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**EVALUATION OF FFTF REACTOR
ASSEMBLY DEVELOPMENT
AND TESTING PROGRAM**

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NOVEMBER 1967

**AEC RESEARCH &
DEVELOPMENT REPORT**

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EVALUATION OF FFTF REACTOR ASSEMBLY
DEVELOPMENT AND TESTING PROGRAM

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TABLE OF CONTENTS

INTRODUCTION	1
SUMMARY	5
DEVELOPMENT AND TESTING REQUIREMENTS	0.1
A. THERMAL-HYDRAULIC DEVELOPMENT.	1.1
B. FUEL SUBASSEMBLY WATER, SODIUM AND MECHANICAL TESTING	2.1
C. CORE AND VESSEL HYDRAULICS	3.1
Core Flow Distribution (C-1)	3.2
Bypass Flows (C-2)	3.16
Fluid Thermal Gradients in the Reactor Outlet Region (C-3)	3.23
Loss of Coolant Accidents (C-4).	3.26
Cavitation (C-5)	3.42
Gas Entrainment (C-6)	3.49
Solids Deposition (C-7).	3.57
Gas or Steam Closed Test Loop Rupture (C-8).	3.64
D. CORE VIBRATION	4.1
E. THERMAL-MECHANICS.	5.1
Plenum Tube Sheets and Reactor Cover (E-1 and E-2)	5.1
Flow Ducts (E-3)	5.11
Integral Structural Analysis (E-4)	5.18
Core Accident Mechanics (E-5)	5.25
Vessel, Nozzle, and Piping Mechanics (E-6)	5.37
Driver Fuel Top Face Hardware (E-7).	5.50
F. MECHANICAL FIT TESTING	6.1
Reactor Cover Arrangement (F-1).	6.1
Configuration of Reactor Internal Components (F-2)	6.5
Self-Welding of Reactor Components (F-3)	6.12
Galling of Reactor Components (F-4).	6.15
Duct Flutter During Insertion (F-5).	6.17
G. CONTROL MECHANISM DEVELOPMENT.	7.1
Drive Mechanism (G-1)	7.4
Scram Latch and Extension Rod (G-2).	7.8
Seals (G-3).	7.10

Accelerator (G-4)		7.14
Poison Element (G-5).		7.17
Decelerator (G-6)		7.18
H. CLOSED LOOP PROTOTYPES.		8.1
Closed Loop Top Face Hardware (H-1)		8.1
Closed Loop In-Reactor Hardware (H-2)		8.6
Closed Loop Interconnecting Piping and Disconnects (H-3)		8.7
Closed Loop Components External to Reactor Vessel (H-4).		8.9
Complete Closed Loop (H-5)		8.11
I. INSTRUMENT DEVELOPMENT.		9.1
In-Reactor Coolant Temperature Sensors (I-1).		9.2
Pressure, DP Flow, Level Sensors (I-2)		9.5
In-Core Sodium Flowmeters (I-3)		9.10
Gas Disengagement for Failed Fuel Monitoring (I-4)		9.14
Delayed Neutron Fuel Failure Monitor (I-5)		9.16
Instrumentation Subassembly (I-6)		9.19
Vibration and Stress Measurements In-Core (I-7)		9.21
Signal Connectors (I-8)		9.25
Reactor Environmental Effect on Signal Cables (I-9)		9.28
Sodium Flow Measurements in Large Pipes (I-10)		9.30
In-Core Neutron Flux Monitors (I-11).		9.32
DEVELOPMENT FACILITIES.		10.1
A. Thermal Hydraulics Test Facilities		10.1
B. Hydraulic Test Facilities		10.10
C. Hydromechanical		10.18
D. Mechanics		10.19
E. Mechanical Fit		10.20
F. Control Mechanisms		10.21
G. Closed Loop		10.21
H. Instruments		10.21

RECOMMENDED PLAN OF SCHEDULING		11.1
Hydraulic Core Mockup		11.1
Seven-Duct Environment and Mechanism Mockup.		11.2
Reactor Cover Mockup		11.2
Large Heat Transfer Loop		11.3
Closed Loop Facility		11.3

LIST OF FIGURES

1	Typical FFTF Split Core Annular Inlet Arrangement	3
2	Typical FFTF Split Core Piped Inlet Arrangement	4
3	Summary of Overall FFTF Reactor Assembly Testing Schedule	13
4	Summary of Key Existing Facilities Utilization	15
5	Summary of Key New Facilities Utilization	17

LIST OF TABLES

I	Problem Identification	10
II	Summary of Proposed Thermal-Hydraulics Test Program	19
III	Summary of Proposed Fuel Subassembly Flow Test Program	21
IV	Summary of Proposed Hydraulic and Vibration Test Program	23
V	Summary of Proposed Mechanics Test Program	27
VI	Summary of Proposed Mechanical Fit Test Program	29
VII	Summary of Proposed Control Mechanisms Test Program	31
VIII	Summary of Proposed Closed Loop Development Test Program	33
IX	Summary of Proposed Instrument Development Test Program	35
10.I	Required Dynamic Facilities	10.2
10.II	Required Static Facilities	10.6
10.III	Partial List of Reactor Model Study Water Flow Facilities	10.12
10.IV	Partial List of Existing Dynamic Sodium Facilities	10.25
11.I	Existing PNL Water Loops and Static Mockups	11.4
11.II	New Hydraulic Core Mockups	11.5
11.III	New Static Mockups	11.6
11.IV	Existing PNL Sodium Facilities	11.7
11.V	Existing Off-Site Sodium Facilities	11.8
11.VI	New Sodium Mockups	11.9

EVALUATION OF THE FFTF REACTOR ASSEMBLY
DEVELOPMENT AND TESTING PROGRAM

INTRODUCTION

Evaluations of various concepts for the Fast Flux Test Facility have led to decisions that the initial core will be of a dispersed split core configuration. The mechanical design of the core involves many considerations such as: vibration, fretting, erosion, corrosion, cavitation, structural integrity, mechanical and thermal stresses, dimensional tolerances, control rod mechanisms, reactor cover penetrations and seals, in-core instrumentation, closed test loops, and proper coolant distribution among and heat removal from the various fuel subassemblies and other core components.

An updated evaluation has been made to identify the overall program of experimentation needed to resolve FFTF reactor assembly design problems. Although based to the maximum extent possible on state of the art, the intricacies and difficult requirements of the design are such that proof testing of the reactor assembly components and systems are absolutely required to support and confirm design, and to provide a high level of assurance that the reactor will function satisfactorily at the design operating conditions.

The objective of this report is to present an overall reactor assembly engineering development and testing program required to support the FFTF design and construction. This program includes:

- 1) Identification of the reactor design problems where required design data must be obtained by experimentation.
- 2) Descriptions and technical justifications of proposed tests.

- 3) An overall test schedule commensurate with the FFTF schedule.
- 4) Identification of testing facilities to be used.

The bounds of the study (as shown in Figures I and II) include the reactor vessel, cover, vessel internals, control rods, driver fuel, core, core support, and closed loops. The report is limited to the evaluation of engineering development requirements exclusive of those areas primarily concerned with materials, chemical behavior, or nuclear physics. Following the Summary and Conclusions, the report is subdivided into the following major categories:

- 1) Fuel subassembly thermal-hydraulics
- 2) Fuel subassembly-hydraulics, sodium and mechanical
- 3) Core and vessel hydraulics
- 4) Hydromechanical (vibration)
- 5) Thermal-mechanics
- 6) Mechanical fit testing
- 7) Control mechanisms
- 8) Closed loops
- 9) Instrument development
- 10) Facilities.

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FIGURE 1. Typical FFTF Split Core Annular Inlet Arrangement

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FIGURE 2. Typical FFTF Split Core Piped Inlet Arrangement

SUMMARY AND CONCLUSIONS

Most of the important reactor assembly design problems which require development or testing programs to support the FFTF design are believed to have been identified. They are summarized below along with the experimental actions needed to solve the problems, and facilities needed for the work.

Problems and Solutions

A. Thermal Hydraulics

Heat transfer measurements in sodium are needed for prototypical and nonstandard electrically-heated simulated FTR fuel pin subassemblies, to verify suitable dimensions and design features of the fuel subassemblies, and specify coolant flow conditions required to prevent failure. Of particular interest are local clad hot-spot conditions, local boiling and voiding, hydraulic instability, and the maximum safe heat flux. Flow studies using water and prototypical fuel subassemblies are planned to measure pressure drop and cross channel mixing.

B. Fuel Subassembly Flow and Mechanical Testing

Long-term flow tests in hot sodium are needed to demonstrate adequate fuel subassembly designs to resist corrosion, erosion, and adverse dimensional changes due to thermal cycling. Vibration tests in air, and in flowing water and sodium, are needed to design for positive damping.

C. Core and Vessel Hydraulics

A substantial program of flow testing with water is required to determine the flow distributions, thermal gradients, cavitation, gas entrainment and solid deposition within the core. The effects of loss-of-coolant accidents as provided by piping or vessel failures and gas or steam closed loop ruptures also require water flow testing. Initial studies will be performed with feature models of selected potential problem areas. These

will be followed by tests conducted in a half-scale model and a full-size 1/4 sector hydraulic core mockup.

Some problem areas will require further testing in a sodium environment to supplement data obtained during water testing. Large scale core testing in sodium was originally planned. However, PNL has accepted the RDT decision that funds cannot be provided for the Sodium Core Mockup. The first chance to test the integral system in sodium and the final verification testing will necessarily include use of the reactor itself during an initial non-nuclear startup period. Based on time periods for non-nuclear shakedown testing in other fast reactors, a schedule slippage of the order of one year may be required to carry out the testing program.

D. Hydromechanical (Vibration)

Vibration of fuel subassembly flow ducts, control rods, baffles, shielding, reflectors and duct supports will cause mechanical fatigue, galling and fretting which foreshortens the useful lives of the structural elements. Dynamic testing in both water and sodium will be performed to determine potential vibration problems; the results will be used to indicate required design changes or to verify the adequacy of mechanical hydraulic design characteristics.

E. Thermal Mechanics

Experimentation using photoelastic models to measure and supply inputs to develop and verify analytical models of the plenum tube sheet(s), reactor cover, driver fuel, driver fuel topface hardware and vessel will be used to formulate the analysis of the more complex reactor systems. The integral core analysis requires a test model of the core structure to obtain the coupling between structural components and to verify the applicability of the analytical models. This testing will be conducted in the hydraulic core mockup facilities.

F. Mechanical Fit

The assembly and disassembly of the reactor internal and external components require testing to insure the adequacy of mechanical fitup. Problems which result from the compactness of the core and the close tolerance requirements, as well as the auxiliary effects of chemical, metallurgical and hydraulic behavior of the components can result in self welding, galling and shroud flutter.

The reactor cover mockup will be required to evaluate the arrangement for assembly fit, interferences, special tools, fuel handling clearance and assembly sequence. The full-size hydraulic core mockup (1/4 sector) and the seven-duct sodium environment and mechanism mockup will be used to determine the prototypic FFTF fuel duct mechanical insertion and removal problems.

G. Control Mechanisms

The control rod drive, seals, accelerator, decelerator and scram release will initially be tested as separate components in air, water and sodium environments to provide data necessary for the timely completion of preliminary design. The assembled prototype control mechanism will be tested in water and stagnant and flowing sodium to determine the working interrelationship of the individual components. Compatibility of the control mechanism with the driver assembly and driver closures will be verified in the seven-duct environment and mechanism mockup. Final testing will be performed in the reactor to determine the relationship of the control mechanism with other reactor components such as reactor vessel and cover, closed loops, gamma shields, electrical and instrumentation leads and fuel handling machine.

H. Closed Loops

The uniqueness and complexity of the closed loops coupled with the interaction of other reactor components and systems require an extensive prototype testing program to assure successful operation. Topface hardware sealing techniques, in-reactor hardware and tubes, interconnecting piping and disconnects, and external components including pumps, heat exchangers, valves, etc., will be tested in water and sodium environments as individual components during development and for design verification. A complete closed loop prototype must be tested to demonstrate compatibility and operability of individual components when integrated into a complete system. This closed loop mockup will duplicate maximum temperatures, pressures, flows, purity levels and spatial layouts anticipated for closed loops in reactor.

I. Instrument Development

Instrument development and testing programs are required to demonstrate, and in some cases to improve, the life and reliability of sensors for temperature, flow, pressure, liquid level, in-core neutron flux, delayed neutron fuel failure, vibration and stress, and for signal cables and connectors in the unprecedented environment and confined space of the FFTF. Hot gas, static and flowing sodium, and in-reactor testing at EBR-II are planned to provide initial acceptable reactor instrumentation. Final verification testing will be performed as part of the total system in the FFTF. This is the only location where the full flux environment is available.

The significant problem areas of the reactor as identified above in A through I are summarized in Table I, Problem Identification. The Summary of Overall FFTF Reactor Assembly Testing Schedule is presented in Figure 3. Each of the major development problems is identified and the milestones are denoted in this

figure. The Summary of Key Existing Facility Utilization for implementing the FFTF development program in a timely manner is given in Figure 4. The available PNL water and sodium facilities and their schedule of use are part of Figure 4.

Offsite existing facilities at the LMEC and ANL will be used for a major portion of the sodium testing as depicted in Figure 4.

Some of the FFTF development testing program requires water and sodium facilities and prototypes which do not now exist due to size or specific requirements unique to the FFTF configuration. The hydraulic core mockups and sodium facilities and prototypes required at PNL to meet these needs are listed below and in Figure 5, Summary of New Facilities Utilization.

Hydraulic Core Mockup

Partial feature assemblies

1/2-scale vessel

Full-scale vessel, 1/4 sector

Subassembly Flow Mockup

Large Heat Transfer Loop

Closed Loop Prototype

Reactor Cover Mockup

Seven-Duct Sodium Environment and Mechanism Mockup

Control Rod Mechanism Test Stand

Closed Loop Static Mockup

FFTF Sodium Facilities Building

A tabulation of tests, test objectives, test description, major variables, information to be gained, facilities required and schedules is provided in Tables II through IX.

The tables and text which follow in the remainder of the report provide additional detail in discussing the design problems, the proposed tests for their solution, and the information to be gained from the tests.

TABLE I. *Problem Identification*

- A. Thermal Hydraulic Development
 - 1. Fuel clad temperature limit
 - 2. Local coolant voiding and propagation
 - 3. Loss of coolant and power excursion accident
 - 4. Pressure drop and cross channel mixing
 - 5. Critical heat flux test verification
- B. Fuel Subassembly Flow and Mechanical Testing
 - 1. Cladding erosion due to flow
 - 2. Cladding corrosion due to purity and temperature
 - 3. Fretting corrosion
 - 4. Vibration/wear
 - 5. Thermal shock
- C. Core and Vessel Hydraulics
 - 1. Flow split between reactor ducts
 - 2. By pass flow around core
 - 3. Fluid thermal gradients in reactor outlet region
 - 4. Loss of region coolant accidents
 - 5. Cavitation
 - 6. Gas entrainment
 - 7. Solids deposition
 - 8. Gas or steam closed test loop rupture
- D. Core Vibration
- E. Thermal Mechanics
 - 1. Lower tube sheet assembly
 - 2. Reactor cover
 - 3. Flow ducts
 - 4. Integral structure analysis
 - 5. Core accident mechanics
 - 6. Vessel
 - 7. Driver fuel top face hardware

TABLE I (contd)

F. Mechanical Fit Testing

1. Reactor cover component arrangement
2. Configuration of reactor internal components
3. Self welding
4. Galling
5. Shroud flutter during insertion

G. Control Mechanism Development

1. Reactor control drive
2. Scram control drive
3. Seals
4. Accelerator
5. Poison section vibration
6. Deceleration
7. Prototype test
8. Production test
9. Mechanical configuration and environmental test
10. Preoperational proof testing

H. Closed Loop Prototype

1. Top-face hardware
2. In-reactor hardware
3. Interconnecting piping and disconnects
4. Components external to reactor vessel
5. Complete closed loop

I. Instrument Development

1. In-reactor coolant temperature sensors
2. Pressure transducers
3. In-core sodium flowmeter
4. Gas disengagement for failed fuel monitor
5. Delayed neutron fuel failure monitor
6. Instrumentation assembly

TABLE I (contd)

7. Vibration and stress measurements
8. Signal connector
9. Signal cables
10. Sodium flowmeter in large pipes
11. In-core neutron flux monitoring

** FFTF Milestones - based on FFTF PERT Charts (Rev. 14)

Fuel Subassembly

- (1) - Pin design complete 7/69
- (2) - Pin Procurement starts 1/70
- (3) - Subassembly design 9/69
- (4) - Subassembly procurement 3/70
- (5) - First Core 9/71

Vessel

- (6) - Vendor design/fab. starts 7/69
- (7) - Installation starts 7/71

Cover

- (8) - Design procurement starts 12/68
- (9) - Fabrication complete 3/71

Nuclear Control

- (10) - Procurement starts 1/70
- (11) - Tests start 1/71
- (12) - Tests complete 7/71

Closed Loop

- (13) - Procurement starts 10/69
- (14) - Tests start 4/71
- (15) - Tests complete 8/71

Instruments

- (16) - Procurement starts 7/69
- (17) - Tests start 1/71
- (18) - Tests complete 7/71

Safeguards

- (19) - Final DBA 8/69
- (20) - Start final Safeguards Analysis 8/70
- (21) - Complete final Safeguards Analysis 6/71

FIGURE 3. Summary of Overall FFTF Reactor Assembly Testing Schedule

* See Table I for problem identification

FIGURE 4. Summary of Key Existing Facilities Utilization

* See Table I for problem identification

FIGURE 5. Summary of Key New Facilities Utilization

TABLE II. Summary of Proposed Thermal-Hydraulics Test Program

Identification No. and Problem	Test Objective	Test Description	Major Measured Variables	Information Gained	Facilities	Schedule	
						Start	Finish
A-1 Fuel clad temperature limit	Prevent fuel clad failure by assurance of clad temperature less than 1100 °F in normal and distorted geometries.	Sodium tests with electrically-heated bundles of 7-pins initially followed by tests of 19 or 37 pins prototypic of FFTF fuel assemblies.	Pressure Power Flowrate Coolant Inlet Temperature Coolant Outlet Temperature Pin Clad Temperature	Location and magnitude of local clad hot-spot temperatures.	PNL Heat Transfer Loop-1 (Na, 20 gal/min., 1500 °F 200 kW electrical heat).	Jan 1968 (test section)	Sep 1968 Jan 1970
		Fabricated with allowance for varying pin displacements and number of blocked channels.		Effect of distorted or blocked channels.	Large Heat Transfer Loop (Na, 250 gal/min., 1500 °F, 6 MW electrical power).	Sep 1968 (test)	Jul 1969 Jul 1970 (test section) Jul 1970 Mar 1971 (test)
A-2 Local coolant voiding propagation	Prevent incidents which can lead to local coolant boiling and voiding propagation throughout subassembly.	Sodium tests with electrically-heated bundles of 7-pins initially followed by tests of 19 or 37 pins prototypic of FFTF fuel assemblies.	Pressure Power Flowrate Fuel Assembly Pressure Drop Coolant Inlet Temperature Coolant Outlet Temperature Coolant Outlet Temperature of Afflicted Channels Pin Clad Temperatures	Conditions under which local boiling and voiding will occur.	PNL Heat Transfer Loop-1	Mar 1970 (test)	Sep 1970
		Fabricated with allowance for varying pin displacements and number of blocked channels.		Consequences of local boiling and voiding.	Large Heat Transfer Loop	Jan 1970 (test section) Sep 1970 (test)	Sep 1970 Mar 1971
A-3 Loss of coolant and power excursion accidents	Prevent hydraulic instability following flow loss accidents by sufficiently steep pump head characteristic and orificing.	Sodium tests with electrically-heated bundles of 19 or 37 pins prototypic of FFTF fuel assemblies, including inlet and outlet plenums.	Power Flowrate Inlet Plenum Pressure Outlet Plenum Pressure Pressure Drop from Inlet to Outlet Plenum Coolant Inlet Temperature Pin Clad Temperatures	The hydraulic demand characteristics of FFTF fuel assemblies.	Large Heat Transfer Loop	Jan 1970 (test section) Sep 1970 (test)	Sep 1970 Jul 1971
A-4 Pressure drop and cross-channel mixing.	Establish acceptable pressure drop characteristics and assure adequate coolant mixing for minimization of local hot spots.	Water tests with prototypic, full-size models of FFTF driver and test fuel assemblies. Bundles of 19 or 37 pins fabricated to allow for varying pin diameter, pin spacing, pin displacements, and blocked channels.	Overall Pressure Drop Entrance and Exit Pressure Losses Fueled Length Pressure Drop Flowrate Coolant Temperature Local Velocity Fields	Pressure drop as a function of Reynolds number for normal and distorted fuel geometries. Flow distribution and mixing data.	PNL 189D Hydraulic Loop (Water, 500 gal/min., 200 °F, 200 psi head).	Jan 1968 (Parametric Δ P) Jul 1968 (Mixing)	Jul 1968 Dec 1969
A-5 Critical heat flux (proof tests)	Demonstrate adequate heat transfer from fuel subassemblies to heat fluxes twice those expected in the reactor.	Sodium test with electrically-heated bundles of 19 or 37 pins prototypic of FFTF fuel assemblies. Fabricated with allowance for varying pin displacements.	Pressure Power Flowrate Coolant Inlet Temperature Coolant Outlet Temperature Pin Clad Temperature	Heat fluxes up to which the critical heat flux does not occur.	Large Heat Transfer Loop	Jul 1970 (test section) Jan 1971 (test)	Jan 1970 ---

TABLE III. Summary of Proposed Fuel Subassembly Flow Test Program

Identification No. and Problem	Test Objective	Test Description	Major Measured Variables	Information Gained	Facilities	Schedule	
						Start	Finish
B-1 Cladding erosion due to flow	Determine effects of high coolant velocity on fuel clad corrosion and erosion	Long-term exposure of small bundles of pins (7) to 25 to 30 fps sodium at temperatures up to 1100 °F. Long-term exposure of fuel assembly to 25 to 30 fps sodium flow at temperatures to 1100 °F with mass mockups above and below assembly to simulate in-reactor vibration.	Mass loss and gain Change of metallurgical characteristics Flow rate System pressure Pressure drop Temperature	Demonstration of corrosion level below that incorporated into fuel cladding design (0.001 in. per year)	PNL Small components evaluation loop.	Apr 1967	Jun 1968
					ANL Core components test loop	Jan 1968	Mar 1970
B-2 Cladding corrosion due to purity and temperature rates	Identify the effects of coolant purity and temperature on corrosion and material removal rates	Long-term exposure of small bundles of pins to 30 fps velocity, to 1200 °F temperature, less than 10 ppm oxygen, and 20 ppm carbon. Long-term exposure of fuel subassembly to above condition.	Mass loss and gain Change of metallurgical characteristics Flow rate System pressure Pressure drop Temperature Oxygen and carbon content	Certification of the cladding and spacer material removal rates	PNL Small components evaluation loop	Apr 1967	Jun 1968
					ANL Core components test loop	Jan 1968	Mar 1970
B-3 Fretting corrosion	Identify the potential fretting corrosion of cladding and spacers and the interfaces between them.	Subassembly vibration measurements in water and in sodium (if measuring techniques are available). Perform accelerated sodium tests if potential wear and fretting corrosion occur.	Deformation of material indentation Surface effects Cross correlating of material loss information from velocity, purity, and temperature effects.	Certification of present design. Suggested design changes if required.	PNL 189D Hydraulic loop		Jul 1968
					PNL Small components evaluation loop	Apr 1967	Jun 1968
					ANL Core component test loop	Jan 1968	Mar 1970
					PNL heat transfer loop-1 Large Heat transfer loop PNL vibration lab	Jul 1969	Jul 1971
B-4 Vibration/wear	Define the vibration/wear characteristics of fuel subassembly and establish optimum design which minimizes vibration	Static air test to establish effect of structural design parameters on pin, subassembly and shroud, individually, for different end fixities using electro-magnetic shaker. Air temperature from 70 to 1000 °F. Response characteristics recorded on magnetic tapes. Static water shroud test with forced vibration for damping data, resonant decay; response characteristics recorded for water temperature to 200 °F. Vibration flow test in water at 80 to 125% of design average velocity for temperature 70 to 200 °F, using fuel subassembly, with prototypic duct. Vibration flow test in sodium at 80 to 125% design flow velocity, 600 to 1200 °F. Some extended term runs to establish wear characteristics.	Natural frequencies Damping ratios Spacer location Wire wrap pitch and tension Temperature Natural frequencies Damping ratios End fixities Spacer location Wire wrap, pitch and tension Temperature Average velocities Local velocities Deflections (or acceleration) at a selected point Temperatures Temperature Average flow velocity Fuel pin deflection Wear	Combination of wire wrap tension and end fixity that optimizes positive damping. Positive damping information for designing around resonant frequency conditions. Design curves showing vibratory amplitude and flow velocity for various temperature and structural designs Design curves of vibration amplitudes, flow and temperature	PNL vibration lab	Aug 1967	Dec 1967
					PNL vibration lab	In progress	Dec 1967
					PNL-EDEL-1 or 189D hydraulic loop	Dec 1967	Feb 1969
					A single duct sodium flow loop, 500 gal/min, 1200 °F. (ANL-CCTL)	Jan 1968	Mar 1970
B-5 Thermal shock	Establish the adequacy of fuel element	Subject fuel pin alternately to hot (1100 to 1200 °F) sodium and to cold (600 to 700 °F) sodium.	Pin bowing Change in pin diameter Change in pin ovality Change ductility of cladding Change in ductility of weld metal Change in cladding surface finish Wire wrap loosening or tightening Change in ductility of wire wrap	Design confirmation of adequacy of fuel pin to undergo thermal shock	Inert gas filled tube connected to hot sodium and cold sodium supply tanks	Not established	

TABLE IV. Summary of Proposed Hydraulic and Vibration Test Program

Identification No. and Problem	Test Objective	Test Description	Major Measured Variables	Information Gained	Facilities	Schedule	
						Start	Finish
C-1 Flow split between reactor ducts	Prevent maldistribution of coolant flow distribution in core.	Single component pressure drop-flow tests in water.	•Pressure differential •Flow rates of coolant.	A complete characterization of steady state core flow splits for a wide range of flow conditions and reactor physical arrangements. Verification of reactor final hydraulic design.	PNL 189D hydraulic loop (500 gal/min, 200 °F)		Jul 1968
		System flow distribution tests performed in water First tests in small sectional models followed by tests in full reactor system models.	•Individual flow duct flow rates •Pressure distribution in inlet and outlet regions •Total flow rate		Hydraulic core mockup, models of inlet and outlet plenum features (3000 gal/min, 200 °F), reactor hydraulic system (15,000 gal/min, 250 °F)	Jan 1968 Jan 1969	Dec 1968 (plenum) Jun 1969 (system) (prel. date) Dec 1969 (final date)
C-2 Bypass flows	Optimize bypass flow arrangement to assure that these flows perform all functional requirements (e.g., cool fixed main radial reflectors)	System flow distribution tests performed in water First tests in small sectional models followed by tests in full reactor system models	•Local coolant velocities •Local pressure distribution •Total flow rate	Optimization of bypass flow system arrangement. Verification of final reactor bypass design.	Hydraulic core mockup Models of inlet plenum features, flow ducts, and bypass region (3000 gal/min, 200 °F) reactor hydraulic system (15,000 gal/min, 250 °F)	Jun 1968	Mar 1969 (feature)
						Jan 1969	Dec 1969 (system)
C-3 Fluid thermal gradients in reactor outlet region	Prevent excessive thermal stress in outlet region components caused by thermal gradients in outlet coolant.	System mixing and flow distribution in full system models with water	•Coolant mixing and flow distribution in outlet region •Total flow rate	Characterization of outlet coolant temperature gradients sufficient to verify reactor final hydraulic design. Final verification	Hydraulic core mockup, reactor hydraulic system (15,000 gal/min, 250 °F) FTR non-nuclear	Jan 1969	Dec 1969
C-4 Loss of coolant accidents	Determine the consequences of loss of coolant accidents to the core and reactor.	Single fuel assembly heated tests in sodium	•Described in subassembly loss of coolant accident problem A-3.	Results of combined single fuel assembly and system-wide loss of coolant tests will provide information on the possible consequences of loss of coolant accident.	Large-Heat transfer loop (250 gal/min, 1500 °F, 6 MW electrically heated 37 pin fuel bundle)	Jan 1970	Jul 1971
		Single fuel assembly fission gas plenum rupture tests	•Residence time and distribution of voiding fission gas.		PNL 189D hydraulic loop (500 gal/min, 200 °F) hydraulic core mockup, reactor hydraulic system (15,000 gal/min, 250 °F)	Jan 1970	Jul 1970
		System loss of coolant accident simulation in full reactor system models with water	•Coolant flow rates in affected region •Pressure changes in affected region •Delay time in operation •Operational lifetimes	Verification of final emergency coolant system operation			
		Emergency coolant system performance tests in full reactor system in sodium			FTR		
C-5 Cavitation	Prevent significant levels of cavitation from occurring in the reactor or primary coolant system	Single component cavitation tests in water and sodium	•Cavitation damage (post inspection) •Flow rate •Fluid temperature and pressure	Identification of cavitation problem areas. Evaluation of design modifications designed to prevent cavitation. Verification in final reactor design of the absence of cavitation.	Single component PNL 189D hydraulic loop. Fuel subassemblies, orifices, ANL-CCTL (fuel, orifices). LMEC-ACTI (pump)	Jan 1968	Jun 1968
		System cavitation tests in a full reactor system model with water	•Visual evidence of cavitation •Flow rate •Fluid temperature and pressure		Hydraulic core mockup, reactor hydraulic (15,000 gal/min, 250 °F)	Jan 1968 Mar 1970	- 1970 - 1971
		System cavitation tests in heated sodium in a full reactor system model	•Cavitation damage (post inspection) •Flow rate •Fluid temperature and pressure			Jan 1969	Dec 1969
					FTR		

TABLE IV (contd)

Identification No. and Problem	Test Objective	Test Description	Major Measured Variables	Information Gained	Facilities	Schedule	
						Start	Finish
C-6 Gas entrainment	Prevent gas entrained in the reactor primary coolant from reaching levels which will adversely affect the operation of the test programs or endanger the safety of the reactor.	Single component gas entrainment tests in sodium and water.	<ul style="list-style-type: none"> •Amount of gas entrained •Loss of function with gas entrained •Flow rate of coolant •Fluid temperature and system pressure 	Identification of gas entrainment problem areas. Evaluation of methods to reduce undesirable levels of gas entrainment.	LMEC-SCTI (IHX) LMEC-ACTI (pump)	Mar 1970	Mar 1971
		System gas entrainment tests in a full reactor system model with water.	<ul style="list-style-type: none"> •Amount of gas entrained (visual and stoichiometric) •Flow rate of coolant 		Hydraulic (water) core mockup of reactor hydraulic system (15,000 gal/min)	Aug 1970	Dec 1970
		System gas entrainment in heated sodium in a full reactor	<ul style="list-style-type: none"> •Amount of gas entrained •Flow rate of coolant •Fluid temperature and system pressure 		FFTF		
C-7 Solids deposition	Verification of methods to prevent solids deposition in critical components of the reactor	System solids deposition tests in a full reactor system model with water	<ul style="list-style-type: none"> •Distribution of deposited solids •Flow rate of coolant 	Verification of design features aimed at preventing solids from depositing in critical areas such as the fuel assembly coolant channels. The consequences of fuel assembly coolant flow channel blockages.	Hydraulic core mockup of reactor hydraulic system (15,000 gal/min)	Jan 1971	Apr 1971
		Heated tests of coolant channel flow blockages in a single fuel assembly	<ul style="list-style-type: none"> •Described in Thermal Hydraulic Development (A-2) 		PNL-Heat Transfer Loop-1 (20 gal/min, 200 kW)	Mar 1970	Sep 1970
					Large Heat Transfer Loop (250 gal/min, 6 MW)	Sep 1970	Mar 1971
C-8 Gas or steam closed test loop rupture	Determine the risk to the reactor of a potential gas or steam test loop rupture	Rupture simulation tests in a full reactor system model with water as the test fluid	<ul style="list-style-type: none"> •Pressure wave propagation in core •Gas void residence time •Source pressure 	Safeguards information on feasibility of including a gas or steam closed test loop in the reactor	Hydraulic core mockup of reactor hydraulic system (15,000 gal/min)	Jun 1971	Dec 1971
D-1 Hydraulically induced vibrations	Prevent hydraulically induced vibration levels sufficient to damage or adversely affect the functional operation of components.	Single component vibration tests performed in sodium and water	<ul style="list-style-type: none"> •See B-4 	Identification of component and component systems where significant hydraulically induced vibration may occur. Evaluation of design modifications to prevent vibration. Verification of the lack of damaging vibration modes in the final reactor design.	See B-4	-	Jul 1968
		Identification tests of hydraulically induced vibration problem areas in water reactor mockup.	<ul style="list-style-type: none"> •Location of vibration (visual) •Frequency and amplitude of vibration •Flow rate of coolant 		Hydraulic core mockup	Jan 1968	Mar 1970
		Verification tests demonstrating lack of hydraulically induced vibration problem areas in final reactor design	<ul style="list-style-type: none"> •Frequency and amplitude of vibration •Flow rate of coolant 		Reduced scale model	Jan 1969	Dec 1969
					Full scale sector	Mar 1970	Jun 1970
					FTR	Jul 1973	--

TABLE V. Summary of Proposed Mechanics Test Program

Identification No. and Problem	Test Objective	Test Description	Major Measured Variables	Information Gained	Facilities	Schedule	
						Start	Finish
E-1 Lower tube sheet assembly	Confirm the analytical model developed for the inlet tube sheet assembly	Ring moments to simulate tube bowing loads and uniform lateral loads to simulate hydraulic pressure will be applied to prototypical replicas	Deflection, isoclinics and isochromatic stress patterns will be measured on photoelastic models. Hydraulic pressure and simulated thermal moments are the load variable	Out-of-plane deflections and in-plane stresses. Establishes allowable thermal moments. Optimized design	Photoelastic laboratory or hydraulic core mockup	Jan 1969	Jun 1969
E-2 Reactor cover	Confirm the analytical model developed for the reactor cover assembly	Hydraulic pressures to simulate sodium spray, and sodium slug loads plus point loads to simulate fuel weight will be applied	Deflections, isoclinics, and isochromatics simulated hydraulic pressure and point loads will be measured.	Establishes allowable deflections and stress levels. Optimize design details such as edge fixity, cover thickness, etc.	Photoelastic laboratory or hydraulic core mockup	Jan 1969	Jun 1969
E-3 Flow ducts	Determine the stresses in geometric transitions or near the perforations in flow ducts and closed loop tubes	Pressure and/or axially load photoelastic models of various duct or tube sections	Isoclinics and isochromatics stress patterns	Optimum design at transition regions and perforations.	Photoelastic laboratory	Jan 1968	Jun 1968
	Determine the creep rates in flow ducts under combined loads.	Hydraulic pressure and simulated thermal bowing loads	Deflections as a function of time. Pressure and applied moment.	Establishes allowable creep rate in flow ducts. Verifies the analytical models.	314 Bldg. Burst Test Facility	Jul 1968	Dec 1968
	Determine duct behavior to prototypic operating conditions of temperature and coolant	Vibration, integral structure, mechanical fit, and sodium flow testing	Deflection, bowing, amplitude frequency, galling and fretting damage, flow, temperature, pressure	Proof testing of driver duct	See D-1, E-4, F-2 sodium flow loop, mockup of process channel, 450 gal/min 1200 °F	Jun 1969	Jun 1970
E-4 Integral core structure	Confirm the analytical model of the integral core structure	Apply temperature gradients and hydraulic pressure to replicas of the tube sheet-ducts-cover-vessel assembly	Temperature, hydraulic pressure, deflections and strains	Optimizes location and rigidity at core supports. Provides means to analyze operational stability from temperature transients	Small dry scale model or hydraulic core mockup facility	Jan 1969	Jun 1969
	Determine the linearity of the core structure movements with uniform heating	Apply uniform temperature to an exact, full scale replica of core structures	Temperature and deflections	Establishes how mating parts interact on heating; confirms basis for linear analysis	Quarter section, full scale, hydraulic core mockup	Sep 1969	Mar 1970
E-5 Core blast analysis	Develop a semiempirical model for propellant explosions in composite wall vessels.	Detonate propellants in sodium filled vessels. Vary charge, wall thickness, gas gap, diameter, 850 °F	Strains in outer and inner vessel; pressure distributions on walls and ends	Defines loadings on the vessel walls	Hazards test cell	Jan 1969	Jun 1969
	To measure the sodium spray and sodium slug loadings	Detonate TNT in sodium filled, model pressure vessels, with and w/o cover. Vary constraint of lower vessel; 850 °F and 200 °F	Strains in vessel walls; pressure distributions on walls and cover; time history for spray and slug	Defines pressure loadings on cover	Hazards test cell	Jul 1968	Dec 1968
E-6 Vessel	Establish the stress concentrations in the vessel nozzles and support reactions	Applied simulated pressure, thermal or weight loads to photoelastic specimens	Load, isochromatics and isoclinic stress patterns	Optimizes design at nozzles and support reactions	Photoelastic laboratory or hydraulic core mockup	Jan 1969	Jun 1969
E-7 Driver fuel top face hardware	Develop reliable top face hardware and sealing methods for driver fuel ducts	Proof-of-principle testing of candidate seals followed by statistical and reliability design verification	Coolant pressure, temperature, vibration frequency, dimensional fit, distortion, fretting and galling	Reliable top hardware seal design	Static test stand, gas and sodium to 1200 °F, water to 200 °F	Jun 1968	Jun 1970

TABLE VI. Summary of Proposed Mechanical Fit Test Program

Identification No. and Problem	Test Objectives	Test Description	Major Variables Measured	Information Gained	Facility	Schedule	
						Start	Finish
F-1	Reactor cover component arrangement	Evaluate arrangement for assembly fit, interferences, special tools required, clearance for fuel handling, and assembly sequence	Assemble a representative full size sector of the vessel cover and top-of-reactor components, pipelines, connectors, etc.	<ul style="list-style-type: none"> •Assembled configuration of the reactor top face •Dimensions, interferences 	Preferred form of reactor component assembly at top face.	•Full size, reactor cover mockup	Sep 1968 (preliminary) Jun 1970 (confirm)
F-2	Configuration of reactor internal components	Evaluate arrangement for assembly, fit, alignment, interferences, duct removal and insertion, and assembly sequence	Assemble a representative full size sector of the duct array; insert and remove shroud under simulated and prototype fueling conditions	<ul style="list-style-type: none"> •Assembled configuration of the partial reactor duct •Mechanical interference location and extent. •Withdrawal and insertion forces 	Confirmation of core assembly design	<ul style="list-style-type: none"> •Seven duct dry mockup Full size, partial sector mockup of the cover, supply plenum tube sheet, and ducts •Duct environment •Seven duct sodium environment, full length duct 	Sep 1968 Mar 1969 Jul 1969 June 1970
F-3	Self-welding of components exposed to liquid sodium.	Determine adequacy of component design to resist self-welding	Immerse contacting components in liquid sodium, hold in static contact, and check forces required for separation. Immerse full length duct in prototype sodium environment simulate heat-up and cool-down cycle	<ul style="list-style-type: none"> •Surface hardness •Surface finish •Sodium temperature •Contact length (time) •Contact force •Sodium purity 	Assurance of no self-welding characteristics	<ul style="list-style-type: none"> •Component size, heated sodium pot connected to an existing loop and fitted with loading and force measuring mechanisms •Seven duct sodium environment, full length duct 	Sep 1968 Mar 1969 Jul 1969 Jun 1970
F-4	Galling of components exposed to liquid	Confirm adequacy of component design to resist galling	Operate assemblies while immersed in liquid sodium, measure forces required, and inspect contacting surfaces	<ul style="list-style-type: none"> •Surface hardness •Surface finish •Sodium temperature •Contact force 	Assurance of no galling characteristics	<ul style="list-style-type: none"> •Same heated sodium facility as used for self-welding tests. •Seven duct sodium environmental mockup 	Sep 1968 Mar 1969 Jul 1969 Jun 1970
F-5	Duct flutter during insertion	Confirm adequacy of design to prevent flutter or develop ways to minimize the disturbance	Lower ducts into a stream of water in a simulated reactor channel	<ul style="list-style-type: none"> •Visual observation •Flow rate •Vibration 	Assurance of satisfactory fuel duct insertion	•Full size mockup of a reactor channel with a simulated vessel cover guide and tubesheet penetration. A circulating pump and piping system are required to produce flows representative of shut-down flow rates.	Apr 1969 Jul 1969

TABLE VII. Summary of Proposed Control Mechanisms Test Program

Identification No. and Problem	Test Objective	Test Description	Major Measured Variables	Information Gained	Facilities	Schedule	
						Start	Finish
G-1 Reactor control drive	Provide reliable drive mechanism	Component test in air	<ul style="list-style-type: none"> •Output force •Drive rate under load •Position readout, reproducibility •Input power 	<ul style="list-style-type: none"> •Overall size •Input power •Output power 	Vertical test stand, simulated rod weight in Bldg. 314	Apr 1968	Dec 1968
G-2 Scram (latch) control device	Minimize scram mechanical time delay	Electromagnet test in air	<ul style="list-style-type: none"> •Release time •Holding force •Input power 	<ul style="list-style-type: none"> •Release time •Holding force •Input power 	Vertical test stand, simulated rod weight in Bldg. 314	Apr 1968	Dec 1968
G-3 Mechanism sodium contamination	Seal control mechanism from reactor environment	Heating bellows test in sodium vapor above 1200 °F	<ul style="list-style-type: none"> •Temperature profile •Cycle rate •Number of cycles 	<ul style="list-style-type: none"> •Type and size of seal 	Vertical test stand, 1200 °F sodium pot, simulated temperature gradient. LMEC CRTT	Aug 1968	Jun 1969
G-4 Accelerated safety scram	Provide an initial acceleration greater than gravity for the safety control element	Materials investigation on spring materials at 1200 °F in sodium.	<ul style="list-style-type: none"> •Percent relaxation versus exposure to temperature •Intergranular corrosion 	<ul style="list-style-type: none"> •Spring material suited for service 	Stagnant sodium pot with provision to apply static load up to 1000 lb	Apr 1968	Jun 1969
G-5 Hydraulically induced vibration	Flow induced vibration in poison sections	Flow tests in 250 °F water; flow rate to 100 gal/min	<ul style="list-style-type: none"> •Flow rate (velocity) •Water temperature •Water pressure (static) •Water pressure loss across control element •Vibration frequency •Vibration amplitude 	<ul style="list-style-type: none"> •Element vibrational instability •Element pressure loss versus flow 	Water flow test EDEL-1 loop 314 Bldg.	June 1968	Jun 1969
G-6 Decelerator	Minimize stress levels introduced in the control element and rod guide tube during scram cycle	Deceleration component test in water at 250 °F to 50 gal/min Simulated rod weight	<ul style="list-style-type: none"> •Rod velocity •Deceleration time •Water flow •Water temperature •Dash pot water temperature 	<ul style="list-style-type: none"> •Acceptable design configuration •Deceleration rate 	PNL EDEL-1 loop 314 Bldg.	Apr 1967	Mar 1968
G-7 Prototype test	Vendor acceptance test Verify design	Water test vendor checkout Life test	<ul style="list-style-type: none"> •Input power •Drive rate •Scram time •Output force 	<ul style="list-style-type: none"> •Preliminary acceptance •Reliability •Power requirements •Performance 	Vendor water test stand. Rod test stand. Sodium flow to 50 gal/min. Sodium temperature to 1200 °F. Inclined angle to 8°. LMEC CRTT	Jan 1970	Feb 1970
						Mar 1970	Sep 1970
G-8 Production test	Vendor acceptance	Water test, vendor checkout, preparational checkout	<ul style="list-style-type: none"> •Input power •Drive rate •Scram time •Output force 	<ul style="list-style-type: none"> •Preliminary acceptance •Final acceptance 	Vendor water test stand Sodium mockup facility	Jun 1971 Sep 1971	Sep 1971 Dec 1971
G-9 Mechanical configuration and environment test	Verify compatibility with top cover hardware	Mechanical fit maintenance, handling, procedures	<ul style="list-style-type: none"> •Elapsed time •Physical interference 	<ul style="list-style-type: none"> •Operability 	Seven duct facility	Sep 1970	Mar 1971
G-10 Preoperational	Performance verification	Installation	<ul style="list-style-type: none"> •Mechanical fit 	<ul style="list-style-type: none"> •Installation complete 	Pressure vessel FTR	Jan 1972	Dec 1972

TABLE VIII. Summary of Proposed Closed Loop Development Test Program

Identification No. and Problem	Test Objective	Test Description	Major Measured Variables	Information Gained	Facility	Schedule	
						Start	Finish
H-1 Closed loop top face hardware	Develop reliable top face hardware and seals for closed loops which involves selection of sealing methods which will prevent liquid sodium leakage and development of hardware that will last at least 5 years	<ul style="list-style-type: none"> •Proof-of-principle tests on various sealing methods and clamping devices •Design of prototype hardware employing proven seals, and statistical reliability testing of seals in same •Design verification testing of final hardware design 	<ul style="list-style-type: none"> •In first two levels of testing-- <ol style="list-style-type: none"> 1) coolant temperature and pressures 2) frequency and amplitude of vibration 3) external and thermal structural forces, static and dynamic 4) effects of galling and fretting •In third level same as above plus <ol style="list-style-type: none"> 1) flow rates of coolant 2) dimensional fit and distortion 3) compatibility with fuel handling machine 	<ul style="list-style-type: none"> •Reliability of seals established •Information to form basis for final design •Verification of final design 	Static (up to 250 psig) test stand, water to 200 °F, helium to 1350 °F, sodium to 1350 °F, thermal transient capability.	Jun 1968	Jun 1970
H-2 Closed loop in-reactor hardware	Develop reliable in-reactor hardware which will last 12 months. This involves preliminary design followed by prototype testing.	Full scale and proto-typical in terms of flows, and geometries in H ₂ O, followed by liquid sodium	<ul style="list-style-type: none"> •Coolant flow rates, temperatures, and pressures •Frequency and amplitudes of vibration •External and thermal structural forces, static and dynamic •Effects of fretting and galling 	Graphs and other data showing vibration amplitudes as a function of flows at various temperatures, would be provided by the preliminary tests. Reliability assurance on the in-reactor hardware will be provided by the endurance tests	Hydraulic (water) loop, 100 to 700 gal/min, 200 °F Sodium loop 100 to 700 gal/min, 1350 °F, w/closed loop in-reactor hardware and external and thermal structure forces	Jun 1969 Jun 1970	Jun 1970 Jan 1971
H-3 Closed loop interconnecting piping and disconnects	Develop reliable interconnecting piping and disconnecting components which will last at least 5 years. This involves: <ul style="list-style-type: none"> •Leak tight joints to contain liquid sodium •Methods of removal without loss of coolant •Containment of radioactivity in case of failure •Detection of leaks in case of failure •Recovery from sodium freeze-up 	Feasibility tests on one or more full scale prototype piping and disconnect arrangements to determine feasible methods of removal, sealing, and leak detection	<ul style="list-style-type: none"> •Coolant temperature and pressure •Vibration frequency and amplitude •External and thermal structural forces, static and dynamic •Galling and fretting •Leak rates •Thermal heating requirements, i.e., current voltage versus temperature 	Experimental data for design and verification of final design	Static (to 250 psi) test stands, water to 200 °F, sodium to 1350 °F, helium to 1350 °F Sodium loop, 100 to 700 gal/min, 1350 °F with interconnecting piping and external and thermal structure forces. Pressure 150 to 250 psi	Jun 1969 Jun 1970	Jan 1970 Jan 1971
H-4 Closed loop components external to reactor vessel	Develop reliable shutoff and check valves for component external to reactor vessel which involves selection of seals and valves that will withstand the liquid sodium environment for at least 5 years	<ul style="list-style-type: none"> •Proof-of-principle tests on various sealing methods and valve arrangement •Statistical reliability testing of sealing method in prototype valve •Design verification testing of final valve designs 	<ul style="list-style-type: none"> •In first two levels of testing-- <ol style="list-style-type: none"> 1) coolant temperature and pressure 2) effects of galling, fretting 3) leakage rates 4) vibration •Same as above plus <ol style="list-style-type: none"> 1) flow rate 2) pressure drop 	<ul style="list-style-type: none"> •Reliability of seals established •Information to form basis for final design •Verification of final design 	Hydraulic (water) loop, 100 to 700 gal/min, 200 °F pressure to 150-250 psi Sodium loop 100 to 700 gal/min, 1350 °F, pressure 150 to 250 psi	Jun 1969 Jun 1970	Jan 1970 Jan 1971
H-5 Prototype closed loop	Demonstrate compatibility and operability of individual closed loop components when integrated into a complete closed loop	Complete full scale system operation in terms of sodium capability and tie-in with interface components	<ul style="list-style-type: none"> •Flows, temperatures, vibration, combined distortion interferences and fitup with fuel handler and associated components •Routine, maintenance, and emergency operations 	Demonstration that all loop components plus the loop as a system, perform to requirements under ex-reactor but otherwise proto-typical conditions.	Prototype closed loop system with flowing sodium, 100 to 700 gal/min, 1350 °F, pressure, 150 to 250 psi. Auxiliary sodium pumping system simulating flows external to the in-reactor hardware	Jan 1971	Jan 1972

TABLE IX. Summary of Proposed Instrument Development Test Program

Identification No. and Problem	Test Objective	Test Description	Major Measured Variables	Information Gained	Facilities	Schedule	
						Start	Finish
I-1	In-reactor coolant temperature sensors	Provide specifications for chromel-alumel thermocouples capable of operating in FFTF environment.	Iterative testing for thermal shock and cycling behavior from 400 to 1400 °F, nondestructive examinations, calibration checks, irradiation in EBR-II.	•Speed of response •Insulation resistance •Calibration as function of neutron exposure •Catastrophic failure rate	Specification and technical backup for FFTF in-reactor thermocouple design.	•Thermal cycle and shock test stand •EBR-II •Hot cells for examination	Feb 1968 Oct 1969 In progress - 1971 Mar 1968 - 1971
I-2	Pressure transducers	Produce transducer capable of measuring coolant pressure above and below core, in sodium at FFTF flux and temperature.	Screen commercial transducers in sodium pot tests. Modify with vendor most promising. Flow test at 1400 °F and 250 psi. Irradiate components in EBR-II.	•Accuracy •Sensitivity •Zero drift •Corrosion effects	Specification for procurement of suitable pressure transducer.	•Sodium pot with level control temperatures to 1400 °F, pressure to 250 psi •Sodium flow loop to 20 ft/sec, 1400 °F •EBR-II for component exposure •Sodium pot 30 ft high by 12 in. diam, low flow, level sensor installation	Oct 1967 - 1970 Mar 1968 - 1970 Dec 1968 - 1971
I-3	In-core sodium flowmeter	Develop and test magnetic and eddy current type flowmeters for use in emergency, surveillance and power data	Magnet testing at temperature, thermal and mechanical shock, and radiation. Evaluate in magnetic hot cell mockup. Coil testing by temperature cycle in furnace and flowing sodium. Prototype magnetic flowmeter test in sodium to 1200 °F and 450 gal/min. Backup eddy current flowmeter in 1200 °F sodium	•Magnetic flux •Coil characteristics •Reliability •Eddy current flowmeter sensitivity, impedance, noise temperature compensation, signal stability, accuracy, calibration.	Specifications for acceptable design of in-channel flowmeters	•Sodium flow loop for low flow to 1200 °F (ANL). •Mockup simulating hot cell magnetic properties for preirradiation measurements with furnace to 1200 °F (ANL). •Hot cell for postirradiation examination •Furnace for 1200 °F coil test •Large sodium flow loop, 450 gal/min, 1200 °F. (LMEC LCTL) •Closed loop prototype, 700 gal/min, 1400 °F	In progress - 1970 In progress - 1971 Mar 1968 - 1970 Dec 1968 - 1970 Jan 1971 Jan 1972
I-4	Gas disengagement for failed fuel monitor	Develop a technique of efficiently disengaging fission gas from sodium. Accumulation and transporting the gas out of the reactor vessel.	Determine disengagement efficiency by injecting gas into sodium using several gas disengagement units. Evaluate means for getting fission gas sample out of reactor without plugging lines with sodium.	•Efficiency of disengagement devices •Evaluation of methods, sizes and locations of sample lines •Threshold of system sensitivity	Recommendation of technique for gas disengagement. Location and size of sample tubes.	•Small sodium loop with sodium at temperatures, pressure, and velocity of FFTF subassembly channel (ANL). •Sodium loop, mockup of process channel, 450 gal/min, 1200 °F, gas injection unit	In progress Jun 1970 Jun 1970 Jun 1971
I-5	Delayed neutron fuel failure monitor	Develop a reliable operating system to detect and locate failed fuel for operation in FFTF sodium, thermal, and nuclear environment.	Determine response of detectors and pre-amplifiers at high temperature and high gamma flux. Experimentally evaluate piping and valving arrangements for reliability and freedom from plugging	•Detector gamma tolerance and neutron sensitivity •Pulse height stability versus temperature •Noise pickup •Lifetime of electronic components	Design data on operable system	•Gamma facility for neutron detector evaluation •Sodium loop for sampling valve evaluation, 1000 °F •Sodium loop, mockup of process channel, 450 gal/min, 1200 °F for system test.	In progress Jun 1969 Jun 1968 - 1970 Jul 1969 - 1971

TABLE IX (contd)

Identification No. and Problem	Test Objective	Test Description	Major Measured Variables	Information Gained	Facilities	Schedule	
						Start	Finish
I-6 Instrumentation subassembly	Assure compatible instrument subassembly operation of individually developed and tested instruments.	Proof test the instrument subassembly package by simulating various reactor conditions for which the package has been assembled. Reactor conditions include temperature abnormalities, gas disengagement for failed fuel, pressure and flowmeter response to loss of coolant, flux monitor probe operation, materials evaluation.	<ul style="list-style-type: none"> •Temperature variations due to channeling •Performance of gas disengagement system •Pressure drop across subassembly •Vibration tests and adequacy of mechanical mountings 	Proof test of instrument package and design data for optimizing location performance characteristics.	•Sodium flow loop, mockup of process channel, 450 gal/min, 1200 °F 12 in. diam test section 35 feet long, 304 stainless steel construction.	Dec 1968	- 1971
I-7 Vibration and stress measurements	Improve sensors for high temperature tolerance, radiation tolerance, and tolerance to immersion in sodium.	Develop encapsulation method for 1200 °F service and radiation environment service. Evaluate sensors for vessel surveillance.	<ul style="list-style-type: none"> •Temperature, radiation and sodium environment tolerance. •Operational data •Nondestructive testing data 	Proof testing of vibration and stress measuring instruments and techniques for core component and vessel surveillance.	<ul style="list-style-type: none"> •EBR-II for radiation effects (strain gauges) •Sodium pot for 1200 °F operation •Sodium vessel for 1200 °F application, size of vessel as required to simulate reactor. 	Jul 1968	Jul 1969
I-8 Signal connectors	Develop signal connectors for nozzle and in-sodium service at 1200 °F	Prove remote disconnect method, mechanical adequacy through cycling.	<ul style="list-style-type: none"> •Remote disconnect performance •Gas and sodium leak tightness •Flux monitor mechanical performance 	Design information and specifications for connectors demonstrated suitable for FFTF use	<ul style="list-style-type: none"> •Top nozzle configuration with simulated fuel handling machine •Sodium pot for 1200 °F service, mechanical loading apparatus for tension compression and bending tests (ANL). 	Mar 1968	- 1970
I-9 Signal cables	Provide technical basis for signal cable specifications.	Subject several types of signal cables to radiation and thermal environment. Determine effect on insulation resistance and amounts of induced signals as function of cable geometry and materials.	<ul style="list-style-type: none"> •Insulation resistance of several mineral oxide cables as function of temperature and radiation. •Inducted signal generation as function of all cable parameters. 	Design data on acceptable cable and material and types of circuitry required.	<ul style="list-style-type: none"> •Gamma source up to 10⁹ R/hr. •Neutron flux up to 10¹⁶ nv. 	In progress	Dec 1970
I-10 Sodium flowmeter in large pipes	Assure 16 to 18 inch sodium flowmeter for 1000 °F operation, in flow range of 800 to 16,000 gal/min	Perform proof tests of standard design flowmeters, in piping assembly simulating FFTF. Compare against flow tube differential head flowmeter. Thermal cycles.	<ul style="list-style-type: none"> •Accuracy, repeatability, rangeability over range of 20 to 120% of full scale flow •Sodium temperature •Magnetic flux as function of temperature. 	Proof tested flowmeter for FFTF service.	•Sodium loop, 800 to 16,000 gal/min flow, 1000 °F temperature, thermal cycling. (LMC-ACTI).	*Mar 1970	Dec 1971
I-II In-core neutron flux monitoring	Provide method to reliably transport and guide sensors in and out of reactor during operation	Mockup arrangement of guide tubes, connectors, thimbles and related essential parts. Heat in-vessel thimble to determine life, test solution to correct unexpected failures.	<ul style="list-style-type: none"> •Life of components •Handling features 	Design adequacy, specification of operational system.	•Mockup of subassembly including one channel of flux mapping system with subassembly at 1400 °F	Jun 1968	Dec 1971

* Assumes availability of ACTI by March 1970.

DEVELOPMENT AND TESTING REQUIREMENTS

Evaluations were made to define the overall reactor system development and testing requirements. For this purpose, the reactor system is defined as the reactor vessel, cover, and all components and internal structure within the reactor vessel. The evaluation was carried out on the basis of technical disciplines where suitable, to establish a set of test requirements more easily amenable to characterizing the facilities needs and their schedules through generally common test facility features; these include thermal hydraulic, hydraulics, mechanics, hydromechanics, and mechanical fit. In some cases, however, component approach to test program evaluation was found to better present the total development needs. Such items as control mechanisms, instruments, and closed loops are in this category.

To conduct this evaluation, the basic format employed was:

- 1) Identify the design problem requiring experimentation
- 2) Describe the technical plan needed to resolve the problem
- 3) Describe the test facility needed
- 4) Determine the trade off on alternate approaches to resolving the problem or alternate action which might be taken.

The problems identified and discussed within this report are presently defined major problems of a broad scope. It is emphasized that certain subproblems associated with the details of these larger problems are inherently included. By their nature, they remain undefined at this time, but the timely and systematic resolution of these subproblems is very important to the particular development effort involved.

The problems identified and discussed in this evaluation are considered preliminary. Undoubtedly some of the specific programs will incur revisions as additional studies are made, and the concepts are firmed. The Reactor Plant Designer will

also provide contributions to the program as his efforts become integrated into the FFTF project. Reliability testing of components and systems to provide high assurance of performance and hence, high plant availability has been covered only to a small degree in this evaluation.

This FFTF reference reactor design is based on the split core concept of reactor arrangement, with a vertical core serving as a backup design. It is the intent that the overall development and testing program will be carried in such a manner that if the fall-back position were resorted to, loss in schedule for the FFTF startup would be minimized. Although there are some differences in the development requirements, the basic program needs in terms of type of information required is essentially the same. Therefore, it should be considered that the plan outlined, while basically directed to the split core, will be applicable for the most part to a vertical core concept as well. Specific development programs and facilities needed for the backup core will be formulated and compared with these discussions after PNL receives General Electric Company recommendations of development requirements for the vertical core. Detailed test plans will be formulated to minimize duplication wherever possible.

A. THERMAL-HYDRAULIC DEVELOPMENT

A plan of the thermal-hydraulics development effort needed to support the core design for the Fast Flux Test Facility was presented in Reference 1.* The plan is oriented to meet requirements dictated by a wide range of core performance criteria. The scope of this plan was limited to thermal-hydraulic problem areas arising from and directly associated with driver and test fuel in-core operation. Specifically, excluded are thermal-hydraulic problems arising in fuel handling, fuel inspection in gas or liquid, and in the heat dump systems.

In the evaluation, design data were compared with limits based on current technology. Problem areas are considered to be those cases in which:

- 1) The design data for the initial concept exceed the limits that may presently be set by current technology
- 2) The estimated design data for anticipated upgraded operation exceed the limits that may be set by current technology
- 3) No limits can be inferred due to the inadequacy of current technology.

The body of problem areas identified was subjected to a technical evaluation which compared alternate, though not necessarily independent, avenues of solution. These include:

- 1) Performing development work to determine more accurate limits for that phase of the FFTF operation
- 2) Changing the reactor design to conform to the known limit
- 3) Changing the basic criteria.

The resultant thermal-hydraulics development program is oriented towards resolving design requirements, identifying

* See end of section for reference details.

design uncertainties, and specifying specific development efforts necessary to reduce these uncertainties. The experiments proposed for the program are summarized from Reference 1 in Table II.

The program was evaluated by several National Laboratories and industry personnel* as to the needs and priorities of the several experiments to solve the problems. Comments and suggestions derived from evaluation have been reviewed by PNL and incorporated into the PNL program. The resulting program established an order of priority for experimental solution to the problems as follows:

- 1) Pressure Drop and Cross-channel Mixing
- 2) Fuel Clad Temperatures
- 3) Local Coolant Voiding and Propagation
- 4) Loss of Coolant
- 5) Proof tests of Critical Heat Flux.

The evaluation also resulted in slight redirection of the experimental program. The primary redirection is the inclusion of tests with smaller scale test sections (7 pin rather than 19 or 37 pin electrically heated test sections) to do preliminary fuel clad temperature and local voiding and propagation tests. The purpose of the smaller scale test sections is to provide information for the FFTF earlier than could be obtained with the larger scale tests. The critical heat flux tests are verification of the absence of the critical heat flux for possible overpower conditions.

The redirection of the program requires a liquid metal heat transfer facility to accommodate a 7 pin electrically heat bundle. The facility requirements are sodium pumping capacity up to 20 gal/min with a head of 60 psi, and coolant

* Argonne National Laboratory; Atomics International; Atomic Power Development Associates; General Electric Company; Oak Ridge National Laboratory.

temperature of 1500 °F. The facility will require a direct current power source of 200 kW to voltages of 450 Vdc to provide current flow through simulated fuel pins. This facility can be modified from the PNL single pin heat transfer loop by the replacement of the pump.

The 37 pin test program requires flow of 250 gal/min, pump head of 200 psi minimum at temperature to 1500 °F. This facility will require an electric power supply of about 6 MW, with approximately 4.5 MW at 450 Vdc to simulate fuel pin heat generation rates up to 2×10^6 Btu/hr/ft². This power requirement was determined by the requirements of a test section of 37 pins, each 3 ft long and 0.25 in. OD and operated at the maximum heat flux, giving 4.25 MW. The loop will have other power requirements—principally pump power and preheat power. The 4.25 MW was increased to 6 MW to provide additional power for the loop and to provide a modest excess for testing.

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1. G. M. Hesson, A. Padilla, and J. M. Yatabe. Thermal-Hydraulic Development, BNWL-CC-1149, Pacific Northwest Laboratory, Richland, Washington. April 1967.

B. FUEL SUBASSEMBLY WATER, SODIUM, AND MECHANICAL TESTING

The integrated project plan for water and sodium flow testing and mechanical testing for the FTR fuel subassembly has been presented in a separate document.⁽¹⁾ The testing program is directed toward resolving design uncertainties empirically. Where analytical techniques are not sufficiently sophisticated to produce guaranteed reliability, observation through testing under prototype or near prototype conditions, is the only available method. The report presents a comprehensive test plan which establishes the performance limits imposed on the FFTF driver fuel assembly. It examines the operating problems and proof testing methods in which partial and full fuel subassemblies are subjected to water hydraulic, sodium dynamic, and mechanical testing. The total fuel operating experience cannot be simulated in any one of these tests since a number of the more important operating conditions are absent, primarily core ΔT and radiation exposure. The program simulates all operating experiences which are possible through testing and couples the results with the effects of irradiation as generated in the FTR fuel and cladding irradiation programs. The problems of erosion and corrosion are examined in light of other reactor operating experiences where a temperature gradient is imposed on the fuel assembly. Analysis will be performed to determine the mechanical effects of vibration, fretting corrosion, fretting fatigue, etc., coupled with the vibration and material removal measurements noted in the hydraulic and the sodium dynamic tests. By using this synthesis testing method, the total experience for the fuel performance will be projected which will lead to high reliability and assure performance of both first and subsequent FFTF core loadings. Additionally, the data generated within these testing programs will be coordinated with the thermal-hydraulics development program and the core hydraulic mockup program to insure that the projected

operating characteristics and design modifications resulting from the testing are reflected in these programs. The water hydraulic, sodium dynamic, and mechanical tests proposed for the program are summarized below:

<u>Test Designation</u>	<u>Test Parameter and Objectives</u>
Water-hydraulics tests	Flow rates to 125% of normal, using full scale subassemblies. Objectives to establish acceptable pressure drop characteristics and assure adequate coolant mixing for minimization of local hot spots.
Small bundle sodium flow tests	Velocities up to 35 ft/sec in shortened 7 and 19 rod clusters. Objectives to examine specific localized problems in the fuel bundle performance. Outlet temperature up to maximum sodium clad interface temperature ~1100 °F.
Full scale CCTL (Core Component Test Loop) testing	Full subassembly flow at maximum hot channel outlet temperature. Examine full subassembly performance, and due to flow induced vibration, corrosion, erosion, etc.

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1. P. D. Cohn. FTR Fuel Design and Analysis -- Hydraulic, Sodium and Mechanical Testing, BNWL-599, Pacific Northwest Laboratory, Richland, Washington. October 1967.

C. CORE AND VESSEL HYDRAULICS

The core and vessel hydraulics development program is oriented toward resolving design and operating requirements, identifying design uncertainties, and specifying development efforts necessary to reduce these uncertainties. The program has been divided into specific types of technical problem areas. These areas are:

- 1) Core flow distribution
- 2) Bypass flows
- 3) Fluid thermal gradients in the reactor outlet region
- 4) Loss of coolant accidents
- 5) Gas entrainment
- 6) Cavitation
- 7) Solids deposition
- 8) Gas or steam closed test loop rupture.

These problem areas will require testing of components (e.g., fuel subassembly), partial assemblies (e.g., inlet plenum), and composite assemblies (e.g., full core mockup). Examination is undertaken by problem areas both to present all the hydraulic problems of concern and to account for interactions between components or component systems.

Subassembly component testing is a vital part of these overall core and vessel hydraulic studies; subassembly hydraulics additionally have major impact on fuel thermal-hydraulics problems and subassembly design. Therefore, the fuel subassembly pressure drop and cross channel mixing problem is covered in detail in both the thermal hydraulic development and driver fuel subassembly water and sodium testing programs. Discussions of subassembly hydraulics as part of other subprograms are not intended to imply separate entity. In fact, information obtained by other subprograms provides an integral input to the total hydraulic evaluations.

CORE FLOW DISTRIBUTION (C-1)

Design Problem and Requirements

Poor coolant distribution will cause certain channels to operate at higher than normal temperature levels. For the ducts containing driver fuel subassemblies, the thermal limits set on fuel operation will cause the hot channels to limit directly reactor power and/or bulk coolant outlet temperature. Thermal limits are:

- 1) Fuel centerline temperature limit: It is currently proposed that the fuel centerline temperature will not reach the melting point of the mixed oxide fuel at reactor powers below steady-state power level of 1.25 times the normal operating power.
- 2) Fuel clad temperature limit: It is currently proposed that the reference clad material TP 304 SS not exceed 1200 °F, based on a maximum strain rate of 0.2% after 45,000 (avg) or 75,000 (max) MWD/Tonne of fuel burnup.

At the present time, a conservative estimate of possible maldistribution of coolant between reactor flow ducts must be made and factored into an engineering hot channel or hot spot analysis and will directly increase calculated maximum fuel clad and fuel centerline temperatures. If these temperatures then exceed the aforementioned limits, the reactor power or coolant temperature must be lowered or the cause of the flow maldistribution corrected in the design.

Coolant maldistribution will also adversely affect fuels and materials test data. For example, coolant temperatures and flow rates that diverge from those specified in open loop testing programs will degrade the value of the test results. In addition, for advanced open loop fuels tests to be performed with higher outlet temperatures, unpredicted flow starvation from various causes may endanger the test fuel assemblies by causing overheating with subsequent material failure.

Coolant starvation in control rod or reflector module assemblies may cause excessive temperature levels which will possibly degrade component performance or in extreme cases cause material failure. On the other hand, excessive coolant flows in the control rod assemblies may produce high hydraulic forces which could adversely affect component operation.

Three factors determine the coolant flow distribution among the reactor ducts:

- 1) Pressure distribution in the inlet plenum
- 2) Flow resistance of the reactor duct and internal components, including orifices
- 3) Pressure distribution in the coolant outlet region.

The effects are coupled, and analytical or experimental models must not ignore any of them in the determination of flow split between the reactor tubes.

Inlet Region

Coolant enters each reactor duct from the high pressure inlet plenum. Duct inlet sections are approximately 2.9 in. outer diameter (2.7 in. inner diameter) arranged in an equilateral triangular pattern with a pitch of 3.1 in. Flow enters through the bottom of each duct. Inlet sections for ducts containing driver fuel assemblies, open loop tests or reflector modules will be identical with the possible exception of inlet orifices.

In a presently proposed single tube sheet inlet plenum design, no local pressure aberrations are anticipated. The flow rates in the plenum proper will be very low (on the order of 1.5 ft/sec) and the region surrounding the reactor tube inlet should be one of virtually constant pressure. However, perturbations in pressure may arise due to entrance effects at the subassembly duct inlet region. For example, in the closely spaced inlet region, a high flow tube adjacent to a

low flow tube (e.g., reflector module) may cause inlet pressure field disturbances sufficient to affect the flow split, or the inlet pipes and baffle arrangement may cause a vortex or vortices with uneven pressure distribution. With the use of hydrodynamic or potential flow techniques, analyses of these extreme effects will indicate whether or not a problem area exists. However, direct experimental data will be required to provide verified flow distribution results useful to the design and operation of the reactor.

There is an alternate inlet plenum design concept which employs two compartments: a high pressure section through which primary coolant flows into the core and a low pressure section below it to provide more balanced hydraulic forces on the reactor flow ducts and fuel assemblies. The major effect of this plenum design on core flow split is that the inlet region geometry will be more restricted. Flow into the ducts will be through a series of holes or slots through the side walls rather than through a large hole in the bottom. Experimental verification will be required to assure that the advantages of balanced forces in this concept outweigh the disadvantages of a more restricted inlet. Frictional forces arising from flow between and through the closely packed tubes may be significant, and the total radial pressure gradient across the inlet region may vary locally and be larger than in the first design. The potential for flow maldistribution will be increased due to the increased gradient. Potential flow analysis will be inadequate, analysis based on boundary layer and separated flow in this region of complex geometry can only be performed on a very approximate basis; therefore, data sufficiently accurate to support reactor plenum design must be obtained experimentally.

In-Duct Flow Resistance

The total flow resistance in each reactor duct will be the major factor in determining the coolant flow split. The majority of the pressure drop will occur in the small equivalent diameter region of the fuel assembly and fuel gas plenum region for driver and open test loop flow ducts. In addition, smaller pressure losses will occur due to the various entrance regions, exit regions and long pipe runs. In reflector modules and control rod positions, the major pressure losses will probably occur at inlet orifices to produce the significantly lower flow rates required to cool these components.

The inlet resistance of the driver and open test loop ducts may or may not be varied by means of orificing. In the driver ducts, this may be done for two possible reasons:

- 1) To match individual channel coolant flow to tube power: All cores, zoned or not, proposed for the FTR will have a nonuniform radial power distribution. If the radial core power gradient is large, duct coolant flows may be adjusted to minimize the temperature gradient across the core and to permit maximum coolant temperatures to be maintained within designated fuel and reactor component thermal limits.
- 2) To provide high inlet pressure drops (steeper hydraulic demand curves) to improve flow stability: Loss of coolant accidents may cause flow instabilities arising from coolant boiling in the fuel assembly which in turn may lead to low flow conditions and possible gross fuel melting. Such instabilities may be prevented from occurring by increasing inlet plenum pressures and orificing reactor ducts to provide a steep hydraulic demand curve. A full technical discussion of the effects of orificing to prevent flow instability has been made in Reference 1 and will not be repeated here.

For the open test loops, orificing may be desirable to provide high temperature coolant outlet conditions for advanced fuel and materials testing programs.

The major factor determining the pressure drop-flow behavior of the driver and test reactor channel will be the fuel assembly configurations: closely packed bundles of long, slender pins with spacers. Hydraulic behavior cannot be predicted analytically with complete accuracy. Nor are there sufficient experimental data on this class of fuel assembly configuration to completely determine their pressure drop-flow behavior as a function of spacer design, pin spacing or pin diameter. The same conclusion holds true for open test loop fuel assemblies of other geometries. The most accurate method of determining pressure drop-flow behavior in FFTF fuel assemblies will be to perform laboratory hydraulic tests on full scale, exact models.

The hydraulic behavior of the inlet orifices may be more easily characterized using currently available correlations. The only complication is that fully developed entrance conditions do not exist immediately upstream of the inlet orifices due to the accelerating fluid field at the entrance to the tubes.

Outlet Region

In the currently proposed design, coolant leaving the core will pass upward through the fuel assembly gas plenum region and then through a straight pipe region containing only instrumentation (negligible flow resistance). Coolant will emerge

from the reactor duct through radial holes located at approximately 8 ft above the top of the core. From this position, flow egresses through very complex flow paths to the outlet plenum of the reactor. An additional complication is momentum exchange between the core outlet, and the cooler bypass flow throughout this region.

Outlet coolant flow streams from the inclined subassembly shrouds act as high velocity, interacting radial sources in a complex region of tightly restricted flow paths. In general, all sources are not equal nor symmetric. The bypass flow represents a small but not negligible three dimensional velocity field mixing with the core outlet fluid. All of the aforementioned effects will be combined into an approximate analytical model. Again, the best method for obtaining the design support information for this region will be to perform hydraulic experiments on a geometrically and dynamically similar model of these systems.

An important problem which will affect flow distribution and pressure drop distribution in all three of the aforementioned reactor regions will be the effect of distorted components. Where misaligned, bowed, and swollen components alter the core flow split, their presence is detrimental to the safe and efficient operation of the reactor. Initial studies of the effects of deformed components will be performed analytically. The results of these analyses will define the further experimental studies required.

The information required for the design is:

- 1) Preliminary hydraulic data as basis for alternate designs, to optimize core flow distribution
- 2) Core flow distribution for the final Fast Test Reactor design for all proposed operating conditions

- 3) Information on the effect of mechanical distortion in reactor components on the core flow split.

These studies support design development and verification of performance characteristics for the reactor.

Test Program

Test Method and Purpose

The first tests performed will determine the flow resistance of individual reactor flow ducts and internal components. These tests will evaluate the pressure drop-flow behavior of driver fuel subassemblies, reflector subassemblies, regulating, driver and safety rods and open loop test tubes as they are defined.

For these tests, water will be used as the test fluid. The use of water will not degrade the accuracy of these data and will be more timely, and economical in the performance of the core flow distribution program. The major hydraulic tests of the driver fuel assemblies are described in detail in Reference 1 and Reference 2. The hydraulic tests on the regulating, shim and safety rods are discussed under the specific subassembly.

The individual component tests will provide important hydraulic design information and will provide information required for later tests on a simulation of the reactor in-vessel systems.

The second series of tests will be performed with feature or sectional models specifically designed to investigate one specific flow region. These tests will determine the gross effect of reactor arrangement and will be semiquantitative in nature. The first test will be performed on a small-scale, transparent model of the inlet plenum and lower flow duct and bypass region which will be used to examine core flow distribution as a function of plenum and core arrangement. The

effect of chamber configuration, antivortex baffles and inlet pipe arrangement will be evaluated. Plenum flow distribution will be determined by visual observation and dye injection. In addition, individual duct flow rates will be measured. The data obtained from these tests are expected to facilitate a choice between the several candidate inlet plenum designs. A companion test will use a single full scale reactor flow duct model to verify the hydraulic balance concept and determine the best method of providing leakage flow.

The third series of tests will be performed in an accurate mockup of the reactor in-vessel systems. The mockup may be a reduced scale, geometric model. The application of models for these studies was evaluated in Reference 3. The basic purpose of these tests will be to measure flow distribution in the core flow ducts as a function of total coolant flow rates and reactor arrangement, i.e., core and plenum geometry, test loop, position. For these tests, the hydraulic behavior of each reactor flow duct must be carefully simulated over a wide range of conditions. This will be accomplished by a series of orifices in the test model. The design of these hydraulic resistances will be based on the individual component hydraulic tests that form the first series of tests in this program.

Simultaneous measurements will be made of the local fluid pressure distribution in the inlet plenum and the tube outlet region. These measurements will determine and isolate the cause of any observed flow maldistribution. Measurements will be made with traversing pressure sensors of the impact type. Velocities and direction may also be determined.

If flow maldistributions are uncovered, further tests of this type will be performed to evaluate design modifications aimed at correcting the situation. These tests will be considered as tentative until their necessity is

clearly demonstrated by the results of the previous test program. Such tests would be run on either the reactor models or a special feature model as would be most useful.

The test program to determine the effect of component and component system distortions has not been defined to date. An extensive analytical effort must be undertaken before the program to determine the effect of mechanical distortion on flow split, vibration, etc. can be scoped.

Range of Parameters

Single Component Hydraulic Tests

Flow rates	5-150% of that proposed for the specific component
Pressure	Open
Fluid Temperature	
Water	70-250 °F
Sodium	500-1200 °F
Mechanical distortion	
Fuel subassembly	as specified in Reference 1
Other components	Open

Partial Assembly Hydraulic Tests

Model flow rates	5-150% of reactor flow rates
Fluid type	Water
Fluid temperature	Open
System pressure	Open

System Hydraulic Tests

Flow rates	5000-50,000 gal/min (or modeled flows)
Fluid temperature (water)	70-250 °F
System pressure	Same as reactor
Core arrangement	Open
Mechanical distortion	Open

Variables to be MeasuredIndividual Component Tests

Pressure drop in various sections

Flow rate

Fluid temperature

Partial Assembly Hydraulic Tests

Flow distribution as a function of reactor arrangement (visual studies)

System pressure

Fluid temperature

Hydraulic forces on flow ducts and components

Individual reactor duct flow rates

System Hydraulic Tests

Individual reactor duct flow rates

Pressure distribution at duct inlets and plenum region

Pressure distribution at duct outlet region

Fluid temperature

Total flow rate

System overall pressures

Reflector flow rates

Control and safety rod flow rates

Information from Test Results to be Applied to Design

The combined series of tests will characterize core flow distribution for a wide range of flow conditions and reactor physical arrangements. These results will cover steady state conditions only. The data will provide important design support, in that if flow is distributed properly these tests will verify the final hydraulic design of the reactor. If flow is improperly distributed, these tests will provide accurate values of the maldistribution and the specific cause and location of the maldistribution. In addition, these

tests would evaluate the effectiveness of design modifications made to correct observed flow maldistribution.

Test Facilities

The single component tests will be performed in already existing water and sodium flow facilities at Pacific Northwest Laboratory (water tests on fuel assemblies and ducts) and Liquid Metal Engineering Center (sodium tests on regulating, shim or safety rods). There will be no special instrumentation requirements.

The feature or partial assembly hydraulic tests will be performed in reduced scale transparent models of the inlet plenum and outlet plenum regions, and in a full sized single duct mockup. The first models will be used to examine the gross effects of reactor geometry on core flow-split with dye or other tracer injection and visual observation techniques. The second model will be used to verify hydraulic balance. Besides the dye injecting no special instrumentation requirement exists.

The system hydraulic tests will be performed in full sized or reduced scale models of the Fast Test Reactor, in-vessel systems. Water will be used as the test fluid. The major flow resistances (e.g., fuel assemblies, control rods, etc.) will be modeled with orifices. Individual duct flow rates will be measured by the pressure differential across these orifices. Local pressure measurements will be made in the fluid regions surrounding the duct inlets and outlets. These measurements will be made with small, traversing impact pressure probes.

Alternates to Performing Experiments

The following alternate procedures may be considered to performing these tests:

- 1) Base the reactor design and operation on analytical hydraulic models of the reactor

- 2) Perform hydraulic tests in the Fast Test Reactor prior to going critical
- 3) Increase reactor power and/or outlet coolant temperatures in small stages until assurance of predicted flow distribution is determined by actual performance.
- 4) Perform only analyses and testing on small scale sectional models to characterize the core flow split.

Alternate 1: Base Reactor Design and Operation on Only Analytical Models

A completely rigorous analytical model of the complex hydraulic behavior of coolant in the Fast Test Reactor is not possible in light of the current state of technology. For example, there is no satisfactory correlation or analytical method of accurately calculating pressure drop-flow behavior in closely packed bundles of wire wrapped, slender fuel pins.

In cases where uncertainties arise due to limits in present technology, a conservative or worst case approach is taken. This conservatism applied in each stage of the analysis often builds up to the point where the reactor operation is unduly penalized. For example, preliminary calculations show a possible worst case pressure gradient of 6 psi in the core outlet, a possible pressure change adjacent to tube inlet of 13 psi due to acceleration of the coolant and uncertainty of $\pm 15\%$ (12 psi) in the calculation of the reactor tube flow resistance. Combining these into the worst case, for one duct we have a possible 39% decrease in pressure drop corresponding to a 12% decrease in the coolant flow. This may be used directly and conservatively in a hot channel factor to calculate maximum fuel component temperatures. Since the proposed reactor operation is right at the fuel centerline melting limit with a transient

overpower factor of 1.25, the reactor power would have to be lowered 12%. Alternately, power could remain fixed while coolant outlet temperatures are reduced by 12%.

Alternate 2: Perform Hydraulic Tests in the Reactor Prior to Going Critical

This is a practical solution and was partially applied in the EBR-II when external system problems caused a delay in reactor startup. (Hydraulic tests on 0.6 scale model were conducted.) The information obtained from such tests is completely prototypic, but it is limited by the reactor instrumentation. Thus, flow rates in the individual reactor tubes and overall reactor flow rates could be measured. But detailed pressure gradients at tube inlet and outlet could not be measured. If no maldistribution exists, these tests provide excellent verification of reactor and core hydraulic design. However, if maldistribution of flow is found to exist, these tests will not necessarily indicate the exact location where the problem area arises. No intelligent modification to the reactor system with the exception of orificing can then be made on the basis of these tests. In addition, modification such as inlet plenum revisions, after all systems are installed, would be extremely expensive in time and money.

To summarize, the performance of core flow split tests on the reactor are a calculated risk. These tests automatically assume the level of go, no go propositions with the success of the test facility riding on the result.

Alternate 3: Increase Power in Small Stages until Complete Assurance of Proper Flow Distribution Is Obtained

The course of action is similar to that outlined for Alternate 2, except that there is no delay in going critical. The hydraulic tests could be accomplished with other startup

tests. There are several important safeguard questions that arise here, the major one being: Can the reactor be permitted to go critical with a large uncertainty in core flow split?

The consequences of performing core flow split tests in this manner are the same as in Alternate 2. If no significant flow maldistribution exists, the tests provide verification of the hydraulic design of the reactor. If on the other hand, significant flow maldistributions exist, the reactor operation will be severely penalized or a major delay in time will occur while expensive modifications to the reactor are made.

Alternate 4: Perform Only Analyses and Testing on Small Scale, Sectional Models to Characterize the Core Flow Split

This course of action represents an abbreviation of the test program rather than a true alternate to performing the tests. The reduced test program will be effective and economical in the early phases of the reactor design. It will provide a flexible instrument to directly compare competing design concepts and to determine the hydraulic effect of gross component changes in a single concept. However, the semiquantitative results obtained from these models will definitely not provide the detailed hydraulic information required to support and verify the final design of the Fast Test Reactor.

References

1. G. M. Hesson, A. Padilla, and J. M. Yatabe. Thermal-Hydraulic Development, BNWL-CC-1149, Pacific Northwest Laboratory, Richland, Washington. April 1967.
2. P. D. Cohn. "FTR Design and Analysis - Hydraulic Sodium and Mechanical Testing," BNWL-599, Pacific Northwest Laboratory, Richland, Washington.
3. D. S. Trent. "Application of Geometric Model for FFTF Hydraulic Core Mockup," BNWL-575, Pacific Northwest Laboratory, Richland, Washington.

BYPASS FLOWS (C-2)

Design Problem and Requirements

A small portion (about 5%) of the primary reactor coolant will be bypassed around the reactor flow ducts and pass upwards through distribution nozzles and leakage annular spaces in the lower tube sheet. The functions of the bypass flow are to:

- 1) Provide cooling to the fixed, main radial reflector region
- 2) Provide a large heat sink in the case of major loss of coolant or power excursion such as primary piping break
- 3) Reduce thermal gradients at the radial boundary of the core.

Flow velocities will be low, in general, and the pressure drops will be correspondingly small. Thus, slight changes in the geometry of the flow paths due to component mechanical distortion will cause a major redistribution of the coolant flows. An additional problem is that pressure drops due to thermal gradients (natural convection flows) may be significant with respect to the imposed pressure gradient in several locations in the bypass regions.

Cooling of the fixed main radial reflector assembly presents two potential problems. The first is to prevent overheating that could cause high temperature failures in the reflector components. The second is to prevent large thermal gradients, due to the exponentially decreasing radial heat generation profiles across the reflector assemblies. These gradients would give rise to large thermal stresses which could damage the reflector assembly.

Due to the low reflector heat generation, preventing high temperatures in the maximum heat generating sections of the reflector will be relatively simple. However, to match

coolant flow to heat generation throughout the reflector may present a much greater problem. Heat generation rates vary by a factor of more than 5 across the reflector requiring a wide range of coolant flow rates.

Mechanical distortion in the reflector assembly would cause a large percentage redistribution in flow rates due to low pressure heads involved. In addition, buoyancy heads due to thermal gradients (natural convection) are of the same order of magnitude as the imposed heads. Thus, changes in the gross or local temperature fields in the bypass region may affect flow distributions, which in turn will affect the temperature fields.

Stress problem arises from the close packing of the reflector rods (presently $P/D = 1.0$, rods in point contact along their axis). This spacing will cause temperature gradients around the rods. These gradients may be roughly predicted from limited experimental data on closely packed heated rod bundles and will contribute to the stress loading on the reflector rods.

Basic thermal hydraulic requirements of the bypass region in terms of coolant flows and distribution must be provided for all modes of operation and with the presence of temperature gradients and the possibility of deformed components. The major unanswered design problem is the characterization and optimization of the bypass flow inlet region. Various concepts involving a combination of distribution nozzles and leakage through the tube sheet must be compared under all proposed conditions of reactor operation. The first characterization will be analytical. However, due

to the complexity of the lower plenum geometry, the final characterizations will be performed experimentally.

The heat sink capacity of the pool in the event of a major loss of coolant or power excursion will be treated analytically. In cases where vessel breaks would provide drainage of the pool, analysis of the accident will be sufficient except for the possible core flow maldistribution caused by the egress. Here, scoping test information would be helpful.

Reduction of radial core thermal gradients at the core boundary by matching the bypass axial temperature rise to that of the core is not difficult since this is a gross rather than a local requirement.

The information required for the design includes:

- 1) Approximate hydraulic data to provide a criteria of choice for alternate schemes of introducing and distributing the bypass flow from the inlet plenum
- 2) Accurate bypass flow distribution and temperature profiles for the final Fast Test Reactor design for all proposed operating conditions
- 3) Information on the effect of mechanical distortion in bypass zone components on the bypass flow distribution.

These studies support design development and verification for the Fast Test Reactor.

Test Program

Test Purpose and Methods

There are no individual component tests planned in this program. The first tests performed will be on models and mockups, each tailored to investigate a specific area of bypass flow problems. A specific example is a reduced scale,

transparent model of the inlet plenum, flow ducts, and bypass region. These test facilities have previously been described in "Core Flow Distribution," page 3.2. In these tests, bypass flow distribution is examined in a semi-quantitative manner using visual observation and dye injection. These tests will be used to obtain the effects of various combinations of inlet nozzles and leakage around reactor ducts on the flow distribution in the bypass region. The test results will provide a basis for choices among the various bypass flow inlet configurations.

The second set of tests will be performed in a mockup of the reactor in-vessel systems. The purpose of these tests will be to obtain accurate pressure drop-flow distribution data in the bypass region over a wide range of overall reactor flow conditions. The tests will be made under isothermal conditions with water as the test fluid. Bypass region components will be accurately modeled. This exact modeling will be to facilitate tests on distorted geometries covered in the third series of tests. Local flow rates and pressure distributions will be measured with impact type probes and conductivity probes or thermistors in conjunction with the introduction of conducting salts in the fluid or heated water. The conductivity probe outputs will register system disturbances traveling between probes at different locations. A spectral analysis data acquisition system will reduce the outputs to time delays from which three dimensional local velocity fields can be calculated.

No attempt will be made to impose thermal gradients in these tests to determine the superimposed flow rates due to thermal buoyancy or natural convection. While this information is desirable to obtain, the difficulty of simulating reactor thermal conditions would render its obtainment unfeasible. The effect of natural convection circulation will be performed analytically using the isothermal flow distribution as its initial condition.

The third series of tests will determine the effect of mechanically deformed components on the bypass flow distribution.

Range of Parameters

Preliminary Plenum Optimization Tests

Model flow rates	10-150% of reactor (model) flow rates
Fluid type	water
Fluid temperature	open
System pressure	open

Bypass Flow tests

Flow rates	5000-50,000 gal/min (or modeled flows)
Fluid temperature (water)	70-250 °F
System pressure	same as reactor
Mechanical distortion	open

Variables to be Measured

Preliminary Plenum Optimization Tests

Bypass flow distribution as a function of lower
plenum arrangement
Total flow rate
System pressure
Fluid temperature

Bypass Flow Tests

Bypass local flow distribution
Bypass local pressure distribution
Fluid temperature
Total flow
System pressure

Information from Test Results to be Applied to Design

The combined test results will provide for the optimization of the bypass system arrangement and then final determination of the hydraulic behavior in the final reactor arrangement. In addition, information on the effects of mechanical distortion in components and component systems will indicate needed design modifications to prevent undesirable flow redistribution.

Test Facilities

The preliminary plenum optimization studies will be performed in a reduced scale transparent model of the core bypass and inlet plenum. The model will be used to look at the effect of changing bypass inlet arrangements on the bypass flow distribution using dye or other tracer injection systems; no other special instrumentation requirements exist.

The bypass flow distribution tests will be performed in full-size or reduced-scale models of the Fast Test Reactor in-vessel systems. This is the planned Hydraulic Core Mockup. Water will be used as the test fluid. The bypass components will be accurately modeled. The measurement of local velocity fields will be made with conductivity probes or thermistors and a spectral analysis data acquisition system. A system of traversing impact probes will be used to determine local pressure profiles. No other special instrumentation requirements exist.

Alternates to Performing Experiments

The alternates to performing the experiment are:

- 1) Base the bypass design and reactor operation only on analytical treatments of the bypass regions
- 2) Perform the experiments in the Fast Test Reactor prior to going critical.

Alternate 1: Base Bypass System Design and Reactor Operation Only on Analysis

A large portion of the test program will be designed to verify calculated behavior, rather than to establish unknown behavior. In addition, the use of isothermal water tests to determine flow behavior, without natural convection due to thermal gradients in the reactor systems will not provide completely accurate determinations of the thermal hydraulic behavior of the bypass system. The use of a heated sodium mockup to simulate the exact thermal hydraulic condition of the bypass system would be expensive with respect to the added accuracy of the results obtained.

The penalty of not performing the experimental tests will be an uncertainty in the operation of the reactor. The low flows and pressure drops in the bypass should not significantly affect other coolant streams in the reactor. However, all these streams (e.g., core outlet) strongly affect the distribution of the bypass flows. In addition, the low flows and pressure drops make the effect of mechanical distortion in components and thermal gradients significant on the flow distribution of the bypass streams. Since a major function of these streams is to cool the fixed reflectors which are replaceable, but only with difficulty, it is essential to characterize bypass flow behavior well enough to prevent failures among these fixed reflectors. The other functions of the bypass, neutron shielding and heat sink capacity, do not require a test program for their characterization.

Alternate 2: Perform Bypass Flow Tests in the Reactor Prior to Going Critical

Thermocouples additional to those normally required for reactor operation will be needed in the bypass and pool regions of the reactor. Normal meaningful readings cannot be obtained until the power ascension period. Tests performed in the reactor would probably only indicate that there

is coolant flow not exceeding a certain temperature in the bypass region. Delays in reactor operation would occur if changes are found necessary.

FLUID THERMAL GRADIENTS IN THE REACTOR OUTLET REGION (C-3)

Design Problem and Requirements

The coolant outlet temperature for all of the reactor flow ducts will not be uniform. Including planned open loop operation, there is a potential spread of up to 150 °F between maximum and minimum outlet temperatures. The temperature gradient across the flow duct walls can be of the same order as the temperature differences of the outlet of adjacent flow ducts. Since a maximum value can be set on the temperature difference across the reactor flow duct wall, analysis alone will be sufficient to determine the resulting maximum thermal stress levels in the flow ducts.

Another problem involves thermal gradients in the upper pool and outlet plenum region. If areas of poor mixing exist between core outlet streams of differing temperature levels, thermal gradients causing intolerably high secondary stresses may exist in outlet region components (e.g., upper tube sheets, outlet nozzles, piping, etc.) The resultant strain may cause mechanical distortions in the components which would affect the coolant flow paths surrounding them. An accurate mechanical analysis of the problem is possible once the temperature gradients and hydraulic forces on the component are fully specified. It is not possible to accurately determine the aforementioned thermal hydraulic quantities in this region of complex geometry. A combined analytical and experimental approach will be necessary.

Design needs verification that in the final design of the Fast Test Reactor no component in the reactor upper

section will exceed maximum permissible stress limits. The information must assure that no component will suffer a serious loss of function or physically fail for any mode of reactor operation.

Test Program

Test Purpose and Method

Tests will be performed in a model of the reactor with isothermal water as the test fluid. The objective of these tests will be to obtain flow distribution and flow mixing data in the outlet region of the Fast Test Reactor. This hydraulic information will then be used to calculate approximate temperature profiles in the outlet region coolant and components. Approximations are inherent in calculating these temperature profiles due in large part to the uncertainty in the ratio for eddy diffusivity for momentum and heat transfer. In addition, the use of isothermal water test will ignore the effects of natural convection. At the lower flow rate range of these tests representing shutdown conditions, natural convection will cause significant pressure differentials which must be accounted for in the thermal analysis.

The determination of local flow distribution and mixing will be performed as described for "Bypass Flows."

The effect of component mechanical distortion on flow mixing requires a great deal more analysis before a test program can be defined. In addition, the information obtained for normal configuration as to total stress levels will be an important base point to determine the possibility of component deformation in this region.

An additional phase of the program will be to evaluate design modifications (e.g., thermal baffling if such modifications are found necessary to prevent excessive thermal stress or temperature levels in components.) For this purpose the test results will be used in a relative, rather

than absolute basis to compare the mixing effectiveness of various upper region configurations.

Range of Parameters

Total flow rate	4000-40,000 gal/min (or modeled flows)
Fluid temperature (water)	70-250 °F
System pressure	same as reactor
Deformation	open

Variables to be Measured

Local flow distribution and mixing
Total coolant flow rate
Coolant temperature
System pressure

Information to be Transmitted to Design

These tests will provide the thermal boundary conditions for component mechanical analyses. The results of the mechanical analyses will determine if the combined stresses on the components will adversely affect their structure or function. If problem areas are discovered, the test program will evaluate the effectiveness of modifications designed to reduce the thermal gradients.

Test Facilities

The test facility required for the water mixing and flow distribution tests is identical to that described for "Core Flow Distribution."

Alternates to Performing Experiments

Alternate 1: Base Design of the Reactor Outlet Region
Components on Conservative Analyses

This is a practical course of action and has been used as a design basis for many currently operating fast and thermal reactors. The major differences between these reactors and the FTR are the higher outlet coolant temperatures

planned for the FTR and the larger disparities in outlet tube temperatures because of the open test loops. To fully assure reliable operation under the aforementioned conditions, it is necessary to obtain more accurate characterizations of the temperature profiles in the outlet region than are possible by analysis alone.

LOSS OF COOLANT ACCIDENTS (C-4)

Design Problem and Requirements

The term loss-of-coolant covers several possible incidents which result in less than normal coolant flows in a portion of or in the entire reactor core. A loss of cooling incident such as the rupture of a single reactor flow duct would reduce coolant flow in the reactor duct and fuel assembly. Even if a scram is initiated the affected fuel assembly can still overheat due to the loss of coolant sink capacity. A severe reduction in flow may initiate coolant boiling. The pressures built up from the phase change (e.g., at atmospheric pressure, the volume change for vaporization of sodium is about 3000:1) will cause the partial or total voiding of the ruptured tube. The critical flow rate of the two-phase mixture limits the outflow. This severe loss of coolant capacity could then cause melting of the affected fuel assembly. Further, the pressure surges accompanying the boilout of a reactor flow duct can lead to further damage by causing flow disturbances in adjacent ducts.

This report covers the loss-of-coolant accidents under the following categories:

- 1) Rupture of fuel assembly gas plenum or individual fuel pin
- 2) Displacement of a reactor flow duct from its hole in the lower tube sheet
- 3) Reactor flow duct rupture

- 4) Plugging of reactor flow duct or fuel assemblies by foreign material
- 5) Loss of pumping
- 6) Plenum or primary coolant system rupture.

The first four categories are single flow duct incidents. They will probably result in damage to the affected reactor duct and may propagate to other ducts. The latter two incidents are more serious in that they affect the entire core.

Rupture of a Fuel Assembly Gas Plenum or Individual Fuel Pin

The fission gas plenum for each FFTF driver fuel pin may contain gas at pressures approaching 1100 psi for end of life conditions. Maximum gas plenum pressures for the test fuel assemblies have not been defined at present. The rupture of one plenum and the escape of all the gas to the sodium coolant would result in a gas void in the coolant of about 0.1 ft^3 . If this gas void passes upwards to the duct outlet it would be too small to significantly affect flow conditions in the duct. Several gas plenums would have to simultaneously rupture before a potential hazard to the flow duct would result from this postulated accident. If the release could occur in such a manner that the gas void is driven downward into the core, the potential hazard is much greater. The volume of gas contained in the plenum could vapor blanket all of the fuel pins since the coolant volume in the subassembly is 0.1 ft^3 . In central flow ducts of the split conical core the void will also cause a positive insertion of reactivity. The combination of the decreased heat transfer coefficient and increased power could raise fuel temperatures sufficiently to cause damage to the fuel assembly. The totally insulated heat up rate for a FFTF driver fuel assembly will be on the order of $1100 \text{ }^\circ\text{F/sec}$ based on a linear power of 15 kW/ft .

Thus, if the void remains for more than a second, clad and fuel temperature limits would be exceeded by a large degree.

Two sets of information are required to determine the consequences of the gas plenum rupture. First, will any gas plenum rupture expel gas downward into the core? If this can be clearly demonstrated to be impossible, there is no further problem. The second set of information required is the maximum residence time and distribution of gas voids in the fueled region. This is not the simple problem of buoyancy and the hydraulic force of the flowing coolant stream driving a gas void upward through the core. Rather it involves a high pressure source forcibly ejecting gas voids downward and through a region of complex geometry with possible breakup and coalescence. Dependent on their size and initial velocity, some voids will reach equilibrium and be pushed upwards while others will possibly pass through to the tube inlet. Without a knowledge of the initial conditions (e.g., time zero velocity vector and size) even an approximate analysis of the problem is not possible.

A subsidiary accident is the rupture of the fueled section of the pin. Again the high pressure fission gas will be released into the primary coolant. The release of molten or hot solid fuel particles may accompany this release of gas. The gas release rates will in general be slower than for plenum breaks because of the semiporous resistance path that the fuel represents between the break and the major source of gas in the plenum. Thus, initially only the gas contained in the fueled section adjacent to the break will be released.

The gas released will blanket other fuel pins, increasing their temperature. In addition, fuel particles released represent a potential flow blockage source or if the released particles are molten, their temperature levels will be

sufficient to vaporize sodium, further disrupting flow and pressure conditions in the coolant channels. The effects of the total void, gas plus sodium vapor, on reactivity would be positive in the central flow ducts of the split conical core.

The analysis of the consequences of this accident is not possible without corroborating data on the release rates of gas and solid materials for each postulated rupture condition. Currently available information is not sufficient to enable a prediction of these release rates.

Displacement of a Single Reactor Flow Duct from the
Lower Tube Sheet

Improper latching or failure of the latch mechanism of the duct could displace a reactor duct. Hydraulic forces could then push the reactor duct upward and possibly laterally. Lateral movement is only possible if the core mechanical restraints break simultaneously. The thermal-hydraulic consequences of the displacement depend on the amount of displacement. However, at worst, two serious consequences may be postulated. First, the displaced duct is instantaneously starved for coolant and begins to heat up. Coolant boiling and eventual meltdown of the fuel assembly may follow. In addition, the hole in the tube sheet provides a lower pressure drop path for the remainder of the coolant in the plenum, reducing flow to adjacent reactor flow ducts. If this reduction in coolant flow is sufficient, overheating may occur, leading to potential damage to these fuel assemblies. In addition, the boiling in the displaced duct creates severe pressure surges which could cause severe flow disturbances in adjacent reactor flow ducts.

If a double plenum arrangement is used, the displacement of a reactor duct will increase the pressure of the lower

plenum chamber of the plenum and cause the partial loss of hydraulic balance. The effect this pressure increase may have on the hold down of other reactor ducts must be determined before the completion of the plenum design.

The determination of the sequence of events following displacement requires two groups of information. The first set of data required is the single phase coolant flow in the displaced reactor duct and the adjacent reactor ducts as a function of the axial and lateral (optional) displacement of the reactor duct from its original position. From this data, calculations can be made to estimate whether coolant boiling will occur and, if it does, approximately what the delay time is after displacement.

The second set of data will be the behavior of the system if boiling started in one or several reactor ducts. Due mainly to the low pressure of the system, boiling presents a potential hazard to the system. Volume changes for vaporization of sodium at typical Fast Test Reactor pressures are 3000 to 1. The volume change gives rise to a corresponding large pressure change which may be sufficient to expel all coolant from the reactor tube in a fraction of a second. Additional pressure surges may arise from the energy stored in liquid in the form of high superheats. With the loss of cooling capacity, gross fuel meltdown may follow.

The hydraulic supply and demand characteristics⁽¹⁾ of the affected reactor tubes which boil are extremely desirable. These data will provide information on what pressure drop must be supplied to a reactor tube to return its operation to stable, all liquid flow conditions.

A detailed discussion of the tests proposed to study loss of coolant in a single reactor duct is contained in Reference 1.

Reactor Flow Duct Rupture

A rupture in a reactor (i.e., driver, open or closed loop) duct, especially upstream of or within the core zone, will constitute a serious loss-of-coolant incident. Such ruptures could result from several causes. These include the loss of creep strength in the duct wall material due to high neutron exposure, high temperature corrosion, thermal stresses or vibration damage. A rupture which decreases coolant flow through a driver fuel assembly on the order of 60% or more may cause coolant boiling. Boiling could cause considerable damage in the affected reactor duct and fuel assembly. If coolant boiling propagates radially through the core, the entire reactor system would be endangered.

As in the duct displacement accident, two separate groups of information are required. The first set of data involves the single phase coolant flow rates in the ruptured duct and the adjacent reactor ducts following the rupture as a function of the size and location of the rupture. From these data, an evaluation may be made on the possibility of coolant boiling in the affected tubes. In cases where coolant boiling does occur, these data will provide an estimate of the delay time after the rupture before the initiation of boiling.

The second set of information required is the behavior of the system if boiling does occur in the ruptured duct. Again, the problem is that the large pressure buildup from the liquid-to-vapor phase change will cause rapid expulsion of the coolant. The critical flow condition limits the two-phase flow rates. This loss of coolant capacity, if not made up, will probably cause fuel assembly melting even with the proper operation of the scram system. In addition, the large pressure disturbances caused by the expulsion from the reaction duct may disturb flow in other reactor ducts and cause serious hydraulic hammer to other reactor components.

The hydraulic single and two-phase demand and supply data for the ruptured duct would provide important information on the amount of pressure drop required across the tube to re-establish stable, all liquid flow conditions. This information would support design and help establish criteria for the emergency coolant systems.

Plugging of the Reactor Duct on Fuel Assembly by Solid Material

This accident is described in the section on "Solids Deposition" and in Reference 1.

Loss of Pumping

The loss of pumping due to failures in one or more of the reactor primary coolant system pumps will significantly reduce coolant flow through the core. The reduction in coolant thermal sink capacity will in turn cause temperature rises in core components. There are several methods of emergency cooling proposed to limit the amount of flow reduction to the core to assure that no gross core damage through overheating will take place in the event of pump failures.

These emergency cooling methods differ according to the cause of pump failure. The two main causes are as follows:

- 1) Loss of electrical power
- 2) Mechanical failure in one or more of the coolant pumps or motors.

If a pump loses electrical power, emergency power systems immediately will come on line. These will be set up so that unless three simultaneous failures occur, power will be provided to the pumps.

If a mechanical failure such as a shaft failure causes loss of pumping, then several proposed emergency methods will prevent damage to the core components. These methods include

the use of redundant pumping capacity, auxiliary cooling systems, and/or the thermally induced natural circulation of coolant that exists within the reactor.

Analysis will provide the criteria of choice for these systems and will provide the data required for the final design. Tests to verify the performance of the final emergency coolant system are described in the following Test Program section.

Plenum or Primary Coolant System Rupture

These separate incidents can be grouped together because major breaks in either area would lead to the same results, severe reduction in overall core coolant flow rates. Emergency coolant systems will prevent overheating leading to gross core damage. The incidents should be differentiated in one sense: ruptures that release sodium to the pool, and ruptures that release sodium outside of the reactor vessel. The pool represents a large heat sink capacity which would provide additional heat removal capacity from the core and is an important factor in the progress of these incidents.

When this incident involves an overall core coolant reduction without maldistribution between individual reactor ducts, standard analytical procedures will suffice to estimate the all liquid flow conditions following the accident. Unless analysis shows that this is true and that significant maldistribution between the reactor ducts cannot occur for a postulated plenum or primary coolant system rupture, no tests will be considered.

If boiling were to occur in the core, information on the single- and two-phase hydraulic supply and demand would be necessary to determine what pressure drop is necessary to re-establish stable and all liquid flow in the core. These data would be vital to set criteria for auxiliary emergency coolant systems designed to prevent core damage.

Additional verification would be required of the performance of the emergency coolant systems in a heated sodium environment. These data would be required over a long term of operation to thoroughly test the mechanical components and to unearth potential corrosion and vibration problems.

Test Program

Test Method and Purpose

The experimental program will be divided into four separate portions:

- 1) Isothermal gas plenum and fuel pin rupture tests performed in a model of a single FFTF flow duct and fuel assembly. Water and gas will model the sodium coolant and fission gas respectively. A high pressure gas injection system will simulate various plenum breaks. The local pressure surges, total coolant flow, and the residence time of the gas voids in the fueled section will be measured. The residence time and distribution of the gas voids will be determined visually and with conductivity probes with salt injection (e.g., measurement of resistance or capacitance).

With the use of a specially designed pin, the second set of tests will simulate a fuel pin rupture. This pin will be filled with compacted powder to model the resistance of the fuel to gas flowing from the fission gas plenum. This pin will be ruptured by pressurizing a predefected region and the local pressures, total coolant flow and the residence time of the gas voids will be measured by the previously described techniques.

The results of these hydraulic tests will provide a firm basis for analytical models to determine the heat transfer and physics behavior of the system following the rupture.

- 2) Loss-of-coolant tests in sodium with heated models of a single FFTF fuel assembly and flow duct. These tests are described in detail in Reference 1 and will provide information on the hydraulic demand requirements necessary to re-establish stable single phase flow in the reactor after coolant boiling has occurred.
- 3) Isothermal water tests on a reactor system wide basis to determine the single phase hydraulic behavior after the loss-of-coolant accident occurs. The accidents simulated will be the displacement of a duct from the lower tube sheet and the rupture of a flow duct. In both cases the major measured variables will be the flow rates in the affected flow duct and the adjacent flow ducts. These tests will determine if coolant boiling could occur from a given displacement or rupture. In addition, if coolant boiling would occur, the test results will be used to calculate the delay time before its initiation.
- 4) Proof tests of the operation of the emergency coolant systems in the FTR. These tests will verify the long term operation of the mechanical components of the emergency coolant systems in a high temperature environment. If problem areas such as fretting, corrosion or oxide formation are unearthed, the test program will be re-oriented to investigate these specific problems.

The purpose of the four test programs will be to provide all practically obtainable data to support the evaluation of serious loss-of-coolant accidents in the Fast Test Reactor. It is considered impractical and uneconomical to attempt to simulate the boiling portion of the accidents on a system wide basis. Thus, the separated groups of data obtained in single tube and system hydraulic tests, heated single fuel assembly tests and emergency coolant system performance tests will be tied together by analyses to determine the complete

sequence of events following a specific loss-of-coolant accident. An additional task of the analytical program will be to extend the results of the experiments to be performed. Obviously all displacements, ruptures, and plugging incidents cannot be examined experimentally. A limited number of specific worst cases must be examined, their results used to verify the accuracy of the analytical models, and the final combined results applied to the overall safeguards evaluation of the reactor.

The analytical model will have to couple heat transfer and fluid flow behavioral equations to those of physics equations describing the neutronics behavior, similar to those written for thermal reactors, such as FARM or FORE. Additional information of a general nature would be necessary for the analytical model such as sonic velocity or critical flow data for the two-phase fluid is not included in the scope of the test program because there are LMFBR programs which are presently proposed to produce these data.

Range of Parameters

Single Reactor Duct Tests

Heated Tests in Sodium

Described in Reference 1.

Gas Plenum Rupture and Fuel Pin Rupture Tests

Flow rate	200-500 gal/min
Fluid temperature (water)	70-250 °F
System pressure	same as reactor
Rupture size and location	open
Plenum gas pressure	900-1200 psi

Water System Loss-of-Coolant TestsDisplacement of a Reactor Duct from the Lower Tube Sheet

Flow rate	20,000-40,000 gal/min (or modeled flows)
Displacement	open
Fluid temperature	70-250 °F
System pressure	same as reactor

Reactor Duct Rupture

Flow rate	20,000-40,000 gal/min (or modeled flows)
Rupture size and location	open
Fluid temperature (water)	70-250 °F
System pressure	same as reactor

Sodium Emergency Coolant System Verification Tests

The range of parameters must remain open until the criteria for the emergency coolant system design have been established.

Variables to be MeasuredSingle Reactor Duct TestsHeated Tests in Sodium

Described in Reference 1.

Gas Plenum Rupture and Fuel Pin Rupture Tests

Gas void residence time and distribution
 Local coolant velocities
 Local coolant pressures
 Total flow rate
 Fluid temperature
 Plenum gas pressure
 Initial conditions of flow and pressure

Water System Loss-of-Coolant TestsDisplacement of a Reactor Duct from Lower Tube Sheet

Coolant flow rate in displaced duct
Coolant flow rate in adjacent ducts
Pressure drop in ducts
Amount of displacement
Fluid temperature
Total coolant flow rate

Reactor Duct Rupture

Coolant flow rate in ruptured duct
Coolant flow rate in adjacent ducts
Pressure drop in ducts
Fluid temperature
Total coolant flow rate

Sodium Emergency Coolant System Verification Tests

Delay time in operation*
Coolant flow
Pressure drop in system
System temperature

Information from Test Results Transmitted to Design

The information from the test results will support the design by determining the consequences of loss-of-coolant accidents. These data will assist in assessing which accidents must be positively prevented in the design of the reactor. They will also help in setting the criteria for safety instrumentation and auxiliary emergency coolant systems design to prevent damage in the event of certain loss-of-coolant accidents. Finally, the tests will provide verification of the proper operation of the emergency coolant systems.

* All measurements made periodically over a long time period.

Test Facilities

The sodium heat transfer facility and the electrically heated test section required to perform the single fuel assembly loss-of-coolant tests are described in Reference 1.

Gas plenum and fuel pin rupture tests will be performed in an isothermal, water test facility. An exact model of an FFTF fuel assembly and flow duct will be used. Gas injection systems capable of simulating plenum and fuel pin ruptures will be incorporated in the model. One test pin will be filled with a compacted powder to simulate the capillary gas transport resistance of the oxide fuel. For other fuels other special pins will be fabricated. Special instrumentation will include high speed motion picture cameras and viewports to follow the void. In addition, small pressure probes and productivity probes (e.g., measuring resistance or capacitance) with salt injection will be used to measure transient local pressures and velocities in the fuel assembly coolant channels.

The water system loss-of-coolant tests will be performed in a full sized or large scale model of the reactor in-vessel systems. The model will permit the controlled displacement of a single flow duct and the insertion of a ruptured flow duct. The measurements of flow and pressure will be on a steady state basis and the only special instrumentation requirement will be additional pressure sensors near the displacement or rupture.

For the sodium emergency coolant verification tests the test facility will be the FTR. The test fluid will be sodium at temperatures up to 1000 °F and of prototypic purity. No special instrumentation requirements beyond the standard pressure and flow sensors will be required.

Alternates to Performing Test Program

- 1) Base the reactor design and safeguards evaluation on analytical evaluations of loss-of-coolant accidents.
- 2) Perform single tube transient tests with heated bundles which carefully simulate actual loss-of-coolant accidents.

Alternate 1: Perform Only Analyses of Loss-of-Coolant Accidents

This course of action has been followed in support of many reactors. With the current state of technology accurate analysis of the sequence of events following loss-of-coolant accidents, in which boiling or severe maldistribution of coolant flow may occur, is not possible. This leads to the compounded use of ultraconservative estimates of what could happen.

The analytical model can be greatly strengthened by experimental data characterizing specific steps in the sequence of the accident.

The program outlined in the previous section attempts to do just this; that is, combine analysis and practical, specific experiments into a more realistic synthesis of the consequences of loss-of-coolant accidents.

An additional problem that cannot be handled analytically is verification tests of auxiliary emergency coolant systems designed to prevent core meltdown in the event of a loss-of-coolant accident. Here, actual tests in heated sodium must be performed.

Alternate 2: Perform Single Tube Tests with Heated Fuel Assembly Models to Completely Simulate Loss-of-Coolant Accidents

These experiments would not be an alternate course of action but represent an addition to the proposed test program. The test results are considered very desirable but not

absolutely necessary at this time. In addition, transient tests would involve great difficulty and expense in their performance. At a later date, after the completion of the proposed test program, the performance of these tests should be reconsidered in light of knowledge available at that time.

References

1. G. M. Hesson, A. Padilla, and J. M. Yatabe. Thermal-Hydraulic Development, BNWL-CC-1149, Pacific Northwest Laboratory, Richland, Washington, April 1967.

CAVITATION (C-5)

Design Problem and Requirements

Cavitation of the reactor coolant can occur in any location where the local pressure falls below the vapor pressure of the fluid. The possibility of cavitation is high when the coolant is at high temperature and flowing in a region where sudden velocity changes occur. Critical areas for possible cavitation in the Fast Test Reactor include:

- 1) Coolant pumps
- 2) Reactor duct orifices
- 3) Fuel assembly region
- 4) Support tube sheet-flow duct clearance (leakage openings).

Cavitation of the reactor coolant could have the following consequences:

- 1) Cavitation damage to components due to the collapse of voids
- 2) In the core region, voids formed will cause unpredicted power perturbations
- 3) Possible reduction in the superheat requirements necessary to initiate boiling by introducing preformed voids.

The first item is the one of major concern. Cavitation damage may severely reduce the operational life of components, adding to fuel cycle costs, reducing plant availability, or leading to a hazardous condition to the reactor. For example, a failed pump impeller in one primary pump will lead to serious loss of coolant condition. In addition, if solid particles were released they would be swept through the reactor system to cause possible channel plugging or damage to other components.

The latter two items, void driven power perturbations and reductions in superheat, do not involve as serious hazards to

the reactor as item one. The reactivity perturbations caused by cavitation bubbles due to the sodium void coefficient should not be large in magnitude. This is due to the short life time of the voids in the subcooled coolant stream. The collapse of cavitation bubbles is almost instantaneous and very few should reach the active core or have a long life there. It is not expected that the power perturbations caused by these voids will be large enough to represent a hazardous condition to the reactor. However, the power perturbations may be large enough to adversely affect some of the short term fuel tests. The effect of the voids on the superheat required to initiate boiling is difficult to assess within the framework of reactor safety. Lowering the superheat requirement per se has no affect on the operation of the reactor. The problem is that another indeterminate factor has been added to the complex problem of determining the behavior of sodium boiling in a reactor channel.

The major problem is to determine where and under what conditions significant cavitation in the reactor coolant can occur. In addition, if cavitation does occur the design modification made in the individual component or system to prevent cavitation must be evaluated.

Analysis alone will not determine whether or not cavitation will occur at a specific location. The prevention of cavitation in components is usually done on the basis of past performance and established design rules. The analysis performed will be to determine fluid velocity head changes in the fuel assembly region, tube orifices, pumps, etc. These will then be compared to the reduction in pressure required to reach the saturation point at the maximum possible fluid temperature in the region. The approximations involved in making these calculations in certain areas of the reactor plus the lack of knowledge of the superheat requirements to

produce voiding will make the results of the analysis less than definitive as to whether or not cavitation will occur. However, the analysis will be valuable in scoping the experimental program and providing a prior estimate where cavitation may be a problem.

In terms of the reactor, the major purpose of this program is to support design verification, component and system characterization and system reliability.

The final information required is assurance that in the final design of the FFTF, cavitation will not occur at high enough levels to cause damage to components or to disrupt test programs. The accuracy of this information must be sufficient to provide a high confidence level, that damaging cavitation will not occur under all proposed conditions of operation.

Experimental Program

Test Method

The test program will be performed in two phases:

- 1) Individual component cavitation tests in heated sodium.
- 2) Qualitative tests in the hydraulic water core mockup to unearth system cavitation problem areas not investigated in the individual component tests.

The individual component cavitation tests will be a portion of the component performance evaluation tests. Cavitation will be detected by post inspection for damage and by the use of acoustical pickups to detect the noise produced by the generation and collapse of voids. The test conditions will cover the range of temperature and flow conditions that the component will experience in the reactor. Every reactor component will not be tested in this manner. However, the major components where cavitation may occur such as the pumps and fuel assemblies will be tested.

The second phase of testing will be performed in a mockup of the in-vessel system with water as the test fluid. The major purpose of these tests is to uncover cavitation problem areas in components which were not previously tested individually or problems which only arise due to conditions which require a simulation of the total system. An example of the latter case would be where a combination of several mechanically distorted components cause a flow aberration involving a large change in the fluid velocity. It must be emphasized that only the detection of cavitation is to be attempted; no attempt will be made to assess the extent of damage that may occur from the test results.

The methods used to detect cavitation will be threefold:

- 1) Visual observation of the flow streams through viewing ports or transparent flow channels. This method will be used wherever the geometry of the system permits.
- 2) Acoustical probes to detect the noise caused by void formation and collapse. This method will be used in areas such as the core inlet where visual observation will not be possible.
- 3) Postinspection of components to find areas damaged by cavitation. This test method will not assess damage rates but only the presence or absence of cavitation. The postinspection observations will serve as a verification of the two previously mentioned techniques and, in addition, may uncover problem areas missed by the other two techniques.

The tests to determine the effect of mechanically deformed components on cavitation of the fluid remain open at this time. An extensive analysis of possible deformations and combination of deformations and their effect on flow velocities must be made before these tests can be defined. Such analyses have not been done, nor is the reactor design firm enough at this time to support such analyses.

Range of ParametersSingle Component Tests

Flow rates	100-125% of proposed flow through the specific component
Fluid temperatures (sodium)	100-110% of the thermal operating conditions for that component
Pressure	50-100% of the operating pressures for that component

Water Core Mockup Cavitation Tests

Flow rates	20,000-50,000 gal/min (or modeled flows)
Fluid temperature (water)	70-250 °F
Pressure	reactor pressure
Component distortion	open

Variables to be MeasuredSingle Component Cavitation Tests

Cavitation damage (post operational inspection)
Cavitation noise (frequency, amplitude and location)
Flow rate
Fluid temperature
System pressure

Water Core Mockup

Visual observation
Cavitation noise
Cavitation damage (qualitative)
Flow rate
Fluid temperature
System pressure

No quantitative measure can be placed on the accuracy of the cavitation detection methods. However, the postinspection tests will reveal any significant damage.

Information from Test Results to be Applied to Design

The individual component tests performed as a part of the performance evaluation tests will determine if cavitation leading to damage occurs in the operation of that component. In addition, if cavitation does occur, further tests will be performed to evaluate design modifications aimed at suppressing void formation.

Test Facilities

Individual component cavitation tests will be performed as part of the component performance evaluation tests. These tests will be performed at ANL-CCTL for the fuel subassembly, at LMEC-ACTI for the primary pumps and similarly suitable facilities for other components.

Semi-qualitative cavitation tests require a water mockup of the in-vessel section of the reactor. The mockup could be to scale or a full size replica of the reactor without affecting these semi-qualitative experiments.

Alternates to Performing Experiments

The following alternate procedures may be considered:

- 1) Perform only individual component cavitation tests in combination with component performance evaluation tests
- 2) Prevent cavitation in the reactor using analytical models and proven design techniques.

Alternate 1: Perform Only Individual Component Cavitation Tests

This is a practical and reasonable alternate to performing the entire test program. In fact, cavitation tests for other

fast and thermal reactors have generally been performed in this manner. However, the Fast Test Reactor will operate with higher coolant temperature and sodium velocities than other fast reactors. These factors would tend to increase the possibilities for cavitation and would increase the desirability for examining the entire system for possible sources of cavitation.

Another question that can not be answered by single component tests is what combinations of mechanical distortions such as bowing, displacement or swelling could cause cavitation in a specific region of the reactors. Obviously, all such cases will not be examined in the core mockup tests. However, such facilities should be available to study certain critical cases.

Alternate 2: Depend on Analysis and Proper Design Guides to Prevent Cavitation from Occurring

This course of action represents a fairly high risk to the operation of the reactor. Analysis and proven design practice alone will not assure freedom from coolant cavitation in reactor components. If during the operation of the reactor no significant cavitation damage occurs, this alternate will result in the savings of several thousands of dollars. If significant cavitations occur, this course of action could lead to lengthy delays in the operation of the vital Fast Test Reactor.

GAS ENTRAINMENT (C-6)

Design Problem and Requirements

The entrainment of inert gas in the coolant stream may occur in the following areas:

- 1) Sodium pool surface-reactor cover gas interface
- 2) Free sodium surface primary pumps
- 3) Intermediate heat exchanger.

Gas entrained in sufficient quantities could have the following detrimental effects on the operation of the Fast Test Reactor:

- 1) Sudden power perturbations due to the sodium void coefficient. These power perturbations could be negative or positive depending on the location of the void within the core
- 2) Reduction of the thermal conductivity and heat capacity of the sodium coolant stream
- 3) Possible reduction in the heat transfer coefficient in the passing of large bubbles through the full assembly due to gas blanketing
- 4) Possible damage to components similar to cavitation damage.

The first three items will affect the safe operation of the reactor, though the effect of item two is probably slight. In addition, items one and two will affect the conditions of fuels and materials test programs. The fourth item may reduce the lifetime of certain components, adding to fuel cycle costs and reducing plant availability.

Location and Extent of Gas Entrainment

The first problem areas to be considered will be those in which gas entrainment could occur in an isolatable component. The two components having high susceptibility to gas entrainment are the primary system sodium pump of free surface design and the intermediate heat exchanger with a gas space on

the shell side of the exchanger to protect the upper tube sheet from extreme thermal gradients. Established design practices and analyses of these components will prevent the occurrence of entrainment. However, tests should be performed to make sure of this.

With respect to the entire system, an area of vulnerability is the interface between the reactor sodium pool and the cover gas space. The analysis of the problem is complicated by the complex and restricted geometry caused by the reactor and closed loop tubes extending through the interface. It will be difficult to even approximate the mass transport governing disturbances at the surface of the pool caused by the underlying flows from core outlet and bypass coolant. Characterization of the surface is vital, as the formation of surface turbulence in the form of waves, vortices or humping will greatly increase the amount of gas entrained. Without experimental collaboration, analytical models fall into the realm of guesswork.

The potential problem of gas entrainment can be more serious with the backup vertical core. Gas entrainment in the core outlet coolant will have a higher probability due to the nature of the outlet designs.

Detrimental Level of Entrained Gas

The shape and distribution of the gas voids affect the amplitude and length of the power perturbations. Although physics information will not be available, physics analytical models should be able to bracket the perturbations caused by a given amount of entrained gas within narrow limits.

Reduction of sodium thermal conductivity and heat capacity due to gas entrainment can not be predicted analytically and must be determined experimentally. With this information, standard heat transfer calculations may be performed to

determine the effect on fuel temperatures. Unless very large amounts of gas are entrained, the reduction in coolant sink capacity should not be significant.

An analysis of the effect of entrained gas on the operation of mechanical components such as the reactor pumps can only be approximated. The final results will have to be obtained by test.

Caused by the accumulation of the entrained gas, large bubbles blanketing the fuel assembly will reduce the heat transfer coefficient in the fuel assembly. In their traverse through the core, such bubbles produce an insulating blanket over the fuel assembly. With heating rates on the order of 1100 °F/sec for an insulated fuel pin, these bubbles would have to be very large to cause damage even if they produce an increase in reactivity of the fuel assembly.

Potential damage to components caused by the collapse of the entrained gas bubbles (like cavitation damage) is amenable to a certain amount of analysis. For example, regions in the reactor where velocity changes are large enough to cause pressure heads sufficiently large to collapse the bubble can be identified. However, the final assessment of damage must be done experimentally.

The information provided by these studies will determine what the maximum permissible levels of entrained gas will be from the standpoint of overall reactor operation and also from the standpoint of specific fuels testing programs.

Design Modifications to Prevent Gas Entrainment

The procedures to prevent gas entrainment once it is detected are standard. For example, at the reactor pool surface flow baffling could be used to damp disturbances at the pool surface. The real problem in the design of these systems

is to characterize the location and magnitude of the gas entrainment. Further, the baffling systems, etc. must be tested in place to verify their effectiveness.

The required information for design is whether or not significant gas entrainment can occur in the present reactor design. Further, will the design changes made to prevent gas entrainment, if it occurs, be effective. This program supports design development.

Test Program

Test Methods

Gas entrainment tests will be made in conjunction with the performance tests of the reactor pumps and intermediate heat exchanger. The first tests are expected to be performed in large, sodium test facilities such as the Sodium Component Test Installation (SCTI), the planned Sodium Pump Test Facility (SPTF), or the proposed Auxiliary Component Test Installation (ACTI) at the Liquid Metal Engineering Center (LMEC).

For the next set of tests, an isothermal air-water system to model the sodium-inert gas system will be used in semiquantitative tests of gas entrainment at the reactor pool surface. The test facility will consist of a mockup of the upper region reactor in-vessel systems. Where consistent with overall planning, a composite mockup may be used. Extensive use will be made of visual studies through view ports located above the pool and at the reactor outlet region. These studies will provide a strong indication as to whether or not a gas entrainment problem exists. If the fluid surface is smooth and no gas bubbles are observed in the outlet coolant stream, gas entrainment will not be considered a serious problem. Visible evidence of surface turbulence and gas bubbles in the outlet coolant stream indicate a gas entrainment problem. In this

case, further tests will be carried out to more quantitatively measure the amount of entrained gas. Periodic sampling and stoichiometric analysis will then be used to measure the gas entrained. In addition, fluid velocities at varying depths below the pool level will provide data for more accurate analyses of the fundamental fluid processes leading to the problem.

To cover the entire range of reactor operation, simulated reactor flows will be operated over a wide range. Where gas entrainment occurs at any flow, the results of these tests will be used in reactor systems modifications to correct the problem. These modifications will in turn be evaluated in further tests of the same nature.

A subsidiary test will inject gas into the coolant stream to determine if this gas will accumulate in any region of the reactor to be released as bubbles large enough to cause core damage. Again, visual techniques will be used.

Final verification tests providing quantitative results on gas entrainment rates must be performed on the Fast Test Reactor. With all components of the primary coolant system in place, the test results would encompass all possible sources of entrained gas. The gases would, of course, be the actual gases to be used in the reactor cover gas system. Periodic sampling and stoichiometric analysis would measure the amount of gas entrained. Dependent on the results of the water system tests and the single component tests, the sodium system tests as a part of the non-nuclear tests may not be considered necessary. This is more fully discussed under Alternates.

Range of ParametersSingle Component Tests

Flow rates	50-125% flow for specific component
Fluid temperature	600-1200 °F
System pressure	dictated by component

Water System Tests

Flow rates	
Primary	0-40,000 gal/min (or modeled flow)
Auxiliary	0-20,000 gal/min
Fluid temperature	70-250 °F
Gas pressures	open
Core arrangement	open

Sodium System Tests - FTR

Flow rates - Primary	0-40,000 gal/min
Fluid Temperatures (sodium)	300-1000 °F
Gas pressures	open
Core arrangement	reference

Variables to be MeasuredSingle Component Tests

Gas entrained in fluid
Level of introduced gas that would affect operation
Fluid temperature
System pressure

Water Systems Test

Surface agitation (visual)
Pool velocity
Air entrained in exit fluid
Reactor flow
Auxiliary flow

Information from Test Results to be Applied to Design

The first experiments performed in feature tests of various sections of the reactor, followed with water core mockup, will determine whether or not a problem area exists at the reactor pool surface. If none appears, the tests provide verification of this segment of reactor design. If surface turbulence is observed in the total core model tests, more quantitative tests will indicate corrective action in the form of design modifications and will in turn evaluate the effectiveness of these modifications. In addition, single component tests will determine if gas entrainment will occur in these components.

A reactor test will provide final, quantitative design verification of the entire reactor system with respect to cover gas entrainment.

Test Facilities

The single component performance tests of the pumps and intermediate heat exchanger are planned for the ACTI and SCTI facilities. These facilities would also be suitable for the component gas entrainment tests.

The water-air system gas entrainment tests at the pool surface will be performed in a full sized or scale model of the reactor upper region and later additional in-vessel systems. View ports will be required through the top cover and at the reactor coolant outlet. Also instrumentation to measure local pool velocities and the level of entrained gas will be required for these tests.

The sodium-gas tests will be performed in the reactor with the primary coolant system in place to examine all potential sources of gas entrainment.

Alternates to Performing Experiments

The following alternate procedures may be considered in performing these tests:

- 1) Perform only gas entrainment tests in the Fast Test Reactor prior to going critical
- 2) Reduce the test program to only visual studies performed in water
- 3) Base the reactor design and operation on analytical estimates of gas entrainment.

Alternate 1: Perform Only Gas Entrainment Tests in the Reactor Prior to Going Critical

These tests will provide excellent design verification and be completely adequate in themselves if no gas entrainment is discovered. However, if significant gas entrained in the primary coolant is discovered, these tests will not indicate where in the pool it is occurring. Thus, no intelligent modification of the reactor system to prevent gas entrainment can be made as a result of these tests.

If no problem area exists this method of testing is adequate economically. However, if a problem area exists, then this method of testing may become extremely expensive in lost time and dollars.

Alternate 2: Reduce the Test Program to Only Visual Studies Performed with Water

This test plus the individual component tests represent a minimum effort program which, while qualitative, would reduce the amount of testing significantly. If these tests lead to a reactor hydraulic design that prevents surface agitation of the sodium pool under all flow conditions (i.e., shutdown cooling to above normal flows) and, in addition, if the individual component tests lead to components with no gas entrainment possibilities, then further test programs are unnecessary except final reactor verification tests of the total systems performance.

In light of the savings in time and money, this reduced test program should receive careful consideration.

Alternate 3: Base the Reactor Design and Operation Only on Analytical Estimates of Gas Entrainment

This course of action involves high risk, because estimates of gas entrainment rates will be very difficult to make and of doubtful reliability. Gambles of this nature when compared to the cost of Alternate 2 seem inadvisable for a project of this magnitude.

SOLIDS DEPOSITION (C-7)

Design Problem and Requirements

The deposition of solids suspended in the reactor coolant may cause significant changes in local flow distributions and pressure drops. For example, if solid particles lodge in a driver fuel assembly and partially or totally block coolant channels, the coolant starved areas will overheat. In extreme cases, the overheating will be sufficient to cause coolant boiling and/or fuel meltdown. The consequence to the reactor to this case is a lengthy outage.

There are more subtle problems than the one cited; for example, in the bypass region where flow rates and pressure drops are low, preferential deposition of solids could significantly change the flow distribution. One possible result would be flow starvation and subsequent overheating in sections of the fixed radial reflector. Solids deposition may also occur in tube sheet leakage openings, affecting hydraulic balance characteristics or making fuel duct removal difficult.

Possible sources of foreign solids in the coolant stream would include:

- 1) Sodium oxide or other chemical complexes
- 2) Molten fuel release

- 3) Failure of components, such as rupture discs, spacers, filters, etc.
- 4) Introduction of foreign objects, such as tools during reactor maintenance
- 5) Mass transfer from hot leg to cold leg, e.g., inlet plenum.

There is presently no method proposed to monitor nonradioactive solids in the reactor coolant streams. Their presence will be induced in a secondary manner by the malfunction of a component or an off normal reading on an instrument. This is a reasonable course of action and has many precedents. However, it is fallible if the selective deposition of solids can be shown to cause a catastrophic incident under a postulated sequence of events. For example, if it were shown that the deposition of solids from a failed component could simultaneously block the coolant channels of several fuel assemblies, serious damage to the core would result.

Obviously, it is not feasible to investigate the problem of solids deposition by carefully simulating every conceivable incident introducing solids in the reactor coolant. The problem, rather, is to determine critical areas in the reactor where solid deposition is likely to occur, methods to prevent deposition and the consequences of deposition at these points to the operation of the reactor. Analytical calculations will be helpful in identifying these areas (e.g., areas of relatively low coolant flow) but not definitive since the geometry of the system is too complex to predict accurate local flow behavior. Even with flow behavior information for the normal condition of the reactor, the deposition of solids will distort the flow field in the same manner as mechanical deformation (e.g., component bowing, displacement, etc.) reducing analysis to approximation. Tests such as those described in the next section should be performed to fully characterize this problem area.

The purpose of the test program will be component and system performance characterization, design development, and design verification. The final information required is what potential hazard exists from a wide range of possible solid particles suspended in the reactor coolant and the effectiveness of preventative measures. This information must exist for all reactor operating conditions. The accuracy required is dependent on the individual component and the final reactor system design.

Experimental Program

Test Method and Purpose

The purpose of these tests is to determine where and to what extent a range of solids suspended in the reactor coolant will deposit in the reactor. The tests will also determine the amount of flow redistribution that may be caused by these deposits and the effectiveness of preventive measures.

The test facility will be a scale model or full sized replica of the in-vessel reactor systems. Water will be used as the test fluid. Solid particles of varying size and density will be introduced in the coolant streams and circulated for periods ranging from hours to several weeks.

Analytical calculations defining points of relative fluid stagnation or restrictive flow channel geometry, where settling is likely, will determine critical areas to be examined. In these areas, the local pressures will be monitored to detect changes indicating flow redistribution. If flow redistribution is indicated in a critical area, such as the inner fixed reflectors, more detailed tests will determine the new flow distribution by local velocity field measurements.

At the conclusion of the tests, the reactor model will be partially dismantled to determine the amount of depositing. The deposits found will be measured, weighed, and sieved to determine their exact nature.

The results of the first problem will locate positions in the reactor where solids deposition may occur and cause undesirable flow redistribution. Further tests will then be performed to evaluate design changes aimed at preventing solid deposition in these areas. All of the tests will be performed in water; no final verification tests in sodium are deemed necessary. In addition, these tests will consider only solid particle deposition without a consideration of chemical reaction. The deposition of oxides, nitrides, and carbides resulting from chemical reaction between the coolant and impurities contained in the coolant will be studied in small test facilities.

Another area that will be covered separately will be that of solid particles lodging in the fuel assembly. The restrictive geometry involved makes it possible that solids suspended in the coolant will lodge in the fuel assembly in spite of the high flow rates involved. Foreign particles blocking fuel assembly coolant channels represents a potentially dangerous situation in the fuel assembly due to its high heat generation, and detailed tests of the consequences of blocked channels will be performed with electrically heated models of FFTF fuel assemblies. These tests are described in detail in Reference I.

Range of Parameters

System Deposition Tests

Flow rates	1000-40,000 gal/min (or scaled values for model)
Fluid temperature	70-250 °F
Pressure	open
Particle size distribution	open

Variables to be Measured

Location, amount, and size distribution of solid deposit
Local pressure and pressure drop
Local coolant flow distribution
Overall coolant flow
Coolant temperature
System pressure

Information from Test Results to be Applied to Design

These tests will determine whether or not a problem area may result from the deposition of solids suspended in the reactor coolant in the present reactor design. Further, if a problem area does exist, further tests will evaluate design modifications proposed to solve it.

The heated fuel assembly tests in sodium will provide information for the safeguards analysis on the consequences of blocking one or more fuel assembly coolant channels with foreign matter.

Test Facilities

The test facility in which tests to determine the consequences of blocking one or more coolant channels in a single fuel assembly will be performed in a large, sodium heat transfer facility described as part of the Thermal Hydraulic program.

The system deposition tests performed in water will require a full sized or scale model of the reactor in-vessel systems. Instrumentation for flow and pressure distribution will be the same as that required for Bypass Flows studies. In addition, a solid particle introduction system will be required.

Alternates to Performing Experiments

The following alternate procedures may be considered in lieu of performing these experiments:

- 1) Operate the reactor without consideration of the problem. This course of action assumes that the normal monitoring system will detect any abnormality caused by solids deposition with a sufficient margin of safety to prevent hazardous conditions from occurring.
- 2) Develop instrumentation to detect nonradioactive solids suspended in the coolant system. Upon detection of a certain concentration of solids, the reactor will be shut down, the source of the solids identified, and the system thoroughly cleaned out.

Alternate 1: Place Full Dependence on Normal Reactor Instrumentation

This is a practical solution and has been used for many operating reactors. However, there is an undetermined risk involved, not so much in terms of a catastrophic design basis accident but rather in the possibility of a limited incident such as a single fuel assembly meltdown or operation under local off normal conditions in areas such as the bypass. There is, of course, no guarantee that the proposed experiments will provide information to positively prevent these incidents. However, they should provide a great deal of information on the behavior of solids in the specific reactor coolant system. A great deal more analysis must be completed on the possible effects of solids deposition before the proposed test program of this or other alternate courses of action may be chosen.

Alternate 2: Develop Detection Techniques for Nonradio-
active Particles in the Coolant

The major question is how to detect solids and what level of solids suspended in the fluid constitute a potential hazard to the reactor. This is a development program in itself, though it may form part of a larger program. Thus, the development of a detection device should not be considered a true alternate to performing the tests, but, rather, as a possible future supplement to the test program.

References

1. G. M. Hesson, A. Padilla, and J. M. Yatabe. Thermal-Hydraulic Development, BNWL-CC-1149, Pacific Northwest Laboratory, Richland, Washington. April 1967.

GAS OR STEAM CLOSED TEST LOOP RUPTURE (C-8)Design Problem and Requirements

Future FFTF fuel and materials testing programs may require tests to be performed in a fast neutron environment with cooling provided by flowing high pressure gas or flowing supercritical steam. Therefore, consideration has been given to conversion of one FFTF closed test loop to a gas or steam loop in the future. Before such a decision can be made, an important safeguards question must be answered. In the event of a gas or steam loop rupture, will serious damage or hazardous conditions result in the remainder of the reactor.

The gas loop will typically use either carbon dioxide at approximately 2000 psi or helium and other inert gases at 1000 psi. Coolant temperature may be high and the loop will be tentatively designed for operation above 2000 °F. A steam loop will operate with steam at pressures up to 2000 psi and temperature up to 1500 °F. Rupture of either of these loops below the sodium pool surface will cause severe flow, thermal, and mechanical shocks to the remainder of the core. In addition, with the steam release, energy release will occur through water-sodium chemical reactions. The power level of the reactor will change suddenly due to mechanical movement, the sodium void coefficient and (in the case of steam) the release of hydrogenous material (e.g., either as water, hydrogen, or sodium hydroxide) which acts as a moderator of fast neutrons. Since the core may be blanketed by hot gas or steam temporarily, core temperature rises may occur, leading to possible damage and/or coolant boiling, though with unimpeded core flow in-core boiling does not appear to be possible.

The sum total of these effects is a complex incident which could be of large magnitude. The forces involved could potentially lead to serious damage to the reactor and possibly to

operating personnel. An accurate analysis of this incident in the geometrically complex region represented by the reactor core is not possible. Accidents of this nature are generally treated by a hypothetical sequence of events and involve a combination of pessimistic or worst case estimates. The purpose of the proposed tests is to provide hydraulic and mechanical data to make a more accurate evaluation of this incident possible. This evaluation will be a major factor in the decisions as to whether or not one of the close test loops will be converted to gas or steam use.

The major purpose of this program is to support the safeguards evaluation of the complete testing function of the Fast Test Reactor. The required information is the consequences to the reactor of a major rupture in a gas or steam closed loop.

Test Program

Test Purpose and Method

The test will require a mockup of the reactor's in-vessel systems with water and air modeling sodium-gas (CO_2 or inert gases) or sodium-steam systems that would be involved in the actual rupture. The release of high pressure gas or steam from a test loop rupture will be simulated by the release of a high pressure air supply from a nozzle. A rupture disc or similar apparatus will be used to initiate the release.

During the brief duration of the test, rapid and violent transients in pressure and flow will occur. These will be followed by:

- 1) Visual (photographic) observation through viewing ports. The use of high speed motion pictures to follow the incident for more than a qualitative estimate of the

sequence of events is dubious due to the complex geometry of the inner structure and the possible multiplicity of interfaces (gas-liquid) on the field of vision.

- 2) Pressure measurements in many specific locations in the core and vessel region. These are the basic measurements in this test and will be made on a continuous basis with fast response sensors and recording equipment.
- 3) Strain rates on critical components such as reactor tubes, tube sheets, etc. Since the mockup tests can not model thermal stresses or thermal shocks of components, these tests will not give fully quantitative results on mechanical behavior but will rather, indicate critical mechanical problem areas requiring further investigation.

The water tests will provide gross hydraulic results of the accident. Power fluctuations, thermal shocks and possible in-core sodium boiling associated with this accident can not be mocked up in this facility. These quantities will be calculated from the hydraulic results obtained. Obviously, the thermal and nuclear effects will in turn affect the hydraulic behavior and these changes must be estimated to modify the data obtained. In spite of these shortcomings, the data obtained will provide a much firmer basis for the evaluation of a steam or gas loop rupture.

Range of Parameters

Pressures	500-2000 psi
Fluid temperatures	70-250 °F
Air temperatures	Open
Location and size of rupture	Open

Variables to be Measured

Fluid pressures (major variables)

Stress in components
Coolant temperature
Exit reactor coolant flow
Limited visual studies of pool level and environment, etc.

Information from Tests to be Transmitted to Design

Information from these tests will not be directly transmitted to design. Test results will be used in analytical safeguards evaluations to determine the feasibility of including a closed test loop using gas or supercritical steam as coolants.

Test Facility

The test facility required for these tests will consist of a mockup of the Fast Test Reactor in-vessel systems. The facility may be a full size or a scaled down simulation of the reactor. Insofar as possible, the mechanical behavior of component and component systems will be modeled. Water and air will be used as the test fluids modeling sodium-gas (e.g., helium or carbon dioxide) and sodium-steam systems.

Instrumentation beyond that planned for a basic hydraulic core mockup will be fast response data acquisition systems to follow transient pressure, flow and stress surges. The minimum system response should be on the order of 0.001 sec.

Alternates to Performing Experiments

The following are alternates to performing the experiments.

- 1) Base an evaluation of a gas or steam loop rupture on conservative analytical calculations to depict a hypothetical sequence of events.
- 2) Perform more realistic tests in a test reactor specifically designed for high risk transient and accident

experiments. Such a reactor has been tentatively proposed by the LMFBR program office.

- 3) Since significant risks may be involved, eliminate consideration of a gas or steam test loop until major liquid metal cooled fuel and materials test programs are completed.

Alternate 1: Base the Evaluation of Gas or Steam Loop Rupture on Conservative Analytical Models Based on Hypothetical Sequence of Events

The question of a high pressure gas or steam system within the core of a liquid metal reactor has not been seriously considered before. Due to the complex nature of the transient conditions following a rupture of such a loop within the core of the reactor, an analysis of considerable magnitude involving a large extrapolation of current technology will have to be made.

Certainly, precedents exist for such a course of action, and this analysis will take the form of a conservative estimate based on a hypothetical sequence of worst case events. Based partially on the results of this analysis, a decision will be made as to whether or not a gas or steam loop will be included in the FFTF.

Alternate 2: Perform the Tests in a Test Reactor Specifically Designed for High Risk Experiments

The accident test reactor proposed by the LMFBR program office is tentative, and this alternate must await further developments before it is given serious consideration. It should be noted that the proposed reactor was not specifically designed to handle tests involving other fluids.

Alternate 3: Delay Installation of a Gas or Steam Test Loop Until Major Liquid Metal Cooled Fuel and Materials Test Program Are Completed

This course of action is presently being followed. The gas or steam loop will not be installed in the initial cores of the Fast Test Reactor. Instead, the conversion of one of the closed loops is slated for consideration at an unspecified future date. However, since steam and gas cooled fast power reactors appear to have promise, a long term delay might cause a serious setback to these portions of the fast breeder and power reactor programs.

The total evaluation of this alternate is far beyond only thermal hydraulic criteria and, as such, requires discussion beyond the scope of this document.

D. CORE VIBRATIONS

A fluid flowing over the surface of a structure may cause certain elements of the structure to vibrate. Generally, flow induced vibrations in reactor systems have deleterious effects.^(1,2)

Within the FFTF core, vibrations may be most detrimental on the fuel elements. This vibration problem is treated thoroughly in Reference 9. The hydraulically induced vibration problem related to the poison section of the control element is discussed with the general subject of the control mechanism. The flow ducts may vibrate also and may amplify the motion of the fuel elements. Other elements, such as baffles, shielding, reflectors, and duct supports may vibrate, too.

In general, vibrations can foreshorten the useful lives of structural elements due to mechanical fatigue, galling, and fretting. Design problems foreseen and supporting test programs to help control or avoid serious vibrations in the core are discussed below.

DESIGN PROBLEMS AND REQUIREMENTS

In certain cases, if both the forcing function acting on the structure and the damping characteristics of the system are known, the vibration frequencies and amplitudes can be calculated from vibration theory. However, in the FFTF core, neither the forcing function or the effective damping can be predicted analytically. The mathematical theory of fluid elasticity (flow induced vibration) is complex;⁽³⁾ the fourth order partial differential equations contain fluid velocity terms which probably are both nonlinear and stochastic.⁽⁴⁾ The forcing mechanism, whether created by self excitation, vortex shedding or random turbulence fluctuations, is a subject of conjecture. Except for simple highly idealized models, fluid elastic problems cannot now be solved analytically.

The formulations of solutions to fluid elastic problems depend on whether the mechanism is primarily caused by a

parallel flow,⁽⁵⁾ or cross flow.⁽⁶⁾ For flow velocities parallel to the axis of the structural elements, most investigators⁽⁷⁾ agree that:

- 1) Vibration frequencies are independent of flow velocity
- 2) The vibration amplitude is directly proportional to approximately the square of the flow velocity
- 3) Amplitude increases with temperature
- 4) The virtual mass of the fluid which depends upon the geometry of the flow channel is a significant parameter
- 5) Self excitation has the character of negative damping and may cause instability.

The flow velocities internal to the FFTF flow ducts are approximately 30 ft/sec, while the bypass flow external to the flow ducts is only 3 ft/sec. Now:

$$\delta = CU^2$$

where

δ is the vibration amplitude induced by parallel flow,

C is a constant, and

U is the average flow velocity.

$$\frac{\delta_{\text{internal}}}{\delta_{\text{bypass}}} = \frac{CU^2_{\text{internal}}}{CU^2_{\text{bypass}}} = \frac{30^2}{3^2}$$

or:

$$\delta_{\text{bypass}} = 1\% \delta_{\text{internal}}$$

Hence, parallel flow should have relatively little influence on the vibrations of the flow ducts. Thus, any flow duct vibrations generated from external sources probably will be caused by cross flow.

One region where cross flow can produce significant vibrations is in the outlet plenum. As the coolant leaves the flow duct through the perforations in the tube wall, the flow

velocity changes from parallel to cross flow. Based on a bulk coolant flow at 30,000 gal/min ($67 \text{ ft}^3/\text{sec}$) and assuming that the outlet flow would be uniformly radial at a depth of two feet, the average cross flow velocity is 1 ft/sec. However, the bulk flow consists of the low velocity bypass flow as well as the higher velocity effluent from the flow ducts. Locally, cross flow velocities as high as 10 ft/sec are conceivable. The Reynolds numbers (10^5) are sufficiently low to cause periodic vortex formation.⁽⁸⁾

The Strouhal number, $S = \frac{fd}{U_c}$, is a measure of the frequency of vortex shedding where f is the shedding frequency, d is the diameter of the flow duct, and U_c is the critical cross flow velocity. The Strouhal number is essentially constant ($=0.2$) for Reynolds numbers in the range of $300 < R < 3 \times 10^5$. For a flow duct diameter of 5 in. and a natural frequency of 3.2 cycle/sec:

$$U_c = 6.7 \text{ ft/sec} \quad .$$

If the natural frequency of the fuel assembly equals the vortex shedding frequency, large resonant oscillations can occur. Thus cross flow velocities, $1 < U < 10 \text{ ft/sec}$, could produce resonant oscillations in the tubes. While mechanical fatigue of the ducts is unlikely, the ducts act as moving grounds to the fuel subassemblies. Hence, large stresses in the fuel elements can be generated by cross-flow-induced tube motion.

Neither the strength of the cross flow forcing function nor the length over which it acts can be calculated analytically. Hence, experiments are required to determine the magnitude and frequency of the cross flow induced vibrations. In any event, the identification of the components or component system where significant hydraulically induced vibration can occur is the first and most important problem.

The second problem is to determine the possible effect of the predicted levels of hydraulically induced vibration on the structure or function of the components and reactor systems. Again, analysis based on currently available materials data will be performed. However, these calculations will be approximate and are not considered adequate to support reactor design or operation without corroborating tests.

The third problem area is the effect of mechanical distortion on hydraulically induced vibrations. Bowing, displacement or swelling not only affect the mechanical behavior of the components and component systems but will also affect hydraulic behavior in the surrounding coolant and thermal gradients within components. This is a complex and very difficult problem to attack analytically or experimentally. However, the whole question of distortions must be adequately handled since they may affect the safety of the reactor. For example, the inward bowing of core tubes of thousandths of an inch will introduce serious reactivity insertions that may lead to coolant voiding and possible core meltdowns.

A great deal of analysis in mechanics and thermal hydraulics will have to be performed in this area before an effective experimental program can be defined.

The fourth problem is to evaluate design modifications proposed to prevent hydraulically induced vibrations in components and component systems uncovered in the three previous studies. Here, reliance will be placed on verification tests performed under conditions prototypic of the Fast Test Reactor.

This program will support design verification. Verification is needed that in the final Fast Test Reactor design no hydraulically induced vibration problem areas exist under any proposed operation. All of the proposed tests and analyses

are directed toward this final goal. The required accuracy for the results can only be defined on a component by component basis.

TEST PROGRAM

Test Objectives and Requirements

The first tests to be performed will be individual component vibration tests on critical items such as the fuel assemblies, flow ducts, emergency regulating and shim rods, intermediate heat exchangers and pumps. The purpose of these tests will be to identify components in which hydraulically induced vibrations are possible. Also to determine, insofar as is possible in isolated tests, the effect of the vibration on the structure and function of these components. These tests will be performed in conjunction with the performance tests of these components and will be performed in heated sodium facilities. Some preliminary tests on the fuel assembly will be performed in water. The detailed description of these tests are to be found in BNWL-599. The individual component tests will be carefully instrumented to measure deflection amplitudes and vibration frequencies. Reactor components will be employed for all the tests. The major problem in the interpretation of the test results will be the lack of proper simulation of the hydraulic boundary conditions at the component surfaces.* Additional drawbacks include; the lack of thermal gradients, radiation induced damage in other components and mechanical coupling to other reactor components (e.g., proper damping).

The second set of tests will be semi-quantitative, vibration tests performed in a flowing water mockup of the reactor in-vessel systems. These tests will use visual, simply instrumented (e.g., strain gages), and post inspection techniques to identify individual components and groups of

* *This is, of course, not true for many components such as fuel assemblies or intermediate heat exchangers.*

components where significant hydraulically induced vibrations may occur. These tests will examine components that were not tested individually, such as core tube sheets and systems of components such as the reactor tube, lower tube sheet structure, and where interactions between components must be considered.

The effect of mechanical distortion on component vibration will be examined in these system tests. However, a great many more analyses will have to be performed before this program may be described in detail.

Because the fundamental nondimensional parameters in fluid elasticity are not well established, the test section should have structural properties, material properties and flow channel configurations as nearly prototypical as possible. The density and viscosity of 700 °F sodium can be sufficiently approximated by 200 °F water. The natural frequency of the ducts can be corrected for the change in the modulus of elasticity with temperature. Each duct should be prototypical and should contain a mass simulating the fuel bundle. The perforated plates, which impose boundary conditions on the ducts (thus influencing the natural frequencies) should also be geometric replicas of the prototype. Because of quadrant symmetry, a 90° sector is sufficient to establish cross flow effects in the outlet plenum.

The true vibration test of all components will occur in reactor operation since only there are all conditions (i.e., thermal gradients, neutron fluxes, hydraulic forces, etc.) applied. The aforementioned test programs can only duplicate these conditions in a limited way. However, the test programs outlined above will unearth many potential problem areas, suggest design modifications and evaluate these modifications in laboratory tests.

Range of ParametersIndividual Component Tests

Flow rates (water or sodium)	50-125% of reactor flows for the individual component
Fluid temperatures	
water	70-250 °F
sodium	500-1200 °F
Pressure	5-125 psig
Mechanical distortions	Open

Water Core Mockup

Flow rates	50-125%
Geometry	1/4 sector or more
Fluid temperature	up to 200 °F
Pressure	125 psig
Structural distortion	Open

Variables to be MeasuredIndividual Component Tests

Amplitude of vibration
Vibration frequency
Hydraulic forces (stress)
Coolant flow rate
Coolant temperature
System pressures
Fretting damage (post-operational inspection)

Hydraulic Core Mockup Tests

Amplitude of vibration
Vibration frequency
Coolant flow rate
Coolant temperature
System pressures

Information to be Transmitted to Design

The test data will be converted into graphs showing the vibration amplitude as a function of bulk flow at various temperatures. The vibration deflection data can be converted to stress levels. The results of these tests will indicate areas where flow induced vibrations in reactor components may occur and the potential damage rates occurring from these vibrations. This information will be important to indicate required design changes or to verify the adequacy of mechanical hydraulic design characteristics of the FTR.

Test Facilities

The individual component tests will require only the flowing water and sodium facilities that already exist. These include the 189D Hydraulic Test Facility (for water fuel assembly tests) at Pacific Northwest Laboratories, the CCTL (i.e., for sodium fuel assembly tests), at Argonne National Laboratory, and existing facilities at the Liquid Metal Engineering Center.

The water tests on a systemwide basis will require a model of the FTR in-vessel systems with water as the test fluid. This may be full scale or a partial segment of at least one quadrant. Component mechanical behavior and mechanical interaction with other components must carefully duplicate that of the reactor. Instrumentation requirements will include view ports at several locations in the top cover and vessel sides, plus strain gauges, deflection monitors, and acceleration gauges monitored on critical components.

Alternates to Performing the Tests

Alternates

The following alternates exist to performing the test program.

Alternate 1: Base mechanical-hydraulic design on previous practice and analytical studies

Alternate 2: Perform only individual component vibration tests

Alternate 3: Perform small scale model tests

Evaluation of Alternates

Alternate 1: Base Mechanical-Hydraulic Design only on Analysis. This approach involves high risk. Analytical models for a complex reactor system require a detailed knowledge of the system mechanical, thermal, and hydraulic conditions. Knowledge of the latter two items does not exist in sufficient accuracy to, say, predict core inward deflections to 0.001 in. Yet, deflections of this order in the core flow ducts can produce significant reactivity insertions.

The cross flow induced vibrations at the outlet of the tubes can be reduced by inserting an intermediate tube sheet near the perforations. By increasing the degree of fixity on the ducts, the vibration effects can be further reduced. However, optimizing the structural configuration for vibration purposes may worsen other facets such as heat transfer or normal refueling operations. Reducing the bulk flow rate will reduce vibrations, too. The tradeoffs imposed by performing no vibration experiments are not obvious.

Without experiments, the vibration problems of baffles and other secondary structural elements cannot be foreseen. A less conservative approach is to observe whether vibration problems develop in normal operation of the prototype.

Alternate 2: Perform only Individual Component Vibration Tests. Performing only individual component vibration tests would represent an abbreviation of the test program rather than a true alternate. A significant savings in money and time may be affected by the course of action since the facilities required presently exist at various sites in this country.

However, in these tests the system hydraulic behavior will, in general, be different from that occurring in the Fast Test Reactor. For example, in the core coolant outlet region, the coolant flow and pressure distribution is the result of radial core outlet flow and bypass coolant flows. This intermingling occurs in a complex geometrical region. The duplication of these hydraulic conditions in a single component test facility would be quite impossible. Thus, in single component tests, the driving forces causing vibration will not be correctly simulated in magnitude, direction, or location. This should be considered as a serious drawback in making a small portion of the overall test program carry the burden of the entire vibration test program.

Alternate 3: Perform Reduced Scale Model Tests to Determine Pressure Profile. Performing small scale model tests would mean that the vibration amplitudes are small and that the deflection of the ducts do not influence the fluid pressure distribution; a small subsize mockup of the outlet flow region can be used to measure the pressure profiles. This mockup should consist of rigid cylinders with prototypical effluent perforations. The resulting pressure profiles would define the forcing function acting on the ducts.

Digital computer programs could be used to predict the undamped vibration amplitudes of the prototypical cylinders. This alternate is questionable because the influence of the deformable structure on the fluid pressure distribution is not known.

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E. THERMAL - MECHANICS

The mechanics program is comprised of several basic areas of investigation. These include:

Plenum tube sheet and reactor cover

Flow ducts

Integral structural analysis

Core accident analysis

Vessel, nozzle, and piping

Nozzle and seals

Much of the program is oriented toward the use of experimentation to develop and verify simple analytical models, with the results being used to formulate the more complex analysis of the reactor system. This procedure is necessary since prototypic loading conditions with prototype components are not easily simulated, and the procedure will yield an acceptable solution economically.

The integral core analysis, however, requires a test model of the core structure to obtain the true coupling between structural components and verify the applicability of the analytical models.

Included in the mechanics area is the driver closure development. The driver closure development deals with tests of prototype components in a prototypic operating environment.

The mechanics program is primarily oriented toward development support of design.

PLENUM TUBE SHEETS AND REACTOR COVER (E-1 and E-2)

Test Requirements and Objectives

The reference concept for the FTR incorporates perforated plates or tube sheets in the inlet plenum to the core and in the reactor cover. The inlet plenum and the reactor cover are the primary vertical and lateral supports for the reactor core. Because of the reactivity sensitivity of fast reactors to

small motions of core components, it is necessary to characterize the mechanical response of core support elements under all potential conditions of mechanical and thermal loadings.

The perforated plate or tube sheet forming the core inlet plenum is an important structural element in a reactor structure since it is directly coupled with the core.⁽¹⁾ Expansion and bending of the plate will cause changes in the fuel sub-assembly configuration and position during operation, thereby influencing the reactor performance. It is essential that tube sheet structures be well characterized to allow prediction of reactor response. In addition to reactivity changes in the core, the deflection of the support plates under the action of loads and temperature gradients, particularly if creep is developed, may cause the core component to bind or seize, complicating loading-unloading operations.

Alternate inlet plenum conceptual designs for the FTR include a single tubesheet core support (no hydraulic balance) and a double coupled plate (grid plate) core support (hydraulic-balance). In the double plate concept, the two perforated plates are connected rigidly together by short tubes or rods. The same general design considerations apply to both single and double support plates to prevent excessive deflections.⁽²⁾

The reactor cover in the reference concept is a large (18 ft diam) circular axisymmetric shell form (hemispherical or torispherical) pierced by numerous (~150) closely spaced nozzle openings. Shell thickness is about 6 in. except at the flanged edge. The dominant loading for the cover is internal fluid pressure but significant dead weight and thermal loads will be applied. The presence of the nozzle holes in an already difficult to analyze thick shell requires a sophisticated analysis. Rupture of the shell or excessive deformation preventing the movement of control rods or

moving core components to more reactive positions constitute potentially serious problems which must be evaluated and, if necessary, corrected.

The main loads which will be considered in the design of the FFTF tube sheets are:

- 1) Hydraulic pressure gradient
- 2) Gamma heating
- 3) Moments due to thermal bowing of the core subassemblies
- 4) Thermal shock from coolant temperature transients

For an isothermal plate under external lateral load,⁽³⁾ the stress is inversely proportional to the thickness squared. The thermal stress in a plate due to gamma heating⁽⁴⁾ is directly proportional to the thickness squared. This inverse thickness relationship complicates the design of a plate when both external lateral load and gamma heating are significant design factors: the thickness required to carry the lateral load may result in a high overstress when considering gamma heating, and the thickness required to keep the thermal stresses within allowable limits may result in a high overstress when considering the external lateral load.

The presence of the holes limits the application of classical plate analysis in obtaining a practical design for a tube sheet. Preliminary calculations for an idealized FTR inlet plenum tube sheet 6 in. thick with fixed edges, based upon current techniques show that the maximum stress developed in the plate is 32,500 psi due to a 100 psi lateral pressure and 49,700 psi due to gamma heating. The yield strength of 304 SS at 550 °F (the tube sheet initial environment temperature) is 17,250 psi.⁽⁵⁾ These preliminary calculations reveal the possible severity of the problem and indicate the need for more refined calculations.

The required design information to satisfy this problem is the complete characterization of the FTR inlet plenum and reactor cover as core support structures. The information must be in the form of analytic expressions confirmed and correlated by experiment.

Initial information required are parametric studies of both the support structures and the potential loadings to permit reasonable evaluation of limiting load conditions which may be imposed as operational or design constraints.

Test Program

Single Tube Sheets. The state of the art⁽¹⁷⁾ for the calculation of thermal stresses due to both internal heat generation (gamma heating) and temperature gradients is well established for the elastic range and simple geometries. However, due to the nature of the problem, a detailed analysis of the more complicated structure is necessary. If the gamma heating temperature distribution can be obtained, the finite element methods (to be discussed later) will provide the resulting thermal stresses. This analysis will take into account the presence of the holes in the plate. There has been considerable research done in connection with isothermal perforated plates. For the most part, this research has been performed on plates with an equilateral triangular array, a rectangular array, or some other symmetrical array of constant diameter holes.

There are three general methods which may be used to solve for the stresses and the deflections in an isothermal perforated plate. One method involves the concept of an "equivalent" solid plate,⁽⁶⁻¹⁴⁾ the second uses the "point-matching" technique, and third, the finite element technique. Of the three methods, the latter is the most versatile. Although it has apparently been little used for the perforated plate problem, the finite element method of analysis shows

great promise for this application. The philosophy of the procedure is to replace the perforated plate with an assembly of simple elements. Equations are then written which will satisfy both equilibrium and compatibility between the elements.

Triangular cylinder elements will be used to model the perforated plates; in the plan view, each element appears as a triangle; each triangle extends through the thickness of the plate. The plate is then approximated as an assembly of triangular prisms. The holes in the mathematical model will appear as six or eight sided prisms. To refine the analysis about the holes, an element with a curved surface may be developed.

One of the advantages of the finite element method is that it is not restricted to any symmetry, fixed patterns, etc. Therefore, it makes little difference whether the tube sheet has axial symmetry, triangular or rectangular hole patterns, different hole sizes, etc. The procedure is adaptable to any geometry.

A relatively good computer code is presently available for immediate use. This is the Structural Analysis and Matrix Interpretive System (SAMIS).⁽¹⁸⁾ The program has a triangular plate bending element which is based on the usual thin-plate assumptions except that shear deformations are possible. Due to the thickness of the plate this represents a rather distinct advantage over the two methods mentioned previously. Loading possibilities are quite general and appear sufficient (at least compared to the other two methods).

One final advantage of the finite element method is that the actual plate support structure can be better approximated.

Therefore, in view of the advantages offered by the finite element method, it will be used for the analysis of the single tube sheet.

Coupled Tube Sheets. The technical literature pertaining to the analysis of coupled tube sheets is rather limited. Marchertas⁽²⁾ has done theoretical and experimental work in connection with the deformation of a coupled tube sheet. He developed a computer program for a triangular array based on the "equivalent" solid plate concept utilizing the Rayleigh-Ritz perturbation method which involves an infinite series of polynomials. Cook⁽¹⁵⁾ has made a theoretical analysis of the stress distribution in an axially symmetric coupled tube sheet.

Neither of the above is applicable or suitable to the FTR problem. However, the finite element method is applicable if sufficient computer capacity and proper elements can be obtained. The SAMIS code does not possess the proper analytical element for this simulation. Even if the computer capacity requirements could be satisfied, the triangular element model for the plate must be provided with an additional constraint of the nodes surrounding the hole to relate the nodal behavior to the end of the coupling tube (which should be represented as a line element). With some development work, the finite element procedure will provide the best analysis.

Reactor Cover. The reactor cover is a perforated spherical shell. There is little literature available for this problem and in many instances such a shell is treated as a flat plate.⁽¹⁶⁾ However, the effect of the curvature can be significant and the behavior of a shallow shell is usually quite different from that of a flat plate.

As mentioned previously, the great asset of the finite element procedure is its generality. The triangular plate

bending element in the SAMIS code mentioned previously has an in-plane stretching capability that makes it suitable for the approximation of shell behavior. Therefore, the same SAMIS code will be used for the analysis of the reactor cover.

Test Specimen and Experimental Method. The proposed test program shall be conducted on both scale models (or prototypical replica) of the lower tube sheet assembly and the reactor cover assembly to determine the effects of the various loadings. The effect of each of these types of loads may be investigated separately (or combined) and the results superimposed to obtain the final solution. All tests will be run at room temperature.

To determine the effects of the lateral hydraulic pressure gradient a static test will be performed utilizing mechanical displacement type gauges, a birefringent coating, and electrical resistance strain gauges to obtain the required data. The specimen will be loaded with a uniform lateral pressure. The test rig will be designed to permit the end conditions to be varied from simply supported to fully clamped. The photoelastic, strain, and deflection data will be obtained at critical points by discrete location of the measurement devices.

The effect of thermal bowing of the shroud ducts on the perforated plate will be obtained in a prototypical mockup of the structure. These proposed experiments serve primarily to confirm the numerical analyses.

Range of Parameters. The parameters are the plate thickness, the uniform lateral pressure, and the boundary conditions. The plate thickness will be varied from the thin plate range to the thick plate range, i.e., $3 < \frac{\bar{a}}{t} < 10$, where $\bar{a}$ is the plate radius and t is the plate thickness. This variation will show the effect of shear forces on the deflection of the

plate. The pressure goes from 0 to 1000 psi in increments of 200 psi. These limits will insure that the actual loads due to various operating conditions fall within the range of the experimental results. The boundary conditions will range from simply supported to fully clamped.

Variables to be Measured

The variables to be measured are the load, the deflection, and the strain. The photoelastic patterns to be observed are the isoclinics (these relate to the direction of the principal stresses) and the isochromatics (these relate to the magnitude of the difference of the principal stresses).

Information Applied to Design

The results of the proposed analyses will optimize the design of the lower tube sheet assembly and the reactor cover. Increasing the thickness or changing the perforation geometry may not be sufficient to reduce the stresses to allowable levels; hence, lowering the fluid pressure or temperature gradients may be necessary. Such design decisions cannot be made until some of the proposed analyses are completed.

Alternates to Performing Experiments

First Alternate: Analysis

As noted in the discussion, other methods of analysis can be used. However, the Equivalent Plate Concept and the Point Matching Technique have less flexibility but require as much development time as the Finite Element Method. Whether the Point Matching Technique can be applied to a coupled grid plate assembly is doubtful.

Second Alternate: Experimental

Rather than simulate the hydraulic pressures with static loads, the lower tube sheet assembly can be tested in a partial water mockup. The deformation of the assembly can be

measured with strain gauges and profilometer. By using sodium rather than water the effects of thermal shocks can be measured.

This alternate provides more accurate lateral loadings for the tube sheet assembly; however, its major disadvantages are timing and cost.

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FLOW DUCTS (E-3)

Design Problem and Requirements

The flow ducts comprise the core as an assemblage of axial elements, supported axially as well as radially to a high tolerance. Design problems and uncertainties stem from two considerations: reactivity effects and structural integrity. Briefly, the problems are described under each category.

Reactivity Effects

Since the flow ducts contain the core fuel, their individual and collective positions affect the core reactivity. Therefore, motion of the flow ducts under thermal and hydraulic driving forces must be well-characterized and of such nature as to not constitute a safety or control problem. The important design characteristics are:

- Bowing
- Warping
- Vibration
- Radial Expansion
- Axial Expansion.

Each dimensional characteristic must be correlated with respect to the driving forces and the response time.

Structural Integrity

Close dimensional tolerances of the ducts are essential to maintenance of balanced coolant flow. Structural failures of the ducts may generate fuel-pin meltdowns and possibly result in a propagating condition. It is the desired design objective to maximize the structural integrity of the ducts by minimizing the stresses imposed by thermal, hydraulic and support forces.^(1,2) Design characteristics required are:

- Creep Deformation
- Strain Induced by Cyclic Thermal Stresses

- Plastic Deformation
- Stress Concentrations
- Material Properties.

Structural characteristics must be correlated with geometry, loading, temperature and time.

Current Technology

Elastic analyses form the foundation for more complex structural analyses. Elastic analyses of the flow ducts are within the current technology. Stress concentrations at perforations or geometric transitions can be evaluated with photoelastic models. The overall elastic stress distribution and the thermal bowing displacements can be predicted with finite element codes such as SAMIS.

Creep analyses of the flow ducts will require extensions to the state-of-the-art. In structural metals, the steady-state creep rate $\dot{\epsilon}$ is related to the stress σ by the relation

$$\dot{\epsilon} = B\sigma^n$$

where B and n are temperature constants. Note: For even a small stress, this power law can generate appreciable strains.

Finnie ⁽³⁾ has shown that the steady-state bending creep rate of a tube is magnified greatly by pressurizing the tube. This magnification does not arise where the strains are strictly elastic--and linear superposition applies.

The breakdown of linear superposition for metallic creep problems complicates their solution. The linear viscoelasticity theories cannot be applied. Integral transforms of the governing field equations must be solved numerically.

Proposed Development Program

The development program proposed for the flow ducts must be correlated with a parallel analytical effort. The program

as presently planned consists of four phases:

- Phase I: Preliminary design studies to optimize the structural configuration.
- Phase II: Detailed elastic analyses of the hexagonal duct region at the active core zone.
- Phase III: Development of an analytical model for creep of irradiated SS.
- Phase IV: Proof tests of full size ducts.

The objective of Phase I is to optimize the location of the core supports to minimize thermal bowing effects. The objectives of Phase II are to determine the local elastic stresses and to evaluate their effects on structural stability. The objective of Phase III is to determine the cumulative creep deformations at critical locations. Phase IV will confirm the combined effects of the previous phases.

Test Plan and Experimental Methods

The Phase I work requires an experiment to verify the analysis. The natural frequencies and mode shapes of the flow ducts can be predicted using the SAMIS code. By comparing the analytical frequencies with the experimental, the accuracy of the analysis can be established. To measure frequencies and mode shapes, a prototypical flow duct with prototypical supports will be excited with an electromagnetic shaker.

The Phase II work will require full size photoelastic models of 1) the transition region from circular to hexagonal cross-sections; 2) the perforated region in the outlet plenum; 3) the slotted region at the inlet plenum; 4) a slice through the hexagonal section; 5) any preload rings, bearing pads or other lockup devices at the core center.

Axial (to simulate insertion and removal) forces, moments (to simulate thermal bowing loads) and/or uniform pressure will be applied.

For the Phase III work, creep tests of hexagonal tubes under combined thermal bowing loads and internal pressure are required. The test section may be any convenient size. The analytical creep methods will be applied to the test section; analytical results will be compared with measured creep data on the test section. Though the creep mechanism is different for unirradiated material than for irradiated, the analytical method will have the capability to include time-dependent irradiation changes in material properties. Proving the analytical methods does not require irradiation creep tests. Once proved, the analytical methods will be applied to the irradiated ducts when the appropriate irradiation creep data are available.

In Phase IV, full size ducts will be subjected to thermal and dynamic hydraulic loadings in conjunction with other related development and testing. These include core vibration (Problem D-1) for hydraulic water testing, integral structural analysis (Problem E-4) for thermally induced characteristics, configuration of reactor internal components (Problem F-2) for dimensional compatibility and sodium environment effects. Sodium flow testing will be required as part of the prestartup prooftesting. Also required is the confirmation of the capability of the fuel duct to prevent propagation of a single channel accident (including meltdown) to other fuel ducts. Specific methods for this experimentation are subject to further study.

Range of Parameters

For Phase I, a forced oscillation, variable frequency survey will be made of the prototypical flow duct at room temperature. For Phase II, the photoelastic models will be

loaded at appropriate elevated temperatures to lock-in the stress patterns; the 3-D models will be sliced to perform the 2-D analysis. Combinations of axial loads, pressure and moments corresponding to thermal bowing will be applied. For Phase III, temperatures of $750 + 40^{\circ}\text{F}$, $750 + 60^{\circ}\text{F}$, and $750 + 80^{\circ}\text{F}$ will be applied to one hexagonal force; temperatures of $750 - 40^{\circ}\text{F}$, $750 - 60^{\circ}\text{F}$, and $750 - 80^{\circ}\text{F}$ respectively will be applied to the opposite force. The uniform pressure will be 50, 75, 100, 125, and 150 psi. Thus, fifteen creep tests--five pressure and three temperature conditions--are required. These tests will be run for 6 months.

The prooftesting parameters are covered in the specific sections concerning vibration, mechanics, and mechanical fit. Sodium flow testing will be conducted at temperatures up to 1200°F . The duct will be subjected to thermal cycling and endurance testing.

Variables to be Measured

For the Phase I vibration tests, frequencies and amplitudes will be measured. For the Phase II photoelastic models, isoclinics and isochromatics will be measured. For the Phase III creep tests, lateral deflections will be measured.

For the Phase IV tests, coolant flow rates, temperatures, pressures; frequency and amplitudes of vibration, external and thermal structural forces, static and dynamic; and effects of fretting and galling will be measured or evaluated.

Information Applied to Design

The results of Phase I will show the optimum location of core supports (such as an upper tube sheet) and will establish the required section properties along the length of the flow ducts. The results of Phase II will establish the elastic stress distribution in the flow ducts. Whether

the flow ducts are in danger of failing by creep, fatigue or brittle fracture will become evident: the results of Phase II also may spawn materials studies other than creep. The results of Phase III are not directly applicable to design; however, these results will establish analytical methods which will be used to predict in-service creep behavior.

Phase IV testing will provide proof-tested duct design under operating conditions except nuclear.

Test Facilities

The vibration tests require a single prototypical duct with conventional dynamics laboratory equipment. The photoelastic tests require a conventional 3-D photoelasticity laboratory. The creep tests require burst safety chambers that can be heated up to 850 °F; quartz space heaters (or equivalent) are required to improve the temperature gradients. Pressure and temperature regulating devices are required.

The water and gas test facilities for the final development phases are described under discussions of vibration, integral structural analysis, and configuration of integral core components. The sodium testing verification will require a sodium loop, having adequate height to accommodate a full length duct, 500 gal/min flow, at 1200 °F coolant. No existing facility specifically meets the requirements to accommodate the duct.

Alternates

The work proposed in Phase I and Phase II is considered necessary to support an adequate structural design of the FTR fuel subassemblies, and in-core supporting hardware, including elastic analyses of core movements due to mechanical and thermal stresses. Only the results of Phase II will show whether Phase III and similar extensions are needed. Hence,

there are no alternatives to these initial two phases. Phase IV activities may be eliminated or decreased in scope. However, the application of the fuel duct in the FTR requires that the characteristics of the duct be identified by testing to assure structural integrity and absence of undesirable reactivity effects.

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INTEGRAL STRUCTURAL ANALYSIS (E-4)

Design Problem and Requirements

FFTF core components form a highly coupled nuclear, thermal and mechanical system. Fuel and flow ducts are mounted in an axial and radial support system to afford minimum strain and achieve high mechanical and hence, nuclear stability. Information required is based upon the need to characterize the core system adequately to assure predictable time temperature response of core geometry and to achieve a structurally sound core support system.

Core Geometry Effects

Geometric characterization of the mechanically coupled system subjected to thermal and hydraulic driving forces is required for:

- 1) Subassembly bowing
- 2) Subassembly axial expansion
- 3) Subassembly radial expansion
- 4) Support (grid) plate deflection
- 5) Interaction with split plane structure
- 6) Interaction with control elements
- 7) Interaction with reflectors
- 8) Vibration.

For a flow duct which is simply-supported at the bottom tube sheet, built-in at the reactor cover and free to expand radially elsewhere, preliminary analyses show a maximum radial inward deflection of 0.34 in. when exposed to a maximum temperature gradient of 60 °F across the tube. Deflections as small as 0.1 in. will induce measurable reactivity changes. Motion of coupled core components must be correlated with direction, time and phase of the driving forces.

Structural Integrity

Loading of the core by its support system in an optimum manner to assure minimum strain throughout the core life requires characterization of the following:

- 1) Thermal stresses (cyclic and steady state)
- 2) Mechanical or hydraulically induced stresses
- 3) Plastic deformation
- 4) Stress concentration
- 5) Vibration.

Strain induced in the coupled components must be correlated with location of support points and with thermal and hydraulic loadings to be anticipated.⁽⁴⁾

Analytical methods currently used for equally complex aerospace structures can be applied to the FTR core. Generally these methods are based on linear analyses so that the influence of one loading condition can be superimposed onto that of one or more others--thus, structural behavior under expected loading conditions can be predicted. Whether the thermal or mechanical deformations are linear in the FTR core will require experimental verification.

Proposed Development Program

Because of the complexity of the core system, the development effort must also be paralleled by an extensive analytical program. Experimental data will be correlated with mathematical models to provide the detailed thermal, mechanical and nuclear behavior characterization. The first phase of this program will use existing technology to the fullest extent practicable and will be highly idealized to gain an understanding of the system. A second phase effort will be aimed at greater refinement in which transient behavior can be studied with detail extended to individual core components.

Experimental Methods

Although the structural analytical methods to be used are founded on well-established theories that have been verified experimentally for aerospace structures,^(1,2,3) the application of these methods to reactor structures is unique. To obtain the true coupling between structural components and verify the applicability of the analytical models, a test model of the core structure is required.

The model does not have to be an exact replica of the FFTF core structure, but should be in a similar class. The model can be any convenient size and shall consist of at least a partial sector of tubes, tube sheets, vessel and cover. A finite element code will be applied to the core model for various thermal and mechanical loadings. Analytically predicted deformations will be compared with those measured on the model.

Because the core is a built-up structure, various joints and mating parts may not respond compatibly to temperature fluctuations. Local buckling, interferences, fits and other nonlinearities may invalidate the proposed linear analyses. Whether these nonlinearities are serious can be established by uniformly heating a full size sector of the core structure. Obviously this model for establishing heat-up linearity could double as the core model used in the above analytical correlation studies. One argument for the latter is timing.

Range of Parameters

The uniform temperature linearity check will require reproducing the absolute temperature ranges expected in the reactor. The thermal stresses are contingent on reproducing the prototypical temperature gradients. Thus, the range of parameters are as follows:

- 1) Uniform heating from 100 to 1000 °F
- 2) A steady state temperature distribution corresponding to inlet temperatures of 500 °F to outlet temperatures of 950 °F
- 3) Longitudinal temperature gradients to 350 °F from the bottom of the active core zone to the top
- 4) Radial temperature gradients from 0 to 60 °F across a typical flow duct in the active core zone
- 5) Hydraulic pressure corresponding to 50 to 125% of the designed bulk flow velocity.

End fixtures for the upper portion of the flow duct corresponding to pinned and clamped ends will be considered. Geometric parameters will be the configuration of the lockup mechanism and location of an intermediate tube sheet.

Variables to be Measured

The linearity of uniform temperature expansion will be established by deflection measurements on the tube sheets and ducts. Duct bowing loads will be established with strain gauges and profiles of translations and rotations at the tube sheets. The lateral deflections of the grid plate assembly can also be established with profilometers and strain gauges. Thermocouples are required to establish the exact temperature profiles.

Information to be Applied to the Design

Analytical results will be plotted on curves; these will show the radial and longitudinal deformations of the core as a function of temperature distribution and coolant flow rate. Stress concentrations at various locations will be delineated.

The flow ducts are the critical elements in the core structure. The integral core analysis will predict their geometric behavior with time. The grid plate assembly, the

reactor cover and any intermediate lockup devices are less critical but excessive deformations in them alter the behavior of the flow ducts. Once the accuracy of the analytical methods are verified, the analytical methods can be applied to the prototypical structure without further experiments.

Test Facility

The fluid elastic loadings on the prototypical ducts can be separated from the thermal effects. The two individual results can later be superimposed to indicate the overall behavior. The mass density of the coolant should have little influence on the deflections of the flow ducts. Hence, a heated-gas model is sufficient to establish thermal effects. The fluid elastic forces arise from two sources:

- 1) Cross flow induced vibrations of the flow ducts
- 2) Lateral hydraulic pressures on the grid plate assembly.

The magnitude of the vibration deflections will be established by experiments suggested in Core Vibration. The hydraulic pressure acting on the grid plate assembly can be simulated by either hydraulic jacks or dead loads in the gas heated mockup.

To summarize, a partial, full scale model is required to check linearity expansions with temperature changes. A non-prototypical structure model, subscale and partial, is the minimum required to confirm the analytical methods. The core model will be subjected to prototypical temperature gradients and to simulated hydraulic loadings.

Alternates to Performing the Proposed Analysis and Experiments

The alternates to the proposed plan are:

- 1) Eliminate experimental work
- 2) Eliminate integral core analysis.

Eliminate the Experimental Work

The experimental work can be eliminated. Sole reliance on analytical methods can be argued cogently on the basis of successful applications in the aerospace industry. Analysis is much less expensive than experiments. Furthermore, an experimental model is not necessarily more accurate than an analytical model.

But not all loadings can be predicted by analysis. Also, the proposed analytical methods have not been applied to integral core structures; we cannot say at this time that a conservative design based on analytical methods alone is adequate. As experience is gained in analyzing integral core structures, the need for proof tests will diminish. This alternative is less expensive but less conservative than the proposed work.

Eliminate the Integral Core Analysis

Conversely, the integral core analysis can be eliminated. The traditional approach--analyzing components in detail followed by proof tests in the components--can be taken. This approach is necessary but not sufficient.

This alternate does not assure compatibility of the individual analyses. Nor does it provide the flexibility and speed to assist preliminary design. Nor does it provide a means to perform system analyses of operational transients. Most important, it does not bridge the gap between non-nuclear and prototypical behavior.

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CORE ACCIDENT MECHANICS (E-5)

Design Problem and Requirements

Behavior of the reactor vessel and top cover under design basis accident (DBA) conditions has a vital effect on the design of blast shields and containment spaces within the containment system. Knowledge of this behavior is necessary for the proper design of the top cover and its hold-down mechanism or other energy absorbing devices to control the top cover. This knowledge is also necessary to predict pressures, temperatures, and missile hazards in surrounding containment spaces. The maximum energy equivalent of the explosive disassembly of the FFTF core is believed to be in the range of 750 to 1500 lb of TNT.⁽¹⁾ Continued effort is necessary in developing an analytical model that defines the nuclear explosion, its magnitude and characteristics. In addition, an analysis of the effects of the DBA energy release on the containment system is necessary to justify the safety of the reactor. Because of the many assumptions in such an analysis, experimental testing using FFTF configurations and conditions is proposed. Existing model test data can be correlated to the FFTF to some extent but these data do not fit all FFTF conditions.

Damage Mechanisms

The damage mechanisms may be attributed to two effects which are separated based on the time scale of events. The shock effects which occur within the first few milliseconds after the explosion and the gas bubble effects which occur later.

Shock Induced Effects

In the radial and downward direction, the shock wave provides a virtually instantaneous load for a short period of time on the containing structure and causes permanent

deformation of the structure. The shock wave propagating upward through the sodium coolant may be reinforced by reflected shocks from the vessel wall. When the shock reaches the gas gap between the coolant and vessel cover, an expansion wave will reflect back through the coolant. Spray and jets of sodium will impinge on the bottom of the vessel cover and induce a high impact pressure. This pressure will be a function of the shock deformation in the core region. Shock waves transmitted along subassembly tubes induce an additional load to the vessel cover.

Gas Bubble Induced Effects

After the passage of the initial shock wave and reflections, vaporized fuel, stainless steel cladding, coolant, and other core materials pressurize the reactor as another damaging mechanism. The gaseous bubble causes vessel deformation if the pressure is sufficiently high. Also it will accelerate a portion of the sodium against the vessel cover. The damage potential of the gas bubble is a function of the blast energy which remains after shock deformation and heat absorption by the coolant and the reactor internals. It may be postulated that a small sodium slug will also occur inside of some subassembly ducts.

The required design information to satisfy this problem is the characterization of the nuclear excursion as a pressure-time effect in the core and the characterization of the reactor vessel response to the excursion. The information must be in the form of analytic expressions confirmed and correlated to the extent possible by experiment.

Current Technology

A theoretical analysis of the loadings induced on the containing structure has been performed for spherical geometries. (12) The model consisted of an explosive core surrounded by

water with a small gas gap between the water and a rigid wall biological shield. Actual reactor designs are not one dimensional systems in either spherical or cartesian coordinates. The shock may initially tend to be spherical but multiple reflections from the vessel will modify this if they overtake the initial shock as it moves up toward the vessel cover. Physical uncertainties of mathematically solving the three dimensional problem with the appropriate shock-surface interactions soon lead to scale model tests.

Experience with explosives has demonstrated scale models may be used to simulate explosion effects in full size tests provided the proper scaling laws are followed.⁽¹³⁾ The most appropriate scaling laws are known as Hopkinson scaling where the linear dimensions of the model and reactor are related by a constant factor. Velocity and accelerations due to gravity do not scale but other quantities such as mass, specific impulse, stress, etc., are simply multiples of the scale factor.⁽³⁾ This is convenient for nuclear containment studies as long as the explosion characteristics may be properly simulated. The energy equivalent in TNT is often used to simulate the nuclear explosion although the nuclear explosion is known to generate smaller pressures and in general causes less deformations.⁽²⁾ TNT provides an extremely rapid conversion of energy and generates a severe shock wave and impulse loading. Conversely, a slow burning source of chemical energy such as a propellant produces relatively small deformations because there is no shock wave. Therefore, a charge of TNT simulates the maximum vessel deformation and maximum shock loading on the vessel cover from spray of coolant at the sodium-gas interface. Seemingly, a propellant should produce the largest vessel cover loadings from sodium hammer because of the relatively smaller vessel deformation volume where explosion gases may be accommodated. However,

experience at N.O.L.⁽⁴⁾ indicated the maximum plug jump for the model of Enrico Fermi reactor occurred for TNT type explosives if rigid wall vessels were used.

Extensive experimental work has been performed at N.O.L.^(3,4) on containing explosions with single-walled cylinders and measuring the height of a blast driven shield plug, called "plug jump." The containment potential of single walled cylinders has been measured and correlated into "containment laws" with reasonable material law assumptions. Based on these laws, the shock induced deformations for a single walled cylinder at least two diameters in length may be calculated, and allowable blast energies predicted with confidence. For composite vessel walls, however, analysis and theory is very limited. Reasonable methods are needed to predict shock induced deformations where the walls of concentric cylinders are separated by a layer of sodium. However shock loadings on the lower portion of the pressure vessel are not the primary design difficulty as will be explained later.

The shock induced loadings for an unsecured vessel cover have been experimentally measured;^(5,6) however, analytical models have not been developed for reactor geometries. The FFTF design uses a bolted down cover which provides more loading time for spray effects. This mechanism should be investigated experimentally and analytically.

The sodium slug effect has been investigated to a small degree.^(7,8) An analytical model has been proposed⁽⁷⁾ which explains the SL-1 accident but should also be applied to British results^(5,6,8) where possible for verification.

Information Gained from Current Technology

The vessel wall material will receive neutron fluence up to 10^{20} nvt. Conventional tensile testing⁽⁹⁾ has shown a

marked decrease in material ductility for neutron exposures in the range of 10^{20} . Reasonable estimates of allowable design ductility are 3 to 5%.⁽¹¹⁾ There is some degree of uncertainty regarding the degradation of material properties over the 20 year life of the vessel and its ability to contain a TNT type explosion. The reactor system must be designed to retain the sodium and radioactivity in a secondary containment. Note: The term primary vessel refers to the pressure retaining stainless steel structure around the reactor core while secondary containment refers to the concrete structure outside the primary vessel. If the primary vessel fails, the sodium slug loadings on the vessel cover would be greatly reduced because of loss of driving pressure.

There are at least two reasons why the vessel might not fail from ductility loss. Experience with explosive deformations have shown that brittle material may deform in a ductile manner under very rapid loading conditions. Thus, a ductile material model may not be unrealistic. There is not sufficient experimental data for blast deformations of irradiated stainless steels to determine the answer. Therefore, blast containment calculations must consider the effect of a ductile material under shock loadings also. Secondly, nuclear explosions do not have the shock deformation characteristics of TNT. The deformation seems to occur from the overpressure effect and is relatively small.

Applying the existing theories to a present conceptual design (SK.3.12574) is informative. The factor method of assigning percentages of blast energy to mechanical damage, loss to coolant and vessel internals, gas bubble, etc., has not been used since this does not reflect the specific reactor conditions such as coolant temperature and may not be realistic. The primary vessel and inner core vessel walls are assumed to be 2 in. thick AISI 304 annealed stainless steel.

The first set of considerations are for the containment potential of the primary vessel and inner core vessel separately for a TNT type explosion. At reactor temperatures of 550 °F, the uniaxial strain to rupture is approximately 40%.⁽⁹⁾ Based on tests at N.O.L.⁽³⁾ a safe allowable strain is one-third the strain to rupture. Applying the containment laws with these factors showed the outer pressure vessel would safely withstand 172 lb of TNT and the inner core vessel would withstand 112 lb. These figures are not additive for total vessel containment, however, because of inertia of the sodium annulus between walls. Also the laws were correlated for water as the medium between the explosive and wall so the containment potential will be reduced somewhat.⁽¹⁰⁾ These values of acceptable explosions indicate that an explosion of 750 to 1500 lb of TNT, which must be considered for the FFTF, will not be contained within the primary vessel even considering ductile materials.

The second set of considerations are for a propellant type of explosion which has no shock wave and causes deformation only through overpressuring the vessel. The propellant was assumed to have a quantity of gas generation equivalent to 750 lb of a TNT type explosion. The slug of sodium above the core is accelerated by an isothermal^(4,7) expansion of explosion gases and impacts the vessel cover. The validity of using an isothermal expansion was based on test data for model reactors with coolant and internal structure which indicated the gases rapidly approached the coolant temperature after blast. Immediately after propellant is burned, the gas bubble retains the original explosive volume of 7.9 ft³. The gas

rapidly approaches the coolant temperature as the vessel is deformed. The equilibrium position was located for the composite wall consisting of the inner core vessel and outer primary vessel with sodium in between. This is calculated by balancing the pressure of the explosion gases which decrease with vessel strain with the pressure containing ability of the vessel which increases from strain hardening. The resultant gas bubble pressure was 1450 psi with a strain of 6.1% in the inner core vessel. This provides an additional volume of 272 ft^3 for gases to be accommodated. The sodium slug is accelerated through a 2 ft gas cover space until it reaches a speed of 166 ft/sec and impacts the top cover. The remaining static pressure in the gas bubble is about 650 psi. Subsequent events depend on the type of impact invoked.

The most favorable containment situation is an inelastic collision where a large portion of the energy is dissipated during the collision process. This situation imparts a velocity to the cover of 82 ft/sec and the energy of the fluid-metal system is dissipated in the top ten feet of the vessel (single wall region) with a strain of 4%, which appears credible.

However, the validity of assuming an inelastic collision is not apparent and a more conservative approach would take the kinetic energy of the sodium slug, subtract the energy of deformation, with the remaining energy lifting the vessel head. The difficulty of this occurs in accounting for the energy of deformation. The deformation may be approximately considered as circumferential and axial strains. The axial strain energy was approximated by a plastic flow stress of 48,000 psi straining the upper 10 ft of the vessel 5%.

For zero circumferential strain, the remaining energy goes to potential energy which could lift the head 160 ft.

There will be circumferential strains; however, they are not easily calculated. The wall and head respond to a varying sodium pressure loading caused by their own motions.

The example energy balance has not accounted for a large portion of energy in the gas bubble which becomes available for accelerating the sodium and vessel cover if the vessel fractures. Most of this energy will not be transferred to the head, however, because the sodium must also be accelerated and the gas bubble will have displaced all the sodium in the vessel before it has fully expanded to atmospheric pressure. Therefore, it would not continue to transfer energy to the sodium. Extensive analytical work will be required to calculate these loadings which should be verified by tests.

Test Program

Test Method and Purpose

In addition to the work for developing an analytical model which defines the nuclear explosion in terms of maximum energy release, release rate, and gas bubble pressure build-up; an analytical model is needed which predicts the effects of the nuclear explosion on the reactor vessel. The analytical model must be based on the FFTF reactor vessel configuration which is difficult to correlate with existing test data. Results from this program will be used for verification of expected pressures, temperatures and missile hazards within adjacent containment spaces of the containment system.

To proceed in this direction, the following test sequence is recommended:

- 1) Development of an analytical model which defines the character of the nuclear explosion. This includes the maximum energy release, energy, release rate, and gas bubble pressure build-up.

- 2) Develop an analytical model for the deformation characteristics of a composite wall with a propellant type explosive. The approach will apply the techniques of Reference 3 (which examined thin walled cylinders) to composite walls. This will require an experimental program to explosively dilate about 50 cylindrical models with diameters ranging up to 20 in., different wall thicknesses, various types of composite walls, size of explosive charge, etc. This information will allow prediction of minimum wall deformations so that maximum sodium slug loadings can be calculated.
- 3) Develop an analytical model for the vessel head loading imposed by the sodium spray and the sodium slug. The flow equations for continuity, momentum, and energy plus the equation of state will be solved numerically along with the Rankine-Hugonot shock equations for one dimensional flow in both spherical and cartesian coordinates. This will provide lower and upper limiting values for the head loadings. The vessel cover motion will be considered in the boundary conditions as this will affect the impulse the cover receives. Experimental results will provide guidance for determining where reactor explosions fit between the upper and lower boundaries.
- 4) The information gained in recommendations 1, 2, and 3 will be applied to the subassembly ducts so their role in vessel cover loadings can be assessed. Mechanisms to be examined include shock loading transmitted to vessel cover through subassembly ducts, sodium slug phenomena within a single duct, and the effect of the subassembly ducts on the sodium slug motion.

- 5) Establish a test program to improve definition of cover loadings within boundaries determined in recommendation 3. A model which is about one-tenth as large as the FFTF will be used. The coolant will be sodium at 900 °F, the nominal outlet temperature for the FFTF unless otherwise specified. Four tests are proposed to obtain specific information:
- a) Test without vessel cover to obtain appearance times of sodium spray and sodium slug. This will aid interpretation of the data by indicating when the different loading mechanisms occur for the closed vessel tests.
 - b) Test with fixed vessel cover and reinforced lower wall near the core to obtain maximum sodium spray and slug effects. This will provide an upper limit for loadings on the vessel cover. The strain rates of the upper region of the vessel will be measured. Pressure cells will record sodium impact pressures on vessel wall and head.
 - c) Test with fixed vessel cover and allow about 6% deformation of lower core vessel which will result in more realistic vessel head loadings. This will be accomplished by a rigid wall vessel which surrounds the model and stops its wall motion after the required strain. This deformation absorbs some of the blast energy and helps accommodate the gas bubble so that less pressure is available to accelerate the sodium. Test data will be identical to test b).
 - d) Repeat test c) with 1200 °F sodium. This provides a different gas bubble temperature and reduced the sodium slug loading. The measured loadings will be used to check the fluid flow model established in recommendation 3.

Information from Test Results to be Applied to Design

When analyzing the damage from nuclear explosions, knowledge of the explosion energy release rate is important to the analyst in addition to the total energy release. This would aid comparing effects of nuclear explosions with calculations for TNT type and propellant explosions. Early analytical results showed a TNT explosion would not be contained within the primary vessel and, therefore, the sodium slug loadings on the vessel cover would be minimized. The 750 lb propellant type explosion releases will make rupture in the vessel wall near the core more probable. The maximum sodium slug loading on the vessel cover occurs with energy releases that do not quite cause rupture. The prediction of the sodium slug loading is limited by inadequate fluid impact models. The approach shown previously indicated the vessel cover could be torn free of the vessel and still possess sufficient energy to impact the roof of the hot cell which is inside the containment dome. Refined analytical techniques are necessary to determine more accurate impact energies so that proper energy absorbers can be designed.

Alternates

If the analytical studies described in recommendation 3 were completed, the vessel head loadings could be calculated on a conservative basis so that a safe design could be obtained. The procedure would be as follows:

- 1) Perform analytical studies of recommendation 3.
- 2) Check validity of analytical studies with existing experiments in References 3, 4, 5, 6, 7, 8, and 10.
- 3) Calculate maximum sodium spray and sodium slug loading on the vessel head.

The advantage of this approach is elimination of the testing program, but, this leaves a large number of uncertainties in the response of the vessel to the explosion.

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VESSEL, NOZZLE, AND PIPING MECHANICS (E-6)

Design Problem and Requirements

The design problems for the FFTF main reactor vessel, the removable core vessel, the sodium inlet and outlet nozzles, and the primary nuclear piping can be solved, for the most part, within the current technology. In general, the design procedures must satisfy the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1965.⁽¹⁾ Not all of the theoretical solutions for stresses in the two vessels, the nozzles and piping are in forms amenable to routine design.⁽²⁾ In addition, analytical results are not always conservative and often should be checked experimentally.⁽³⁾ Thus, combined analytical and experimental stress analysis methods are required.

The FFTF main reactor vessel is a large diameter to thickness ratio (thin walled) cylindrical shell with a tori-spherical head at the base.* The main reactor vessel is suspended by its support flange at the top. The removable core vessel, suspended inside the main reactor vessel, is basically a thin walled cylindrical shell with a conical section and shell at the top. Attached to the base of the core vessel is

* See FFTF project drawing SK-3-12579, Split Core Reactor Arrangement V.

the toroidal shaped high pressure plenum. Both shell configurations have several discontinuity regions (i.e., abrupt changes in geometry), particularly in the support flange areas.

The main reactor and removable core vessels must withstand primary and local bending stresses due to the pressure in the main vessel plus the suspended weights of the vessels, core, and coolant, and secondary stresses due to thermal gradients.⁽¹⁾ Each stress classification mentioned is redundant or indeterminate in the discontinuity regions. Also, in the discontinuity regions, peak (cyclic) stresses due to differential expansion may cause fatigue but not shell distortion. An accurate detailed stress analysis of these two vessels will be essential in their design program. Design conservatism obtained by merely increasing the wall thicknesses of the FFTF main reactor vessel, and the removable core vessel is not a preferred alternate. It is desirable to keep the walls of these vessels as thin as practical to limit the stresses due to the suspended weight of the vessels and to minimize the thermal stresses from gamma heating and nonuniform or cyclic sodium temperatures.

The main reactor vessel configuration shows eight 12 in. inlet nozzles and four 24 in. outlet nozzles for the sodium coolant. Stress concentration factors will exist at these openings for membrane and local stresses and for secondary thermal stresses. These stress concentration factors could approach a value of three of that on the edge of a small circular hole in the center of a rectangular plate subjected to axial tension. The nozzle walls could also be subject to cyclic fatigue stresses: thermal fatigue could arise from rapid changes in coolant temperature and mechanical fatigue could arise from cyclic external loads.

In summary the design procedure of the FFTF main reactor and core vessels and the coolant nozzles is: to determine the

stresses at all vessel discontinuities, to adequately compensate for stress concentrations at the nozzle openings, and to do a fatigue evaluation of the vessel shells and nozzles.

The following is a list of representative loads or loading mechanisms which will exist, to at least some extent, in the FFTF primary nuclear piping.⁽⁵⁾

- 1) Internal Pressure
- 2) Gross Thermal Expansion and Restraint, or Imposed Moments and Forces.
- 3) Steady State and Transient Local Thermal Stresses.
- 4) Impact
- 5) Hydraulic Hammer
- 6) Dead and Live Gravity Loads
- 7) Forced or Self-excited Vibration (Including earthquake loads).

Generally only a few of the above listed loads will be significant in any specific component of a power piping system. A design study of the primary nuclear piping system is necessary to completely understand all possible failure mechanisms. Adequate analysis methods for computing local stresses and strains in common piping components such as tees, wyes, and laterals must be found if the fatigue behavior of the piping system is to be predicted.

The design of any nuclear pressure vessel must satisfy the requirement of the ASME Boiler and Pressure Vessel Code, Section III. ASME, Section III contains rules for the design, fabrication, material specifications, and inspection of nuclear vessels.

The stresses in axisymmetric cylindrical, pressure vessels with abrupt changes in geometry (discontinuities) may be determined analytically by a discontinuity analysis,⁽¹⁾ or by experimental stress analysis.^(6,1)

The discontinuity analysis is initiated by separating the vessel into individual shell elements of a simple geometric shape for which the structural behavior is known. The pressure, mechanical, and thermal loads are applied to the shell elements along with a system of forces required to keep the static equilibrium of each element. These loads and forces cause individual element deformations which, in general, are not equal at adjoining edges. Therefore, a redundant shear and moment must exist at each discontinuity to have compatibility of deformations. Equations for the redundant loads are written in terms of each element's influence coefficients, i.e., edge deformations due to a unit shear and a unit moment. Several computer programs are adaptable to the discontinuity analysis of an axisymmetric shell. After the equations are solved for the redundant loads, the stresses at the edge of each element or discontinuity are calculated by conventional means. The procedure and information required for a discontinuity analysis are given in article I-7 of ASME, Section III. The rules for the stress classification and stress allowables in nuclear pressure vessels are given in paragraph N-414 of ASME, Section III.

The torispherical head at the base of the main reactor vessel is relatively flat and the vessel shell is relatively thin. The head to shell junction (knuckle) can be a region of high stress (both tension and compression) with the result that the material in this region may yield. A limit analysis will be needed for this junction because, for such a head and shell configuration, elastic stability theories could indicate buckling even before yielding,⁽²⁾ i.e., the stability of the head must be insured.

The coolant inlet and outlet nozzles penetrating the FFTF main reactor vessel must be adequately reinforced to compensate for their increased membrane and local stresses due to pressure, and their increased secondary thermal stresses (stress concentration factors). ASME, Section III satisfies this requirement with the code rules for design and fabrication of nozzles given in paragraphs N-452 to N-456.⁽⁷⁾ The design technique proposed in the code for the reinforcement of the penetrations is the area replacement method. The amount of reinforcement placed around openings by this method is considered reasonably conservative. The reinforcement areas required, by ASME, Section III, for the conceptual outlet and inlet nozzle configurations are, 7.5 in.² and 5.5 in.², respectively. The effective reinforcement areas available for the outlet and inlet openings are 47.8 in.² and 41.0 in.², respectively. Therefore the ASME, Section III nozzle reinforcement requirement is satisfied by the conceptual design.

The critical or governing stresses at the discontinuity regions and nozzle penetrations may be found by experimental stress analysis. This is accepted by the ASME, Section III nuclear vessel design code if the code's general requirements given in Article I-10 are met.

An experimental stress analysis of a pressure vessel may be necessary if the shell has a complex configuration, i.e., the manipulative difficulties of the discontinuity analysis may be prohibitive. For example, the upper region of the FFTF main reactor vessel is difficult to separate into simple shell elements. These simple shell elements can only be approximated at some of the discontinuity areas, and

the number of elements will become large. The number of redundants and equations necessary for their solution, associated with numerous shell elements, is also large and tends to complicate the discontinuity analysis greatly. The accuracy also lessens because of the approximations.

The stress distributions around the nozzle openings in a shell may be determined with good accuracy by experimental methods.^(6,8) Such experiments could probably best show the adequacy of the ASME, Section III reinforcement requirements for the nozzles penetrating the FFTF main reactor vessel.

Two types of tests for determining pressure vessel and nozzle stresses could be used here. They are strain measurement tests and photoelastic tests. Strain gage data may be taken from the actual vessel or from a model vessel. Both two-dimensional or three-dimensional photoelastic techniques may be used.

Preliminary design calculations show the maximum alternating stress (S_a) in FFTF main reactor vessel, for a radial thermal gradient of $\Delta T = 300$ °F, to be 16,500 psi.⁽⁹⁾ This thermal gradient and potential fatigue stress represents a fast scram from full power with full pump flow condition or the most severe fatigue condition possible. The numbers of FFTF fast scrams or fatigue cycles may be as high as 1000.

The present method of evaluating fatigue for a nuclear pressure vessel is discussed in paragraph N-415.2 of ASME, Section III and the code's design fatigue curves based on strain-controlled fatigue tests.^(10,11) The fatigue curve applicable to the FFTF main reactor vessel is Code Case 13-31 (1200 °F). The assumptions necessary to make the present code's fatigue rules and curves apply to the FFTF design are: the irradiation of the vessel does not change the vessel's

material properties, and the rapid cycle fatigue test data, from which the design curves were compiled, are representative of the vessel's actual fatigue cycling (potential damage at 1200 °F would actually result from a combination of fatigue and creep phenomenon).

Preliminary fatigue stresses (S_a) for a 200 °F step-change at the nozzles were found to be 22,900 psi and 24,075 psi maximum for the outlet and inlet nozzles respectively.⁽⁹⁾ The ASME, Section III fatigue curves, using these preliminary data, indicate that fatigue failure in the nozzles is not imminent until 40,000 cycles ($S_a = 24,075$ psi controls). Further calculations show that the Code, stress limit ($3 S_m$) will not be exceeded in either nozzle region for a 200 °F step-change. (S_m = the allowable stress for the vessel material, ASTM A-240 TP 304 SS.)

The design of the FFTF primary nuclear piping system will also rely on the ASME, Section III code. The code requires a detailed stress analysis and a fatigue evaluation of the system.⁽⁵⁾ The possible load sources for the piping are listed in Part I of this code. The code also includes rules to guard against brittle fracture. The stress analysis cannot be handled analytically for many of the piping systems unsymmetrical shell-like discontinuities such as tees, wyes, laterals, etc.⁽⁵⁾ The code allows for experimental stress analysis methods to be used, however. Photoelastic analysis or correlation with existing experimental data is recommended.

Proposed Experimental Programs and Expected Results

The following photoelastic model tests are recommended for the FFTF main reactor vessel and the removable core vessel design program.

Two-Dimensional Models

- 1) Upper Region of Main Reactor Vessel Shell
- 2) Upper Region of Removable Core Vessel
- 3) Inlet Nozzle
- 4) Outlet Nozzle.

Three-Dimensional Models

- 1) Inlet Nozzle Region
- 2) Outlet Nozzle Region
- 3) Torispherical Head.*

The results of these photoelastic experiments can be combined with analytical stress analysis methods to readily distinguish between primary, secondary, and local stresses. This capability is important because each stress classification has an applicable code limit.

The proposed two-dimensional models would be used to give the primary and local bending stresses due to suspended vessel, core, and coolant or mechanical loads at the gross discontinuities and nozzle regions. The stresses due to pressure and thermal gradients, determined analytically, could be superimposed on the model stresses to give the overall stress picture at these regions.

Three-dimensional models 1 and 2 could be used to accurately predict local peak and fatigue stresses in the nozzle regions due to stress-changes in temperature or from other external loads transmitted to the nozzles from the piping.^(6.8) These two photoelastic experiments would, in

* Suitable experimental data may be available in the literature.

effect, show the adequacy of the ASME, Section III reinforcement requirement for the nozzles penetrating the FFTF reactor vessel.

Three-dimensional model 3, would give the elastic stress distribution in the head to shell junction (knuckle). With this distribution known, the limit analysis of the head can be readily performed.

As previously noted, some photoelastic stress experiments, etc., are anticipated for the discontinuities of the primary nuclear piping system. The details of this portion of the experimental program can be readily worked out when the piping system configuration becomes fixed.

An important reactor vessel design problem has been identified for which a development program has not yet been defined. In the design of the FTR vessel, strong effort is directed toward the use of design features and design concepts which are amenable to theoretical analysis with the least need for experimental correlation. In effect, this design approach should permit development of analytical models of the reactor structure which may be used to predict steady state and transient response under virtually all modes of operation. To the extent that valid analytical structural models can be developed, the performance of the reactor system may be predicted at any time during the life of the reactor, even after the occurrence of severe (not destructive) accidents.

Assuming a nearly ideal structure characterized by a practicable analytical model, there is a high incentive for correlation of the physical system response to the analytical model. Expected correlative data to be obtained are structural strains and displacements as a function of imposed mechanical or thermal loadings and time. In its present

status, the CCTL at ANL has been structurally characterized by analytical models with correlative instrumentation for strain and displacement affixed to the loop piping and components. It is expected that operation of the CCTL will yield valuable information concerning 1) our ability to establish valid analytical models which are confirmed and improved by experimental data acquired from the physical system, and 2) the state-of-art capability of strain sensors to withstand long time environmental conditions at or near 1200 °F. The results of the CCTL structural program will assist in definition of the FTR vessel structural response development program.

Design Alternates

Nozzles Stresses

An analytical solution for the state of stress in a cylindrical shell with a circular hole has been proposed by A. C. Eringen.⁽¹²⁾ The solution is applicable if the opening has parameters within the following limits: $d_m/D_m \leq 0.35$; $D_m/T \geq 20$ and $0.155 \leq (d_m/D_m)(D_m/T)^{1/2} \leq 3.90$ (see end of section for notation). However, the solution does not account for the reinforcing effect of the nozzle, i.e., ($t/T = 0$).

The stress concentration factor for a single opening reinforced by a nozzle can be calculated by hypothetically replacing the cylindrical shell with a plate.⁽¹³⁾ This solution becomes less accurate as the vessel radius decreases.

The major value of either of the two above solutions for nozzle stresses is to check the safety of the ASME, Section III nozzle design. Most likely these methods could not reduce the code reinforcement requirements.

The stress components in the nozzle walls at their junction to the reactor vessel, due to possible external thrust, moments, torsion, and shear, can be calculated with the aid of a set of nondimensional curves plotted in terms of nozzle to shell parameters. These nondimensional curves are the results of analytical work by P. P. Bijlaard.⁽¹⁴⁻¹⁶⁾ Reference 14 gives equations, also derived by Bijlaard, for the various stress components in terms of the external loads and appropriate values from the nondimensional curves (Table 5, Reference 14). The external loads would be those derived from temperature step-changes, gradients, etc.

The accuracy of the above solution is highly dependent upon the selection of certain stress concentration factors for membrane stresses and bending stresses.⁽¹⁴⁾ These stress concentration factors are selected from curves plotted in terms of (r_2/T) , where (r_2) = the outside filled radius and (T) = the vessel wall thickness. These curves conform to empirical formulas that were derived from experimental data. Also, the nondimensional curves, used in the method, have been adjusted to suit certain experimental data,⁽¹⁴⁾ i.e., the solution is configuration dependent. This indicates that for unique nozzle configurations subjected to external loads, the stresses should be found experimentally.

Fatigue Evaluation

As pointed out in a previous section, some assumptions that may not be truly realistic have to be made to do a fatigue design of a nuclear vessel operating at 1100-1200 °F, with the present ASME, Section III code. The recommended alternative is that the problem of high fatigue or cyclic stresses from thermal gradients be "designed around." Thermal fatigue stresses in the vessel and nozzles can be reduced to a minimum by use of thermal liners and by

immediately reducing the coolant flow from the pumps during a "fast scram." Also most emergency shutdowns probably can be handled with an "intermediate scram" which would not produce an extreme fatigue condition.

A postoperative inspection capability should be designed into the FFTF reactor vessel and nozzles to allow the early detection of any fatigue cracking that may occur. The non-destructive test methods available for early detection of fatigue cracking are the eddy current and the ultrasonic tests. Such tests would be performed routinely during reactor shutdown.

If necessary the fatigue testing of critical vessel components could be included in the previously proposed experimental stress analysis program. In many cases such tests demonstrate higher allowable peak stresses than the fatigue design curves.⁽¹⁾

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Notation

The following letter symbols, used in Sections E-1 through E-6, conform essentially to Standard Letter Symbols adopted by the Pressure Vessel Research Committee, PVRC.

- D_o = Outside diameter of pressure vessel
- D_i = Inside diameter of pressure vessel
- D_m = Mean diameter of pressure vessel
- d_o = Outside diameter of nozzle
- d_i = Inside diameter of nozzle
- d_m = Mean diameter of nozzle
- R_o = Outside radius of pressure vessel
- R_i = Inside radius of pressure vessel
- R_m = Mean radius of pressure vessel
- r_o = Outside radius of nozzle
- r_i = Inside radius of nozzle
- r_m = Mean radius of nozzle
- T = Wall thickness of vessel
- t = Wall thickness of nozzle
- r_3 = Inside fillet radius
- r_2 = Outside fillet radius
- p = Internal design pressure
- S_m = Stress allowable for pressure vessel material
- S_a = Alternating or Fatigue Stress.

DRIVER FUEL TOP FACE HARDWARE (E-7)Design Problem and Requirements

The major problem related to top face hardware is the selection of sealing methods for prevention of cover gas and sodium vapor leakage. Current design concepts indicate that there may be several joints required to provide the necessary capability for removal and replacement of driver fuel assemblies

and the associated instrument jumper connectors. During removal of fuel, joints will very likely be exposed to liquid sodium. Mechanical seals are a logical consideration, but the limited experience with them in a sodium vapor environment restricts their use where a high degree of reliability is required. Grayloc and Conoseals have been used in applications that are accessible for manual removal of seals, clean-up of mating, surfaces, and manual fit up and replacement of the new assembly. An insufficient number have been employed to make a realistic evaluation of their statistical reliability under conditions to be expected in the FFTF where removal and replacement will be by remote means. A further complication is the lack of space available for seal locking devices. Galling of threads in such devices is certain to be a serious problem. Conventional locking device approaches are ruled out for the most part because of the lack of space and the galling problem.

Since the primary purpose of the FFTF project is to provide reliable testing facilities, it follows that advance testing of hardware is justified.

Experimental Program

Seal testing will involve proof-of-principle tests on mechanical seals prior to design of a hardware prototype, followed by statistical reliability testing of the prototype and design verification testing of the final design prior to installation of final hardware in FFTF. The first two tests are primarily directed at the seals while the third test evaluates a final design of the complete hardware. It should be anticipated that available mechanical seals and locking devices may prove to be unsuitable. For this reason, it is important that a backup sealing method be investigated concurrent with the above program.

The sealing method or methods that survive the proof-of-principle testing would be incorporated in a hardware prototype design for statistical reliability testing. This would involve repeated making and breaking of seals under conditions expected in actual service. Environmental conditions between seal replacement would be essentially the same as those employed during feasibility testing. The number of tests performed would be based on the level of statistical reliability and confidence desired. For example, if the joint employing the seal were connected and disconnected 100 times in succession without producing a leak under the condition listed above, there is 90% confidence that the statistical reliability would be 97.7% or higher.

The sealing methods which survive the above test would then be incorporated in a final design of top face hardware for design verification testing. These tests would involve the same environment as the feasibility tests, but would also involve dimensional fit and distortion, fretting and galling and interface with the fuel handling machine.

The tests needed to establish feasibility will involve subjecting the sealing methods to static and dynamic pressure, thermal expansion, and external forces as well as vibration in an inert gas and sodium vapor environment.

Test Facilities

The facilities required for the feasibility tests should allow testing full scale seals and joints in nonflowing He, H₂O, and Na vapor environments. Water testing may be a means of rejecting seals in advance of sodium vapor tests. The helium test anticipates that certain seals may not be suitable even in a gas environment at FFTF conditions. The facilities required for final testing would involve a full scale system in terms of nonflowing sodium capability.

Alternates

Since evaluation of alternates in this case is largely a matter of pointing out the consequences of not performing certain tests, it is important for purposes of this discussion that the objectives of the various levels or types of tests are defined. In general, testing of components falls into five general categories:

- 1) Proof-of-principle tests or tests which demonstrate the feasibility and/or characteristics of a particular approach
- 2) Design verification tests which demonstrate that certain design characteristics have been provided
- 3) Statistical reliability and/or endurance tests which demonstrate that functional and materials requirements will be met, throughout the predicted life of the component
- 4) Materials testing
- 5) System operability tests.

System testing may involve 1-3 above, but as a rule, is intended to demonstrate compatibility and operability of individual components when integrated into a complete system. Tests 2 and 3 are usually the basis for acceptance or rejection of individual components developed, while test 5 is the basis for integrated systems.

In the case of top face hardware development, alternates could involve any one, a combination of several, all or none of the tests. The plan described above involves tests 1, 2, and 3. Tests 4 and 5 are part of a separate program. If proof-of-principle tests (1) on several sealing and fastening methods are not carried out in advance of the first prototype hardware design, it is possible that a complete prototype would eventually have to

be built and tested for each type of seal available (and there are many). This is true because the hardware design is dictated largely by the particular seal employed, and if the seal selected fails, the complete prototype also fails. Proof-of-principle testing of seals and sealing methods per se does not require a complete hardware prototype, and as a consequence, could eliminate considerable unnecessary expense.

Statistical reliability testing (3) of seals and sealing methods could be eliminated with the possible consequence that top face hardware would leak after several loop changeouts and a new design would have to be developed and installed after start up, including appurtenances permanently attached to the reactor cover. This would be costly and could result in considerable delay in reactor start up.

Elimination of the design verification tests (2) in advance of start up could result in the same consequences mentioned above, but for different reasons. The tests above have dealt with seals and sealing methods only and have not considered, for example, routing of electrical jumpers and interface with other top face hardware and handling systems, or any of the many other functional and materials requirements specified for top face hardware. Failure to meet any one of these requirements could put in jeopardy achievement of the major goals of the FFTF project.

F. MECHANICAL FIT TESTING

The assembly, disassembly, and reassembly of the reactor system cover and internal components are being subjected to detailed analytical, graphic, and model evaluations. Although these efforts yield no evidence of unsurmountable problems, it is strongly indicated that supporting test work to assure the adequacy of fit up of the many components, both at the reactor cover and in the vessel, are essential.

The problems associated with the normal fit up result from the pure mechanical considerations of the compactness of the core and close tolerance requirements, as well as the auxiliary effects of chemical, metallurgical, and hydraulic behavior on the components.

The area of mechanical fit is directed toward providing development support of design and design verification.

REACTOR COVER ARRANGEMENT (F-1)

Test Requirements and Objectives

With the exception of the primary sodium coolant lines, all penetrations into the reactor vessel pass through the vessel cover. The principal penetrations consist of more than 150 nozzles which accommodate the driver fuel shrouds, safety rods, shim rods, regulating rods, closed loop tubes, reflector rods, oscillating rods, and neutron source. These nozzles are located within a circle of approximately seven-foot radius and are arranged on about a ten-inch triangular pitch pattern resulting in a compact array.

The space between and around the nozzles is used for routing many auxiliary components. These components include: gas coolant piping, closed loop piping, electrical power cables or conduit, instrumentation jumpers, sample lines, purge lines,

and leak monitor lines. All of the components in this crowded area must be carefully arranged to avoid interferences and to ensure practicality of assembly, compatibility with the fuel handling machine, and relative ease of maintenance.

During the reactor design, a good deal of work will be devoted to the layout of the cover area; however, many disciplines are involved and dimensional errors, interferences, and awkward configurations can come about to cause interference problems. Background information for design is needed with respect to required clearances, shape and size limitations, handling problems, assembly sequences, accident fault analyses, mechanical loads developed by handling equipment or component interactions, shielding limitations, and thermal problems.

The information required by design is the parametric data listed in the preceding problem description. The data are required during the course of design to avoid technical or schedule failures that may jeopardize the project. The penalties for failure to provide an acceptable arrangement are summarized as follows:

Technological Failure

Tradeoff

Awkward cover configuration.

Increased assembly time, completion delayed, increased maintenance.

Failure to locate interferences or dimensional errors.

Assembly delays, component rework, field changes, inability to use ordered parts, extra cost.

Failure to establish assembly sequence.

Dismantling of adjacent components to install other new components, delays, increased construction costs.

Poor operational clearance.

Limited fuel handling capability, redesigned FHM tools, lengthened startup program, assembly rework.

Failure to confirm assembly drawing information.

Delays in assembly.

Failure to define locations requiring special tools.

Tool development delayed, tools not available for reactor assembly.

Schedule Failure

Tradeoff

Failure to define shielding problems.

Operational limitations, streaming problems, excess shield loads, delays and extra costs resulting from rework.

Design development confirmation and refinement not completed before ordering components.

Changes to contracts, delays and extra cost.

Test Program

Test Method and Purpose

The "test" will consist of assembling the full size components of a reactor cover complex. In order to cover all combinations and relationships of components, the cover mockup will include all of the different kinds of shroud and closed loop nozzles and all types of auxiliary parts such as instrumentation leads, electrical runs, coolant piping, remote connectors, etc. The test plan will be conducted to:

- 1) Evaluate design for practicality of assembly, proper fit, accessibility, and ease of maintenance.
- 2) Define areas which will require special tools for assembly, disassembly, and maintenance.

- 3) Check layout for interference, dimensional errors, and inefficient or awkward arrangement.
- 4) Refine and optimize the arrangement of components and the routing of connectors, instrumentation leads, electrical runs, etc.
- 5) Confirm operational clearance and compatibility of configuration with the fuel handling machine.
- 6) Determine preferred or mandatory assembly sequence.
- 7) Confirm adequacy of assembly information given on the design drawings.

Test Facilities

This test will require a full size mockup of the reactor cover and associated components. The cover can be simulated by a mild steel flanged and dished head. The nozzles and auxiliary components can be made from mild steel or other common easily worked material. However, the related application of this mockup to sodium environment evaluation of top face hardware suggests that the cover be designed and fabricated for sodium service.

Alternates for Performing Experiments

Alternates

- 1) Use smaller scale model
- 2) Rely on drawing overlays on composite drawings
- 3) Rely completely on design drawings.

Evaluation of Alternates

Use Smaller Scale Model. This test could be carried out using a reduced scale mockup. This approach would probably be a little less expensive because it would require less material; however, since most of the cost will be for labor, no great savings would result.

If a smaller scale model is used, the test will be less meaningful. Areas having close fitting components may appear as points of interference when using the reduced scale, whereas in full size the clearance is adequate. Situations requiring special tools would not be as clearly evident; also, testing of tools would not be as realistic. In general, dimensional accuracy and tolerance testing loses significance when performed at less than full size.

Rely on Drawing Overlay or Composite Drawings. Dimensional and configurational checks could be attempted by making drawing overlays or composite drawings of all cover components. However, when an arrangement is complex, composite drawings become cluttered and are not a very reliable check of a dense three dimensional array of components.

Rely Completely on Design Drawings. Dimensional and configurational checks could be eliminated, placing full reliance on the accuracy and completeness of the design drawings. However, interferences, errors and omissions do occur even in well engineered designs and these would almost certainly lead to difficulties, delays, and extra expense during construction and possibly to reduced operational capabilities after startup.

CONFIGURATION OF REACTOR INTERNAL COMPONENTS (F-2)

Test Requirements and Objectives

The reactor fuel ducts and closed loop tubes are assembled in a compact array fitting closely together at the bottom of the active zone. During assembly, the subassembly ducts and tubes are aligned by the reactor cover guide tubes and are lowered and inserted into the inlet plenum tube sheet. Since all ducts are not vertical, they will tend to deflect as they are lowered. This deflection may create an alignment and insertion problem at the inlet plenum tube sheet.

Accordingly, the sequence of assembly may prove to be very important. If subassembly duct or tube deflection is excessive, the core may have to be assembled by first placing the outer reflector tubes, then proceeding toward the center using previously placed subassemblies as guides. In addition, after the assembly is complete, it may be found that only one subassembly in a cluster can be removed and inserted at a time in order to use surrounding ducts as a guide.

The alignment and compactness of the subassembly arrays are affected by thermal expansion of the vessel and ducts. Four different temperature ranges require consideration: room temperature, refueling temperature, operating temperature, and temperature differentials across the duct. The reactor will be initially assembled at room temperature. Fuel handling operations take place at a temperature in the vicinity of 400 °F. Reactor operating temperatures will be 500 °F to approximately 1200 °F with temperature differentials up to 100 °F from one side of the duct to the other.

The dimensional changes resulting from the above temperature changes will be relatively large (in the neighborhood of three inches vertically, 3/16 in. horizontally, and up to 1.0 in. bow). The reactor cover guides and inlet plenum tube sheet holes must be aligned well enough at the refueling temperature to allow subassembly removal and insertion without damage or excessive time, and the preload device must maintain the desired core configuration at operating temperatures.

The required design information is 1) the definition of acceptable clearances between core components under the full range of operating temperatures and 2) confirmation of core assembly design under full range of operating temperatures.

The penalties for failure to resolve adequately the problems are summarized as follows:

Technological Failure

Tradeoff

Failure to define dimensions and tolerances required for proper fit.

Fuel handling problems, damage, high maintenance.

Failure to eliminate dimensional errors and interferences.

Delays in assembly, rework, extra cost.

Failure to determine assembly sequence.

Delays in assembly

Failure to confirm or refine lead-in device design.

Fuel handling problems.

Failure to establish feasibility of random assembly.

Potential fuel handling problem, jamming, high maintenance

Failure to eliminate areas of potential damage.

High maintenance, extra cost

Schedule Failure

Design confirmation and refinement not complete before ordering components.

Changes to contract, delays and extra cost

Test Program

The test program will be directed to:

- 1) Evaluate design for practicality of assembly, proper fit, and alignment.
- 2) Check arrangement for dimensional errors, interferences and suitability of tolerances.
- 3) Determine preferred or mandatory assembly sequence.
- 4) Confirm adequacy of the size, configuration, location, and orientation of the lead-in device at the inlet plenum tube sheet.

- 5) Determine the practicality or limitations of random sub-assembly removal and insertion.
- 6) Define areas of potential damage to the reactor internals while raising and lowering subassemblies.
- 7) Verify the compactness of fit in the active core zone.
- 8) Confirm the effectiveness and adequacy of the preload and hold-down mechanism.
- 9) Develop maintenance procedures for repair of failed components.

The program will be divided into three phases:

Phase I

A preliminary test will be run using seven full sized subassembly ducts. The test assembly will be a cluster consisting of a simulated control rod surrounded by six fuel ducts arranged to simulate seven adjacent ducts in the reactor. These ducts will be assembled in fixtures representing seven positions in the reactor cover and inlet plenum tube sheets.

The purpose of this initial experiment will be to evaluate quickly the feasibility of the basic design concept and gain experience and information in the early stage of design. Gross errors or formidable obstacles will be exposed and can be eliminated before advancing to the more comprehensive second and third phases of testing.

Phase II

This test will consist of assembling the full size components of a sector of the reactor core. In order to duplicate the relationships of components, the representative sector will include all the different kinds of subassembly ducts. About twenty ducts will be required.

The core sector will be assembled and disassembled initially in various sequences at room temperature to simulate the initial fueling. A simulated reactor cover, inlet plenum tube sheets and subassembly duct will be used. Visual and measured data will be obtained to determine the complex interaction of the core behavior.

Gross evaluations of the core behavior to increasing temperatures of start up refueling and operation will be made by simulating the axial thermal expansion of the ducts by increasing the distance from the cover to the tube sheet. Elongation of the ducts can be achieved by adding to their lengths. Changing spacers at the duct penetrations through the cover and tube sheet will simulate radial expansion. For each simulated condition, the mechanical fit effects will be determined. At refueling temperature simulation, the selected ducts for fuel, control elements, reflectors, etc. will be removed and reinserted to evaluate the refueling sequence. Withdrawal and insertion forces required for the fuel assemblies will be determined.

The information from these tests will be integrated with core heat-up tests needed for integral structural analysis.

Phase III

Design verification will be completed by evaluating fit parameters in a static sodium pot. The seven-duct assembly, as described for Phase I, will be used. The evaluations will be concerned with the effect that sodium may have on insertion and removal of the ducts. The sodium will be heated to refueling temperature, the duct inserted, sodium temperature cycled between refueling and reactor operating temperature, the ducts removed and reinserted. The forces necessary to remove and insert ducts will be measured and the surfaces of the ducts will be examined for wear or severe sliding contact.

The procedure employed for the evaluation will be to insert and remove each element of the seven duct units, and to observe and measure the degree of difficulties associated with the mechanical fit.

Test Facilities

The required facility for Phase I is a test assembly arranged to simulate seven adjacent ducts in the reactor. These ducts will be supported by a simulated reactor cover section and its hardware, tube sheet(s) and a restraining ring near the top of the core.

The assembly will be made up of full size subassemblies. The assembly will be located in unheated air, requiring only mechanical supporting auxiliaries.

The required facility for Phase II testing will be a simulated reactor cover, inlet plenum tube sheets and subassembly ducts. Only about a quarter of the core assembly will be simulated. The cover will be situated above the tube sheet to represent the actual reactor arrangement. No vessel configuration is required but capability for providing temperature gradients will exist. This open arrangement, without a vessel, will allow direct observation of tube insertion and easy visual inspection of the core area in some of the succeeding tests.

Phase III tests require sodium environment in a pot 2 to 3 ft diam by about 35 ft long (reactor vessel height). Internally, the pot will contain a simulated core support tube sheet, and reactor cover guide tubes. The "cover" will be fitted with nozzles and other simulated hardware. The pot will be heated to provide up to 1200 °F temperature sodium. A small circulation will be required to provide purification and to provide a means for slow temperature cycling in the vessel.

Alternates to Performing Experiments

Alternates

- Use an enclosed mockup vessel for dry tests (for Phases I and II)
- Use a reduced scale mockup
- Use only seven subassembly ducts for performing tests
- Eliminate testing.

Evaluation of Alternates

Use an Enclosed Mockup Vessel for Dry Tests. The test could be run by assembling reactor components into a vessel mockup. The thermal expansion effects could then be achieved by hot gas heating of the vessel. This is a good final check of the reactor configuration; however, this arrangement would not be well suited for early development work because of the complexities of instrumentation and test setup. Vessel system testing for integral structural analysis is covered elsewhere in this report.

Use a Reduced Scale Mockup. The test could be carried out using a reduced scale mockup. However, here again, as in the reactor cover mockup, dimensional accuracy and tolerance checking loses significance when performed at less than full size.

Use Only Seven Ducts for Performing Tests. Testing could be terminated after the proof-of-principle experiments, using seven ducts (Phase I). This test is of limited significance, however, because it does not explore the interfaces of all the different kinds of subassemblies.

Eliminate Testing. Testing could be eliminated. The reactor internals could be fabricated from the design drawings without mockup verifications.

If, however, errors, omissions, interferences, improper tolerances, poor fits, etc., are present, shop or field changes would be costly and wasteful and long delays in assembly would result. It cannot be assumed that the components will fit correctly because they appear to do so in the drawings. It must be known that they will fit and function as intended.

SELF-WELDING OF REACTOR COMPONENTS (F-3)

Design Problems and Requirements

Various components of the reactor must be capable of separation after long periods of stationary contact in liquid sodium. Because of the "cleansing"⁽¹⁾ action of liquid sodium, these parts are subject to possible self-welding.

Self-welding potential is known to be reduced by minimization of the true contact area and minimization of the adhesive bond strength in the true contact area. Surface hardness is the property most effective in decreasing self-welding potential.

Areas within the reactor that are subject to potential self-welding are:

- 1) Holddown mechanisms at the inlet plenum tube sheet
- 2) Preload mechanism above the active core zone
- 3) Shroud assemblies at the inlet plenum tube sheet
- 4) Safety and control rods.

The conditions leading to self-welding, which could lead to lockup of holddown and preload mechanism, inability to remove shrouds without damage to tube sheet, seizure of safety and control rods, must be identified and eliminated.

Program

Tests to establish the adequacy of design to prevent self-welding will be needed. The principal variables involved are:

- 1) Surface hardness
- 2) Surface finish
- 3) Sodium temperature
- 4) Length of time in contact
- 5) Contact force
- 6) Sodium purity.

Representative contacting components of the reactor internals will be tested in a liquid sodium environment which simulates reactor coolant temperature conditions. The components will be assembled and held in static contact under loads which correspond to the forces present within the reactor during operation.

These assemblies will be immersed in liquid sodium and the force or torque required to separate the components will be measured at various time intervals. The time interval between measurements will correspond to the most likely contact period in actual reactor operation.

Initially, component type testing will be performed to obtain early indication of the self-welding potential for the general geometric and fluid environments of the FTR. Component tests must be viewed as lead in screening tests, since the long duct and core assembly reaction forces to heat up may introduce somewhat different behavior. Thus, eventual confirmation is necessary that self-welding does not pose problems to the FTR operation. A seven shroud geometry will be used to test those parameters which were identified as the potential problems in the "component" tests.

Facilities

A heated sodium system with a pot large enough to hold the contacting portions of full size reactor components will be required. The pot will be in the neighborhood of 2 ft long and about 2 ft in diameter. The pot will be fitted with an access port and force or torque applying and measuring mechanisms. This arrangement will be located in an inert gas filled glove box.

The verification test using full lengths of the reactor shrouds will require a vessel capable of holding seven shrouds in prototypic sodium environment with simulated internal support structure. The vessel will be 2 to 3 ft in diameter by about 35 ft long, and contain simulated reactor cover nozzles.

Alternates

Perform No Experimental Verification

Experimental verification that self-welding conditions are not present could be eliminated. A review of the literature indicated that some authors feel that there is no problem;⁽¹⁾ others, however, state that self-welding is very definitely a potential source of trouble.^(2,3)

In view of these contradictions, it would appear that experimental work should be done to be assured that the surface hardness, finish, and materials of construction are adequate to prevent self-welding in the FFTF.

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1. *Reactor Handbook, 2nd edition, vol. IV, p. 330. 1964.*
 2. *J. M. Davidson, et al. Progress Report Conceptual Design Fast Flux Test Reactor, BNWL-CC-175, vol. 3, Section 3.4.11. Pacific Northwest Laboratory, Richland, Washington.*
 3. *W. J. Freede, et al. Static and Sliding Contact Behavior of Materials in Sodium Environments at Elevated Temperatures, NAA-SR-12446, September, 1967.*

GALLING OF REACTOR COMPONENTS (F-4)

Design Problems and Requirements

The reactor vessel contains many moving components and mechanisms which operate in the liquid sodium environment. These parts are made of austenitic stainless steel which is notorious for its tendency to gall and seize. The sodium environment tends to make galling worse because of the "cleansing" action of liquid sodium. Surface hardness is the property most effective in decreasing the tendency to gall.

The areas that are subject to galling are:

- 1) Holddown mechanism
- 2) Preload mechanism
- 3) Shroud assemblies at the inlet plenum tube sheet
- 4) Safety rods, shim control rods, and regulating rods.

Galling or seizure of the mechanisms or components would result in unreliable operation of central mechanisms, seizure of holddown or preload devices, damage to shrouds and vessel and high maintenance. The galling locations and materials must be identified and eliminated prior to reactor construction.

Experimental Program

The purpose of this test is to confirm the adequacy of design to prevent galling. The following variables are to be considered:

- 1) Surface hardness
- 2) Surface finish
- 3) Sodium temperature
- 4) Contact force.

Representative components of the reactor internals which are in sliding contact in liquid sodium will be tested. The components will be assembled immersed in sodium and subjected to loads (estimated subassembly weight at 700 lb; loads hundreds to thousands of pounds) and motion that correspond to those present in actual reactor operations.

The number of motion cycles given the test components will correspond to cycles the parts can be expected to undergo in their design life within the reactor. The sodium temperature will simulate that found in the reactor vessel in the area in which the components operate. The force or torque required to produce motion will be measured and the condition of the sliding contact surfaces will be inspected periodically.

These component tests will provide the proof-of-principal information for the design of the reactor internals. However, final verification of the design adequacy will require the prototypic forces which characterize the inserting and removing of shrouds in a packed bundle of adjacent shrouds. These data will be obtained in conjunction with other tests which evaluate the mechanical problems of refueling.

Facilities

The basic facility required will be a heated sodium pot large enough to hold the galling sections of the reactor components. Motion generating mechanisms and additional force and torque measuring equipment will be required.

Final verification tests require a sodium filled vessel capable of holding at least seven full length, full size shrouds (one control rod), with simulated internal support.

Alternates

Eliminate Testing

Experimental work to insure adequacy of design and material selection could be eliminated. The reactor components could be installed without testing. If, however, galling developed after the reactor was assembled, considerable extra expense and lost time would result. This condition would be even more serious if damage was done to a large permanent part of the reactor such as the inlet plenum tube sheet

DUCT FLUTTER DURING INSERTION (F-5)

Design Problem and Requirement

As a duct is being inserted into the reactor during refueling, the leading end of this long slender cantilevered member must move down through the flow of coolant streaming up from the vacant opening in the inlet plenum tube sheet. Potentially, a condition of instability could be produced causing the duct to flutter as it approaches the tube sheet. Such an occurrence would make insertion into the tube sheet more difficult and damage could result from the duct fluttering against neighboring components. Potential damage can be postulated for the duct, control rods, or tube sheet during the fuel handling operation, with attendant downtime. Additional channeling and crossflow, the latter from outlets of surrounding ducts, could affect the duct being inserted. Assurance that damaging flutter does not exist is necessary. Presence of flutter potential would require design modifications to bring this disturbance within acceptable limits.

Test Plan

Full sized ducts, representative of the various kinds in the core, will be lowered into a channel with water flow to simulate the actual reactor coolant flow velocity at

refueling conditions. The shape of the channel will represent the configuration of ducts surrounding a position in the reactor core array.

The fact that various ducts are tilted at different angles will tend to influence their dynamic behavior because some are hanging virtually plumb while others with greater slant are acting partially as cantilevers carrying their own dead weight during insertion into the core. Accordingly, tests will be run with the ducts inclined at different angles which represent various positions in the reactor.

Duct behavior during insertion and extraction will be observed. Design modification will be implemented, if flutter poses serious problems to the reactor design.

Test Facility

This test will require a full size mockup of a reactor channel with a simulated inlet plenum tube sheet penetration and a reactor cover guide tube. The duct will be surrounded by a prototypic envelope of simulated subassemblies. The channel will be fitted with viewing ports and will be trunnion mounted to allow tilting at different angles.

A circulating water system with a pump, instrumentation, and associated piping will be required to produce and measure variable flow to the test channel.

Alternate

Eliminate Testing

Experimental stability verification could be eliminated. It could be argued that there is insufficient reason to suspect the possibility of fluttering to justify testing.

Indeed, it is very difficult to state with complete assurance that the duct will become unstable and flutter during inserting. Conversely, failures due to mechanical instability have happened all too frequently when this problem was ignored during design. The history of many of the operating failures of the past attests to this error.

G. CONTROL MECHANISM DEVELOPMENT

The control rod mechanism for the Fast Test Reactor (FTR) although treated as a separate entity in this section of the development and testing program document must, by necessity be integrated not only into the reactor and core design, but also the fuel handling and maintenance concepts. The following discussion is limited to the control mechanism mechanical and electrical aspects of the design and development and does not include material properties, possible chemical reactions nor nuclear physics criteria. The source of electrical energy to actuate the control mechanism is not considered in this discussion. One of the basic criteria for the control mechanisms is that proven components of operating reactor systems will be used whenever possible. Since the FFTF is by design at the forefront of current technology, thorough evaluation is required even where near identical components are selected to verify components and establish high confidence levels required for safe and efficient operation of the FTR.

The tentative criteria for the FTR control mechanisms includes three types of devices: 1) in-core safety rods, 2) in-core shim-scram rods, and, 3) reflector shim rods. Each control rod occupies one complete reactor lattice position and contains a neutron poison material which is lifted out of the active core region by the drive mechanism. It has been further proposed that some or all of the in-core rods have a materials test section follower and the reflector rods have a follower composed of reflector material.

The control mechanism development and evaluation test program is subdivided into three areas. The first is a series of component evaluations. The second is testing of the assembled prototypes and the final phase is preoperational testing.

The first phase of testing is the component development and testing program. The control mechanism component evaluation program is required to provide data for the timely completion of preliminary design. To provide a safe and efficient integrated control system, not only must the design objectives of the individual components be complied with, but the interrelationships of other control system components and reactor systems must be considered. The proposed component evaluation is outlined in the following sections of this development plan. The order of the component discussions which follow is not intended to convey the relative importance of these components as all individual parts are equally important to attain a functional unit. Initial component evaluation will be in the vertical attitude, with the final tests performed at inclined angles simulating the typical FTR control element guide tube orientation. Generally, the component evaluation program will initially use water and air to simulate the reactor environment with following use of sodium and cover gas as required.

The second evaluation phase will use the assembled prototype control mechanism. Initial vendor plant tests will be performed in air and water to model the reactor gas and coolant. Complete prototypical conditions will not be attempted at the vendor plant because of the large expense and additional safety hazard imposed to heat and circulate sodium. After the control mechanisms have passed an acceptance test in water, they will be installed in the prototypical sodium test facility, (probably the CRTT at LMEC). In this facility, the prototype control mechanism will complete the life test conditions. The life test will, by necessity, be accelerated. This acceleration will be accomplished by reducing the test periods to minimum values and maintaining the drive and travel rates as required in the FTR design. The prototype evaluation will be conducted at the typical FTR inclined attitudes. The evaluation testing will be divided between hot isothermal flowing sodium and stagnant sodium with the predicted axial temperature gradient. The additional small risk of not testing the prototype control

mechanism simultaneously in both hot flowing sodium and at predicted temperature gradients is offset by the added complexity and increased cost of providing the necessary facilities, which would involve large heaters, heat exchangers and a relatively complex control system.

The final phase is the preoperational checkout of the production control mechanism. This phase will begin with testing of the prototype in the seven-duct test facility. This facility will contain an opening in the central position for the control rod. The preoperational tests in the seven-duct facility will include an extensive operational evaluation, including measuring drive rates and scram times. The accuracy and reproducibility of the control element position readout devices will be checked. The preoperational test data will be recorded for each control mechanism to establish a baseline for future operational checks to detect incipient failures. The compatibility of the control mechanism with the driver assembly and driver closures will be checked.

Upon completion of this final mockup test series, the control mechanisms will be transferred to the FFTF. In the FFTF, the relationships of the reactor vessel and cover head, closed loops, driver assembly closures, gamma shields, electrical and instrumentation leads, and fuel handling machines to the control mechanism will be checked. A series of demonstrations will be performed to evaluate preliminary replacement, handling, and maintenance procedures using prototype equipment. The need for additional handling equipment will be verified.

The following discussions are directed to component design development testing only.

DRIVE MECHANISM (G-1)

Design Problem and Requirements

Requirements for the control mechanism drive component for the FTR, include accurate and predictable movement of the control elements. The control signal may be derived from the automatic control circuit, from the automatic safety circuit, or from the manual control panel. This control signal is translated into physical movement of the control element, which in turn changes the reactivity of the core in the desired direction. Many types of control rod drive mechanisms have been designed, built, and operated on nuclear reactors. Since the most common control energy is electrical, a majority of drives have utilized electric motors. Some designs, however, use alternate power sources such as pneumatic or hydraulic actuators because of physical size limitations on power requirements or need for either high control element velocity and/or high actuating force. Both the pneumatic and hydraulic systems use electricity as the primary power, however, all include stored energy devices.

Pneumatic drive mechanisms are relatively simple and fulfill the requirement in light duty applications. With proper selection of the actuating gas, leakage through seals would not be a problem nor a safety hazard. Due to the compressibility of the gas, accurate positioning of the actuator at a given intermediate location is not practical. However, drive for a two-position actuator may be easily accomplished with a pneumatic system. In the pneumatic type system the compressed gas may also be used to provide the scram accelerator force and drive force to free a stuck rod.

Hydraulic actuators have the advantage of extremely high speed capability and infinite position control. Hydraulically actuated drives provide large driving forces and can be locked in any position if the fluid is trapped in the actuating cylinder. In addition, an advantage of the hydraulic control system

is that the cylinder is relatively small in diameter. Electrical control systems are generally used in conjunction with servo and/or solenoid valves to complete the hydraulic control loop. The disadvantages are the requirement for a high pressure fluid system, fire and safety hazards when leaks occur and possible rapid actuator movement in the event of a component failure.

Electrical control element drive units have been developed to many specifications and have been powered by both alternating or direct current. Alternating current motors are generally preferred because of ease of directional control and limited maximum output speed in the event of a component failure. Other requirements of a drive component are minimum inertia or overtravel and limited maximum force which can be transmitted to the control element. Several motors have been developed which combine these functions with the drive function. One such motor is the stepping motor which requires a series of electrical pulses to step the motor a given number of degrees. These motors have essentially zero overtravel because each step is, by design, a stable position. This same quality provides a static brake as long as the electrical power is applied. Since the stepping motor's torque is limited by the size, an additional clutch is unnecessary to protect other parts of the system. The output torque of the stepping motor decreases as the speed (pulse rate) increases; therefore, in addition to the electrical pulse rate to limit the maximum speed, the drive will slip pulses even if the pulse rate could increase above the design value.

Two proven designs in pressurized water reactor systems have been developed which could be utilized in the FTR if engineering studies prove favorable. These designs have the drive coils and positioning devices external to the pressure vessel and the actuating mechanism internal in the reactor environment. In the FTR it would appear that the reactor environment would be deleterious to the drive mechanism

actuating functions; therefore, additional seals would probably be required. The Westinghouse magnetic jack drive consists of two groups of pawls, one stationary and the other movable. These pawls grip a central grooved rod which is moved or jacked in a series of steps by sequentially energizing three electromagnet coils. The step length is determined by the spacing of the grooves. This system has excellent reproducibility, but the large step length may make control of the reactor difficult. The most accurate position indication is counting the jacking pulses. Removing all electrical power allows the rod to drop under the action of gravity.

The Marbel Schebler rod drive system utilizes four roller nuts which ride on a lead screw to shim the control element. Two roller nuts are mounted on each rotor half. The magnetic force developed by the stator winding attracts the upper section of the rotor outward and forces the roller nuts to engage the lead screw. The control circuit sequentially energizes the stator windings to rotate the magnetic field, which in turn shims the control element. Removing all power scrams the control element by gravity.

In the FTR, the drive component will be located near the reactor vessel head to permit ready removal, adjustment and maintenance. The total travel of the drive mechanism will be designed to move the control element from its least reactive position to its most reactive position plus any additional travel required for latching or drive mechanism removal operations. The maximum smallest control element jog increment will be consistent with the control element worth and reactor kinetic nuclear properties. Since the control drive is critical to reactor safety, a thorough testing program must be conducted to prove the principle, verify the design, and assure the high required reliability.

Test Program

The drive mechanism component tests will be performed in air at ambient conditions. The drive unit will be mounted vertically with provision to operate at small inclined angles to simulate the FTR control rod guide tube orientation. A weight will be added to simulate the extension rod and control element, and to apply representative forces to the drive mechanism.

To evaluate the drive mechanism, travel rates will be measured. Also, the maximum insertion and withdrawal forces of which the drive is capable will be measured along with the input power and temperature rise of the drive motor. Position readout accuracy and reproducibility will be recorded and the minimum jog step and overrun will be measured. A complete life test will be performed during the prototype evaluation test series to establish the reliability and expected life of components.

Test Facility

The drive component test stand is mounted in the PNL 314 Building. The drive is mounted vertically with provision for testing at small inclined angles to simulate the control element guide tube. The tests will be performed in air at ambient conditions. A weight to simulate the control element will be attached to evaluate drive rates, jog increment, and overtravel. The maximum available force to free a stuck rod will also be measured.

Alternates

Alternate drives will be evaluated if the above proposed drive mechanisms are not acceptable for the FTR. Initially several drives will be considered and the most attractive followed thru complete development with the second most attractive held as a backup.

SCRAM LATCH AND EXTENSION ROD (G-2)

Design Problem and Requirements

The scram latch and extension rod couple the control poison element to the drive mechanism during operation and decouple the safety control element during emergency safety action. In EBR-II, SRE and Hallam sodium reactor designs, the extension rod always remains connected to the control element. In the FERMI, the poison section is decoupled during the scram cycle. The major advantage of not uncoupling the poison element from the drive assembly or the extension rod is that without decoupling the ability is maintained to accurately determine the rod position or that the rod has been fully inserted following receipt of the scram signal. One advantage of disconnecting the control element from the extension rod are that the element can be scrammed in the unlikely event that sodium surface crud binds the extension rod. Another advantage is that the drive seals do not have to experience rapid relative motion.

The FTR conceptual design is based on the use of an electromagnetic latch with the armature connected directly to an extension rod similar to the FERMI design. The electromagnet will be located below the drive mechanism in the cooler region of the control rod guide tube. This rod configuration is consistent with current fuel handling machine refueling concepts. This concept will provide an indication that the rod is fully withdrawn from the core. The scram delay time will be minimized because the latch magnet armature is attached directly to the extension rod, eliminating additional mechanical linkages.

The certainty of the operation of the electromagnetic latch in the proposed configuration is not assured. The latch must securely hold the rod to prevent spurious scrams, at the same time the latch must be capable of reliably

dropping the rod on scram demand. The problems in operability and reliability associated with crud buildup from dirt or failure of a sodium seal must be explored. If the resulting FTR design calls for a rod latch in the sodium environment the crud buildup of sodium becomes of critical importance to the operability, therefore, testing must be performed to assure that the most severe conditions can be accommodated by the final design. The armature of the electromagnet must be designed such that residual magnetic flux will not interfere with prompt dropaway of the control element.

Test Program

Electromagnet tests will be conducted to determine the scram release time. Time from breaking magnet current to rod breakaway will be measured with a recording oscillograph.

Initial testing will be done in air because the current conceptual design locates the magnet external to the cover gas. If the concept changes to include a latch mechanism located below the sodium free surface, the latch will be evaluated in stagnant sodium. The sodium will be heated to predicted operating temperatures (up to 1200 °F) during the latch and unlatch operations. The latch will be subjected to predicted external forces imposed by the weight of the control element and accelerating spring (approximately 600 lb). Axial forces through 1000 lb in both compression and tension will be imposed on the latch to simulate freeing a stuck control rod. The compression and tension thrust capability of the rod will be determined. The complete extension rod and latch assembler will be life tested during testing of the assembled prototype. The latch magnet component evaluations are required to verify that scram mechanical release times can be achieved under all conditions within the design requirements of the FTR.

Test Facility

The test facility for the electromagnet evaluation will be mounted vertically. Since the conceptual design located the magnet above the reactor head, the electromagnet will be tested in ambient air. Provisions will be made for recording coil temperature, electrical current, magnet separation force, and release delay from interruption of the energizing current to dropaway. Plans do not include component testing of the extension rod but exhaustive tests will be conducted during the prototype test under prototypical conditions.

Alternates

If, during preliminary reactor design, the advantages of a mechanical latch outweigh the disadvantages, an evaluation program will be started. The FERMI design appears to offer several advantages and is proven in sodium service. Component evaluations will be conducted in 1200 °F sodium in the vertical orientation with a weight to simulate the control element weight and initial accelerating spring force.

SEALS (G-3)

Design Problem and Requirements

Control mechanism seals are one of the important components that contribute to the reliability of the control mechanism design. The major functions of the seals are to separate the reactor cover gas and sodium vapor from the drive mechanism, reduce leakage of cover gas to a minimum and reduce contamination of the cover gas from in-leakage of hot cell atmosphere. These functions must be met in a manner that will minimize frictional drag and reduce the potential for binding.

The selected seal must meet as nearly as possible the following requirements:

- 1) Have zero leakage
- 2) Be capable of 36 to 48 in. axial motion
- 3) Add a minimum of diameter to the parent assembly
- 4) Have minimum potential for binding
- 5) Have low frictional drag
- 6) Capability of accommodating high acceleration, deceleration and high axial velocity
- 7) Must be simple and easily replaceable
- 8) Have potential for high reliability
- 9) Be impervious to sodium and sodium/gas atmospheres
- 10) Have ability to operate over long periods of time in the 250 to 1200 °F temperature range.

Size of components in the FTR is of critical importance and since the reference concept seal, which is a bellows seal is inherently space consuming, great attention must be given to acquiring a seal that meets all the above requirements within a minimum diameter.

Experience at both ANL and FERMI indicate that for long bellows life, the seal should be located in the cool region of the system. Sodium freezes at 208 °F. The temperature in the bellows region should not span this temperature because excessive stresses will be developed in the bellows with frozen sodium in the convolutions.

Test Program

Seal component evaluation tests must be performed in sodium test stands because testing at other than prototypical environmental conditions will not evaluate the design problems nor indicate incipient failures.

A simulated section of the control element guide and extension rod will be mocked up. The test section will have capabilities of maintaining predicted temperature gradients. The seal/extension rod assembly will be cycled in hot sodium

up to 1200 °F. The cover gas will be supplied at the appropriate interface location. The mechanism will be operated at design conditions of acceleration, deceleration and velocity of the FTR control element, including scram cycles. Accelerated tests must be performed to complete the required number of life cycles within a relatively short time; therefore, the normal rest periods will be reduced.

The seal components tests must, of necessity, be full scale to adequately evaluate problem areas. Initially, the test temperature gradients will be based on the best conceptual design information. These selected gradients may change as the preliminary design progresses, with a resultant base of a more severe operating condition. The seal will be subjected to predicted operating stresses imposed by both the operational requirements, physical orientation and thermal gradients. The effects of chemical corrosion and grain boundry defects will be evaluated by metallurgical examination and destructive testing at the successful completion of the component tests.

The reference seal concept is the nesting bellows. All known vendor/manufacturers of this type of seal will be contacted to determine their capability and interest in providing test bellows seals for this application for evaluation. Concurrently, other potential sealing techniques will be reviewed and several of these with high potential will be examined as a backup concept.

If, during the preliminary design, the space required for a nesting bellows seal is found to be not compatible with the other design features, effort will be concentrated on evaluation of alternate seal techniques. Cyclic life testing based on a 20 year design life of the seals will be performed under prototypical conditions in the prototype testing facility.

Test Facilities

The test stand will consist of a pot which contains sodium and cover gas above the sodium. Individual heaters will be provided to allow the sodium temperature to be raised to 1200 °F, and temperature gradients to be applied axially over the test specimen.

Alternates

Two alternate seal designs are under consideration which require only minimum volume requirements. The two types are the labyrinth seal and the chevron packing seal. Of these two, the labyrinth type is preferred because of the low frictional drag. The major disadvantages of the labyrinth design is the requirement of a buffer gas flow to minimize sodium vapor diffusion back into the drive mechanism area. This gas flow may introduce oxygen and other contamination into the cover gas. A pump and filter system is required to circulate the seal buffer gas. In addition, as the control mechanism moves, the control element extension rod penetrating the seal will transfer deposited sodium across the seal area. Preliminary calculations indicate that high buffer gas velocity is required to prevent back flow. The second alternate seal design is the chevron packing type seal which will minimize sodium transport on the control element extension rod because of the wiping action; however, this wiping action also increases the frictional drag. The gas leakage problems of the labyrinth seal are reduced in this concept but not eliminated.

Failure to develop a suitable nesting bellows located in the cool top region of the control rod guide tube to seal the control mechanism from the reactor cover gas will require the use of a possibly less desirable seal design with increased control envelope above the reactor cover.

ACCELERATOR (G-4)

Design Problems and Requirements

A control rod accelerating device is required in the FFTF safety control mechanism to rapidly insert the control poison element into the core to shut down the reactor. Possible acceleration methods include gravity, springs, compressed gas, flywheels and clutches, hydraulic motors or cylinders, or a combination of these. Stored energy must be available for acceleration in case a loss of normal system power occurs. Gravity is assessed as not being adequate for the safety rod application since initially required acceleration is 2 to 3 "g's" to meet conceptual safety criteria.

Springs are a recognized stored energy device that do not rely on an external power source. Springs have two major disadvantages: 1) they may relax when subjected to high temperatures, and 2) they may break if operated in a corrosive atmosphere that weakens grain boundaries. An advantage of springs is that they can generally be positioned concentric to other components in a mechanical assembly, thus, the overall dimension of the control rod drive mechanism can be minimized. However, since space limitations on the top cover of the FTR are so restrictive, design of the spring to perform the desired function within a small envelope becomes critical. The spring cross section, outside and inside diameter, free and compressed lengths are of prime importance.

Compressed gas has good stored energy potential. However, in the FTR with quite a restrictive space limitation, the gas accumulator or pneumatic storage vessel becomes a problem in the height of the overall assembled mechanism. If the accumulator were placed in a less congested location, the piping from the storage to the control mechanism would still be a major deterrent to good space utilization due to the relatively large pipe size required to avoid a high pressure drop during scram.

Springs have been selected as the reference stored energy source for the FTR control drive mechanisms. They will be used to provide the initial acceleration requirements of between 2 and 3 "g's" for the safety rod insertion.

Mechanical accelerating mechanisms, such as flywheels, weights with mechanical advantage, etc., will not be considered at this time due to their inherent size and relative complexity. Hydraulic systems are not considered due to the safety disadvantages of most hydraulic fluids in relation to the sodium environment.

Test Program

The basic spring design using Inconel as the reference material will be tested in air for basic properties. The integrated dynamic force over the extension length will be measured by use of load cells. High stress points will be located by use of the Stresscote process. Strain gauges will be placed at high stress points to measure the strain at stress concentration locations through the spring cycle. A simulated, representative load will be used and acceleration will be measured using a calibrated accelerometer and light beam oscillograph. These data will be reviewed in relation to existing data in the literature on creep, strain rate, fatigue, yield and ultimate strength of the material.

The above test will provide verification that the spring, as designed, will meet the basic mechanical requirements. A series of tests specifically aimed at material performance will be conducted. Initial tests will be conducted in inert gas at elevated temperatures up to 1200 °F to evaluate spring relaxation rates. The spring material will be pre-stressed during this test and the temperature will be held constant for several thousand hours. During this test, the reduction in spring force versus time will be measured. The second

material test phase will be an investigation conducted using the spring in hot (to 1200 °F) stagnant sodium to evaluate intergranular corrosion. Spring relaxation rates will also be measured in this sodium test.

Final testing of the selected accelerator design will be performed during the testing program for the assembled prototype control rod. During this testing, life cycle tests will be performed, as well as periodic measurements to determine incipient failure, ability to maintain required force throughout the required travel and ability to continue to maintain required acceleration rates.

Test Facilities

The initial design verification tests will be performed in a simple vertical spring test jig.

The materials testing will require an inert gas electric furnace capable of continuous 1200 °F operation. The furnace will require modification to provide a preload of the spring to approximately 600 lb and appropriate force measuring instrumentation to monitor relaxation.

The sodium phase of testing will be done in a sodium pot heated to 1200 °F. The spring in the pot will be preloaded during this portion of the test to 600 lb and monitored for relaxation.

Alternates

If an acceptable spring design cannot be achieved from a space, materials, or performance standpoint, a pneumatic system will be evaluated. The pneumatic system will be similar to that used in the EBR-II and FARET design.

POISON ELEMENT (G-5)

Design Requirements

The poison element must maintain a high degree of integrity throughout its life. In both the proposed FTR and FERMI designs, the control element is located within the control guide tube in the pressure vessel. Sodium coolant flows up around the control element to remove the heat generated within the element and structural members. Previous experience with similar flow geometries have shown that flow induced vibration can result in material fatigue or fretting wear. Flow testing in water to simulate sodium at various flow rates will reveal if any flow instabilities or induced vibrations are inherent in the FTR design. The proposed design has an advantage in that the poison is contained in six individual tubes; therefore, failure of any single tube will not result in complete loss of poison from the cluster.

Test Program

Test Plan

Component flow tests on the poison section will be conducted in hot (250 °F) water to simulate kinematic viscosity of reactor sodium at 900 to 1200 °F. The flow system will be pressurized to 50 psig to prevent boiling of the water. Although normal coolant flow will be approximately 20 gal/min, the flow will be varied from zero to 100 gal/min to investigate a wide range of induced vibrational frequencies.

Test Facility

The poison element component will be tested in the EDEL loop #1 at PNL. The test rig will not only be instrumented to detect element vibrations, but also the pressure loss of the flowing fluid to provide hydraulic data for the preliminary design. The test setup will be full scale and will be

able to simulate all operating modes including scram cycles. Final testing under prototypical conditions will be completed during the prototype life test cycle using hot circulating sodium at LMEC.

Alternates

If design studies indicate that additional poison volume is required for adequate reactor control, the redesigned poison element will be evaluated as outlined above.

DECELERATOR (G-6)

Design Problem and Requirements

A decelerator or snubber is required to reduce resultant stress level within the control element and guide tube during the scram cycle. Several decelerator designs using gas, oil, or sodium for the operating fluid have successfully been used on sodium cooled reactors. In the SRE and Hallam reactors, the control element is contained within a thimble. This thimble is filled with gas and, therefore, gas was the logical fluid for the snubber. The snubber is located in the shield assembly above the active core and consisted of a piston on the control element extension rod and a dash pot in the guide tube.

The EBR-II decelerator is located above the reactor vessel head to allow ease of inspection. Vertical space above the EBR-II is not at a premium as is the case in the FTR design. The decelerator is filled with oil and consists of a hollow, cylindric tapered piston and an annular dash pot. This design was also proposed for the FARET concept.

The FERMI design uses a sodium filled dash pot arrangement located below the active core region because the control element separates from the extension rod during the scram cycle. The use of a hydraulic dash pot decelerator, using reactor

coolant fluid below the active core is preferred due to the extreme difficulty in installing any other type in the core region. The design minimizes the space requirement in the region of the reactor head.

The reference design for the FTR is similar to the FERMI design in respect to method and location. The tapered dash pot section will be attached to the control rod and can therefore be changed by removing the rod as required by future design changes. Loss of snubber fluid is not possible because the dash pot is submerged in the reactor coolant. However, the design calculated piston cylinder relationship and flow annulus of the dash pot must be confirmed.

Test Program

The effects of mechanical forces imposed by acceleration will be investigated on a full size model of the dash pot. The control element will be mocked up to simulate the flow passages in the dash pot and the weight of the extension rod. The dash ram, control element, and weight simulator will be accelerated by a spring to simulate the required initial acceleration and velocity as the ram enters the dash pot. Design changes in the ram will be made as necessary to achieve a uniform deceleration near the end of the stroke.

If, during the preliminary design, the proposed method of deceleration proves to be not compatible with the core design, alternate methods will be investigated to provide the required decelerator design data for preliminary design activity. Life testing will be conducted during the prototype test program using hot flowing sodium. The effect of sodium buildup on the dash pot will be checked although this is not anticipated to be a major problem.

Test Facility

A water test rig has been assembled at PNL. This assembly uses hot water (250 °F) to model the sodium hydraulic conditions. The simulated control element is accelerated by a spring to the designed velocity of 10 ft/sec upon entering the dash pot.

Alternates

If the below core location becomes unacceptable from a materials and/or radiation damage criteria, an above core design will be investigated. The alternate snubber design would likely be a gas filled cylinder.

H. CLOSED LOOP PROTOTYPES

The closed loop systems push current materials technology because of the combination of high neutron flux and high temperature to which they will be subjected. They also interact strongly with other reactor components and systems. Specific closed loop hardware design is highly dependent upon details of the selected reactor concept and also upon functional testing requirements. The functional testing requirements will be the subject of continuing studies. Without a firm reactor concept or functional basis at this time, it is possible only to assess development testing requirements on a typical or representative basis. It can be expected that the actual development activities required will vary in scope and depth in proportion to and in phase with future changes in design requirements.

CLOSED LOOP TOP FACE HARDWARE (H-1)

Design Problems and Requirements

The major problem related to top face hardware is the selection of sealing methods for prevention of liquid sodium leakage. Current design concepts indicate that there may be between three and five joints required to provide the required capability for removal and replacement of test specimens or fuel, removal of complete loops, and replacement of coolant jumpers. Mechanical seals are a logical consideration, but the limited experience with them in a sodium environment restricts their use where a high degree of reliability is required. Grayloc and Conoseals have been used in loops that are accessible for manual removal of seals, clean up of mating surfaces, and manual fit up and replacement of the new assembly. An insufficient number have been employed to make a realistic evaluation of their statistical reliability under conditions to be expected in the FFTF where the removal

and replacement will be by remote means. A further complication is the lack of space available for seal locking devices. Galling of threads in such devices is certain to be a serious problem. Conventional locking device approaches are ruled out for the most part because of the lack of space and the galling problem.

Since the primary purpose of the FFTF project is to provide reliable closed loop testing facilities, advance testing of hardware is required.

Considerable work has been done on programmed and remote welding and cutting of pipe and tubing in a wide range of sizes down to 1 in. ID. The work to date apparently has not involved systems containing liquid sodium or sodium vapors. The effects of sodium and/or vapors on arc behavior, weld penetration, weld quality, and the possible need for prior clean up of the joint needs to be investigated. Also weld joint design will very likely be extremely important.

Experimental Program

Seal testing will involve proof-of-principle testing of mechanical seals prior to design of a hardware prototype, followed by statistical reliability testing of the prototype and design verification testing of the final design prior to installation of final hardware in the FFTF. The first two tests are primarily directed at the seals while the third test evaluates a final design of the complete hardware. It should be anticipated that available mechanical seals and locking devices may prove to be unsuitable. For this reason it is important that a backup sealing method be investigated concurrent with the above program.

The sealing method or methods that survive the proof-of-principle testing would be incorporated in a hardware

prototype design for statistical reliability testing. This will involve repeated making and breaking of seals under conditions expected in actual service. Environmental conditions between seal replacement will be essentially the same as those employed during feasibility testing. The number of tests performed will be based on the level of statistical reliability and confidence desired. For example, if the joint employing the seal were connected and disconnected 100 times in succession without producing a leak under the condition listed above, there is 90% confidence that the statistical reliability would be 97.7% or higher.

The sealing methods which survive the above testing will then be incorporated in a final design of top face hardware for design verification testing. These tests would involve the same environment as the feasibility tests, but would also involve evaluation of the effects of flow reaction, dimensional fit and distortion, fretting and galling and interface with the fuel handling machine.

Feasibility tests on both the mechanically driven TIG and the PIGME arc welding processes should be initiated and if successful be followed by reliability and design verification testing. Cutting equipment development should parallel this activity.

The tests needed to establish feasibility will involve subjecting the sealing methods to static and dynamic, pressure, thermal expansion, and external forces as well as vibration and hydraulic and thermal shock in a liquid sodium environment.

Test Facilities

The facilities required for the feasibility tests should allow testing full scale seals and joints in nonflowing

He, H₂O, and Na environments. Water testing may be a means of rejecting seals in advance of sodium tests. The helium test anticipates that certain seals not suitable for sodium may be entirely suitable for cover gas or sodium vapor applications, perhaps for driver hardware. The facilities required for final testing amount to a full scale prototype of a complete closed loop system. This facility is described in more detail under the complete closed loop system testing description.

Alternates

Since evaluation of alternates in this case is largely a matter of pointing out the consequences of not performing certain tests, it is important for purposes of this discussion that the objectives of the various levels or types of test be defined. In general, testing of closed loop components falls into five general categories: 1) proof-of-principle tests or tests which demonstrate the feasibility and or characterizations of a particular approach, 2) design verification tests which demonstrate that certain design characteristics have been provided, 3) statistical reliability and/or endurance tests which demonstrate that functional and materials requirements will be met throughout the predicted life of the component, 4) materials testing, and 5) system operability tests. System testing may involve 1) through 3) above, but as a rule, is intended to demonstrate compatibility and operability of individual components when integrated into a complete system. Tests 2) and 3) are usually the basis for acceptance or rejection of individual components developed, while test 5) is the basis for integrated systems.

In the case of top face hardware development, alternates could involve any one, a combination of several, or all of the tests. The plan described above involves tests 1), 2), and 3). Test 4) is part of a separate program, and test 5) is part of the complete closed loop test discussed later. If proof-of-principle tests 1) on several sealing and fastening methods are not carried out in advance of the first prototype hardware design, it is possible that a complete prototype would eventually have to be built and tested for each type of seal available (and there are many), and for each location of seals (3-5) in the hardware. This is true because the hardware design is dictated largely by the particular seal employed, and if the seal selected fails, the complete prototype also fails. Proof-of-principle testing of seals and sealing methods per se does not require a complete hardware prototype and if performed as an initial phase can eliminate considerable unnecessary expense.

Statistical reliability testing 3) of seals and sealing methods could be eliminated with the possible consequence that topface hardware would leak after several loop change-outs and a new design would have to be developed and installed after start up, including appurtenances permanently attached to the reactor cover. This would be costly and could result in considerable delay in the irradiation testing programs for which the reactor is being constructed.

Elimination of the design verification tests 2) in advance of start up could result in the same consequences mentioned above, but for different reasons. The tests above have dealt with seals and sealing methods only and have not considered, for example, freeze-up and thaw-out problems, flow consideration, or any of the many other functional and materials requirements specified for topface hardware. Failure to meet any one of these requirements could put in jeopardy the timely achievement of the major goals of the FFTF project.

CLOSED LOOP IN-REACTOR HARDWARE (H-2)

Design Problems and Requirements

One of the major problems that may be encountered with in-reactor hardware including the closed loop tube, meltdown cups, fuel hangers, and data readout instrumentation, is vibration of the in-reactor duct and internals due to flow of the coolant. Because mechanical fatigue, galling, and fretting may foreshorten the useful lives of these components, vibration must be minimized. It is important that the forcing frequency set up by fluid flow not be in the same range as the natural mechanical frequency of the components. For the complex shapes and distribution of masses that are proposed for the incore components, it is very likely that analytical methods will not suffice, and experimental determination will be required. Since a wide range of flow velocities, temperatures, and mass flows may be involved to accommodate the variable testing function of FFTF closed loops, an even greater number of experimental determinations than for driver ducts may have to be made to guarantee reliable operation. Mechanically induced vibration due to coolant pumps and thermal shock also deserve consideration.

Experimental Program

In order to make a realistic evaluation of the problems involved, it will be necessary that the test be as nearly prototypical as possible. Initial tests with water can be run to simulate flowing sodium, however, final evaluation should be made on the basis of actual coolant variables. Typical test data will include graphs showing vibration amplitudes as a function of flow at various temperatures. The deflections can be converted into stress levels, and with frequencies known, a design can be developed which has a reasonable life expectancy. In order to provide reliability

assurance on these components, it will be necessary to carry out endurance tests to verify design and fabrication methods. The tests will involve subjecting the in-reactor hardware to prototypical conditions in terms of pressure, temperature, flow induced vibration, thermal shock, hydraulic shock, mechanically induced vibration and dynamic external forces. Effects of flow-induced vibration created by liquid sodium streams both inside and outside the in-reactor closed loop tube and internals will be of interest.

Test Facilities

The facilities required to carry out these experiments involve a full-scale closed loop system having flowing water for initial tests and sodium for final verification. A more complete description is given under the section on Complete Closed Loop System.

Alternates

The experimental tests with water could be eliminated and design be based on very limited theoretical analysis. This could lead to failures in the endurance testing phase and cause considerable delay in the overall development program. Both phases of testing could be eliminated with the consequence that the testing and development would be carried out after reactor start-up. If failures occur after reactor start-up, in all probability it would be necessary at that time to initiate a testing program similar to that described above.

CLOSED LOOP INTERCONNECTING PIPING AND DISCONNECTS (H-3)

Design Problems and Requirements

The closed loop interconnecting piping has design problems related to piping disconnects, containment of sodium in

case of leaks, heating apparatus for thaw-out of sodium during replacement of loop components, and electrical and instruments with respect to mechanical fit-up and routing.

The primary problem has to do with developing leak tight joints for the required disconnect points which will withstand the routine operating conditions, as well as thermal, hydraulic, and mechanical transients and vibration. A means must also be found to contain the sodium during removal of piping. Retention of sodium and containment of radioactivity in case of pipe failure is another consideration. This may involve a double-walled piping system. Methods for fabricating, replacing, and inspecting the double piping will be required. It is important that a means of detecting leaks be developed.

Experimental Program

Development of components will involve building one or more full scale prototype pipe and disconnect arrangements and subjecting them to the conditions typical of closed loop operating parameters. Feasibility tests on these components will also include removability, and methods for inspecting and detecting leaks. Water, helium, and sodium at temperatures of 200 °F for water and 1350 °F for gas and sodium, and up to 250 psig pressure will provide the test media. The test specimen will be subjected to various steady static and transient thermal, hydraulic and mechanical loadings.

The above tests will provide a basis for final design of interconnecting piping components, but will not provide assurance that the designed hardware will be reliable. For this reason, endurance testing of the final hardware will also be required.

Test Facilities

As in the case of in-reactor hardware, endurance tests on inter-connecting piping components will require a complete closed loop system.

Alternates

The above tests could be eliminated and design be based on theory, extrapolation of other applications, and judgement. In all probability, it would lead to some sort of failure after start-up and considerable delay in the irradiation testing program.

CLOSED LOOP COMPONENTS EXTERNAL TO REACTOR VESSEL (H-4)

Design Problems and Requirements

Closed loop components external to the reactor vessel include pumps, valves (shut off and check), cold traps, heat exchangers, sodium sampling and purification systems, system instrumentation and fission product transport and decontamination systems. The development effort discussed below does not discuss traps, sampling or purification devices. Pumps and heat exchangers are discussed only because they relate strongly to the facility requirements.

In spite of the numerous liquid sodium facilities that are in existence at the present time, there has been very little experience involving check valves and shut off valves in the 3 in. range and above. Most sodium valves have been custom built and tested prior to actual service. It follows that a similar approach is needed in this case, particularly since reliability in closed loop components is the primary objective of the FFTF Project.

The pumps as conceived for the closed loop system are beyond current technology. The anticipated temperature of 1200 °F, coupled with a high head requirement of 175 to 200 psig make the pumps unique.

Heat exchangers are also of questionable reliability since difficulty with internal flow distribution has been a problem in sodium systems.

Experimental Programs

The development plan for the valves, pumps and heat exchangers would involve: 1) proof-of-principle or feasibility testing, followed by 2) construction of prototype for statistical reliability tests, and finally, 3) design verification testing of the final design in the complete closed loop system.

The performance in sodium over wide flow and temperature ranges need to be verified. The pump and heat exchanger characteristics need to be determined.

Test Facilities

The facilities required for this testing should be full scale water where suitable followed by full scale sodium at the closed loop operating conditions of flow, pressure and temperature. Temperatures will range from 1100 to 1350 °F at flows from 100 to 700 gal/min under pressure ranging between 150 and 250 psig.

Alternates

As in the case of top face hardware, the various alternates available involve elimination of one or more of the proposed development tests described above. The possible consequences are similar since in each case elimination of tests can result in failures after start-up and delays in the irradiation testing programs.

COMPLETE CLOSED LOOP (H-5)

Design Problems and Requirements

The major requirement of the complete closed loop test is to evaluate the performance, operability, and compatibility of all the individually tested closed loop components after they are integrated into a complete system. Each of the component categories will have had some feasibility or characterization type tests followed by endurance and/or design verification tests, but none will have been tested on a system basis. A system test is required to demonstrate compatibility and operability of individual components when integrated into a complete system. The various levels of testing of individual components dictates the use of a complete closed loop under completely prototypical conditions.

The maintenance or maintainability problems with closed loop components need to be determined as thoroughly as possible prior to reactor installation. The adequacy of pump speed control, the efficiency of the heat transport system, the cover gas control and pipe reactions of the closed loop system must be determined in the complete system prototype test.

Experimental Program

The closed loop system will be operated under typical reactor operating conditions, and off-normal events will be simulated. The operating mode will be directed toward the evaluation of performance, operability and compatibility of the total system, the interfaces to the system and individual component performance in the system.

Test Facilities

These tests require a full-scale prototype of a complete FTR closed loop. This should duplicate the closed loop in-reactor duct plus the geometry of adjacent driver hardware, with a sodium pumping system and heaters capable of producing flow conditions around the closed loop tube equal to that expected in actual service. Adequate space above the pot will be required for removal and insertion of in-reactor hardware. Capability for simulation of fuel handling apparatus should be incorporated. Lay-out of the facility should also provide space for the length of inter-connecting piping that is prototypic of the final reactor concept. In addition, a separate pumping and heat-exchange system will be required to provide the flow conditions expected inside the closed loop system. In each, flows will be in the order of 100 to 700 gal/min at temperatures up to 1350 °F, and at pressures between 150 and 250 psig. The facility should be planned on the basis of exclusive use by closed loop testing activities.

Alternates

The alternate to the complete system test described above, is to carry out the test after reactor start-up. The possible consequences are that individually proven components would not prove satisfactory when integrated into complete systems because of mechanical fit-up and interferences. Also, interface with the fuel handling equipment could cause problems and delays in getting underway with the irradiation testing program.

I. INSTRUMENT DEVELOPMENT

Detailed analyses have been carried out for power excursions, loss of flow, release of fission products, and surveillance instrument needs to establish the technical basis for the FFTF driver fuel instrumentation.⁽¹⁾ Reference 1 reviewed the present technologies as applied to fast sodium-cooled reactors for temperature, flow, failed fuel, coolant pressure, in-core neutron flux, and signal transmission and data reduction with respect to such characteristics as response time, sensitivity, and reliability.

The primary need and, hence, the goal of the instrument development and testing program is to improve the life and reliability of the sensors in FFTF service, rather than improvement in some characteristics such as response time compared with the values usually associated with that particular type of device in customary applications.

Since the state-of-the-art experience seldom describes sensors that have been stressed by environments comparable to those expected in-core and near core in the FFTF, comparisons of present performance in other applications and predicted performance in the FFTF are not practical to make in a realistic, quantitative manner. The first problem faced by the developer for the FFTF is to make the sensor perform at all reasonably in the unprecedented environment and confined space. Following this, the development effort strives to approach the performance achieved in existing, less demanding applications and to provide an acceptable lifetime in the FFTF environment. The following is a list of instrument development programs presently underway under PNL direction:

- 1) Gas Chromatograph Fuel Failure Detection
- 2) Delayed Neutron Fuel Failure Monitor

1. R. E. Peterson. Technical Basis for FFTF Driver Fuel Instrumentation, BNWL-555. Pacific Northwest Laboratory Richland, Washington, September, 1967.

- 3) Gas Disengagement for Failed Fuel Monitoring
- 4) Pressure, DP Flow and Level Sensors for In-Reactor Sodium Service
- 5) In-Core Sodium Flowmeters
- 6) In-Reactor Coolant Temperature Sensors
- 7) Reactor Environmental Effects on Signal Cables
- 8) Signal Connector for Sodium Service.

These efforts, as well as others which are known to require development and testing support, are discussed in this section. This program is consistent with attainment of 1400 °F in the closed loops.

IN-REACTOR COOLANT TEMPERATURE SENSORS (I-1)

Design Problem and Requirements

Technical bases are needed for the specification of FFTF coolant-temperature thermocouples whose response in the FFTF environment has been determined and whose reliability has been statistically established. No attempt will be made to develop a new type of thermocouple wire alloy.

One major design problem is to accommodate or circumvent the differential expansion of the 304 stainless steel sheath and the Chromel-Alumel thermocouple wires.

The environment will be about 10^{14} n/cm²-sec, 10^8 R/hr, in 1200 °F (and possibly to 1400 °F) flowing sodium. The effects of fast neutron exposures of the flux and fluence magnitudes expected have not been measured.

Experimental Program

Test Purpose and Methods

The test program is a repetitive one in that the data from the tests will cause new specifications to be written and more tests performed on the improved thermocouples. An outside contractor will be given a contract to write specifications based

on ASTM-E-235 and his own experience with reactor grade thermocouples. A quantity of short (30 in.) thermocouples will be purchased, using an extensive manufacturing inspection, quality control and acceptance testing program.

Thermal cycling and thermal shock testing in the range of 600 to 1200 °F (and possibly to 1400 °F) will be performed. Radiographs, calibration, speed of response measurements, dye tests, and insulation resistance measurements, will also be performed.

Uncycled thermocouples will be placed in EBR-II to obtain radiation damage effects information and data on decalibration due to radiation. Full length thermocouples, on the order of 35 ft. in length, will be ordered and tested. Destructive testing will be undertaken if the thermal tests show a large failure rate. Corrective specifications will then be written and the above tests will be repeated. The fast neutron fluence will be limited by the time operating efficiency of EBR-II and the flux level in the core positions made available.

The number of couples that require testing for statistical verification of five-cycle lifetime in FFTF with an acceptable failure rate will probably not be achieved before FFTF operates. Therefore, the coolant temperature sensor development program may continue for several years after FFTF goes into operation, using FFTF as the testing facility.

A destructive testing program will be undertaken to identify causes of failures due to radiation exposure. Since no fast flux facility exists that allows measurements to be taken during irradiation, pre- and postirradiation calibrations, speed of response and insulation resistance measurements will be made. Possible instantaneous errors due to extreme gamma and neutron fluxes are being investigated in gamma pit and pulsed-reactor facilities.

Range of Parameters

The calibration temperatures will range from 400 to 1400 °F. Neutron exposures will be as great as can be obtained during the development period. Speed of response will be measured from 800 to 1200 °F. The number of couples tested will be limited to about 200 in the initial program due to funding.

Variables to be Measured

Speed of response, insulation resistance, change in wire resistance, changes in calibration as a function of neutron exposure, and lifetime are to be measured. Other measurements will probably be made as the program develops.

Information from Test to be Applied to Design

The specifications and technical backup information will be formally reported as bases for ordering FFTF construction thermocouples.

Test Facilities Required

The only facility for radiation exposure in a typical fast neutron spectrum is EBR-II, where sample thermocouples are now being irradiated. There are many hot cell facilities for doing postirradiation calibrations and metallographic examinations, including two at PNL. The thermal shock testing will require a small, specially built loop. Several laboratories have space available and equipment in storage to build the loop facility.

Alternates

Present state-of-the-art thermocouples could be used at the risk of an unacceptable lifetime and perhaps false temperature indications. Since FFTF is a test facility, knowing the environment is of prime importance. No acceptable alternate to testing is known.

PRESSURE, DP FLOW, LEVEL SENSORS (I-2)Design Problem and Requirements

The objective of the development and associated tests is to produce a pressure transducer capable of measuring the coolant pressure above and below the FFTF core while immersed in the sodium coolant and subject to the flux and temperature environment near the core.

The testing and development program is required because commercial pressure transducers are not rated for operation immersed at temperatures to 1200 °F, the effects of sodium corrosion on diaphragms are not completely determined, and the effects of radiation at the levels predicted for FFTF are not known.

The pressure transducers will also be developed and tested for use in measuring flow of the sodium coolant in the fuel channel by measuring the differential pressure of a venturi-type flowmeter. This is a backup effort for permanent magnet, turbine and eddy current flowmeters being developed by ANL.

Development must also be directed toward miniaturizing the units as required to fit in the allowable space of an FFTF fuel channel.

Pressure transducers will also be developed for measuring pressure out-of-vessel in other parts of the primary loop. In this less stringent temperature and radiation environment, a different, more inexpensive transducer may be specified.

Level measuring transducers will be required with both a narrow range of a few inches for level monitoring and control, and with a range up to approximately 30 ft for drain and fill monitoring. No activities are planned in FY-1968. Long range prooftesting will be done when specific design requirements are known, provided sufficient uncertainties exist at that time to justify such activities.

Experimental Program

Selected candidate pressure transducers will be procured from commercial vendors for tests in the sodium pot facility at PNL.

Initial tests are screen tests whereby the transducer is immersed in sodium while temperature and pressure are gradually increased to the manufacturers' maximum rating. Accuracy, sensitivity and zero drift will be measured and recorded.

In the second stage, the transducers will be modified in cooperation with the vendor in order to improve tolerance to high temperature and radiation. The materials and construction of each transducer will be examined in detail. Alternate materials for the diaphragm, bonding, insulation, and other components will be incorporated until a unit is developed to meet the following specifications as a design goal:

Pressure range	0-15 psi through 0-250 psi both gage and absolute
	0-20 in. through 0-400 in. water column differential
Temperature range	77 to 1200 °F
Radiation	In-reactor 10^{16} n/cm ² /sec fast neutrons
	10^{23} n/cm ² (total fast fluence)
	10^9 R/hr gamma
	Piping system 5×10^5 R/hr gamma
Nonlinearity	Less than 1% of full range
Hysteresis	Less than 0.5% of full scale
Thermal zero shift	Less than 0.03% of full range/°F
Response time	Less than 0.1 sec

This specification is a design goal and defines a very good commercial instrument. It is not a documented FFTF requirement. It is expected that the tolerances will be sacrificed in a trade-off to procure a unit which will operate in the high temperature and flux environment of the FFTF.

In a final test phase the transducers developed in the sodium pots will be tested in flowing sodium at 1400 °F and 250 psi to simulate, as nearly as possible, the FFTF in-reactor conditions. These tests should reveal the effects of flow on the corrosion rate and fatigue of the transducer diaphragms.

Radiation tolerance will be achieved by replacing questionable materials with other material of known radiation tolerance. Transducer components or the completed units will be irradiated in a facility such as the EBR-II.

The Instrument Design Section will be informed of the performance parameters of the developed transducers.

Prooftesting will be required as follows:

- The development program will continue until a specification for procurement of a suitable pressure transducer is completed along with identification of qualified vendors. The units produced by these vendors for use in FFTF require prooftesting before installation in the reactor. Prooftesting would only require tests for accuracy, sensitivity, and related parameters in a facility capable of varying only temperature and pressure.
- This assumes that the corrosion effects and radiation effects on the transducer have been determined and compensated for during development.

Additional design testing may be required.

Operation of a developed prototype unit in its designated location in a fuel channel could be affected by other parameters not present during development testing or prooftesting:

- Locating the signal lead wires in a conductor with other power or signal leads may cause error signals to be induced in the pressure signal lines.
- The particular flow and turbulence pattern of the coolant across the transducer face, and vibration may cause error or transient signals to be generated.

- Similarly, the placement of the transducer may introduce error in the readouts of other types of instrument sensors.

For these reasons, it is desirable to place a prototype transducer in the complete configuration of instruments in a prototype fuel channel. These instruments should then be tested to the temperature, pressure, and flow in the presence of sodium coolant simulating FFTF conditions, as nearly as possible, to proof-test the complete instrument package.

Test Facilities

Pressure transducers will be tested in an existing sodium pot facility at PNL. The facility consists of four pots, approximately 2 ft high and 1 ft wide, with a removable cover on each pot. Sodium with controllable level and temperature to 1400 °F may be introduced into each pot separately from a reservoir. The cover gas pressure may be adjusted to 250 psi. Small differential pressures may be obtained with a standpipe or by use of two separate pots with different pressure settings determined by commercial differential pressure instrumentation.

The following describes a suitable flowing sodium test loop simulating the FFTF:

- Material: all 304 SS on both hot and cold sides
- Duplicate the flow of the FFTF fuel channel for the diameter of the instrument under test (Approximately 15 ft/sec)
- Four test pots in hot leg and one test pot in cold leg
- Loop pressure to 250 psi
- Temperature: to 1400 °F
- Include hot carbon, hot oxygen, cold traps and plugging meter to insure sodium purity as near FFTF as possible.

Anticipated test facilities for liquid level measurement development will consist of a vertical tank of flowing sodium

approximately 30 ft high and 12 in. in diameter. The flow rate may be lower than in the FFTF vessel, but the sodium should not be stagnant. Cover gas type, pressure and flow should be similar to FFTF. Sodium purity will be similar to FFTF. The facility should include sodium level control over the vertical range of the tank and a secondary standard for making a reference sodium level measurement. The vertical tank shall have access to the top for introducing level sensors undergoing testing.

Alternatives to Performing These Experiments

No Testing Program

Without tests and development to withstand high temperature and radiation, the chances of a commercial pressure transducer surviving in the FFTF in-vessel environment are very small, and the degree of confidence in the accuracy of a pressure readout signal is unacceptable.

Modifying Commercial Units, But Without Testing

Materials in commercial units may be investigated and replaced for high temperature or radiation tolerance. However, the effects on the unit's performance could not be quantitatively assessed, and the confidence in the pressure signal of an in-vessel unit becomes very small even though the unit may survive in the environment.

Tests to Temperature and Pressure Without a Sodium Environment

This alternative does not test the effect of sodium corrosion on the thin diaphragm of the instrument. Changes in the properties of strength, ductility, and creep rate could only be estimated. A unit so tested might survive in the FFTF in-core environment with some initial accuracy. However, confidence in the unit would decrease as the length of time the unit was immersed in sodium increased.

Entire Prototype Assembly Not Tested in Radiation Environment

Due to unavailability of test thimble access in fast test reactors such as EBR-II or due to size restrictions, the entire assembly may not be tested in a radiation environment. If each component material is examined and selected for radiation tolerance separately, with irradiation tests of individual components performed as required, then the confidence in the assembly made up of these components remains high.

Tests in Static Sodium Pots and Radiation But Not in a Flowing Sodium Facility

Pressure transducers tested in static sodium without flow, ΔT or sustained purity, will not necessarily experience corrosion rates identical to those in FFTF. The result will be a transducer with proven operation in high temperature and radiation, but with a corrosion rate interpolated from static corrosion data. Confidence in the pressure signal is thereby reduced as the period of use in the FFTF increases.

However, confidence in this unit is greater than in the unit which is not tested in sodium at all.

IN-CORE SODIUM FLOWMETERS (I-3)

Design Problem and Requirements

A discussion of the need for individual channel flowmeters is shown in BNWL-555, Technical Basis for FFTF Driver Fuel Instrumentation.

The objective of this task is to develop and test magnetic and eddy current type flowmeters which can be used to measure the flow in individual FTR channels. Other types will be considered if they show sufficient promise.

Flowmeter Requirements:

- 1) The time response of the flowmeter must be rapid enough to allow for the timely protection of the reactor core in the event of a rapid flow reduction.
- 2) The flowmeter shall have sufficient accuracy and sensitivity to determine:
 - Significant changes in the hydraulic characteristics of the fuel channel assembly
 - Heat generation rate information of usable quality when combined with coolant temperature data
 - Significant differences in heat generation rates between fuel channels caused by such things as fuel element loading errors.
- 3) The flowmeter shall be designed to operate reliably in the defined FFTF environments for a minimum time interval of one fuel cycle.
- 4) Measure flow of about 450 gal/min at temperatures to 1400 °F while interfaced with sodium.

An additional design problem is the effect on calibration of magnetic flowmeters when they are operated in close proximity as in the FFTF channels.

Experimental Program

Basic testing of magnetic materials and coil structures will be made in order to determine their performance in the predicted FFTF environment. Depending on the results of the material tests, one or more of the flowmeter types will be developed for FFTF use.

The following represents a brief itemized summary of the proposed work:

- 1) Determine experimentally the significant changes in the properties of several selected magnetic materials as the result of temperature, thermal and mechanical shock, and

radiation. The magnets will be placed in a mockup of a hot cell which includes a furnace. The magnets will be temperature cycled and the magnetic flux will be measured with a magnetometer. The magnets will then be irradiated in a capsule in EBR-II. After irradiation, the magnets will be placed in a hot cell and magnetic flux again measured under conditions identical to the preirradiation tests.

- 2) High temperature coils will be developed and tested for use in turbine and eddy current flowmeters. These coils will be temperature cycled in a gas furnace and later in a flowing sodium loop.
- 3) A prototype of each candidate flowmeter type will be fabricated and tested. ANL is now operating a prototype permanent magnet flowmeter in a flowing sodium test facility with limited flow capacity and temperature. The temperature capability of this facility is being increased to 1200 °F. Additional tests rated at about 450 gal/min will require a larger flow facility.

The eddy current flowmeter, a prime backup to the prototype, will be fabricated and tested. In these prototype tests, the following characteristics will be determined.

- Flowmeter sensitivity in sodium temperatures up to 1200 °F
 - Flowmeter impedance and field noise signal levels
 - Temperature compensation methods
 - Long term signal stability and reliability
 - Flowmeter accuracy and calibration.
- 4) Report the performance limits and capabilities which can be expected of an FFTF flowmeter and prepare the material specifications, drawings and fabrication information needed to procure commercial flowmeters for installation in the FFTF.

Test Facilities

Feasibility studies, conceptual design, and prototype fabrication and testing are being performed for permanent magnet and eddy current flowmeters. ANL is performing this work and providing the following development test facilities:

- 1) Sodium flow test facility with limited flow at 1200 °F.
- 2) Mockup which simulates the magnetic properties of the hot cell that will be used for postirradiation measurements. This mockup saves a great deal of hot cell cost during pre-irradiation testing.
- 3) Hot cell magnetic flux measurement facility with electric furnace where flux is measured with a magnetometer. This facility tests irradiated magnets for permanent magnet flowmeters.
- 4) Gas furnace facilities capable of 1400 °F.

A larger sodium flow facility, not now available at ANL, capable of about 450 gal/min flow, at 1200 °F, will be required to complete the development testing, and for proof testing of production flowmeters. Tests at 1400 °F will require a closed loop prototype.

Alternates to Performing Experiments

No Test Program

Without testing and calibration at high temperatures in sodium, and in radiation, the confidence in an in-vessel FFTF flowmeter signal will be very low. It is not possible to predict the performance of magnetic materials at temperatures higher than those for which adequate test data are available.

Eddy current flowmeters are still in a developmental stage. Mathematical means of predicting performance and optimizing design have not been developed to the extent necessary.

GAS DISENGAGEMENT FOR FAILED FUEL MONITORING (I-4)

Design Problem and Requirements

The fission gas monitor technique must be used if failed fuel detection and/or location is required when the reactor is not operating. Several reactors have used fission gas monitors of the cover gas with various degrees of success, but not as a rupture location technique. The major design problems with the fission gas monitor are insuring the efficient disengagement of a sample of released fission gas, accumulating it, and transporting it out of the reactor vessel for detection. The plugging of sample lines is a major problem.

The tentative requirements for the disengagement device are that it must fit into a 5 in. process channel along with coaxial cables and tubes, withstand 1200 °F, withstand the FFTF nuclear environment near the top shield, and work in liquid sodium.

It is required that experiments be performed early so the design of a failed fuel monitor will not delay reactor startup. Sufficient space needs to be available inside the driver fuel shroud, in the reactor building for shielding of detectors, and for routing of sample lines. The space required will be learned as a result of a successful testing program.

Experimental Program

Test Purpose and Methods

An experimental program will be performed with several different sampling and disengagement devices to determine which is most effective, which will work in a confined space, and how to trap the sodium vapor before entering the gas lines. The approach is to incorporate a gas injection unit and several gas disengagement units in a sodium loop at ANL. Known volumes of gas will be injected, and the disengagement effectiveness measured.

Various gas disengagement techniques will be tried such as, increasing surface area, gas sparging, temperature gradients, and a stilling tank.

Various techniques for getting the fission gas sample out of the reactor will be evaluated, such as the use of a continuously flowing carrier gas, the application of pressure on demand, and bringing the gas sample out of the reactor along with a sodium sample and disengaging outside the reactor vessel.

Range of Parameters

The range of parameters to be tested is not well enough known to be stated at this time.

Variables to be Measured

The major variable to be measured is the effectiveness of the disengagement device. Also measured will be the performance of the sodium vapor trap, the most effective size and location of sample line within the simulated subassembly, and the threshold of sensitivity of the system.

Information from Test to be Applied to Design

The data on performance of the various disengagement techniques will tell the designer which technique to use. The location and size of sample tubes will be known. The technique of reliably transporting the fission gas sample out of the reactor vessel will be known. Some operational data may be available.

Test Facilities Required

A small loop being modified at ANL will be used with sodium at the temperature, pressure and velocity expected in an FFTF subassembly channel.

A second facility will be required for design testing of FFTF prototype components. This facility has been described under the Instrumentation Subassembly discussion.

Alternates to Performing Experiments

The tests could be performed on the reactor at the expense of delaying the startup date and perhaps additional expense of developmental modifications shown to be necessary by the startup tests.

Some of the tests could be performed in water if the solubilities of various gases in sodium and water were known. In some cases, it would be easier to work with water as the flowing media, but the results, as applied to sodium, would be uncertain. The demonstration of the validity of the water experiments could be more difficult than the actual performance of work in sodium. Reactor operation could be delayed if modification were required after testing in sodium in the reactor.

DELAYED NEUTRON FUEL FAILURE MONITOR (I-5)

Design Problem and Requirements

Failed fuel monitoring is a two fold problem of detection and location. The detection of a few neutrons in the presence of a large gamma background (10^4 R/hr and up) and at elevated temperatures has not been reliably solved.

It is complicated further by the presence of tramp uranium in the sodium sample, high temperatures and the short decay time of the neutrons. A sodium sample must be obtained from each major fuel assembly to locate the fuel failure. There are problems of maintaining fixed time delay from core to detector, keeping the sodium temperature high to avoid plugging of sample lines, valving to reduce the number of detectors, and of getting the sodium sample out of the core through the vessel. The requirements are to develop a reliable operating system which shall detect and locate a failed fuel subassembly and operate in the thermal-nuclear environment of the FFTF sodium.

Experimental Program

Test Purpose and Methods

One major effort will be directed toward testing the present off-the-shelf detector and electronic components beyond the manufacturer's specifications rather than develop new devices. This effort will include measuring the response of neutron detectors at high temperatures and in high gamma fluxes, and the response of preamplifiers to their temperature and radiation limits. The response stated above includes sensitivities, stability, gamma pile-up rejection, signal to noise ratios, cable length effects on signal, unwanted signals due to gamma interaction with the cable, and the ability to operate after being subjected to excessive temperatures for a period of time (simulating the loss of cooling).

Analytical studies will be performed to determine the optimum thicknesses of thermal and gamma shielding, and the optimum size of the sodium sample as a function of specific activity of the sodium.

The piping and valving required to provide sodium samples to a minimum number of detectors consistent with acceptable sample transport time and minimum pump capacity will be tested. The minimum acceptable sample line size, and the effects of intermittent flow will be determined.

Range of Parameters

The final values must reflect the FFTF design. However, until that is finalized, the following is being used as a guide:

Neutron flux	1-10 n/cm ² /sec	At 250 keV
Gamma flux	10 ⁴ to 10 ⁶ R/hr	
Preamplifier environment	200 to 400 °F	
Sodium sample temperature	600 to 1000 °F	

Variables to be Measured

The variables to be measured will include but not be limited to:

- 1) Detector neutron sensitivity as a function of gamma background at several temperatures
- 2) Pulse height stability versus temperature
- 3) Noise pickup as a function of cable type and length
- 4) Lifetime of electronic components in the FFTF environment.

Information from Test to be Applied to Design

Design will receive data on the optimum size, shape, and materials of shielding and moderator, the sodium sample size, the choices between detector and preamplifier combinations, and the effects of temperature on components.

Test Facilities Required

The following facilities are required:

- 1) A test facility is needed to determine the tendency of sodium sample lines to plug as functions of line size, sodium purity, line temperature and temperature gradients, flow rate, flow duty cycle, and related parameters.
- 2) A gamma facility is needed where one can expose a heated neutron detector to a small neutron source in the presence of intense gamma fluxes.
- 3) A facility is needed to test the design of a sampling valve which:
 - Can select one of several sample lines
 - Be remotely operated
 - Operate at temperatures up to 1000 °F
 - Be compatible with sodium
 - Be radiation resistant
 - Have no leakage to the outside
- 4) A facility to show proof-of-design and to check any interactions of one system with another is described under the Instrumentation Subassembly discussion.

Alternates to Performing Experiments

A system can be designed without further inputs from testing programs. The probability of its being adequate is small, and the cost of a new design and the replacement of components in relatively inaccessible locations would be great.

INSTRUMENTATION SUBASSEMBLY (I-6)

Design Problem and Requirements

The instrumented section of the driver fuel channel may include thermocouples, a flowmeter, a flux monitor thimble, pressure transducers, straightening vanes in the stream below the flowmeter, fission gas disengagement section, and sodium sample lines.

All these items will be developed and tested in separate programs. Additional testing of the assembled group of instruments is required to assure their compatible operation. Needed information includes:

- 1) Effect of wiring tubes and flux monitor tubing on the sodium flow straightening section below the flowmeter
- 2) Effect on flowmeter calibration
- 3) Adequacy of assembly to withstand the vibration, temperature and flow conditions typical of those expected in the reactor vessel
- 4) Assurance that the sodium sample lines will acquire an adequate sample of sodium plus fission products with other instrument obstructions in the channel.
- 5) The coolant pressure drop across the assembled instrument should be measured. This affects flow and thus the heat removal efficiency.
- 6) Testing is needed to determine whether the close proximity of wire routing in the fuel channel will cause signal crosstalk and whether there are grounding or shielding problems.

After reactor startup, the design of future instrument assemblies or experiment assemblies should also be tested for operation in simulated reactor conditions before being placed in the reactor.

Experimental Program

Assemble the instruments above a fuel section mockup in a fuel channel prototype which is installed in a flowing sodium test loop to simulate reactor temperature and flow.

- 1) Introduce a ΔT between simulated fuel pins and read the difference between in-channel thermocouples to determine the efficiency of coolant mixing.
- 2) Adjust the distance of the thermocouples above the simulated fuel pins to study thermocouple response and determine optimum thermocouple locations.
- 3) Introduce partial and full blockage of the coolant in the fuel zone as functions and determine flowmeter, thermocouple, and pressure instrumentation response.
- 4) Introduce inert gas for test purposes from simulated fuel pins, then operate the fission gas disengagement system to test the collection, separation, and detection of the gas.
- 5) Instrument the tube to measure pressure drop across the instrument package.
- 6) Operate the flux monitor traveling probe through the thimble at 1200 °F to determine any adverse affects due to thermal expansion, sticking at high temperature, or sodium leakage.
- 7) Vary the liquid sodium level above the channel outlet by ± 5 ft to observe the effects on flow. This capability may also be used to test liquid level instrumentation.
- 8) Disassemble the instrumented duct after operational testing and inspect for corrosion points, fatigue, sodium leakage or other design deficiencies.

Data from the above tests will be used by the design group to correct the instrument assembly designs as required.

Test Facilities Required

A flowing sodium test facility is required to simulate as near FFTF conditions as possible. The loop should include a test section ~12 in. diameter and high enough for a full size fuel channel with access from the top. Sodium flow will be about 450 gal/min through the fuel channel, at 1200 °F.

The loop should be capable of producing 120 psi coolant pressure at about 450 gal/min at the bottom of the test section for entry to the prototype fuel assembly.

The loop should be constructed of all 304 SS in contact with sodium, and should include a sodium purification system similar to FFTF. Secondary standard instrumentation for temperature, flow and pressure measurements should be included.

The test section should have a controlled sodium level to correspond with FFTF reactor sodium level and cover gas pressure (~25 psig).

Alternates

- 1) Each in-core instrument is developed and tested individually. Without testing of the assembled instruments as described, confidence in the instruments operation in-reactor is reduced. Unknown error due to instrument interaction in the final assembly may be introduced.
- 2) Some of these tests could be performed less adequately in a water facility.

VIBRATION AND STRESS MEASUREMENTS IN-CORE (I-7)

Design Problem and Requirements

Development of sensors such as strain gages, accelerometers, and Linear Variable Differential Transformer (LVDT) type instruments, to be used to measure vibration, stress, movement or displacement of in-vessel elements

is needed. The major development problems are to improve high temperature tolerance, radiation tolerance, and tolerance to immersion in sodium in order that the sensors may survive for a useful period in the vessel.

Sensors for use in a vessel surveillance program are also needed. This broad program for FFTF calls for the monitor of the vessel, in-vessel components, primary piping, heat exchangers, and similar components for such parameters as cracks, wall thickness, vibration, stress and strain, visual observation, temperature expansion, and corrosion rates as required by metallurgy, fuels, and sodium chemistry groups.

Both short term surveillance (instrumentation in fuel channels which can be removed and replaced), and long term surveillance using permanently mounted transducer systems are planned. Nondestructive testing techniques using conventional eddy current devices, ultrasonics, acoustic emission, and noise analysis techniques will be studied for application to the FFTF vessel. It is presently planned to develop a suitable surveillance program for the FFTF vessel. It is also planned to make base reference records of structures and metals characteristics to be placed on file for comparison at some future time. Methods of remote operation and remote replacement of instruments may also be developed as required.

Experimental Program

An LMFBR sponsored program for development of high temperature strain gages is underway at the LMEC. Gages have been developed which can operate in excess of 1000 °F. Methods of encapsulating the gages in stainless steel so they can be immersed in sodium are also being developed. At some future time, the gages will be developed for use in a radiation environment. This last effort should be moved ahead to aid in the FFTF application.

Studies are in progress to establish the surveillance requirements for the FFTF. Concurrently, the various instrumentation methods will be studied and proposed along with remote operation and replacement methods. The approved instrumentation prototypes should be assembled and tested in as near FFTF operating conditions as possible. The following are possibilities which should be considered:

- 1) Operate the CCTL to yield valuable information on ability to establish analytical structural models and to determine the state-of-the-art capability of strain sensors to withstand long time environmental conditions at or near 1200 °F. In its present status, the CCTL at ANL has been structurally characterized by analytical models with correlative instrumentation for strain and displacement affixed to the loop piping and components.
- 2) Construct and operate a full-scale model of an ultrasonic probe mounted in a wheeled carriage to traverse completely around the outside of a test vessel to measure wall thickness, cracks and defects. The carriage can also traverse vertically to inspect the entire outer surface of the vessel. This machine is remotely operated.
- 3) Mount vibration sensors on studs extending from the test vessel sides, on the nozzles of the fuel channels, or on internal members, for vibration and noise analysis. Stress the vessel or members by filling with sodium or otherwise simulating stress, and record the noise level and burst of noise.
- 4) Place accelerometers or suitable noise sensors at predetermined points on the outside of the vessel. Then stress the test vessel, and locate faults in the vessel wall by triangulation methods. (Acrojet SWAT program)
- 5) Place strain gages and LVDT gages at required points in the vessel and on in-core members, fill the vessel with sodium, circulate at 1200 °F with flow at FFTF ratings, and measure strains and displacement as required.

- 6) Construct and test the prototype equipment for remote replacement of in-core, out-of-core, and primary piping sensors.

Test Facilities

For strain gage development, the LMEC has the capability for high temperature tolerance and encapsulation development. This work is part of the LMFBR program. LMEC does not do irradiation testing and some external facility such as EBR-II may be used.

The test facilities required for the vessel surveillance program depend upon the surveillance requirements and the chosen instrumentation. Developed instruments such as accelerometers or LVDTs could be tested to operating temperatures in sodium environment in a sodium pot facility or preferably in a flowing sodium test facility.

Larger systems such as ultrasonic monitoring of the entire vessel or triangulation to find fault points may require a full size vessel mockup for complete testing and development.

Full size mockups are also desirable for testing remote replacement or remote operating mechanisms in a sodium environment at high temperatures. This testing might be done on the reactor during the non-nuclear startup period.

Alternates

No Testing Prior to FFTF Installation

If only existing laboratory facilities are used to bench test the individual sensors for acceptable operation, then the effects of interface with high temperature sodium (if required) are unknown along with the effects of radiation. Sensors would be subject to initial exposure to high temperature, radiation,

and sodium for the first time in the FFTF. With regard to remotely operated equipment, if no prior testing is done, then the first prototype must be tested and developed on the FFTF vessel during construction, or later during operation.

Testing of Sensors to Rated High Temperature and in Sodium Environment

Sensors which are so tested environmentally may not perform correctly in their assigned application. Additional tests should be performed with the sensor attached to the member it measures to assure the method of attachment, the method of remote operation, and the method of replacement. If prior testing is not done, then successful operation on the FFTF vessel is not assured, requiring testing and development on the FFTF vessel during construction or operation. For example, vibration pickups could be tested individually but the application of the signals to triangulation for locating a fault on the vessel could only be thoroughly tested on an FFTF vessel in conjunction with an appropriate computer program and computer.

SIGNAL CONNECTORS (I-8)

Design Problem and Requirements

A nozzle connector is required to transmit instrumentation information from sensors in the fuel channel to readout devices outside the reactor. One or more connectors will carry ~35 to 40 electrical lines, two to four gas pressure lines and one flux monitor thimble.

The connector requirements are:

- 1) Capable of being remotely separated and joined by the fuel handling machine which may be sodium cooled.
- 2) A flux monitor thimble of large radius will be coupled into the fuel channel making a 90° bend at the connector.

- 3) The design shall fit the space on the top reactor cover at 8 to 10 in. center-to-center spacing between connectors.
- 4) The electrical connectors with associated wiring termination must be reliable enough to carry reactor safety circuits at 1000 °F and maintain an insulation resistance of at least 5 to 10 MΩ.
- 5) The connector may be required to carry a sodium sample line with no possibility of sodium leakage into the electrical contacts or the gas line.
- 6) Reliable tubing connections will be made without leakage.

A second connector may be used in the center of the fuel channel to shorten the assembly for interior examination. This connector would have similar signal carrying and reliability requirements and in addition must maintain a seal while operating under sodium of 1200 °F near the active zone.

Test Program

An approved prototype of the nozzle connector will be built and tested. Development testing will include tests to prove the remote disconnect method using the fuel handling machine, gas pressure tests to determine the adequacy of the gaskets, electrical insulation tests and introduction of flux monitoring instrumentation. If a sodium sample line passes through the connector, the connector should be tested with a prototypical sodium sample removal system, similar to that planned for the FFTF. The sample line should be heated as required with sodium flowing at the temperature and speed of the final reactor delayed neutron fuel failure detection design. The connector should be mechanically cycled thoroughly in the test to prove the technique of remote operation with the fuel handling machine and to prove the connector operation after the wear of cycling.

The mid-channel connector will undergo similar fabrication and testing at ANL. Additional tests will include immersing the connector in a sodium pot of 1200 °F, temperature cycling the connector, and mechanical loading to test the effects of tension compression and bending.

Data from the above tests will be made available to design.

Test Facilities

The nozzle connector prototype will require a mockup of the vessel top nozzle with a simulated fuel handling machine to test all mechanical compatibilities. Instrumentation for measuring electrical conductivity and insulation and pin contact resistances is required along with a method of pressurizing and monitoring for leaks.

The flux monitoring thimble should be tested by introducing the flux monitor wire or aeroballs to assure that the mechanical entry is adequate.

Prototype testing of the mid-channel connector requires a sodium pot which can be sealed, filled and cycled to 1200 °F and which incorporates a mechanical loading apparatus for tension, compression and bending tests. ANL presently has these pot facilities or will provide them as part of their performance of Task IC-18.

Alternatives to Performing These Experiments

No Prototype Testing

One alternative to adequate development testing of connector prototypes is accepting the risk of maloperation during and after reactor construction. Confidence in the connectors performing the required tasks during operation without extensive developmental testing would be low.

Testing Only During Construction

If testing is deferred until the reactor is being constructed then deficiencies discovered in the design may cause major changes in equipment already built causing expensive modifications and associated delays in construction.

REACTOR ENVIRONMENTAL EFFECTS ON SIGNAL CABLES (I-9)

Design Problem and Requirements

Technical bases are needed for the specification of signal cables whose response in the FFTF environment has been determined. The magnitude of interfering signals injected into cables by radiation must be determined and methods developed to reduce or reject these false signals. The polarity of the induced current depends on whether the radiation is generated inside or outside the cable.

The insulation resistance of a cable decreases by six to eight decades at the high temperatures and high nuclear radiation levels in FFTF. This shunting effect decreases the output signal as transmitted by the cable. The reduced signal shows up as reduced accuracy, lower signal to noise ratio, and in a shifted power spectrum. It requires more circuitry to process the already low voltage signals. The environment may be as high as 10^9 R/hr and 1400 °F. The signal level of most sensors is in the mV range.

Experimental Program

Test Purpose and Methods

Theoretical calculations will be made on the expected reduction in insulation resistance due to both temperature and radiation.

The amount of induced current as a function of cable length and intensity of radiation will be measured to confirm the theoretical calculations.

Studies and experiments will be conducted on electronic processing to minimize the error introduced by the cable in a thermal and nuclear environment.

Range of Parameters

The radiation levels will be as follows:

- 1) Up to 10^9 R/hr
- 2) Neutrons up to 10^{16} n/cm²-sec fast
- 3) High energy electrons generated both inside and outside the cable due to nuclear interaction of the neutron and gamma flux on materials in the reactor.

The temperature measurements will be made up to 1400 °F.

Variables to be Measured

Insulation resistance of several mineral oxide insulators as a function of combined temperature and radiation will be measured. The effect of wire and cable size on the magnitude of inducted current generation will be measured.

Information from Test to be Applied to Design

The cable size and material and types of circuitry necessary to process the signals will be given to the Design Section so that the system design can be finalized.

Test Facilities Required

A gamma field of up to 10^9 R/hr is required. The gamma facility should be capable of irradiating a heated wire. Neutron flux of up to 10^{16} n/cm²-sec is also required as well as a source of electrons. All the needed irradiation facilities exist, including a pulsed reactor with adequate flux intensities.

Alternates to Performing Experiments

No alternate to testing is known.

SODIUM FLOW MEASUREMENTS IN LARGE PIPES (I-10)

Design Problem and Requirements

The purpose of the development and procurement program is to produce large sodium flowmeters of 16 to 18 in. diam pipe size for flow measurements from 800 to 16,000 gal/min suitable for installation and operation in FFTF primary and secondary coolant loops. The sodium temperature range is expected to be 350 to 1000 °F with surrounding ambient air temperature of 90 to 120 °F.

The flowmeters will probably be of the permanent magnet type, subject to review of design requirements in light of the flowmeter state-of-the-art report to be issued by LMEC early in 1968. Existing technology will be used with little development required.

A major problem is the calibration of the units. Flowmeter accuracy of $\pm 1\%$ is desired as a design goal; however, $\pm 5\%$ will more probably be realized.

Experimental Program

Standard design flowmeters will be fabricated to FFTF specifications and tested using sodium in a facility large enough to accommodate the pipe diameter and flow.

Where the FFTF primary or secondary loop design requires an elbow, valve, tee or discontinuity to be installed in the sodium pipe ahead of the flowmeter, the entire questionable section of pipe will be duplicated in the test facility ahead of the flowmeter in order to calibrate the flowmeter at prototypical operating conditions.

The accuracy, repeatability and rangeability of the magnetic flow system shall be tested against a flow tube differential producer or other suitable secondary standard connected in series.

The repeatability of the magnetic flowmeter, compared with the flow tube sensor, shall be $\pm 1\%$ of full scale over the range of 800 to 16,000 gal/min.

Each flowmeter shall be checked at a minimum of six flow rates (normally 20,40,60,80,100, and 120% of full scale flow).

Each magnetic flow element shall be thermally cycled a minimum of three times and checked for thermal effects. Records shall be made of sodium temperature and magnetic flux measurements. Calibration curves of flow versus output in millivolts shall be supplied for each test run. Data from the above tests will be made available to design.

Test Facilities

Due to the large size of the flowmeters, the only facilities which would be suitable are the proposed Auxiliary Component Test Installation (ACTI) or the Sodium Pump Test Facility (SPTF) at the Liquid Metal Engineering Center.

Alternates

No Tests

Without calibration curves, thermal shock, and repeatability tests, reliance on calculations of the predicted flowmeter would be necessary, with further refinement by indirect measurements through observations of the overall heat transport system.

Tests Without Pipe Discontinuities

In this case, the flowmeters might have been correctly calibrated using sodium, but might give erroneous readings as installed in the FFTF, due to flow discontinuities.

Sample Testing

After initial prototype flowmeters are tested, subsequent duplicate flowmeters might be installed without proof testing. However, only eight flowmeters are planned for FFTF. The

investment to modify or build a test facility for these flowmeters is better justified if all eight are calibrated. Calibration of each individual flowmeter will also compensate for any manufacturing differences.

IN-CORE NEUTRON FLUX MONITORING (I-11)

Design Problem and Requirements

Knowledge of the spatial and energy distributions of the neutron flux within the core is necessary for the efficient operation of the reactor and for the evaluation of test data taken from in-reactor test loops. A monitoring system is required that will permit scanning the core in a sufficient number of positions to adequately characterize the core. Two primary problems are the provision of thimbles to guide the sensors being inserted and withdrawn during operation, and the provision of a reliable transporting mechanism for the sensors. The thimbles must maintain a seal so the sodium cannot leak out of the vessel, and the vessel penetration design may require a remotely operated connector. Sensor transport tubes that lead to the in-vessel thimbles may have to be removable from the reactor top works by remote means during reactor charge-discharge operations. The sensor transport mechanism must operate at a distance from the reactor vessel, moving the sensors through a guide tube that has whatever bends and joints are dictated by the building structure. The difficulty of transporting sensors reliably in this sort of device has been shown to be dependent upon several factors such as the length of the tube, the number and radii of the bends, the length of vertical runs, the tube and cable materials, and the operating temperature. Means must be provided for remote handling of activation sensors after irradiation and transporting them to a counting facility.

Experimental Program

This program is based on the knowledge that the actual neutron sensors have been or are being developed by other programs. Activational techniques have been used with a variety of sensing materials in the form of wires, foils, and balls, and miniature chambers have been successfully operated in an environment approximating that expected in the FFTF at a fraction of full power. The remaining problems being treated in this discussion are mainly mechanical in nature.

The conceptual arrangement of sample tubes, connectors, thimbles, and related essential parts that is formulated in preliminary design will be mocked up in a test facility to demonstrate the system feasibility, and thus to provide an experimental base for the improvement of the system during detailed design. This mockup shall simulate all the difficult aspects of the problem that can be anticipated, including temperature effects, material compatibilities, and connections capable of being handled remotely. Life testing shall be performed to determine the adequacy of the design, and to test the solutions proposed to correct the failures observed.

The detailed designer will be given the results of this testing, including dimensions of critical components, materials used, drawings, and information needed for the specification of a system for operational use.

Test Facilities Required

A mockup of the proposed monitoring system is required that simulates the true size and shape of at least one channel of the proposed mapping system. The in-vessel thimble will be heated to simulate the reactor environment. A high-bay laboratory will be required to accommodate the vertical runs, which will be on the order of 40 ft in height. It is anticipated that this work could be done by a vendor experienced in this particular type of equipment, such as Westinghouse or General Electric.

Alternates To Performing Experiments

It is possible to design such a system as an extrapolation of past work and install it without having demonstrated that it will actually function reliably. This alternate is considered to be very unwise because of the difficulty in predicting the performance of unusual mechanical systems of this order, and the difficulty of replacing components built into the reactor or building structure.

DEVELOPMENT FACILITIES

The foregoing chapters presented many of the reactor system problems requiring experimental solutions. The general characteristics of the test facilities which are required to obtain the experimental data were identified in terms of the coolant or environmental needs, static or flow conditions, temperature, pressures, and special test operation methods.

In Tables 10.I and 10.II, specific problems are identified and the basic characteristics of the facilities which are needed have been tabulated. In some cases, the problems require several different types of testing and test facilities. For each problem, the many different test facility requirements were identified. Test facilities are identified which are needed to perform the required testing. The specific considerations of the facilities are provided in the following sections.

A. THERMAL HYDRAULIC TEST FACILITIES

The Thermal Hydraulic development program is formulated on the basis of partial subassembly testing of 1, 7, and 37 simulated full size, fuel pins in sodium. The flow requirements are about 10, 20, and 250 gal/min respectively. (Each fuel pin bundle does not necessarily correspond in 1/1 relation of number of pins to flow rate, since the objectives in the use of different sizes are not necessarily coincident.) Generally, the basic loop requirements are similar to many "conventional" sodium loops except that a very large electrical power source (up to 6 MW) with matching heat dump is required to permit the simulation of nuclear heat generation. PNL is developing a simulated heated fuel pin with the objective of eventually producing a workable multi-pin, partial assembly heater which would be needed for the thermal-hydraulic development program. The 1 and 7 pin tests can be performed

TABLE 10.1. Required Dynamic Facilities

Component or Assembly	Key	Problem	Fluid	Flow gal/min	Temp., °F	Pressure, psig	Special Features	Recommended Facilities	
								Existing	New
Fuel subassembly	A-1	Fuel clad temperature limit	Sodium	20	1500	200	Electrically heated simulated 7 pin fuel bundle, 200 kW electrical power	PNL Heat Transfer Loop 1	
			Sodium	250	1500	200	Electrically heated simulated 37 pin fuel bundle, 6 MW electrical power		Large Heat Transfer Loop
	A-2	Local coolant voiding and propagation	Sodium	20	1500	200	Electrically heated simulated 7 pin fuel bundle, 200 kW electrical power	PNL Heat Transfer Loop 1	
			Sodium	250	1500	200	Electrically heated simulated 37 pin fuel bundle, 6 MW electrical power		Large Heat Transfer Loop
	A-3	Loss of coolant and power excursion	Sodium	250	150	200	Electrically heated simulated 37 pin fuel bundle, 6 MW electrical power		Large Heat Transfer Loop
	A-5	Critical heat flux (proof test)	Sodium	250	150	200	(as above)		(as above)
	A-4	Pressure drop and cross channeling	Water	500	200	200	Hydraulic facility plus mixing analysis equipment	PNL 189D Hydraulic Loop	
	B-1	Cladding erosion	Sodium	20	1100	---		PNL Small Components Evaluation Loop	
			Sodium	500	1100			ANL Core Components Test Loop (CCTL)	
	B-2	Cladding corrosion	Sodium	20	1100	---		PNL Small Components Evaluation Loop	
			Sodium	500	1100	---		ANL Core Components Test Loop (CCTL)	
	B-3	Fretting corrosion	Water	500	200		Vibration measurements	PNL 189D Hydraulic Loop	
			Sodium	20	1100	---		PNL Small Components Evaluation Loop	
			Sodium	20	1100	---	Vibration measurements heated bundle	PNL Heat Transfer Loop 1	
			Sodium	250	1200	---	Vibration measurements heated bundle		Large Heat Transfer Loop
			Sodium	500	1200	---		ANL Core Components Test Loop (CCTL)	
	B-4	Vibration/wear	Water	500	200		Vibration measurements subassembly duct	PNL 189D Hydraulic Loop	
			Sodium	500	1200	---	Vibration measurements subassembly duct	ANL CCTL	
	B-5	Thermal shock	Sodium	---	1200	---	Thermal shock 700 to 1200 °F	SEFOR (GE APO)	

TABLE 10.1. (cont'd)

Component or Assembly	Key	Problem	Fluid	Flow, gal/min	Temp., °F	Pressure, psig	Special Features	Recommended Existing	Facilities New
Inlet Plenum	C-1	Characterize hydraulic balance.	Water	250	Room		Single tube hydraulic balance model.	PNL 314 Hydraulic Test Stand	
	C-1	Characterize inlet plenum flow behavior.	Water	5000	200		Inlet plenum feature models, a 1/4 scale.		Hydraulic Core Mockup
Core	C-2	Characterize core and reflector flow distribution	Water	5000	200		Core and bypass feature model, a 1/4 scale.		Hydraulic Core Mockup
Upper Plenum	C-1	Characterize outlet mixing	Water	5000	200		Upper plenum and mixing device feature models.		Hydraulic Core Mockup
Vessel	C-1	Vessel flow distribution	Water	15,000	250	200			
	C-2	Bypass flows	Water	15,000	250	200			
	C-1	Upper plenum mixing	Water	15,000	250	200			
	C-3	Thermal gradients	Water	15,000	250	200	Reduced scale (1/2) core and vessel model.		Hydraulic Core Mockup
	D-1	Core vibration	Water	12,500	200	200	Full vessel, 1/4 sector of ducts		Hydraulic Core Mockup
	C-4	Loss of coolant							
	(A-3)	Single assembly heated	Sodium	250	1500	200	Heated subassembly bundle, 6 MW		Large Heat Transfer Loop
	C-4	Single full assembly fission gas plenum	Water	500	200		Full scale subassembly	PNL 189D Hydraulic Loop	
	C-4	System loss of coolant	Water	15,000	250		Scale model		Hydraulic Core Mockup
	C-5	Cavitation single component	Water	500	250		Subassembly	PNL 189D Hydraulic Loop	
			Sodium	500	1200			ANL CCTL	
			Sodium	12,000	1200		Prototype IHX pump	LMEC ACTI	LMEC SPTF or ACTI
		System	Water	15,000	250		Scale model core		Hydraulic Core Mockup
	C-6	Gas entrainment							
		Single component	Sodium	12,000	1200		Prototype IHX pump	LMEC ACTI	LMEC SPTF or ACTI
			Water	15,000	250		Scale model core		Hydraulic Core Mockup
		System							
	C-7	Solids deposition	Water	15,000	250		Scale model core, simulated solids		Hydraulic Core Mockup
	C-8	Gas or steam closed loop rupture	Water	15,000	250		Scale model core, high pressure air injection		Hydraulic Core Mockup
Flow Ducts	E-3	Proofterest flow duct design and characterize response to temperature and sodium flow.	Sodium	500	1200		Vertical section for reactor flow duct		FFTF channel size sodium loop, with channel mockup
Seven-Tube Module	F-5	Confirm adequacy of design to prevent flutter	Water	500	Ambient		Simulated envelope of seven ducts, center shroud capable of inserting and removing, vessel model.	PNL 189 Hydraulic Lab Pumping System	

10.3

BNWL-614

TABLE 10.I. (contd)

Component or Assembly	Key	Problem	Fluid	Flow gal/min	Temp., °F	Pressure, psig	Special Features	Recommended Facilities	
								Existing	New
Control and Safety rods	G-5	Flow induced vibration in poison section	Water	to 100	250	50	Vertical test section	PNL EDEL 1 loop	---
	G-6	Minimize stress levels introduced on the control element and rod guide tube during scram cycle.	Water	to 50	250	50	Vertical test section	PNL EDEL 1 loop	---
	G-7	Verify operation, input power, output force drive rate. (Vendor acceptance tests)	Water	to 50	250	50			Vendor
	G-7	Life test (test stand)	Sodium	to 50	1200		Vertical test section	LMEC Control Rod Test Tower	
	G-8	Product assurance (acceptance)	Water	to 50	250	50	Vertical test section		Vendor
Closed Loop In-Reactor Hardware	H-2	Closed loop in-reactor hardware i.e., insulated pressure tubes and internals, which involves selection of material, fabrication methods and other design parameters dictated by the FFTF Environment. (pressure, temperature, vibration, thermal shock)	Water	100-700	250			PNL EDEL-1 loop	
			Sodium	100-700	1350	250			Closed Loop Mockup
	H-3	Develop reliable interconnecting piping	Sodium	100-700	1350	250			Closed Loop Mockup
Closed Loop Components External to Reactor Vessel	H-4	Develop reliable valves, pumps, heat exchangers	Water	100-700	ambient	ambient			Closed Loop Out-of-Reacto Components
			Sodium	100-700	1350	250			Closed Loop Mockup
Complete Closed Loop System	H-5	Compatibility and operability of individual closed loop components when integrated into a complete system to gain acceptance of complete closed loop	Sodium	100-700	1350	250	Fuel handling equipment and associated components. Sodium mockup interface		Closed Loop Prototype Mockup

TABLE 10.I. (contd)

Component or Assembly	Key	Problem	Fluid	Flow gal/min	Temp., °F	Pressure, psig	Special Features	Recommended Facilities	
								Existing	New
Instrument Pressure Transducer	I-2	Pressure transducer capable of measuring the pressure above and below core	Sodium	500	1400	250	304 SS, 20 ft/sec velocity, four test pots in hot leg, one test pot in cold leg, hot and cold traps		FFTF Channel Size Sodium Loop (An existing loop modified)
Sodium Flow Measurements in Large Pipe	I-10	Assure 16-18 in. flowmeter for 1000 °F operation, 800-16,000 gal/min	Sodium	800-16,000	1100	---	Thermal cycling		LMEC ACTI (planned) LMEC SPTP (design)
Gas Disengagement for Failed Fuel Monitor	I-4	Ensure efficient disengagement of sample from released fission gas	Sodium	low 500	---	---	Gas injection, Reactor top nozzle and channel mockup required	ANL	FFTF Subassembly Size Sodium Loop, with channel, nozzle mockup
In-reactor Coolant Temperature Sensors	I-1	Provide specifications for thermocouples capable of operating in FFTF environment	Sodium	10	1400	---	Thermal shock at 100 °F/min for 3 sec and 600 to 1000 °F in 6 sec	Several available	
Delayed Neutron Fuel Failure Monitor	I-5	Develop reliable operating system to detect and locate failed fuel for FFTF environment	Sodium	20	1200	---	Valve selection, several in-put lines	Several available	
			Sodium	450	1200	---	Reactor channel mockup		FFTF Channel Size Sodium Loop, with channel mockup
In-core Sodium Flowmeter	I-3	Develop magnetic and eddy current type flowmeter	Sodium	10 450 700	1200 1200 1400	---		ANL LMEC LCTL	Closed Loop Prototype
Instrumentation Subassembly	I-6	Assure compatible instrument assembly operation for in-channel location	Sodium	450	1200	---	Instrument package in test section 12 in. in diameter, channel mockup		FFTF Channel Size Sodium Loop, with channel mockup

TABLE 10.II. Required Static Facilities							Recommended Facilities	
Component or Assembly	Key	Problem	Fluid	Temp., °F	Pressure, psig	Special Feature	Existing	New
Fuel Subassembly	B-3	Fretting corrosion	Water	to 200	---	Air, electromagnetic shaker	PNL Dynamic Vibration Lab	
	B-4	Vibration	Gas	to 1000		Air, electromagnetic shaker fuel duct test stand, magnetic tape, electromagnetic shaker, fuel duct test stand, magnetic tape.	PNL Dynamic Vibration Lab with test section added	
			Water	to 200	---			
Flow Ducts	E-3	Stresses in geometric transition	Air	Ambient		Photoelastic equipment	Photoelastic Lab	
		Creep rate	Air	850		Burst chamber	314 Bldg. Burst Test Facility	
Core Structure	E-4	Confirm analytical model of the integral core structure	Gas	1000		Scale or full scale replica of tube sheet-ducts-cover vessel assembly		Hydraulic Core Mockup
		Determine linearity of core structure movements with uniform temperature	Gas	1000		Full scale replica of core structure		Hydraulic Core Mockup
Lower Tube Sheet	E-1	Establish analytical model for tube sheet deflection, stresses	Air	Ambient		Scale model of tube sheet, photoelastic lab		
Reactor Cover	E-2	Establish analytical model for reactor cover deflection stresses	Air	Ambient		Scale model of cover, photoelastic lab		
Driver Duct Top Face Hardware	E-7	Develop reliable top face hardware and seal	Gas	1200				Static pots
			Water	200				
			Sodium	1200				

10.6

BNWL-614

TABLE 10-II. (contd)

Component or Assembly	Key	Problem	Fluid	Temp., Pressure,		Special Features	Recommended Facilities	
				°F	psig		Existing	New
Reactor cover	F-1	Evaluate arrangement for assembly; fit, interferences, special tools required, etc.	Gas		Ambient	Complete reactor cover mockup.		Reactor cover mockup.
Partial Reactor Core Component	F-2	Evaluate arrangement for assembly fit, alignment, interference, removal and insertion and assembly sequence.	Gas		Ambient	Seven ducts		Seven ducts, dry assembly
			Gas		Ambient	Partial cover plenum, and about 20 shrouds.		Partial reactor assembly mock-up.
			Sodium	1200	---	Seven duct, sodium pot capable of heatup.		Seven duct environment, and mechanical mockup.
Contact surface	F-3	Determine adequacy of component design to resist self-welding.	Sodium	1200	---	Sodium pot, loading and force measuring mechanism contact location only.	AI Sodium mechanism	
	F-4	Determine adequacy of component design to resist galling.	Sodium	1200	---	Sodium pot for full length shrouds, supports covers, etc.		Seven duct environment, and mechanism mockup.
Vessel	E-5	Core blast analysis	Sodium	1200	---	1/10 scale vessel model TNT detonation		Hazards test cell
Vessel	E-6	Stress concentration in vessel nozzles and support reaction	Air		Ambient	Scale model of nozzle penetration support, photoelastic lab.		

10.7

BNWL-614

TABLE 10-II. (cont.)

Component or Assembly	Key	Problem	Fluid	Temp., Pressure,		Special Features	Recommended Facilities	
				°F	psig		Existing	New
Control and Safety Rods	G-1	Provide reliable drive mechanism	Gas			Vertical test stand, air, simulated rod weight	PNL Test Stand	
	G-2	Minimize scram mechanical time delay	Gas			Vertical test stand, air, simulated rod weight	PNL Test Stand	
	G-3	Seal drive mechanism from reactor environment	Sodium	1200		Vertical test stand, cover gas, simulated rod, simulated temperature gradient		
	G-4	Provide an initial acceleration greater than gravity for the safety control	Sodium	1200		Sodium pot, provision to apply static load up to 1000 lb		
	G-7 G-8	Verify mechanical fit, refueling maintenance, handling procedures	Sodium	1200		Sodium mockup with minimum six ducts surrounding rod position, simulated FHM		Seven Duct Module Sodium Mockup
Closed Loop Top Face Hardware	H-1	Top face hardware involves development of reliable seals and disconnect joints which will last at least five years.	Gas	1350	250	Vertical sodium pot with glove box above later complete closed loop		Closure Test Facility
			Water	200	250			
			Sodium	1350	250			
Closed Loop Interconnecting Piping and Disconnects	H-3	Closed loop interconnecting piping and disconnects involving: • Leak tight joints to contain liquid sodium. • Methods of removal without loss of coolant. • Containment of radioactivity in case of failure. • Detection of leaks in case of failure. • Recovery from sodium freeze up.	Gas	1350	250	Closed loop piping mockup		Closed Loop Piping Mockup
			Water	200	250			
Closed Loop Components External to Reactor Vessel	H-4	Develop reliable shut off and check valves for components external to reactor vessel, involving seals in components that will withstand the liquid sodium environment for at least five years.	Sodium	1350	250			Static Sodium Facility

TABLE 10-II. (contd)								
Component or Assembly	Key	Problem	Fluid	Temp., °F	Pressure, psig	Special Features	Recommended Facilities	
							Existing	New
Instrument Pressure Transducer	I-2	Pressure transducer capable of measuring the pressure above and below the core	Sodium	1400	250	Apply small differential pressures	PNL static pots	
			Sodium	1400	250	Vertical pot 30 ft x 12 in. for level		Seven Duct Sodium Environment and Mechanism Mockup
Signal Connector Mid-channel	I-8	Connectors for service in sodium at 1200 °F, carrying approximately 35 to 40 electrical lines, 2 to 4 gas pressure lines, one flux monitor thimble	Sodium	1200	---	Cycling, mechanical loading apparatus for tension, compression and bending	ANL pots	
Nozzle Connector	I-8	Connector for nozzle to transmit instrumentation information per above	Sodium	1200	---	Reactor nozzle, fuel handling function		Reactor nozzle simulated, fuel handling function
In-core Spatial Flux Distribution	I-11	Provide thimbles to guide sensors and reliable transporting mechanism	Gas	1200		Reactor nozzle, partial core mockup		Reactor nozzle simulated
			Sodium	1200	---	Reactor nozzle, partial core mockup		Reactor nozzle simulated, sodium
Vibration and Stress Measurements	I-7	Improved sensors for high temperatures tolerance, radiation tolerance, and tolerance to immersion in sodium	Sodium	1200	---	Ultrasonic probe mockup, full size sodium vessel mockup Accelerometer or LVDT pot.		Reactor vessel mockup
Signal Cable	I-9	Technical basis for cable specifications				Gamma facility		

by the use of the existing PNL Heat Transfer Loop No. 1, with replacement of the circulating pump required for high head availability. The second loop having the larger flow and the very large power requirements would have to be constructed. Current plans are to construct the sodium loop from AEC excess loop components such as the LASL Fast Reactor Core Test Facility (FRCTF). This assumes availability of the FRCTF for this use.

B. HYDRAULIC TEST FACILITIES

The hydraulic test requirements for the FFTF can be divided into the following classifications by sizes:

- 1) Partial subassembly
- 2) Subassembly
- 3) Reactor.

Each facility size provides specific and required information on the hydraulic character and behavior within the FFTF system. They provide the detail of each component and the integrated response of the system.

Partial and Fuel Subassembly Hydraulics

The program of water and sodium tests for fuel development is first concerned with seven pin (full size) assemblies to obtain early erosion, corrosion, and vibration wear data in sodium. The tests are presently being performed in the PNL Small Components Loop, which provides 25 gal/min sodium at temperatures up to 1500 °F. The sodium endurance tests will continue on a fuel subassembly using the ANL-Core Components Test Loop (CCTL). This program is also supported by water testing, first to determine pressure drop characteristics of candidate fuel subassemblies, and finally to characterize cross-channel mixing. These water tests on the full size fuel subassemblies are being performed on existing PNL hydraulic test loops. The same water loops, in conjunction with the pressure drop and cross-channel mixing evaluation, will be used to provide

flow-induced vibration information. These PNL loops to be used are tabulated in Table 11.I.

The fuel development program also includes thermal shock tests for fuel pins. The facility would be similar to that used for the SEFOR development program. Specific details of scheduling the required facilities for thermal shock testing are being established.

Based on the presently identified water and sodium tests for the fuel development, facilities presently exist to conduct the required testing and for the most part, arrangements have progressed to assure the availability of the facilities for the tests.

Core and Vessel Hydraulics

The hydraulic testing program for the reactor core and vessel requires mockups which represent the whole FTR primary system flow of approximately 40,000 gal/min. This flow rate is needed for such reactor system testing as core flow split, bypass flow, upper plenum mixing, gas entrainment, loss of coolant accidents, and solids deposition.

The best results from experiments would, of course, be obtained under conditions of complete ideality in which prototype components serve as the test model and the coolant environment is sodium at the expected operating conditions of the FTR. However, certain types of experiments are well suited to other techniques less costly and easier to operate. Use of water, for example, permits much simpler experimental procedures and more data accumulation than sodium. Fortunately, laws of modeling provide techniques by which the fluid and the test specimen scale can be modeled to provide a more appropriate means

TABLE 10.III. *Partial List of Reactor Model Study Water Flow Facilities⁽¹⁾*

<u>Site</u>	<u>Flow, gal/min</u>	<u>Pressure Drop, psi</u>	<u>Material</u>
Argonne National Laboratory	2,700*		
Atomics International	750		Stainless Steel
Babcock and Wilcox	4,000		Stainless Steel
Franklin Institute	2,500	110	Coated Carbon Steel
General Electric APED	1,700		Stainless Steel
Oak Ridge National Laboratory	18,000	100	Carbon Steel
Los Alamos	2,500		
Westinghouse	8,000	95	Stainless Steel
PNL	500	500	Stainless Steel
	250		Stainless Steel
FFTF Hydraulic Core Mockup**	15,000	200	Stainless Steel

1. K. D. Hayden. Survey of Hydraulic Modeling Experience for Reactor Core Systems, BNWL-518, Pacific Northwest Laboratory, Richland, Washington, August, 1967.

* Proposed system for FARET

** Proposed system for FFTF

for experimentation. The application of modeling to reactor core mockups of hydraulic and mechanical problems has been examined.⁽¹⁾

The study, conducted concurrently with this program evaluation, was addressed specifically to the major hydraulic and mechanical problems identified for the FTR. The study objectives were to resolve answers to the following questions:

- 1) What is the most feasible model fluid in each of these cases?
- 2) Is scaling down model length feasible and if so, what is the minimum practicable scale consistent with obtaining reliable, comprehensive model data?

Because of the direct application of the study results to the formulation of the concept of hydraulic and mechanical test facility criteria, specific conclusions from the study are as follows:

- 1) Water is the most feasible model fluid.
- 2) It is feasible to use a scale model, but the scale should not be less than ~ 0.2 to 0.5 . For steady-flow, a model scale of ~ 0.2 is feasible, however, for unsteady testing, the scale should be closer to ~ 0.5 . Determination of an absolute minimum scale for adequate modeling would require more specific details concerning the prototypic design than are available at the present time and would require a more detailed analysis than is presented in this study.
- 3) The model should consist of a full (360°) core as opposed to a core solid segment.
- 4) In the case of unsteady-flow simulation, the model should be geometrically similar in all respects which include

1. D. S. Trent. Applications of Geometric Models to FFTF Hydraulic Core Mockup, BNWL-575. Pacific Northwest Laboratory, Richland, Washington, October, 1967.

scaled tolerances where prototypic characteristic lengths are small. It is sufficient to make all model surfaces smooth.

- 5) In the case of steady-flow modeling at or near a pre-determined reactor flow, the fuel bundles may be simulated with orifice plates. The orifice plates serve a dual purpose in that they can be used as flow metering devices and at the same time eliminate the cost, time, and difficulty of fabricating redundant scaled fuel pins which are prototypically small. Orifice plate simulated fuel rod bundles are not valid for unsteady flow modeling.
- 6) Dynamic similarity is dictated by the Euler number and the Reynolds number. If in the final core design there is significant dynamic interaction between the sodium pool level and the core hydraulics, Froude number similarity must also be considered.
- 7) Ideal or true modeling is unnecessary in obtaining adequate model results; that is, it is not necessary to insist that $Re_r = Eu_r = 1.0$. In view of the relatively weak dependence of viscous energy losses on Reynolds number (i.e., $F = Re^{-0.2}$) it is appropriate to assume that Euler number similarity insures sufficient dynamic similarity for adequate modeling in all cases, except where Reynolds number ratios are small while the model flow is significantly viscous dependent.

Model tests or investigations which are feasible using the full core steady flow hydraulic studies model include:

- 1) Plenum flow distribution
- 2) Plenum pressure distribution
- 3) Plenum mixing
- 4) Bypass mixing
- 5) Gas entrainment (possibly only qualitatively)
- 6) Cavitation

Model tests or investigations which are feasible using only an exact geometry, transient flow, hydraulic studies model include:

- 1) Hydraulic hammer (possibly only qualitatively)
- 2) Partial or complete pumping failure
- 3) Local flow stoppages
- 4) Vessel rupture
- 5) Closed loop rupture.

A composite or complete vessel model is needed for the hydraulic evaluation of the core and vessel. Some of the presently identified tests are concerned with reactor safety requiring transient evaluations. Based on the foregoing information, a geometric scale of at least 1/2 is required to obtain the testing accuracy. This scale will of course satisfy the steady state testing requirements in the composite models. The model flow requirements for a 1/2 scale model of the 40,000 gal/min FTR flow based on constant Euler number for the reactor and the model is about 10,000 gal/min. Allowing a minimum of 125 to 150% of normal flow for the test flow range, a water supply of from 12,500 to 15,000 gal/min is needed to meet the needs of the core hydraulic testing program. Additional requirements for the test program and the facility are; pump head of 200 psi, capability to heat water up to 250 °F to simulate the kinematic viscosity of sodium in the range of 1000 to 1100 °F, and stainless steel construction to avoid auxiliary system corrosion problems.

No existing facility will meet the requirements for the core hydraulic testing. A summary of various loops available or used in past reactor model tests is given in Table 10.III. The Table is based on results of a recent survey.⁽¹⁾ None

1. K. D. Hayden. Survey of Hydraulic Modeling Experience for Reactor Core System, BNWL-518. Pacific Northwest Laboratory, Richland, Washington, August 1967.

of the existing water systems have the capability to meet the head and flow requirements of FTR. Oak Ridge National Laboratory has the required capability in a system used for the High Flux Isotope Reactor (HFIR), but the head is inadequate, and the piping is carbon steel. The proposed PNL Hydraulic Core Mockup would provide the needs of the FTR hydraulic study program.

In the program discussion, the use of feature sectional models as well as composite models was indicated. The purpose of the "feature" or sectional tests is to investigate feasibility of concepts of specific flow regions, e.g., inlet plenum geometry. These tests will investigate the gross effects of reactor arrangement and will be semi-quantitative in nature. The tests are intended to screen the various problem areas in the region. Such items as the effect of chamber configuration, anti-vortex baffles, and inlet pipe arrangement on pressure and velocity distribution can be evaluated in this manner. Simple techniques such as visual observation and dye injection will be used to provide rapid criteria of choice between several candidate inlet plenum designs. The intended use allows the model construction to be simple and less precise than the later confirming tests on more elaborate models. The models will probably be constructed from plastic. The results of these tests should provide preliminary guides to reactor vessel design.

The above models are not intended to substitute for the composite, geometric scale models. The feature models will definitely not provide the detailed hydraulic information required to support and verify the final design of the Fast Test Reactor. The question arises whether these feature sectional models can be designed and instrumented to provide the accurate hydraulic information required to support the reactor design and still result in overall savings in time and money. It does not appear likely. First, the functional purpose of the models will be circumvented since the model design must be more precise, and

instrumentation and fabrication effort would prevent these models from assisting the early phases of reactor design. Second, the hydraulic characterization of the in-vessel system is composed of many complex sections, in which integrated performance is not easily calculated. Basically, the problem is that the hydraulic behavior within the vessel is a highly complex process. The core flow split, for example, depends upon the characteristics of the inlet plenum, core, and outlet plenum. In turn, hydraulics characteristics of each of these may vary depending on the characteristic of the others.

The core hydraulic program needs sodium testing. The tests are to provide confirmation data for outlet mixing for operation of the core up to 1200 °F, mixing of core and parallel flow streams, gas entrainment, sodium oxide deposition, loss of coolant, and long term effects of hydraulic related phenomena such as cavitation. Sodium testing on a vessel system is considered necessary when the thermal or physical property of the sodium coolant would have a significant influence on the process behavior. For the hydraulics concerning flow distribution in the core and inlet plenum, sodium testing is not considered to be essential.

Although the need for large composite reactor system, sodium testing is indicated, a decision to remove the previously recommended sodium core mockup from developmental plans has been made by the AEC. This decision has required the consideration of alternate plans to obtain the information to support design activities. The primary alternate to testing in an out-of-reactor sodium core mockup is to utilize partial mockups and to perform extensive nonnuclear proof tests in the FTR. These reactor tests will undoubtedly delay the reactor power operation by as much as a year. The delay could be even longer if the nonnuclear testing dictates long and costly developmental changes.

C. HYDROMECHANICAL

Flow induced vibrations will be investigated by testing subassemblies, and by testing the reactor system including the core. The subassembly vibration program will initially define the effect of structural design parameters. Flow vibration tests will be run in water and sodium. The air tests require a vibration laboratory. The subassembly flow tests in water will be performed in existing PNL water loops in conjunction with subassembly pressure drop tests. The sodium test is planned to be made in conjunction with the fuel development, subassembly endurance tests in ANL-CCTL.

The reactor vibration testing will include vibration evaluation of the core unit and reactor internal components including baffles, shielding, reflectors, and duct support. The identification of potential vibration problems and resolution, if required, must be made with a geometric model in which all essential structural elements are present. Because the fundamental nondimensional parameters in fluid elasticity are not well established, the test section should have structural properties, material properties and flow channel configuration as nearly prototypical as possible. Because of quadrant symmetry, a 90° sector is sufficient to establish a cross flow effect in the outlet plenum. A 90° sector of the full size reactor is needed. The hydraulic requirements for this facility are 12,500 gal/min to provide the 125% of normal flow stipulated in the program discussion.

As shown in the previous discussion, a facility to provide the hydraulic conditions cannot be met by existing facilities, and the Hydraulic Core Mockup is the logical solution. The test unit will be a geometric model of the FTR. This model can also be used in thermal-mechanical studies of integral structural analysis discussed later.

Sodium testing where parallel and cross flow can be simulated is necessary. Tests should be made in prototype environment and conditions of the FTR, viz, during the nonnuclear FTR startup period.

D. MECHANICS

Resolution of mechanical problems requires varied facilities. The integral structural analysis requires geometric and loading (stress) characterization to assure predictable time temperature response of core geometry and to achieve a structurally sound core support system. The experimental data needed are those resulting from hydraulic and thermal driving forces. The fluid elastic loadings on prototypical ducts can be separated from the thermal effects. The two individual results can be superimposed to indicate the overall behavior. The information from vibration experiments will be integrated for the integral structural analysis. For the thermal effects, a heated gas model is needed to check linearity expansion with temperature changes. The core model will consist of prototype ducts, tube sheet and vessel cover. Only partial duct loading will be required, although the number must be sufficient to provide a good replica of the core restraint. For this, a 90° sector would be a minimum with the split core geometry. The same model can be used for the hydraulic loading. The details are presented in the hydromechanical discussion. The model will be similar to that suggested for the core vibration study and the same one can be used. The facility is as proposed in the Hydraulic Core Mockup.⁽¹⁾

Core accident mechanics require explosively dilating 1/10 scale cylinders up to 20 in. in diameter under a variety of simulated nuclear explosions to predict the effects of nuclear explosion on the reactor vessel.

1. D. S. Trent. Applications of Geometric Models to FFTF Hydraulic Core Mockup, BNWL-575. Pacific Northwest Laboratory, Richland, Washington, October 1967.

The tube sheet and cover will be initially evaluated by the use of reduced scale models, with static loadings applied in the member using photoelectric techniques. More prototypic loadings will be obtained in the full scale thermal-mechanical model with water flow.

The flow duct will require full size flow ducts including photoelastic models of stress joints and burst test models.

E. MECHANICAL FIT

The problems in the general category of mechanical fit are resolved through spatial mockups for initial evaluations and finally integrated testing in a sodium environment. The reactor cover assembly presents a difficult arrangement with the duct nozzles, instrumentation leads, sample lines, purge lines, leak monitor lines, closed loop piping, etc. in the tight configuration. The basic problem of initial arrangement is difficult; the problem is further complicated when it is considered that remote handling techniques must be used to assemble and disassemble the many components. The reactor cover mockup is essential to identify interference points and to provide corrective actions.

The internal fit up of the core components will evolve through a dry mockup of seven ducts to identify the major problem areas in the basic mechanism of inserting and removing ducts from an assemblage, to identify problems of fitup of a representative sector of the core, and finally, in a seven duct assemblage in sodium to determine the problems which may be influenced by the sodium environment. The latter will be integrated with sodium mechanism testing in existing off-site facilities.

The purpose of the mechanical fit mockups requires that emphasis be given to maintaining tolerance controls which are representative of the reactor components. The extremely tight fit anticipated on the cover and in the core make the use of full scale components for the mockup mandatory.

F. CONTROL MECHANISMS

The control mechanisms development program requires the use of air and water static facilities to simulate the reactor environment, followed by component verification in sodium and cover gas. Based on component tests, an assembled prototype control mechanism will be procured, water and air tested in the vendor's plant, and sodium tested in the existing Control Rod Test Tower at the LMEC. The production control mechanism will be tested in a seven-duct test mockup which can provide the suitable configuration and environment to evaluate the fit-up, holddown and maintenance aspects of the rod assemblage.

G. CLOSED LOOP

The closed loop development and testing program requires a series of helium, water and sodium static testing to evaluate the performance of individual hardware components in the initial phase of the program. These facilities are generally existing, or are not of major investments.

High performance and availability requirements of the closed loop require that the system be adequately proof-tested before commitment to the reactor. These tests will require the in-reactor hardware, reactor top hardware, inter-connecting piping, and the pump, valve, heat exchanger, etc. No existing facility can be adapted to provide the essential loop system; a new facility is required. This basic loop would also be used to support the operation of the FFTF by testing future closed loop in-reactor hardware for design and for operating characteristics. Such a facility in reasonable proximity to the reactor will provide an important support capability for effective long-term use of the closed loops as prime FTR test facilities.

H. INSTRUMENTS

The development of instruments suitable for the FFTF environment is largely directed to improvements of present

techniques. Static sodium pots of both short length and full vessel length are required for pressure transducer evaluation. Reactor nozzle mockups are needed for connector and in-core spatial flux monitor operation. A vessel mockup is desired for vibration and stress measurements tests. The large sodium flow-meter for the primary or secondary loops will require the proposed ACTI or the SPTF located at the LMEC.

A major need is a subassembly size flow test loop to proof-test pressure transducers, gas disengagement of fission gas, delayed neutron monitor operation, in-core sodium flow-meter, and the instrumentation subassembly. Existing facilities in this flow range and 1200 °F temperature rating are limited; larger flow loops are available but they will be used in other test work for the FFTF.

REQUIRED NEW FACILITIES

Several mockups not now existing are required to carry out this overall plan of reactor system development and testing. These include the following:

Thermal-Hydraulics

Large Sodium Heat Transfer Loop, 250 gal/min flow, 1500 °F, 200 psi pump head, 6 MW electric power supply for a heated pin bundle.

Core and Vessel Hydraulics

Hydraulic Core Mockup, 15,000 gal/min water flow rate, 200 psi head, 250 °F isothermal with plastic feature test models (e.g., inlet plenum), 1/2-scale composite model.

Core Vibration

Hydraulic Core Mockup, 15,000 gal/min water flow rate, 200 psi head, 250 °F isothermal with full-scale vessel and 1/4 segment of full size core components.

Mechanics

- 1) Full scale thermal-mechanical and hydraulic mockup (Hydraulic Core Mockup) with prototypic loading of 90° sector, with reactor cover, tube sheet, ducts, restraints. The mockup to be capable of heating up to 1000 °F and variable hydraulic loading, simulating up to 125% of bulk flow velocity.
- 2) One-tenth-scale cylinders for evaluation of nuclear explosion effects.
- 3) Scale models of reactor cover and plenum tube sheet with static loading.

Mechanical Fit

- 1) Full-scale Reactor Cover Mockup, with complete nozzle arrangement, piping, instrument lead.
- 2) Seven-duct dry and water mockup for evaluation of insertion and removal and shroud flutter.
- 3) Partial dry core mockup of approximately 20 ducts, simulated cover, tube sheets.
- 4) Seven-duct static sodium environment and mechanism mockup, with 2 to 3 ft diam vessel about 35 ft long, cover gas and level control.

Control Mechanisms

Seven duct static sodium environment and mechanism mockup with simulated control rod in-core position, 1200 °F. Cover gas and level control.

Closed Loop

A prototype closed loop system including in-reactor hardware, reactor cover, interconnecting piping, and out-of-reactor components including pump, valves, heat exchanger, 1350 °F, 700 gal/min.

Instruments

Single subassembly flow loop, 1200 °F, 500 gal/min, with prototype simulated fuel subassembly and duct. This facility may possibly be modified from existing loops.

Existing Sodium Facilities

The reactor assembly testing requires various types and sizes of sodium facilities. Requirements for the program have been identified and where possible, specific facilities have been named. The selection does not imply that other comparable facilities do not exist, although this would generally be the case in the larger flow capacity systems. A list of the many sodium facilities presently in use or standby is given in Table 10.IV. The table is separated into facilities by sizes. Each facility is identified as to its primary design use and the more important secondary use it might serve. This listing indicates alternate sodium loops which could be used in the reactor assembly development and testing program.

FFTF Sodium Facilities Building

The majority of PNL's sodium facilities are concentrated in the basement of 324 Building. This area is best adapted to small, radioactive loop experiments. The present building is becoming congested, and a steel frame building is needed to provide space for the facilities and prototypes recommended in the foregoing sections. These equipment items include a reactor cover mockup, large heat transfer loop and closure test facility. The seven duct Environment and Mechanism Mockup would be contiguous to this building. This building is needed in mid FY-1969 to house some of the required mockups.

TABLE 10.IV. Partial List of Existing Dynamic Sodium Facilities

Size	Facility (Flow, gal/min)	Site	Hydraulics	Thermal Hydraulics	Hydraulic Mechanical	Mechanical Structural	Mechanical Configuration	Mechanical Functions	Materials	Instruments
Multi-pin (<300 gal/min)	Fast Reactor Fuel Cladding Test Loop	(25) AI		X					X	
	Forced Convection Loop	(6) AI		X						
	System Quality Sodium Loop	(10) AI								
	Loop 2	(25) ANL							X	
	Quality Control Loop	(15) ANL								X
	Components Test Loop (construction)	(150) ANL	X		X	X				
	High Temp Na-U Paste Flow	(10) APDA	X							
	Sodium Endurance Loop	(300) APDA	X		X	X				
	Sodium Technology Loop	(7.5) APDA							X	X
	Loop No. 1	(2) GE		X						
	8	(6) GE		X						
	9	(6) GE		X						
	11	(80) GE	X	X						
	12	(-) GE				X				
	Heat Transfer System	(225) MSA	X							
	Pump Development Loop 2	(200) MSA	X							
	Heat Transfer System	(25) PNL		X						
	Small Components Evaluation Loop	(30) PNL	X							
	Large Size Loop (Design)	() B&W	X							
	Large Potassium System	(11) ORNL	X							
	Intermediate K System	(8) ORNL							X	
	FCL-1	(5) UNC	X			X				

TABLE 10.IV. (contd).

Size	Facility (Flow, gal/min)	Site	Hydraulics	Thermal Hydraulics	Hydraulic Mechanical	Mechanical Structural	Mechanical Configuration	Mechanical Functions	Materials	Instruments
	FCL-2	(5) UNC	X			X				
	FCL-3	(5) UNC								
	FCL-4	(5) UNC								
	GPL-1	(200) WEST	X	X	X	X				
	6-Mass Transfer Loop	(50) WEST							X	
	GPL-									
Subassembly (300-800 gal/min)	Core Component Test Loop	(380) ANL	X		X	X		X	X	
	EM Flowmeter Calibration	(300) ANL	X						X	
	Pump Development 1	(500-1000) MSA	X						X	
	GPL-2 (Plan)	(600-1200) WEST	X							
Multi- Subassembly (800-5000 gal/min)	Sodium Component Test Installation	(3600) AI	X		X	X		X	X	
	Large Component Test Loop	(1200) AI (130)	X		X	X			X	
	Pump Test Loop (to be modified- ACTI)	(2200) AI	X	X	X				X	
Reactor (>5000 gal/min)	Sodium Pump Test Facility (Design)	(120,000) AI	X						X	
	Auxiliary Components Test Installation	(7200/20,000) AI	X						X	
Control Rod	Control Rod Tower	AI				X		X		

10.26

BNWL-614

RECOMMENDED PLAN OF SCHEDULING

The Recommended Plan of Scheduling for the reactor system development and testing program is presented in Tables 11.I through 11.VI. The schedule has been arranged to identify the plan with specific facilities and their utilization in the overall program. This plan is subject to modification as the development and testing program proceeds, and the objectives and requirements may be redefined. However, it is believed that the development and testing program presented herein substantially represents the needs for the FFTF reactor system, and may be used as a blueprint for proceeding with the work discussed.

The development and testing schedule presented in this plan is generally consistent with the overall FFTF schedule requirements. In some cases, these FFTF requirements need to be satisfied by preliminary data to allow design commitment on a reasonable basis, with the final data to be supplied later as confirming information. Although programs, schedules, facilities and plans for their utilization are vital to the FFTF program, there are a few key areas which must be emphasized. These are considered key items in part because the successful and timely completion requires an inherent leadtime in putting together the facilities required to carry out the job. These include the following:

- 1) Hydraulic Core Mockup
- 2) Seven-Duct Environment and Mechanism Mockup
- 3) Reactor Cover Mockup
- 4) Large Heat Transfer Loop
- 5) Closed Loop Prototype

HYDRAULIC CORE MOCKUP

The Plan of Scheduling for the Hydraulic Core Mockup is based on performing key feature tests of concepts to characterize the gross hydraulic behavior to support the preliminary

design. The schedule anticipates that these tests will commence in January, 1968, and continue into the composite model tests a year later. The plan will provide the required hydraulic information on a timely basis for the FFTF schedule if the necessary continuity is maintained in the design and construction of the Hydraulic Core Mockup, and its use for design-supporting tests. At the present time, site preparation activity in the AEC-PNL facility of 321 Building, 300 Area, is in progress. Feature tests requiring up to 3000 gal/min flow can proceed at this site with slight additional work. It is considered feasible to begin feature tests in January, 1968. However, a significant delay in the start of the feature tests would delay information needed to support inlet and outlet plenum design decisions.

SEVEN-DUCT ENVIRONMENT AND MECHANISM MOCKUP

The Seven-Duct Environment and Mechanism Mockup will be used for the final verification of key in-core components before commitment to the reactor. This facility is largely oriented to the mechanical support of the reactor in-vessel components. The mockup is currently scheduled to be available for test work in June, 1969. Availability at this time will allow testing data input during the detail design period of the reactor vessel and core.

REACTOR COVER MOCKUP

The reactor cover arrangement has a direct bearing on the vessel design. To support the design activities, information leading to the best arrangement needs to be obtained early in the vessel design. Further, interfaces with the many systems requires that early assurance of design compatibility be provided so that each of these systems can be developed, designed, and constructed on an assured basis. The mockup must be fabricated beginning in early FY-1969, and preliminary information

on desirable cover arrangement should be available in mid to late CY-1969. This timing will yield arrangement information in the early vessel design period so that the major problems, if any, will be resolved on a timely schedule.

LARGE HEAT TRANSFER LOOP

The Large Heat Transfer Loop will provide resistance-heated testing capability up to 37 pins. This is considered the smallest number that can adequately characterize a 217 pin fuel subassembly. The alternate to doing the experiments is to do extensive heat transfer testing during the reactor start-up, followed by a very cautious power ascension schedule which would delay attaining operation at the design power level.

The schedule is based on starting tests with the Large Heat Transfer Loop about July 1969. This testing will follow a seven-pin test period in another facility which exists. The Large Heat Transfer Loop will require about 6 MW of electrical power to provide a simulated heat flux of 2×10^6 Btu/(hr)(ft²), and a flow rate of up to 250 gal/min. The facility meeting these basic requirements would be available in the planned schedule of July 1969, if two conditions can be met:

- 1) Plans for initiation of the 37 pin tests are firmed by January 1968.
- 2) The reactor coolant loop components of the Fast Reactor Core Test Facility (LASL) is made available for the heat transfer loop by January 1968.

CLOSED LOOP PROTOTYPE

The Closed Loop development will involve early component testing followed by complete prototype system testing beginning in January 1971. The sodium components such as pumps, valves, heat exchangers, and in-core ducts, will require additional evaluation before the complete system can be fully designed, fabricated and tested. However, a late FY-1968 or early FY-1969 start on the design is essential if the prototype loop is to be completed to meet the indicated schedule.

TABLE 11-I. Existing PNL Water Loops and Static Mockups

Facility	Key	Problem	Information Gained	Start	Finish
PNL 189D Hydraulic Lab (water)	A-4	Subassembly pressure drop and cross-channeling	Subassembly hardware design, channel flow pressure drop characteristics	Jan 1968	Jul 1968
				Jul 1968	Dec 1969
	B-3	Subassembly fretting	Adequacy of design for vibration	Jan 1968	Dec 1969
	B-4	Subassembly vibration			
	F-5	Duct flutter	resistance	Apr 1969	Jul 1969
PNL 314 Hydraulic Test Stand (water)	C-1	Hydraulic balance	Hydraulic balance concept for inlet plenum	Jul 1967	Nov 1967
PNL EDEL I (water)	G-5	Control rod •vibration •flow characteristic	Preliminary hydraulic data	June 1968	Jun 1969
	G-6	Control rod deceleration	Acceptable decelerator	Apr 1967	Mar 1968
	H-2	Closed loop in-reactor hardware	Preliminary hydraulic data	Jun 1969	Jun 1970
Control Mechanism Component Test Stand (static) Ambient	G-1	Drive mechanism	Acceptable drive design	Apr 1968	Dec 1968
	G-2	Scram mechanism	Acceptable scram device	Apr 1968	Dec 1968
Closed Loop Static Mockup (water)	H-3	Interconnecting pipe	Proof of principle of connectors	Aug 1969	Oct 1969
	H-1	Top face hardware	Proof of principle of seals, hardware	Oct 1968	Jan 1969
	E-7	Driver duct top face hardware	Proof of principle of seals, hardware	Oct 1968	Jan 1969
Dynamic Vibration	B-3	Subassembly fretting	Definition of natural frequency, damping ratio	Aug 1967	Dec 1967
	B-4	Subassembly vibration			
Tube Burst Facility	E-3	Flow duct creep rates	Allowable creep rate in flow ducts	Jul 1968	Dec 1968

TABLE 11-II. New Hydraulic Core Mockups

Facility	Iden Key	Problem	Information Gained	Start	Finish
Hydraulic Core Mockup Feature Models	C-1	Inlet plenum hydraulics	Plenum concept characteristics; design data.	Jan 1968 Jun 1968	Jun 1968 (Prelim.) Dec 1968
	C-1	Outlet plenum	Plenum concept characteristics; design data.	Mar 1968 Jun 1968	Jun 1968 (Prelim.) Dec 1968
	C-2	Bypass flows	Characterization of leakages, reflector flow rates.	Jun 1968 Dec 1968	Dec 1968 (Prelim.) Mar 1969
Hydraulic Core Mockup-Composite 1/2-Scale Geometric Model	C-1	Vessel flow distribution	Complete characterization of steady state core flow split.	Jan 1969 Jun 1969	Jun 1969 (Prelim.) Dec 1969
	C-1	Upper plenum mix	Thermal gradient minimization for high outlet temperature	Jan 1969 Jun 1969	Jun 1969 (Prelim.) Dec 1969
	C-2	Bypass flows	Optimization of bypass flow system.	Jan 1969 Jun 1969	Jun 1969 (Prelim.) Dec 1969
	C-3	Thermal gradient	Characterization of outlet coolant temperature gradients	Jan 1969 Jun 1969	Jun 1969 (Prelim.) Dec 1969
	D-1	Core vibration	Identification of component or component systems where significant vibration exists	Jan 1969 Jun 1969	Jun 1969 (Prelim.) Dec 1969
	C-4	Loss of coolant	Possible consequences of loss of coolant accidents	Jan 1970	Jul 1970
	C-5	Cavitation	Identification of problem areas. Evaluation of design modifications to prevent cavitation	Jan 1969	Dec 1969
	C-6	Gas entrainment	Identification of gas entrainment areas	Aug 1970	Dec 1970
	C-7	Solids deposition	Identification of design features needed for preventing solid deposition.	Jan 1971	Apr 1971
	C-8	Gas or steam closed test loop rupture		Jun 1971	Dec 1971
Hydraulic Core Mockup Sector Core Mockup Full-Scale Vessel	F-2	Fitup	Confirmation of core assembly design	Jul 1969	Sep 1969
	D-1	Core vibration	Define fatigue stress levels, extent of galling, fretting	Mar 1970	Jun 1970
	I-4	Integral structure analysis	Optimizes location and rigidity of core supports, establishes how mating parts interact on heating.	Sep 1969	Mar 1970
	I-7	Vibration and stress measurements	Prooftesting of vibration and stress measuring method for core components and vessel surveillance	1969	1971
	I-8	Signal connector-nozzle	Design data for acceptable connector	--	1970
	I-11	In-core flux distribution monitor	Design adequacy, specification for operable system	--	Dec 1971

TABLE 11-III. New Static Mockups

Facility	Iden. Key	Problem	Information Gained	Start	Finish
Reactor Cover Mockup	F-1	Cover arrangement, fit, tolerances.	Adequacy of design Preferred form of assembly	Sep 1968	Jun 1969 (Prelim.) Jun 1970
Subassembly Module Sector Core	F-2	Duct fit up, refueling and heatup characteristics	Adequacy of core assembly design	Sep 1968	Mar 1969
Seven Duct Sodium Environment and Mechanism Mockup	F-2	Duct insertion and fit in sodium	Confirmation of core assembly	Jul 1969	Jun 1970
	G-9	Duct, control rod assembly	Mechanical fit verification	Sep 1970	Mar 1971
	G-10	Duct control rod assembly	Preoperational verification	Jan 1972	Dec 1972
	G-10	Control rod product assurance	Preoperational test	Sep 1971	Dec 1971
	F-4	Galling	Assurance of no major galling	Jul 1969	Dec 1972
	F-3	Self-welding	Assurance of absence of self welding	Jul 1969	Dec 1972
	F-2	Level instrument	Operational verification	Jul 1969	Jul 1970
	I-8	Signal connector nozzle	Design data for acceptable connector	Mar 1969	1970
Control Mechanism Component Test Stand (static) Sodium	G-3	Seal	Proof test of acceptable seal	Aug 1968	Jun 1969
	G-4	Accelerator	Proof test of acceptable spring	Apr 1968	Jun 1969
Closed Loop Static Mockups					
Hot gas	H-3	Interconnecting piping	Proof of principle of piping system	Jun 1969	Sep 1969
	H-1	Top face hardware	Proof of principle of seals	Jun 1968	Dec 1968
	E-7	Driver duct top face hardware	Proof of principle of seals	Jun 1968	Dec 1968
Sodium	H-3	Interconnecting piping	Feasibility of disconnector, removal, sealing	Dec 1969	Jan 1970
	H-1	Top face hardware	Feasibility of seals, statistical reliability	Jan 1969	Jun 1971
	E-7	Driver duct top face hardware	Feasibility of seals, statistical reliability	Jan 1969	Jun 1970
Photoelastic Lab	E-1	Plenum tube sheet	Out-of-plane deflections and in-plane stresses, allowable thermal movements	Jan 1969	Jun 1969
	E-2	Reactor cover	Allowable deflection and stress level	Jan 1969	Jun 1969
	E-3	Flow ducts	Optimum design at transition regions and performances	Jan 1968	Jun 1968
	E-6	Vessel, nozzles	Optimized design at nozzles and support reactions	Jan 1969	Jun 1969
Hazards Test Cell	E-5	Vessel, cover	Defines loadings on vessel and cover from simulated nuclear explosion	Jul 1968	Jun 1969

TABLE 11-IV. Existing PNL Sodium Facilities

<u>Facility</u>	<u>Key</u>	<u>Problem</u>	<u>Information Gained</u>	<u>Start</u>	<u>Finish</u>
Heat Transfer Loop 1	A-1	Fuel clad temperature limit	Location and magnitude of local clad hot spot	Jan 1968	Jan 1970
	A-2	Local coolant and voiding and propagation	Condition under which local boiling and voiding will occur	Jul 1969	Sep 1970
	B-3	Vibration with heated bundle	Vibration potential from heated pins	Jan 1968	Sept 1970
Small Components Evaluation Loop	B-1	Cladding erosion	Demonstration of low erosion corrosion	Apr 1967	Jun 1968
	B-2	Cladding corrosion	Design adequacy of cladding and spacer materials removal rates	Apr 1967	Jun 1968
	B-3	Fretting corrosion	Adequacy of present design for corrosion resistance	Apr 1967	Jun 1968
Instrument Static Pot	I-2	Pressure transducer	Transducer component suitability	Oct 1967	1970

11.7

BNWL-614

TABLE 11-V. Existing Off-Site Sodium Facilities

Facility	Iden. Key	Problem	Information Gained	Start	Finish
LMEC SPTF or ACTI	I-10	Calibration of sodium flowmeters in large pipes.	Calibration data, with sodium	Mar 1970	Mar 1971
	C-6	Gas entrainment in pump	Identification of potential problem area	Mar 1970	Mar 1971
	C-5	Cavitation in pump	Identification of potential problem area	Mar 1970	Mar 1971
LMEC SCTI	C-6	Gas entrainment in intermediate heat exchanger	Identification of potential problem area		
LMEC LCTL	I-3	In-core sodium flowmeter	Reliability of flowmeter for in-core operation	Dec 1968	1970
LMEC Control Rod Test Tower	G-7	Prototype control rod life test	Performance and life evaluation	Mar 1970	Sep 1970
LMEC Sodium Mechanism	F-3	Mechanisms of self-welding.	Potential for self welding and identification of condition of forces.		
ANL CCTL	B-1	Subassembly erosion	Demonstration of acceptable corrosion limit, adequacy of design	Jan 1968	Mar 1970
	B-2	Subassembly corrosion			
	B-3	Subassembly fretting			
	B-4	Subassembly vibration			
	E-6	Vessel and nozzle stresses	Establish valid analytical models confirmed and improved by experimental data State-of-art capability of 1200 °F sensors	Mar 1968	Mar 1970
	I-7	Vibration and stress measurements	Evaluation of vibration and stress measuring equipment	Mar 1968	Mar 1970
SEFOR Thermal Shock Facility	B-5	Pin thermal shock resistance	Design confirmation of adequacy of fuel pin to sustain thermal shock		
(To be selected)	I-1	In-reactor coolant temperature sensors	Ability of sensors to resist thermal shocks	Feb 1968	Oct 1969
ANL Sodium Pot	I-8	Signal connectors mid-channel	Design data for acceptable connector in sodium	In progress	Jul 1968
ANL Small Flow Loop	I-3	In-core flow meter magnet	Acceptable magnet for thermal and mechanical shock service.	In progress	1970
ANL Small Flow Loop	I-4	Gas disengagement technique	Gas disengaging efficiency.	In progress	Jun 1970

TABLE 11.VI. New Sodium Mockups

Facility	Iden. Key	Problem	Information Gained	Start	Finish
Subassembly Flow Mockup With Full Size Sub- assembly Channel (an existing loop probably modified)	E-6	Flow duct	Proof test and characterization	Jun 1969	Jun 1970
	I-6	Instrument package	Proof test of performance.	Dec 1968	1971
	I-4	Gas disengagement	Proof tested technique for gas disengagement. Location and size of sample tube.	Jun 1970	Jun 1971
	I-5	Delay neutron monitor	Confirmed data on operable system.	Jul 1969	1971
	I-2	Pressure transducer	Proof test for specifi- cations of FFTF transducers.	Mar 1968	1970
Closed Loop Prototype Facility	H-5	Complete system	Proof of performance operating charac- teristics.	Jan 1971	Jan 1972
	H-2	In-reactor hardware	Reliability assurance.	Jun 1970	Jan 1971
	H-4	External hardware	Reliability assurance.	Jun 1970	Jan 1971
	I-3	In-core sodium flowmeter	Specifications for closed loop flowmeter	Jan 1971	Jan 1972
Large Heat Transfer Loop	A-1	Fuel clad temperature limit	Reactor operating parameters.	Jul 1969	Mar 1971
	A-2	Local coolant voiding and propagation	Conditions and conse- quences of local boiling and voiding.	Jan 1970	Mar 1971
	A-3	Loss of coolant and power excursion accident	Hydraulic demand characteristics of fuel subassemblies.	Jan 1970	Jul 1971
	A-5	Critical heat flux (proof-testing)	Assurance of no critical heat flux problem.	Jul 1970	---
	B-3	Vibration with heated bundle	Confirmation of adequate vibration- caused damage pre- vention.	Jul 1969	Jul 1971

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 J. H. Westsik
 R. G. Wheeler
 L. A. Whinery
 R. D. Widrig
 T. M. Withers
 N. G. Wittenbrock
 M. R. Wood
 F. W. Woodfield
 J. M. Yatabe
 F. R. Zaloudek
 FFTF File (2)