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FAST FLUX TEST FACILITY
SUMMARY TECHNICAL REPORT
JANUARY-DECEMBER 1966

June 1970

AEC RESEARCH &
DEVELOPMENT REPORT

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PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

operated by

BATTELLE MEMORIAL INSTITUTE

for the

UNITED STATES ATOMIC ENERGY COMMISSION UNDER CONTRACT AT(45-1)-1830

FAST FLUX TEST FACILITY

SUMMARY TECHNICAL REPORT
JANUARY-DECEMBER 1966

Work reported herein was directed by
E. R. Astley

Report compiled by
C. P. Cabell

June 1970

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FAST FLUX TEST FACILITY
SUMMARY TECHNICAL REPORT
JANUARY-DECEMBER 1966

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 design-related technical progress made in the Fast Flux Test Facility Program during the period January-December 1966.

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BNWL-325	FFTF Summary Technical Report	January-December 1966
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
BNWL-1275	FFTF Quarterly Technical Report	9-1-69 to 11-30-69

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FAST FLUX TEST FACILITY
SUMMARY TECHNICAL REPORT
JANUARY-DECEMBER 1966

CHAPTER I. INTRODUCTION AND SUMMARY

This report is being issued to summarize major CY-1966 technical studies carried out in support of the FFTF primary effort of concept selection. The actual concept selection effort itself is described in other reports and only a brief outline of this work is included herein.

The arrangement of the report generally conforms to the LMFBR Program elements of plant design, components, instrumentation and control, sodium technology, core design, fuels and materials, fuels recycle, physics, and safety.

The present tense is used throughout this report to make it consistent with other periodic technical reports in this series. Appendix A gives a list of reports issued in CY-1966.

CHAPTER II. PLANT DESIGN - BNW DESIGN STAFF

A. PURPOSE OF FFTF

The purpose of the FFTF Project is to provide a sodium-cooled fast neutron reactor plant for fast flux irradiation testing of reactor fuels and materials of interest to the USAEC Liquid Metal Fast Breeder Reactor Program.

B. INTERIM REFERENCE CONCEPT

In CY-1965 BNW was requested by the AEC to provide a conceptual design for a fast test reactor to meet the design requirements shown in Column I of Table 2.1. These requirements included maximum flux of 1.35×10^{16} n/cm²-sec, minimum of six closed loops (6 in. diameter preferred) and minimum of six open loops. No specific requirements were imposed for core coolant outlet temperature or for verticality of closed test loops.

To meet these requirements, a PuO₂-SS cermet-fueled skewed core reactor concept was proposed by BNW in a CY-1965 informal budget study document. A more formal report describing the concept was compiled and was issued early in CY-1966 as the "Interim Reference Concept"^(1,2) for the FFTF. (Figure 2.1)

C. STUDIES OF ALTERNATE CONCEPTS

Early in CY-1966 the AEC requested BNW to provide overall management for continued FFTF conceptual design studies and for related development work, and to define and evaluate alternate design concepts for the reactor. In compliance,

-
1. E. R. Astley. Progress Report - Fast Flux Test Facility - Interim Reference Concept, BNWL-CC-400. February 1, 1966.
 2. E. R. Astley. The Conceptual Evaluation of a Fast Flux Test Facility, BNWL-SA-978. April 12, 1967.

Neg 41120-1

FIGURE 2.1. FFTF Interim Reference Concept

BNW developed two alternate concepts and requested Phillips Petroleum, Westinghouse, General Electric, and Atomic Power Development Associates (APDA) to submit other conceptual designs for the FFTF reactor.

As a result of the engineering studies, a total of seven "reactor concepts" were defined.^(1,2) Three of the concepts were based on a "skewed" reactor core of the same general type as shown in the "Interim Reference Concept." Three of the concepts were based on a vertical core arrangement. The seventh concept proposed a "split core" design in which several of the closed loops were plumb vertical, and the remainder of the core lattice positions were slightly tilted.

As study work proceeded during CY-1966, requirements governing the FFTF design were modified by the AEC, as shown in Column II of Table 2.1. Minimum peak flux was lowered to 0.7×10^{16} n/cm²-sec. Minimum required number of closed loops was reduced to four and minimum required number of open loops was not specified. New requirements for 1200 °F core outlet temperature and for "essentially vertical" orientation of test loops were established.

Evaluation of the concepts was completed late in CY-1966 and the evaluation results^(1,2) were submitted to the AEC for their subsequent judgement. The AEC selection was a split core reactor,⁽³⁾ with three to five vertical closed loops as shown in Figure 2.2.

-
1. E. R. Astley. The Conceptual Evaluation of a Fast Flux Test Facility - Interim Reference Concept, BNWL-SA-978.
 2. E. R. Astley. Fast Flux Test Facility Quarterly Progress Report, January, February, March, 1967, BNWL-501.
 3. E. R. Astley. Progress Report - Fast Flux Test Facility - Reference Concept, BNWL-470. August 1967.

TABLE 2.1. FFTF Design Requirements Comparison

	I CY-1965	II CY-1966
1. Flux, n/cm^2 -sec	1.35×10^{16}	0.7×10^{16}
2. Flux Spectrum % >0.1 MeV	70%	75%
3. Number of Instrumented Closed Loops	6	4-6
4. Desired Diameter of Closed Loops, in.	6**	NS†
5. Burnup, MWd/ton	1×10^5	NS
6. Number of Instrumented Open Loops	6	NS
7. Closed Loop Testing Subassembly Per Year	4	NS
8. Open Loop Testing Subassembly Per Year	5	NS
9. Test Loop Orientation	NS	Essentially Vertical
10. Core Outlet Temperature	NS	Up to 1200 °F
11. Closed Loop Outlet Temperature	NS	1400 °F
12. Specific Power, watts/gram fissile	500	NS

* Both concepts were also required to meet generally comparable safety and operational criteria and requirements.

** Concept provided four 6 in. diameter loops and two 4 1/2 in. loops.

† NS - Not specified

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FIGURE 2.2. Split Core Concept

III. COMPONENTS

A. HEAT TRANSPORT SYSTEM COMPONENTS

1. Double-Walling Piping Requirements

V. W. Gustafson

Problems in the use of double-walled piping for the primary system* were examined. The following list outlines typical problems found:

- Costs appear to be double that of single-wall pipe as a minimum
- Use of bellows (for differential expansion) in the secondary pipe is required
- Pipe repair is difficult
- Vibrations and stresses are hard to control
- Sodium heating is more difficult.

Due to the numerous problems associated with double-wall piping use, a very basic question evolves as to the value it serves. Because of this a critical examination will be made to determine if its use in FFTF is essential.

2. Water Heat Sink for FFTF Main Heat Removal System

H. G. Johnson

FFTF heat dumps must meet the following requirements:

- The heat exchange equipment must be capable of operation from 0 to 100% of reactor design power.
- For operational flexibility, the heat exchange equipment must be capable of operation over a broad band of primary loop temperatures at any given reactor power below the design power.
- Secondary coolant temperatures must be kept safely above the freezing point of the coolant. For sodium, the freezing point is 208 °F so the minimum temperature probably should be at least 240 °F.

* *Double wall piping is described in BNWL-CC-400. E. R. Astley. Progress Report - Fast Flux Test Facility - Interim Reference Concept, BNWL-CC-400. February 1, 1966.*

Although simple in concept, a dump of secondary coolant heat directly to river cooling water in a liquid/liquid heat exchanger is far from routine. Specific design problems associated with this concept are as follows:

- The difference in heat source and heat sink temperatures is very large. For operation of the primary loop at 1000 °F, one can expect temperature differences between secondary coolant and tertiary cooling water of over 800 °F.
- The bulk outlet cooling water temperature must be limited to below the saturation temperature at atmospheric pressure for safe return of water to river or pond.
- The tube surface temperature in contact with cooling water must be limited to some safe maximum value which depends on the water treatment process used. If pH is carefully controlled, the water film temperature (Columbia River water) can be as great as 290 °F before excessive fouling occurs.
- Sodium and water chemical reactions are possible.

Feasibility of Concept

It appears as if a heat exchanger could be designed and constructed to operate at design power and design temperatures. An abnormally large tube wall resistance is required to absorb the large temperature difference between the source and sink. A double-walled tube with a selected annulus fluid could be used to satisfy the required resistance and at the same time would solve, to some extent, the safety problem of preventing accidental sodium and water chemical reactions due to failure of a heat exchanger component. The large wall resistance tends to result in a more or less constant overall heat transfer coefficient over the full range of reactor power and desired operating temperatures, and therein lies a serious drawback of

the water heat dump concept. Because the overall coefficient is nearly constant and because of the limitations on bulk water and water film temperatures, a given heat exchanger, with fixed heat transfer surface, can only operate over a very small band primary coolant temperature for any power less than the design power. In other words, the water heat dump concept lacks operating flexibility which can be achieved easily with the air dump concept and fairly readily with the steam/water concept. Operating flexibility is discussed in detail under "Flexibility of Concept."

The mechanical design problems are recognized as substantial in a unit designed for steady-state temperature differences on the order of 800 °F and for high rates of temperature changes under transient conditions. However, these problems were not investigated inasmuch as the concept probably can be rejected on the basis of process design considerations.

Flexibility of Concept

The concept lacks operating flexibility in that with fixed heat exchanger surface, the FFTF would have to be operated over a very narrow band of temperatures for a given reactor power. This is shown in Figure 3.1, for a representative sodium/water heat exchanger containing 7550 ft² of surface designed to dissipate 200 MW of heat with secondary coolant cold and hot leg temperatures of 600 and 875 °F. These temperatures correspond to primary coolant cold and hot leg temperatures of 725 and 1000 °F, assuming an intermediate heat exchanger with a log mean temperature difference of 125 °F and approximately equal primary and secondary coolant flow rates at the design condition.

FIGURE 3.1. Required Primary Coolant Hot Leg Temperatures for a Typical Sodium to Water Dump Heat Exchanger

3. Number of Heat Removal Loops Required

L. M. Finch

It has been determined that a two-loop primary heat removal system, as outlined in the current interim reference design, may not be practical. This is due in part to the requirement, as stated in the January 28 AEC program directive,

Attachment A, "A minimum of two reactor primary system loops, two secondary loops, and two heat sinks, designed to permit continued plant operation with one loop in a nonoperating status." Use of a two-loop system would require that each loop be capable of dissipating full reactor power, which is considered impractical from a component sizing standpoint. This leads to consideration of a three- or four-loop primary system, in which the three-loop system components would have the same capacity as the reference design, and the four-loop system would use the Fermi design. These are being reviewed in considerable detail, looking at various component arrangements as well as other types of heat exchanger designs.

4. Dynamic Simulation of Heat Exchangers

C. L. Goss and H. G. Johnson

Dynamic simulation of the main heat removal system is to be done using analog and digital models for process control and hazards evaluation. Traditionally, modeling of heat exchangers is one of the more difficult problems of such dynamic simulations. The heat exchangers are usually modeled as multiple nodes in series wherein various techniques are used to give coolant temperatures for each node. Use of an average of the inlet and outlet temperatures to represent coolant temperature in a node leads to completely unrealistic results. For example, a sudden increase of inlet temperature to a node will produce an equally sudden decrease of outlet temperature, which is impossible. Use of a model wherein perfect mixing is assumed in each node such that the average node temperature is taken as the outlet temperature may eliminate the unwanted sawtooth effect described for the average temperature model, but the assumption of perfect mixing is hardly satisfactory for a tubular type heat exchanger.

Hence, the work reported here was started in FY-1966 to attempt to develop a satisfactory heat exchanger model.

Development of Model

Every point in a multinode FFTF intermediate heat exchanger model may be described by the following set of equations:

$$W_s C_s \frac{\partial T_s}{\partial x} = U_a (T_p - T_s) V_s \rho_s C_s \frac{\partial T_s}{\partial t}$$

$$U = f(W_p, W_s, T_p, T_s)$$

$$C = f(T)$$

$$\rho = f(T)$$

where W = coolant flow rate, lb/sec
 C = coolant specific heat, Btu/lb-°F
 T = coolant temperature, °F
 X = longitudinal distance along tube, ft
 U = overall heat transfer coefficient, Btu/sec-ft²-°F
 a = unit heat transfer surface, ft²/ft
 V = unit volume coolant, ft³/ft
 ρ = coolant density, lb/ft³
 t = time, sec

subscripts p denotes primary side
 s denotes secondary side.

In this development, heat storage in metal posts of the exchanger is neglected. Future, developments are planned either to account for heat storage in metal parts or to justify the assumption of insignificance of these heat storage terms.

For incompressible fluids, the rate of change of weight flow rate with distance, W/x , is zero by the continuity equation. This appears to be a valid assumption to make with a liquid sodium system. Also, variations of coolant physical properties and the heat transfer coefficient with distance are neglected so that the same average values, which are based on

the average of the exchangers, coolant inlet and outlet temperatures, are used for all nodes at each time plane. Thus,

$$\bar{U} = f(W_p, W_s, \bar{T}_p, \bar{T}_s)$$

$$\bar{C} = f(\bar{T})$$

$$\bar{\rho} = f(\bar{T})$$

where
$$\bar{T} = \frac{T_H + T_C}{2}$$

subscripts H denotes hot end
C denotes cold end.

Since $dT/dX = (\partial T/\partial X)(dX/dt) + \partial T/\partial t$,

and since $dX/dt = 0$,

$$dT/dt = \partial T/\partial t = \dot{T}.$$

Thus, the two partial differential equations can be rewritten as:

$$W_p C_p (\partial T_p / \partial X) = U_a (T_p - T_s) + V_p \rho_p C_p \dot{T}_p$$

$$W_s C_s (\partial T_s / \partial X) = U_a (T_p - T_s) - V_s \rho_s C_s \dot{T}_s.$$

The models are evaluated by digital simulation using the GE Dynamic System Analyzer computer program (DYNASAR).

5. FFTF Analog Simulation

R. D. Benham, L. H. Gerhardstein, and W. H. Nielson

This activity is part of the effort to provide a facility for investigating the dynamic behavior of design configurations of reactor and heat removal plant.

Modeling of the tertiary coolant system was completed and tested. Components of the model are:

- Tertiary fan
- Dump heat exchanger air intake valve
- Tertiary bypass valve
- Intake filter
- Friction losses
- Entrance and exit losses
- Draft.

The basic assumptions of the model are:

- Compressible flow
- Bernoulli's Equation applies at inlet and exists
- Ideal gas.

B. FUEL HANDLING COMPONENTS

1. Sodium-Cooled Fuel Handling Machine

H. D. Lenkersdorfer

Work is in progress at Atomics International on development of a so-called "wet" (sodium cooled) fuel handling machine (FHM) with a closure that will permit primary sodium to be drawn into and contained by the FHM during the removal and/or replacement of reactor fuel assemblies or tubes. A spherical lapped closure valve with a freeze seal gave encouraging results in preliminary tests.

2. Capabilities of Gas-Cooled Fuel Handling Machine

A. Padilla

An analysis was made to determine the conditions under which the use of a gas-cooled fuel handling machine would be feasible.

A chopped cosine axial heat flux distribution was assumed for 36 in. long fuel sections with a peak-to-average heat flux ratio of 1.3. Equations were developed to predict the maximum fuel clad temperature and the required gas flow rate as a function of the decay heat generation. The results are summarized in Table 3.1 for an inlet helium coolant temperature of 300 °F.

TABLE 3.1. Required Helium Flow Rates for Gas-Cooled Fuel Handling Machine

Decay Heat Generation, kW	Maximum Allowable Fuel Temperature, °F	Minimum Helium Flow Rate, lb/hr	Helium Inlet Velocity, ft/sec
40	1300	1772	1496
20	1300	74	62
	1200	88	75
	1100	90	76
	1000	153	129
	900	267	226
10	1300	30	55
	900	56	79

The results show that gas cooling at a decay heat generation rate of 40 kW is impractical because of the excessive helium velocity. However, gas cooling for 20 kW appears to be feasible.

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CHAPTER IV. INSTRUMENTATION AND CONTROL

A. CONTROL SYSTEM PROTOTYPE

J. L. Deichman

The purpose of this activity is to provide an experimental facility for evaluating digital control methods, developing data acquisition and control program structure, and combining with the analog facility to perform overall control-process analysis and simulation.

The digital computer has been reassembled in new DEC cabinets. The unit is now five bays long with console mounting hardware and a wing table. The cabinet floor fans were removed and holes cut in the floor to allow plenum air flow to cool the computer modules. The data conversion equipment and control interface for the hybrid has also been mounted in the new cabinets.

A few features of the controller are: 1) the integral and derivative mode constants are independent of the proportional constant, 2) the operator may choose to use an analog setpoint from the analog computer, or a fixed one he can type in, and 3) the sampling interval for each controller can be independently adjusted from 5 msec to 4 sec in 5 msec increments.

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CHAPTER V. SODIUM TECHNOLOGYA. SODIUM STUDIES FACILITY

R. L. Knecht and M. E. Shockley

Scope studies and advanced planning for a centralized Sodium Studies Facility to conduct state-of-the-art and technological sodium demonstration work in support of FFTF have been completed. The facility, to be located in the basement of the 324 Building, has been partitioned to separate the radioactive and nonradioactive work areas and a project proposal is being prepared for the building alterations necessary to provide adequate ventilation, electrical, water, air, fire detection, and radiation monitoring services to the laboratory.

The facility will house and service the following equipment: forced convection sodium flow systems, sodium analytical equipment, laboratory furnishings and safety equipment. Laboratory equipment items that are available, on order, or in shop fabrication include the following: sodium analytical equipment including an atomic absorption spectrometer, a gas chromatograph, a flame photo spectrometer and analytical glassware; laboratory furnishings such as radiochemical fume hoods; vacuum dry boxes and an associated gas purification train; safety showers and sodium fire extinguishing equipment; and an alpha, beta, gamma hand and shoe contamination counter. Technical proposals and bids for a forced convection flow system have been reviewed.

B. RADIOISOTOPE TRANSPORT LOOP

V. H. Troutner

Fabrication of a loop for radioisotope transport studies is progressing satisfactorily. The control and instrumentation

console is complete. The sodium piping is about 50% complete and is expected to be finished by mid-November. After the piping is complete, all that will remain will be trace heating, thermocouple installation, and lagging.

C. OXYGEN ANALYSIS

T. F. Demmitt

The BNW oxygen analysis technique utilizes mercury amalgamation to separate the insoluble residue from the sodium sample. Residue analysis is then performed by titrimetric or spectrophotometric methods. The analytical facilities and equipment are in place, and preliminary demonstration tests have been performed.

D. HYDROGEN ANALYSIS

T. F. Demmitt

The existing BNW method for hydrogen analysis in sodium involves mercury amalgamation to separate the insoluble residue from the sodium sample, followed by hydrolysis of the residue with collection and analysis of the evolved gases. This method is limited to measurement of molecular (free) and hydride hydrogen only. The analysis system is being modified to permit vacuum refluxing of the amalgamated sample to release hydroxide hydrogen. The evolved hydrogen will be measured by gas chromatography.

E. NITROGEN ANALYSIS

T. F. Demmitt

The BNW nitrogen analysis involves a modified Kjeldahl technique in which the sodium sample is dissolved in ethyl alcohol and the nitrogen is then distilled and collected in boric acid for analysis. The analysis system is in place and is currently being calibrated to establish sensitivity and reproducibility values. Initial results indicate this system will be satisfactory.

F. CARBON ANALYSIS

T. F. Demmitt

The BNW analytical system for measuring total carbon in sodium is only partially complete. The analytical method will employ a high-temperature combustion step to form carbon dioxide which will be collected and analyzed by gas chromatography. Completion of this system is being delayed pending receipt of a combustion furnace and gas chromatograph.

G. TRACE METALS

T. F. Demmitt

Measurement of trace metal corrosion product impurities in sodium will be performed by atomic absorption spectrophotometric (AAS) techniques with hydrolyzed samples. The AAS unit is set up and preliminary checkout tests are now in progress. Results indicate the instrument is quite sensitive to the impurities of interest (Fe, Cr, Ni). Final checkout tests will be delayed until the unit is moved to the 324 Building sodium facility.

CHAPTER VI. CORE DESIGNA. CORE MECHANICS1. Reflector Control Rod Study

L. M. Finch

The objective of this task is to provide a conceptual design and analysis of a supplemental reflector control system. The investigation of alternate control concepts is essential to ensure capability for providing the required degree of reactivity and flux control for the wide variety of FFTF core loadings of interest.

Nuclear calculations, to choose feasible materials for the poison backed reflector drums, were completed. The results of the calculations show that twelve 16 in. diameter poison backed reflector drums can supply 6% $\Delta k/k$ shim control, plus additional for boron burnup. The feasible reflector materials appear to be the Inconel alloys and Type 304 SS. These materials were selected from among the following: beryllium, tungsten, carbide, molybdenum, gold, cobalt, copper, and nickel. For the poison backing, tantalum, tungsten, europium oxide, and boron carbide are acceptable from a neutronic standpoint. Tantalum and tungsten offer metal strength and no gas production. Europium oxide also produces no gas. Boron carbide offers operating experience, but produces gas.

Mechanical design of the reflector control system has progressed concurrently with the nuclear analysis. The current concept involves 12 drums, each 15 in. diameter at the top, tapering to 14 in. at the bottom. Problems which are currently being studied include: 1) the types of bearings to be used; 2) provisions for adequate sodium flow around the

drums, bearings, and fixed reflectors; and 3) suitable means of installation and removal of the drums.

2. Analyses of Round Versus Hexagonal Process Tubes

M. T. Jakub and C. D. Flowers

It is desirable to design the FFTF reactor core with a core geometry that is as compact as possible in the interests of both safety and costs. This analysis is to provide the basis for a selection of tube geometry. With low pressure sodium coolant, the hexagonal shape for the process tube cross section makes a convenient receptacle for the high-density, triangular pitch packing of fuel pins. Also, the sodium volume between the triangular pitched tube bundle is at a minimum. A problem with hexagonal tubes is the way the required tube wall thickness increases as internal pressure is increased. The corners of the hexagonal tube provide areas of high bending stress which, of course, are absent in a round tube. Analyses are required 1) that provide tube thicknesses as a function of temperature and pressure, and 2) that yield cost estimates to support the core tube geometry selection.

Pressure-induced primary membrane stresses and tube bending stresses have been calculated for several possible FTR reactor tube sizes. Table 6.1 presents preliminary results; more detailed calculations are in progress. The stresses for both a round and a hexagonal tube are given for each of the tube beam models. The three cases investigated are as follows (Figure 6.1):

Short Tube. The reactor tube is supported axially at the upper tubesheet and extends down through the lower tubesheet. The tube is assumed to be supported radially by only one of the two lower tubesheets in all cases.

Long Tube. The reactor tube is supported at top cover and passes through the upper and lower tubesheets where it is only restrained radially. The long tube model is assumed to be cantilever supported at the upper tubesheet to account for the longer tube above the upper tubesheet.

Open Vertical Core (OVC). The reactor tube is assumed to be free standing, and the tube is assumed to have no top restraint.

TABLE 6.1. S_m Values for FTR Reactor Tubes

S_m Criteria	Short Tube, 5 in. OD		Long Tube, 5 in. OD		OVC 3 in. Hex. Across the Flats
	Hexagonal $t = 0.18$ in. $R/t = 4.0^{**}$	Round $t = 0.1$ in.	Hexagonal $t = 0.18$ in. $R/t = 4.0$	Round $t = 0.1$ in.	$t = 0.1$ in. $R/t = 4.0$
Primary Membrane	1380 psi	2400 psi	1380 psi	2400 psi	980 psi
$\sigma_{\Delta p} = S_m$					
Δp { 100 psi- Tubed 70 psi- OVC					
Primary Bending and Primary Membrane $\sigma = 1.5 S_m$	7530	4550	8220	5230	4320
Δp { 60 psi- Tubed 50 psi- OVC					
Secondary and Primary (Discontinuity Stresses added to above) $\sigma = 3 S_m$	Estimated S_m values lower than above line				
Fatigue	Must wait for detail design				

* S_m is material design number given in ASME Nuclear Vesselcode--suitably modified for FTR conditions of temperature, irradiation and criticality of service

** R/t = corner radius to wall thickness ratio for hexagonal tube.

W, P, F - Preload Support Reactions

FIGURE 6.1. Reactor Tube Beam Models for Calculating Bending Stresses

Table 6.1 gives the assumed tube thicknesses and pressure loads. The primary membrane stresses for hexagonal and round tubes are calculated in the same manner. These stresses are directly proportional to the OD and inversely proportional to the tube wall thickness. The primary bending stress consists of the pressure-induced bending stress set up in the corners of the hexagonal tube (zero for round tube) and the bending stress due to the deflection of the tube beam model.

The reactor tube deflection is the sum of the preload deflection, the radial thermal growth deflection, and the tube bowing deflection due to the approximate temperature gradients across the diameter of the tube. The tube beam bending stress is then calculated as MC/I .

B. CORE PROCESS TECHNOLOGY

1. Selection of Mixed-Oxide Driver Fuel for FFTF

Fuels Development Staff

As part of the initial concept study⁽¹⁾ for the FFTF, cermets, oxides carbides, nitrides, alloys, and other fuel materials were investigated as possible driver fuel candidates. The PuO_2 -SS cermet, metallurgically bonded to stainless steel cladding, was recommended as the reference driver fuel for 800 °F reactor temperature (outlet). UO_2 - PuO_2 was selected as the backup fuel because of the well-developed technology and the justifiably intense interest both here and abroad in the application of oxides to fast reactors. These recommendations were made subject to further review.

As part of the further reviews on fuel selection, an FFTF Driver Fuel meeting, sponsored by AEC and BNW, was held in Richland, March 8-10, 1966. The purpose of the meeting was to promote technical information and discussion relative to the choice of FFTF driver fuel candidates. Emphasis was given to evaluations of cermet and mixed oxide fuel types and their potential for meeting FFTF criteria. Of particular concern was the apparent limit⁽¹⁾ of 800 °F outlet temperature for PuO_2 -SS cermet fuels. This limit was not compatible with newly revised criteria calling for FFTF operational capability in the range of 1000 to 1200 °F outlet temperature.

The meeting was followed up with a comprehensive evaluation of driver fuel requirements. These fuel evaluations culminated in a decision to proceed with the development of an interim or

1. E. R. Astley. Progress Report - Fast Flux Test Facility - Interim Reference Concept, BNWL-CC-400. February 1, 1966.

first-core loading of mixed UO_2 - PuO_2 fuel clad in stainless steel, and to accept a current penalty of lowered neutron flux. Development of a full performance core was retained as an ultimate objective in the overall program.

The fabrication technology for oxide fuels is extensive and well established. There is a large quantity of irradiation data for UO_2 and a limited quantity for UO_2 - PuO_2 . UO_2 - PuO_2 has the potential for achieving the desired FFTF burnup goal. In addition, the mixed oxide would produce nearly as high a fast flux (for a given core size and power) as the PuO_2 -SS cermet fuel.

Mixed oxide fuels are being studied extensively for fast reactors in the United States, the United Kingdom, and in Europe. Transient tests with fast reactor fuel have recently been reported, and the UO_2 specimens failed in a comparatively nonviolent manner. Transient experiments with mixed oxides have been published recently and it was found that no cladding deformation was evident in any of the three tests; melting of 70% of the fuel caused extensive fuel relocation, but no cladding damage; and cladding melting occurred before any fuel vapor pressure effects became significant. It was further pointed out that the absence of cladding deformation is significant, because it means the fuel volume increase on melting can be accommodated by axial expansion (if unrestrained) of the fuel. In this country, oxide fuel has been irradiated to >10% burnup of the heavy metal atoms, and to 9% burnup in Dounreay Fast Reactor. The Russians have irradiated both UO_2 and PuO_2 in the fast flux of BR-5 to 6.5 at.% burnup, but some 20% of the pins failed by splitting of cladding. The UO_2 , irradiated at a low temperature, showed a volumetric expansion of 1.5% ΔV per at.% burnup.

2. Mixed-Oxide Fueled Core Design Fuels Development Staff

Additional parametric studies have been initiated for an FFTF core of 800 liters using mixed-oxide for an interim core loading. The larger volume (as compared to the 600 liter core of the interim reference concept) is essential to achieving a neutron flux near the lower end of our flux range for concept evaluation (0.7×10^{16} n/cm² sec) because of the substantially reduced flux capability of PuO₂-UO₂ fuel, as opposed to the PuO₂-SS cermet. There are a number of considerations which restrict the range of parameters from which a core design can be selected.

The number of basic subdivisions and hence, lattice positions, in the core determines the subassembly size. In a skewed or tilted tube core design, it has a large effect on the maximum skew or tilt angle. Also to be considered is the reactivity worth of a subassembly for both driver and test elements. Since control elements in the core also occupy lattice positions, they may be considered to displace driver or test positions. Finally, the maximum diameter of closed loops to be accommodated in the lattice is a significant factor.

There are several factors influencing the choice of a height-to-diameter ratio. These include change in pressure drop with fuel length, plutonium fraction, pin diameter, length, test length, and the manufacturing parameters associated with pin size and the number of pins per subassembly. Reactor pressure drop is increased as the core length increases, if the same neutron flux is maintained. The pressure drop can be maintained by reducing the pin diameter, but this requires an increase in the plutonium fraction.

Pin diameter and plutonium fraction are interrelated and both affect the core performance. Pin diameters are chosen on

the basis of fabricability and number of pins/subassembly. The plutonium fraction should be as low as possible to maintain a high negative Doppler coefficient in the fuel.

Two subdivisions have been reviewed that form a hexagonal structure for the 800 liter core. The two subdivisions consist of 61 and 91 subassemblies. The first is assumed to have four control elements in the core (3 safety, 1 regulating) and the remaining shim controls in reflector positions; the total number of driver and test positions is then 57. In the core with 91 subdivisions, a minimum of 10 control positions were assumed within the core, resulting in a maximum of 81 driver plus test locations. Analyses have indicated that better performance is achieved with a core division of 61 rather than 91 subassemblies. This may not be true, however, for an open pool type of core. While a core with 61 subassemblies is limited in the variety of testing locations, it appears to be a preferred design from the standpoints of skew or tilt angle, accommodation of 6 in. loops, and reduction of nonfissile material in the core.

Several height-to-diameter ratios were examined for the current study. It was determined that the flux penalty for increased core height is quite severe. Larger core heights also tend to make the sodium coefficient more positive. A height-to-diameter (H/D) ratio of 0.9 still gives a core height of 3.06 ft, an acceptable value.

The longer core (H/D of 1.25) may have an advantage if full performance driver fuels are considered. This core has a subassembly size of 4.5 in. across flats compared with 5.0 in. across flats for a core with height-to-diameter of 0.9. The longer core is readily shortened to a 600 liter core using an advanced driver fuel at an H/D of 0.9, for example. One method of increasing the core performance for a longer core is to

increase the permissible pressure drop. However, the maximum sodium velocity in the core becomes limiting in this regard.

Another method of increasing core performance is to decrease the core overpower factor. This raises the operating temperature of the fuel closer to the melting point. The presently assumed overpower factor is 1.25, although values down to 1.1 have been considered. It is felt that a value of 1.25 represents a safe operating condition for a test core such as the FTR. Detailed accident studies must finally determine this factor, however. Lowering the overpower factor raises the allowable kW/ft, and thus the flux. However, a more complex relationship is involved. The pressure drop increases as the power grows and the pin density (number of pins/subassembly) must be reduced to maintain the core pressure drop. This, in turn, has the effect of raising the plutonium fraction. Table 6.2 shows selections from the most recent survey for a number of cases to illustrate some of the considerations discussed. A core design that appears to represent a good compromise of all factors is the core listed in the first column. This design yields a peak centerline neutron flux slightly above the minimum of the specified range. The number of pins per subassembly and pin diameter are acceptable from a fabrication standpoint. The plutonium fraction is low enough that the Doppler effect may be satisfactory, although detailed accident studies must finally resolve whether or not BeO is necessary. The test positions allowed are sufficient to satisfy FTR testing requirements as presently envisioned. Core power is near 400 MW and can be handled by the heat removal system.

3. Cermet Fuel Expansion in Fast Transient Fuels Development Staff

In a fast reactor flux ramp, a plutonium stainless steel cermet fuel must provide a negative overall thermal coefficient

TABLE 6.2. FFTF Mixed Oxide Core Survey

Characteristic	Common Parameters						Major Parameter Change		
	Basic Core	Long Core	Larger Fuel Pins	Higher Flux	Highest Flux	Low Over- power	Underlined	Long Core, High ΔP Low Over- power	High ΔP , Low Over- power
Height, ft	3.06	3.82	3.06	3.06	3.06	3.06	3.82	3.82	3.06
H/D	0.9	1.25	0.9	0.9	0.9	0.9	1.25	1.25	0.9
Na vol%**	41.1	57.5	37.5	45.4	55.5	51.8	41.1	57.5	41.1
Fuel vol%***	32.2	26.3	33.3	29.1	24.8	26.3	32.2	26.3	32.2
SS vol%**	26.7	16.2	29.2	25.5	19.7	21.9	26.7	16.2	26.7
PuO ₂ , vol%***	25.1	32.8	25	30	37	31.1	24.5	31	25.2
Crit. Mass, K fiss.	457	458	457	457	457	457	457	458	457
Overpower factor	1.25	1.25	1.25	1.25	1.25	1.1	1.25	1.1	1.1
Pins/SA	331	217	271	331	397	271	331	217	331
Pin OD, in.	0.20	0.20	0.225	0.190	0.160	0.20	0.20	0.20	0.20
SA Flat dimension, in.	5.0	4.5	5.0	5.0	5.0	5.0	5.0	4.5	5.0
Flux, n/cm ² -sec $\times 10^{-16}$	0.83	0.74	0.74	0.88	1.0	0.88	0.88	0.84	0.94
Core Power, MW	360	320	320	373	430	373	373	360	420
Core Pressure drop, psi	60	60	60	60	60	60	90	90	90
Pins/yr at 45,000 MWd/T	19,500	14,500	14,000	22,000	27,000	21,000	21,000	15,000	22,000

* Includes one package loop

** Total core volume fractions (inc. test fuel in 10 tests)

*** Driver fuel only.

of nuclear reactivity through expansion of the stainless steel matrix of the cermet. In the BNW recommendation⁽¹⁾ for selection of Pu-SS cermet as the reference fuel, the need for additional study to evaluate the adequacy of the cermet expansion was defined.

"Three areas which have not been thoroughly investigated for cermet fuel pins are the stress conditions under operation, the reliability of the thermal expansion at high burnup, and the behavior under transient conditions."

"The (UO₂-SS) cermet thermal expansion tests indicate that experimental and calculated values do not agree. This suggests that there is little cohesion between phases and that the stainless steel network expands independently of the UO₂ particles. In addition, the tensile tests show that the ultimate and yield (0.2%) stress values at 700 °C are about one-half the values at 20 °C.

In view of these considerations, analytical studies have been made to evaluate the performance of PuO₂-SS cermet during a fast flux ramp. Temperature profiles from the calculations show that a large fraction of the stainless steel matrix is at temperatures above 1100 °F. At these temperatures the matrix stress behavior becomes partially plastic, and the expansion is less than would be computed by assuming elastic behavior of the matrix.

The calculational model used a modified version of the TIGER V transient heat transfer code. The calculations determined the radial temperature profiles and average temperatures of both the stainless steel and the PuO₂. At temperature differences less than 360 °F (185 °C), a thermal contact coefficient of 500 Btu/hr-ft²-°F was used. Between temperature differentials of 360 °F (185 °C) and 400 °F (205 °C) the thermal contact coefficient increases to 100,000 Btu/hr-ft²-°F. Above temperature differentials of 400 °F the thermal contact coefficient was held constant at 100,000 Btu/hr-ft²-°F.

1. E. R. Astley. Progress Report - Fast Flux Test Facility - Interim Reference Concept, BNWL-CC-400. February 1, 1966.

4. FFTF Driver Fuel Element Pressure Loss

G. J. Crandall

Spiral wire wrap is being considered as a means of spacing driver fuel element fuel pins. A preliminary step in determining the feasibility of using this type of spacer is to determine the pressure loss of a driver fuel element employing spiral wire wrap. To this end, a preliminary study to determine driver fuel element pressure loss as a function of pitch of the spiral wire wrap was performed.

Driver fuel element pressure loss as a function of pitch of the spiral wire wrap was calculated (Figure 6.2). Pressure loss due to the spiral wire spacers and skin friction in the fuel pin bundle were determined by the de Stordeur⁽¹⁾ method. Other fuel element pressure losses (i.e., skin friction in the fuel element above and below the core, entrance and exit losses, etc.) were determined by conventional methods.

In the calculation of these pressure losses, it was assumed that:

- Reactor operating power is 300 MW_t.
- The equivalent number of average driver fuel elements is 53-1/2.
- Coolant temperature differential across an average driver fuel element (axially) is 300 °F.
- The driver fuel element is cermet.
- Fuel pin diameter, lengths, and spacing are as indicated in BNWL-CC-400.
- Coolant velocity in a central fuel pin bundle flow cell approximately equals the average coolant velocity in the fuel pin bundle.
- Each fuel pin is wire wrapped.

1. A. N. de Stordeur. "Drag Coefficients for Fuel-Element Spacers," Nucleonics, June 1961.

FIGURE 6.2. Driver Fuel Element (Cermet) Pressure Loss (FFTF) as a Function of Fuel Pin Spiral Wire Spacer Pitch

The driver fuel element pressure loss values indicated in Figure 6.2 are expected to be accurate within $\pm 20\%$.

Considering present pump technology and assuming that a single stage pump will be used, driver fuel elements with pressure losses above 80 psi for reactor operation at 300 MW_t should not be considered.

If it is decided to use spiral wire spacers, a more detailed hydraulic analysis of a cermet and oxide driver fuel element employing spiral wire spacers will be necessary.

The driver fuel element pressure loss values indicated in Figure 6.2 are satisfactory for preliminary studies and will be used in determining the feasibility of using spiral wire wrap as a means of spacing driver fuel pins.

5. Liquid Superheat Necessary for Nucleation

A. Padilla

A study of the superheat required for initiation of boiling has been conducted to provide the information for the establishment of reactor operating limits and to furnish the basic data for the safeguards evaluation. The various analytical methods for calculating the liquid superheat necessary for nucleation have been reviewed and the uncertainties in their derivations assessed.

The theory of a spherical bubble in static equilibrium has been used to calculate the superheat as a function of saturation temperature. A comparison of the theory with experimental superheat data for sodium and potassium gives merely qualitative agreement but a conservative limit for the superheat can be obtained if the cavities on the heating surface of which nucleation takes place can be characterized.

Figures 6.3 and 6.4 show the superheat as a function of saturation temperature for various cavity radii for potassium and sodium, respectively. For a given cavity size, the superheat decreases quite rapidly with saturation temperature. From the experimental data for potassium in Figure 6.3, it is seen that the points for 0.006 in. diameter cylindrical holes drilled in the surface agree qualitatively with the curve for a cavity radius of 0.003 in. The data for potassium also indicates that an upper limit for the cavity radius of a smooth surface is 0.003 in.

FIGURE 6.3. Superheat Required for Nucleation in Potassium

FIGURE 6.4. Superheat Required for Nucleation in Sodium

The superheat data for sodium is plotted in Figure 6.4. Based on the experimental data for potassium, a conservative estimate of the superheat for smooth surfaces is given by the 0.003 in. radius curve and the superheat required for nucleation at 1 atm is 800 °F.

6. NUTIGER Transient Analysis Code Development

C. E. Leach and K. R. Merckx

NUTIGER is a combined nuclear kinetic and transient heat-transfer code that allows the simultaneous solution of reactor kinetics and heat-transfer with Doppler, coolant density, fuel expansion and bowing reactivity. It is anticipated that this code will be one of the principal analytical tools for investigating transient behavior of the driver fuel.

The code principally consists of the KAPL TIGER code and a zero-dimensional, six group nuclear kinetics code programmed especially for the purpose at hand. Reactivity feedbacks are included as self-contained subroutines with NUTIGER delivering the required detailed input. Since these feedbacks are included as subroutines they can be programmed at will according to the purpose at hand. TIGER is a finite difference approach to solution of transient heat transfer problems. These combine to allow considerable flexibility and detail in selection of geometry and lead to a code of general applicability to FFTF problems.

Both open (no reactivity feedback) and closed loop operation of the NUTIGER code is now available. Doppler, fuel clad expansion and coolant density reactivity feedbacks are included and operable. Fuel bowing requires additional debug and verification.

a. Nutiger Test Case

A test case was programmed to verify all code options except fuel bowing. The test case parameters are as follows:

Fuel Geometry

Pin OD	0.2 in.
Length	3.0 ft
Clad Thickness	0.012 in.
Fuel	20 vol% PuO ₂ 80 vol% UO ₂
Fuel Form	Pellet
Gas Gap	0.0025 in.
Pin Spacing	0.050 in. equilateral triangular lattice

Coolant

Sodium	
Inlet Temperature	700 °F
Outlet Temperature	1000 °F
Velocity	25 ft/sec

Physics Input

Reactivity Insertion Rate	+50 cents
Generation Time	4* 10 ⁻⁷ sec

Group Constants

β_i	λ_i
0.000114	0.0129
0.000840	0.0311
0.000649	0.1340
0.00986	0.3310
0.000306	1.26
0.000106	3.21

Reactivity Feedback Coefficients Fuel Expansion

$$\frac{\partial}{\partial K} \frac{dK}{d\epsilon} = -3.08 \times 10^{-1}$$

Coolant Density

$$\frac{1}{K} \frac{dK}{d\epsilon} = -2.4 \times 10^{-4} \text{ ft}^3/\text{lb}$$

Doppler

$$\frac{T}{K} \frac{dK}{dT} = -4.3 \times 10^{-3}$$

Melting of the fuel was allowed. The fuel was assumed to melt at 4980 °F, absorbing 115 Btu/lb as a consequence of fusion.

The fuel was subdivided into 3 radial nodes, a clad node, and coolant node. These were in turn split into 3 axial nodes. The axial normalized power profile was assumed as 90, 120, and 90% of average for the lower, middle, and top nodes, respectively. The bonding gas was assumed to be helium. For this case, the fuel expansion was assumed to depend solely on clad expansion. A subroutine was programmed, based on this model.

The initial computer run was made assuming coolant flow rate and inlet temperature were constant with time. Additional runs are being made assuming only a coolant flow rate decrease of 10% per second. In all instances, the run is intentionally terminated at the temperature at which the sodium would experience equilibrium pool boiling.

Preliminary analysis of the results from these runs revealed no obvious anomalies.

b. Comparison of NUTIGER and K-III

Comparative runs, both open and closed loop, were made with the K-III code.

Open loop comparison was accomplished for positive and negative step, linear and quadratic reactivity inputs. The NUTIGER code is capable of arbitrary reactivity inputs for which no comparison is available.

Closed loop comparison was accomplished by linearizing the NUTIGER fuel expansion and coolant density coefficients in the region of interest and using these numbers as input to K-III (Table 6.3).

TABLE 6.3. Test Case Reactivity Coefficients

	Doppler, 10^{-3}	Fuel Expansion	Coolant Density
NUTIGER	$\frac{TdK}{KdT} = -4.3$	$\frac{\ell dK}{Kd\ell} = -3.08 \times 10^{-1}$	$\frac{dK}{d\rho} = -2.4 \times 10^{-4} \frac{\text{ft}^3}{\text{lb}}$
K-III	$\frac{TdK}{Kdt} = -4.3$	$\frac{dK}{dT} = -3.49 \times 10^{-6}$	$\frac{1}{K^0} \frac{dK}{dT} = 3.46 \times 10^{-6} \frac{1}{K^0}$

The model chosen to be on a comparable basis with K-III consists of three regions and no axial dimensions. The three regions are: Region 1 - fuel; Region 2 - cladding; Region 3 - coolant. Doppler, fuel expansion, and coolant density reactivity coefficients apply to the regions in that order. A reactivity ramp of 0.0015 t was used to perturb the system. The results are compared in Table 6.4.

Analysis has demonstrated that discrepancies are due to three items: (1) comparison at slightly different times, (2) linearization of coefficients, and (3) differences in calculated temperatures. The comparison is thought to demonstrate adequate correspondence.

Additional noncomparative closed loop runs were made to observe NUTIGER's behavior under conditions of varying coolant flow rate and coolant inlet temperature. The results are deemed to be satisfactory.

C. IRRADIATION TESTING FACILITIES

1. Comparison of FFTF Closed Loop Testing Capabilities with Requirements for 1000 MW_e FBR Studies

Design Analysis Staff

The characteristics and performance of FFTF fuel test loops were not studied in detail previously, and the equivalent number of average driver fuel subassemblies was only roughly approximated. Therefore, an investigation of fuel test loop operation was initiated to provide more accurate information for use in fuel cycle analysis, cost analysis, etc. In this study it was assumed that:

- Reactor Operating Power: 300 MW (Cermet Core)
402 MW (Oxide Core)
- Flux: 1.0×10^{16} n/cm²-sec (Core Center, Cermet Core)
 0.92×10^{16} n/cm²-sec (Core Center, Oxide Core)

TABLE 6.4. Comparison of Reactivity and Power for K-III and NUTIGER

<u>Time, sec</u>	<u>Doppler</u>	<u>K-III</u>			<u>Total Reactivity</u>	<u>Power Ratio</u>
		<u>Fuel Expansion</u>	<u>Coolant Density</u>			
0.01607	-9.75×10^{-8}	-1.15×10^{-9}	$+1.73 \times 10^{-10}$		2.4×10^{-5}	1.007
0.1008	-3.68×10^{-6}	-2.43×10^{-7}	$+1.80 \times 10^{-7}$		1.47×10^{-4}	1.052
0.2010	-1.45×10^{-5}	-1.48×10^{-6}	$+1.22 \times 10^{-6}$		2.87×10^{-4}	1.108
0.3571	-4.55×10^{-5}	-6.16×10^{-6}	$+5.18 \times 10^{-6}$		4.90×10^{-4}	1.205
		<u>NUTIGER</u>				
0.01584	-8.80×10^{-8}	-2.17×10^{-9}	$+1.05 \times 10^{-9}$		1.77×10^{-5}	1.006
0.1069	-5.09×10^{-6}	-2.52×10^{-7}	$+1.86 \times 10^{-7}$		1.50×10^{-4}	1.053
0.1980	-1.76×10^{-5}	-1.32×10^{-6}	$+1.09 \times 10^{-6}$		2.74×10^{-4}	1.104
0.3563	-5.69×10^{-5}	-5.87×10^{-6}	$+5.16 \times 10^{-6}$		4.72×10^{-4}	1.200

- Flux: 0.45×10^{16} n/cm²-sec (Average in Peripheral Loops, Cermet Core)
 0.313×10^{16} n/cm²-sec (Average in Peripheral Loops, Oxide Core)
- Loops: Occupied by fuels proposed in the GE,⁽¹⁾ CE,⁽²⁾ AC,⁽³⁾ and Westinghouse⁽⁴⁾ 1000 MW_e studies

Allocation of loops was such that a loop with high flux and one capable of accepting large test sections was provided for testing each of the 1000 MW_e fuels. The following results were obtained:

- Total operating power of all fuel test loops: 29.7 MW (Cermet Core) and 34.4 MW (Oxide Core)
- Power of an average driver fuel subassembly: 6.1 MW (Cermet Core) and 5.7 MW (Oxide Core)
- Equivalent number of average driver fuel subassemblies: 48.9 (Cermet Core) and 71.0 (Oxide Core)
- Ratio of average loop to driver fuel power: 0.487 (Cermet Core) and 0.439 (Oxide Core)

The testing capabilities of FFTF appear to be very good in that predicted lineal average and maximum power values for each of the 1000 MW_e fuels operating in FFTF were, in most cases, above and never below the values expected for operation in the 1000 MW_e reactors by more than 10%. This indicates that FFTF will be able to perform prototypic fuel tests and will not be

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1. Liquid Metal Fast Breeder Reactor Design Study (1000 MW_e UO₂-PuO₂ Fueled Plant), GEAP-4418, Vol. I and II. General Electric Co., Schenectady, N. Y., January 1964.
 2. Liquid Metal Fast Breeder Reactor Design Study, CEND-200, Vol. I and II. Combustion Engineering Inc., Idaho Falls, Idaho, January 1964.
 3. Large Fast Reactor Design Study, ACNP-64503. Allis-Chalmers Mfg. Co. Milwaukee, Wisconsin. January 1964.
 4. Liquid Metal Fast Breeder Reactor Design Study, WCAP-3251-1. Westinghouse Electric Corp., Pittsburgh, Pa. January 1964.

hindered by burnup values lower than necessary to obtain the desired results in the required time.

2. "Rabbit" Irradiation Test Facility

M. K. Mahaffey

A so-called "rabbit" facility for performing brief irradiation tests is being considered. The test specimen would be moved into, and out of, the reactor core by flow of inert gas.

Figure 6.5 shows a schematic of the system proposed. The main components are:

- Gas sampler and counter for failure detection
- Gas heat exchanger for in-core cooling of the sample
- High pressure gas supply for insertion and removal
- Blower for circulating the gas during cooling or optional insertion and removal
- In-core loop for positioning the sample for irradiation
- Out-of-core insertion terminal and receiver.

Valves shown are remotely operated to provide automatic operation.

Normal operation includes a preliminary purging and gas analysis of the pneumatic system both to check integrity of the system and to remove possible sources of contamination. Insertion and return of the sample at the end of the preset irradiation period is automatic. The sample can also be returned manually or by high temperature trips in the outlet coolant and gas line, by a high detected radiation level in the cooling gas, or by a containment trip or reactor scram. Unique features of the system include⁽¹⁾ optional in-core cooling of the sample and⁽²⁾ the ability to propel the sample by a pressure pulse from the high pressure gas supply of "free blown transfer from inert gas system or blower.

FIGURE 6.5. Rabbit Gas System Schematic

D. OTHER CORE DESIGN

1. Gas Cooling of Fuel Subassemblies During Inspection

D. R. Dickinson

An analytic study was conducted to evaluate the feasibility of inert gas cooling of discharged FFTF fuel subassemblies during examination and disassembly. Consideration was given to driver fuel subassemblies of the reference design generating 20 kW of decay heat during interim inspection without disassembly and 10 kW during disassembly and more complete examination. A maximum permissible cladding temperature of 600 °F was assumed; primary consideration was given to argon as the coolant. Cooling will require axial gas flow before the removal of the liner tube, and transverse flow after its removal. The study indicates that gas cooling will be feasible; however, certain problems have to be overcome by proper design and adequacy of cooling demonstrated by tests. The principal problem areas disclosed by this study are:

- Flow Requirements. For axial flow with the liner in place, about 300 ft³/sec of argon will be required at 20 kW; 150 ft³/sec at 10 kW. For transverse flow at 10 kW, the flow requirement is difficult to estimate as it will depend on the ability to cool the fuel pins near the support and leakage around the edges as discussed below, but is estimated as about 1000 ft³/sec of argon. If helium were used, the volumetric flow required would be the same as for argon; for nitrogen, 71% as great.
- Pressure Drop. Because of the close spacing of the fuel pins, significant pressure drop will be required to provide adequate gas flow. If argon is used, the pressure drop will be about 7 psi for axial flow at 20 kW, and 0.5 to 1.0 psi for transverse flow at 10 kW. Compression

to 7 psi will heat the gas by about 100 °F; it will be desirable to cool the gas before it is used to cool the fuel. These pressures will impose significant forces on the fuel bundle which must be adequately supported (about 100 lb axially and 70 to 140 lb laterally). The use of helium or nitrogen would be better in this respect than argon, as the pressure drop would be only 16 and 41% as great, respectively.

- Spacers. The spacers in the fuel pin bundle will hinder gas cooling, especially with transverse flow. Either the reference spacer design (described in BNWL-CC-400), or wire wrap on a 10 in. pitch will block transverse flow over a length of 1/2 or 1 in. It is not possible to predict the amount of gas flow and mixing in this region without experimental data. Axial conduction can provide adequate heat removal to better cooled portions of the pin with temperature differentials on the order of 50 to 100 °F for cermet and 100 to 200 °F for oxide fuel.
- Flow Leakage. Cooling in transverse flow will require baffles which will channel the gas flow through the fuel bundle and prevent leakage around the sides. These must be flexible enough to permit rotation of the fuel and to function after removal of part of the fuel pins. This will make close tolerances difficult and leakage a major problem. Leakage equal to 100% of the flow through the fuel bundle would be possible with gaps on the order of 0.1 in.

2. Reactor Tank Model

G. J. Crandall, C. L. Goss, and H. G. Johnson

Dynamic simulation of FFTF requires that a detailed mathematical model be developed that will describe the reactor

tank during all types of reactor operation (i.e., reactor startup, shutdown, steady state, power perturbations, etc.) so that the effects of these types of operation on out-of-core components (e.g., heat exchangers, coolant pumps, etc.) can be determined.

Work has been done on modeling of the FFTR reactor tank. Reactivity terms, δk , to account for fuel movement due to reactor tube and fuel pin bowing and for thermal expansion or contraction of the reactor tube, are being added to the model to give an equation that will describe reactor power during all types of reactor operation.

CHAPTER VII. FUELS AND MATERIALSA. FUELS EVALUATION*1. Mixed Oxide Fuel Irradiation Testing Program
Fuels Development Staff

An irradiation testing program was formulated to support development of mixed oxide fuel for the FFTF, in particular, and the LMFBR program in general. This program is based on the need for specific data for design of the FFTF driver fuel elements, and incorporates appropriate contributions expected from the LMFBR mixed oxide fuel development programs being conducted by GE, ANL, and NUMEC. Thus, duplication and redundancy of experiments are minimized. As presently planned, the program involves irradiation of 92 mixed-oxide specimens in EBR-II and 54 mixed-oxide specimens in thermal flux test reactors. Twelve mixed-oxide specimens are to be transient irradiated in TREAT without prior steady-state irradiations.

The scope of the program was derived from formulations of detailed irradiation testing needs and specific tests to be performed on the fuel. The major performance factors to be evaluated are as follows:

- Fuel swelling
- Fission gas release characteristics
- Plutonium and fission product migration characteristics
- Fuel structural changes
- Fuel slumping
- Compatibility of cladding and structural materials
- Failure behavior
- Transient behavior
- Mechanical stability of fuel pins and subassemblies
- Design verification and FTR fuel surveillance
- Fuel element performance improvement.

* See also section VI-B-1, *Selection of Mixed-Oxide Driver Fuel for FFTF.*

Design of fuel specimens and capsules has been completed for fast flux and thermal flux specimens. These capsules are designed to operate at the capsule cladding temperatures and power generations regardless of fuel form or stoichiometry of the fuel material. With these two parameters held constant, any variation in irradiation performance should be directly attributable to fuel form, stoichiometry, or flux spectrum.

2. Effect of Temperature on Mechanical Properties of Cermets

E. A. Snajdr

The yield strength and ultimate strength of 20 vol% UO_2 -304 SS cermet (in the as-extruded condition) was studied as a function of temperature. The data are shown in Table 7.1.

TABLE 7.1. Mechanical Properties of a 20 vol% UO_2 -304 SS Cermet as a Function of Temperature

<u>Testing Temperature, °F</u>	<u>Yield Strength, psi</u>	<u>Ultimate Strength, psi</u>
72	38,200	54,500
752	24,500	37,000
1,328	16,500	23,000
1,472	16,000	18,700
1,724	6,500	7,300
1,832	4,000	5,000
2,012	3,000	3,200

Metallographic examination of the test samples is in progress. Preliminary examination shows the UO_2 to be fractured when tested at temperatures between ambient and 900 °C. At temperatures above 1,652 °F fracturing of the UO_2 decreases and the stainless steel begins to pull away from the UO_2 particle, creating voids.

3. Cermet Fuel Irradiations

FFTF cermet pin irradiations are tabulated below.

<u>Ebr-II Subassembly</u>	<u>No. of Pins</u>	<u>Pin Composition</u>
X 009	2	PuO ₂ -SS
X 011	2	PuO ₂ -SS
	1	UO ₂ -SS

B. MATERIALS TECHNOLOGY

1. Fuel Pin Cladding Creep Analysis

G. L. Fox

During FTR operation, the fuel pin cladding will be subjected to stresses caused by internal temperature gradients and pressure from fission gas buildup. The resulting creep is arbitrarily limited to approximately 0.1 to 0.2% because of decreased cladding ductility. Since the expected fuel pin operating life is about 5000 hr, it becomes necessary to consider primary creep in addition to secondary creep for more accurate creep calculations. Creep data for both irradiated and unirradiated Type 304 SS are being examined. The data are being fit to empirical curves of the form

$$\dot{\epsilon} = c\sigma^n t^a$$

where $\dot{\epsilon}$ = creep rate

σ = stress

t = time

c, n, a = material constants which are dependent on temperature.

A computer program was written to optimize the constants c , n , and a . More specific information about material constants and uncertainties in the curve fit will be given when the analysis is completed.

2. Irradiation Testing of Fast Reactor Structural and Cladding Materials

J. J. Holmes and R. E. Robbins

The objective of this program is to determine the combined effects of environment and fast reactor irradiation on the mechanical properties of candidate fast reactor structural and cladding materials. The program currently in progress is specifically directed at providing a basis for selection of optimum alloys for FFTF application and at providing a description of material behavior in fast reactor service.

Four MK-A irradiation pins containing 18 tensile and 10 creep specimens of AISI 304 SS have been discharged from the EBR-II and have been shipped to PNL for postirradiation testing. These specimens have accumulated a total fluence of 8×10^{21} to 2×10^{22} at 750 °F (400 °C) in a helium environment. Subsequent testing will be directed at establishing the stress rupture properties at 1200 °F (650 °C) to 10^4 hr and the effects of test temperature (room temperature to 1500 °F (815 °C) and strain rate on tensile properties. All testing will be closely supported by optical and electron microscopy.

Two full EBR-II subassemblies having the MK-B-7 configuration are being prepared for insertion into the EBR-II in January. One subassembly, scheduled for irradiation to 1×10^{23} fluence, is to be placed in the reactor core; the other will be placed in Row 7 and is scheduled for 1×10^{22} fluence. Each subassembly contains a series of sodium-filled, gamma-heated capsules designed to operate at three temperatures, namely, 900, 1200 and 1400 °F (480, 650 and 760 °C). Both units will contain mostly AISI 304 and 316 SS tensile, biaxial creep, and uniaxial creep specimens. However, also included are backup FFTF alloys, such as 348 and 321, as

well as Hastelloy X-280, Inconel 600, Incoloy 800, Inconel 718, Inconel 625, and Inconel X-750, and materials for basic damage studies, namely pure molybdenum, iron, and rhenium single crystals. The data packages for both the Rows 2 and 7 experiments have been sent to the EBR-II site.

3. Liquid Metal Capsule Irradiations

K. R. Wheeler

Capsule GEH-22-3, containing convecting NaK coolant, has begun its third irradiation cycle in the ETR. A temperature record of the capsule is still being maintained by means of one remaining thermocouple. This couple indicates a central nominal sodium temperature of 1100 °F (600 °C), the equilibrium temperature throughout past cycles at this capsule location. The capsule will be discharged at the end of the cycle during which all communication with the capsule is lost.

CHAPTER VIII. FUELS RECYCLE

A. TENTATIVE DESIGN FOR MIXED OXIDE FUEL

Fuels Development Staff

A tentative mixed-oxide fuel element and subassembly design has been prepared to incorporate the latest results of parametric analyses. Details are summarized below. A split fission gas plenum has been incorporated into an alternate fuel pin design, as it may offer the advantages of smaller total plenum volume (due to lower temperature operation of the upstream plenum), lower flow resistance, and self-orificing.

Mixed Fuel Pin Description

Fuel Composition	UO ₂ -30% PuO ₂
Fuel Density (Smear)	85% TD (75% TD for optional pins)
Fuel Material Preparation	Physical mixing and impacting following by 1) vibrational compaction, <u>or</u> 2) pressing and sintering
Fuel Diameter	0.220 in. for vibrationally compacted pins; 0.216 in. for pellets
Diametral Clearance, Pellet-to-Cladding	0.004 in.
Cladding Material	304 SS
Cladding Thickness	0.015 in.
Cladding OD	0.250 in.
Total Pin Length	50 in.
Fuel Length	30 in.

The 85% dense fuel will operate with peak pin power in the range of 13.0 to 17.7 kW/ft. The 75% dense fuel will operate with peak power between 11.5 to 15.5 kW/ft. The calculated exposure for these pins is approximately 50,000 MWd/ton.

Subassembly

Configuration	Fuel pin bundle, hexagonal spacing
Number of Pins	214 fuel pins + 3 guide rods
Spacing	P/D = 1.24 (minimum distance between pins 0.045 in.)
Process Tube	4.1 in. across flats (outside dimension)
Spacers	Open (several candidates being investigated)
Pin Anchor	Open (several candidates being investigated)

B. MIXED-OXIDE PELLET FABRICATION

Fuels Development Staff

Existing equipment is being modified and relocated to establish an integrated mixed UO_2 - PuO_2 experimental pellet fabrication facility. The possible need to fabricate 61 pin EBR-II test assemblies will require about 4000 pellets per assembly. It would be difficult to make the number of pellets with present manual pressing facilities. As a part of the present modifications, an automatic pelleting press will be needed.

CHAPTER IX. PHYSICSA. CORE PHYSICS1. Physics Characteristics of an 800 Liter Oxide Reference Reactor

R. W. Hardie

Neutronics calculations were performed for an 800 liter, PuO_2 - $^{238}\text{UO}_2$ core ($L/D = 0.8$). The following parameters were determined: critical mass, reactivity coefficients, flux, flux spectra, breeding ratio, sodium worth, and Doppler coefficient. All computations were performed using a modified version of the Russian 26 group, cross-section compilation; shielding factors were generated using the fcc fundamental mode code. The one-dimensional runs were done using LASL's 1-D code, DTF-IV (S_4 option).

The principal statics characteristics of the ceramic reference reactor are described in Table 9.1. Critical mass

TABLE 9.1. Neutronics Characteristics of an 800 Liter Oxide Reactor

Parameter	Values
k_{eff}	1.04
vol% PuO_2	5.66
vol% $^{238}\text{UO}_2$	19.34
vol% SS	25.0
vol% sodium	45.0
vol% void	5.0
Mass ^{239}Pu (kg)	412
Mass ^{240}Pu (kg)	46
Mass ^{238}U (kg)	1495
ϕ_T ($\text{n-cm}^{-2}\text{-sec}^{-1}$)	0.55×10^{16} ($\bar{P}$ = 0.5 MW/l)
$\phi_{\text{peak}}/\phi_{\text{avg}}$	1.76
$\phi(>1.4 \text{ MeV})/\phi_T$	0.10
$\phi(>0.1 \text{ MeV})\phi_T$	0.67
$\phi(>0.01 \text{ MeV})\phi_T$	0.94
Reactivity Life (days, $\delta k = 0.03$)	86 ($\bar{P} = 0.5 \text{ MW/l}$)
Breeding Ratio	0.49

calculations were done in cylindrical geometry; axial leakage was simulated using a reflector savings of 12 cm on each end. The reactor was assumed to consist of a homogeneous core surrounded by a 30 cm Inconel radial reflector.

Table 9.2 summarizes some of the important kinetics parameters. Spherical geometry was used for sodium worth calculations instead of cylindrical geometry because the shift in spectrum due to sodium loss causes a change in reflector savings.

Reactivity coefficients were calculated for several materials using DTF-IV. Table 9.3 gives the change in k_{eff} for a homogeneous change in mass in the core. Material worths in the center test loop ($r < 5$ cm) were also computed; these are tabulated in Table 9.4.

TABLE 9.2. Kinetics Parameters

Parameter	Value
Doppler coefficient (Tdk/dT)	
Reference Core	-3.89×10^{-3}
Reference Core (no Na)	-1.84×10^{-3}
"Dirty" Reference Core ($\delta N_{\text{F.P.}} = 0.003$)	-3.26×10^{-3}
β_{eff}	0.0030
Λ_0 (sec)	3.9×10^{-7}
Na Worth	
Reference Core - Total	-0.0130
Reference Core - $r < 10$ cm	+0.0002
"Dirty" Reference Core - Total	
($\delta N_{\text{F.P.}} = 0.003$)	-0.0093
"Dirty" Reference Core - $r < 10$ cm	+0.0003

TABLE 9.3. Material Reactivity Worths

<u>Material</u>	<u>$\delta k/\delta M$ (kg/l)</u>	<u>Material</u>	<u>$\delta k/\delta M$ (kg/l)</u>
^{239}Pu	1.095	Be	0.238
PuO_2	0.879	BeO	0.133
^{238}U	-0.045	Na	0.062
$^{238}\text{UO}_2$	-0.037	B^{10}	-17.294
F.P.	-0.144	B_4C	-2.437
^{240}Pu	0.068	SS	~ 0.0

TABLE 9.4. Material Reactivity Worths in Center Loop

<u>Material*</u>	<u>vol% in Center Loop, at Theoretical Density</u>	<u>δk</u>
PuO_2	6	0.0097
$^{238}\text{UO}_2$	40	-0.0021
B_4C	60	-0.0290
BeO	60	0.0041
SS	60	0.0004

* *Material is assumed to displace sodium.*

2. Effect of Fuel Zoning on Power Distributions and Doppler Coefficients

L. L. Maas

Power distribution, flux distribution, and Doppler coefficients were compared for unzoned and a two-zone ceramic fueled FTR core. Table 9.5 shows the core characteristics and results. Figures 9.1 and 9.2 are the power distributions for the unzoned and two-zone cores, respectively.

TABLE 9.5. Comparison of One and Two-Zone Cores

	Unzoned	Zoned	
		Zone 1	Zone 2
Vol% PuO ₂	5.657	5.0	6.85
²³⁸ UO ₂	19.343	20.0	18.15
SS	25.0	25.0	25.0
Na	45.0	45.0	45.0
Void	5.0	5.0	5.0
Outer Core Radius, cm	54.2	38.32	54.2
Outer Reflector Radius, cm	84.2		84.2
Core Length, cm	86.7	86.7	86.7
Total Refl Savings, cm	22.0		22.0
Volume, liters	800.0	400.0	400.0
L/D	0.8		0.8
kg ²³⁹ Pu	411.5	181.86	249.15
k _{eff}	1.026		1.018
Fertile-Fissile Mass Ratio	3.648	4.267	2.827
$T \frac{dk}{dt}$	-4.15×10^{-3}		-4.24×10^{-3}
P _{pk} /P _{avg} , core	1.439	1.161	1.137
ϕ_{pk}/ϕ_{avg} , core	1.472	1.352	0.993

FIGURE 9.1. Power Distribution--Unzoned Core

FIGURE 9.2. Power Distribution--Two-Zone Core

3. FTR Phase A Critical Experiments

R. A. Bennett, S. Engstrom, and Q. Baird

FTR Phase A critical experiments were started on schedule December 1, 1966, in ZPR-III, Assembly 48, by ANL personnel at NRTS.

The main objective of the Phase A experiments is to provide "clean" control rod worth measurements to supply a basis for experiment-theory correlation. These correlations should either demonstrate the adequacy of present models used in FTR control rod design or provide useful data for necessary improvements of the models. The Phase A, Assembly 48 experiments are outlined below:

Central Control Rod Experiments

- Measurements of spatial worth and reaction rates
- Central sodium column, base point for central rod experiments
- Stepwise insertion of a highly enriched boron rod
- Stepwise insertion of a natural boron rod
- Rod heterogeneity test

Reflector Control Rod Experiments

- Addition of Inconel reflector sectors
- Stepwise insertion of a highly enriched boron rod
- Boron enrichment comparison
- Rod interaction experiments
- Symmetric two-rod experiment
- Measurement of spatial distributions of worth and reaction rates

4. Preliminary Evaluation of Cross-Section Data

W. W. Little, Jr., L. L. Maas, and R. W. Hardie

All FTR neutronics calculations are based on a modified version of the Russian 26-group data compilation. To assess

the accuracy of this cross-section set, numerous critical assemblies have been analyzed. A partial list of the assemblies studied, which cover a wide range of flux spectra, is given in Table 9.6. The prominent spectrum characteristics of these assemblies are tabulated in Table 9.7.

One method of checking cross-section data is to compare calculated and experimental fission ratios. Tables 9.8 and 9.9 give the $^{239}\text{Pu}/^{235}\text{U}$ and $^{238}\text{U}/^{235}\text{U}$ fission ratios for the assemblies listed in Table 9.6. In general, the computed fissile fuel fission ratios are in reasonable agreement with the experiment, whereas the computed $^{238}\text{U}/^{235}\text{U}$ fission ratios are consistently higher than experimental values.

Cross sections can also be evaluated by comparing calculated and experimental reactivity coefficients. Table 9.10 gives the reactivity worth of various materials in the SEFOR critical assembly in terms of δk ; Table 9.11 gives reactivity worths in the ZEBRA Assembly-2 in terms of σ_p . Note that fuel worths are predicted well, whereas the calculated light-element worths--especially ^{10}B --exhibit rather large discrepancies with experimental values.

5. Evaluation of Burnup Calculation Routines

W. W. Little, Jr. and R. D. Kidman

The 1-D ZODIAC F burnup code is now operational on the 1107; the 1-D and 2-D ASSAULT burnup code--using FCC input--is operational on the 7090. ASSAULT can also be used for k_{eff} calculations in X-Y, R- θ , and RZ geometry.

To understand the effect of using different numbers of energy groups on burnup calculations, a brief calculational survey was performed. Unless indicated otherwise, all calculations were for an 800 liter, oxide-fueled FTR operating at

TABLE 9.6. Central Core Compositions, atoms/cm³ × 10²²

Core	²³⁵ U	²³⁸ U	²³⁹ Pu	²⁴⁰ Pu	Be	C	O	Na	Al	Cr	Fe	Ni	Mo
ZPR-III-14	0.4500	0.0330	--	--	--	5.3000	--	--	--	--	0.790	--	--
29	0.2390	0.4800	--	--	--	--	1.3900	--	1.470	--	2.110	--	--
34	0.2250	0.4950	--	--	--	0.7600	--	--	1.540	--	2.110	--	--
FERMI 35	0.1954	0.0141	--	--	--	--	0.3994	0.7825	--	0.7556	3.180	0.2975	--
38	0.3638	3.5052	--	--	--	--	--	--	--	0.1507	0.572	0.0716	--
RAPSODIE 44	0.4480	0.2730	0.2147	0.0103	--	--	1.9580	--	1.027	0.1970	1.467	0.0920	--
45	0.0015	0.7230	0.1070	0.0051	--	2.4970	--	0.6350	--	0.2490	0.928	0.1230	--
45A	0.0015	0.7230	0.1070	0.0051	--	0.8330	--	1.0580	--	0.2030	0.757	0.1000	--
FARET 46B	0.3380	0.5680	0.1622	0.0078	--	1.1140	--	0.8460	--	0.2840	1.057	0.1400	--
SEFOR 47	0.0015	0.7744	0.1480	0.0132	0.568	0.3360	1.5660	0.6630	0.699	0.2731	1.045	0.1080	0.0515
48	0.0016	0.7406	0.1644	0.0106	--	2.0765	--	0.6230	0.011	0.2658	0.990	0.1308	0.0206
ZPR-VI-2	0.2285	0.9853	--	--	--	1.2900	--	0.7910	--	0.2983	1.038	0.1390	--
ZEBRA Core-2	0.2400	1.5000	--	--	--	3.8000	--	--	--	0.1000	0.380	0.0390	--
FTR Oxide	--	0.4728	0.1296	0.0144	--	--	1.2335	1.0280	--	0.4127	1.580	0.1625	--

TABLE 9.7. Neutron Flux Spectrum

	<u>% of Flux Above Indicated Energy</u>		
	<u>0.01 MeV</u>	<u>0.1 MeV</u>	<u>1.0 MeV</u>
ZPR-III-14	94.4	74.8	30.0
29	97.2	73.4	17.5
34	97.2	74.7	17.8
FERMI B 35	97.0	76.3	15.7
38	99.9	85.0	14.0
RAPSODIE 44	99.5	87.5	31.5
45	86.8	59.0	14.6
45A	93.3	67.5	14.8
FARET 46B	99.0	85.5	28.3
SEFOR 47	89.5	61.2	15.0
48	92.1	67.5	18.3
ZPR-VI-2	96.7	73.6	17.0
ZEBRA Core 2	92.8	66.5	17.1
FTR Oxide	93.6	67.0	14.5

TABLE 9.8. Core Center Reaction Rate Ratios,

$$\frac{\bar{\sigma}_f^{239}\text{Pu}}{\bar{\sigma}_f^{235}\text{U}}$$

<u>Assembly</u>	<u>Experimental</u>	<u>Calculated at PNL</u>	<u>E/C^(a)</u>
ZPR-III-14	1.050 ± 2%	1.100	0.940
29	1.060 ± 2%	1.080	0.981
34	1.067 ± 2%	1.090	0.978
FERMI B 35	1.090 ± 2%	1.090	1.000
38	1.170 ± 2%	1.190	0.983
39	1.180 ± 2%	1.180	1.000
RAPSODIE 44	1.186 ± 2%	1.230	0.967
45	---	---	---
45A	1.047 ± 2%	0.980	1.068
FARET 46B	---	1.210	---
SEFOR 47	0.916 ± 2%	0.920	0.996
48	---	---	---
ZPR-VI-2	1.090 ± 2%	1.080	1.010
ZEBRA Core-2	0.987 ± 0.010	1.003	0.958
FTR Oxide	---	0.976	---

(a) Ratio of Experimental Value to Calculated Value

TABLE 9.9. Core Center Reaction Rate Ratios,

<u>Assembly</u>	$\bar{\sigma}_f^{238\text{U}} / \bar{\sigma}_f^{235\text{U}}$		<u>E/C^(a)</u>
	<u>Experimental</u>	<u>Calculated at PNL</u>	
ZPR-III-14	0.0600 ± 2%	0.071	0.838
29	0.0380 ± 2%	0.040	0.960
34	0.0366 ± 2%	0.039	0.926
FERMI 35B	0.0310 ± 2%	0.034	0.912
38	0.0330 ± 2%	0.035	0.944
39	0.0480 ± 2%	0.050	0.960
RAPSODIE 44	0.0820 ± 2%	0.086	0.945
45	---	---	---
45A	0.0279 ± 2%	0.029	0.962
FARET 46B	---	0.078	---
SEFOR 47	0.0264 ± 2%	0.027	0.989
48	---	---	---
ZPR-VI-2	0.0360 ± 2%	0.038	---
ZEBRA Core-2	0.0320 ± 0.0005	0.035	0.912
PTR Oxide	---	0.029	---

(a) *Ratio of Experimental Value to Calculated Value*

TABLE 9.10. SEFOR Central Reactivity Coefficient Comparisons

<u>Sample</u>		<u>Reactivity Worth, mk</u>	
<u>Material</u>	<u>Mass (g)</u>	<u>Experimental</u>	<u>Calculated</u>
Pu-SS-Al	69.1	0.297	0.300
EnUA1	72.1	0.223	0.220
Dep. U	2460.4	-0.535	-0.722
B ₄ C (Enr)	32.4	-1.445	-1.420
Na	91.2	-0.031	-0.043
SS	1026.1	-0.114	-0.152
Fe	1028.0	-0.099	-0.147
Be	240.7	0.082	0.045

TABLE 9.11. ZEBRA Core-2 Small Sample Perturbation Cross Section, σ_p (barns)

<u>Sample</u>	<u>Experimental</u>	<u>Calculation</u>
²³⁹ Pu	3.236	3.384
²³⁵ U	2.282	2.291
²³⁸ U	-0.177	-0.170
¹⁰ B	-3.136	-2.179
C	0.010	0.006
Na	0.005	0.004
Al	-0.009	-0.013
Fe	-0.019	-0.029
Ta	-0.790	-0.863

400 MW_t (homogeneous core, L/D = 0.8). In essence, the results show that it is permissible to use few (2 to 4) energy groups in burnup calculations (Tables 9.12 through 9.15).

TABLE 9.12. Initial k_{eff} (Eight Groups)

DTF IV	1.02319
ZODIAC F	1.01697

TABLE 9.13. Reactivity Change for Various Energy Groups

Groups	- δk After 50 Days		- δk After 100 Days	
	FCC (O-D)	ZODIAC F (1-D)	FCC (O-D)	ZODIAC F (1-D)
2	0.016914	0.018480	0.033634	0.036617
4	0.017274	0.018841	0.034348	0.037298
8	0.017363	0.018936	0.034523	0.037470
12	0.017364	-	0.034527	-
26	0.017364	-	0.034527	-

TABLE 9.14. Spectrum Averaged ^{239}Pu Absorption Cross Section

Groups	$\bar{\sigma}_a$, T = 0		$\bar{\sigma}_a$, T = 50 days		$\bar{\sigma}_a$, T = 100 days	
	FCC	ZODIAC F (a)	FCC	ZODIAC F (a)	FCC	ZODIAC F (a)
2	2.2653	2.2625	2.2668	2.2651	2.2684	2.2675
4	2.2653	2.2587	2.2673	2.2627	2.2703	2.2666
8	2.2653	2.2552	2.2685	2.2603	2.2717	2.2652
12	2.2653	--	2.2687	--	2.2721	--
26	2.2653	--	2.2687	--	2.2721	--

(a) In central region of core

TABLE 9.15. Peak-to-Average Power Densities from ZODIAC F

Groups	T = 0 days	T = 50 days	T = 100 days
2	1.4668	1.4462	1.4266
4	1.4786	1.4570	1.4363
8	1.4753	1.4539	1.4334

6. Algorithms for Collapsing Fast Reactor Cross Sections

W. W. Little, Jr. and R. W. Hardie

Fast reactor cross sections are often compiled in multi-group arrays--the number of groups usually lying between 15 and 40. Since many design calculations--especially multidimensional calculations--do not require such fine energy detail, a study was made of the methods, applicability, and errors involved in collapsing fine group cross sections. The collapsing problem is of special interest in the analysis of large sodium cooled breeder reactors. In these reactors, the sodium reactivity coefficient depends on a precarious balance between spectral and leakage terms. Thus, if few group cross sections are used to calculate sodium worth, the collapsed cross sections must preserve both spectral and leakage effects.

For the calculation of k_{eff} , reaction rates, etc., the usual flux weighting recipe--with reciprocal weighting of the transport cross section--was found to be most suitable. The accuracy of this collapsing method for a large oxide-fueled breeder reactor is illustrated in the first four columns of Table 9.16.

The perturbation in the power densities are similar to those shown for the fluxes in Figure 9.3.

Since the majority of the tests in the FTR will be conducted in the core, it appears that the reflector control rod scheme may be more desirable from the standpoint of their effect on test assemblies. It should be noted, however, that quite significant axial flux perturbations in the materials test region (Row 6) are caused by the reflector control rods.

It can be seen from Table 9.16, however, that the collapsed cross sections computed by flux weighting are totally inadequate

FIGURE 9.3. Beginning of Life Axial Flux Distribution in FTR
for In-Core and Reflector Rods

TABLE 9.16. Parameters Computed from Exact and Collapsed Cross Sections (a)

Parameter	Number of Energy Groups				4 (Perturbation Cross Sections)
	20 (Exact)	12	8	4	
k_{eff}^{-1}	0.0	+0.00009	+0.00356	+0.00517	--
$\bar{\sigma}_f^{239}$	2.01493	2.01494	2.01684	2.01678	--
<u>Worths (δk)</u>					
^{239}Pu (+ 10%)	+0.05321	+0.05392	+0.05366	+0.05377	+0.05311
^{10}B ($\delta N = 2 \times 10^{-4}$)	-0.07663	-0.07489	-0.07550	-0.07579	-0.07798
Na (- 20%)	-0.00126	-0.00090	-0.00104	-0.00077	-0.00121
Be ($\delta N = 10^{-3}$)	+0.00060	-0.00072	-0.00084	-0.00152	+0.00062

(a) Calculations for large oxide fueled fast reactor

for calculating the reactivity coefficient of a moderator (e.g., beryllium or sodium). For such calculations, a new collapsing algorithm was devised. This algorithm, which uses both flux and adjoint flux, is obtained by conserving the fundamental mode reactivity coefficients. This is most conveniently accomplished by introducing a pseudoabsorption cross section in each coarse group.

The last column in Table 9.16 demonstrates that a four-group model using such "perturbation" cross sections is much more accurate for computing moderator reactivity coefficients than corresponding models using flux weighted cross sections.

Many other numerical experiments using these and alternate algorithms have been performed. In general, the two algorithms mentioned were found most appropriate, although other collapsing recipes may yield better results for specific reactors.

B. RADIATION AND SHIELDING

1. Sodium Activation in Primary Coolant System

W. L. Bunch

Activation of the sodium in the primary reactor coolant system has been estimated by using the results of core analyses for an 800 liter, oxide-fueled core operating at 400 MW. Generation of ^{24}Na (per cm^3) within the core was obtained directly by the triple product of the calculated volume average flux, the assumed sodium concentration, and the calculated spectrum weighted sodium absorption cross section. Generation of ^{24}Na external to the core was estimated by considering the calculated total neutron leakage rate from the core and an evaluation of the fraction of these neutrons that would be absorbed within sodium. For the reference core, activation within the core would generate approximately 7×10^{16} atoms of ^{24}Na per second. Assuming the axial reflectors to contain

20 vol% sodium, 80 vol% Inconel, leakage neutrons would be expected to generate about 5×10^{17} atoms of ^{24}Na per second. Thus, total generation rate in the reference system would be approximately 6×10^{17} atom/sec. If it is assumed that the radioactive atoms will be homogeneously distributed throughout the sodium system, the saturated specific activity in a 5800 ft^3 ($1.6 \times 10^8 \text{ cm}^3$) system would be about 3.7×10^9 disintegrations/ cm^3 or 0.1 Ci/cm^3 .

Generation of ^{22}Na by the (n,2n) threshold reaction was also estimated by using the results of the core analyses; however, the calculation is not so straightforward and is subject to more uncertainty. The highest energy group in the core calculations includes all neutrons with energy greater than 6.5 MeV, whereas the threshold energy for the (n,2n) reaction is $\sim 12.4 \text{ MeV}$.

2. Sodium Activity in Closed Loops

W. L. Bunch

Saturated specific activity of the sodium contained in closed experimental loops has been calculated as a function of loop flow parameters based on results of reactor analyses for an 800 liter, oxide-fueled core operating at 0.5MW/liter. Activation calculations assume that every sodium atom follows the same simple cyclic flow path:

$$A = \phi N \sigma t/T$$

where ϕ is the activation flux, N is the number of atoms of ^{23}Na per cm^3 , σ is the effective activation cross section for the neutron flux energy distribution, t is the length of time the sodium resides in the activation flux during each cycle, and T is the loop cycle time. Table 9.17 shows the results of the calculations.

TABLE 9.17. Saturated Sodium Activity in Closed Loops as a Function of Loop Flow Parameters

Loop Heat Removal Capacity, MW	Flow Parameter, t/T		Sodium-24, Ci/cm ³		Sodium-22, μCi/cm ³	
	Minimum	Maximum	Minimum	Maximum	Minimum	Maximum
10	0.0036	0.0054	0.046	0.073	30	43
5	0.0055	0.0091	0.073	0.12	43	73
3	0.0123	0.0192	0.16	0.26	100	160
1	0.0238	0.0330	0.32	0.46	190	270

These calculations serve only to illustrate the strong influence that the currently postulated loop parameters have on the activation level of the sodium. In addition, the core-reflector design will also influence the activation level. No attempt was made to determine the magnitude of this effect.

3. Fast Reactor Library for RIBD and ISOSHLD

W. L. Bunch and D. R. Marr

A fast reactor library has been generated for use with the computer programs RIBD and ISOSHLD to determine gamma shielding requirements associated with fuel irradiated in the FTR. The fission yield factors and burnup cross sections in the library have been selected to approximate the FTR environment. The library, which has been used in the FFTF shield studies completed to date, is discussed in the informal report, "Fast Reactor Library for use with RIBD," BNWL-CC-868, W. L. Bunch and D. R. Marr, October 17, 1966.

A brief survey has been completed of the neutron sources that might be associated with irradiated FTR fuel and test assemblies to determine the neutron shielding problems that might be encountered. The four potential neutron sources

that were considered are: delayed neutrons, spontaneous fission neutrons, neutrons from the (α, n) reaction, and neutrons from the (γ, n) reaction. As indicated in the following discussion, significant neutron shielding will probably be required only if beryllium is present in the fuel.

Delayed neutrons will not require special shielding under the present concept that fuel will only be discharged after the reactor has been shut down. The neutron generation rates due to spontaneous fission and the (α, n) reaction are summarized in Tables 9.18 and 9.19 based on fuel from an 800 liter oxide reactor. For a typical oxide fuel assembly containing about 40,000 grams of uranium and plutonium, the two neutron sources combined would be expected to generate about 1.6×10^6 n/sec. Approximately the same rate would also be generated in an unirradiated fuel assembly. The need for neutron shielding will depend on the handling techniques to be employed; it is unlikely that special neutron shielding would be required for handling the irradiated fuel.

TABLE 9.18. Neutrons from Spontaneous Fission in FTR Fuel

Isotope	Spontaneous Fission Rate, n/gi sec	Neutrons per Fission	Composition Assumed	
			gi/gm	n/gmsec
^{238}U	7.0×10^{-3}	2.0	0.7630	0.011
^{238}Pu	1.1×10^3	2.2	0.0002	0.48
^{239}Pu	1.0×10^{-2}	2.2	0.2100	0.0046
^{240}Pu	4.6×10^2	2.2	0.0230	23.
^{242}Pu	8.2×10^2	2.2	0.0002	0.36

Total = 24.

TABLE 9.19. Neutrons from (α ,n) Reaction in FTR Fuel

<u>Isotope</u>	Reaction Rate, $n/g_i \text{ sec}$	<u>Composition Assumed</u>	
		g_i/g_m	$n/g_m \text{ sec}$
^{238}Pu	1.4×10^4	2.0×10^{-4}	2.8
^{239}Pu	4.5×10^1	2.1×10^{-1}	9.5
^{240}Pu	1.7×10^2	2.3×10^{-2}	3.9

Total = 16.

NOTE:

 g_i = grams of isotope; g_m = grams of plutonium and uranium.

A significant number of neutrons will be generated within irradiated fuel that contains beryllium because of the (γ ,n) reaction. The threshold energy for this reaction is approximately 1.67 MeV; it is estimated that the photon flux above this energy at the center of a typical fuel assembly will exceed 10^{13} photons/cm²sec eight hours after shutdown. The beryllium photoneutron reaction cross section is about 5×10^{-28} cm² for these photons. One volume percent of beryllium in the fuel corresponds to a concentration of about 10^{21} atoms of beryllium per cubic centimeters. The triple product (photon flux above 1.67 MeV, photoneutron cross section of beryllium, and atomic density of beryllium) yields a neutron generation rate of about 10^7 n/sec cm³ per volume percent of beryllium in the fuel. The material contained in a fuel assembly would be expected to generate about 10^{11} n/sec per volume percent beryllium. A neutron source of this magnitude would require special neutron shielding in many of the

concepts for handling irradiated fuel. More detailed analyses need to be made to determine neutron shield requirements for specific beryllium-loaded fuels and fuel handling concepts.

4. Criticality Safety Reviews and Studies

C. L. Brown

Evaluation of a proposed design for decay storage pots for irradiated and nonirradiated subassemblies was begun.

Recommendations were submitted to the design engineers on the design of shipping containers for nonirradiated and irradiated subassemblies. The nonirradiated shipping containers are to accommodate 4 1/2 in. diameter $\times$ 8 ft long subassemblies transported from the fabrication site to the reactor. The containers must assure criticality safety as well as protect the fuel from damage. The basic design will be a metal birdcage or box--about 20 $\times$ 20 $\times$ 92 in.--inside of which is mounted a 5 in. diameter steel watertight tube. At present, only one subassembly will be packaged per container. A spring absorption system will be used to protect the fuel element against vibration and impact.

Calculations to obtain nuclear safety parameters for FFTF fuel pins are in progress. The fuels under study are $\text{PuO}_2\text{-UO}_2$, PuO_2 -stainless steel, and PuC-UC . It has been decided to use the BNL-SRL HAMMER code with Sher plutonium cross sections for these calculations. A comparison (Table 9.20) of calculations to critical experiments with 1.5, 2.0, and 6.6 wt% $\text{PuO}_2\text{-UO}_2$ water lattices indicates the code should be acceptable for nuclear safety surveys.

TABLE 9.20. Comparison of Measured and Computer Criticality

<u>PuO_2/UO_2, wt%</u>	<u>Rod Diam, in.</u>	<u>No. Lattices</u>	<u>Avg Δk, % (Calc-Exp)</u>
1.5	0.372	6	+1.0
2.0	0.508	6	-2.3
6.6	0.337	5	+1.1

CHAPTER X. SAFEGUARDSA. APPROXIMATE METHOD FOR MILD REACTIVITY TRANSIENT ANALYSIS

Safeguards Analysis Staff

Approximate analysis of reactor response to input reactivity transients at power result in the following relationship for the time available for the safety system to react and effectively terminate the transient before exceeding a given fuel temperature limit:

$$\Delta t_a = \frac{1}{a} \left[1 + \frac{\gamma \lambda P_o}{a} \ln \left(1 - \frac{a}{a + \gamma \lambda P_o} \right) \right] - \frac{1}{\lambda} \ln \left[1 + \frac{\gamma \lambda P_o}{a} (1 - F) \right] - \frac{F - 1}{aF}$$

where a = input reactivity ramp rate (\$/sec)

α = prompt negative thermal power coefficient (\$/MW)
(Doppler)

λ = inverse thermal time constant of fuel (sec^{-1})

P_o = initial power level (MW)

F = overpower factor; the ratio of steady-state power at which the fuel temperature limit is reached to the initial steady-state power.

The term $\frac{F - 1}{aF}$ is the time between initiation of the reactivity input and existence of a trip signal, assuming a high flux trip signal set at a power of FP_o .

This relationship points up the interrelationship of prompt negative thermal power coefficient (Doppler), overpower factor, thermal time constant of fuel, and the input ramp which can be accommodated with a given safety system response time.

For a representative case, we assumed the following:
Input reactivity ramp = 5\$/sec; fuel thermal time lag = 0.5 sec^{-1} ; reactivity temperature coefficient, $T \frac{dk}{dT}$ = -4.0×10^{-3} which is equivalent to -2.15×10^{-3} \$/MW at equilibrium. The calculated time to take action, Δt_a , is 0.158 sec.

B. DESIGN BASIS ACCIDENT ANALYSIS

R. A. Harris

Investigations of the so-called "Bethe-Tait" type core accident are being continued. The accidents are quite hypothetical in nature. At present, such accidents are used as a design basis for the reactor shield and containment structure design. Considerable development work is required to describe the behavior of the FTR core during the accident and to obtain the required hydrodynamic and thermodynamic properties of the core materials involved.

Most of the accident analyses are being carried out by APDA. The energy releases for a 600 liter PuO_2 cermet and an 800 liter PuO_2 - UO_2 ceramic core are being calculated, and preliminary results indicate that they are comparable.

C. CONTAINMENT AND SITING

N. G. Wittenbrock

A site analysis study was initiated to provide the preliminary value for the design leakage rate for the containment vessel. A typical exclusion area radius will be assumed. Using the suggested doses of the AEC Site Criteria, 10CFR100, the maximum allowable leakage rate from the containment vessel will be estimated, taking credit for distance from the containment vessel and for atmospheric dilution and dispersion.

APPENDIX A

CY-1966 DOCUMENTS AND REPORTS

APPENDIX A

CY-1966 DOCUMENTS AND REPORTSINFORMAL DOCUMENTS

BNWL-CC-400	E. R. Astley. <u>Interim Reference Concept</u>
BNWL-CC-644	L. H. Gerhardstein. <u>Two Alternate Nuclear Kinetics Simulations</u>
BNWL-CC-691	H. G. Johnson. <u>Evaluation of Water Heat Dump</u>
BNWL-CC-704	W. L. Bunch. <u>Shielding Calculations - Underwater Examination Facility</u>
BNWL-CC-728	L. L. Maas. <u>Reflector Core Schematics</u>
BNWL-CC-730	K. R. Merckx. <u>Temperature Program for Model for Transient Behavior</u>
BNWL-CC-743	W. L. Bunch. <u>Shielding Calculations - FFTF Conceptual Design, Rabbit Facility</u>
BNWL-CC-817	D. R. Dickinson and J. M. Yatabe. <u>Gas Cooling of FFTF Fuel During Inspection</u>
BNWL-CC-871	R. A. Harris and L. L. Maas. <u>Comparison of In-Core and Reflector Control for Cermet</u>

FFTF MONTHLY INFORMAL TECHNICAL PROGRESS REPORTS

BNWL-CC-479 R1	January 1966
BNWL-CC-524 R1	February 1966
BNWL-CC-589 R1	March 1966
BNWL-CC-647 R1	April 1966
BNWL-CC-683 R1	May 1966
BNWL-CC-719 R1	June 1966
BNWL-CC-762 R1	July 1966
BNWL-CC-806 R1	August 1966
BNWL-CC-848 R1	September 1966
BNWL-CC-882 R1	October 1966
BNWL-CC-936 R1	November 1966
BNWL-CC-973 R1	December 1966

FORMAL REPORTS

BNWL-234	W. W. Little, Jr. and R. W. Hardie. <u>FCC-A Fundamental Mode Code for Fast Reactor Analysis</u>
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FORMAL REPORTS, continued

BNWL-252 W. W. Little, Jr. K-III, A Three Region Kinetics Code in Fortran-IV

BNWL-263 C. D. Flowers. Structural Analysis of a Hexagonal Tube or Ring under Internal Pressure

BNWL-264 G. J. Crandall. Preliminary Hydraulic Analysis of the Conceptual FFTF Cermet Driver Fuel Element Design

BNWL-268 Compiled by R. E. Peterson. Fast Flux Test Facility Driver Fuel Meeting March 8, 9, 10, 1966.

BNWL-281 W. W. Little, Jr. and R. W. Hardie. Neutronics Characteristics of Possible Driver Fuels for the Fast Test Reactor (FTR)

BNWL-308 C. L. Goss and H. G. Johnson. Program for Steady-State Process Design and Cost Optimization of FFTF Heat Removal Systems Using Heat Dumps to Air

BNWL-342 W. L. Bunch. Shield Calculations for Irradiated FTR Fuel

BNWL-347 W. W. Little, Jr. R. W. Hardie and L. L. Maas. Multigroup Analysis of Selected Fast Critical Assemblies

BNWL-388 W. L. Bunch. Shielding Calculations for Line Sources of Radioactive Sodium

BNWL-962 W. L. Bunch and D. R. Marr. Fast Reactor Library for Use with "RIBD"

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