Rejects - 0

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

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

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

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

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

Emphasis in the welding development program in the immediate future will be in two major areas: (1) The effect of end cap machining tolerances and fit between end cap and tube on the completed weld and (2) an intensive physical testing program on the completed welds. This testing program will be similar to the program currently in progress for the fuel pin cladding and will include tensile tests, stress rupture tests, and burst tests.

2. Acoustic Emission Weld Monitor

R. M. Crawford (733), J. C. Crowe, and W. D. Jolly (MOO)

Acoustic emission denotes the energy released in the form of elastic waves by fracture or plastic deformation within a solid material. Acoustic emission detection has been used to predict failure in pressure vessels and piping, to determine the onset of fatigue cracking, to detect stress corrosion cracking, and to detect cracking, porosity and inclusions in welds.

The acoustic emission technique for weld monitoring has been demonstrated on materials such as stainless steel, steel, aluminum and titanium using electron beam, gas-tungsten arc, or submerged arc welding techniques. Since this technique is not inhibited by physical size or weld geometry, it is ideally suited to the monitoring of FFTF fuel pin closure welds.

The system used to monitor the fuel pin welds is shown schematically in Figure 8.11. The weld monitor instrument integrates the amplified sensor output over a short time interval and presents this sum to the chart recorded at the end of each time interval. This approach was necessary to overcome the random nature of the acoustic emission pulses. Repetition rate of acoustic emission pulses varies from one every few minutes to about twenty per second while amplitude varies from a few microvolts to a hundred millivolts. The emission rates output of the monitor instrument is in units of $\text{volt-sec/sec} = \text{volts}$ and is proportional to the square root of the average acoustic power received by the sensor. Thus, the emission rate is sensitive to variations of the acoustic energy released by cracking, porosity, and inclusions in the weld metal as the joint cools.

Earlier experiments with artificially induced gross cracking in fuel pin welds indicated a difference of orders of magnitude between defective and normal welds, but naturally

FIGURE 8.11. Block Diagram of the Acoustic Emission System
Used to Monitor FFTF Fuel Pin Closure Welds

occurring defects such as microcracks are not expected to cause such a large change in emission rate over normal (defect free) welds.

The present FFTF fuel pin weld monitoring program is aimed at determining the range of variation of acoustic emission rate for normal welds and the reliability of distinguishing between normal and defective welds on the basis of recorded acoustic emission data.

During this quarter, a group of 55 fuel pin and closure welds was monitored with the FFTF acoustic emission monitor prototype. Analysis of the acoustic emission recordings revealed five welds which produced relatively large amounts of emission (above 15 V-sec). This represents approximately 10% of the welds examined. Three welds (or approximately 5% of those monitored) recorded a moderate amount of emission (8.0 to 15 V-sec). The remaining 47 (85%) of the welds produced acoustic emission totals below 8.0 V-sec.

All welds were monitored during and after the welding procedure for a total of 75 sec. The emission totals were determined by summing the emission output during the entire monitoring period.

D. FUEL SUBASSEMBLY TECHNOLOGY

1. CCTL Mark II Subassembly Fabrication

R. L. Myers, R. B. Baker, D. E. Blahnik (732)

An FTR Driver Fuel Subassembly is being fabricated for testing in the sodium environment of the CCTL.

a. Lower Grid Assembly

An electron-beam welding technique was developed for fabrication of the layer spacing grid plates. This technique was

then used to weld the round rail to the plates. The weld parameters established for this weld are listed in Table 8.I.

TABLE 8.I. Layer Spacing Grid Plate Weld Parameters

Welding Machine	Hamilton Standard Electron Beam Machine, Model No. EBW(6)52-36-36
Beam Size	0.003 - 0.005 in. (fine spot)
Current, Filament beam	1.525 A 2.0 mA
Voltage	105 kV
Speed	30 in./min
Start	Running, full power
Overshoot	On rail extension
Atmosphere	Vacuum, 5×10^{-5} Torr
Focal Height	5.0 in.
Beam - Plate Angle	20°

A flat was machined 0.005 in. deep on the rails to allow for weld build-up and assembly of pins onto the finished plate. Figure 8.12 shows a cross-section of the grid plate and the relative position of the electron beam, the 0.005 in. flat, the rail, and grid plate.

FIGURE 8.12. Electron Beam Weld Position

A metallographic examination cross-section of a weld in Figure 8.13, shows the weld location and penetration. A weld sample of 0.010 in. deep flats milled on the rails is shown in Figure 8.14. As can be seen, the weld build-up is minimal and the lower end caps fit without any additional preparation after welding. The same is also true when a 0.005 in. deep flat is used.

b. Pin Assembly

Development of a welding procedure for the lower end cap closure welds was completed for the modified design lower end cap. The welding parameters are shown in Table 8.II, the operating procedure for lower end cap welding. Approximately 250 end caps were welded and passed radiographic examination.

A loading procedure was completed, along with the loading of approximately 250 pins. Each pin was loaded, (including the upper reflector rods) and sealed with plastic end bags. The fuel pellets were loaded 6 in. at a time and seated with a seating rod. Fuel column length and weight were recorded.

4C594

FIGURE 8.13. Electron Beam Cross Section

PNL-0694709

FIGURE 8.14. Electron Beam Weld Sample

TABLE 8.II. Welding Procedure

Project:	CCTL Mark II Lower End Caps	
Process:	TIG	
Alloy:	316 SS	
Power Source:	Merrick E.S.R. 300	
Torch:	HWS20 (Welding Head)	Initial Current: 16 A
Cup:	Gas Screen #4 Cup	Upslope Rate: 0
Shielding Gas:	He-Ar	Weld Current: 16 A
Type:	75% - 25%	
Volume:	60 CFH	
Electrode:		Voltage: 14 - 16
Type:	2% Thoriated Tungsten	
Diam:	1/16 in.	
Ext.:	0.250 in.	Weld Time: 14 sec
Shape:	30°	
Filler Alloy :	None	
Cleaning:	Clean end cap with alcohol in ultrasonic cleaner for 15 min. Clean cladding according to attached procedure.	1 Rev Time: 12 sec Down Slope Rate: 5 sec Finish Current: 5 A Stop Delay : 0 Down Slope Time: 5 sec Fixture Speed in./min : 4 Fixture Dial (Ref.): 35 Start Arc: Running Start

CHAPTER IX. PHYSICSA. CORE PHYSICS1. An Analysis of Subcritical Measurements of Reactivity Worth of Peripheral Control Rods in ZPR-3 Assembly 56B

V. O. Uotinen (833)

a. Introduction and Summary

Reactivity worths of control rods, inferred from responses of neutron sensors located in various positions with respect to the rods, can be expected to be different. The magnitude of the variation is, of course, dependent upon the proximity of sensor and rod, and dependent upon the magnitude of the worths of the rod. This has been demonstrated in recent FTR Critical experiments⁽¹⁾ performed in ZPR-3 Assembly 56B. The reactivity worth of four large peripheral B_4C rods, as inferred from detectors in different locations, ranged from -1776 ± 6 lh to -2021 ± 4 lh. Similarly, the worth of four small B_4C rods ranged from -925 ± 5 lh to -1000 ± 2 lh. The measured rod worths inferred from the standard out-of-core ZPR-3 flux detectors were larger in magnitude than those inferred from a detector located in the center of the reactor, by 12% and 3.3%, respectively, for the large and small B_4C rods. Intuitively, one would expect the flux at the center to be the least perturbed, and thus, the central detector is expected to yield the most correct result. However, on the basis of the experiment alone, one cannot say definitely what the measured worths are. Furthermore, if, as suspected, the central detector gives the correct result, this implies that peripheral rod worths, measured using the standard ZPR-3 detectors, are consistently wrong. If this is the case, one should apply correction factors

1. Reactor Development Program Progress Report, April - May, 1969, Argonne National Laboratory, ANL-7577, July 7, 1969.

to peripheral rod worths that have been measured in the past (Assembly 48A) with the standard detectors.

A theoretical analysis is essential so that the measured rod worths could be interpreted meaningfully. The purpose of this report is to describe an analysis that has been performed, using two-dimensional diffusion theory, and to present the significant results and conclusions of the analysis.

The results of these calculations indicate that a detector in the center of the reactor gives correct results, and that the standard ZPR-3 flux detectors can be expected consistently to overestimate the worth of such peripheral rod configurations. Furthermore, it appears that two-dimensional theory calculations can be used to calculate reasonable correction factors, which can be applied to peripheral rod worths that have been inferred from the standard ZPR detectors.

Finally, a rather obvious suggestion arising from the present analysis is that in future subcritical measurements of peripheral rod worths, one should place a detector in the center of the reactor where possible.

b. Measured Reactivity Worths

Experimental Method. The degree of subcriticality $\Delta k_{\text{ex}}(\text{sub})$ of the reactor is given by the relationship

$$\Delta k_{\text{ex}}(\text{sub}) = -Q/CR,$$

where Q is an effective source term, which, when divided by the count rate, CR , yields $\Delta k_{\text{ex}}(\text{sub})$. The source term Q may be inferred using two techniques. The first is a period technique, which involves recording the count rate as a function of time when the reactor is on a positive period, then applying inverse kinetics analysis to the data to obtain Q . The second method is a two-power-level-technique, in which the subcritical count rate is measured at two levels of subcriticality

which differ by a known reactivity amount (e.g., movement of a calibrated control rod). The source constant Q can be evaluated by taking the difference between the Δk in the two cases.

The effective source term is different for each detector, and even for the same detector when it is in different locations, because the source term is proportional to the absolute detection efficiency (counts detected/total events in reactor) of the detector. Once Q has been measured for a given detector in a given location, the degree of subcriticality is obtained by simply dividing Q by the observed count rate.

This simple theory assumes that the effective source and the detection efficiency remain constant as a perturbation is introduced to change the k of the reactor. In practice this is not necessarily a valid assumption. Any change in fuel mass changes the effective source because the spontaneous neutrons from the fuel material constitute the source. Also, any change that alters the composition and/or geometry at the periphery of the core will cause a perturbation which will change the detection efficiencies of detectors. In some cases these perturbations may be negligible, but this is not necessarily so. One needs to consider each measurement carefully, and if necessary, apply calculated correction factors to assure that measured results are interpreted correctly.

Measurements in 56B. Reference 1 (ANL-7577) contains a detailed description of these "peripheral poison control composition worth experiments." The poison control drawers, which contained platelets of B_4C and of Na, are illustrated in Figure 9.1. Homogenized material densities are listed in Table 9.I.

The measurements were performed in the following sequence. First, the reactivity was measured for the base case. Then the reactivity was measured after the poison control material replaced

DRAWER LOADING USED FOR MEASUREMENTS IN THE RADIAL REFLECTOR
FIGURE 9.1. Drawer Loading for Peripheral Poison Drawer

TABLE 9.I. Homogenized Material Densities
(a/b-cm) for Poison Control Drawers

Na	0.008736
B-10	0.007145
B-11	0.02897
C	0.009358
Fe	0.01054
Cr	0.002651
Ni	0.001177
Mn	0.0001192
Si	0.0001391

radial reflector material in four single matrix locations (marked I in Figure 9.2). After this, radial reflector was replaced by poison control material in twelve additional matrix locations (marked II in Figure 9.2). Finally, the reactivity of the base case was again measured. Corrections were made for control rod movement and differences in temperature, so that all cases could be compared on the basis of common reference condition. The following description of the various detectors is from ANL-7577:

S - SAFETY ROD
 C - CONTROL ROD
 T - SOURCE TUBE
 N - NEUTRON LEVEL DETECTOR

I POISON CONTROL I LOCATIONS
 II POISON CONTROL II LOCATIONS

NOTE: ALL DIMENSIONS IN CENTIMETERS

FIGURE 9.2. Detector and Matrix Locations for the Peripheral Poison Control Worth Measurements

"The standard flux detectors are the two current chambers labeled 4a and 4b in Table 2* (see also Figure 2*). These chambers, filled with BF_3 gas at 1 atm, are 7.6 cm in diameter with an active length of 19 cm. The two chambers are connected in parallel, and the current is measured with a high-stability electrometer. The electrometer output is digitized by a voltage-to-frequency converter, and the resultant pulses are summed in a precision EPUT meter to yield a digital count rate proportional to the chamber current.

The three additional detectors were all BF_3 gas-filled proportional counters. Detector No. 1 was 0.9 cm in diameter with an active length of 5.1 cm. Detectors 2 and 3 were 2.5 cm in diameter with active lengths of 15 cm. The data for Detector 1 was taken on a pulse-height analyzer. The locations of the detectors were chosen to evaluate their influence upon the individual count rates as interpreted by the standard subcritical technique. Detector 1 measured the change reaction rate at the core center; hence, a simple analysis based on results from this detector would seem intuitively to yield the most nearly correct answer.

Detector 2 predominantly measured the change in axial leakage; hence, it could be expected to yield too small a change and therefore too small a reactivity worth for the poison control compositions. Detectors 4a and 4b, which measured the change in radial leakage, could be expected to

* Found in Table 9.II and Figure 9.2 this report.

yield too large a change and, therefore, too large a reactivity worth for the poison control composition. The actual worth could be expected to be between those calculated from the change in axial leakage and the change in radial leakage. Detector 3 measured a mixture of many effects with possible localized effects due to the proximity of the poison control composition itself; hence, the results based on this detector should be the most difficult to interpret as to physical meaning."

TABLE 9.II. Detector Locations for Peripheral Boron Rod Worth Experiments

Detector No.	Type ^(a)	Signal	Matrix Element	Geometric Location	
				Radius ^(b) , cm	Distance from Midplane ^(b) , cm
1	PC	Pulse	1-P-16	0	7.6
2	PC	Pulse	2-P-16	0	63
3	PC	Pulse	1-T-12	31-3	63
4a + 4b	IC	Current	External	~110	15

a. All detectors were BF_3 gas-filled chambers.
PC--proportional counter; IC--ion chamber.

b. Distance to center of the active volume.

The measured worths deduced from the various detectors did indeed follow the pattern that was expected. The measured results are presented in Table 9.III.

c. Calculated Reactivity Worths

Description of Calculations. The problem of calculating the change in flux and reaction rate at all detector locations is a three-dimensional problem. However, a two-dimensional analysis, such as has been performed here, allows one to compare the central detector (which was expected to yield the

most correct worths) with the standard ZPR-3 detectors. Such a comparison can shed some light on some of the important questions involved.

TABLE 9.III. Results of the Peripheral Poison Control Composition Worth Experiments (Replacing Radial Reflector Composition)

		Experiment ρ (lh), Core Excess Reactivity				
		Reference 1:	34.2			
		Reference 2:	32.4			
		Average:	33.3			
		Units	1	2	3	4a + 4b
<u>Poison Control, Configuration I</u>						
Effective Source (Q)	ct/sec		$1.692 \pm 1\%$	$38.56 \pm 1\%$	$9.395 \pm 1\%$	$271.0 \pm 1\%$ ^a
Corrected Count Rate ^b	ct/sec		$188.9 \pm 0.5\%$	$4,510 \pm 0.5\%$	$1,101 \pm 0.4\%$	$29,310 \pm 0.2\%$
$k_{\text{eff}}(\text{sub})^c$	% k		-0.896 ± 0.005	-0.855 ± 0.005	-0.903 ± 0.004	-0.925 ± 0.002
ρ (sub) ^c	lh		-939 ± 5	-896 ± 5	-947 ± 4	-971 ± 2
$\Delta\rho$ (Temp)	lh		4.6	4.6	4.6	4.6
Total $\Delta\rho^c$	lh		-968 ± 5	-925 ± 5	-976 ± 4	$-1,000 \pm 2$
<u>Poison Control, Configurations I and II</u>						
Corrected Count Rate ^b	ct/sec		$100.4 \pm 0.6\%$	$2,338 \pm 0.4\%$	$560.9 \pm 0.3\%$	$14,362 \pm 0.2\%$
$k_{\text{eff}}(\text{sub})^c$	% k		-1.685 ± 0.010	-1.649 ± 0.006	-1.675 ± 0.005	-1.887 ± 0.004
ρ (sub) ^c	lh		$-1,781 \pm 11$	$-1,743 \pm 7$	$-1,771 \pm 5$	$-1,998 \pm 4$
$\Delta\rho$ (Temp)	lh		9.9	9.9	9.9	9.9
Total $\Delta\rho^c$	lh		$-1,804 \pm 11$	$-1,766 \pm 7$	$-1,794 \pm 5$	$-2,021 \pm 4$

^aFor 10⁻⁸ scale on the electrometer.

^bUncertainties include statistics and uncertainty in the dead time correction to the count rates.

^cQuoted uncertainties are for count rate only. An additional 1% uncertainty should be added in quadrature to account for the uncertainty in the effective source term measured for the reference configuration.

Two-dimensional diffusion-theory calculations in X-Y geometry were performed using a slightly modified version of the code, 2DB.⁽¹⁾ The modified version, 2DB, was developed for FTR shielding calculations, and one of its features is the ability to directly calculate activations of interest. The reactor model was a quarter (37 × 35 mesh) of the reactor as shown in Figure 9.3. In the model the y-dimension of the ZPR-3 was extended out beyond the physical boundary of 85.746 to 106 cm. This was done by extending the dilute stainless

1. W. W. Little, Jr., and R. W. Hardie. 2DB Users Manual BNWL-831, Battelle-Northwest, Richland, Washington, July, 1968.

FIGURE 9.3. Map of a Quarter of the Reactor, Showing the % Overestimation of the Worth of Four Large B_4C Rods as a Function of Position. The Lines are Lines of "equal-overestimation."

steel "mesh" and was done in order to obtain fluxes and reaction rates at the locations of the ZPR-3 standard flux detectors, which are on top of the reactor. This extension of the mesh has a negligible effect on the calculated reactivity.

The one-dimensional diffusion-theory code, 1DX,⁽¹⁾ was used to generate 13-group cross sections⁽²⁾ from a modified 26-group Russian set. Cross sections for materials in each zone were averaged over the 1DX spectrum appropriate for that zone. Each of the 13 broad groups contained two of the narrow groups.

An eigenvalue calculation and a source calculation were performed for each of three reactor configurations:

- A base case, no peripheral poison
- Poison in four single matrix locations
- Poison in four groups of four matrix locations.

For the source calculations a neutron source uniformly distributed in the fuel zones was included to account for spontaneous fissions in isotopes of plutonium and uranium. The following source strengths (in neutrons/g-sec) were used for $^{238}\text{Pu}/^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/^{242}\text{Pu}/^{238}\text{U}$, respectively: 2420.0/0.022/1012.0/0.022/1804.0/0.014. An α -n reaction in oxygen, which constitutes an additional source of neutrons in plutonium oxide fuels, was not present in these experiments because the plutonium was metallic and clad.

Activations were calculated for absorptions in ^{10}B and for fissions in ^{235}U , using both cross sections averaged over the core spectrum and cross sections averaged over the reflector

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1. R. W. Hardie and W. W. Little, Jr. 1DX, A One Dimensional Diffusion Code for Generating Effective Nuclear Cross Sections, BNWL-954, Battelle-Northwest, Richland, Washington, 1969.
 2. S. L. DeMyer, Unpublished Data. (Private Communication.)

spectrum. The axial buckling was adjusted to $5.2391 \times 10^{-4} \text{ cm}^{-2}$ to give k_{eff} of 0.9995 for the base case.

It might be argued that one could never hope to calculate accurate fluxes at points as far from the core ($\sim 110 \text{ cm}$) as the locations of the standard detectors. However, we are not interested here in the absolute value of the flux at these locations but rather in the ratio of fluxes from two cases. It is maintained that the ratios of fluxes are meaningful even though the absolute values are systematically wrong. The calculated results give credence to this contention.

Calculated Results. Reactivity worths deduced from calculated fluxes and reaction rates at detector locations are presented in Table 9.IV. A ratio R is also given, which is the ratio of the worth obtained from reaction rates to the "correct" worth given by the eigenvalues. In the center of the reactor all three, i.e., total flux, $\sigma_a^{10\text{B}}$, and $\sigma_f^{235\text{U}}$ reaction rates, yield reactivity worths in good agreement with the worths based on calculated eigenvalues. This agreement indicates that the insertion of peripheral control rods introduces a negligible distortion at the center of the reactor.

TABLE 9.IV. Calculated Reactivity Worths Deduced from Calculated Reaction Rates of Various Detectors

Detector Location	Detector	4 Small B ₄ C Rods		4 Large B ₄ C Rods	
		Cal. Worth (Ih)	R*	Cal. Worth (Ih)	R*
Center of Reactor	Reaction Rate in ^{10}B	1005	1.000	1923	1.003
	Fission Rate in ^{235}U	1003	0.998	1915	0.999
	ϕ Total	1002	0.997	1909	0.996
Location of Standard ZPR-3 Detectors	Reaction Rate in ^{10}B	1056	1.051	2255	1.176
	Fission Rate in ^{235}U	1052	1.047	2215	1.155
	ϕ Total	1045	1.040	2153	1.123

* R is the ratio of the worth deduced from reaction rates to the worth as obtained from the calculated eigenvalues.

At the locations of the standard ZPR-3 flux detectors, all three calculated ratios overestimate the change in reactivity. These results indicate that the insertion of peripheral control rods introduces rather large distortions in the flux at the locations of the standard ZPR-3 detectors. This, of course, was what the measured results seemed to indicate also. With regard to the question of how much the reactivity worth is overestimated when deduced from the standard detectors, the calculated results agree fairly well with the measured results. The calculated total flux gives the closest agreement with measurements indicating an overestimation of 13 and 4.3%, respectively, for the large and small B_4C rods, as compared with measured overestimations 12 and 3.3%. This is surprising at first thought, because one would rather expect the calculated ^{10}B reaction rate to give the best agreement with measurement since BF detectors were used. However, the standard BF_3 detectors are surrounded by a hydrogenous material which thermalizes the fast neutrons, thus enhancing the count rate. Therefore, it is not surprising that the total flux would give results that agree most closely with measured results. The amount of overestimation (%) of the worth of four large control rods as a function of detector position is shown in Figure 9.3.

It appears that the calculated ratio of total flux (i.e. base case/case with peripheral control rods) at the standard ZPR-3 detector location, when compared to the same ratio at the center of the reactor, provides a reasonable correction factor which can be applied to measured worths deduced from the standard detectors.

2. Expected Count Rates in the Shutdown FTR

V. O. Uotinen (833)

a. Introduction and Summary

Calculations have been performed to determine what count rates can be expected from in-vessel sensors in the shutdown ($k = 0.9$) FTR.

Subcritical source calculations have been performed using the code 2DBS to obtain reaction rates directly for the subcritical FTR at $k = 0.9$. Thus, no extrapolation to shutdown was necessary. Also, the reactor model contained peripheral control rods. Thus, no correction was necessary for the flux depression caused by the peripheral control rods. Activations were calculated using fission cross sections for ^{235}U that had been averaged over a spectrum characteristic of the shield.

These calculations indicate that the expected count rate at 96 cm, using a commercially available 2 in. ^{235}U fission chamber, is 50 counts/sec when operating in a gamma-field of 10^6 R/hr.

The estimate of 50 counts/sec in a gamma-field of 10^6 R/hr is considered to be a conservative estimate of the expected count rate.

b. Description of Calculations

Calculational Model. Two dimensional diffusion-theory calculations in R-Z geometry were performed using the code 2DBS. The two-zoned reactor model was essentially the same as that used by Sheff⁽¹⁾ in his subcritical calculations, except that in this case the densities recommended by Westinghouse were used. The one-dimensional diffusion-theory code 1DX was used to generate 13-group cross sections (from a modified 26-group Russian set), for materials in the various zones. The calculational method is the same as that being used to analyze current FTR critical experiments. The calculated eigenvalue for this model of the shutdown FTR was 0.908.

Source. A distributed neutron source was inserted in all zones that contained fuel, to account for spontaneous fissions and α -n reactions with oxygen. The major contributor to the

1. J. R. Sheff. FFTF Monthly Informal Technical Progress Report, BNWL-1062, p. 3.17. Battelle-Northwest, Richland, Washington, April, 1969.

source (about 65% of the total) is the spontaneous fission of ^{240}Pu , for which the half-life is known to better than $\pm 10\%$. The yields from α -n reactions are less well known, but the uncertainty in the total source strength, due to uncertainties in neutron yields, is estimated to be ± 10 - 15% . There is an additional uncertainty due to the uncertainty of isotopic concentrations, particularly that of ^{240}Pu which was assumed to be 12% . If this concentration is uncertain by an additional $\pm 6\%$. Thus, one can arrive at an estimate of $\pm 20\%$ for the uncertainty in the total source strength due to spontaneous fission and α -n reactions in the fuel pins.

Count Rate. For a $k_{\text{eff}} = 0.9$, the ^{235}U reaction rate at a radius of 96 cm is calculated to be 1.3×10^5 reactions/sec 10^{24} atoms. For a commercial 2 in. diam fission chamber containing 1.8 g of ^{235}U (2 gm/cm^2) this is equivalent to ~ 600 reactions/second. Assuming a conservation⁽¹⁾ 50% efficiency and 15% absorption loss, we have 250 counts/sec. When one compensates for a gamma-field of 10^6 R/hr, the efficiency of the detector is reduced to 20% of its original value, thus, giving us 50 counts/sec.

Experience in using this calculational method to calculate traverses in FTR critical experiments, has shown that reaction rates are consistently underestimated at large radii. This is a further reason why it is felt that the estimated 50 counts/sec is a conservative estimate of the expected count rate.

Alternate Sensor Locations. According to the calculations, one could gain about a factor of two in count rate by moving the detector in to a radius of ~ 80 cm. To move it in as far as 7 cm would not result in a further gain; in fact, it could result in a loss if the detector is shielded by a peripheral control rod, as indicated by R- θ calculations of this same problem.

1. W. Abson, et al. "The Design Performance and Use of Fission Counters," J. Brit. Nucl. Energy Conf., p. 201, July, 1958.

c. Counting Immediately After Shutdown

Immediately after shutdown the gamma field is expected to be above the proposed 10^6 R/hr discrimination level. Thus, the detector will be indicating counts due to gamma pile-up. However if the gamma-field were as large as 10^7 R/hr (which is at least a factor of two larger than anticipated⁽¹⁾ for saturation gamma intensities) the expected count rate, due to gamma pile-up, is only ~600 counts/sec.⁽²⁾ This gamma-caused count rate will, of course, decay rapidly in the first few hours following shutdown according to the usual sum-of-exponentials decay scheme, leveling off at an intensity dictated largely by the -intensity of CO^{60} (5.26 year half-life).

3. Reactivity Effects of Stainless Steel Swelling

S. L. DeMyer (833)

Type 304 SS is expected to exhibit significant swelling as a result of exposure to neutron flux in the FTR. Differential swelling is expected where there are neutron flux and temperature gradients across the shroud tubes. The magnitude of the bowing of the individual shroud tube and the resultant whole core distortion is a function of the steepness of the gradient, the flux, the temperature, and the diameter of the shroud tube.

For the reference design of the FTR,⁽³⁾ and 30,000 MWd/tonne exposure, the distortion is characterized by a maximum radial contraction at the midplane of 0.29%, or 0.1886 cm on a radius of 65.8686 cm, and a maximum expansion along the central axis of 0.32%, or 0.1473 cm on a half height of 45.7200 cm.

It has been calculated that this distortion will yield a reactivity effect of $+0.34\% \text{ dk} \pm 0.04\% \text{ dk}$. This uncertainty

1. Memo, C. A. Mansius to W. L. Bunch. December 16, 1968.
2. D. C. Thompson, Unpublished Data. (Private Communication.)
3. W. W. Little, Jr., and L. L. Maas. Nuclear Parameters and Parametric Studies for the Fast Test Reactor (FTR), BNWL-1067, Battelle-Northwest, Richland, Washington, June, 1969.

reflects the precision of the convergence but not the limitations of the calculational model. Thirteen energy group cross sections were used in an R-Z geometry, two dimensional diffusion calculation. The computer code 2DB⁽¹⁾ was used.

The cross sections were collapsed from the 26 group revised Russian set⁽²⁾ over zone average spectra in 1DX.⁽³⁾ The atom densities used in the 1DX calculation and the undistorted core calculation are listed in Table 9.V.

TABLE 9.V. FTR III Atom Densities

Inner Core	
239+241 _{Pu}	0.99399998-03
240 _{Pu}	0.84999999-04
235 _U	0.99999999-05
238 _U	0.45500000-02
O	0.11752000-01
Fe	0.13463000-01
Cr	0.32590000-02
Ni	0.14120000-02
Na	0.10249000-01
Outer Core	
239+241 _{Pu}	0.14930000-02
240 _{Pu}	0.17500000-03
235 _U	0.11000000-04

1. W. W. Little, Jr. and P. W. Hardie. 2DB User's Manual - Revision I, BNWL-831, Battelle-Northwest, Richland, Washington, February, 1969.
2. J. V. Nelsen and S. L. DeMyer. Group Constants (Tape U1576) for Analysis of FFTF Critical Experiments, BNWL-1044, Battelle-Northwest, Richland, Washington, March, 1969.
3. R. W. Hardie and W. W. Little, Jr. 1DX, A One-Dimensional Diffusion Code for Generating Effective Nuclear Cross Sections, BNWL-954, Battelle-Northwest, Richland, Washington, February, 1969.

TABLE 9.V. (contd)

Outer Core (contd)	
^{238}U	0.50820000-02
O	0.14925000-01
Fe	0.18880000-01
Cr	0.31280000-02
Ni	0.13550000-02
Na	0.87379999-02
Mo	0.43000000-03
C	0.10530000-02
Radial Reflector	
Fe	0.86189999-02
Cr	0.24190000-02
Ni	0.47647000-01
Na	0.66319999-02
Axial Reflector	
Fe	0.14802000-01
Cr	0.40430000-02
Ni	0.29866000-01
Na	0.92429999-02

Figure 9.4 represents the 2DB calculational model in R-Z geometry of the undistorted core. Zones 1-4, 7-10, 13-16 are the inner core while zones 5, 6, 11, 12, 17 and 18 are the outer core. The calculation contains 37 radial and 31 axial-mesh intervals, in an evenly distributed pattern.

Figure 9.5 represents the calculational model of the distorted core. There are three levels of axial expansion and six levels of radial contraction allowed. As a result, 18 core zones are formed, 12 in the "inner" core, and 6 in the "outer" core. The resultant atom densities in these zones are determined by the ratio of "initial" to "final" volume in these zones which

FIGURE 9.4. 2DB Mesh Structure - Undistorted Core

FIGURE 9.5. 2DB Mesh Structure - Distorted Core

are enumerated in Table 9.VI. The radial expansion data at the core midplane is summarized in Figure 9.6. From data describing the total axial expansion,⁽¹⁾ zone dependent expansions were inferred by making the following assumptions:

- Maximum axial expansion occurs at the core center, which allows the use of 1/2 core, reflected, calculations. The actual maximum occurs above the center as the thermal profile is not symmetric about the core midplane.
- The expansion along the axis can be described by a cosine centered at the midplane. While the axial expansion is not linear, it is not a cosine either, although the function does resemble the latter.
- The decrease in the total axial expansion with increasing radius can be described as a cosine centered at the core center.

The following assumptions were also made:

- It was permissible to use a core with radius equal to the radial distance across six rows of subassemblies lined up flat to flat (65.8688 cm) rather than the average core radius. As a result, calculated core k_{eff} 's will be larger than for the FTR, but the change in k should be little different.
- Initial (unburnt) atom densities were used rather than those representative of 30,000 MWd/tonne.

The calculated increase in k_{eff} produced by distortion in the core is 0.40% dk. The convergence on k in the calculations is two parts in 10^{-4} in k . This will introduce an uncertainty of the order of $\pm 0.03\%$ dk. However, a larger uncertainty is introduced by the use of different mesh structure. Both calculations have the same number of mesh intervals, but the

1. R. Jackson. Unpublished Data. (Private Communication.)

TABLE 9.VI. Zone Dependent Atom Density Modifiers

Zone	Ring	Initial Volume, cm ³	Final Volume, cm ³	Initial/Final
1	1-A	1716.777	1707.379	1.0055
7	1-B		1707.519	1.0054
13	1-C		7710.166	1.0039
2	2-A	13,734.060	13,665.740	1.0050
8	2-B		13,665.999	1.00498
14	2-C		13,684.593	1.0036
3	3-A	27,346.548	27,354.966	0.9997
9	3-B		27,340.201 +0.0208	1.0002
15	3-C		27,326.855 +0.09877	1.0007
4	4-A	40,881.360	41,100.139	0.9947
10	4-B		41,050.147	0.9959
16	4-C		40,954.782	0.9982
5	5-A	55,378.167	54,803.746	1.0105
11	5-B		54,867.781 +0.0472	1.0093
17	5-C		55,066.400 +0.2200	1.0057
6	6-A	68,669.976	68,474.73	1.0028
12	6-B		68,485.491	1.0027
18	6-C		68,541.860	1.0019

distribution is varied. An estimate of this effect on k would be that the k of the distorted calculation is about 0.06% dk too high.⁽¹⁾ The uncertainty on this, however, would be about $\pm 0.02\%$ dk . As a result, the reactivity increase should be stated as $0.40\% dk - 0.06\% dk \pm 0.04\% dk$, or $0.34\% dk \pm 0.04\% dk$.

1. L. D. O'Dell, et al. A Numerical Comparison of Diffusion and Transport (Sn) Codes for Selected Fast Reactor Configurations, BNWL-992. Battelle-Northwest, Richland, Washington, March, 1969.

FIGURE 9.6. Core Distortion, Fuel Assembly Ducts

B. RADIATION AND SHIELDING

1. ZPPR/FTR2 Shielding Experiments

E. T. Boulette (813)

A series of shield experiments on the Zero Power Plutonium Reactor is to provide an evaluation basis for the FTR conceptual design. The ZPPR experiments employed a cylindrical core 3 ft high with an effective radius of 2 ft. A depleted zone with an effective radius of about 5 inches was located along the core central axis. Surrounding the core was a 12 inch nickel reflector extending both radially and axially. A 2 ft thick shield section, with an azimuthal span of about 80°, was located outside the reflector.

Reaction rate distributions for $B(n,\alpha)$, $Mn(n,\gamma)$, $Na(n,\gamma)$, $U^{238}(n,f)$, and $Pu^{239}(n,f)$ were measured radially at three elevations from the core center, and thermoluminescent dosimeters (TLD's) were irradiated to obtain gamma flux distributions. Also, neutron spectra were measured in the radial reflector and in the shield. A comparison of the measured spectrum and the calculated spectrum in the shield is shown in Figure 9.7. The two-dimensional diffusion theory calculation in X-Y geometry used a core axial buckling of $5.82 \times 10^{-4} \text{ cm}^{-2}$ to account for axial leakage. No axial leakage was assumed in the shield. Since the measurements were not on an absolute basis, normalization was based on equal areas below the curves. The overall agreement is quite good; however, for certain energies, such as the sodium resonance at 2.85 keV, the calculation is too coarse. Similar agreement was found for the reflector spectrum.

Further comparisons will be made as the experimental results are reduced and made available.

FIGURE 9.7. ZPPR/FTR2 Shield Spectrum Comparison

2. Shielding Fallback Positions for Concept V-A

E. T. Boulette (813)

Shielding is required to protect the reactor vessel and the core support grid plate from excessive radiation damage and thereby assure its structural integrity throughout the design lifetime of the facility. For a fixed diameter vessel, a number of radial shield concepts were examined to establish potential shielding fallback positions that might be employed. For the vessel 17 ft in diameter, a 19.2 in. thick radial shield composed of 66 vol% SS and 34 vol% sodium is calculated

to be adequate. To meet more restrictive criteria, or to accommodate modifications in the design, consideration can be given to employment of more efficient shielding materials and/or to increasing the shield thickness. Results of calculations for five fallback positions are given in Table 9.VII, where a total fluence criterion is assumed to control.

TABLE 9.VII. Possible Radial Shield Fallback Arrangements

<u>Shield Arrangement</u>	<u>Calculated 20 Year Fluence (nvt)</u>	<u>Relative Lifetime</u>
Concept V-A - 19.2 in., 66 vol% SS, 34 vol% Na	8.3×10^{20}	1
(1) 19.2 in., 75 vol% SS, 25 vol% Na	6.8×10^{20}	1.2
(2) 19.2 in., 37.2 vol% SS, 35 vol% Na, 27.8 vol% B ₄ C	5.3×10^{19}	15.7
(3) 31.2 in., 66 vol% SS, 34 vol% Na	1.8×10^{20}	4.6
(4) 31.2 in., 75 vol% SS, 25 vol% Na	1.3×10^{20}	6.4
(5) 31.2 in., 37.2 vol% SS, 35 vol% Na, 27.8 vol% B ₄ C	2.8×10^{18}	296.4

Similar calculations were made for possible axial shield designs. The basic shield length, as specified in Concept V-A, was 19.6 in. The 1.5 streaming factor used in the calculations was based on preliminary experimental results and are conservative since a more recent estimate is 1.2. The results of the calculations are listed in Table 9.VIII.

3. FTR Lower Shield Penetrations

D. R. Marr (813)

The FTR lower axial shield between the core and the grid support structure will consist of stainless steel cooled by the inlet sodium. In some design concepts, relatively large, sodium-filled regions penetrate the shield, for example, below control rod followers and closed loops. The calculations described below indicate that the local flux below such penetrations would be extremely high.

TABLE 9.VIII. Possible Axial Shield Fallback Arrangements

	Calculated 10 Year Fluence (nvt)	Relative Lifetime
Concept V-A (sodium filled guide tubes)	1.2×10^{21}	1
(1) SS pin array in guide tubes with 1.5 streaming factor	3.6×10^{20}	3.3
(2) SS clad B ₄ C pins in guide tubes with 1.5 streaming factor	4.5×10^{19}	26.7
(3) Elimination of line-of-sight path		
--SS shield	2.4×10^{20}	5.0
--SS clad B ₄ C shield	3.0×10^{19}	40.0
(4) False grid plate (3- to 4-in. SS plate)	6.0×10^{20}	2.0
(5) SS content increase (2 in. effective)	7.5×10^{20}	1.6
(6) Increase in shield length (2 in. B ₄ C pin array) (1.5 streaming factor)	3.0×10^{19}	40.0

For calculational simplicity, the idealized model of the lower shield region consisted of concentric cylindrical regions. The central, sodium-filled region was assumed to be 4.55 in. in diameter. The surrounding shield, containing 56.6 vol% SS and 43.4 vol% sodium, had an outside diameter of 32.3 in., which provided a shield thickness around the sodium of at least three-mean-free paths at all neutron energies. Axial thickness of the shield was 30 in. A constant, isotropic source with the spectral distribution at the FTR core-reflector interface was assumed at the top surface of the cylinder. The radial periphery and lower face were leakage boundaries with no neutron return.

Calculations were made with three codes: the 2-d diffusion code 2DBS, the 2-d discrete ordinates transport code DOT, and the Monte Carlo code BMC. Both the 2DBS and the DOT calculations

were made by using the HOMSET cross sections employed in the FTR conceptual design shielding studies, while the BMC calculation employed ENDF cross sections. The 2DBS results were tightly converged; however, because of computer time involved, convergence in DOT and statistics in BMC were limited. Two calculations were made with each code, one with the sodium-penetration described above and the other an imperforate case with shield material in the central region.

Although the total leakage at the bottom of the shield is somewhat different for each code, the effect of the sodium-filled penetration on this quantity is about the same, as shown in Table 9.IX. Of particular concern is the local flux intensity immediately below the penetration, also shown in Table 9.IX. A comparison of the bottom radial flux profiles indicates that the Monte Carlo total flux ratio is high because of statistical limitations. Differences in boundary treatment, particularly in the diffusion theory calculation, probably influence the flux ratios somewhat.

TABLE 9.IX. Effect of Sodium-Filled Penetration

<u>Computer Code</u>	<u>Ratio - Penetrated Shield/Imperforate Shield</u>	
	<u>Total Bottom Leakage</u>	<u>Total Flux at Bottom Center</u>
2DBS	1.6	5.0
DOT	1.7	4.0
BMC	1.7 $\pm$ 0.2	8.5 $\pm$ 3.6

4. Concept V-A Above-Core Shielding Calculations

E. T. Boulette (813)

Two-dimensional diffusion theory calculations were made with 2DBS on Concept V-A to evaluate the shield design. The geometry assumed in the calculations was R-Z, and the resulting zone configuration is illustrated in Figure 9.8 with the material compositions by zones listed in Table 9.X. All calculations

FIGURE 9.8. Zone Arrangement for Concept 5-A

TABLE 9.X. Material Composition vol% by Zone

Zone No.	Zone Description	Fuel	Ni	SS	Na	Carbon Steel	SiO ₂	B ₄ C	Void
1	Core Inner Zone	25.91	2.27	29.40	38.89	---	---	---	3.53
2	Core Outer Zone	26.26	3.12	28.06	38.98	---	---	---	3.58
3	Peripheral Control Follower	---	27.80	37.20	35.00	---	---	---	---
4	Axial Reflector	---	32.90	23.70	43.40	---	---	---	---
5	Gas Plenum	---	---	23.70	46.30	---	---	---	30.00
6	Peripheral Control Ring	---	---	37.20	35.00	---	---	27.80	---
7	Radial Reflector	---	32.30	33.70	34.00	---	---	---	---
8	Radial Shield	---	---	66.00	34.00	---	---	---	---
9	Core Barrel	---	---	100.00	---	---	---	---	---
10	Assembly Top and Instrument Tree	---	---	17.00	83.00	---	---	---	---
11	Sodium Annulus	---	---	---	100.00	---	---	---	---
12	Vessel-Thermal Liner	---	---	83.00	17.00	---	---	---	---
13	Sodium Pool and Holddown	---	---	5.20	94.80	---	---	---	---
14	Reactor Cover	---	---	17.50	---	39.60	---	---	42.90
15	Blast Shield Zone	---	---	---	---	34.70	54.40	---	10.90
16	Instrument Chaseway	---	---	6.00	---	---	---	---	94.00

were normalized to a two-zone core operating at 400 MW. In-vessel storage of fuel subassemblies was not included because meaningful representation in R-Z geometry is not possible. Previous calculations indicate that fuel stored within the vessel will not significantly distort the total flux distribution. Zones 15 and 16 shown in Figure 9.8 were positioned to simulate the radial blast shield and instrument chaseway respectively, and their locations in Figure 9.8 are not intended to comply with Concept V-A descriptions.

The total neutron flux along the core centerline and along the vessel wall are plotted in Figure 9.9. The difference in the attenuation through the cover is due to the different spectra incident at the cover, and not to the incident flux intensity. The incident spectrum at the centerline is harder than that at the vessel wall because of the cross section minima in iron. The presence of iron in Zone 13 and the lack of iron in Zone 11 account for this behavior. On the basis of current radiation damage limits and personnel protection requirements, the above-core FTR shield design indicated in Concept V-A is conservative. Neglecting penetrations, the imperforate cover thickness could be reduced materially.

5. Shielding of FTR Irradiated Fuel and Fission Gases

C. A. Mansius (813)

We have calculated that about 6 1/2 ft of structural concrete or equivalent will be required to reduce the radiation intensity to less than 0.2 mR/hr for a single FTR irradiated driver fuel subassembly which has decayed for 1 day. For a single fuel pin about 5 ft of concrete will be required for the same level of radiation intensity. The irradiated fuel subassembly and pin were assumed to have operated continuously to goal exposure at 5 MW and 0.025 MW, respectively. The above values will establish the minimum shield requirements for the Fuel Examination Facility.

FIGURE 9.9. Total Flux Distributions Along the Core Centerline and the Vessel Wall

Provisions will also be made in the fuel examination facility to measure the amount of fission product gas generated in FTR fuel. Radiation levels will be high in the vicinity of the gas collection equipment. An estimate was made of the curies of fission product gases in a goal exposure fuel pin as a function of decay time, and the radiation intensity at 2 in. from spherical source volumes containing all of the radioactive iodine, xenon and krypton. The results are presented in Tables 9.XI and 9.XII. About 11 in. of lead or equivalent are required to reduce the radiation intensity to <0.2 mR/hr for 8-day decay gases encapsulated in a spherical source of 100 cm³.

TABLE 9.XI. Fission Product Gases
in a Goal Exposure Fuel
Pin (curies)

	Time After Shutdown, days				
	1	3	8	30	100
Iodine	1900.0	1100.0	500.0	53.0	0.12
Xenon	1600.0	930.0	484.0	29.0	0.04
Krypton	6.0	1.5	1.5	1.5	1.4

TABLE 9.XII. Gamma Intensity from Fission
Product Gases^(a)

	Gamma Intensity at 2 in., R/hr Spherical Source Volume, cm ³		
	100	1,000	10,000
Iodine	50,000.0	25,000.0	10,000.0
Xenon	1,400.0	700.0	280.0
Krypton	0.32	0.16	0.07

a. 8 day decay.

The safe transportation by truck of aqueous solutions containing radioactive fission products must also be provided for. The amount of radioactive fuel which may be in either of two casks and meet the regulations for the safe transport of radioactive materials is shown in Table 9.XIII. The cavity of Cask A was assumed to be 31.2 in. in diameter surrounded by a lead shield 4.44 in. thick. For Cask B, the cavity was taken as 27.75 in. in diameter with an 8 in. lead shield. In each case, 100% of the fission product was assumed uniformly distributed in the aqueous solution. For 1 day decay, over 80% of the radiation intensity is contributed by solid fission products, while for 30 day decay time, solids make up over 95%.

TABLE 9.XIII. Amount of Irradiated Fuel in Cask to Give 10 mR/hr at 100 cm from Cask Surface

Decay Time, days	Cask A		Cask B	
	Fuel Operating Power, MW	Fuel (Total Pins)	Fuel Operating Power, MW	Fuel (Total Pins)
1	0.0083	0.33	2.0	80.0
3	0.0109	0.44	2.63	105.0
10	0.0167	0.67	4.0	160.0
30	0.0476	1.9	11.1	444.0

Note: Fuel pin power assumed to be 0.025 MW with fuel operated to goal exposure (~300 days).

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

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

Critical approach and exponential experiments are in progress to measure material bucklings, extrapolation distances and critical masses as a function of lattice spacing for 25 wt% PuO₂-U(Nat)O₂ fuel pins in water. The data obtained

* Reported by W. W. Little (831).

from these experiments will provide a firm basis for criticality safety for the handling, storage, and shipment of FFTF fuel pins. The composition and dimensions of the fuel pins being used in experiments are given in Table 9.XIV.

TABLE 9.XIV. Average Composition and Dimensions
of 309 Fuel Pin Used in CAF Experiments

Clad OD, in.	0.230
Clad ID, in.	0.200
Clad Length, in.	28.7
Clad Material, in.	316 SS
Fuel diam, in.	0.195
Fuel Length, in.	27.25
Fuel Density, g/cm ³	10.31 $\pm$ 0.22 (93 $\pm$ 2% TD)
Fuel Form	PuO ₂ -UO ₂ - pellets
Fuel Pin Composition	
Fuel, g	138.40 $\pm$ 1.33
PuO ₂ , g	34.87
Pu, g	30.75 $\pm$ 0.30
U, g	91.16 $\pm$ 1.03
O, g	16.52 $\pm$ 0.17
²⁴⁰ Pu Content of Pu, wt%	11.93 $\pm$ 0.06
²³⁵ U Content of U, wt%	
Enrichment, wt% PuO ₂	25.20

The experiments are being performed in a 5 ft diam $\times$ 6 ft high water tank located in the 305 B Building, Critical Approach Facility (CAF). Fuel pins are systematically loaded into vertical, triangular pitch, lattices, which are reflected by water on all sides. Pin spacings are maintained by Lucite locator plates at the top and bottom of the lattices. A typical arrangement of pins is shown in Figure 9.10. Results of the lattices measured and evaluated to date are shown in Table XV. The error limits indicated in the table represent 95% confidence limits.

Neg 0694288-1

FIGURE 9.10. Fuel Pin Lattice in Water (Spacing Between Pins: 0.45 in.)

TABLE 9.XV. Results of Criticality Experiments
with 25 wt% PuO₂ (Nat)O₂ Pins in Water

Critical Approach Experiments

<u>Lattice Pitch, in.</u>	<u>Water to Fuel Volume Ratio</u>	<u>No. Pins in Experi- mental Lattice</u>	<u>No. Pins for Criticality (Extrapolated)</u>
0.779	15.423	148	156.2 ± 0.8
0.900	21.007	141	149.8 ± 0.6

Exponential Experiments

<u>No. Pins in Experi- mental Lattice</u>	<u>b₁₁, cm²</u>	<u>Material Buckling M⁻²</u>	<u>Extrapo- lation Distance, cm</u>
78	-12.72 ± 0.10	160.9 ± 1.3	6.88 ± 0.06
90	-15.05 ± 0.12	160.8 ± 1.4	6.89 ± 0.06
Average		160.9 ± 1.0	6.89 ± 0.04
66	-12.10 ± 0.16	138.3 ± 1.3	6.91 ± 0.09
78	-14.27 ± 0.11	138.6 ± 1.0	6.89 ± 0.06
Average		138.5 ± 0.8	6.90 ± 0.05

CHAPTER X. SAFETYA. PLANT SAFETY ANALYSIS1. Protection System Dynamics

C. L. Fies (821)

Dynamics of the FTR protection system were selected as a basis for accident analysis in the Preliminary Safety Analysis Report. Analyses covered a range of hypothetical reactivity range up to 5\$/sec, for which the protection system proved to be adequate.

A reactor trip (scram) is accomplished by the insertion of poison elements into the reactor core. All six in-core control and safety elements are inserted into the core accelerated by a 14 in. spring. The minimum worth of these six in-core rods is 7% $\Delta k/k$. The design of the reactor is based on redundant shutdown systems; however, the analysis takes credit only for the primary shutdown system composed of the three in-core safety rods. Assuming one stuck rod, the remaining two rods have a minimum worth of 3.2% $\Delta k/k$. The safety rods are positioned at the top axial interface of the core for improved speed of response.

The time delays associated with the instrumentation and rod systems were selected to meet the speed of response requirements for the accidents described above. The rod release time is 75 msec. This time, combined with a delay of 25 msec in the power range nuclear channels and protection logic, results in a total time delay of 100 msec from the time the overpower trip point is reached until the beginning of rod movement.

The system described above is capable of terminating an excursion due to a reactivity ramp of 5\$/sec with a total insertion of 4\$. Figure 10.1 shows negative reactivity

FIGURE 10.1. Negative Reactivity Insertion of the Protection System Versus Time

insertion of the protection system versus time. The lower curve assumes a reactor trip using two-of-the-three safety rods (stuck rod provision). This curve was used in the analysis. The top curve shows the response of the system as designed when all in-core rods are accelerated into the core.

Figures 10.2 and 10.3 show the reactor power and hot channel fuel temperature response to a 5\$/sec reactivity insertion. The assumption that only two safety rods scram is quite conservative and thereby allows a considerable safety margin below fuel melting.

2. Evaluation of Various Configurations of the Plant Protection System for Unavailability

O. B. Monteith (821)

A study was initiated to investigate the unavailability of various configurations of the plant protection system. Results are given in Figure 10.4. The term "unavailability" used herein is defined as, the probability that the system will not be ready to perform its intended function at some future point in time. This definition takes into account the periodic testing of the system, and random failures of the PPS components. It does not include the effects of systematic failures of the equipment, i.e., failures from environmental effects, human error, or unrecognized dependence on a common element.

The testing interval employed in this study is 170 hr (each week). This testing interval implies that all components are tested and returned to "as good as new condition" after each test. As a consequence, a test each week on all safety rods would be required to confirm the designed scram capability.

FIGURE 10.2. Reactor Power Response to a 5\$/sec Reactivity Ramp with Protection System Action

FIGURE 10.3. Reactor Hot Channel Fuel Temperature Response to a 5\$$/sec$ Reactivity Insertion with Protection System Action

FIGURE 10.4. Unavailability of Configurations (Probability that system will not perform for a valid requirement to scram)

3. Plant Protection System Reliability Analysis

O. B. Monteith (821)

Some preliminary evaluations have been made on the probability of spurious responses (false scrams) of the PPS resulting from equipment failures. An initial result for a typical instrumentation subsystem is about 0.0004 false trips per year. Based on the present nominal number of 50 such subsystems, the expected spurious scram rate for the PPS instrumentation is about 0.02/year. These early results do not include the effect of failures in the ac or dc power supplies, the safety circuits, nor the rod release circuits.

A report was issued on the reliability of the sensors and logic circuits of the PPS: BNWL-1201, October 1969, Reliability Analysis of the FFTF Conceptual Plant Protection System.

B. ANALYSIS OF HYPOTHETICAL ACCIDENTS

1. Sodium Fires in Containment Spaces

P. R. Shire (821)

The design basis for containment spaces within the FFTF includes pressures and temperatures that might result from accidental release of sodium. Analyses were completed for equipment cells and the main containment space above the operating floor, assuming spray and pool reactions.

Although the main containment space encloses an air atmosphere, sources of sodium are limited. For the case in which a large pool fire (540,000 lb Na) occurs in an equipment cell open to the main containment, the analysis yielded pressures less than 8 psig. For other credible sources of sodium, pressures were found to be substantially lower and well within the main containment capability.

Equipment cells are inerted with nitrogen to prevent potentially severe reactions although a small fraction of oxygen is present.

2. Driver and Test Channel Instrumentation Analysis

J. J. Regimbal (823)

Protection against flow obstructions afforded by in-core instrumentation was investigated for driver and test fuel assemblies. Specifically, the capability of flowmeters and thermocouples to initiate protective action was evaluated.

The greater part of the analysis utilized the TIGER V program to survey the parameter conditions which barely produce (or prevent) the occurrence of coolant boiling. Special modifications to that program facilitated the survey, including a) an automatic scram from trip feature and b) provision for problem restart from tape. The SINTR program was used to calculate refined pin conditions based on nominal operation.

From nominal, steady-state conditions of power and coolant flow, each TIGER V transient case was initiated by the occurrence of a flow loss ramp which reduced flow steadily to a specified level; constant flow was then reestablished at that throttled level. The more slowly flowing coolant receives more heat per unit mass, and the sodium temperature rises, with rising fuel and cladding temperatures as the result. The highest coolant temperature occurs at the exit from the core; some heat is lost to the plenum material above this point during a rising temperature transient. Incipient temperature elevation information reaches the thermocouple position (selected 100 in. above the core) in a time which depends upon the flow velocity.

Most of the transient cases specified power reduction by a full scram with thermocouple trips set 30 °F above nominal. T/C time-constants of 0, 1 sec, 2 sec, and ∞ time were used. A variety of flowmeter trip level settings were examined.

The principal results of the study are given in Figures 10.5 and 10.6 for a hottest driver fuel pin and for a high performance (43.5 kW/ft) sodium-bonded carbide test fuel, respectively. We can conclude that adequate protection is furnished in both cases against blockages of 70% or less which take place over 5 sec or greater. The flowmeter, which is not dependent upon channel flow velocity for reaction speed, can provide a greater protective coverage, even with relatively relaxed trip settings. Due to the higher flow velocities required for heat removed from the high performance fuel bundle, a fast T/C actually furnishes an increased boiling protection capability.

FIGURE 10.5. Instrumented Protection Against Driver Channel Boiling

FIGURE 10.6. Protection Against Boiling in a Carbide Fuel Test Channel

APPENDIX A

ORGANIZATION CODES FOR FFTF PERIODIC TECHNICAL REPORTS
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ORGANIZATION CODES FOR FFTF PERIODIC TECHNICAL REPORTS
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