BNWL-1461
UC-80

FFTF IRRADIATION TEST PROGRAM
PRELIMINARY

August 1970

AEC RESEARCH &
DEVELOPMENT REPORT

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ERRATA

Page 4, 2nd Paragraph, last line should read:

adequate for non-fuel materials and low-power, blanket pin tests.

Page 9, 4th line from bottom should read:

These irradiations will not have contact instrumentation or other

Page 11, Reference 10 should read:

10. *D. P. Schively, Users' Guide for Irradiation of Experiments in the FTR, WHAN-FR-23, WADCO, Richland, Washington. September, 1970.*

Page B-8, 2nd line should read:

a radial gap. Both tubes are closed at the bottom end. The gap is main-

Page B-10, 3rd line from bottom should read:

flow is directed on the cup to remove decay heat. The in-core portion of the

Page E-3, 4th line from bottom should read:

800 °F. Coolant density, viscosity and heat capacity are thus held

Page Distr-2, 8th listing should read:

4 General Electric Company
 Advanced Products Operation
 Dr. K. Cohen (3)

3 3679 00061 7151

BNWL-1461
UC-80, Reactor
Technology

FFTF IRRADIATION TEST PROGRAM

PRELIMINARY

Compiled by

L. M. Finch

REACTOR AND PLANT TECHNOLOGY DEPARTMENT

FFTF PROJECT

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FFTF IRRADIATION TEST PROGRAM

L. M. Finch

ABSTRACT

A preliminary "FFTF Irradiation Test Program" has been prepared as an initial definition of the first in-reactor irradiation experiments in the FTR. The general program description is appended with more detailed, but tentative, test plans. Appendices also compare the designed test capability against required test capability and evaluate the open test positions for tests and reactor performance optimization.

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ACKNOWLEDGEMENT

The information compiled in this document and its appendices was acquired from many sources within and without the FFTF project over a period of five years. The principal contributors to these data and conclusions were the following members of Battelle Northwest FFTF staff.

J. G. Bradley
L. M. Finch
R. J. Hennig
W. R. Lewis
D. P. Schively

FFTF IRRADIATION TEST PROGRAMINTRODUCTION

The Fast Flux Test Facility (FFTF) is conceived as a coordinated facility for investigations into the behavior of fuels and structural materials when subjected to extreme conditions of neutron flux in environments prototypic of the LMFBR. The goal of this work is a test reactor which will be uniquely suitable for closed loop testing under controlled conditions.* The direct application of the FFTF is fuel development to facilitate the orderly development of fast breeder reactors by the USAEC and industry. The experimental facilities and the FFTF testing capabilities are being planned so that a wide range of tests can be performed. Since the function of the FFTF will be to perform tests which are predicated on future technological development, all such tests cannot be completely defined today.

This description of the experimental program is divided into two main parts. First a short outline of the reactor design provisions for experiments is given, and second, the types of experiments and irradiations which are expected to be performed are described. The appendices and references to this program provide comprehensive details for both the capability of the facility and the requirements for testing under the LMFBR program. The experimental work described as "initial" is that which would be expected during the operation of the reactor with the first core. The word "future" is used to indicate facilities and tests beyond the time period covered by the specific reactor core description in this report.

It is expected that this test program will receive evolving documentation. Subsequent issues of this plan should become more definitive as specific

*A closed loop is defined as an isolated circuit which permits precise control over all experimental parameters (such as flow, temperature, purity, pressure) and which includes in its circuit an in-core test section exposed to known, predictable neutronics. Implicit in the definition is a capability of changing or replacing irradiation specimens in the test section.

tests, sponsors and funding are identified and committed to by the fast breeder program participants.

The test program outlined in the following paragraphs and more explicitly defined in Appendix A, FFTF Irradiation Test Program (Tentative), is the result of an extended effort whereby the apparent needs of the LMFBR developmental program were reconciled against the practical capabilities of the FFTF. The present FFTF testing capability is described in BNWL-1118, FFTF Testing Capability.⁽¹⁾ The safety considerations of the present design have been evaluated for the FFTF Preliminary Safety Analysis Report (in preparation at this date). The test facilities are described in summary manner in Appendix B, Irradiation Testing Facilities. A more detailed description is available in the Conceptual Design Descriptions for Closed Loops, BNWL-500, Vol. 61.

The testing requirements for the FFTF have been acquired from many sources; in fact, from all sources known to have an interest in high energy neutron irradiation, particularly under liquid metal cooled fast breeder conditions. The AEC issued its basic guideline requirements for testing in January, 1966, in the form of the first FFTF project directive.⁽²⁾ Subsequently, potential experimenters in the FFTF submitted their appraisals^(3,4,5,6,7,8,9) of testing requirements as well as recommendations for the test facility itself. These user appraisals were consolidated in a working paper, "Preliminary Correlation of Studies of the Functional Testing Requirements of the FFTF," by R. J. Hennig, December 30, 1966. Appendix C includes abstracts from that requirements compilation. Appendix D compares the present testing capability of the FFTF against, first, the AEC requirements in the January 1966 program directive and, second, the characteristics of the 1000 MWe LMFBR as it is defined by two rounds of conceptual design effort by the industry.

Finally, this program document includes in Appendix E a review of the importance of careful management and planning of open position tests to assure a continuing high performance (neutron flux, fuel cycle economies, fuel management, test optimization) for the test reactor.

Companion documents to this Test Program are the FFTF Users' Guide⁽¹⁰⁾ and the closed loop characteristics.⁽¹¹⁾

DESIGN PROVISIONS FOR EXPERIMENTS

Reactor Test Positions

Nine in-core test positions are provided in the initial core design as shown in Figure 1. A test position includes an access port through the reactor cover with direct access to the core (or reflector). Test facilities may remain in place in a test position on a semi-permanent basis with the exception of one position which will have to be cleared to provide space for reactor fuel handling. Access space is provided on the reactor cover for sodium coolant lines and instrument leads. The nine in-core test positions are suitable for as many as six closed loops, for capsule experiments and for instrumented tests of fuel assemblies cooled by the primary coolant (open loop tests). The test positions may be charged with standard driver fuel assemblies when not in use for tests.

Experimental Facilities for Test Positions

The initial complement of closed loops will consist of two general purpose loops of about 2 MW capacity each. Four closed loop equipment cells will be provided in the containment building with space, services and piping to the reactor. Provisions will be made for the possible addition of two closed loop cells outside the containment building.

Test positions which do not have closed loops installed initially will be available for other uses such as capsules and open loop tests. The users will supply equipment designed for specific experiments. Provisions will be made in the plant design for instrumentation leads and other services to the in-core test positions and to the closed loop equipment cells.

Appendices C and D offer concise descriptions of the FTR and the test apparatus.

Core Driver Positions

Standard driver positions can be used for tests of fuel assemblies using the standard reactor fuel instrumentation. These positions can also be used for non-fuel materials irradiations. The instrumentation for each standard driver fuel consists of a sodium flowmeter, outlet coolant temperature, and fuel failure detection.

Reflector Positions

The first row reflector is cooled from the high pressure plenum to assure that coolant flow is available at fuel cooling rates. The second and third row reflector positions are provided with a reduced coolant flow adequate for non-fuel materials and low power blanket pin tests.

Core Reactivity

Initial core reactivity and reactivity control requirements are based on operation of six test positions, each with a fissile fuel loading equivalent to 15% of a driver assembly, plus three open test positions with 75% fuel loadings. Although such a specific test loading may not be in the reactor, the corresponding reactivity allowance will be available for experiments and irradiations. Appendix E describes the importance of proper open position test management to assure optimum core performance.

EXPERIMENTS AND IRRADIATIONS

Standard Closed Loops

The standard closed loops provide facilities which can be used for a wide range of fuel and non-fuel materials tests. It is expected that emphasis will be placed on fuels tests because of the limited number of loops available. Typical tests are described below.

LMFBR Fuel Tests

The purpose of these tests is to determine fuel performance characteristics under high power conditions.

Typical test specimens:

<u>Core Material</u>	<u>Carbide</u>	<u>Oxide</u>
No. of pins	19	37
No. of active pins	17	34
kW/ft peak	50	25
Power MW	2.1	2.1
Approx. Enrichment %	45	40
Total kg fuel	5	5.5
Flow gpm	165	180
AT °F	350	320
Axial blanket in.	12	12
Pin diameter in.	0.29	0.23
Exposure per 3 month cycle MWd/tonne	25,000	22,000

Test specimens may have contact instrumentation to measure inlet temperature, outlet temperature, sub-channel temperatures, fuel core temperatures, and fission gas pressure. A flux wire may be included to measure neutron flux during operation. Non-fueled pins may be used in test assemblies for instrumentation.

FTR Driver Defect Tests⁽¹¹⁾

The purpose of these tests is to **determine** the effects of reactor operation with a clad failure.

A typical test specimen will consist of a 37 pin bundle of standard FTR fuel pins and spacers. One or more small holes will be drilled in new or previously exposed pins.

The pins will be designed to operate at central driver pin powers. Enrichment will be established for the location of the loop used.

Instrumentation is expected to include fission gas pressure measurement, in-fuel thermocouples, coolant sub-channel thermocouples, and a vortex generator plus sodium sample line to test failed element detection and location by detection of fission gasses in the sodium outlet stream.

Single Pin Experiments

The number of irradiations which can be performed on bundles of identical fuel pins is limited. About two years at 75% operating efficiency are required to reach 100,000 Mwd/tonne for fuel of LMFR enrichments. It is, therefore, expected that some test bundles will be made up of a number of different fuel pins so that a number of single pin tests can be performed concurrently. These tests will involve contact instrumentation up to the capability of the test facility.

Coolant Chemistry Experiments

The closed loop systems are being designed so that each loop has a primary system which is independent of the reactor and other loops. Provisions are planned so that impurities may be introduced and controlled at the desired levels. Experiments can, therefore, be conducted to measure the effects of impurity levels on the clad surface and piping, including those due to corrosion, mass transport, film buildup and fission product deposition.

Special Purpose Closed Loops

It is anticipated that experiments will be performed in up to two special purpose closed loops. Design of such experiments and associated facilities will be developed in the future. Safety analyses will be required for each such experiment including effects of the test on the FFTF.

Open Test Position - Fuels Tests

Instrumented fuels tests are planned for several of the open test positions. The fuel will be cooled by the reactor primary coolant. Because of the risk of contaminating the primary coolant and because of lack of thermal isolation between test assembly and other reactor components the experiments will be restricted in scope as compared to the closed loop tests. During initial operations it is expected that the tests will be limited to fuel that has been demonstrated in other facilities such as closed loops or the EBR II. The tests below are illustrative of the mechanical-thermal-hydraulic capabilities of the FFTF but do not necessarily reflect the administrative restrictions with respect to safety and reliability which will be developed. Typical examples of possible open loop fuel test characteristics are given in Table B-1.

FTF Driver Characteristics

A standard driver assembly will be instrumented to measure sub-channel temperatures, temperature profiles, fission gas pressure and fuel core temperatures. Instrumentation to replace standard driver fuel instrumentation will be included.

Enrichment for the initial test will be established so that the power is the same as the high power driver assemblies. A subsequent test may be enriched to give a power slightly above the maximum driver fuel power to permit increased core specific power limits.

These typical tests involve a bundle of instrument leads 1 1/2 inches in diameter including thermocouples, pressurized gas capillary tubes and a sodium sample line. The test specimen plus leads is essentially a rigid assembly about 40 feet long. Methods of assembly, fuel handling and interim inspection are yet to be defined. The performance of such tests will be contingent on establishing satisfactory mechanical design features and operating procedures.

LMFBR Fuels Tests

A demand for irradiation of larger bundles of fuel pins than can be accommodated in a closed loop is foreseen. These tests would include instrumentation as described above. The in-core fuel duct would have the same OD dimensions as a driver assembly duct. Within this duct, pins of a wide range of pin diameters can be used. The pitch can be adjusted to provide the desired flow at the available core differential pressure. The diameter and pitch can be selected for an exact fit into the duct for bundles of 217, 169, 127 or 91 pins. Under these conditions the fuel weight is about the same as for a driver assembly and there would be little reactivity loss. If an exact pin diameter and pitch is required for test purposes, it may be necessary to use fillers in the duct and some reactivity loss would result. A wide range of fuel types can be irradiated within the mechanical and coolant flow restrictions. Typical test assemblies designed to fit the fuel duct are described in Table B-I.

Open Test Position - Materials Tests

Facilities are provided to irradiate a wide range of capsules designed for materials experiments such as creep, stress rupture and swelling. The experimental apparatus will be designed to fit into the reactor and facilities as built.

Core Driver Positions - Fuels Tests

During the initial operations it is expected that fuel assemblies incorporating minor variations from the driver fuel design will be irradiated in standard driver positions. Such tests will be limited by administrative restrictions so that the fuel will correspond to the description of the first core given in this report with respect to all safety considerations.

Materials Tests

Irradiation of a wide variety of structural materials specimens in the high flux region is foreseen during operation with the initial core. Tentative

plans are to convert at least one driver fuel duct to a holder for irradiation of materials. This specimen would include FFTF surveillance samples as well as R&D tests.

Reflector Positions

Blanket Pin Tests

Blanket pin irradiations have been proposed for reflector positions. These tests would involve pins of various enrichments to simulate the fissile content projected for LMFBR blankets at various points in their irradiation cycle. Analyses during FFTF design will define instrumentation requirements and safety restrictions for these irradiations.

Materials Tests

The first row reflector positions are recognized as the primary location for irradiation of non-fuels materials. It is estimated that as many as 2000 small specimens may be under irradiation shortly after startup. Several reflector assemblies will be converted to holders for materials irradiations. The outside dimensions will be identical with those of the reflector pieces. These will be charged and discharged from the reactor using the plant fuel handling equipment. These irradiations will not have contract instrumentation or other means of measuring operating conditions. Temperatures and flux will be estimated from general reactor measurements. Activation dosimeters will be used to measure cumulative neutron flux and spectra.

TABLE 1. Typical Examples of Open Loop Fuel Test Assemblies

<u>Number of Pins</u>	<u>91</u>	<u>91</u>	<u>127</u>	<u>127</u>	<u>169</u>	<u>169</u>	<u>217</u>	<u>217</u>	<u>217</u>
Pin Diameter, In.	0.38	0.35	0.32	0.30	0.275	0.25	0.24	0.23	0.22
Power, average MW	8	10	8	10	8	10	8	9	10
kW/ft peak	36	45	26	33	20	25	15	17	19
Percent oxide Fissile	--	--	21	--	21	33	22	27	33
carbide	15	23	15	23	16	25	16	20	25
U + Pu, Kg Oxide	--	--	39	--	39	32	37	34	32
carbide	52	44	52	42	52	42	50	46	42

Test Section AT 400 F

Flux: 7×10^{15} nv peak, 1.24 axial peak/average

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8. Attachment to Zetter, D. R. deBoisblanc-27-67, "FFTF Functional Test Requirements," dated January 20, 1967. (drafts transmitted by Idaho Nuclear Corp in September, 1966).
9. TRT-0082-BNW, Preliminary Correlation of Studies of the Functional Testing Requirements of the FFIF, Battelle-Northwest, December 30, 1966.
10. D. P. Schively, Users' Guide for Irradiation of Experiments in the FTR, BNWL-1460, Battelle-Northwest, Richland, Washington. July 1970.
11. R. E. Keyes, FFIF Closed Loop Characteristics, BNWL-1251, June 1970. Battelle-Northwest, Richland, Washington.

BNWL-1461

APPENDIX A

FFTF IRRADIATION TEST PROGRAM

(Tentative)

APPENDIX A

FFTF IRRADIATION TEST PROGRAM (Tentative)INTRODUCTION

The program presented herein is intended to cover the first core and the initial few years of FFTF operation. Tests have been selected to present a general idea of the type of work planned and to define the more difficult tests for which initial approval will be requested. It is therefore of importance to consider whether the maximum test conditions indicated for any test are high enough to cover all probable experimental work. On the other hand it is also important to consider whether the indicated capabilities are above those which might be used or are too high with respect to cost, time and risk.

It is intended that the tests proposed will be within the planned reactivity available for tests. This will require that LMFBF fuels tests in standard driver positions be designed within the required fuel loading restrictions. If excess reactivity is available materials irradiations designated for reflector positions could be moved into the core.

The program has been prepared on the basis of 4 cycles per year. Each cycle represents exposure equivalent to 1/4 of the annual exposure with the FTR operating at 75% efficiency at 400 MW.

TEST PROGRAMIn-Core Test Positions

Figure A-1 shows the test sequences and in-core test position assignments for a four year period after startup of the FTR facility. Figure A-2 provides identification of the test positions on the initial core map. It should be noted that the open test and the closed loop positions are interchangeable except for position 3 which is constrained by refueling system design to be a proximity instrumented open test position.

REFLECTOR TEST POSITIONS

(1) STIF - Flux Monitor

Cycles 1 - 4 Nonfuel materials and instruments 25 tests/year.

Cycles 5 - 10 Some fuels irradiations contingent upon design of the facility.

(2) Flux Monitor - Experimental Positions

Cycles 1 - 2 No experiments - Shutdown Flux Monitors.

Cycles 3 - 10 Blanket pin tests plus shutdown flux monitors.

NONINSTRUMENTED IRRADIATIONSDriver Positions

	Cycle Number			
	<u>1 & 2</u>	<u>3 & 4</u>	<u>5 & 6</u>	<u>7 to 10</u>
LMFBR Fuels (oxide)	2	4	6	8
LMFBR Fuels (carbide)	0	0	0	2
Nonfuel Materials	0-2	0-2	0-2	0-2

Reflector Positions

Nonfuel Materials	5	5	5	5
Blanket Pins	0	0	1	2

Year	Cycle	IN-CORE TEST POSITION NUMBER (Approx. Full Power Flux x 10 ⁻¹⁵)								
		1	2	3*	4	5	6	7	8	9
		(7)	(7)	(7)	(5.7)	(5.7)	(5.7)	(3.7)	(3.7)	(3.7)
1974	1	CL	OT <u>FTR Driver</u>	OT <u>Materials Irradiation</u>	CL <u>Vented Carbide Fuel</u>	OT <u>FTR Driver</u>	OT <u>Enriched FTR Driver</u>	OT <u>Creep Experiment</u>	OT <u>Regular Driver</u>	OT <u>Oscillator</u>
1975	2	<u>FTR Defect Test</u>	thermo-couples fission gas flux wire	Insulated Capsules thermo-couples	alternate irradiations of type A and B 30 kW/ft	thermo-couple flowmeter	10% over power			
	3									
	4									
1976	5	<u>FTR FUEL Failure Propagation</u>	CL <u>Over Power LMFBR Fuels</u> Carbide 40 kW/ft. oxide 50 kW/ft oxide 25 kW/ft			CL <u>Life Test LMFBR Fuel oxide 13-18 kW/ft</u> two types alternate irradiation and examination	OT <u>LMFBR Fuels Instrumented oxide 12-15 kW/ft</u>	CL <u>Materials 1200°F</u>	OT <u>LMFBR Fuel Instrumented Oxide 5-7 kW/ft</u>	CL <u>Sodium chemistry corrosion</u>
	6									
	7									
1977	8						Carbide Fuel 30 kW/ft			
	9									
	10									

CL - Closed loop

OT - Open test position

* - Proximity instrumented only

FIGURE A-1 - FTRF Irradiation Test Program (tentative)

FIGURE A-2 - Core Map with Numbered Test Positions

FFTF IRRADIATION TEST

TEST INFORMATIONFTR Defect Test

The purpose of this test is to determine the effects of reactor operation with a clad failure. This involves questions of reactor safety and operability. One or more small holes will be drilled in new and previously exposed pins. The test specimen will consist of standard FTR fuel pins with a few dummy pins containing instrumentation. A 61 or 37 pin bundle will be used.

Instrumentation

Fission gas pressure in defect pins
 Thermocouple in defect pin core
 Thermocouples in subchannels
 Fission gas, dissolved f.p.'s and solid f.p. in loop sodium
 FEDAL system test - vortex generator and sodium line flux wire
 for measurements during operation.

Operating Conditions

600 to 700 °F inlet '900 - 1000 °F outlet
 Standard driver 30 ft/sec flow.

Enrichment and Specific Power

Standard driver enrichment, 13 kW/ft avg.

Power

1.8 MW

Interim Examination

The test train will be examined at the end of each cycle, defect pin removed and hole drilled in another pin. New pins will replace pins removed for detailed radiometallurgy examinations.

Test References:

<i>Approvals:</i> 	<i>Date</i> 	<i>Core Position No.</i> <u>1</u> <i>Reactor Cycle No.</i> <u>1 thru 4</u> <i>Type of Test</i> <u>Closed Loop</u> <i>Average Flux</i> <u>7×10^{15} n/cm²-sec.</u>
<i>Issue Date</i> June 30, 1970	<i>Supersedes Issue Dated</i> 	<i>FTR -</i> <u>0001 - CL</u>

FFTF IRRADIATION TEST

TEST INFORMATIONFTR Driver Characteristics

A standard driver assembly will be instrumented to measure subchannel temperatures and temperature profile. Fission gas pressure measurements and fuel core temperatures will also be monitored. The purpose of the test is to measure performance of a standard high power driver assembly.

Instrumentation

- 16 Subchannel thermocouples
- 3 Core thermocouples
- 4 Fission gas pressure lines
- 1 Standard FEDAL
- 1 Flux wire for measurements during operation

Interim Examination

No scheduled interim examinations will be performed. One unplanned examination may be required for instrument repair.

Enrichment

Standard driver enrichment.

*Test References:*Approvals:

DateCore Position No. 2Reactor Cycle No. 1 thru 4Type of Test Open PositionAverage Flux $7 \times 10^{15} \text{ n/cm}^2\text{-sec.}$

Issue Date
June 30, 1970

Supersedes Issue Dated

FTR - 000 2 - 0T

FFTF IRRADIATION TEST

TEST INFORMATIONMaterials Irradiation

This test will provide high flux irradiation for plant surveillance and R&D materials. A sample holder with outside dimensions the same as a fuel assembly duct will provide space for 200 or more test specimens. Some insulated capsules may be used to provide elevated temperature irradiation.

Instrumentation

thermocouples for insulated capsules
 thermocouples to measure sodium coolant temperatures
 flux wires for integrated flux and spectrum

Interim Examination

The test assembly will be removed at the end of each cycle. This will be required so that the instrument tree can be moved. Individual test specimens will be removed at the end of their scheduled irradiation and new specimens inserted.

Operating Conditions

The inlet temperature will be reactor inlet (600 °F). The sodium flow will be orificed to give a 200 to 300 °F AT (~ 30 gpm).

Test References:

<i>Approvals:</i> <hr/> <hr/> <hr/> <hr/>	<i>Date</i>	<i>Core Position No.</i> <u>3</u>
		<i>Reactor Cycle No.</i> <u>1 thru 10</u>
		<i>Type of Test</i> <u>Open Position</u>
		<i>Average Flux</i> <u>7×10^{15} n/cm²-sec.</u>
<i>Issue Date</i> June 30, 1970	<i>Supersedes Issue Dated</i>	<i>FTR -</i> <u>0003 - OT</u>

FFTF IRRADIATION TEST

TEST INFORMATIONVented Carbide Fuel

Two types of vented carbide fuel will be irradiated with alternate irradiation and inspection cycles. Extensive loop instrumentation will be provided to measure released fission products in the cover gas, in sodium and plate-out on the piping.

	Type A	Type B
Number of pins	19	37
Pin Diam. in.	0.38	0.29
kW/ft.	30	30
Power MW	1.3	2.5
Flow gpm	125	240
Inlet temperature °F	800	800
Outlet temperature °F	1100	1100
Enrichment	18%	32%

Instrumentation

- 3 fuel core thermocouples,
- coupons for f.p. plate out,
- 3 fission gas pressure lines,
- 6 inlet and outlet thermocouples,
- 1 flux wire for measurements during operation.

Interim Examination

The test assemblies will be inspected at the end of each irradiation cycle.

Test References:Approvals:

DateCore Position No. 4Reactor Cycle No. 1 thru 10Type of Test Closed LoopAverage Flux 5.7 $\times 10^{15}$ n/cm²-sec.

Issue Date
June 30, 1970

Supersedes Issue Dated

FTR - 0004- A

FFTF IRRADIATION TEST

ZEST INFORMATIONFTR Driver Characteristics

A standard driver assembly will be instrumented to measure subchannel temperatures and temperature profile with concurrent precision flowmeter measurements. Fission gas pressure and fuel core temperatures will be maintained. The purpose of the test is to obtain thermal-hydraulic data during in-core operation of a standard high power fuel assembly.

Instrumentation

- 16 Subchannel thermocouples
- 3 Core thermocouples
- 4 Fission gas pressure lines
- 1 Standard FEDAL
- 1 Flux wire for measurements during operation
- 1 Test flowmeter

Interim Examination

No scheduled interim examinations will be performed. One unplanned examination may be required for instrument repair. An additional examination may be required for flowmeter recalibration.

Enrichment

Standard driver enrichment.

Test References:

<i>Approvals:</i> <hr/> <hr/> <hr/> <hr/>		<u>Date</u>	Core Position No. <u>5</u>
			Reactor Cycle No. <u>1 thru 4</u>
			Type of Test <u>Open Position</u>
			Average Flux <u>5.7×10^{15} n/cm²-sec.</u>
Issue Date June 30, 1970	Supersedes Issue Dated	FTR - <u>0005 - 0T</u>	

FFTF IRRADIATION TEST

TEST INFORMATION10% Over Power FTR Driver

A driver fuel assembly will be enriched to give a power 10% above the maximum central heat generation rate. Enrichment will be approximately 26% Pu.

Instrumentation

3 fuel core thermocouples,
15 subchannel thermocouples,
3 fission gas lines,
outlet thermocouples,
standard flow and FEDAL,
flux wire for measurements during operation.

Interim Examination

No planned interim examinations.

*Test References:**Approvals:*

*Date**Core Position No.* 6*Reactor Cycle No.* 1 thru 4*Type of Test* Open Position*Average Flux* 5.7×10^{15} n/cm²-sec.*Issue Date*

June 30, 1970

*Supersedes Issue Dated*FTR - 0006 - OT

FFTF IRRADIATION TEST

ZEST INFORMATIONCreep Experiment

A creep experiment package will be placed in the open test position. This package will include temperature controls, instruments for measuring creep during operation and controls for stress loading of the test specimen.

Instrumentation,

thermocouples,
 pressurized gas tubes for stress loading,
 mechanical read out devices,
 electrical leads for temperature control.

Test References:

<i>Approvals:</i> <hr/> <hr/> <hr/> <hr/>	<u>Date</u>	<i>Core Position No.</i> <u>7</u>
		<i>Reactor Cycle No.</i> <u>1 thru 5</u>
		<i>Type of Test</i> <u>Open Position</u>
		<i>Average Flux</i> <u>3.7×10^{15} n/cm²-sec.</u>
<i>Issue Date</i> June 30, 1970	<i>Supersedes Issue Dated</i>	FTR - <u>0007 - OT</u>

FFTF IRRADIATION TEST

TEST INFORMATIONRegular Driver

A regular driver assembly will be located in this position. This will require an instrument stalk to provide standard monitoring.

*Test References:**Approvals:*

*Date*Core Position No. 8Reactor Cycle No. 1 thru 4Type of Test Open PositionAverage Flux 3.7×10^{15} n/cm²-sec.

Issue Date
 June 30, 1970

*Supersedes Issue Dated*FTR - 0008 - OT

FFTF IRRADIATION TEST

TEST INFORMATIONOscillator Test

An oscillator will be installed for experiments during critical tests and initial power operation.

*Test References:**Approvals:*

DateCore Position No. 9Reactor Cycle No. 1 thru 4Type of Test Open PositionAverage Flux 3.7×10^{15} n/cm²-sec.

Issue Date
 June 30, 1970

Supersedes Issue Dated

FTR - 0009 - OT

FFTF IRRADIATION TEST

ZEST INFORMATIONFuel Failure Propagation

This is a continuation of the FTR defect tests into higher risk areas. The objective will be to investigate the safety of the first core under standard conditions then at 20% over power. In some tests only a few pins (7 or 19) will be used to limit the extent of any release. Prepressurized pins may be used. Mockup tests will be required prior to reactor experiments to define shock and pressure from boiling and to define required in-reactor tube wall thickness. Defects to be introduced will include clad penetration during operation, pin bowing and sub-channel blockage.

Instrumentation

Same as for defect tests

Acoustic monitor for sodium boiling

Operating Conditions

Flows as low as 30 - 40 gpm with 7 pins to give 300 °F AT inlet temperature 600 to 800 °F - outlet 900 to 1100 °F

Enrichment

Standard driver and standard plus 20% 13 kW/ft and 16 kW/ft.

Power

200 KW to 1.8 MW fission heat

Interim Examination

The test train will be discharged and transferred to the inert gas cell at

Test References:

Approvals:

Date

Core Position No. 1

Reactor Cycle No. 5 thru 10

Type of Test Closed Loop

Average Flux 7×10^{15} n/cm²-sec

Issue Date

June 30, 1970

Supersedes Issue Dated

FTR - 0010 - CL

FFTF IRRADIATION TEST

TEST INFORMATION (continued)

the end of each cycle. The pin bundle will be disassembled and defect pins given a detailed examination then sent to radiometallurgy. The next experiment will then be assembled and charged.

Hazards

These tests will involve risks of local sodium boiling and local melting of fuel pins. The tests will be designed to duplicate possible defects which could occur in the FTR core. The specific power will duplicate first core and future FTR core powers.

Test References:

<i>Approvals:</i> <hr/> <hr/> <hr/>	<i>Date</i>	<i>Core Position No.</i> <u>1</u> <i>Reactor Cycle No.</i> <u>5 thru 10</u> <i>Type of Test</i> <u>Closed Loop</u> <i>Average Flux</i> <u>7×10^{15}</u> <i>n/cm²-sec.</i>
<i>Issue Date</i> June 30, 1970	<i>Supersedes Issue Dated</i>	FTR - <u>0010 - a</u>

FFTF IRRADIATION TEST

TEST INFORMATION [A closed loop will be installed and checked out during Cycle 5]

Over Power LMFBR Fuel Tests

Several types of fuel will be tested during alternate cycles. The purpose will be to determine performance characteristics of LMFBR Fuel under high power conditions.

Enrichment will be required to produce the temperature and heat flux conditions. It should be noted that while typical fission heat ratings will be obtained the fast flux exposure will only be 1/2 or 1/4 of that in a large LMFBR.

	A	B	C	D
Core Material	Carbide	Carbide	Carbide	Oxide
No. of pins	19	37	37	61
No. of active pins	17	34	37	55
kW/ft. peak	40	40	50	25
Power MW	1.7	3.3	4.1	3.3
Approx. Enrichment %	20	35	45	40
Total Kg	10	10	10	8.7
Flow gpm	170	300	320	280
AT °F	300	300	350	330
Axial blanket in.	12	12	12	12
Pin diam. in.	0.38	0.29	0.29	0.23
Exposure/cycle MWd/MT	10,000	20,000	25,000	22,000

Test References:

Approvals:

Date

Core Position No. 2

Reactor Cycle No. 5 thru 10

Type of Test Closed Loops

Average Flux 7×10^{15} n/cm²-sec.

Issue Date
June 30, 1970

Supersedes Issue Dated

FTR - 0011 -CL

FFTF IRRADIATION TEST

TEST INFORMATION (Continued)Instrumentation

- 16 Subchannel thermocouples
- 6 Inlet and outlet thermocouples
- 3 Fuel core thermocouples
- 3 Fission gas pressure lines
- 1 Flux wire for measurements during operation

Remarks

The 50 kW/ft test represents the maximum specific power and enrichment proposed in the program.

Test References:

<i>Approvals:</i> <hr/> <hr/> <hr/>	<u>Date</u>	Core Position No. <u>2</u> Reactor Cycle No. <u>5 thru 10</u> Type of Test <u>Closed Loop</u> Average Flux <u>7</u> X 10^{15} n/cm ² -sec.
Issue Date June 30, 1970	Supersedes Issue Dated	FTR - <u>0011 - CL</u>

FFTF IRRADIATION TEST

ZEST INFORMATIONLife Test LMFBR Fuel

Two types of oxide fuel for LMFBR reactors will be irradiated until 50% of the pins have failed. There will be alternate irradiation and examination cycles. At the end of each irradiation cycle the test assembly will be disassembled and the pins tested for clad defects if indications of fission product release were present during the irradiation. This will require a method of detecting the failed pins. Defective pins will be sent to radiometallurgy for detailed examination and replaced with new pins.

Test Assembly

Number of pins - total	61
- fuel	55
diameter	0.230
kW/ft.	15 av. max., 18 limit
enrichment	12% Pu 18% U-235
power	2 MW
exposure per cycle	13,000 MWd/Mt
inlet temperature	850 °F
outlet temperature	1150 °F
flow	200 gpm
axial blanket	12"

Instrumentation

Loop fission gas, dissolved f.p., solids, delayed neutron.
Inlet and outlet thermocouples.
Flux wire for measurements during operation.

Interim Examination

At end of each cycle.

Test References:Approvals:

Date

Core Position No. 5

Reactor Cycle No. 5 thru 10

Type of Test Closed Loop

Average Flux 5.7×10^{15} n/cm²-sec.

Issue Date

June 30, 1970

Supersedes Issue Dated

FTR - 0012 - CL

FFTF IRRADIATION TEST

TEST INFORMATIONLMFBR Oxide Assembly

An LMFBR oxide fuel prototype will be assembled in an FTR driver fuel assembly. Maximum power will be about 15 kW/ft. Axial blankets will be included. The enrichment will be about 12% Pu and 14% U-235. The planned exposure will be 100,000 MWd/Mt.

Instrumentation

3 fuel core thermocouples
 15 subchannel thermocouples
 3 fission gas lines
 coolant outlet thermocouples
 standard flowmeter
 standard FEDAL
 flux wire for measurements during operation.

Interim Examination

No planned interim examination.

Test References:

<i>Approvals:</i> <hr/> <hr/> <hr/>		<u>Date</u>	Core Position No. <u>6</u>
			Reactor Cycle No. <u>5 thru 8</u>
			Type of Test <u>Open Position</u>
			Average Flux <u>5.7×10^{15} n/cm²-sec.</u>
Issue Date June 30, 1970	Supersedes Issue Dated	FTR - <u>0013 - OT</u>	

FFTF IRRADIATION TEST

TEST INFORMATIONInstrumented LMFBF Fuel

An LMFBF oxide fuel assembly will be adapted to FTR driver duct dimensions and irradiated at a low power to duplicate conditions in the fringe of a power reactor. Axial blankets will be included.

Instrumentation

- 16 Subchannel thermocouples
- 3 Core thermocouples
- 4 Fission gas pressure lines
- 1 Standard FEDAL
- 1 Flux wire for measurements during operation.

Interim Examination

No interim examinations will be scheduled. The specific power will be about 7 kW/ft peak. Exposure will be about 6,000 MWd/Mt per cycle.

*Test References:**Approvals:*

DateCore Position No. 8Reactor Cycle No. 5 thru 10Type of Test Open PositionAverage Flux $3.7 \times 10^{15} \text{ n/cm}^2\text{-sec.}$ *Issue Date*

June 30, 1970

*Supersedes Issue Dated*FTR - 0014 - OT

FFTF IRRADIATION TEST

TEST INFORMATIONCoolant Chemistry

A closed loop will be installed and used in studies of the effects of impurities in the sodium, mass transport, etc. The loop will be fueled as required to produce the desired test conditions.

*Test References:**Approvals:*DateCore Position No. 9Reactor Cycle No. 5 thru 10Type of Test Closed LoopAverage Flux 3.7×10^{15} n/cm²-sec.

Issue Date
June 30, 1970

Supersedes Issue Dated

FTR - 0015 - CL

FFTF IRRADIATION TEST

TEST INFORMATIONMaterials Irradiations

At the completion of the creep experiment (FTR-0007 - OT) a closed loop will be installed. This loop will be used to provide an elevated temperature controlled environment for irradiation of a wide variety of materials specimens. Possibly an in-core experiment such as creep will be included.

*Test References:**Approvals:*

*Date**Core Position No.* 7*Reactor Cycle No.* 6 thru 10*Type of Test* Closed Loop*Average Flux* 3.7×10^{15} n/cm²-sec.

Issue Date
 June 30, 1970

*Supersedes Issue Dated*FTR - 0016 - CL

FFTF IRRADIATION TEST

TEST INFORMATIONCarbide Fuel Test

An instrumented carbide assembly will be irradiated in the open test position. This may be a test of an advanced FTR driver assembly or of an LMFBR assembly. The fuel design, specific power and enrichment would be based on information from closed loop tests.

Test References:

<i>Approvals:</i> <hr/> <hr/> <hr/>		<u>Date</u>	Core Position No. <u>6</u>
			Reactor Cycle No. <u>9 and 10</u> Type of Test <u>Open Position</u> Average Flux <u>5.7×10^{15} n/cm²-sec.</u>
Issue Date June 30, 1970	Supersedes Issue Dated	FTR - <u>0017 - OT</u>	

APPENDIX B

IRRADIATION TESTING FACILITIES

TABLE 1

APPENDIX 1

TRANSITION TESTING FACILITIES

APPENDIX B

IRRADIATION TESTING FACILITIES

INTRODUCTION

The purpose of the Fast Flux Test Facility is to provide means for investigating the behavior of nuclear reactor fuels, structural materials, and reactor components when subjected to environmental conditions of liquid-metal-cooled, fast reactors. A principal means of investigation is closed loop-testing under closely controlled and instrument environmental conditions. A related method is by open-loop testing.

A closed loop is an isolated circuit which permits precise control over experimental parameters such as flow, temperature, coolant purity, and pressure. An in-reactor test section is included in the closed-loop circuit for irradiation of tests in a known neutron flux. An open loop (or open test position) is a core position which is cooled by the reactor primary coolant, similar to a driver fuel position, but which has additional specialized test instrumentation.

Because fuel testing by its nature implies uncertainties in the design of subassemblies and operation near the design limits of tests, special attention is given to safety features of test facilities. The basic safety philosophy is to provide sufficient test isolation and independence to limit propagation of adverse consequences of test conditions, including malfunctions and failures, to the reactor core fuel components.

CLOSED LOOPDesign Bases

The design of the FFTF provides equipment space for four closed loops. Two additional areas outside the containment vessel are reserved for future construction of closed-loop cells bringing the ultimate capability of the facility to six. Equipment spaces and permanently installed piping allow

for a heat removal capacity of up to 4 MW for each loop. The projected closed-loop utilization plans for the initial FFTF core include two general-purpose loops of 2 MW capacity installed initially, and up to two special-purpose loops of up to 4 MW capacity for either installation.

This Appendix describes the conceptual design of general-purpose loops of 2 MW. Special-purpose loops would be expected to provide essentially equivalent safety characteristics, but may incorporate different design features.

Principal design data for the closed-loop system are given in Table B-1.

Principal Design Requirements

- A. Closed loop system accidents are not to impair the capability to shut down the reactor.
- B. Each closed-loop system is designed to prevent test meltdown in the event of:
 - (1) A loss of power, or
 - (2) Primary or secondary sodium leakage.
- C. The in-reactor tube, interconnecting piping, and portions of the closed-loop system used for the prevention of test meltdown are designed to withstand the effects of the Design Basis Earthquake without loss of cooling capability.
- D. The primary closed-loop heat removal components, with the exception of the in-reactor tube, branch-arm, and jumper piping, are located in a nitrogen atmosphere cell. The cell and primary system components are located within the FTR containment system.
- E. The closed-loop in-reactor tube is designed to maintain integrity after a test or system failure, including meltdown of a test.
- F. Operating pressures are such that any leakage between primary and secondary loops will be from the secondary to the primary.
- G. The secondary system is designed to serve as a part of the containment boundary.

TABLE B-1. A Summary of Closed-Loop Design Parameters

<u>Test Element</u>	<u>Initial Loops</u>
Test Power Level, MW	2
Test OD, In	2.5
Test Outlet Temperature, °F	1200
Test Length (Active), Ft.	3
Fissile Material (At Peak Flux Position), Kg.	2.5
<u>In Reactor Tube</u>	
Design Temperature, °F	1200
Design Life, Yrs.	1
Length, Ft.	39
Assembly OD, In.	4.7
Material	Type 316 SST
Design Pressure, Psi	250
<u>Primary Piping</u>	
Coolant	Sodium
Design Temperature, Hot Leg °F	1200
Design Temperature, Cold Leg °F	1000
Design Pressure, PSI	300
Material	Type 304 SST
<u>Secondary Piping</u>	
Coolant	Sodium
Design Temperature, Hot Leg °F	1200
Design Temperature, Cold Leg °F	1000
Design Pressure, Psi	300
Material	Type 304 SST
Size, In.	3

Primary Pumps

Number
 Type
 Total Developed Head, Psi
 Flow, Gpm

Initial Loops

2
 Electromagnetic
 250
 300

Intermediate Heat Exchanger

Type

 Duty, MW
 Design Temperature °F
 Material

Tube in Shell,
 Primary Sodium
 On Shell Side
 2
 1200
 Type 304 SST

Secondary Pump

Type
 Total Developed Head, Psi
 Nominal Flow, Gpm

Electromagnetic
 50
 300

Air Blast Heat Dump

Type

 Duty, MW
 Design Temperature, °F
 Material

Finned Tube,
 Forced Draft
 2
 1200
 Type 304 SST

Codes And Standards

Each closed-loop system is designed and constructed to the applicable requirements of ASME Boiler and Pressure Vessel Code Section III, Nuclear Vessels, and ASA Standard Code for Pressure Piping Section B.31-7, Nuclear Power Piping, and special, high-temperature design criteria. Primary loop components and piping conform to the requirements for Class A vessels and Class I piping, respectively. Secondary loop components and piping conform at least to the requirements for Class C vessels and Class II piping, respectively.

General Description

The closed-loop systems circulate sodium coolant through in-reactor tubes containing irradiation test specimens. Each closed-loop system consists of three subsystems; the tube to contain the experiment, the primary heat transport system to transfer heat from the irradiation experiment to an intermediate heat exchanger, and the secondary heat transport system which absorbs heat from the primary system sodium and rejects it to the atmosphere. Figure B-1 shows the flow diagram of a typical closed-loop system.

In-Reactor Tube Description

The tube assembly consists of a flow tube within a pressure tube assembly arranged to form a re-entrant flow system. Inlet and outlet openings are provided in the top ends of the pressure and flow tubes, respectively. A test assembly is introduced into the tube assembly through the top of the flow tube. The test is supported on the end of a hanger tube (also called the "instrument stalk") which has axial spacing fins and contains appropriate instrumentation leads, gas leads, pressure sensor capillaries, and flux monitor thimble. Coolant flow is upward through the test assembly to the outlet (see Figure B-2).

FIGURE B-1 Closed Loop Test Schematic

FIGURE B-2 -
In-Reactor Closed Loop
Pressure Tube

The pressure tube assembly consists of two concentric tubes separated by a radial gap. Both tubes are closed at the bottom end. The gap is maintained by spacers or insulating material in the core region, and by spacers over the remainder of the tube. The tube assembly is positioned in the reactor cover through a cover penetration, and hangs from a spool piece installed in that penetration. Additional radial positioning is provided at the bottom of the tube by a hexagonal duct which fits into the core support structure; areas of hard facing on this duct mate with the corresponding areas on adjacent driver assemblies and transmit radial core restraint forces.

The assembly is designed to retain molten test fuel in a refractory metal cup at the bottom of the tube without rupturing the outer pressure tube wall. In the unlikely event of a test meltdown, backup cooling is provided by the reactor coolant as it flows past the bottom of the tube.

The flow tube assembly consists of two concentric tubes similar in construction to the pressure tube. The outer wall of the flow tube has axial fins which serve to locate it radially within the pressure tube. The flow tube hangs from an internal ledge in the spool piece.

The tube assemblies are connected to branch arm piping and thence to the ex-reactor closed-loop system through a tubular spool piece. The spool piece is attached to the reactor cover and contains internal seating surfaces on which the pressure and flow tubes are supported and sealed. The shield plug is positioned into the spool piece and holds the flow and pressure tubes in place. The final closure between the pressure tube and spool piece is made by remote seal-welding.

Primary Heat Transport System

In the primary system, sodium flows from the in-reactor tube to the intermediate heat exchanger, which transfers heat from the primary sodium to the secondary sodium. Sodium flows in a parallel path to the two parallel

electromagnetic pumps. One of the two parallel paths is direct; the other contains a sodium supply tank between the IHX and pump. Sodium from the pumps passes through check valves. The sodium flows are rejoined before entering the reactor tube. Sodium then passes through a strainer and into the reactor tube. The strainer removes particles from the sodium that otherwise could enter the tube and cause blockage. Piping connects the primary system with the in-reactor tube. The system piping and components are arranged to minimize the potential for introduction and circulation of gas bubbles in the coolant. Primary components are located in a nitrogen-filled, concrete vault near the reactor.

During decay heat removal, circulation in the primary system is provided by either electromagnetic pump. Each pump is capable of providing 100% of the flow but in normal operation each pump provides 50% of the loop flow. When required, flow is increased automatically. Both pumps are powered by both normal and emergency power.

A purification system is provided for each closed-loop primary system. Sodium is bypassed from the main primary system and circulated through the purification components. A plugging meter is installed in a primary sampling loop for measurement of sodium oxide content.

Secondary Heat Transport System

Each closed-loop system is provided with a separate secondary heat removal system.

Secondary sodium flows from the IHX through a dump heat exchanger (DHX) and then to an electromagnetic pump. Sodium from the pump discharge returns to the IHX. The DHX and pump are located outside the containment vessel. The DHX is elevated to enhance natural circulation capability.

One common purification and sodium processing unit, on a "feed-and-bleed" basis, serves all closed-loop secondary systems. The rate of secondary sodium "feed-and-bleed" is set by the closed-loop requirements.

Safety Considerations

The closed-loop system is designed to prevent a meltdown of the test due to loss of power or leaks within the system. Meltdowns occurring due to failure of the test itself cannot be precluded. The closed-loop tube is designed to prevent the escape of any material from a cladding failure. The reactor can adjust to the reactivity effects of a 2-MW-rated fuel test failure by automatically positioning the control rod system. However, to minimize damage to the closed loop, the reactor will be shutdown upon indication of test failure. Instrumentation is provided to input critical process parameters to the Plant Protection System and reactor control system.

The in-reactor tube is hung from the vessel head and penetrates the upper core grid plate. A top closure device maintains tube position within a stepped plug. The weight of the tube assembly is greater than the normal upward hydraulic forces to prevent dislodging of the tube assembly in the event of closure failure.

In order to prevent interference with the operation of the control and safety rod drive mechanisms, the spool piece closure and branch-arm piping are designed with sufficient strength to prevent distortion or displacement of these components as a result of a loop accident. Heavy shielding around the pipes guards against movement of the branch-arm piping.

The tube is designed to sustain test failure without propagation of mechanical damage to adjacent channels. A meltdown container in the lower end of the tube retains the fuel within the confines of the tube itself. Reactor coolant flow is directed on the cut to remove decay heat. The in-core portion of the tube is also designed to transfer decay heat from the in-core experiment to the reactor pool. This arrangement allows the ex-reactor system cooling function

to be terminated for refueling.

Any sodium leakage between the primary QLS sodium and the reactor primary sodium through the in-reactor tube will be detected by means of sodium inventory control of the primary QLS sodium. Inventory is measured by level sensors at the primary storage tanks.

OPEN TEST POSITIONS

Design Bases

The FTR open test positions provide for fuel and materials irradiation in a sodium environment prototypic of future LMFBR's. Neutronically, hydraulically, and thermally, the open test positions are integral components of the reactor core. There can be up to seven open test positions in the initial core. Six of these positions can be equipped with contact instrumentation. Proximity instrumentation is provided at the seventh open test position. Each test position is supplied with reactor primary coolant in the same manner as the driver fuel channels. For location of the open test positions refer to Figure A-2 in Appendix A.

Principal design data for open tests are given in Table B-2.

Principal Design Requirements

- A. Two independent hold-down mechanisms are provided.
- B. Tests will be designed such that nuclear, thermal, and hydraulic limitations established for adjacent driver fuel assemblies will not be exceeded.

TABLE B-2 - Design Parameters of an Open Test Position

Heat Removal

Power, Mw	Maximum of 10
-----------	---------------

Coolant

Inlet, °F	Core Inlet (600°F)
Test Outlet, °F	± 100°F from adjacent driver elements
Test ΔT, °F	400 (maximum)
Test ΔP, psi	140 (core)
Maximum Flow, gpm	1000

Fuel

Maximum temperatures	
Oxides, incipient melting	4850°F
Carbides, gas bonded	3000°F
Carbides, sodium bonded	2350°F

Material

Type 304 SST (core)

General Description

The open test positions provide the capability for irradiating specially instrumented assemblies. Six of the seven test positions provide for the use of contact instrumentation while the seventh position has provision for proximity instrumentation only.

Typical instrumentation for a contact-instrumented open test position could include process instrumentation consisting of an eddy-current flowmeter, up to five outlet-coolant thermocouples, a sodium sample line, a 112-in. diameter flux monitor thimble, and the experimenter's instrumentation. This latter may include as many as six thermocouples and two fission gas pressure lines.

The in-core portion of a contact-instrumented open test position assembly is similar in cross section and identical in extension configuration to a driver fuel assembly. Above the core, a transition in cross section from hexagonal to circular cross-section occurs with a reduction in size. The open test position coolant flow exits from the duct near the shape transition. The instrumentation leads, flux thimble, and gas lines pass up into the head through an instrument stalk which is sealed and held down in a manner similar to a closed loop. The sodium sample line mates radially with a sample line spool piece (located between two metallic face seals). The instrumentation leads and flux thimble are brought out through a connector in the top of the shield plug closure.

A proximity instrumented open test position consists of an assembly designed for a specific experiment within the geometric boundary of a driver fuel assembly, and identical in exterior configuration. Instrumentation will include flow and outlet temperature.

BNWL-1461

APPENDIX C

ABSTRACT OF EXPERIMENTER'S REQUIREMENTS
FOR FFTF TESTING

APPENDIX C

ABSTRACT OF EXPERIMENTER'S REQUIREMENTS
FOR FFTF TESTINGINTRODUCTION

The contents of this Appendix are abstracted from the FFTF working paper entitled "Preliminary Correlation of Studies of the Functional Testing Requirements of the FFTF," by R. J. Hennig, January 11, 1967. The data have value, **i f** only to emphasize the urgent need for another experimenter survey for high energy neutron irradiation requirements. **I t** was apparent in 1966, and **i t** is equally apparent today, that the degree of realism applied to testing requirements by **potential** experimenters depends upon the level of immediate commitment in funds and technical personnel.

INTRODUCTION TO WORKING PAPER - JANUARY 1967

The basic objective of this study was to determine the functional test requirements of the FFTF, **i.e.**, What is to be tested? What test environments are needed? And what observations of test behavior are needed? An important consideration in this study was to achieve maximum assurance that potential FFTF applications had not been over-looked. For this purpose, all **participating** organizations were encouraged to consider the broadest scope consistent with their capabilities and PNL's schedule requirements.

Six organizations participated in this study: Atomics International, Atomic Power Development Associates, General Electric Company, Idaho Nuclear, Inc., Los Alamos Scientific Laboratory, Westinghouse Electric Corporation. Figure C - I indicates the basic programs considered by each organization. Figure C-II indicates the technical functions and/or components considered by each organization.

FIGURE C-1

POTENTIAL PROGRAMMATIC APPLICATIONS OF THE FFTF

Applications

CENTRAL STATION FBR

GW 1
LP
GW Steam
A
L He
CO₂

RESEARCH

WL Solid state physics
PI
Physical metallurgy
Physical chemistry)
Nuclear physics
L Plasma physics
L Biology

"DIRECT" CONVERSION

Thermoelectric
GW Thermionic
L
MHD
EGD
Fuel cell

SPACE POWER - A(General)

Hg
G Li
G Rb
K
Ne
A
He + Xe
Steam
G N₂
Na

all

SPACE PROPULSION

LW Solid fuel
Plasma fuel .

ISOTOPES

GI

WEAPONS

L

MILITARY

W

OCEANOLOGY

FUSION

L

A = AI
G = GE
W = W

L = LASL
I = IN
P = APDA

FIGURE C-II

LMFBR Cores

A = AI L = LASI.
G = CE I = IN
W = W P = APDA

SUMMARY OF CONTRACTORS' FUELS TESTING REQUESTSAtomic Power Development Associates (APDA)

This site submitted ~ 17 to 26 percent of all the fuels tests. About half of the APDA tests involved closed loops. All of the tests were connected with the LMFBR Program. The vast majority of the tests utilized oxide, carbide, or nitride fuel.

Idaho Nuclear Corporation (IN)

About one percent of all fuel tests were furnished by this site and all the tests involved closed loops. The tests were all associated with the LMFBR Program. The IN fuel tests tended to favor the use of nitrides, followed by oxides and carbides, and then by sulfides, phosphides, and others.

Westinghouse (W)

This site contributed about two percent of all the fuels tests. More than half the tests involved closed loops. The majority of the tests utilized carbide or nitride fuel and were associated with the LMFBR Program. A few tests, using oxide fuel, were in connection with the steam-cooled FBR Program. There was an unspecified (but large number) of tests indicated for the Space Reactor Program which would use carbide and nitride fuels.

Los Alamos Scientific Laboratory (LASL)

About two percent of all the fuels tests were furnished by this site and all involved closed loops. Almost all the tests used molten fuel (Pu-Co-Ce alloy) and were associated mainly with the LMFBR Program but some (~ 30%) were with the Mobile Reactor Program. A few tests called for oxide fuel and were associated with the Space Reactor Program. An unspecified (but larger) number of tests using carbide fuel were indicated for the LMFBR Program.

Atomics International (AI)

This site submitted ~ 6 to 7 percent of all the fuels tests and more than a third involved closed loops. All the tests were associated with the LMFBR Program. With respect to fuels to be tested, AI favored carbides, followed by oxides, then nitrides and molten plutonium.

General Electric Company (GE)

This site contributed ~ 63 to 71% of all the fuels tests, thus the entire analysis is heavily weighed by their input. The majority of tests were associated with the LMFBR Program and most tests involved carbide or oxide fuel, followed by nitride and other types of fuel. A few fuel test needs were indicated for the Steam-Cooled FBR, Gas-Cooled FBR, and Space Reactor Programs. These tests involved mainly oxide, carbide, and nitride fuels. The great majority of the GE tests were of the open loop type (including capsules). They indicated a significant useage (about half of the non-closed loop tests) of short-time (five minute to ~ 50 day) tests, in particular in axial positioner facilities.

SUMMARY OF CONTRACTORS MATERIALS TESTING REQUESTSAtomic Power Development Associates (APDA)

Five of the seventeen tests proposed were materials tests, or approximately 30%. A series of tests at 750°F and 1200 °F for several unspecified materials, in sodium coolant, were proposed. Half of these tests were to be clad in stainless steel to determine effects of coolant (environment). All of these tests, which amount to 80% of the materials tests proposed were to be the "pre-post" type tests where the mechanical testing would be performed after irradiation. An additional test "series" was specified for "in-reactor" testing of several alloys by fatigue, stress-relaxation, and creep-rupture. A range of testing temperatures was specified from 800 to 1300°F and the coolant environment would be inert gas.

Idaho Nuclear Corporation (INC)

Approximately 30% of the total tests were devoted to structural materials. Almost every alloy currently under consideration for liquid metal systems was included in proposed tests between 800°F and 1400°F. The majority of the tests were to be performed in the 1200 to 1400°F range. In-reactor strain, fatigue tests, and in-reactor fretting or wear tests were described, accounting for about 30% of the materials tests. Most of the tests propose either Na or NaK as a coolant. Pre-irradiation heat treatments and weldments are included in the proposed pre-post type tests.

Westinghouse (W)

Two tests series were proposed **by** covering reflector materials, bearing materials, and refractory metal coating for fuel tests. These tests represent a continuing series of testing over the range of 700 to 1300°F for the conventional alloys and bearing materials, and temperatures to 2300°F on the refractory coated fuel for space applications. The materials testing proposed is subordinate to the fuels tests which account for about 85% of the total tests.

Los Alamos Scientific Laboratory (LASL)

No materials tests were proposed as such, although the alloys which are considered to be candidate for LMFBR applications were specified for cladding in the fuels tests.

Atomics International (AI)

Ten out of the sixty-four tests proposed cover fuel cladding, structural materials, and bearing materials. Test series were proposed for all candidate reactor materials in several alloy classes. Nine series which include twenty-three individual tests were scheduled to extend over the period from FY-1972 through FY-1990. Only one in-reactor tests at relatively low temperatures was proposed. All other tests were pre-post type **ir**-radiation tests at temperatures between 500 and 1400°F. Most of these tests would require capsules or "closed-open loops" for testing.

General Electric Company (GE)

Although the input from this contractor was not in a standard form which was amenable to listing individual tests, **it** was obvious that a broader range of testing was implied than represented by other contractors. Greater weight was placed on dynamic (in-reactor) tests over the temperature range of 800 to 1600°F. Corrosion evaluation and static (pre-post) testing was also covered over the **same** range. A few unusual tests were identified which were applicable to fast reactor technology in general.

SUMMARY OF CONTRACTORS' INSTRUMENTATION REQUIREMENTS

Several prospective users of the FFTF have submitted descriptions of proposed experiments test facility requirements for fuels, materials and instrumentation development that will provide much needed information for the design and operation of liquid metal cooled fast reactors. The experiments were reviewed to determine the instrumentation facilities that should be provided to fulfill the instrumentation requirements as expressed by the experimenter. Of course, not all experimenters desire the same degree of instrumentation for similar tests and some experiments demand the use of a certain type of test environment such as a closed loop. The following is a summary of the test facility instrument capability needed to meet the needs of the many and varied experiments.

FACILITY INSTRUMENT REQUIREMENTS

1. External Readout

Most, if not all, tests require that one or more variables be monitored continuously to provide indication of incipient failure, to allow the separation of cause and effect, and to allow the correlation of test specimen response with reactor operating conditions. The most universal instrument requirement is for readout of fuel pin temperature and/or materials test specimen temperature. The important consideration for the facility operator is that access to and from the core test positions should be provided for instrumentation leads and tubing. Tubing is required for pneumatic signals and/or sodium sampling.

The capability to continue the monitoring of a specimen during its removal from the core and storage before examination should be provided. This information is important because the experimenter needs

to know the maximum temperature or other test variable to which his test specimen was subjected.

The test facility should provide the instrument readout capability of at least 35 separate leads at each core test position as well as two pneumatic lines and one sodium sampling line.

2. Process Control

All the test data required by the experimenters cannot be obtained by instrumenting test specimens. The reactor operating conditions must be known by the experimenter and, in some cases, control of a process system is required to obtain the desired test data. Some typical process variables that must be known by the tester are coolant flow and temperature, neutron flux, coolant purity and reactor operating history. In general, these parameters will not be varied for a particular experiment but still the experimenter needs to know what they are. Coolant purity can be correlated with test specimen corrosion.

The operation of a closed loop should be done at the direction of the experimenter after receiving approval from the reactor operating group. Each closed loop should have its own instrumentation and control equipment to allow greater variation and control of the test specimen environment than can be obtained in the other reactor test positions. Higher temperatures are more easily obtained and controlled in a closed loop.

Some materials test require that the test specimen be maintained at a specific temperature within a variation of 10 °F. This test is used to determine the mechanical characteristics (such as creep, strain, etc.) of reactor materials in a fast flux environment. The temperature of the specimen must be controlled closely for good test results.

3. Data Acquisition

Large amounts of experimental data will be generated for even a small test because the cost of reactor time will tend to promote extensive instrumentation to obtain the maximum amount of information from each test. In addition to the direct test data, the experimenter needs to know the reactor operating conditions so that he can correlate the specimen test data and the specimen environments. The reactor operating conditions as well as reactor operating history should be available to a number of different groups of people. An extensive automatic data processing system will be required to collect, process and record this large amount of test data for a number of experimenters.

4. Building Space for Instrumentation & Control

The largest space requirements are for the placement of the instrumentation and controls for the closed loops. Each closed loop should have the capability of having its controls and instrument readout device separated in its own cubicle.

The instrument racks should be designed with a view to the future development of new and improved instruments. Sufficient space should be provided to facilitate the addition of improved instrumentation as it becomes available.

5. Instrument Clean Rooms

A clean room should be provided where thermocouples and other instruments can be attached to test specimens in a contaminant free environment. Re-instrumentation of some specimens whose instruments have failed may have to be done. This type of facility will expedite the completion of some tests.

FUNCTIONAL TESTS

The majority of the experiments proposed by users of the FFTF fall into one of three general groups---fuels tests, materials tests or instrument development. The following is intended to point out the need for instrumentation on these tests.

1. Fuel Tests

At present there is not enough information available to select a "best" type fast reactor fuel. A number of possible candidates are the oxides and carbides, cermets and metallic fuels. A great deal of experimental work needs to be done to determine such fuel properties as burnup, dimensional stability, power, operating temperatures, lifetime and corrosion. A well instrumented fuel subassembly can shorten testing time and provide considerably more data than just a post irradiation inspection. Instrumentation that will give a continuous readout of fuel pin temperatures at several places on the pin and fuel pin pressure are the most important measurements. Temperature measurement is almost universally requested for both fuels and materials testing. A means of detecting fuel element failure would be a definite benefit. In addition to specimen instrumentation, process instrumentation provides important test data. The more important process measurements are coolant flow and temperature, flux and coolant purity.

2. Materials Tests

The environment for structural materials in a fast reactor is very severe because of the temperature and flux. There is a need to measure the mechanical properties of structural materials specimens in the core of the reactor. To do this requires control of the test specimen temperature within 5 °F and at the same time the specimen must be stressed a known amount and the resulting strain, creep and other mechanical properties measured. In addition flux level at the test position and total experiment dosimetry should be known by the experimenter.

Control materials for fast reactors also need to be tested in a fast reactor environment and similar materials instrumentation will be required.

3. Instrument Development

A great need exists for the development of instrumentation for fast reactors because data from thermal reactors have not been shown to be applicable to fast flux conditions and the measurement of process variables such as temperature have not been attempted to date in a significantly intense fast flux environment. Improved and new instruments are needed to achieve longer operating cycles, operation nearer maximum limits, and observation and correction of undesirable reactor conditions.

The following is a list of possible areas for instrument development:

- . Temperature measurement
- . Flow measurement
- . Fuel failure instrumentation
- . Pressure measurement
- . Vibration
- . Stress-strain monitoring
- . In-core flux monitoring
- . Radiation detection

Emphasis should be given those instruments which provide greatest increase in reactor power and efficiency.

The in-reactor space required for instrument development will include closed loop, open loop, primary sodium and rabbit facility because of the wide variety of irradiation required to test different instruments. A means must be provided to allow a large number of electrical leads to be brought out from the core to the external readout equipment. Some tests will require samples of primary sodium and so a sampling system should be provided as well as tubing for pneumatic lines to various in-core positions.

BNWL-1461

APPENDIX D

COMPARISON OF FFTF REQUIREMENTS WITH
PRESENT DESIGN CAPABILITY

APPENDIX D
COMPARISON OF FFTF REQUIREMENTS WITH
PRESENT DESIGN CAPABILITY

A Special Task Force effort was completed in February 1970 to define an FFTF conceptual design that was consistent with the funding provided for the FFTF. At the conclusion of that effort, the resulting testing capability was evaluated against the basic requirements as set forth in the January 1966, program directive establishing the FFTF project and its goals. The evaluation is presented in tabular form in Table D-1. A similar evaluation is shown on Table D-2 where the characteristics of the 1000 MWe LMFBF's are compared to the FFTF design capability.

TABLE D-1

COMPARISON OF FFTF REQUIREMENTS WITH PRESENT DESIGN CAPABILITY

FAST FLUX TEST FACILITY - PROGRAM DIRECTION JAN 28, 1966							PRESENT FFTF DESIGN					JAN. 1970		
ATTACHMENT A							ATTACHMENT B							
	REQUIRED BY PARAGRAPH	RANGE	REMARKS	REQUIRED BY PARAGRAPH	RANGE	REMARKS	CAPABILITY RANGE	DIRECTIVE NO		REQUIREMENTS YES		SATISFIED PARTIALLY	UNCERTAIN	REMARKS
I. PLANT CHARACTERISTICS														
A. OBJECTIVES														
1. CLOSED LOOP TESTING	A.1.a.(1)		(INFERRED)	B.2.a.(1)		"RELIABLE"	1-6							LOW IN NUMBER & SIZE
2. LMFR APPLICATION (PROTOTYPICALITY)	A.1.a.(1)		(FLUX ONLY)	B.2.a.(2)										
3. PLANT FACTOR (AVAILABILITY)	A.1.a.(3)	75%	(HIGH AND PREDICTABLE)	B.2.a.(3)	HIGH	"MAX. ATTAINABLE"								LESS THAN 75%
4. FUEL FAILURE TESTING	A.1.a.(2)		IN DYNAMIC Na											TYPE OF FAILURE LIMITED
5. SHORT TERM TESTS	A.1.a.(4)		~ SECONDS											NOT INCL. IN INITIAL TEST CAPABILITY
6. EXAMINATION CAPABILITY	A.1.a.(5)		INTERIM, NONDESTRUCTIVE											NO REINSTRUMENTATION, NO ON-SITE EXAM FACILITIES
7. NON-LMFR TESTING CAPABILITY	A.1.a.(6)		ALL US FAST FLUX REQMTS											NOT INCL. IN INITIAL TEST CAPABILITY
B. REQUIREMENTS														
1. CORE														
a. FLUX														
(1) VALUE	A.1.c.(1)	2×10^{16} nv		B.2.b.(1)	$0.7-1.35 \times 10^{16}$ nv		0.7×10^{16} nv							LOW END OF REQUIRED RANGE
(2) SPECTRA	A.1.c.(1)	$75\% \pm 0.1$ Mev		B.2.b.(1)	$75\% \pm 0.1$ Mev		$85\% \pm 0.1$ Mev							
(3) SPECTRA CHANGE				B.2.b.(1)	1000 MW LMFR (SOFTENING)		NONE							
b. FUEL														
(1) COMPOSITION				B.2.b.(2)	$240_{Pu}/Pu > 0.1$		(Pu-U) ₂							
(2) FORM				B.2.b.(2)	CERMET OR CERAMIC		OXIDE CERAMIC							
c. SPECIFIC POWER	A.1.c.(2)	500 TO >2000 w/g Pu												
d. COOLANT														
(1) TYPE	A.1.c.(3)	Na		B.1.c.(4)	Na		Na							
(2) TEMPERATURE	A.1.c.(3)	800-1200°F		B.1.b.(5)	800-1200°F		800-1100°F							LOW END OF REQUIRED RANGE
(3) PURITY	A.1.c.(4)	O, C, N, H	CONTROLLED	B.1.c.(4)	O, C, N, H	CONTROLLED	OXIDE							
e. INSTRUMENTATION														
(1) DRIVER FUEL				B.2.b.(4)		CONTROL & MONITORING	NONE							
(2) COOLANT				B.2.b.(4)		CONTROL & MONITORING	TEMP. & FLOW							
(3) FEDAL				B.2.b.(4)		POTENTIAL CAPABILITY								GAS TAG PREFERRED OVER Na SAMPLING
(4) INCIPIENT BOILING				B.2.b.(4)		POTENTIAL CAPABILITY	NONE							
f. REPRESENTATIVE OF LMFR	A.1.a.(1)		(FLUX ONLY)	B.2.b.(1)										
2. REACTOR														
a. COOLANT														
(1) TYPE				B.1.c.(2)	Na	(REFERENCE COOLANT)								
(2) TEMPERATURE				B.1.b.(5)	800-1200°F	COOLANT BYPASS OF CORE PERMITTED	800-1050°F							CORE BYPASS REQUIRED AT 1100°F CORE COOLANT
(3) PURITY				B.1.c.(4)	O, C, N, H	CONTROLLED	OXIDE							
b. LIFE				B.1.c.(3)	20 YEARS	SHORT LIFE COMPONENTS REPLACEABLE								
c. ACCESSIBILITY														
(1) CLOSED LOOPS			EXTERNAL TO REACTOR	B.3.g.										
(2) INSTRUMENTS - CLOSED LOOPS				B.1.c.(1)	MAKE & BREAK	EXTERNAL TO REACTOR								REMOTE ACCESS RESTRICTED
(3) INSTRUMENTS - OPEN TESTS	A.1.d.(5)	MAKE & BREAK	EXTERNAL TO REACTOR	B.1.c.(1)		(INFERRED)								ACCESS RESTRICTED
(4) INSTRUMENTS - DRIVER FUEL COOLANT				B.1.c.(1)		(INFERRED)								ACCESS RESTRICTED
3. FUEL HANDLING														
a. PRIMARY HTS LOOP														
(1) NUMBER				B.1.e.(1)	2 MIN		3							COMMON AT REACTOR
(2) CAPACITY				B.1.e.(1)	$\leq 75\%$ EACH	(INFERRED)	33 1/3% EA							
(3) COOLANT				B.1.e.(5)	Na		Na							
(4) TEMPERATURE				B.1.e.(6)	800-1200°F	CORE BYPASS PERMITTED	800-1050°F							CORE BYPASS REQUIRED
b. SECONDARY HTS LOOP														
(1) NUMBER				B.1.e.(1)	2 MIN		3							
(2) CAPACITY				B.1.e.(1)	$\leq 75\%$ EACH	(INFERRED)	33 1/3% EA							
(3) COOLANT				B.1.e.(5)	Na		Na							
c. HEAT SINK														
(1) NUMBER				B.1.e.(1)	2 MIN		3							3 MODULES PER SINK
(2) CAPACITY				B.1.e.(1)	$\leq 75\%$ TOTAL	(INFERRED)	33 1/3% EA							
(3) COOLANT				B.1.e.(5)	AIR OR WATER OR WATER-STEAM		AIR							
d. COVER GAS				B.1.e.(2)	A OR H ₂		A							
e. CONTAINMENT														
(1) CONTAINMENT				B.2.e.(3)		MAINTAINED DURING REFUELING								CONTAINMENT VIOLATED DURING COVER OPERATIONS
f. PROOF TEST PLAN														
g. PROOF TEST PLAN				B.1.e.(4)										
II. CLOSED LOOP TESTS														
A. OBJECTIVES														
1. CLOSED LOOP TESTING	A.1.a.(1)		(INFERRED)	B.2.a.(1)		"RELIABLE"								
2. OPTIMIZATIONS OF PHYSICS, RELIABILITY, SAFETY, ECONOMICS														
a. TEST LENGTH				B.3.c.			36 INCH							
b. TEST DIAMETER				B.3.d.			2 1/2 INCH							
c. TEST LOCATION				B.3.f.										
d. TEST COOLANT TEMPERATURE				B.3.k.			$\leq 1400^\circ\text{F}$							
B. REQUIREMENTS														
1. TYPES OF TESTS, FUEL														
a. CAPSULES	A.1.a.(1)	(INFERRED)		B.8.e.(1)										
b. PIN-ENCAPSULATED	A.1.a.(1)			B.8.e.(2)										
c. PIN	A.1.a.(1)			B.8.e.(3)										
d. PIN, CLUSTER				B.8.e.(4)										
e. SUBASSEMBLY	A.1.a.(1)			B.8.e.(5)										LMFR & FFTF SUBASSEMBLY EXCEED TEST POSITION SIZE
2. TYPES OF FUEL MATERIALS														
a. (Pu-U) CERAMICS				B.8.a.	OXIDES, CARBIDES, CARBON NITRIDES, NITRIDES									
b. (Pu-U) ALLOYS				B.8.a.										
3. TYPES OF TESTS, CLADDING														
a. TUBE BURST	A.1.a.(1)			B.8.b.										
b. CREEP				B.8.b.										
c. TENSILE				B.8.b.										
4. TYPES OF CLAD MATERIALS														
a. METAL				B.8.b.		INCL. ALLOYS								
b. REFRACTORY				B.8.b.		INCL. ALLOYS								

TABLE D-1 (Cont'd)

COMPARISON OF FFTF REQUIREMENTS WITH PRESENT DESIGN CAPABILITY

FAST FLUX TEST FACILITY - PROGRAM DIRECTION JAN 28, 1966						PRESENT FFTF DESIGN JAN. 1970					
ATTACHMENT A			ATTACHMENT B			CAPABILITY RANGE	DIRECTIVE REQUIREMENTS SATISFIED				REMARKS
REQUIRED BY PARAGRAPH	RANGE	REMARKS	REQUIRED BY PARAGRAPH	RANGE	REMARKS		NO	YES	PARTIALLY	UNCERTAIN	
5. EXPOSURE			B.8.d.	10^5 MWD/IT						●	
a. FUEL BURNUP			B.8.d.	10^{24} nvt						●	
b. NEUTRON FLUENCE			B.3.a.	4-8		1-6			●		2 MW, 2 LOOPS INITIALLY
6. NUMBER OF LOOPS	A.1.b.(1)	6-8									
7. SIZE OF TEST (LOOP)			B.3.d.	"WILL BE DETERMINED"		4.715 INCH			●		LATTICE SPACE
a. EFFECTIVE DIAMETER (O.D.)	A.1.b.(1)	6 INCH	B.3.e.	"SUFFICIENT TO ACCEPT MODEL SUBASSEMBLY"		36 INCH		●			CORE HEIGHT
b. EFFECTIVE LENGTH (CORE HEIGHT)	A.1.b.(1)	3-4 FT									
8. SIZE OF TEST FUEL PIN			B.3.b.	0.2-0.4 INCH				●			
a. DIAMETER			B.3.c.	2-4 FT		36 INCH		●			
b. LENGTH			B.3.f.	CENTRAL POSITION	(OTHERS TO BE DETERMINED)	ROWS 2, 4, 6			●		NO CENTRAL TEST
9. LOOP LOCATION			B.3.i.	VERTICAL	"ESSENTIALLY"	PLUMB VERT		●			
10. LOOP ORIENTATION											
11. LOOP COOLANT	A.1.a.(2)	Na	B.3.n.	Na	REFERENCE	Na		●			
a. FLUID		"DYNAMIC"	B.3.n.	GAS, STEAM OR LIQ	(NOT IN INITIAL CORE)					●	NO CONCEPT
b. FUTURE FLUID			B.3.k.	1400°F MAX.		1200°F		●			SPECIMEN BYPASS REQUIRED
c. TEMPERATURE											
12. INST. AND CONTROL	A.1.d.(3)	"ADEQUATE PROVISION"	B.3.g.		CONTROL AND/OR MONITOR			●			
a. LOOP ENVIRONMENT			B.3.g.		CONTROL AND/OR MONITOR			●			
b. TEST FUEL			B.3.g.					●			
c. FEEDAL								●			
13. LOOP EQUIPMENT ARRGT			B.3.h.		EXTERNAL TO REACTOR			●			
a. LOCATION, SERVICE EQUIPMENT			B.3.j.	COMPLETE	INCL. HEAT SINK	INDEPENDENT		●			COMMON COOLANT PURIF.EQUIP.
b. INDEPENDENCY			B.3.m.	AS REQUIRED TO MINIMIZE EFFECT OF FAILURE				●			
c. REDUNDANCY			B.3.i.	TO BE DETERMINED				●			INITIAL CORE 2 MW
14. LOOP POWER											
15. FUEL MELTDOWN CAPABILITY	A.1.a.(2)	DYNAMIC SODIUM	B.3.i.					●		●	
III. OPEN POSITION TESTS											
REQUIREMENTS					INTERCHANGEABLE WITH OTHER CORE COMPONENTS	9 - n		●			n = NUMBER OF CLOSED LOOPS
1. NUMBER OF LOOPS	A.1.b.(2)	ABOUT 6	B.4.a.	VARIABLE	(MULTIPLE OF)	4.715 HEX (ACROSS FLATS)			●		MULTIPLES OF CORE LATTICE NOT ALLOWED
2. SIZE OF TEST SPACE			B.4.d.	SAME AS DRIVER SUBASSEMBLY		36 INCHES			●		
a. EFFECTIVE DIA.(O.D.)			B.4.c.	DEPENDENT ON CORE LENGTH							
b. EFFECTIVE LENGTH (CORE HEIGHT)			B.4.b.	0.2-0.4 INCH				●			
3. SIZE OF TEST FUEL PIN			B.4.c.	DEPENDENT ON CORE LENGTH		36 INCHES		●			
a. DIAMETER			B.4.f.	ANY CORE POSITION	(EXCEPT CLOSED LOOPS OR CONTROL ROD)			●			9 POSSIBLE FIXED POSITIONS
b. LENGTH			B.4.f.	VERTICAL	(ESSENTIALLY)	VERTICAL		●			
4. LOCATION			B.4.	REACTOR COOLANT	(SODIUM INFERRED)	REACTOR COOLANT		●			SODIUM
5. ORIENTATION			B.4.g.		SEPARATE INST. CONTROL AND/OR MONITOR			●			
6. COOLANT			B.4.e.		CONTROL AND/OR MONITOR				●		NO INLET CONTROL
7. INST. AND CONTROL	A.1.d.(4)										
a. TEST ENVIRONMENT			B.4.e.						●		
b. TEST FUEL			B.4.e.					●			
c. FEEDAL								●			
8. ARRANGEMENT	A.1.d.(5)	MAKE & BREAK INSTRUMENT	B.4.g.	EXCEPT FOR COOLANT					●		POSSIBLY REMOTE EXCEPT FOR COOLANT
a. LOCATION		EXTERIOR TO REACTOR									
b. INDEPENDENCY											
IV. SPECIAL TEST FACILITIES											
A. OBJECTIVES											
1. SHORT TERM TESTS	A.1.a.(4)	~SECONDS						●			DEFERRED CAPABILITY
B. REQUIREMENTS					REFERENCE TO ATTACHMENT "A" REQUIREMENTS						DEFERRED
1. SHORT TERM TESTS	A.1.b.(3)	RABBIT						●			
a. TYPE	A.1.a.(4)	FUEL						●			
b. MATERIAL								●			
c. EXPOSURE PERIOD	A.1.a.(4)	SHORT TIME	MINUTES, EVEN SECONDS					●			
2. CAPSULE TEST HOLES					TEST HOLES IN REFLECTOR AND IN AREA OUTSIDE REFLECTOR			●			NO CONCEPT
a. LOCATION			NO MENTION	B.6.							
V. EXAMINATION											
A. OBJECTIVES											
1. EXAMINATION CAPABILITY	A.1.a.(5)	INTERIM, NONDESTRUCTIVE						●			NO ON-SITE FACILITIES
B. REQUIREMENTS											
1. INTERIM EXAM OPERATIONS (SPECIMENS)	A.1.b.(5)	FUEL	B.7.a.					●			NO ON-SITE FACILITIES
a. DISASSEMBLY	A.1.b.(5)		B.7.a.(1)					●			
b. CLEANING			B.7.a.(1)					●			
c. NEUTRON RADIOGRAPHY			B.7.a.(2)					●			
d. OTHER N-D EXAMINATION			B.7.a.(2)					●			
e. REINSTRUMENTATION	A.1.b.(5)		B.7.b.					●			
f. REASSEMBLY	A.1.b.(5)		B.7.c.					●			
g. HANDLING			B.7.d.					●			
h. SHIPPING			B.7.e.					●			
2. DRIVER FUEL EXAMINATION			B.7.f.		CLEAN EXAM. STORAGE				●		FUELL SPECIMEN SHIPMENT

TABLE D-2

LIQUID METAL FAST BREEDER REACTORS
(US-AEC 1000 MWe STUDIES)

FEATURE	ALTERNATE	FIRST STUDIES, 1964						FOLLOW-ON STUDIES, 1969					TYPICAL LMFBR	FFTF
		ANL	AC	CE	GE	W	AI	CE	GE	W	B&W	AI		
HTS LOOP ARRANGEMENT	SEPARATE (PIPED) INTEGRAL (POT)	6	6	6		2	5	6		2		3	X	X
HTS PUMP ARRANGEMENT	HOT LEG PUMPS (NO.) COLD LEG PUMPS (NO.)			6	X	4	5	6	X		X	3	X	X
EMERGENCY CORE COOLING LOOPS	INTEGRAL SEPARATE STANDBY	6	6		6					2	6			
		NR	X		X	NR	X		X	NR	X	X	X	X
HTS PIPING	ISOLATION VALVES NO ISOLATION VALVES	NR		X		NR		X		NR				
						X					X			X
REACTOR VESSEL	LOW PRESSURE HIGH PRESSURE	X	X	X	X	X		NR	X	X	X		X	
									X		X	X	X	X
CHANNEL ORIENTATION	VERTICAL	X	X	X	X	X	X	X	X	X	X	X	X	X
CORE LATTICE	TRIANGULAR - HEXAGONAL SQUARE - SQUARE	X	X	X	X	X	X	X	X		X	X	X	X
CORE SUPPORT	FULLY RESTRAINED PARTIALLY RESTRAINED	X	X	X	X				X	NR	X			
						X	X	X				X	X	X
CORE INSTRUMENTATION	FLOWMETER, (ZONED OR COMPLETE) TEMPERATURE COOLANT (Z OR C) FAILED ELEMENT (LOC & DETECT)								C		Z			
		Z	Z	C	C	Z	Z	NR	C	NR	Z	C	Z	C
CENTRAL VOID COMPENSATION	CORE GEOMETRY T-H DESIGN & POWER COEFFICIENT	MOD	AN	GAP	PAN	MOD	MOD		PAN	MOD				
								X			X		X	X
FUEL MATERIAL	CERAMIC ALLOY CERMET		OX	C	OX	C	OX	C	OX	C	OX	OX	Pu, U Oxide	Pu, U Oxide
		U ₂ Ti ₃												
FUEL PIN SPACER	WIRE WRAP (W) BANDS (B) GRID SPACER (G) TUBES (T)			B	W			W	W		W		W	W
		T	G		T		G			G		G		
FUEL CLADDING	ALLOY VENTED OR PLENUM (V) OR (P)	V-Ti	316	19-9DL	316	316	304	316	316	316	304	304	316	316
		P	P	P	P	V	V	V	V	P	V	P	V	P
AVERAGE PIN POWER	kW/ft, LINEAR	6.6	8.6	28.2	10.4	12	11.1	34	9.6	18.5	8.7	9.6	9.5	7.8
REFUELING SYSTEM	UNSHIELDED FHM HOT CELL SHIELDED FHM, NO HC IN-VESSEL FHM ONLY EX-VESSEL FHM ONLY IN & EX-VESSEL FHM'S	X			X	X			X	X			X	X
			X	X			X					X	X	X
		X	X	X	X	X	X							
								X			X	X	X	X

TABLE D-2 (Con't)

LIQUID METAL FAST BREEDER REACTORS (continued)
(US-AEC 1000 MWe STUDIES)

FEATURE	ALTERNATE	FIRST STUDIES, 1964						FOLLOW-ON STUDIES, 1969					TYPICAL LMFBR	FF IF
		ANL	AC	CE	GE	W	AI	CE	GE	W	B&W	AI		
FUEL DECAY STORAGE	IN-VESSEL ONLY EX-VESSEL ONLY IN & EX-VESSEL						X				X			
		X		X		X		X	X	X			X	
			X		X							X		X
ACCIDENT PROTECTION	PRESSURE VESSEL CONFINEMENT SYSTEM	NR	X	X	X	NR	X	NR		X	X	X	X	X
		NR				NR			X					
OUTLET TEMPERATURE		1000	1200	1100	1100	1200	1075	1095	1290	1000	1100	1060	1100	1050
VESSEL ALLOY		NR	316	316	304	304	304	304	SS	NR	304	304	304	304
SHUTDOWN POISON	B ₄ C Ta	X	X	X	X	X	X	X	X	NR			X	X
											X	X		
MELTDOWN PAN		NR	YES	YES	YES	NO	NO	NO	NO	NO	YES	NO	NO	NO
* NR - Not studied or reported.														

BNWL-1461

APPENDIX E

FUEL TESTING IN OPEN FTR POSITIONS

APPENDIX E

FUEL TESTING IN OPEN FTR POSITIONS

J.G. Bradley

INTRODUCTION

The FTR lattice as now conceived consists of 91 hexagonal core positions on a 4.715-inch pitch and similar positions surrounding the core for reflector and shielding assemblies. Nine (9) positions in the core are planned as test positions which may be variously charged with fuels or materials tests. Closed loops may be installed in eight of the nine core positions, up to a total of six closed loops, while fuels or nonfuels testing may be carried out in any of the nine available core positions.

This Appendix is concerned exclusively with the fuels testing potential of core test positions. Potential, as used here, means the maximum use of available space, flux, pressure drop, and heat removal capability. Maxima selected for this study are:

- Space - assemblies fully occupy a "standard" fuel duct,
- Flux - 7×10^{15} nv, central zone spectrum,
- Pressure drop - 100 lb./in² across assembly,
- Heat removal - up to 400 °F coolant AT.

SUMMARY

Simplified conventional calculations of expected pressure drop for flowing sodium through hexagonal array bundles of wire-wrapped, fueled pins were carried out.

Based on a maximum 400 °F coolant temperature rise across the fueled section of the test assemblies and a limit of 100 lb./in² pressure drop, peak power values for different assembly geometries have been derived.

Selection of design operating power as 75% of peak power (e.g. 300 °F coolant AT) was made to select fuel enrichment values for oxide and for carbide fuels of varying assembly geometries. A highlight summary of derived data is shown in Table E-1. More detailed data are given in the accompanying descriptive text.

TABLE E-1

Fuel Test Assemblies for Operation at 300 °F Coolant ΔT

Total Pins	Range of(a) Pin O.D.	Peak Initial(b) Power, kW/ft	Approximate Range of Power-Related Compositions			
			Oxide Fuels		Carbide Fuels	
			U + Pu, kg	% E	U + Pu, kg	% E
Limited by flow velocity maximum 35 ft/sec or assembly ΔP = 100 lb./in ²						
61	0.45 - 0.50	41 - 28	Not operable due to predicted fuel rod melting.		49 - 61	11.7 - 5.6
91	0.35 - 0.40	31 - 20	Predicted fuel melting		44 - 58	15.1 - 6.6
Limited by flow velocity maximum 30 ft/sec or assembly ΔP = 100 lb./in ²						
127	0.30 - 0.31	16 - 14	Predicted fuel melting		45	9.6
	0.32 - 0.35	13 - 9	39 - 46	8.9 - 3.7	to 62	to 2.8
169	0.25	14	Predicted fuel melting		31	12.5
	0.26 - 0.30	13 - 7	26 - 34	13.9 - 4.4	to 45	to 3.3
217	0.20 - 0.25	13 - 8	26 - 41	26.5 - 8.1	34 - 54	19.9 - 6.1
271	0.15 - 0.23	12 - 5	18 - 43	43.8 - 6.3	24 - 57	32.9 - 4.7

(a) Smaller diameter pins could be used in every case at the expense of fuel inventory reduction and, in the smaller pin sizes, impractical dimensions. Larger diameter pins, up to limits of pin-to-pin contact, lead to inoperable values for pressure drop and correspondingly very low potential power generation.

(b) Pressure drop or flow velocity limited.

CONCLUSIONS

It is possible to design and operate fuel tests such that no loss of fuel inventory will result. Such test assemblies would be similar in design to standard driver fuels and exhibit similar operating and power generating characteristics.

Program gains anticipated by designing high fuel inventory fuel tests include:

- Improved statistical validity of test data through the use of larger numbers of specimen pins in each test.
- Lowered demand for driver fuel neutrons with resultant improvement in driver fuel reactivity lifetime. Related effects of this lowered demand include:
 - (a) potentially longer operating periods before refueling is required.
 - (b) more available neutrons for nonfuel tests of materials and components.
- More test parameters can be evaluated concurrently in multiple-pin test assemblies. (Component test specimen compatibility must be of course assured.)
- Fewer charge-discharge operations, for fuel test purposes, will be required.

ASSUMPTIONS AND LIMITATIONS

A few notes concerning the development of data given in this appendix and its attachment are given here to provide an estimate of confidence in their applicability. These are:

- Pressure drop calculations assume coolant temperature constant at 800 °F. Coolant density, viscosity and heat capacity are thus held constant.
- A smooth-tube friction factor is assumed to apply to test sections containing wire wrapped pins. The effect of this calculational

technique has not been evaluated nor is any design known for an actual test assembly. It is noted that the friction factor used is expressed in terms of a Reynolds number which in turn is affected by coolant viscosity values.

- Heat generation rates are based specifically on arbitrarily selected fuel isotopic compositions, chemical compounds and fabricated densities for a single flux spectrum. More detailed fuel design data and knowledge of preselected irradiation test locations are required to better predict heat generation.
- Fuel central temperature calculations used in part to prescribe limits for fuel specific power, calculated as noted above, are based on both uniform heat generation across pins and invariant fuel thermal conductivities. Heat generation $[q(r)]$ varies in actual fuels, particularly in larger pin sizes and, more significant to fuel performance the thermal conductivity varies with both exposure and temperature.
- All pins in each assembly are assumed to be fuel pins. Actual test assemblies will probably be instrumented, at least with flux wires and thermocouples, with some fuel pins thus being displaced.

A wide selection of fuel materials, including mixtures of different compounds, is possible. Subjects of this study are only mixed oxides or mixed carbides of plutonium and uranium having the following selected properties:

<u>Selected Fuel Materials Properties</u>		
<u>Plutonium</u>	<u>Oxide Fuels</u>	<u>Carbide Fuels</u>
Compound	PuO ₂	PuC
Pu239/240	80/20	80/20
Theoretical Density, gm/cm ³	11.46	13.6
Melting Point, °F	4300	3000

<u>Selected Fuel Materials Properties</u> (cont'd)		
	<u>Oxide Fuels</u>	<u>Carbide Fuels</u>
<u>Uranium</u>		
Compound	UO ₂	UC
U235/238	Nat'l	Nat'l
Theoretical Density gm/cm ³	10.96	13.6
Melting Point, °F	5100	4500
<u>Thermal Conductivity of mixtures, cal/(cm)(sec)(C)</u>	0.008	0.04

Enrichment, wherever used here, is:

Wt.% Pu239 + U235 metal in Pu + U metal.

Fuel heat generation rates represent fuel mixtures to selected enrichments with ceramic fuels at 93% of TD and isotopic heat generation⁽¹⁾ as follows:

<u>Isotope</u>	<u>Watts/gm^(a)</u>
Pu239	1078
Pu240	210
Pu241	1386 (not used)
U235	1134
U238	28

(a) $\phi = 7 \times 10^{15}$ - central zone spectrum.

Equations and numerical data appropriate to design and pressure drop calculations for wire-wrapped assemblies, as used in this study, are given in the following section.

(1) W.W. Little, Jr. and L. L. Maas, "Nuclear Parameters and Parametric Studies for the Fast Test Reactor (FTR)", BNWL-1067, April 12, 1969, (Table 4.1).

EQUATIONS AND CALCULATIONSSingle Pin Dimensions

For each selected clad O.D. a clad thickness and fuel rod diameter is calculated to yield area fractions identical with those for 0.23-in. O.D., 0.015 thick clad containing a 0.194-in. diameter fuel rod.

Heat transfer surface = $\pi \frac{d_c}{12} a = 0.25 \pi d_c \text{ ft}^2/\text{pin}$ where d_c = clad O.D., in., and $a \approx 3 \text{ ft.}$ fueled length.

$$\text{Fuel volume} = \frac{\pi}{4} \left(\frac{0.194}{0.23} d_c \right)^2 L \times 36 \times 2.54^3, \text{ cm}^3/\text{pin}.$$

ly Dimensions

Wire wrapped pins are assembled in filled hexagonal arrays. All pin bundles are 4.918-in. from **point-to-point**, including wire wrap, to **fill** a "standard" 4.335-in. flat-to-flat inside dimension fuel duct.

$$\text{Pin pitch, } p = \frac{4.918 + d_c}{n + 1}, \text{ in.}$$

where n = number of rows of pins.

$$\text{Wire diameter} = p - d_c.$$

$$\text{Test section flow area } (A_1) = \text{Duct area } (A_2) - [\text{Pin plus wire displacement}]$$

$$A_1 = 16.27455 - N \frac{\pi}{4} [d_c^2 + (p - d_c)^2], \text{ in}^2$$

where N = total number of pins in assembly.

$$\text{Heat transfer area} = 0.25 \pi N d_c \text{ ft}^2.$$

Pitch to diameter (P/d_c) is a simple ratio ≥ 1.0 .

Overall pin length, $L \approx 93$ -in. including 36-in. fueled section.

Coolant Flow

$$M = \frac{Q}{3600 C_p \Delta T}, \text{ lb./sec}$$

where Q = power Btu/Hr,

$$C_p = 0.301 \text{ Btu/(lb.) } (OF),$$

$$\Delta T = (\text{Outlet} - \text{Inlet}) \text{ Temp., } ^\circ F.$$

Test section flow velocity

$$v = \frac{144 M}{A_1 \rho} , \text{ ft/sec}$$

where $\rho \equiv 52.5 \text{ lb/ft}^3$.

Pressure Drop

Assembly $\Delta P = \text{Non-test section } \Delta P + \text{Test section } \Delta P$

Non-test section ΔP data⁽²⁾ used in this study are:

	Flow, (lb/hr) $\times 10^{-5}$				
	<hr/>				
ΔP , lb/in ²	$\frac{1}{8}$	$\frac{1.5}{9.7}$	$\frac{2.0}{12.1}$	$\frac{2.5}{15}$	$\frac{3.0}{19}$

Test Section ΔP

$$\text{Entrance } \Delta P_C = \frac{K_C M^2}{2g_c \rho A_1^2 \times 144}$$

$$\text{Exit } \Delta P_E = \frac{144 M^2}{2g_c \rho A_1^2} (1 - A_1/A_2)^2$$

$$\text{Pin Region } \Delta P_f = \frac{18 f L P_w M^2}{A_1^3 \rho g_c}$$

where: K_C = resistance coefficient related to A_1/A_2 (3)

Units
1

g_c = 32.17

ft/sec²

f = friction coefficient

1

P_w = "wetted perimeter"

in.

$$P_w = N\pi p + 15.0169$$

(M, A_1 , A_2 , N, p, and ρ are defined previously).

(2) D. L. Koreis to P. D. Cohn. "FFTF Driver Fuel Pressure Drop Calculations", March 20, 1969 (Selected graphic data).

(3) Yevick, Fast Reactor Technology: Plant Design. Fig. 2.97.

Further definition of f , as used in this study, is:

$$f = 0.184/Re^{0.2} \quad \text{where} \quad \text{Units} \quad 1$$

where: $Re = \frac{M D_e \times 3600 \times 12}{\mu A_1} \quad 1$

and $D_e = \frac{4A_1}{pw} \quad \text{in.}$

$$\mu \equiv 0.6207 \quad \text{lb/(hr)(ft) .}$$

All of the test section ΔP increments when summed yield a single equation, representing pressure drop, as follows:

$$\Delta P_{TEST} = 0.04263 \frac{M^2}{A_1^2} \left\{ \frac{0.3486(N_{TP} + 15.0169)^{1.2}}{A_1 M^{0.2}} + \left[K_c + \left(1 - \frac{A_1}{A_2}\right)^2 \right] \right\} .$$

Note:

(Plotted ΔP data, in later sections, represent selected assembly geometry and sodium flow (power) relationships using the above equation for ΔP_{TEST} calculations.)

Fuel Composition

General	Pu239/240 = 80/20
	U235/238 = 0.0072/0.9928 (Nat'l U)
	Fuel fabrication density = 93% TD
Oxide Fuels	Pu/PuO ₂ = 0.88201
	U/UO ₂ = 0.88151
Carbide Fuels	Pu/PuC = 0.95219
	U/UC = 0.952

Solutions of equations for selected enrichments are readily derived for mixed oxides or carbides of Pu and U.

Fuel Power

Using isotopic composition data for selected fuel compositions and enrichments, and heat rates from Reference (1), along with fuel geometries, expression of values **as**: w/cm^3 of fuel; KW/ft of fueled length; $\text{Btu/(hr)(ft}^2\text{)}$ of clad; MW per assembly etc. are readily derived.

Note:

(Appropriate corrections are required for other flux intensities and spectra.)

Fuel Outside to Center AT

$$\Delta T = \frac{qr^2}{4k}, \text{ } ^\circ\text{C}$$

where:

q = specific power	cal/cm^3
r = fuel rod radius	cm
k = thermal conductivity	$\text{cal/(\sec)(cm)(}^\circ\text{C)}$

$q(r)$ is assumed constant

$k(T)$ is assumed constant and for this **study**

$$k, \text{ cal/(\sec)(cm)(}^\circ\text{C)} = \frac{\text{oxide}}{0.008} \quad \frac{\text{carbide}}{0.04}$$

Note:

(Although not specifically a part of this study, the estimated ΔT 's, calculated by the **simplified relationship** shown above, are used to **approximate "limits"** for **KW/ft**, fuel enrichment etc. based on 1100 °F fuel surface and fuel rod centers at the **melting points** of PuO_2 or PuC .)

Test Fuel Design - Geometry

A range of fuel clad diameters from 0.15 + 0.50 in. is expected to cover the probable sizes of interest to fuels development. Considering designs which permit operation within available pressure drop at maximum coolant ΔT of 400 °F leads to the following values for hexagonal array assemblies:

TABLE E-2

Test Assembly Geometry

<u>Range of Clad O.D., In.</u>	<u>Range of Operable (p/d)</u>	<u>Assembly Total Pins</u>
0.50 - 0.45	1.08 - 1.2	61
0.40 - 0.35	1.11 - 1.25	91
0.35 - 0.30	1.08 - 1.24	127
0.30 - 0.25	1.09 - 1.29	169
0.25 - 0.20	1.15 - 1.42	217
0.23 - 0.15	1.12 - 1.69	271

Operating characteristics for the assemblies described in Table E-2 are illustrated in Figs. E-1 thru E-6. These illustrations are indicative of power-coolant flow- ΔP relationships which may be realized when a 400 °F coolant temperature rise (ΔT) is experienced. The effect of reducing permitted coolant ΔT to 350 °F and 300 °F (increased coolant flow required per MW of heat removed) is illustrated for the 271 pin assembly designs in Figs. E-7 and E-8. Similar effects, not illustrated, for lower ΔT 's are appropriate to other assembly designs. Initial FTR operation is planned at 300 °F coolant ΔT . Power generation related to flux and enrichment and pressure drop related to assembly geometry and coolant flow are considered together in the following sections.

Test Fuel Design - Composition

Heat generation and heat flux of materials under irradiation are related to specific power as determined by nuclear cross-sections and neutron flux at

FIGURE E-1. Operating Characteristics of 61-Pin Test Assemblies (400 F° AT)
 (flux, and composition independent)

FIGURE E-2. Operating Characteristics of 91-Pin Test Assemblies (400 F° AT)
 (flux and composition independent)

FIGURE E-2 Operating Characteristics of 127-pin Test Assemblies (400 F° DT) (flux and composition independent)

FIGURE E-4. Operating Characteristics of 169-Pin Test Assemblies (400 F° AT)
 (flux and composition independent)

FIGURE E-5. Operating Characteristics of 217-Pin Test Assemblies (400 F° AT)
 (flux and composition independent)

FIGURE E-6. Operating Characteristics of 271-Pin Test Assemblies (400 F° ΔT) (flux and composition independent)

FIGURE E-7. Operating Characteristics of 271-Pin Test Assemblies (350 F° AT)
(flux and composition independent)

FIGURE E-8. Operating Characteristics of 271-Pin Test Assemblies (300 F° AT)
 (flux and composition independent)

the point of interest. A convenient expression for fuel pin parametric relationships is the **KW/ft** rate.

Fig. E-9 shows peak specific power, **KW/cm³**, for oxide and carbide fuels at 93%TD versus percent fissile metal content. Data represent fuel compositions of this study, $\phi = 7 \times 10^{15}$ nv, at central FTR zone spectrum and specific isotopic heat output values from Reference (1). Axial and radial flux gradients will lead to lower specific power at locations away from the power peak while related spectral softening, with concomitant fission cross-section increases, tends to increase specific power. Figure 10 illustrates **KW/ft** values versus %E for 0.20 in. diameter clad fuels. Relationships among %E, clad diameter and **KW/ft** for any two (2) selected values may be developed by application of factors to the values illustrated or calculated from the following equations.

For oxide fuels

$$\text{KW/ft} = [11.3314 E + 27.67021 d^2] \text{ clad}$$

and for carbide fuels

$$\text{KW/ft} = [14.4971E + 37.0456] d^2 \text{ clad} .$$

The reader should note that Figures 9 and 10, and the above equations represent fuel compositions, single pin geometries, and FTR central zone flux at 7×10^{15} nv for unirradiated fuels specific to this study.

Figs. E-1 thru E-6 illustrate peak operating limits for various assembly geometries at coolant ΔT 's of 400 °F. These limits lie at intersection points of Test ΔP curves with available test ΔP , coolant velocity **ft/sec**, or **Btu/(hr)(ft²)** curves, whichever selected parameter leads to the lowest MW per assembly and **KW/ft** rate as read on the ordinate scales. Power limits at ΔT 's other than 400 may be determined by multiplying observed values by (Selected $\Delta T/400$). Figs. E-7 & E-8 illustrate this effect of selection of alternate ΔT 's for 271-pin test assemblies.

Fuel enrichment levels for illustrated test assemblies are listed in Table E-3 and have been developed within the following constraints:

- Coolant ΔT is 300 °F,
 - Maximum coolant velocity is 35 ft/sec for 61 and 91-pin assemblies and is 30 ft/sec for others,
 - Oxide fuels will not be enriched to levels leading to initial heat rates in excess of 13 KW/ft,
 - Carbide fuels may be enriched to provide initial heat rates to about 40 KW/ft,
 - Test ΔP plus Non-Test ΔP does not exceed 100 lb/in².
- .

FIGURE E-9. Fuel Specific Power Versus Percent E

FIGURE E-10. Fuel Linear Heat Output Versus Percent E (0.20 in. OD fuel illustrated)

TABLE E-3

Initial Composition of Fuel Test Assemblies

Pins per Assembly N	Clad Diameter, d _c , In	Power. Limit, KW/ft	Power-Related Compositions				
			Oxide Fuels		Carbide Fuels		
			Contained Metal, kg	Approx. %E	Contained Metal, kg	Approx. %E	
61	Flow velocity 35 ft/sec or ΔP = 100 lb/in ²						
	0.45	41.4	(36.9) ^(a)	(15.6)	49.1	11.7	
	0.46	39.0	(38.5)	(13.8)	51.8	10.4	
	0.47	36.4	(40.2)	(12.1)	53.4	9.1	
	0.48	33.8	(42.0)	(10.5)	55.8	7.9	
	0.49	30.8	(43.7)	(8.9)	58.1	6.7	
	0.50	28.1	(45.5)	(7.5)	60.5	5.6	
	91	0.35	31.3	(33.3)	(20.1)	44.3	15.1
		0.36	29.3	(35.2)	(17.5)	46.8	13.2
		0.37	27.5	(37.2)	(15.3)	49.5	11.5
0.38		25.5	(39.2)	(13.1)	52.1	9.8	
0.39		23.4	(41.3)	(11.1)	54.9	8.3	
0.40		20.3	(43.5)	(8.8)	57.8	6.6	

TABLE E-3 (Con't)

Initial Composition of Fuel Test Assemblies

Pins per Assembly N	Clad Diameter, d _c , In	Power. Limit, KW/ft	Power-Related Compositions			
			Oxide Fuels		Carbide Fuels	
			Contained Metal, kg	Approx. %E	Contained Metal, kg	Approx. %E
			Flow velocity 30 ft/sec or ΔP = 100 lb/in ²			
127	0.30	15.5	(34.1)	(12.8)	45.3	9.6
	0.31	14.4	(36.4)	(10.8)	48.4	8.1
	0.32	13.1	38.8	8.9	51.6	6.7
	0.33	11.9	41.3	7.2	54.9	5.4
	0.34	10.6	43.8	5.6	58.2	4.2
	0.35	8.5	46.4	3.7	61.7	2.8
169	0.25	13.5	(23.7)	(16.6)	31.1	12.5
	0.26	12.5	25.6	13.9	34.0	10.5
	0.27	11.5	27.6	11.5	36.7	8.7
	0.28	11.1	29.7	10.1	39.5	7.6
	0.29	8.9	31.9	6.9	42.4	5.2
	0.30	7.0	34.1	4.4	45.3	3.3
217	0.20	13.1	25.9	26.5	34.4	19.9
	0.21	12.5	28.6	22.6	38.0	17.0
	0.22	11.6	31.4	18.7	41.7	14.0
	0.23	10.7	34.3	17.1	45.6	12.9
	0.24	9.5	37.3	12.0	49.6	9.0
	0.25	7.5	40.5	8.1	53.8	6.1

TABLE E-3 (Cont'd)

Initial Composition of Fuel Test Assemblies

Pins per Assembly N	Clad Diameter, d _c , In	Power Limit, KW/ft	Power-Related Compositions			
			Oxide Fuels -		Carbide Fuels	
			Metal, kg Contained Metal, kg	Approx. %E	Contained Metal, kg	Approx. %E
			<i>Flow velocity 30 ft/sec or 100 W/in²</i>			
271	0.15	11.8	18.2	43.8	24.2	32.9
	0.16	11.5	20.7	37.2	27.5	28.0
	0.17	11.0	23.4	31.2	31.2	23.5
	0.18	10.4	26.2	25.9	34.8	19.5
	0.19	9.8	29.2	21.5	38.8	16.2
	0.20	9.1	32.4	17.6	43.1	13.2
	0.21	8.1	35.7	13.8	47.5	10.4
	0.22	6.8	39.2	9.9	52.1	7.4
	0.23	5.3	42.8	6.3	56.9	4.7

Notes to Table 2

General. Values for enrichment and heat output are for as-fabricated fuel and will vary as irradiation exposure is accumulated. In some cases, e.g., the lower initial enrichment values, the power at discharge time can exceed initial power generation rate. Limits noted are derived from Figures 1 through 6.

- (a) Numbers shown () for oxide fuels represent those which could be made but which would probably not be used since operation in FTR flux is predicted to result in fuel melting at peak values. Some of these compositions and geometries, when and if of interest to fuel development programs, may be tested at peripheral core positions or in facilities operating at Lower flux Levels.

Supplementary Data

The illustrative figures and descriptions of the paragraphs to follow are given here for ready reference purposes. Limiting KW/ft rates, noted in Table E-3, are based on the following assumptions:

- Fuel thermal conductivity remains constant,
- Maximum fuel center temperature may not exceed the melting point of the plutonium compound in the fuel.
- .. Fuel surface temperature will reach 1100 °F peak.

Figure E-11 illustrates fuel surface-to-center ΔT 's versus KW/ft heat rates for oxide and carbide fuels. When oxide fuels operate at about 21.5 KW/ft the estimated ΔT is 3200 °F and, with the fuel surface temperature of 1100 °F, a 4300 °F (PuO_2 melting point) is reached at the fuel rod center. Allowances for uncertainties in operating instrumentation and control and in local power peaking led to selection of about 13 KW/ft as a "limit" for oxide test fuel. Similar considerations for carbide fuels lead to a 3000 °F central fuel rod temperature at about 63.5 KW/ft adjusted to about 40 KW/ft "limit" to allow for uncertainties.

Figure E-12 relates enrichment and clad diameters to the unadjusted KW/ft peak limits for oxide and carbide fuels.

Pin Geometry - Enrichment - Peak Power relationships are illustrated for oxide fuels in Figures E-13 and E-14 for the range of pin diameters from 0.15 to 0.50 inches from zero to 80% enrichment.

Figure E-13 illustrates theoretically attainable peak power, expressed as KW/ft, for oxide-fueled pins and may be used for selection of enrichment values for pin diameters and compositions within the range of plated data to yield a predetermined initial KW/ft peak heat rate or vice versa.

Figure E-14 presents data of Figure E-13 as $\text{Btu}/(\text{hr})(\text{ft}^2)$ of clad surface at peak values. This figure illustrates that: (a) no limiting value of clad heat flux is likely under anticipated operating conditions and, (b) clad heat flux rates above about $10^6 \text{ Btu}/(\text{hr})(\text{ft}^2)$, which may be of fuels development interest, can probably not be tested with enriched oxide fuels.

Figures E-15 and E-16 are for carbide fuels and provide parallel data for carbide fuels to those for oxide fuels in Figures E-13 and E-14. Figure E-16 shows also that carbide fuels, unlike the oxides, may be operated to clad heat flux levels of interest to advanced cladding materials development.

FIGURE E-11. Fuel Surface-to-Center AT Versus kW/ft

FIGURE E-12. "Limiting" Fuel Enrichment Versus Pin Diameter

FIGURE E-13. Linear Heat Output Versus Clad Diameter (Oxide Fuels)

FIGURE E-14. Clad Heat Flux Versus Diameter (Oxide Fuels)

FIGURE E-15. Linear Heat Output Versus Clad Diameter (Carbide Fuels)

FIGURE E-16. Clad Heat Flux Versus Diameter (Carbide Fuels)

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