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FAST FLUX TEST FACILITY
QUARTERLY PROGRESS REPORT
JANUARY, FEBRUARY, MARCH 1967

E. R. ASTLEY
PROJECT MANAGER

APRIL 1967

**AEC RESEARCH &
DEVELOPMENT REPORT**

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PACIFIC NORTHWEST LABORATORY

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By

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Project Manager

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SPLIT CORE CONCEPT

FAST FLUX TEST FACILITY
QUARTERLY PROGRESS REPORT
JANUARY, FEBRUARY, MARCH 1967

INTRODUCTION

This report is the first of a series of Pacific Northwest Laboratory (PNL) Quarterly Reports that will describe the technical progress of the Fast Flux Test Facility Project.

Information is reported here under the FFTF Project organization that was responsible for the work during the period being reported. When appropriate, contributions from offsite organizations are included; PNL is subcontracting some supporting inputs to the Project from offsite government and industrial organizations.

The Project's efforts during the first quarter of 1967 were directed into two main channels: 1) engineering and development that will continue throughout the life of the Project, and 2) a major program of concept evaluation.

Because of their basic importance to the Project, the subjects of concept evaluation and a thorough description of the final concept choice (frontispiece) are highlighted at the beginning of Section I of this report. Familiarity with the facility's concept brings the total report into perspective.

Some familiarity with the Project's history will also be helpful here. A capsule FFTF history is included below.

HISTORY

Studies and conceptual designs for a fast neutron flux irradiation testing facility have been in progress at the Hanford atomic complex near Richland, Washington since 1963.* In April, 1965, the AEC authorized the Pacific Northwest Laboratory to develop cost studies and conceptual designs for an FFTF. This effort was concluded in August, 1965; the result was the "reference concept."

The "reference concept" described a facility centered about a sodium-cooled fast test reactor equipped with a variety of instrumented test loops and specimen irradiation positions for performing tests under conditions that would closely simulate the environments of future fast-breeder power reactors. The reactor was to be rated at 400 MW_t. The core was to develop a neutron flux of approximately 1.35×10^{16} n/cm-sec at 400 MW_t.

* FFTF work taking place at Hanford during 1963 and 1964 was performed by Hanford Laboratories under the management of the General Electric Company. In 1965, the FFTF work proceeded in the same facilities; however, Battelle Memorial Institute (known as Battelle-Northwest at this site) replaced GE as manager and the facilities were renamed the Pacific Northwest Laboratory (PNL).

In addition to the reactor, the FFTF complex was to include fuel and specimen handling facilities, non-destructive examination cells and equipment for irradiated specimens, a Nuclear Proof Test Facility, and a Support Engineering Development and Test Facility.

In the "reference concept" the core was composed of 73 tubes arranged in a hexagonal lattice and skewed 4° per lattice ring. (The arrangement can be likened to a twisted bundle of pencils.) This concept demonstrated the advantage of dispersing or spreading out the cluster of fuel elements and test loops above the core. If the fuel elements and test loops were all brought out vertically, only the relatively inaccessible area of the hexagon would be available for instrumentation, changing of fuel elements, etc.

Early in 1966, the description of the facility was updated to reflect conceptual design changes and refinements up to January 15, 1966. This updated concept was termed the "interim reference concept." During 1966, this concept was reviewed with principal emphasis on safety, cost, reliability, R&D requirements, design and complexity, and, wherever possible, potential alternate means of accomplishing the required functions were identified.

During the review of the "interim reference concept" PNL solicited proposals for and initiated conceptual design programs for promising alternates to the concept. This program produced seven alternate design concepts, all of which were thoroughly reviewed by PNL (See Section I).

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JANUARY, FEBRUARY, MARCH 1967

SUMMARY
SECTION I

DESIGN AND CONSTRUCTION ENGINEERING - W. B. MC DONALD

Structures and Utilities - F. H. Shadel

A comprehensive evaluation of seven FFTF design concepts was completed.

Preliminary building arrangement Plan XV was prepared. A vertical hot cell containment structure was used.

Possible respiratory hazards associated with FFTF were reviewed. It was determined that either an air supply mask or a self-contained breathing unit will provide the required protection under conditions of potential respiratory hazard.

Reactor Systems Design Engineering - L. M. Finch

Twelve studies were completed by PNL to establish an understanding of the wedge conical geometrical arrays and to determine their influences on FFTF performance capabilities.

The pressure distribution at the region of coolant discharge from fuel subassemblies of a conical FTR core was assessed.

Tube sheet thicknesses for various designs of lower tube sheet were calculated.

Heat Removal Systems - J. M. Westsik

Desire to run the secondary pump at lower temperature and conditions with low potential for gas accumulation in the higher parts of the secondary piping led to a recommendation to place

the pump in the cold rather than the hot leg. A piping arrangement was worked out including provisions for venting and draining. A secondary pump design where the pump is separated from the surge tank is under consideration. This separation could be effective in reducing gas entrainment and turbulence problems at the pump suction.

A comparison was made of the merits of the Hanford PIFLEX and Bechtel's MEC-21/7094 piping flexibility codes. The MEC-21/7094 offers sufficiently greater flexibility and input capability to warrant its preferred use. Steps have been taken to acquire both the "short" and full length versions of the Bechtel code.

The optimization code SWAP is under development for studies of steam/water heat dump. This code is a parallel to the earlier developed ADOPT code for air heat dumps. The SWAP code is based on a tertiary water loop with kettle type evaporators and water cooled condensers. First results with the code indicate that the steam/water heat dump is significantly more expensive.

Loop Test Systems Design - D. R. Doman

Closed loop cell layouts have been completed in which primary system components are located inside reactor

containment and cleanup, sampling, and storage components are located outside. Connecting coolant lines passing through containment would be provided with valves which close on a containment trip.

Fuel Handling and Radioactive Maintenance - *H. D. Lenkersdorfer*

A conceptual design for a wet fuel handling machine was developed for Atomics International to aid their FFTF design effort.

Calculations were completed to update earlier estimates of gas cooling feasibility. The present study indicates that under ideal conditions, helium gas cooling is feasible up to about 40 kW if maximum cladding temperatures are not required much below 1000 °F.

Fuel Examination Facilities - *C. L. Boyd*

Layouts were completed for an inert gas cell.

Three phases of nondestructive examination have been defined.

The conceptual cooling capacity of equipment to be used for gas cooling tests on simulated fuel subassemblies was increased to permit controlling maximum surface temperatures down to

500 °F. The previous limit was 600 °F. This change was made to permit back-up to lower subassembly surface temperatures if problems are encountered in the development of sodium vapor control equipment.

The concept design for the PNL heat transfer laboratory has been completed. It utilizes two separate gas blowers and ducting systems to independently provide axial or transverse coolant flow across the test fuel assembly during inspection and testing.

Electrical Systems - *G. H. Strong*

Two diagrams have been completed that show preliminary layouts for FFTF 115 kV power supplies. Also shown are an associated switchgear and a plot plan.

The primary pump driver is being developed. The function of the driver components has been analyzed in the light of a BPA power loss.

To increase reliability, an effort is being made to use solid shaft couplings instead of a clutch connection between the drive motors and the pony motors of the primary pump driver. Overspeed requirements of the pony motors have not yet been established.

FFTF QUARTERLY REPORT
JANUARY - MARCH 1967

SECTION I

DESIGN AND CONSTRUCTION ENGINEERING

W. B. MC DONALD

STRUCTURES AND UTILITIES - F. H. SHADEL

Concept Evaluation - F. H. Shadel

The evaluation of seven conceptual designs (Figures I-1 through I-8) was completed during the first quarter of 1967. An outline of the evaluations is presented below.

COMMON FEATURES

Important features common to all of the conceptual designs are as follows.

- All designs have a roughly cylindrical core, with a height-to-diameter ratio of about unity, and with a volume of about 800 liters.
- The neutronics and heat removal characteristics are similar; significant differences result more from design choices than conceptual distinctions.
- All of the cores use upward coolant flow.
- The top of the reactor gives access to the core for component removal and replacement.
- All designs use primary and secondary sodium systems and air heat dumps.
- All designs specify austenitic stainless steel for vessels and piping in the primary and secondary sodium systems.
- Control and safety rods occupy fuel or reflector positions; the designs do not provide for a fuel and a safety element in the same lattice position. Each design uses poison control, sometimes combined with reflector removal.
- The potential energy release in a prompt critical excursion has not been calculated for all designs, but the similarity in core geometry and neutronics indicates that no substantive differences exist. An energy release of up to 2000 to 3000 MW-sec is assumed for evaluation purposes.

FIGURE I-1. Skewed-Tube Core

FIGURE I-2. Skewed-Nontube

FIGURE I-3. Bundled-Skewed Core

FIGURE I-4. Split Core

FIGURE I-5. Split Core (90° Horizontal Shift)

FIGURE I-6. Pool Vertical Core

FIGURE I-7. Piped Vertical Core

FIGURE I-8. Modified FERMI

DISTINCTIVE FEATURES

Significant distinctive features of the design studies are as follows.

Primary Conceptual Variables

Two Systems for Shielding and Atmosphere Control for Fuel Handling

- 1) Shielded handling area (hot cell) which may be used as an inert gas cell.
- 2) Unshielded handling area, with shielding incorporated in the fuel and component handling machine(s).

Access to Reactor

The evaluation suggested three different degrees of access for instrumentation, fuel handling, and installation of test facilities.

- 1) Access to individual core positions through nozzles in the reactor cover; nonvertical fuel array.
- 2) Access to reactor by removing the cover; vertical fuel array.
- 3) Access to reactor with under-shield remote handling mechanisms, together with access to closed loop test positions through nozzles; vertical fuel array.

Primary Heat Removal System Arrangement

- 1) Pool concept: a single pool of sodium contains the reactor and entire primary heat removal systems.
- 2) Piped concept: piping loops connect the reactor and primary heat removal system components.

The following combinations of concepts were evaluated.

- 1) Hot cell, open reactor access
- 2) Hot cell, nozzle access
- 3) Unshielded handling area, under-shield access.

The pool vent core study incorporates the pool concept; the skewed tube core design incorporates the piped concept. The remaining studies assumed the piped concept.

Distinctive Design Variables

Significant design distinctions (not conceptual differences) are of interest for two reasons. First, they may represent a range of design options available within certain concepts. Second, they may represent conditions dictated by the concept which limits capability or flexibility.

Core Arrangement

Subdivision of fuel varies from 34 driver fuel assemblies to 115.

The number of control and safety rod assemblies varies from 5 to 24.

With exception of the piped vertical which has a perforated shroud, all designs provide a flow channel for each fuel assembly.

Only the pool vertical core design uses a BeO moderator. All designs except one have externally mounted mechanical control rod drives. The pooled vertical core (GE) uses sodium-hydraulic actuated rods (with a mechanical design as backup).

Core arrays are uniform, with in-core closed loops incorporated in fuel positions, with the following exceptions:

- 1) The split core designs utilize vertical closed loops in a central plane bisecting the core.
- 2) The designs identify the alternate possibility of placing larger loops occupying 3 to 7 core positions.

The study suggests natural B_4C , enriched B_4C , and tantalum for control rod poisons.

Two designs incorporate reflector-located control and/or safety rods, in addition to in-core rods.

Nonvertical fuel configurations include:

Skewed Array
Conical and Wedge Arrays
Skewed Clusters of fuel Assemblies

Test Facilities

The concepts present two types of closed test loops:

- 1) Re-entry type: coolant flow passes down through an annular passage and up through the test section.
- 2) Once-through type: coolant flows down through a separate tube before entering the test section from below.

All designs show re-entry type loops; in addition, two concepts show once-through type loops.

The number, diameter, and angularity to vertical of the closed loops vary with the different concepts (Table I-1).

Fully instrumented open loops (Table I-2) can be located in any driver fuel position of the nonvertical core concepts. Only limited instrumentation is possible in nonclosed-loop

positions of the vertical core concepts.

Packaged loops (Table 1-3) can occupy closed-loop positions or instrumentable open-loop positions in any of the concepts. Transient test facilities, rabbit devices and trail cables, are feasible in all concepts. All designs incorporate capsule tests in reflector positions on the core periphery.

Fuel Handling

Approaches to in-transit, high decay heat cooling of driver and open loop fuel included:

- 1) Use of a wet fueling machine where the fuel assembly is lifted with a portion of the primary sodium coolant and is cooled by a system in the fuel handling machine.
- 2) Transfer of fuel under the surface of the primary sodium into a decay storage position.

A liquid metal-cooled fuel handling machine handles closed loop fuels. In most concepts, the central closed loops in the modified FERMI must rely on thermal convective coolant flow within the re-entrant loops. An auxiliary heat exchanger is attached to the loop during closed loop fuel handling of the pool vertical core concept.

Some designs use a multipurpose handling machine to refuel both driver and test positions; others break down the functions into several special purpose tools. All designs provide a separate hoist or gantry crane for lifting large components inside the containment.

TABLE I-1. Closed Loop Facilities

<u>Concepts</u>	<u>Number of Closed Loops</u>	<u>Outside Diameter, in.</u>	<u>Angularity, Degree</u>	<u>Core Position</u>
Skewed Tubed Core	1	5.21	0	Fixed*
	1	5.21	11.8	Variable
	2	8.25	15.6	Fixed
Skewed Nontubed Core	1	4.26	0	Fixed
	1	4.26	11.8	Variable
	2	~6.0	~17.5	Fixed
Bundled Skewed Core	2	4.9	7.5	Variable
	2	6.0	15.0	Variable
Split Core	5	5.0	0	Fixed**
	2 (reflector)	5.0	10.1	Fixed
Modified FERMI	3	3.5	0	Fixed
	2 (reflector)	6.0	3.5	Fixed
	2 (reflector)	8.0	3.5	Fixed
Pool Vertical Core	4	3.375	0	Variable
Piped Vertical Core	8	3.75	0	Fixed
	2 (reflector)	3.75	0	Fixed

* Location of closed loops is considered variable if only changes in exterior piping are required for changing location.

** Closed loop location is considered as fixed if modifications in the core are required before loops can be relocated.

TABLE I-2. Open Loop Facilities

<u>Concept</u>	<u>Number of Open Loops</u>	<u>Outside Diameter, in.</u>	<u>Angularity, Degree</u>	<u>Core Position</u>
Skewed Tubed Core	2	5.21	4.0	Variable
	1	5.21	8.0	Variable
	1	5.21	15.6	Variable
	2	8.25	15.6	Fixed
	8 (reflector)	5.21	~16.8	Variable
Skewed Nontubed Core	2	4.26	3.0	Variable
	1	4.26	6.0	Variable
	3	4.26	10.5	Variable
	2	4.26	15.0	Variable
	2	~6.00	~17.5	Fixed
Bundled Skewed Core	2	2.76	0	Variable
	8	2.76	7.5	Variable
	7	2.76	13.0	Variable
	1	2.76	15.0	Variable
	2	6.0	15.0	Variable
Split Core	8	4.88	Not shown	Variable
Modified FERMI	3 (up to)	3.5	0	*
Pool Vertical Core	4	3.375	0	Variable
Piped Vertical Core	8 (up to)	~3.9 in. max	0	*

* Can only be installed in closed loop positions

TABLE I-3. Special Test Facilities

<u>Concept</u>	<u>Facility</u>	<u>Number</u>	<u>Angularity</u>
Skewed Tubed Core	Packaged Loop	1	4°
	Rabbit	1	~16.8°
Skewed Nontubed Core	Packaged Loop	3	1 at 3°, 2 at 10.5°
	Rabbit	1	~17.5°
Bundled Skewed Core	Packaged Loop	2	~10°
	Rabbit (in-core)	2	~15°
Split Core	None shown		
Modified FERMI	Rabbit	1	3.5°
Pool Vertical Core	Packaged Loop	2	0°
	Capsule (2 in-core)	15	0°
	Trail Cable (in-core)	1	0°
Piped Vertical Core	Rabbit	1	0°

Changeout of a single fuel assembly is accomplished through nozzle connections in the dispersed core designs nozzle opening for seven positions. The vertical core designs remove the reactor cover for refueling.

The approaches to providing instrumentation connections included:

- 1) Individual guide tube or hanger rod mounted leads and connectors
- 2) Boom mounted retractable arms
- 3) Retractable instrument mounting plate.

In the case of individually mounted instrumentation, the leads and connectors are part of the assembly handled with the fuel handling machine. In other designs, the instrumentation is moved out of the way of refueling via a separate mechanism as a preparatory operation.

The length of driver fuel and open loop assemblies to be handled varies from 8 to about 40 ft. Concepts that provide more complete instrumentation of driver and open loop assemblies invariably have long assemblies to handle. Concepts employing only proximity instrumentation have short assemblies containing only the fueled section, reflectors, axial shielding, and pieces required for holddown and grappling.

In all designs except the pool vertical core, test facility fuel is handled by a wet fuel handling machine. The exception, open loop fuel handline, provides no method for removing decay heat from specimens

being transferred directly to fuel examination; the closed loop tests cannot be handled separately from the loop tube since fuel transfer requires attachment of an auxiliary heat exchanger.

Heat Removal

The important design distinction between alternate concepts is that between the pool type and the piped type primary coolant systems as discussed previously.

Containment

The containment for the reactor design evaluated includes a primary containment vessel with various combinations of features for attenuating the effects of the design basis accident and other reactor incidents.

Basic major combinations are:

- 1) Strong reactor vessel and cover, with shielded fuel handling machine in air atmosphere.
- 2) Reactor vessel with removable cover, within hot cell and cavity; remote refueling in inert atmosphere.

CONCEPT CAPABILITIES

Specific Objectives

Capabilities and limitations of the alternate designs are discussed below with respect to specific design objectives. Only those major conceptual alternates which affect the parameter under consideration are discussed.

Quantitative ParametersNeutron FluxNonvertical Core

- 1) The dispersed core array results in a flux penalty relative to a tightly packed vertical core of 6% to 7%. Peak fluxes range from about 0.8 to 1.0×10^{16} , for ~ 800 liter core operated at ~ 400 MW with mixed oxide fuel.
- 2) Spatial flux distribution is nearly prototypical of fast breeder reactors.

Vertical Fuel Core

- 1) Maximum flux level, per unit power, is achieved with the tightly packed core array; peak fluxes range from 0.95 to 1.05×10^6 , for ~ 800 liter core, operated at ~ 400 MW with mixed oxide fuel.
- 2) Spatial flux distribution is nearly prototypical of fast breeder reactors.

Flux Spectrum

There is no significant difference between vertical and nonvertical cores. All have the following characteristics:

- 1) Flux > 0.1 MeV ~ 65 to 70%
- 2) Near prototypic of fast oxide breeders; departing slightly in and near nickel reflector.

Reactor and Closed Loop OutletTemperatures

These are dependent on materials properties, and independent of basic concept. There are no inherent conceptual limitations with respect to meeting the objectives.

Test Fuel Power

All concepts are essentially equal in capability to test fuel pins at up to 50 kW/ft in closed loops.

Number of Closed Loops

- 1) The nozzle access, non-vertical cores are capable of incorporating at least 12 closed loops.
- 2) The open reactor access, vertical cores are less compatible with a large number of closed loops because of space restrictions and requirements. Designs show 4 to 10 closed loops.
- 3) The undershield manipulator, vertical core is more restricted than the open reactor access concept. The design shows seven closed loops; three central and four peripheral.

Length of Test Specimen

Test specimen length is independent of concept, prototypic neutronics are limited by core height of about 3 ft.

Diameter of Closed LoopsNozzle Access; Nonvertical Core

- 1) In-core loops are limited to fuel assembly diameter of $4 \frac{1}{4}$ to 6 in. OD, except the bundled skewed core design can incorporate closed loops in-core at various diameters from 2.75 to 9.5 in.
- 2) Peripheral loops are readily incorporated with same diameter as fuel assembly, and oversize loops up to ~ 8 in. can be provided.

- 3) Re-entry loops are readily incorporated; once-through loops can be incorporated by permitting coolant leakage or by providing special oversize reactor cover nozzles.

Open Reactor Access; Vertical Core

- 1) In-core loops can occupy multiples of subassembly positions. Loop diameters up to 8 in. could be incorporated in up to 7 core positions.
- 2) Either re-entry or once-through loops can be incorporated.

Number of Open Loops

Nozzle Access; Nonvertical Core

Fully instrumented open loops can be incorporated in any number, limited only by test specimen reactivity.

Open Reactor Access; Vertical Core

- 1) The number of fully instrumented open loops is restricted by access provisions and interface between open loop leads and closed loop piping and leads.
- 2) The number of proximity-instrumented open loops is limited to instrumented core positions.

Undershield Access; Vertical Core

- 1) The number of fully instrumented in-core open loops is limited to closed loop positions.
- 2) The number of proximity-instrumented open loops is not limited except by test specimen reactivity.

Diameter of Open Loops

Maximum open loop diameter is same as fuel subassembly. Size is limited by core subdivision, but not by concept.

Diameter of Test Fuel Pins

Any size can be accommodated within power and overall test specimen size limited, independent of concept.

Qualitative Parameters

Coolant Chemistry and Foreign Material Control

Nozzle Access; Nonvertical Core

Closed method of refueling reduces chance of foreign material entering reactor.

Open Reactor Access Vertical Core

Complete removal of reactor cover increases chance of foreign matter entering reactor.

Under Shield Remote Access; Vertical Core

Closed method of fuel handling reduces chance of foreign matter entering reactor.

Pool Concept

- 1) Large sodium volume dilutes impurities.
- 2) Potential stagnation areas in sodium pool are collecting areas for debris.
- 3) Provisions for mechanical cleanup of debris need development.

Piped Concept

- 1) A given amount of impurity produces a higher concentration because of smaller sodium inventory; however, rapid

cleanup is possible with less equipment.

- 2) Generally, high sodium velocity keeps surfaces flushed.

Vent and drain lines are potential settling places for solid sediments and gaseous contaminants.

Test Orientation

Nonvertical Core

- 1) Closed loop test orientation is vertical only at the core center except in the closed vertical loops of the split core design,
- 2) Nonverticality varies from about 2° per ring of core positions to about 4° per ring. (Maximum nonverticality for core positions in the designs considered ranges from about 6 to 16).

Vertical Core

All test positions are vertical, except the peripheral loops in one design which have a 3 1/2° tilt.

Alternate Coolant Capability

All concepts have the capability to use alternate coolants in closed loops. Hydrogenous coolants present substantial risk.

Testing to Failure

All concepts have the capability to test fuel to failure in closed loops.

Short Term Testing Facilities

Such facilities can be incorporated in any concept.

Testing Isolation

Nozzle Access; Nonvertical Core

- 1) Effective nuclear isolation of tests can be achieved, close spacing of the vertical closed loops in the split core design can lead to interaction between tests.
- 2) Hydraulic isolation between core channels is possible, i.e., there is no cross flow between channels, and flow can be individually orificed.

Open Reactor Access; Vertical Core

- 1) Effective nuclear isolation of tests can be achieved by choice of loop locations.
- 2) Hydraulic isolation is possible as in nonvertical cores.

Under Shield Manipulator; Vertical Core

- 1) Nuclear isolation is limited by constraints on location of closed loops and control rods; the 3 central loops may interact significantly.
- 2) Hydraulic isolation is possible as in the other concepts.

Loop LocationNozzle Access; Nonvertical Core

- 1) Skewed designs offer flexibility for location, insertion and replacement of closed and open loops (any core position); in-core loops are restricted to fuel assembly size.
- 2) The split core design offers multiple vertical closed loops in the central plane of the core. Open loops, and additional closed loops can be incorporated in any core position.
- 3) The skewed cluster design offers flexibility for location of loops; oversize closed loops can be incorporated in core; open loops can be placed at node or cluster positions.

Open Reactor Access

Closed loop and instrumented open loop locations are restricted by piping and instrumentation leads.

Undershield Manipulator; Vertical Core

- 1) Closed loops are restricted to center or peripheral positions.
- 2) Open loops, with attached specimen instrumentation, cannot be accommodated except in closed loop positions.

Instrumentation CapabilityNozzle Access; Nonvertical Core

- 1) This concept provides accessibility for instrumentation of closed and open loops and driver positions, with a minimum refueling interference; all channels can be instrumented with sensors attached to test specimen.
- 2) Instrumentation connection and disconnection is provided outside reactor vessel.

Open Reactor Access; Vertical Core

- 1) Capability for instrumenting closed loop positions is provided.
- 2) Capability for instrumenting open loop positions, with sensors contacting the test specimen, is limited by closed loop piping and other instrumentation leadouts.
- 3) Instrumentation connection above the cover is limited to positions with a corresponding cover penetration of have connections below the cover.

Undershield Manipulator; Vertical Core

- 1) Capability for instrumentation of closed loop positions is provided.
- 2) Instrumentation of driver positions is limited to that mounted on a movable mounting plate; fixed leads would interfere with fueling.

- 3) Connection of instrumentation from closed loop positions and the mounting plate is provided outside the reactor.

Spectrum Softening Capability

All concepts are compatible with spectral softening by BeO addition, at penalty in flux.

General Goals

Safety

Shielding and atmosphere control concepts with and without the hot cells are shown in Figure I-9.

Hot Cell

- 1) The hot cell concept provides maximum flexibility for containment design, with option for design of the hot cell as either total or intermediate containment. It also provides maximum ability to control contamination or other consequences of accidents in the handling area. An inert gas atmosphere above the reactor will minimize sodium fire potential.
- 2) This concept also provides maximum flexibility for core subdivision of nozzle access cores, as discussed below.

Nonhot Cell Concept

The nonhot cell concept provides maximum accessibility for equipment maintenance and repair for normal operation.

FIGURE I-9. Refueling and Maintenance, Nonhot Cell Versus Hot Cell

Access to Reactor

Nozzle Access; Nonvertical Core

- 1) Core subdivision flexibility of the nozzle access concept is related to the choice of hot cell or nonhot cell shielding and atmosphere control. The design of a blast resistant cover, for incorporation of nozzle access in the nonhot cell concept, leads to a practical upper limit of some 70 to 95 nozzles. The corresponding number of driver fuel assemblies, assuming an outer ring of replaceable reflector assemblies, is about 40 to 60. Further subdivision of the core beyond the number of nozzles is possible by locating more than one fuel assembly for each nozzle position.

In the hot cell concept, with no requirement for blast containment by the reactor cover, at least 127 nozzles are practical; the corresponding number of driver assemblies is about 70 to 80.

Core subdivision is of safety concern in connection with a single channel meltdown. Safety rod scram acceleration of more than 3 g is required with less than about 45 to 50 fuel assemblies; with more than 70 to 80 fuel assemblies, the required acceleration is less than 1 g.

- 2) This concept has maximum flexibility for varying the number, location, and worth of

control and safety rods, since the control rod positions are identical and interchangeable with fuel and replaceable reflector positions.

- 3) Control and safety rod drives remain operational during reactor fueling, providing the option to employ "cocked safety" or to withdraw rods for critical measurements or shutdown margin verification at loading intervals.
- 4) Ejection of control and safety rods to initiate a nuclear accident can be made incredible by design.
- 5) Hydraulic and mechanical fuel hold-down methods are both applicable.
- 6) The capability of tubes or flow channels to prevent propagation of a local disturbance in one channel to adjacent channels does not appear significantly difficult

Open Reactor Access; Vertical Core

- 1) This concept is capable of core subdivision to the extent necessary to satisfy channel meltdown requirements.
- 2) The number of control and safety rods incorporated with mechanical drives is limited by access space for drive; this in turn limits the capability to use low worth rods. Little or no flexibility exists for varying the location of rods.

- 3) Mechanical rod drives generally must be disconnected for reactor fueling; thus there are no options for use of "cocked safety" or for rod withdrawal at loading intervals.
- 4) One design incorporates a relatively large number (18) of in-core rods, plus ganged peripheral rods, operated by sodium-hydraulic systems controlled by sodium-fluidic valves. This system has the advantages of eliminating the rod drive structures above the core, permitting rods to remain connected and operational at all times, and permitting a large number of relatively low worth rods in the core; the hydraulic-fluidic drive system has the added advantage of no moving parts except the rod itself. The major disadvantages of the design are that there is no direct control of rod position during withdrawal, and no direct indication of rod position. The system is undemonstrated for the intended service and would require development and extensive proof testing under prototypic conditions.
- 5) The adequacy of hydraulic fuel holddown in vertical cores needs to be demonstrated; capability to provide mechanical hold-down assist, or backup, is necessary. Mechanical

holddown is feasible by use of a core cover plate.

- 6) The capability of flow channels to prevent propagation of local disturbances does not appear significantly different among the concepts.

Undershield Remote Access; Vertical Core

- 1) Single channel meltdown considerations would require a fast response safety system.
- 2) The location of control and safety rods which remain connected during fueling is restricted by fuel handling mechanism access requirements to positions on three core radii. Additional rods could be incorporated if the drives were disconnected for fueling operations. There is no flexibility for varying rod locations, once fixed.
- 3) Hydraulic holddown needs to be demonstrated. Mechanical backup or assist could be provided by a core cover plate.
- 4) Capability to prevent propagation of local disturbances is the same as other vertical concepts.

Alternate Primary Heat Removal Systems, Figure I-10

Pool Concept

- 1) The pool concept provides inherent capability to maintain coverage of the core and flow paths for natural circulation decay heat removal.

FIGURE I-10. Primary Heat Removal,
Piped Versus Pool Concepts

- 2) The sodium pool also provides a large heat sink, and is readily adaptable for incorporation of an emergency cooling loop.

Piped Concept

- 1) The loss of sodium to the extent of uncovering the core must be prevented by design features, such as an outer vessel.
- 2) The capability to remove decay heat in event of pipe breakage must be provided by design features, such as isolation of defects or provision of auxiliary cooling.

Reliability and Availability

Qualitative evaluations were made of the relative effects of major conceptual features on reliability and availability.

No major difference is apparent between concepts in time required for normal shutdown operations. The main differences in plant availability arise from the capability to handle non-routine operations.

Shielding and Atmospheric Control

Hot Cell Concept

- 1) Can accommodate an inert atmosphere above the reactor to minimize sodium fire.
- 2) Provides potential for minimizing downtime and speeding recovery from abnormal situations such as contamination releases and sodium spills.
- 3) Good visibility is available for attaching fuel handling equipment to reactor nozzles.

Nonhot Cell Concept

- 1) For normal operations, provides direct accessibility for contact maintenance and repair of equipment.

Access to ReactorNozzle Access; Nonvertical Core

- 1) Provides greatest ease for replacing instrumented test specimens.
- 2) Replacement and repair of instrumentation is facilitated by the single channel leadout.
- 3) Location of connectors on the nozzle cap provides an environment conducive to high reliability by having relatively low radiation, low temperature and low potential for sodium vapor.
- 4) The relatively loose-packed core lessens the potential for fuel elements sticking.

Open Reactor Access; Vertical Core

- 1) Permits proven vertical refueling.
- 2) Requires removal and replacement of the reactor cover plug.
- 3) The potential exists for having a backup fuel handling method for abnormal conditions, either by extra manipulators or special tools.
- 4) Open access facilitates changing large core components such as grid plates.
- 5) Control rod connection and disconnection is required with mechanical rod drives.

Undershield Remote Access; Vertical Core

- 1) The small number of openings and infrequent entry into the reactor core minimizes atmospheric contamination of cover gas or sodium coolant.
- 2) Minimum preparation is required for refueling, with potentially high refueling rates.
- 3) Presence of test facilities, control rods, and core instrumentation hampers and complicates fuel handling.
- 4) Access to handle abnormal fuel handling situations such as stuck fuel is greatly restricted and potentially could extend outages resulting from such situations.

Development

Areas requiring development efforts significantly different from the base program described previously are summarized in the following sections for each of the primary conceptual variables.

Shielding and Atmosphere ControlHot Cell

- 1) Requires development of equipment to operate in sodium vapor and inert atmosphere.
- 2) Equipment and techniques are needed for performing maintenance remotely or with limited direct personnel contact.
- 3) Methods for adequate viewing, either directly or indirectly, require perfecting; viewing

windows must withstand the design basis accident.

- 4) Requirements for successful development of leak-tight seals are less than described in the base program.

Nonhot Cell

- 1) Methods to control leaks and contamination spread will be required.
- 2) Equipment and methods for performing work on the top face through gamma shields are required.
- 3) Methods for sealing fuel handling equipment to reactor openings without spread of contamination are needed.

Access to Reactor

Nozzle Access; Nonvertical Core

- 1) Satisfactory nozzle seals must be developed.
- 2) Capability to insert >25 ft long assemblies reliably at an angle must be developed.
- 3) The core arrangement requires extensive flow modeling to assure adequate heat removal.

Open Reactor Access; Vertical Core

- 1) Reliable methods for removing and replacing core cover plugs around test facilities and control rods penetrating the plugs must be proven.
- 2) Reliable disconnects are required for mechanical control rod drives.
- 3) Sodium-hydraulic-driven control rod drives would have to be developed and proof

tested before they could be accepted for use.

- 4) Offset fuel handling by vertical machines may have to be developed for replacing fuel not directly accessible.
- 5) Satisfactory methods for removing stuck fuel in closed loops must be developed.

Undershield Manipulator; Vertical Core

- 1) Variable reach offset handling mechanism operating under sodium requires development. For the design presented, a double offset to reach some core positions must be provided.
- 2) Methods for handling stuck fuel assemblies must be developed.

Primary Heat Removal

Pool Concept

- 1) Extensive hydraulic modeling will be required to establish flow and temperature characteristics under all reactor operating conditions. Evaluations must determine effects with various pump and heat exchanger combinations in operation.
- 2) Methods for accommodating thermal expansion between the reactor cavity and the sodium tank and between primary system components must be proven.
- 3) Under sodium flow instrumentation methods require development.
- 4) Materials or methods for location of neutron shielding in sodium must be demonstrated.

Piped Concept

- 1) Double-walled piping fabrication and inspection methods on large diameter pipes may be required.
- 2) Methods must be developed for installing instrumentation in double-walled piping, if used.

Summary

An extensive evaluation of several alternate Fast Flux Test Facility (FFTF) design concepts was completed. It became clear as the evaluation was concluded that the basic requirements could have been met by any of the concepts that were reviewed, but it was also clear that there were intrinsic differences of significance. Both concept-dependent and concept-independent problems associated with the several concepts were studied.

In the area of concept-dependent features, special attention was given to fuel handling methods, heat removal systems (pool or piped), verticality of test loops, closed-loop designs and access, interaction of containment systems with fuel handling modes, instrumentation capability (both in effectiveness and quantity) and, of course, safety. Also involved in studies of the concept-dependent features, was a study of on-line effectiveness.

In concept-independent areas of study, evaluations were made of the problems associated with basic technology intrinsic to all designs, particularly metallurgical and sodium chemistry requirements at the advanced conditions required for FFTF.

CORE ARRANGEMENT

Studies were made of several possible core arrangements that would have relative value to FFTF. The basic evaluations of the conceptual reactor cores were centered on a vertical close-packed core and two dispersed cores. These are illustrated in Figure I-11.

Vertical Core

Advantages

A vertical core permits vertical testing in driver fuel positions. A higher unit flux and power density is attained. The vertical core is closely prototypic of FBR's. It has the most simple in-core reactivity stability.

Disadvantages

The number and location of test facilities are more limited. The driver fuel instrumentation is limited. Control rod drives must be disconnected to remove the cover. There is a substantial limitation on the number and location of control rods. Replacing test facilities is very difficult. Most vertical core arrangements require a hot cell for refueling.

Dispersed Core

Advantages

There is greater freedom in the location of test facilities. Greater flexibility in the number of test facilities is possible.

FIGURE I-11. Core Geometry, Vertical Versus Dispersed

Test and driver fuel can be completely instrumented. Removal of test subassemblies from the facility is simplified. There is good access for test facility installation and replacement. There is greater flexibility in the number and location of control rods. Control rod drives remain connected at all times. A high degree of thermal isolation is inherent.

Disadvantages

There is a penalty of 6 to 17% compared to a compact vertical core. Less inherent vertical test positions are available. Present critical facilities inadequate for accurate engineering simulation. In-core reactivity and stability control is more difficult. We have no direct operating experience with nonvertical fast reactors. A complicated tilting, indexing fuel handling machine is required.

CONTAINMENT STRUCTURES

A prime consideration in the concept evaluation, and one which was considered to be equally feasible for all concepts studied, was whether FFTF should have a hot cell over the reactor in addition to standard containment. Figure I-9 shows facilities with and without the hot cell.

A hot cell is a necessity for open reactor refueling as demanded by a vertical core; it is optional with a dispersed core provided a

shielded fuel handling machine is used. In the latter case a hot cell would be advantageous in many maintenance situations where severe contamination is present. The disadvantage of a hot cell enclosure with a shielded fuel handling machine is the additional capital and operating cost. Without a hot cell, a shielded fuel handling machine is, of course, mandatory.

Hot Cell

Advantages

Operation is possible under conditions of extreme contamination. The inherent capability to recover from contamination incidents is provided for. The demands on the final containment structure are reduced. Flexibility for handling unshielded specimens is provided. Fuel handling machine requirements are reduced. Atmospheric control capability over the reactor is provided.

Disadvantages

Normal refueling and maintenance may be hampered. Design of in-cell equipment is more difficult because of environment conditions. An additional barrier to fuel and equipment transfer is imposed. Higher capital and operating costs are incurred.

HEAT REMOVAL

The principal choice in the primary heat removal system was between

a pool- or a piped-heat removal system concept. These concepts are shown in Figure I-10.

Piped Concept

Advantages

It has greater flexibility in hydraulic and heat transfer processes.

A higher allowable pressure drop in the intermediate heat exchanger and in the piping is possible.

Because of the well defined flow path, less testing is necessary to confirm calculations.

Neutron shielding and instrumentation can be located outside the sodium environment.

Disadvantages

High integrity is required for the primary pipe and components and a separate last ditch shutdown cooling method is mandatory.

Design is complicated by the requirement for substantial thermal expansion in the piping.

Failure of one pump or one heat exchanger shuts down an entire loop.

Pool Concept

Advantages

Inherent shutdown cooling is provided.

Design problems with regard to thermal expansion are eased.

More flexible operation is possible because the pumps and heat exchangers are not intertied.

Disadvantages

More expensive full-scale mock-up flow modeling is required.

There is less flexibility in the hydraulic and heat transfer processes.

The sodium tank support may be quite difficult.

Neutron shielding and instrumentation must be located in the sodium environment.

FINAL DETERMINATIONS

Pacific Northwest Laboratory carried the evaluation to the determination of the overall advantages and disadvantages of the several reactor concepts reviewed above. The evaluation was transmitted to the AEC for their subsequent judgment. The final selection was dispersed core reactor, with a piped heat removal system and a hot cell, but utilizing a shielded fuel handling machine. The dispersed core (Frontispiece) selected will accommodate a range of both vertical and inclined fuel sub-assemblies and test loops. Vertical loops traverse the entire reactor core. Dispersal, providing for greater top access, is obtained by inclining the test loops and driver sub-assemblies to this vertical plan on either side to form the balance of the core. This array has the general appearance of a wedge or a cone, but the angles are very small; thus, the dispersal within the area of the actual core is also small. Nevertheless, by extending these up to the reactor top cover where the test loops are brought out, the necessary overall dispersal is obtained as required for instrumentation connections and closures.

The initial core arrangement of this concept will provide space for

3 to 5 vertical, closed test loops. The initial core volume will be about 800 to 900 liters with mixed uranium and plutonium oxide as the driver fuel. All test and driver fuel will be instrumented for coolant temperature, flow, and fuel failure.

A hot cell will be placed over the reactor vessel to accommodate a wide range of normal and abnormal operations. Under normal operation, personnel access to the hot cell, with direct-contact operation and maintenance, will be possible. If

accidental spillage or contamination occurs, the hot cell will be put in operation. This contingency system will also permit the replacement of the entire reactor core at some future date if such a change is desired. The facility will be capable of operating as a nonhot cell since a heavily shielded reactor vessel head with nozzle access to all main reactor core components will be included in the design. A fully shielded wet fuel handling machine will be provided for the same reason.

Building Arrangement - *F. H. Shadel*

The integrated functional requirements of the FFTF building arrangement is being studied. Special attention is being given to the requirements of equipment handling and vertical fuel transfer. Recent refinements are shown in Figures I-12 through I-20.

Respiratory Protection - *F. H. Shadel*

The Occupational Hygiene Department, Hanford Occupational Health Foundation, has reviewed the potential respiratory hazards associated with the sodium cooled FFTF for the purpose of recommending adequate respiratory protection for personnel. The FFTF will inherently provide a potential exposure to combustion products of sodium, oxygen deficient inert gas atmospheres, and radioactive aerosols. Each of these

hazards alone is unique in its respiratory requirement. Because these hazards may be encountered concurrently, personnel will need a respiratory protective device that gives protection against all respiratory hazards.

When personnel are exposed to potential respiratory hazards and can safely work within the radius of a compressed air station, the use of a supplied air mask is recommended. Breathing air manifolds should be placed near all areas of potential hazards such as loop cells and service galleries, etc.

For more mobility, a self-contained device with full face mask will also be considered for all-around respiratory protection.

34

BNWL - 501

0 10 20 FT
SCALE: 3/32" = 1'-0"

FIGURE I-14. Building Plan XV at Elevation -44 ft 0 in. and -65 ft 0 in.

FIGURE I-15. Building Plan XV at
Elevation +16 ft 0 in.

FIGURE I-16. *Building Plan XV Section*

FIGURE I-17. Building Plan XV Section

FIGURE I-18. Building Plan XV Sections

FIGURE I-19. Building Plan XV Elevations

FIGURE I-20. Building Plan XV Elevations

REACTOR SYSTEMS DESIGN ENGINEERING - L. M. FINCHConceptual Design Studies -*L. M. Finch*

PNL reviewed the Westinghouse and General Electric FTR design studies, with representatives from these companies, in January, 1967. Westinghouse and General Electric submission of informal design reports in February, 1967 allowed timely comparison and evaluation of these concepts with previously developed FTR conceptual designs. Discussions with GE and W personnel regarding FFTF programmatic requirements led to identification of several engineering support activities which could beneficially be performed by these industrial contractors. Engineering activities, identified as appropriate for performance by General Electric, included:

- Design Analysis Support Studies
- Sodium Vapor Transport and Detection
- Sodium Nucleate Boiling Detection
- Fluidic Control Demonstration

Study areas for further design participation by Westinghouse were identified as follows:

- Magnetic Latch Control Rod Drive System for the FTR
- In-Core Instrumentation for the FTR
- Evaluating the Influence of an Open Lattice Liquid Metal Cooled Core on Thermal-Hydraulic Coolant Distribution and Related Nuclear-Mechanical Interactions
- Carbide Core Design Development and Optimization Study
- Evaluation of LMFBR Irradiation Testing Requirements and FFTF Loop Designs
- Grid Spacers for Driver Fuel Assemblies

Typical Geometry of Split Wedge and Conical Core Arrangements for theFFTF - L. M. Finch

Twelve investigations were completed by PNL during March concerning split conical and wedge array geometries as they affect FFTF performance capabilities.

The wedge array consists of elements which recede from the plane of the closed loops in planes normal to the plane of the loops. Thus, in the wedge the closed loops may be considered as central elements of the wedge with zero tilt angle. All other elements of the wedge array are tilted away from the vertical loop plane. The particular advantage gained in the wedge array over the conical array is that the row of closed loops is not anomalous to the general wedge array elements, hence interference is limited only to the region of geometrical mismatch near the lines of wedge convergence. Figure I-21 shows a typical split wedge geometry.

For the conical array, the two dimensional tilt of the typical conical array element is anomalous to the vertical orientation of all except the central closed loop. The consequence of this anomaly is either progressively more severe interferences between closed loops and conical array elements at greater radial positions or greater spacing between the "envelope" of the row of closed loops and the "envelope" of the conical array half. Since the physical system does not permit interferences, the greater spacing requirement must hold. This spacing to avoid interference, pronounced in the conical

FIGURE I-21. Conical Array Study Arrangement I

array, is the reason the split conical arrangements can result in relatively poor active core compaction with consequent degradation of flux, doppler coefficient, enrichment requirements, etc. Figure I-22 shows a typical split core geometry.

Radial Pressure Drop at Outlet Region of a Conical FTR Core - M. T. Jakub

At the outlet of the hexagonal fuel sub-assemblies of a conical core, the sodium coolant exits the tube from a series of holes. This analysis assessed the pressure drop necessary for the fluid (exiting from the central subassemblies of the core array) to reach the outlet plenum region outside the core zone.

Each subassembly of the array provides a source for the flow (See Figure I-23) thus the velocities will increase with increasing radius if only low axial flow occurs. With the model described below, a pressure drop of 1.5 psi is required to overcome the flow resistance from the central subassembly outlet to the plenum region outside the reflector subassemblies. Since this model used hexagonal prismatic shapes, the use of cylinders should have a lower pressure drop for the same spacing and diametral sizes. Maximum velocities were approximately 15 ft/

sec outside the shapes. The flow outlet axial length was assumed to be 1 ft, and the effective length, because of axial flow, was taken to be 2 ft. From the preliminary estimates, the lateral forces do not appear to be large enough to create mechanical problems.

Lower Tube Sheet Design on a Conical FTR Core - M. T. Jakub

Thicknesses for various designs of lower tube sheet for a conical core were calculated by the ASME Nuclear Vessel Code method. The results are shown

<u>Ligament Efficiency</u>	<u>Minimum Ligament, in.</u>	<u>Lateral Pressure</u>	<u>Required Plate Thickness, in.</u>
0.286	1.2	150	4
0.200	0.75	150	6
0.200	0.75	100	5

in terms of ligament efficiency, which is the minimum thickness of the metal web divided by the pitch of the holes. Velocity in the 3 in. OD hole (42 = driver + open channels) was approximately 40 ft/sec at the inlet region; this velocity gives a pressure drop of approximately 0.4 psi/ft for the inlet portion of the fuel subassembly.

FIGURE I-22. Conical Array Study Arrangement II

FIGURE I-23. INC-II Core at Outlet
(~80.5 ft elevation)

HEAT REMOVAL SYSTEMS - J. H. WESTSIK

Alternate Concept Studies -

L. A. Whinery

Studies indicate that it is physically feasible to install a skewed-core reactor together with four heat exchangers, four primary pumps, and two 6 ft diam fuel decay tanks all within a 42 ft diameter pool vessel.

Preliminary structural calculations show that the pool vessel, with sodium contents, can be adequately supported from a steel and concrete structure above the vessel. The calculations assume that the reactor, neutron shield, and all primary system components are hung directly from the support structure with no loads transmitted to the vessel walls or bottom. Further study of this concept was abandoned because a choice was made to use the piped system. The choice was based largely on much better developed technology supporting the piped design.

Secondary Loop Engineering Studies -

L. G. Henke

The coolest operating environment for the secondary coolant loop pumps is recommended regardless of certain obvious advantages of a hot leg pump location, such as the inherent self-venting of process gas and the simplified complete emergency drainage of the loop.

Because of the limited information available concerning the cover-gas solubility of sodium at high temperatures, a cold leg location for the secondary loop pumps would be advantageous in maintaining necessary operational availability of the FFTF reactor coolant system.

A new secondary loop piping arrangement has been developed wherein these desirable features are retained with a cold leg pump installation, and with only a slight increase in the complexity of the piping. Self-venting will be accomplished with a continuous bleed-type vent line, which will be provided with optional cold-plugging facilities or a blocking valve should operating experience indicate that a continuous venting of gas is not necessary. Complete emergency drainage of each secondary loop will be possible with the addition of one short, check valved, drain line. Complete drainage will be accomplished with the opening of only one drain valve. For the avoidance of possible thermal shock, both of these secondary loop adjuncts will have simple built-in provisions for the equalization of sodium temperatures.

Also under consideration is the physical separation of the secondary loop pump and the sodium expansion and surge tank, in lieu of the integrated assembly of these items as shown to date. This would make possible a cooler pump environment, a shorter pump drive shaft requirement, and a slightly cooler, less turbulent, functioning of the expansion tank. Maintenance would also be simplified. With this arrangement the expansion tank would essentially be isolated from the recirculating coolant flow, to thereby prevent the vortex entrainment of the cover-gas, and to lessen the effects of sodium solubility. Freezing of expansion tank sodium, and serious thermal shock, would be impossible

because of the inherent natural circulation and the vent line bleed flow.

Evaluation of Piping Flexibility

Programs - L. G. Henke

A comparison of the Bechtel Corporation's MEC-21/7094 computer program for Piping Flexibility Analysis, and the Hanford developed PIFLEX program has been made to determine whether the MEC-21/7094 program has sufficient additional capabilities to merit its adoption for the various FFTF pipe stress analyses.

The MEC-21/7094 program, which is a fifth generation program of the one originally developed in 1958 by the Mare Island Naval Shipyard, and the PIFLEX program will both handle three-dimensional piping geometries consisting of tangents and bends parallel or skewed to the axis of the coordinate system. Both programs will compute stresses, forces, moments, deflections and rotations at the anchor and branch intersection points, and at the end of each bend and tangent. In addition, the MEC-21/7094 program will compute the stresses, etc., for all pipe line components, such as valves, fittings and expansion bellows, including flexible joints such as hinges and gimbals. It will also evaluate closed loop configurations.

The MEC-21/7094 program has a capacity for 2997 elements, 99 branches, 99 branch points, 2997 material types and any number of anchors or restraints. PIFLEX can handle 150 elements, 30 branches, 11 branch points, 19 material types and 15 anchors. However, computer time with the PIFLEX program is very dependent upon the number of branches used, and they must, therefore, be kept

to a minimum. This PIFLEX restriction severely limits the complexity of piping configurations, whereas the MEC-21/7094 program is suitable for very complex piping systems.

Procurement of the MEC-21/7094 program has been initiated and will, therefore, be available for the more complex piping configurations after conversion and debugging.

Steam/Water Heat Dump Concept -

H. G. Johnson

Heat dumps to air and to steam/water are being considered for the FFTF's main heat removal system. Computer codes are being developed for each concept to provide steady-state process design and cost optimization data for the entire system. The air dump concept is described by the ADOPT Code.⁽¹⁾ SWAP is the code for the steam/water dump concept; SWAP is divided into two sections. The first section provides steady-state process design and cost optimization data for given design conditions, while the second section will provide steady-state equipment performance data at conditions of power and primary coolant temperatures other than design. The first section of SWAP was completed, the second section is in the debug stage.

The SWAP Code is based on horizontal, recirculating, kettle-type evaporators coupled with horizontal surface condensers in closed tertiary loops. Secondary

1. C. L. Goss and H. G. Johnson. ADOPT, Program for Steady-State Process Design and Cost Optimization of FFTF Heat Removal Systems Using Heat Dumps to Air, BNWL-308, Pacific Northwest Laboratory, Richland, Washington, September 1966.

sodium is on the tube side of the steam generator and river cooling water is on the tube side of the condenser. Steam generator designs are based on nucleate boiling with high quality, saturated steam generated. For safety reasons, a double-walled tube (with helium or NaK

in the annulus) may be required in the steam generator designs. Such a construction can provide a large heat transfer resistance which in turn allows reasonable designs based on nucleate boiling even for primary sodium temperatures in the range of 1100 °F.

LOOP TEST SYSTEMS DESIGN - D. R. DOMANClosed Loop Layout Studies -

D. R. Doman

A closed loop cell equipment arrangement (Figure I-24) was prepared to determine minimum practical space needed for an all-purpose loop. This study incorporated:

- A regenerative heat exchanger and heater in addition to the intermediate heat exchanger to provide high in-reactor test section inlet temperatures for materials testing.
- An out-of-reactor materials test chamber in the primary coolant piping. Although shown in an in-line position, this chamber can also be installed on a bypass line to minimize main line flow restriction, if desirable.
- Equipment arranged in relatively small semi-modular packages to facilitate remote removal and replacement in minimal sized casks.
- Mechanical disconnects to determine potential problems in positioning for remote access.

- A helical rotor pump and separate expansion tank.
- Capability for either contact or remote maintenance.

All of the above appear feasible in this layout. The semimodular approach is also adaptable to separate locations for primary system components and purification system components. Such an arrangement would be desirable if space for only loop primary system components is available within the containment vessel. Such an arrangement would also increase accessibility to purification system equipment.

Disadvantages shown by the study are:

- Relative inaccessibility to purification system components in the equipment arrangement
- This particular layout is more compact than desired for easy maintenance. No problems are foreseen in increasing access if larger cell space is available.

UPPER PLAN

SECTION A-A

SECTION B-B

LOWER PLAN

SECTION C-C

SECTION D-D

FIGURE I-24. Closed Loop Piping for 5 MW Loop Cell Arrangement

FUEL HANDLING AND RADIOACTIVE MAINTENANCE - H. D. LENKERSDORFER

Fuel Handling Machine Component Test Stand - H. D. Lenkersdorfer, W. M. Noble, and J. F. Lubeck

Atomics International (AI) was provided a components test stand conceptual design (Figure I-25) for the FHM-interface to help guide their design effort.

Gas-Cooled Fuel Handling Analysis - A. Padilla

The preliminary calculations presented in BNWL-973 have been extended to utilize the most recent FFTF fuel design data. Table I-4 gives the differences between earlier design data.

The present study uses a chopped-cosine distribution for the axial heat generation. Also, the heat transfer coefficient was calculated by assuming a conservative value of 3.65 for the Nusselt number.

Figure I-26 shows the results of the present study. The helium coolant flow rate and the maximum helium velocity are plotted against the maximum fuel clad temperature for decay heat generation rates of 40 and 20 kW.

Table I-5 compares AI calculations based on the earlier design supplied

by PNL and the present study for a decay heat of 40 kW.

The lower pressure drop calculated for the present study is due to the assumption of a wire wrap instead of grid type spacer even though allowance has been made for an additional length of 24 in. for the fission gas plenum. The calculations indicate that the pressure drop for a subassembly with grid spacers is about one and one-half times that of a subassembly with wire wrap.

A large difference exists in the values of the calculated maximum clad temperature. This difference is due to the assumption regarding the axial heat flux distribution and also to the method of calculating the heat transfer coefficient. In the Atomics International work, the value of the heat transfer coefficient at the outlet was obtained from the Reynolds analogy and gave the maximum clad temperature of 700 °F in Table I-5. The magnitude of this coefficient is about one-half that used in the present study and essentially doubles the required temperature driving force between the coolant and the surface,

TABLE I-4. Basis for Gas Cooling Calculations

	<u>BNWL-CC-400</u>	<u>Present Study</u>
Pin Diameter, in.	0.226	0.221
Pitch, in.	0.264	0.2495
Fuel Length, in.	33.56	34.1
Pin Length, in.	35.56	58.1
Pins Per Subassembly	217	217
Axial Peaking Factor	1.0	1.24
Spacer Type	Grid	Wire Wrap

TABLE I-5. Comparison Between AI Calculations and the Present Study

	<u>AI</u>	<u>Present Study</u>
Decay Heat, kW	40	40
Helium Inlet Temperature, °F	250	250
Helium Outlet Temperature, °F	556	556
Helium Flow Rate, lb/hr	360	360
Maximum Clad Temperature, °F	700	598
Pressure Drop Through Subassembly, psi	7.3	6.6(a)
Maximum Helium Velocity, ft/sec	775	515(a)

(a) Based on an inlet pressure of 10 psig. For an outlet pressure of 0 psig, the maximum helium velocity is 736 ft/sec according to Figure I-5

FULL SCALE TEST STAND REQUIREMENTS:

1. PROVIDE SUFFICIENT SPACE AROUND TEST STAND FOR REMOVAL & REPLACEMENT OF COMPONENTS
2. TEST STAND SHALL PROVIDE FOR SUBCOMPONENT TESTS, DRY SUB SYSTEM TESTS, & WET SUB SYSTEM TESTS ON FULL SCALE INDIVIDUAL COMPONENTS OR FULL SCALE COMPLETE ASSEMBLIES TO SATISFY THE FOLLOWING CONDITIONS:
 - A. F.N.M. & REACTOR INTERFACE TEST
 - (1) F.N.M. & REACTOR INTERFACE ALIGNMENT REQUIREMENTS
 - (2) F.N.M. CLOSURE VALVE & FREEZE SEAL
 - (3) F.N.M. CLOSURE VALVE ACTUATING MECHANISMS
 - (4) F.N.M. DRIP PAN & SHIELD VALVE ACTUATING MECHANISMS
 - (5) F.N.M. SODIUM SYSTEM (VACUUM)
 - (6) F.N.M. SHROUDEL (SHORT LENGTH)
 - (7) REACTOR TUBE (SHORT LENGTH)
 - (8) REACTOR TUBE SHIELD VALVE
 - B. F.N.M. STRUCTURAL & COMPONENT TESTING
 - (1) F.N.M.
 - (2) F.N.M. INTERNAL MECHANISMS, VALVE, GRAPPLE, DRIP PAN & SHIELD VALVE
 - (3) F.N.M. ELECTRICAL INTERLOCKS & INSTRUMENTATION CONTROLS
 - (4) STRUCTURAL INTEGRITY
 - (5) COOLING SYSTEM FOR CASK, INTERNALS & FREEZE SEAL

FIGURE I-25. Components Test Stand for Wet Fuel Handling Machine

FIGURE I-26. Gas Cooling of Fuel Subassembly

thus substantially increasing the clad temperature at the outlet.

The present study shows that gas cooling is feasible at a decay heat generation rate of 40 kW if the maximum allowable clad temperature is not too low. For example, the required helium flow rate is 158 lb/hr and the maximum helium velocity is 440 ft/sec for a maximum clad temperature of 1000 °F. The maximum helium velocity becomes ex-

cessive for much lower clad temperatures; for a maximum clad temperature of 550 °F, the maximum velocity becomes 825 ft/sec. Note that the calculations apply to the ideal condition of no fuel blockage. Flow blockage, bowed pins, the mechanics of hooking up to the subassembly, and other practical considerations may determine that gas cooling is analytically possible, but technically infeasible.

FUEL EXAMINATION FACILITIES - C. L. BOYD

Inert Gas Cell Examination - C. L. Boyd

An assessment was made in December, 1966, of the interim examination capability that can be provided during a regular reactor outage period and during one reactor cycle (normal outage plus one reactor operating period). Figures I-27 and I-28 show an alternate inert gas cell arrangement that was prepared to provide additional space for the equipment needed to perform the operations listed in Table I-6 and Figure I-29. The cell arrangement differs from previous designs in the following areas:

- A deep cell is located between the inert gas cell bank and the fuel transfer cell to permit examination, handling, and cleaning of complete test assemblies.
- The equipment storage and transfer cell was enlarged to provide space for preliminary decontamination of equipment before removal from the cell and wall space for a personnel access door.
- The cell floor in the subassembly, disassembly, and reassembly area was lowered about 5 ft to provide clearance for vertical movement of subassemblies. This allows complex disassembly-reassembly operations to be performed on 10 to 12 ft long subassemblies in the viewing-manipulating zone directly in front of the viewing window.

Two factors influence the need for a deep inspection area:

- Limited inspection during a reactor outage can be done more quickly if the hanger rod or instrument tube is not removed and replaced.

- Separation of interim examination operations from reactor refueling operations in the fuel transfer cell will eliminate interference between these two operations.

Examination Requirements in the Functional Testing Requirements Study -

C. L. Boyd

To provide basic information needed for facility planning and conceptual design, a review of the contractor's input to the Functional Testing Requirements Study for the FFTF^(1,2) was made. Further analysis of this information showed that postirradiation examination support needed for the FFTF could be divided into the following phases:

<u>Examination Phase</u>	<u>Definition</u>
Limited interim examination	Remove from reactor, perform limited inspection and return to reactor during a scheduled outage.
Extended interim	Remove from reactor, perform detailed non-destructive inspection and return to reactor after one or more cycles.
Final nondestructive examination	Remove from reactor, disassemble, perform necessary nondestructive inspection, package, and ship to on-site or offsite radio-metallurgy facilities.

1. E. R. Astley. *Unpublished Data. Pacific Northwest Laboratory, Richland, Washington, July 15, 1966. (Personal Communications. FFTF Design Review and Alternate Concept Studies Functional Testing Requirements.)*
2. E. R. Astley. *Unpublished Data. Pacific Northwest Laboratory, Richland, Washington, September 19, 1966. (Personal Communications. Functional Testing Requirements Study.)*

FIGURE I-27. Inert Gas Examination Plan and Sections

FIGURE I-28. Inert Gas Examination Sections

TABLE I-6. Current Assessment of Maximum Examination Capability that can be Provided During Reactor Outage^(a)

Type of Assembly	Operation								
	Requires Cooling	Cleaning	Remove Hanger Rod	Remove Tube or Liner ^(b)	Visual Inspection	Photograph	Measure	Limited Pin Replacement ^(b)	Radiograph ^(c)
Driver or Prototype Fuel Bundle	Yes	Optional	No	Yes	Yes	Yes	?	No	Yes
Fuel Pin Bundle	Yes	Optional	No	Yes	Yes	Yes	?	No	Yes
Fuel Capsule (High Decay Heat)	Yes	Optional	No	Yes	Yes	Yes	?	No	Yes
Fuel Capsule (Low Decay Heat)	No	Optional	No	Yes	Yes	Yes	Yes	Yes	Yes
Material Coupons	No	Optional	No	Yes	Yes	Yes	Yes	Yes	Yes
Material Capsules	No	Optional	No	Yes	Yes	Yes	Yes	Yes	Yes

a. Assumed 2 wk outage period

b. Depends upon design

c. Radiography possible without removing the liner

FIGURE I-29. Current Assessment of Examination Capability During One Reactor Cycle (8 weeks)

Table I-7 shows the types of inspection specified for the majority of tests.

The types of inspection listed for the three phases of examination in Table I-7 have overlapping equipment and handling requirements.

Water Treatment System - D. J. Myers

The current Water Treatment Concept (74), see Figures I-30 and I-31 for the equipment list and flow diagram, includes a change in drain pump type. A submersible pump is now used to meet the high-suction lift requirements of

an above-water-level pump installation. The suction head required is approximately 35 ft of water (neglecting pressure drop) during the final basin draining stage.

Gas Cooling of Fuel During Inspection - J. M. Yatabe

A tentative limit of 600 °F (315 °C) had been set as the maximum allowable surface temperature during examination of discharged FFTF fuel assemblies in an inert gas atmosphere. This limit has been tentatively lowered to 500 °F (260 °C) in light of possible sodium

TABLE I-7. Types of Inspection Specified for Each Phase of Examination⁽¹⁾

<u>Limited Interim Examination</u>	<u>Extended Interim Examination</u>	<u>Final Nondestructive Examination</u>
Radiography	Radiography	Radiography
Gamma Scan	Gamma Scan	Gamma Scan
Remove Sodium (optional)	Remove Sodium (optional)	Remove Sodium
Remove Liner	Remove Liner	Remove Liner
Measure	Measure	Measure
Visual	Visual	Visual
Inspection	Inspection	Inspection
Photograph	Photograph	Photograph
Disassemble Pins ⁽²⁾	Disassemble Pins	Disassemble Pins
Inspect Pins	Inspect Pins	Nondestructive Inspection of Pins
Reassemble Pins ⁽²⁾	Reassemble Pins	Package for Shipment
Reassemble Liner	Reinstrument (limited)	
Clean	Reassemble Liner	
Reinsertion	Clean for Reinsertion	

(1) Compiled from Functional Testing Requirements Study.

(2) Requires extended interim examination unless operations are extremely simple and decay heat output is low.

EQUIPMENT LIST						
EQUIPMENT PIECE NUMBER	SYSTEM PIECE NUMBER	DESCRIPTION	SIZE (INCHES) (NORMAL) PRESSURE END	CONNECTION END	MATERIAL	REMARKS
RECIRCULATION						
F-1-1		FILTER			WOOL	STACKED CARTRIDGES
P-1-1		PUMP				CENTRIFUGAL
HE-1-1		HEAT EXCHANGER				30 KW @ 30 GPM
MBD-1-1		DEMINERALIZER				MIXED BED, 4 Cu Ft
DRT-1-1		DEPLETED RESIN TANK			C-STL	4 Cu Ft
RST-1-1		RESIN SUPPLY TANK			SST	4 Cu Ft
VI-1-1		VACUUM INDICATOR				0-30" Hg
TI-1-1		TEMP. INDICATOR				0-200° F
TI-1-2		TEMP. INDICATOR				0-200° F
TA-1-1		TEMP. ALARM-HI				85° F
PM-1-1		POISON MONITOR				0-30g-B ³ /LITER
PA-1-1		POISON ALARM-LOW				
FM-1-1		FLOWMETER				0-50 GPM
FM-1-2		FLOWMETER				0-20 GPM
QD-1-1		QUICK DISCONNECT				
QD-1-2		QUICK DISCONNECT				
QD-1-3		QUICK DISCONNECT				
QD-1-4		QUICK DISCONNECT				
QD-1-5		QUICK DISCONNECT				
V-1-1		VALVE, FOOT	2 1/2		SST	
V-1-2		VALVE, GATE	2 1/2			
V-1-3		VALVE, GATE	1 1/2			
V-1-4		VALVE, GATE	1 1/2			
V-1-5		VALVE, GATE	1			
V-1-6		VALVE, GATE	1			
V-1-7		VALVE, BALL	1			
V-1-8		VALVE, GATE	1			
V-1-9		VALVE, GATE	1			
V-1-10		VALVE, BALL	1			
V-1-11		VALVE, NEEDLE	1/4			
V-1-12		VALVE, BALL	1			
V-1-13		VALVE, BALL	1			
V-1-14		VALVE, BALL	1			
V-1-15		VALVE, BALL	1			
V-1-16		VALVE, BALL	1		SST	

EQUIPMENT PIECE NUMBER	SYSTEM PIECE NUMBER	DESCRIPTION	SIZE (INCHES) (NORMAL) PRESSURE END	CONNECTION END	MATERIAL	REMARKS
DRAIN & OVERFLOW						
QD-2-1		QUICK DISCONNECT			SST	REMOTELY OPERABLE
QD-2-2		QUICK DISCONNECT				
QD-2-3		QUICK DISCONNECT				
QD-2-4		QUICK DISCONNECT				REMOTELY OPERABLE
T-2-1		TRAP				50 MESH
F-2-1		FILTER			SST	50 MICRONS
P-2-1		PUMP				SUBMERSIBLE
OD-2-1		OVERFLOW DRAIN				
V-2-1		VALVE, FOOT			SST	
V-2-2		VALVE, GATE				
V-2-3		VALVE, GATE				
V-2-4		VALVE, GATE			SST	
SUPPLY & MAKEUP						
PST-3-1		POISON SUPPLY TANK			SST	
M-3-1		MIXER				
LLI-3-1		LIQUID LEVEL INDICATOR				
LLA-3-1		LIQUID LEVEL ALARM-LO				
FS-3-1		FLOAT SWITCH-LO			SST	
SV-3-1		SOLENOID VALVE			SST	N.C.
SV-3-2		SOLENOID VALVE				N.O.
MV-3-1		VALVE, MIXING				
V-3-1		VALVE, GATE				
V-3-2		VALVE, NEEDLE	1/4		SST	

LEGEND

	HEAT EXCHANGER
	ALARM, HORN
	FLOW METER OR INDICATOR
	INDICATOR
	PUMP, CENTRIFUGAL, OR OTHER ROTARY
	AGITATOR
	LIQUID LEVEL INDICATOR
	MIXING VALVE
	QUICK DISCONNECT
	GATE VALVE
	BALL VALVE
	SOLENOID VALVE
	CHECK VALVE, OR FOOT VALVE
	NEEDLE VALVE
	FLOAT SWITCH

NOTES
1. PIPING SHOWN ON SHEET 2 TO BE ALL WELDED CONSTRUCTION
WITH MECHANICAL JOINTS AT EQUIPMENT CONNECTIONS.

FIGURE I-30. Equipment List for Water Treatment System Process Flow Diagram

RECIRCULATION

DRAIN & OVERFLOW

SUPPLY & MAKEUP

FIGURE I-31. Process Flow Diagram for the Underwater Cell Examination Facility Water Treatment System

entrainment and vaporization into the gas stream. Test facility specifications have been changed to meet this new criteria, and the design of the facility has been started by the Laboratory Engineering Section.

The PNL Heat Transfer Laboratory -
J. M. Yatabe and V. R. Hill

The PNL heat transfer laboratory will be used for the test program and will serve as a first prototype of the FFTF fuel examination decay heat removal system. Scope design for the facility has been completed.

The major features of the facility include two separate gas blowers and ducting systems to independently provide axial or transverse coolant flow across the test fuel assembly during inspection and disassembly. In addition, any combination of axial and transverse flow may be provided to cool

the fuel assembly during liner removal. For basic parametric studies to determine the thermal-hydraulic effects of fuel assembly configuration (e.g., pin diameter, pin spacing, and spacers), bypass systems have been provided for both loops to permit optional operation at low flow rates (below 100 scfm).

Model of an FFTF Driver Fuel Assembly -
J. M. Yatabe and V. R. Hill

Design of an electrically heated model of an FFTF driver fuel assembly is about 20% complete. Basic design problems such as instrumentation of test sections, electrical power connections and materials of construction have been solved. However, the configuration of the test section (e.g., pin spacing, number of pins, spacer pitch) has been kept open due to fluctuations in the proposed configuration of the first FFTF driver fuel assembly.

ELECTRICAL SYSTEMS - G. H. STRONGOutside Power - T. W. Jeffs and
D. T. Thonn

Two one-line diagrams, Figures I-32 and I-33, have been completed showing 115 kV power supplies to FFTF. Associated switchgear and a plot plan are also shown on the drawings. The drawings are preliminary.

Primary Pump Drives - G. H. Strong
and T. W. Jeffs

Details of the primary pump drives are being developed. One attractive configuration consists of the following units, starting at the top: pony motor, thrust bearing, main motor, possibly a flywheel, eddy current clutch, thrust bearing, sodium seal, and pump. We have not been able to discover a standard pony motor which can meet the engineering speed requirements. We are now slightly modifying a standard motor so as to meet the requirements.

In the event of BPA power loss, present plans are to de-energize the eddy current clutch immediately and allow the speed of the pump and rotor of the clutch to decay at the maximum possible rate (i.e. by allowing the

motor and drum of the clutch to spin freely during this period) to match thermal power generation in the reactor as closely as possible until 25% pump speed is reached. The eddy current clutch will be gradually energized at that time (using the spinning kinetic energy of the motor and clutch drum) to hold pump speed at 25% until motor speed is reduced to 25%, at which time the pony motor will be connected to drive the pump. The pony motor will receive power from gas turbine generators. About 15 to 20 sec are required to bring 750 kW gas turbines up to operating speed. Preliminary calculations indicate that the kinetic energy of the main drive motor, pony motor, and drum of the eddy current clutch is sufficient to operate the pump at 25% speed for 38.5 sec, or a total of 50.2 sec following a scram. This includes the 12 sec required for the pump and eddy current clutch rotor to decay to 25% speed following loss of power. A 30.2 to 35.2 sec safety margin between the availability and the need of emergency power is thus provided.

FIGURE I-32. One Line Diagram and Plot Plan for 115 and 13.5 kV Electrical Power and Relaying System

RELAYING DIAGRAM
FOR CONTINUATION SHEET 1

FIGURE I-33. One Line Diagram and Plot Plan for 115 and 13.8 kV Electrical Power and Relaying System

SUMMARY
SECTION II

NUCLEAR AND SYSTEMS ENGINEERING - D. L. CONDOTTA

Fast Reactor Technology -

R. J. Hennig

A study was completed to determine the effects of test verticality on irradiation-test results in a dispersed FFTF core concept. Literature surveyed indicated only a few tests, principally involving molten fuel, were affected by test orientation. No adverse effects were reported in the BR-2 reactor which utilizes a skew angle as high as 10.7°.

Core and Systems Analysis -

R. E. Peterson

Coolant systems steady-state studies were continued to provide fundamental process design and cost data for conceptual evaluations and design descriptions. In addition, studies were completed to update previous results obtained for the number of parallel circuits, importance of reactor ΔT and significance of the log-mean ΔT for the IHX. Results confirm earlier estimates that four or five circuits yield lowest costs. Also, a reactor ΔT below 300 °F has an undesirable cost penalty and an IHX log-mean ΔT of 125 °F is not significantly more costly than a design for 150 °F.

Heat transfer performance of an IHX as a function of tube bundle bypassing was investigated. Five per-

cent bypass significantly lowers the apparent overall heat transfer coefficient.

Parametric studies were performed to investigate the effect of uncertainties in the fuel thermal conductivity; fuel centerline temperature, fissile fraction and pin diameter relationships; flux peaking factor, fissile fraction, and number of pins per subassembly relationships; and the effect of pin diameter versus flux for three set-points of fuel enrichment while varying core pressure drop.

Helium cooling requirements for a proposed rabbit capsule were estimated.

An analytical study to calculate the time response of ungrounded thermocouples was performed.

Dynamic simulation of the FFTF using the Dynamic Systems Analyzer (DYNSAR) Program is in progress.

A model for digital simulation of the FFTF reactor and heat removal system was formulated. Equations for the thermal response of inlet plenum and lower support plate were programmed and debugged. Programming of the primary loop portion of the model was completed.

Analog simulation of the FFTF reactor and heat removal systems continued. Improvements were made in component analytical models. Debug runs were continued with the systems model;

an oscillation due to pin bowing feedback was discerned.

Analog runs investigated natural coolant circulation with varying height of intermediate heat exchanger (IHX) and dump heat exchanger (DHX) above the reactor core and loss of system power and scram simulated in the reactor transient response. All combinations of 5, 10, and 20 ft for IHX location above the reactor and DHX above the IHX were investigated.

Eight programs have been developed for the PDP-7 computer for use with the FFTF Dynamics Hybrid Simulation.

Four computer codes, SAFE-SHELL, SAFE-PLANE, SAFE-PCRS, and AXI-SYM, have been converted to the UNIVAC 1107 and are available for general use. These programs are oriented towards analyses of heterogeneous structures such as reinforced or prestressed concrete vessels.

A study of the mechanical loads in the various FTR tube concepts was completed.

Reactor Safeguards - D. E. Simpson

Analysis of single-channel meltdown scram requirements was made. It was concluded that the design selected should be capable of core subdivision to at least eighty fuel assemblies; this quantity does not place a severe demand on the reactor safety system in the event of a single channel meltdown. Eighty assemblies is not a minimum number, and the final optimum may be between 50 and 80.

Instrumentation and Control -

C. D. Swanson

Sheathed thermocouples for the EBR-II capsule have been analyzed and radiographically tested.

A system for detecting the onset of central melting in UO_2 and $(\text{U,Pu})\text{O}_2$ fuels was designed.

Equipment is being fabricated to permit tests for studying immediate effects of intense gamma and neutron flux in cables and insulators.

Mechanisms which govern the emf generated in thermocouples have been studied. Preliminary analysis indicates that a high radiation field (neutrons or gamma photons) can alter the thermocouple. The theory as it is now formulated is not precise enough to predict the flux level at which calibration changes become significant.

The schedule for the Central Control and Data Handling (CC&DH) System shows that the specification development for major digital components should be completed by April 1968. A similar schedule developed for programming activities indicates that general program design should be started early in FY-1968. The evaluation of candidate CC&DH concepts has been postponed until March.

Control room arrangement drawings have been prepared and distributed for comment. These drawings show space requirements for typical analog and digital control concepts. Provision is made for experimenters' equipment rooms.

The A/D and D/A converters are ready for use in the hybrid facility. A set of programs for the PDP-7 is now being developed for FFTF and general simulation use.

FFTF Reactor Physics - *P. L. Hofmann*

A survey has been performed to investigate the effect of BeO on FTR neutronics. Items considered included the sodium coefficient, Doppler coefficient, critical mass, and flux spectrum dependence on BeO volume fractions.

The effects of fuel failure and fuel relocation (slumping) in the FTR are being re-examined. A series of kinetics calculations have been performed for a range of reactivity ramps and shutdown coefficients. The possible use of fuel-rod plugs to minimize the effects of fuel slumping has also been considered.

To check calculational techniques used in FTR design, analysis of the Doppler coefficient measurements in the SEFOR mockup in ZPR-III is being performed. Incorporation of crystalline binding effects in the analysis caused the Doppler coefficient to diverge from the $1/T$ dependence in qualitative agreement with the experimental results.

The Phase-A critical experiments in ZPR-III have been completed, and analysis of the experiments has started.

The attenuation characteristics of a number of alternate shield arrangements have been compared analytically. The removal-diffusion code MAC was used for these calculations.

Fast Reactor Fuels - *W. E. Roake*

A seven-rod cluster was water flow tested over a range of flow con-

ditions matching the range of Reynolds Numbers to be experienced in the FTR. Results, converted to pressure drop in flowing sodium, show that pressure drop at maximum flow conditions would be less than 30 psi across the 24 in. long cluster.

Creep data for 304 SST were evaluated to obtain both primary and secondary creep rates. Empirical functions were integrated to provide total creep strain over different stress histories during a 5000 hr period.

A mixed oxide FFTF core may require cylindrical restrictors in the fuel pins to limit axial fuel movement during excursions. Several potential restrictor materials are being tested to determine fuel-restrictor and cladding-restrictor compatibility.

Design of test capsules for fuel motion restrictors was completed. These capsules will be rapidly inserted into the VH-4 hydraulic rabbit facility of the MTR at power, thus simulating a reactor transient. Restrictor materials to be tested include tungsten, molybdenum, nickel, 304 SS, and depleted UO_2 .

The diameter of the floating restrictors for the VH-4 screening capsules was reduced to prevent restrictor-cladding contact. The irradiation test capsules that will contain these restrictors are designed to generate 28 kW/ft. An analysis of the possible use of PRTR for thermal flux irradiation testing of FFTF fuel indicated that the FERTF has a flux level high enough to make it an attractive test facility.

A capsule design for the MTR tests was approved for fabrication. Three fuel forms, of three stoichiometries,

will be irradiated at 16 kW/ft to exposures of 50,000 and 100,000 MWD/tonne.

Two cermet pins were irradiated in the EBR-II subassembly X009. Two others, (PuO_2 -SS and UO_2 -SS) are being irradiated in EBR-II subassembly X011.

Neutron radiography and gamma scanning of two PuO_2 -SST cermet pins that were irradiated in EBR-II subassembly X009 was completed.

Meetings between BNW personnel and AEC-DRDT Headquarters personnel, on January 24, and EBR-II Evaluations Committee, on February 1, were successful in advancing plans for EBR-II irradiation tests. As a result of these meetings, detailed designs for specific tests can be prepared. Approval in principle for the use of the TREAT facility was obtained from Argonne National Laboratory.

Fuel pin temperatures are being computed for steady state and transient conditions. MANTA, a thermal hydraulic code, was converted for operation on the 1107 computer. This code will aid in predicting the coolant temperature and flow distribution in a fuel bundle.

An oxide fuel temperature model which includes the effect of sintering was developed. The model will aid fuel slumping analysis by allowing more accurate computations of fuel temperatures and the extent of the core void region at equilibrium operation.

Seven fuel-cladding compatibility test capsules were sodium bonded. De-

structive examination of the first three capsules showed excellent wetting of both fuel and cladding.

Nondestructive testing was initiated on a heat of 304 SS tubing (0.250 in. OD, 0.016 in. wall) for MTR/ETR irradiation test pin cladding. The tests resulted in such a high rejection rate that this heat is considered suspect. Correlation between nondestructive testing results and destructive testing has improved, but is still not on a high confidence level. A semi-automatic gaging machine was developed for continuous measurement and recording of inside and outside diameters of cladding for FFTF irradiation test pins.

Vibrational compaction tests were begun by using substoichiometric UO_2 -25% PuO_2 fuel in 0.220 in. ID pins. The fuel was compacted to 82 to 84% TD.

Three hundred and fifty UO_2 -25 wt% PuO_2 pellets were fabricated for the MTR-ETR irradiation test series. Four hundred depleted UO_2 pellets were fabricated. One of these pellets will be placed at each end of the fuel column to reduce end cap temperatures. Vibrational compaction tests are being conducted to improve compactability and powder utilizations.

A preliminary analysis of FFTF hot channel factors was completed. The 3 σ variance (99.87% confidence) in the hottest channel was calculated to be 423 °F, resulting in an engineering power multiple of 1.14.

SECTION IINUCLEAR AND SYSTEMS ENGINEERING*D. L. CONDOTTA*FAST REACTOR TECHNOLOGY - R. J. HENNIGVerticality Requirements in Irradiation Testing - R. J. Hennig

For some time Battelle-Northwest has been engaged in design studies of the Fast Flux Test Facility with major emphasis centering on skewed core reactor concepts. Concern has been expressed about the applicability of irradiation tests conducted in nonvertical facilities to vertical cored LMFBF's.

A study was undertaken to evaluate the effects of test attitude on the results of irradiation testing. The basic question to be answered was, "Are the results of irradiation testing affected in any way by small angles of tilt of the test or the test bed?" And, if so, "To what degree, and what can be done about it?" The study was composed of three main parts:

- A survey of the literature and experience at various sites
- A request to all contractors participating in the FFTF Functional Testing Requirements Studies to specify and justify verticality requirements for all proposed tests
- The definition and analysis of the various problems postulated for nonvertical testing.

Some pertinent aspects of the study are summarized below.

Literature and Experience Survey

An extensive literature survey revealed voluminous data on irradiation

testing of a very wide range of materials in a variety of shapes and forms, in many different types of test facilities and rigs, and in both horizontal and vertical attitudes. The vast majority of the data indicated no effects attributable to test attitude. In a few tests, involving either accidental or purposeful melting of fuel, it seems certain that fuel relocation was significantly (though not solely) affected by gravity and, thus, test attitude. Possibly the most pertinent of these data is some recent work by Battelle-Northwest involving high power irradiations of oxide fuels in the horizontal position. In these tests the centers of the molten zones were displaced upward by amounts ranging from 4 to 12% of the rod radii.

The Belgian Material and Engineering Test Reactor, BR-2, is a high flux machine, which utilizes a skewed core arrangement. The center tube is vertical and the rest of the tubes are inclined at angles of 3.5° to 10.7°. The Belgians have discovered no adverse effects from tilted irradiation tests.

Evaluation of Postulated Effects of Skewing

The effects of skewing on neutron flux and power distributions were analyzed. The principal effect is a slight power peaking; in conical and bottom-skewed cores, there is a small downward shift in power. At the extremes, these power and flux perturbations are comparable to those caused

by nominal changes of the axial reflectors, nearby tests, or distant control rods.

Most of the problems postulated for tilted testing are based on changes in test gravity forces. At the tilt angles of interest, axial forces are reduced 2 to 5%; transverse forces may range up to 20 to 30% of the vertical gravity force. Fast reactor fuel pins weigh a few tenths of a pound per foot of length; spring-grid type fuel pin spacers provide lateral support forces which are two orders of magnitude larger than the lateral gravity forces. Similarly, the measured gap heat transfer coefficients of gas-bonded fuels indicate interface forces about three orders of magnitude larger than the lateral gravity forces.

In sodium-bonded fuels, pellets tend to line up along the lowest generatrix of the clad. The maximum expected eccentricity will cause the surface temperature to deviate less than 20 °F from that of an ideal concentric pellet (central temperatures are not measurably affected). In vertical cores, manufacturing tolerances, irradiation effects, etc. will cause pellet-clad contact, thus yielding similar deviations in fuel surface temperatures. Furthermore, these contact forces will be large compared to gravity forces (which are probably too small to break the sodium film); thus inclining the fuel is not expected to alter fuel-clad chemical interactions significantly. Fission gas bubbles rising to the upper plenum will tend to follow a single preferred path in inclined fuels versus a radially random path in vertical fuels; however,

bubble rise rates should be essentially the same in both cases, and no clad temperature differences are expected.

Molten alloy fuels, molten-center ceramic fuels, and liquid metal bonded fuels may have important phenomena occurring at the liquid-gas and liquid-gas-solid interfaces. At the tilt angles of interest, the liquid-gas interface area increases only 2 to 5%; the liquid-gas-solid interface perimeter increases only 1 to 2%.

Under normal forced-flow operating conditions, circuit hydraulic characteristics are independent of the attitudes of various portions of the systems. Even with sodium boiling, the maximum perturbation is only a few tenths of a percent of pump induced forces. At very low flow rates, the problem is more complex. Under normal conditions of natural convection, bulk flows will be essentially the same in vertical and tilted elements, although local flow patterns may be altered slightly. In the extreme case of boiling with no flow, it seems certain that local flows will be altered. It is not clear at which point between normal convective flow and complete flow blockage that the effect of tilting becomes important.

It was concluded that the vast majority of tests postulated for the FFTF will not be affected by inclinations of interest. However, test attitude may affect local flow patterns at very low coolant flow rates and may affect relocation of melted fuel. The few percent of postulated tests wherein such phenomena are important may require verticality within 5 to 10°.

CORE AND SYSTEMS ANALYSIS - R. E. PETERSONCoolant Systems Analysis -

H. G. Johnson

Coolant systems steady-state studies were continued to provide fundamental process design and cost data which can be used for the preparation of the heat removal systems design descriptions. Results are reported for loop concepts in four areas, and IHX process design for the main heat removal system.

Construction Cost Studies for the Main Heat Removal System

Construction cost studies have been made to update previous studies on the optimum number of parallel cooling circuits, importance of reactor core temperature rise, and the significance of the log-mean-temperature difference for the intermediate heat exchanger. In addition, costs were developed for several new system cases covering a variety of power levels, primary coolant temperatures, and redundancy situations.

The ADOPT Code⁽¹⁾ was used for the main heat removal system costs for the external primary loop concept with basic design input data as listed in Table II-1 for all of the reported studies. Incremental costs for systems other than the main heat removal system were developed independent of the ADOPT Code.

The only significant and practical incremental design development cost appears to be related to the proof testing of pumps, large valves, and the IHX. This incremental cost was estimated for each case as 10% of the cost of an uninstalled IHX, set of primary and secondary valves, primary pumps, and secondary pump. The pump costs include drives.

In evaluating the performance of a given design at various power and temperature conditions, the assumption was made that whenever a cooling circuit is not used, it is completely isolated with no primary coolant flowing through the IHX.

Optimum Number of Cooling Circuits.

Redundant and nonredundant designs were considered. Two through six circuits were considered for the nonredundancy case, and three through seven circuits were considered for the case of one redundant circuit.

Process designs were based on a 400 MW total heat removal capability with primary coolant hot leg bulk temperatures of 1050 to 1100 °F. The reactor vessel pressure loss allowance was set at 100 psi. The bulk temperature changes in the primary and secondary coolants were held constant at 300 °F. IHX log-mean-temperature differences of 100, 125, 150 °F were considered for all cases. Figure II-1 shows the cost data. Four or five circuits give the least cost in the case of redundancy. The exact number depends on the LMTD of the IHX, but there is little difference between

1. C. L. Goss and H. G. Johnson. ADOPT, Program for Steady-State Process Design and Cost Optimization of FFTF Heat Removal Systems Using Heat Dumps to Air, BNWL-308, Pacific Northwest Laboratory, Richland, Washington, September 1966.

TABLE II-1. Basic Design Input Data for
the Main Heat Removal System

IHX

Tube OD, in.	1
Tube Wall, BWG	16
Tube Pitch:	
Type	Equilateral triangular
Spacing, in.	1.375
Effective Tube Length, ft	16
Fouling	None
Number of Passes	
Shell side (primary)	1
Tube side (secondary)	1
Amounts of Tube Bundle Bypassing:	
Shell side primary coolant, %	5
Tube side secondary coolant, %	2.5

PRIMARY PIPING ARRANGEMENT

	Reactor Vessel to IHX	IHX to Pump	Pump to Vessel
Straight Pipe, ft	60	35	65
Number of Fittings and Valves			
90° Long Radius Elbows	2	1	2
90° Short Radius Elbows	2	2	2
Check Valves	0	0	1
Sudden Expansions	1	0	1
Sudden Contractions	1	1	0

PRIMARY PIPE CONTAINMENT

Percent Primary Pipe Contained 28

SECONDARY PIPING ARRANGEMENT

	IHX to DHX	DHX To Pump	Pump to IHX
Straight Pipe, ft	200	125	100
Number of Fittings and Valves			
90° Long Radius Elbows	4	3	5
90° Short Radius Elbows	4	2	0
45° Elbows	2	1	2
Gate Valves	2	0	2
Sudden Expansions	0	1	1
Sudden Contractions	1	0	0

Pumps

Type	Free Surface Centrifugal
Location	Cold Legs
Speed, Rpm	
Primary	870
Secondary	1170
Submergence, ft	
Primary (Distance impeller below level coolant in re- actor vessel)	10
Secondary	5
Drive Motor: Wound rotor induction	(50 to 100% Speed Range)

DUMP HEAT EXCHANGER

Type:	Forced Convection, Cross Flow Air
Material	
Tube	304 SS
Fin	304 SS
Fan type	Axial flow
Inlet air temperature, °F	100

MAXIMUM ALLOWABLE REACTOR VESSEL
PRESSURE, psig

1

FIGURE II-1. Cost Optimization of the Number of Main Heat Removal System Cooling Circuits

four and five circuits. These results are essentially the same as those reported previously⁽¹⁾ as far as the selection of the optimum number of circuits is concerned.

Reactor Core Temperature Rise

This construction cost study was made for four installed cooling circuits, including one redundant circuit, at a total system heat load of 400 MW with a constant 1050 °F primary coolant hot leg bulk temperature, and over

a range of core bulk temperature increases of 200 to 400 °F.

The study was not made for a fixed fuel design but for a varying reactor vessel design. The reactor vessel pressure loss was assumed to be 100 psi, consisting of 80 psi core loss and 20 psi for other losses, at a 300 °F bulk temperature rise through the core. The core loss was then taken as being proportional to the coolant flow rate squared for other temperature rise cases, but the 20 psi loss was kept constant.

Two alternates were considered for the secondary coolant conditions. In one case, the secondary coolant temperature change was set equal to the primary coolant change while in the other case, the secondary coolant temperature change was held constant at 300 °F. The IHX log-mean-temperature difference was constant at 125 °F for all situations. The data indicate an important cost advantage for higher core temperature changes. Results further indicate that temperature changes less than about 300 °F have an undesirable cost penalty. Finally, the preferred alternate is the one with secondary temperature changes equal to the primary coolant temperature changes.

IHX Log-Mean Temperature Difference

Previous studies⁽¹⁾ indicated that the optimum LMTD for the IHX was over 150 °F when the IHX was coupled with a dump heat exchanger designed with carbon steel fins and stainless steel tubes. This combination of materials may not be suitable for the FFTF service.⁽²⁾ Recent studies, including the ones reported here, have been based on stainless steel fins on stainless steel tubes, which results in an increased unit cost of the DHX because of poorer heat transfer and greater unit cost of materials. It was thought that this change to stainless steel fins might cause an important shift

2. D. W. Shannon and S. M. Gill. Unpublished data. Pacific Northwest Laboratory, October 14, 1966. (Selection of Materials and Construction of Finned Tubes for the FFTF Air Dump Heat Exchangers, Memorandum to J. M. Davidson.)

downward in the optimum LMTD for the IHX, but this is not the case. Data for the redundant designs using four and five installed circuits are plotted in Figure II-2. As shown, the incremental construction cost decreases with increasing LMTD (for the IHX) out to 150 °F. However, the curves are quite flat over the range of 125 to 150 °F such that the 125 °F value is to be preferred over 150 °F because of mechanical design considerations. The IHX costing was done with the use of a constant unit surface cost, which does not allow for differences in the mechanical design to tolerate large LMTD values.

FIGURE II-2. Incremental Construction Cost as a Function of the IHX Log-Mean Temperature Difference

Comparison of Construction Costs for
Various Process Design Requirements

Table II-2 defines the thirteen cases considered in this study. The initial conditions dictated heat exchanger sizes while ultimate conditions set pipe and pump sizes for the first four cases with the 300 MW initial power and 400 MW ultimate power. A maximum steady-state IHX log-mean temperature difference of less than 125 °F was used in this part of the study. Thus, the initial condition LMTD values had to be less than 125 °F so that the LMTD criterion would not be violated at the ultimate condition. The initial phase design value used was 100 °F and, as shown in Table II-2, the calculated ultimate phase LMTD was only 120 °F, but this check is considered to be good enough for the purpose of this work.

Table II-3 gives the cost data for cases 1 through 4. Redundancy increases the direct construction costs by \$2,491,000 and \$2,097,000 for the four- and five-circuit cases, respectively. These increases are equivalent to incremental capital investments of about \$4,100,000 and \$5,000,000. The data also verify earlier conclusions that there is little cost difference between the four- and five-circuit cases when one redundant circuit is specified.

Because initial conditions dictate heat exchanger component sizes, the heat exchanger components for cases 1 through 4 are larger than they need be to meet the given ultimate phase conditions. Thus, the ultimate phase primary temperatures can be less than the specified 1050/750 °F. The calculated minimum possible primary cool-

ant hot leg bulk temperature is 975 °F with a 675 °F cold leg temperature.

Table II-4 reports the cost data for the series of 400 MW design cases, numbers 5 through 12. Except for cases 8 and 12, initial phase conditions again dictate heat exchanger sizes and costs while ultimate phase conditions dictate pipe and pump sizes. This is also true for cases 7 and 11, wherein a redundant circuit is provided at the ultimate condition but not at the initial condition. For case 7, it turns out that the heat exchanger equipment provided to meet initial conditions allows operation with three of the four circuits in operation with a minimum primary hot leg temperature of 955 °F and cold leg temperature of 655 °F. Similar temperatures for case 11 are 910 °F and 610 °F. However, cases 7 and 11 have less temperature capability than cases 5, 6, 9 and 10 in that these latter four cases require only 800 °F hot leg primary temperature at the ultimate conditions. Table II-4 includes the minimum required primary coolant bulk temperatures at ultimate conditions.

The data in Table II-4 can be used to arrive at the cost of the various forms of redundancy. The data also indicate very little difference in the cost of four and five circuits under comparable conditions when redundancy is provided. In the four-circuit designs, redundancy for both the initial and ultimate conditions (i.e., case 6) costs about \$2,948,000, whereas redundancy for the ultimate operation (i.e., case 6) only costs about

TABLE II-2. Process Design Conditions for Heat Removal Systems

Case Number	Total System Heat Load, MW		Number Parallel Circuits Total Installed	Redundant		Primary Coolant Bulk Temperatures, °F				Design LMTD for IHX, F(1)	
	Initial	Ultimate		Initial	Ultimate	Hot Leg		Cold Leg		Initial	Ultimate
						Initial	Ultimate	Initial	Ultimate		
1	300	400	4	0	0	800	1050	500	750	100	(120)
2	300	400	4	1	1	800	1050	500	750	100	(120)
3	300	400	5	0	0	800	1050	500	750	100	(120)
4	300	400	5	1	1	800	1050	500	750	100	(120)
5	400	400	4	0	0	800	1050	500	750	125	(121)
6	400	400	4	1	1	800	1050	500	750	125	(121)
7	400	400	4	0	1	800	1050	500	750	104	(<125)
8	400	400	4	0	1	(2)	1050-1100	(2)	750-800	<125	125
9	400	400	5	0	0	800	1050	500	750	125	(121)
10	400	400	5	1	1	800	1050	500	750	125	(121)
11	400	400	5	0	1	800	1050	500	750	108	(<125)
12	400	400	5	0	1	(2)	1050-1100	(2)	750-800	<125	125
13	500	500	5	0	1	(2)	1050	(2)	750	<125	125

1. LMTD values in parentheses are calculated based on the other specified design value

2. Initial phase temperatures not specified. Will accept whatever temperature possible with equipment designed for ultimate conditions. See text for these values.

TABLE II-3. Direct Construction Cost Data for Heat Removal Systems and Related Systems Designed for an Initial Operation at 300 MW with 800/500 °F Primary Coolant Bulk Temperatures and for an Ultimate Operation at 400 MW with 1050/750 °F Primary Coolant Bulk Temperatures

	Case Number			
	1	2	3	4
Number parallel cooling circuits	4	4	5	5
Number redundant circuits:				
Initial conditions	0	1	0	1
Ultimate conditions	0	1	0	1
IHX Log Mean Temperature Difference, °F:				
Initial conditions	100	100	100	100
Ultimate conditions (approximate)	120	120	120	120
Minimum possible primary coolant bulk temperature at ultimate conditions (hot/cold leg), °F	975	975	975	975
Direct construction costs as 1,000 dollars:				
IHX complex/circuit	\$ 417	\$ 517	\$ 353	\$ 417
DHX complex/circuit	824	1,079	641	824
Primary loop/circuit	475	586	422	475
Secondary loop/circuit	439	530	386	439
Subtotal/circuit	\$ 2,155	\$ 2,712	\$ 1,802	\$ 2,155
Incremental electrical supply system/circuit	66	83	45	66
Incremental cost for other closely related systems/circuit ⁽¹⁾	319	319	309	319
Subtotal/circuit	\$ 2,540	\$ 3,114	\$ 2,156	\$ 2,540
Subtotal cost for all circuits	10,160	12,456	10,780	12,700
Incremental cost for all other related systems ⁽²⁾	524	560	730	755
Uninstalled spares for heat removal system	898	1,044	762	901
Subtotal increment of direct construction cost	\$11,582	\$14,060	\$12,272	\$14,356
Incremental design and development	78	91	65	78
Total	\$11,660	\$14,151	\$12,337	\$14,434

(1) Includes instrumentation, inert gas, sodium, safety systems.

(2) Includes reactor vessel, building and structure, equipment maintenance systems.

TABLE II-4. Direct Construction Costs Data for Heat Removal Systems and Incremental Costs of Related Systems for 400 MW Heat Load at Initial Condition Primary Coolant Bulk Temperatures of 800/500 °F and/or to Allow Ultimate Condition Temperatures of 1050/750 °F.

	Case Number							
	5	6	7	8	9	10	11	12
Number installed parallel cooling circuit	4	4	4	4	5	5	5	5
Number redundant circuits:								
at initial condition	0	1	0	0	0	1	0	0
at ultimate condition	0	1	1	1	0	1	1	1
Minimum possible primary coolant bulk temperatures at ultimate conditions with redundancy °F (hot/cold)	800/500	800/500	955/655	1050/750	800/500	800/500	910/610	1050/750
Direct construction cost as 1000 dollar								
IHX complex/circuit	\$ 415	\$ 514	\$ 503	\$ 498	\$ 351	\$ 415	\$ 408	\$ 402
DHX complex/circuit	1,105	1,446	1,110	1,009	876	1,105	876	761
Primary loop/circuit	475	585	583	587	422	475	471	475
Secondary loop/circuit	427	516	489	513	375	427	432	427
Subtotal/circuit	\$ 2,422	\$ 3,061	\$ 2,685	\$ 2,607	\$ 2,024	\$ 2,422	\$ 2,187	\$ 2,065
Incremental electrical supply system/circuit	74	96	91	79	60	74	65	57
Incremental cost for other closed related systems/circuit (2)	329	349	339	339	329	349	319	319
Subtotal/circuit	\$ 2,825	\$ 3,506	\$ 3,115	\$ 3,025	\$ 2,413	\$ 2,845	\$ 2,571	\$ 2,441
Subtotal parallel circuits	11,300	14,024	12,460	12,100	12,065	14,225	12,855	12,205
Incremental cost for all other related systems (3)	524	560	560	560	730	755	755	755
Uninstalled spaces for heat removal system	860	1,032	993	1,011	753	863	859	850
Subtotal incremental direct construction	\$12,684	\$15,616	\$14,013	\$13,671	\$13,548	\$15,843	\$14,469	\$13,810
Incremental design and development	74	90	86	88	63	74	74	73
Total	\$12,758	\$15,706	\$14,099	\$13,759	\$13,611	\$15,917	\$14,543	\$13,883

(1) For 300 °F temperature rise on primary and secondary coolants and 100 °F inlet air.

(2) Includes instrumentation, inert gas, sodium, safety systems.

(3) Includes reactor vessel, building and structure, equipment maintenance systems.

\$1,341,000. These are direct construction costs, so equivalent capital investments run about \$4,900,000 and \$2,200,000, respectfully. The comparable capital investments for redundancy for the five-circuit designs are \$3,800,000 and \$1,500,000 for cases 10 and 11.

Requirements for cases 8 and 12 are quite similar to the requirements for cases 7 and 11, respectively. The difference is in the specification of required initial condition temperatures. Cases 7 and 11 have the definite 800 °F primary coolant bulk hot leg temperature requirement, whereas cases 8 and 12 are designed for ultimate conditions, and the performance which can be achieved with the use of equipment so designed is accepted at the initial condition. Figures II-3 and II-4 show the performances of heat exchange equipment designed to the case 7 and 11 conditions. Performances for cases 8 and 12 are shown in Figure II-5. Comparison of cases 7 and 8 then shows that the greater capability of case 7 is achieved for a capital investment of about \$560,000. The similar capital investment difference for the five-loop situations, described by cases 11 and 12, is \$1,100,000.

Table II-5 lists cost data for the 500 MW case number 13, and Figure II-6 illustrates performance of the system at reduced powers with and without the use of the redundant circuit. A comparison may be drawn between cases 12 and 13, which are fairly comparable, to determine the cost of providing an extra 100 MW heat removal capability using five installed circuits. The

incremental direct construction cost for the added 100 MW of heat removal capability is about \$2,280,000 which is equivalent to a capital investment of over \$3,700,000.

Effect of Tube Bundle Bypassing of Intermediate Heat Exchangers Designed with Removable Tube Bundles

The intermediate heat exchangers may be designed with easily removable tube bundles to facilitate maintenance repairs. However, such designs introduce the probability of significant tube bundle bypassing which can result in poor apparent heat transfer performance as indicated by low overall heat transfer coefficients. Low coefficients require high log-mean temperature differences for given surface and heat duty. Hence, the low coefficients become a problem only if the temperature differences between the shell and tube side coolants become high enough to cause mechanical design problems.

A study was made to show how the apparent heat transfer performance of an IHX is influenced by tube bundle bypassing. Data were calculated for isothermal and nonisothermal flow of the bypassing coolants around the tube bundle followed by complete mixing of the main stream and bypassing coolant before entering the outlet nozzle. Actually, heat exchange will occur between the main stream and bypassing coolant so that the isothermal model yields worst case limiting conditions.

Considered were bypassing amounts of from zero to 5% for the following assumed fixed conditions.

Case 7 Design

FIGURE II-3. Performance of a Four Circuit Heat Removal System Which is Designed for 400 MW at 800 °F Primary Hot Leg Temperature When Using All Four Circuits

Case 11 Design

FIGURE II-4. Performance of a Five Circuit Heat Removal System Which is Designed for 400 MW at 800 °F Primary Hot Leg Temperature When Using All Five Circuits

Cases 8 And 12

FIGURE II-5. Performance of a Heat Removal System Designed for 400 MW at Primary Coolant Hot Leg Temperatures of 1050 - 1100 °F

TABLE II-5. Construction Cost Data for 500 MW Heat Removal System at 1050/750 °F Primary Coolant Bulk Temperatures with Five Installed Cooling Circuits Including One Redundant Circuit (Case Number 13)⁽¹⁾

Construction Costs as 1,000 Dollars:

IHX complex/circuit	\$ 474
DIHX complex/circuit	877
Primary loop/circuit	568
Secondary loop/circuit	533
Subtotal/circuit	\$ 2,452
Incremental electrical supply system/circuit	86
Incremental cost for other closely related systems/circuit ⁽²⁾	319
Subtotal/circuit	\$ 2,857
Subtotal cost for five circuits	14,285
Incremental cost for all other related systems ⁽³⁾	787
Uninstalled spares for heat removal system	1,004
Subtotal incremental direct construction cost	\$16,076
Incremental design and development cost	87
Total	\$16,163

(1) Based on IHX log mean temperature difference of 125 °F.

(2) Includes instrumentation, inert gas, sodium, safety systems.

(3) Includes reactor vessel, building and structure and equipment maintenance systems.

FIGURE II-6. 500 MW Heat Removal System Performance With and Without the Use of the Redundant Cooling Circuit

IHX surface, sq ft	2,300
Tube OD, in.	1
Tube gage, BWG	16
Tube length, ft	16
Heat duty, Btu/hr	113.5×10^6
Shell side fluid	Primary sodium
Tube side fluid	Secondary sodium
Number shell side passes	1
Number tube side passes	1
Type IHX	Vertical, counter flow, unbaffled
Temperatures at nozzles, °F	
Shell inlet	800
Shell exit	500
Temperature rise secondary coolant across nozzles	75

These temperature conditions correspond to a maximum flow of secondary coolant and a reduced flow of primary coolant for a fractional power type operation.

The overall heat transfer coefficients were computed (1) for the tube bundle (based on the nonbypassing streams flowing uniformly through the bundle) and (2) for the entire IHX. The latter is the apparent value for the IHX based on temperatures at the nozzles and corresponds to the normally measured coefficient for an exchanger installed in a facility such as the FFTF. The log-mean-temperature differences based (1) on bundle inlet and exit temperatures and (2) on IHX nozzle conditions were also calculated along with secondary coolant temperatures. These calculated values are listed in Table II-6 for the bypassing conditions shown there and as noted for the isothermal or non-isothermal models. The nonisothermal conditions result from a rough approximation of the heat transfer through the shroud shell which separates the main shell stream and shell side bypassing coolant.

The very low apparent coefficients for the bypassing situations should be noted.

Fuel Parametric Analyses - J. C. Draper

An expansion of the studies on the PNL-II concept, with emphasis on parameters that have not previously been closely examined was made at the Systems Analysis and Simulation hybrid facility. These parametric studies place in sharper focus the need for more detailed studies.

TABLE II-6. Performance of an IHX with Varying Amounts of Tube Bundle Bypassing

Bypassing Model	Bypassing %		Coolant Temperatures, °F						Log Mean ΔT , F		Overall μ	
	Shell Side	Tube Side	Tube Side			Shell Side			Tube Bundle	Apparent for IHX	Btu (hr-sq ft-F) Bundle	Apparent
			Inlet Nozzle	Exit Nozzle	Exit Bundle	Exit Bundle	Bypass	Exit				
--	0	0	498	573	573	500	800		44.5	44.5	1110	1110
Isothermal	3	0	488	563	563	490	800		44.8	75.5	1100	655
Isothermal	5	0	483	558	558	484	800		45.2	84.8	1090	582
Isothermal	5	2.5	483	558	560	484	800		45.2	84.5	1090	583
Nonisothermal	5	0	490	565	565	492	652		45.2	71.2	1090	694

The effect of the uncertainty of the fuel thermal conductivity was studied by multiplying a percentage factor (0.8 to 1.2) by the linear maximum power (Figures II-7 and II-8). This gave the effect of increasing or decreasing the thermal conductivity by 20%. The Linear Maximum Power percentage factor was swept for different setpoints of f_{PuO_2} (0.20, 0.22, and 0.25). The results of these parameter studies are on the X-Y plots and the strip chart recordings included in this report.

A study on the centerline temperature parameter showed that for a larger fissile fraction (0.22 and 0.25), the pin diameter can be held below 0.025 ft for a centerline temperature as high as 6000 °F (Figures II-9 and II-10). A fissile fraction of 0.20 reached the 0.025 ft pin diameter at a 5000 °F centerline temperature. The program showed the expected marked decrease in the number of pins per subassembly as the centerline temperature was increased. An example of this is that for the fissile fraction of 0.20, the number of pins per subassembly was 217 at 4000 °F centerline temperature and about 117 at 5000 °F. The increasing centerline temperature slightly increases the fuel cycle cost.

A sweep of the flux peaking factor (F_T) from 0.8 to 1.6 showed a linear increase in the number of pins per subassembly (Figures II-11 and II-12). For a fissile fraction of 0.22 and F_T of 0.8, there were approximately 117 pins per subassembly while at an F_T of 1.6, there were 227 pins per subassembly. These studies again point out that the fuel cycle costs for a given core is in direct proportion to the fissile fraction.

Sweeps were made on the number of open and closed test loops (N_{TEST}) at different setpoints of peak neutron flux (Figures II-13 through II-15). The X-Y plots of these sweeps were N_{TEST} versus f_{PuO_2} for different setpoints of flux. The fissile fraction appears to be a straight line function of N_{TEST} . Power and flux were held constant during these sweeps, while the fissile fraction was allowed to calculate constantly as N_{TEST} was ramped.

The final study plotted the effect of pin diameter versus flux for the three setpoints of f_{PuO_2} while ramping ΔP (Figures II-16 and II-17). While ΔP was started at an unrealistic value, the plots do come into and show useful data. The strip chart recording has a pin diameter recording as the lower scale to enable the reader to correlate the X-Y plot with the strip chart recording.

Thermal Hydraulics Analyses -

A. Padilla

Analyses of Heat Transfer from Proposed Rabbit Capsule

An analysis was performed to determine the feasibility of using helium coolant to remove the heat generated in the proposed rabbit capsule.

The capsule consisted of a single 0.250 in. OD fuel pin contained inside a 1.0 in. OD by 0.75 in. ID capsule. The space between the capsule and the fuel pin is filled with sodium. The capsule is suspended inside a 1.25 in. ID vessel. The heat generated within the fuel pin is assumed to flow by conduction across the sodium inside the capsule and is then removed by the

FIGURE II-2. The Effects of the Uncertainty of the Thermal Conductivity of the FFTF

FIGURE II-8. The Effects of Thermal Conductivity

FIGURE II-9. Flux Versus Temperature Centerline

FIGURE II-10. Centerline Temperature Study

FIGURE II-11. Flux Versus F_T

FIGURE II-12. Flux Peaking Factor Ramps

FIGURE II-13. F_{PuO_2} Versus N_{TEST}

FIGURE II-14. Number of Fueled Test Loop Assemblies Surveyed

FIGURE II-15. Number of Fueled Test Loop Assemblies Survey

FIGURE II-16. Outside Pin Diameter/
Inches Flux Versus Pin Diameter

FIGURE II-17. Pressure Drop Sweeps

helium coolant flowing in the 1/16 in. annular gap between the capsule and the vessel.

The thickness of the cladding on the fuel pin was assumed to be 0.015 in. The material of the clad and capsule wall was taken as stainless steel with a gamma heating rate of 12 W/g. Assuming a packed powder fuel, the gap coefficient was assumed to be 2000 Btu/hr-ft²-°F. The equation used to calculate the heat transfer coefficient between the helium coolant and the capsule is that recommended by McAdams⁽¹⁾ for convection heat transfer in an annulus. The results are summarized below:

Heat Generation

<u>kW/cm</u>	<u>kW/ft</u>	<u>Packed Powder Fuel</u>
<u>Based on incipient fuel centerline melting (2750 °C)</u>		
0.6	18.3	Helium coolant must be below 100 °F
0.5	15.2	Maximum fuel clad temperature 1596 °F. Helium flow rate 147 lb/hr. Velocity of helium for 1 atm and 100 °F 1360 ft/sec.

Transient Thermal Analysis of Flow Blockage

An analysis was performed to determine the transient fuel and coolant temperatures during a simulated flow blockage at the bottom of a subassembly. The values of the parameters assumed for the calculations were:

Fuel Pin Dimension	0.200 in. OD by 36 in. long
Fuel Type	UO ₂ -PuO ₂ pellet
Internal Heat Generation	Variable along length of fuel pin; constant with time
UO ₂ -PuO ₂ Thermal Conductivity	Variable with temperature
Fuel to Clad Contact Coefficient	1000 Btu/hr-ft ² -°F
Clad to Coolant Heat Transfer Coefficient	32,000 Btu/hr-ft ² -°F
Inlet Coolant Temperature	700 °F
Outlet Coolant Temperature With No Blockage	1000 °F
Flow Rate	Varies from full flow to zero in one second

The fuel was divided into three radial and three axial sections. The parameters listed above were used as input for the TIGER-V program which was run on the digital computer.

The results are shown in Figure II-18. The flow rate was decreased somewhat faster than linearly, although any type of flow change can be programmed. The ratio of coolant temperature across the fuel pin to that with no flow blockage rises gradually. Even at the end of the curve of 0.75 sec, the coolant is still more than 250 °F below the boiling point of sodium. At a point 3 ft downstream of the fuel pin, the temperature rises very slowly. If a thermocouple were located at this point, it would detect a 20% increase in the coolant temperature rise in about 0.57 sec after the start of the flow blockage. However, this should be sufficient time

1. *Heat Transmission, Third Edition, McGraw-Hill Book Co., 1954.*

FIGURE II-18. Simulated Flow Blockage

to take action in order to avoid boiling in the subassembly.

The above calculations would not apply if the flow blockage occurred across a very small number of pins. The increase in coolant temperature rise for the blocked channel would be diluted by the other unblocked channels and a thermocouple located downstream from the fuel pins probably would not be able to detect significant change.

Analytical Study of Time Response for Ungrounded Thermocouples - A. Padilla

An analytical study to calculate the time response of ungrounded thermocouples was performed to obtain a measure of the importance of the various parameters. The three parameters studied thus far are size, type of electrical insulation, and the thermal resistance between different materials within the thermocouple.

The three sizes studied were 0.040 in, 1/16 in., and 1/8 in. OD. Table II-7 shows the specifications for these metal-sheathed, ceramic-insulated thermocouples taken from ANL-HT1-FFTF-507.

For the analysis, the minimum sheath wall thickness and insulation thickness was used, thus maximizing the size of the hot junction. The size of the ungrounded hot junction is unknown and would depend largely on the method of fabrication.

Three materials were considered for electrical insulation between the hot junction and the sheath: BeO, MgO, and Al₂O₃. Since the transient considered was a step change from 800 to 900 °F of the temperature of sodium flowing axially to the thermocouple, the properties for the three insulation materials were taken at 850 °F and assumed to be constant. Table II-8 shows these properties. Although BeO has the highest thermal

conductivity, the product of its density and heat capacity is also higher than that of MgO and BeO.

Three modes of heat transfer were considered between the hot junction and the insulation and between the insulation and the sheath: metallurgical bonding, conduction across 0.002 in. gaps filled with helium, and radiation across 0.002 in. gaps in vacuum.

Figure II-19 shows the thermocouple configuration and the model used for the analysis. The sheath material was taken to be 304 SST. The thermoelements for the thermocouple were Chromel-Alumel. The forced convection coefficient between the sheath and the sodium flowing at 30 ft/sec was assumed to be constant at 50,000 Btu/hr-ft²-°F.

Figure II-20 shows the effect of thermocouple size on the time response to a step change in coolant temperature from 800 to 900 °F. The time constants (time required to reach 63.2% of the total change) are:

TABLE II-7. Specification for Metal-Sheathed, Ceramic-Insulated Thermocouples

<u>Nominal OD, in.</u>	<u>Minimum Sheath Wall, in.</u>	<u>Minimum Wire Diameter, in.</u>	<u>Minimum Insulation Thickness, in.</u>
0.040	0.006	0.005	0.004
0.062	0.009	0.010	0.005
0.125	0.012	0.020	0.012

TABLE II-8. Properties of BeO, MgO, and Al₂O₃ at 850 °F

<u>Material</u>	<u>Density, Lb/in³</u>	<u>Heat Capacity, Btu/Lb-°F</u>	<u>Thermal Conductivity, Btu/Hr-Ft-°F</u>
BeO	0.104	0.432	42.0
MgO	0.126	0.288	8.8
Al ₂ O ₃	0.137	0.275	6.2

FIGURE II-19. Model for Thermocouple Time Response Study

FIGURE II-20. Temperature Response of SST Sheathed Chromel-Alumel Thermocouples to 800 to 900 °F Transient

Nominal OD*	Time Constant
0.040	0.090
0.062	0.182
0.125	0.510

*BeO insulation; contact coefficients
2000 Btu/hr-ft²-°F.

Figure II-21 shows the effect of assuming various values for the sheath-insulation and insulation-hot-junction thermal resistance (these two resistances were assumed to be equal) for a 0.040 in., Al₂O₃-insulated thermocouple. The time constants are:

Contact Coefficient, Btu/hr-ft ² -°F	Time Constant, Seconds
16,000	0.036
8,000	0.051
4,000	0.080
Conduction across 0.002-in. helium gaps.	0.177
Radiation across 0.002-in. gaps.	>>1

Radiation is unrealistic and unacceptable as a mode of heat transfer. It is not possible to determine the value of the internal thermal resistances with much confidence because of the lack of data for ungrounded thermocouples. The data of Green and Hunt⁽¹⁾ for 0.044 in. OD, SST 304-sheathed, Al₂O₃ insulated, swaged thermocouples with ungrounded junctions are plotted in Figure II-20 and are close to the curve for a contact coefficient of 8000 Btu/hr-ft²-°F. Until further experimental data is obtained for comparison, a realistic range for values

of the thermal contact resistances within the thermocouple is between 8000 Btu/hr-ft²-°F and conduction across 0.002 in. helium gaps.

Figure II-22 shows the effect of different electrical insulation materials for 1/8 in., OD thermocouples.

The time constants are:

Insulation	Contact Coefficient, 8000 Btu/hr, ft ² -°F	Conduction Across 0.002 in. Helium Gaps
BeO	0.215 sec	0.950 sec
MgO	0.248	0.935
Al ₂ O ₃	0.271	0.961

As the thermal contact resistances within the thermocouple are decreased, the high thermal conductivity of BeO results in better time constants. The relative order of the time constants follows the order in the thermal conductivities. As the thermal contact resistances are increased, the thermal capacity becomes more important. For conduction across a 0.002 in. gap (which corresponds to a contact resistance of about eight times that of a contact coefficient of 8000 Btu/hr-ft²-°F), the time constant for the MgO-insulated thermocouple is better than that of the BeO-insulated thermocouple due to the much higher heat capacity of BeO. As the internal contact resistances become controlling, the variation between the different insulating materials becomes negligible.

The results indicate the importance of insuring good thermal contact within the thermocouples in order to obtain reasonable time constants. The effect of size is also very important and it may be necessary to go to

1. T. W. Hunt and S. J. Green. "Accuracy and Response of Thermocouples for Surface and Fluid Temperature Measurements," *Temperature, Its Measurement and Control in Science and Industry*. Reinhold Publishing Corporation, New York, 1962. Vol. 3, Part 2., pp. 695-722.

FIGURE II-21. Temperature Response of 0.040 in. OD, SST 304-Sheathed, Al₂O₃ Insulated, CR/AL TC's to 800 → 900 °F Transient

FIGURE II-22. Temperature Response of 1/8 in. OD, SST-Sheathed, CR/AL TC's to 800 → 900 °F Transient

0.040 in., OD thermocouples for measuring coolant temperatures if abnormal operating conditions are to be detected in sufficient time to take action.

FFTF Digital Computer Simulation -
C. D. Flowers and H. G. Johnson

Dynamic simulation of the FFTF with the use of the Dynamic Systems Analyzer, DYNASAR,⁽¹⁾ Program is under way. DYNASAR is a digital computer program developed for the study of dynamic systems. The program can handle systems with up to 500 variables, 150 transfer functions, and 350 other algebraic functions.

A reactor model for digital simulation of the FTR was developed and is shown schematically in Figure II-23. The model consists of an inlet plenum that receives the primary loop cold leg flows from two complete primary loops. A lower support structure is assumed to be subjected to the inlet plenum environment for calculating structural motions due to temperature changes. Three flow paths through the reactor are modeled:

1. Reactor tube flow, w_{cc}
 2. Leakage flow around the reactor tubes, w_{lc}
 3. Radial reflector flow, w_{rc} .
- A three node core model will be assumed with separate fuel and coolant axial nodes.

Lower and upper axial reflector regions are also included in the model. The leakage region is divided into

three axial nodes and the radial reflector region is assumed to be a single node. Linear coolant temperature gradients are assumed through the radial reflector region. Radial heat transfer between each of the adjacent parallel regions will be included in the models.

Equations for the inlet plenum and lower support plate have been programmed and debugged subject to the following assumptions:

1. Complete mixing in inlet plenum
2. Constant overall heat transfer coefficient for lower support plate
3. Constant conductivity of lower support plate

FIGURE II-23. Digital Simulation Reactor Tank Model

1. J. M. Watson, *Engineers' Manual DYNASAR III*, General Report No. 63GL36, January 1, 1963.

4. Infinite convection coefficient, lower support plate to coolant
5. Constant specific heat of lower support plate.

Primary coolant flows and temperatures were input as forcing functions for the debug runs. The temperature response for this portion of the model was found to be satisfactory.

Two heat balance models were formulated for the lower axial reflector region and are presently being compared. In one model the lower axial reflector material is lumped together with the sodium volume within the reactor tubes below the core. The second model simulates separate nodes for the lower reflector material and the coolant regions.

The section of the program for primary loop hydraulics and flow control was completed and is nearly ready for debug. The model includes two separate primary loops which can be used to represent various combinations of two or more actual installed loops. Each loop in turn is divided into a hot leg run between the reactor vessel and intermediate heat exchanger (IHX), and a cold leg run between the IHX and reactor vessel. The pump is in the cold leg in this model. The two primary loops are joined together at the reactor vessel for common flowpaths through the vessel. Detailed simulation is provided for the pump's main motor and pony motor drives including eddy current clutches between the main motors and pump. Variation in the friction factor with Reynolds number and changes of coolant physical prop-

erties with temperatures are accounted for in all sections of the loops and vessel.

FFTF Analog Simulation - R. D. Benham, L. H. Gerhardstein and W. H. Nielson

Debug runs were made by using the overall FFTF Dynamic Analog Simulation with a three + one loop configuration. Several problems were detected and corrected as a result of these runs.

A comparison study of a complex (three coolant nodes, four fuel nodes) and simplified (one coolant node, one fuel node) reactor thermo-dynamic model was made. The results showed that steady-state outputs for coolant outlet and average fuel temperatures are the same for the two models. Temperatures were different by a maximum of ~5% for the transient case. The curves in Figure II-24 show the maximum difference in responses of the two models for a decaying change in inlet temperature.

$$T_{in} = T_{in}(0) + (T_{in}(\text{final}) - T_{in}(0))e^{-t/\tau}$$

where τ^{-1} is the decay time constant. The percentage deviation was computed from

$$\text{dev.} = \frac{\epsilon}{\Delta T}$$

where ϵ is the maximum difference in temperature responses ($^{\circ}\text{F}$).

and

ΔT is the total change of that temperature from one steady-state condition to the next.

FIGURE II-24. Maximum Deviation in the Responses of the 1- and 3-Node Models for a Step Coolant Inlet Temperature

The curves in Figure II-25a show that the maximum deviation tends to increase with decreasing flow rate and when higher frequency components are present in the input wave form. In all cases, the deviation exists for only a short time (a few seconds or less) depending on the speed of the input signal. The simplified one-node model will be used in the FFTF dynamics simulation.

A possible method for simplifying the Dump Heat Exchanger (DHX) Model is

being considered. The simplified model would use two differential equations and an approximation to the log-mean temperature difference calculation.⁽¹⁾ Initial studies showed that accurate steady-state responses can be generated with minimal equipment. Further study of the accuracy of the transient response of the model will be made.

1. L. H. Gerhardstein, Analog Computer Simulation of Log-Mean Temperature Difference Equation, April 18, 1966.

Runs were made to study the transient responses of the FFTF dynamic simulation to various perturbations in power level. When the reactor is scrammed from ultimate phase conditions, the times for secondary coolant solidification were:

<u>Flow Variation</u>	<u>Time for Solidification, min.</u>
No flow variation	9
Primary coolant flow rate decreased from 100% to 20% design conditions at the time of scram	12
Tertiary coolant flow rate decreased from 100% to 20% design conditions at the time of scram	35

The data given above illustrates that variations in tertiary flow are much more effective than primary flow variations in regulating the coolant temperatures over long periods of time.

It was found that structural reactivity feed back due to pin bowing causes oscillations in power level after a negative reactivity step of 1.3 to 2.5 dollars, as shown in Figure II-25b. These runs were made with a maximum pin bowing reactivity feedback coefficient characteristic of cermet fuel. A more realistic figure for oxide fuel will be used in future runs to determine the significance of the oscillation. Proof that the oscillation is due to pin bowing feedback was made by removing the term from the reactivity summer. With the feedback removed, the oscillation is not present in the transient response of the simulation.

Runs will be continued after several reactor simulation changes are made. These changes include: Lumping of the two inlet plenums into one, adding heat transfer and heat generation in the reflector, including conduction heat transfer from the reflector outlet zone to vessel outlet region, and changing the pin bowing reactivity feedback coefficient. Studies of failure in one or more coolant loops will be accomplished by simulating the "failing loops" on the PDP-7 digital computer. Studies of direct digital control will also be made by using the PDP-7 as the controller and the analog computer as the process.

A modification of the simulation was made to include the following:

1. Heat transfer between reflector outlet and vessel outlet mixing regions
2. Core fission heat decay
3. Reflector fission heat decay
4. Pin bowing reactivity feedback coefficient changed from 2.165 to 1.42\$ maximum
5. Lumping of inlet plenums into one.

After changing the maximum pin bowing reactivity feedback to 1.42\$ (at 225 °F), runs were made to investigate the nature of the slow oscillation described above. It was found that, at 1.42\$ maximum reactivity, the sustained oscillation does not occur, and that an over-damped oscillation following a step negative reactivity is present. Runs were made while increasing the slope of the pin bowing reactivity which left the break point constant at 225 °F (Figure II-26). Sustained oscillation after a step negative reactivity begins when δK maximum is

FIGURE II-25a. System Response to a Reactivity Step of -1.3% . Primary flow = 1100 lb/sec/loop; secondary flow = 1100 lb/sec/loop; Tertiary flow = 1 lb/sec/module; $N_p = 4$ loops; NMTY = 4 modules per loop; Tertiary inlet temperature = $100^{\circ}F$.

FIGURE II-25b. System Response to a Reactivity Step of -1.3% . Primary flow = 1100 lb/sec/loop; secondary flow = 1100 lb/sec/loop; Tertiary flow = 1 lb/sec/module; $N_p = 4$ loops; NMTY = 4 modules per loop; Tertiary inlet temperature = $100^{\circ}F$.

FIGURE II-26. Maximum Deviation in the Responses of the 1- and 3-Node Models for a Step Coolant Inlet Temperature

increased to 1.48\$. Beyond this value, there is a range of step δK for which sustained oscillation will occur. Figure II-27 shows the profile of the region over which sustained oscillations were found.

Analog Simulation of Natural Coolant Circulation - R. D. Benham,

L. H. Gerhardtstein and W. H. Nielson

Analog runs were made to investigate natural circulation. Loss of system power and scram were simulated in the reactor transient response while varying the heights of IHX and DHX above the reactor core. All combinations of 5, 10, and 20 ft for the IHX above the reactor and the DHX above the IHX were used.

The initial part (150 sec after scram) of the reactor outlet temperature wave form is not affected by the DHX height above the IHX for the above heights studied. This is expected since the flow rate in the secondary loop has been greatly reduced causing a transport delay greater than 150 sec. The primary effect is by varying the IHX height above the reactor (Figure II-28). Figure II-28 indicates that the peak in reactor outlet temperature is reduced by increasing the height difference between the reactor core and IHX (H_{XR}). The initial decrease in the reactor outlet temperature is due to the immediate decrease in power level (reactor was scrammed at time of loss of system power). The following increase in temperature is a result of a combination of coolant flow rate decreasing and the fission heat that is present.

Figure II-29 illustrates the DHX secondary outlet temperature wave forms for various heights. This figure shows the DHX secondary outlet temperature from the point of loss of system power to freezing of the secondary sodium. This temperature is not affected by the IHX height above the reactor core except when the IHX is 20 ft above the reactor. The transport delay of the primary loop is less than 200 sec at a height difference of 20 ft for the IHX above the reactor.

The height of the DHX above the IHX, however, does significantly affect the DHX secondary outlet temperature. As this height is increased, the secondary coolant flow rate is increased which causes a greater heat transfer rate in the IHX which tends to raise the temperature of the cold leg in the secondary loop.

Figures II-30 through II-33 illustrate the effect of closing the DHX air flow valve after a loss of system power. The sodium is kept from freezing if the valve is closed shutting off the tertiary heat dump.

The digital and analog computers were used to demonstrate direct digital control of the FFTF dynamics simulation. Two control loops were simulated:

1. Power level control with control reactivity controller
2. Reactor inlet temperature control by varying the DHX air intake valve position.

Figure II-34 shows the response of power level to step reactivity insertions of 0.5, 0.1, 1.5, and 2.0 mk with the power level controller operating. The controller gains used were

FIGURE II-27. Profile of Sustained Oscillation Region Following a Negative Reactivity Step

FIGURE II-28. Effect of IHX Height Above the Reactor on the Reactor Outlet Temperature

FIGURE II-29. Effects of DHX and IHX Height Differences on the DHX Secondary Outlet Temperature

FIGURE II-30. Reactor Response to Loss of System Power with DHX Butterfly Valve Open

Primary Flow (initially)
 = 1130 lb/sec/loop,
 Tertiary Flow (initially)
 = 180 lb/sec Module,
 $N_P = 4$ Loops, $N_{MTY} = 4$ Modules per Loop,
 Tertiary Inlet Temperature = 100 °F

FIGURE II-31. Reactor Response to Loss of System Power with DHX Butterfly Valve Open

FIGURE II-32. Reactor Response to Loss of System Power with DHX Butterfly Valve Closed

Primary Flow (initially)
 = 1130 lb/sec/loop,
 Secondary Flow (initially)
 = 1130 lb/sec/loop,
 Tertiary Flow (initially)
 = 180 lb/sec Module,
 $N_P = 4$ Loops, $N_{MTY} = 4$ Modules Per Loop,
 Tertiary Inlet Temperature = 100 °F

FIGURE II-33. Reactor Response to Loss of System Power with DHX Butterfly Valve Closed

$K_p = 0.5 \times (1 \text{ mk/W})$ and $K_I = 1.0 \text{ sec}^{-1}$. The digital computer was operating with an iteration time of 0.1 sec. The response in power level after the disturbance is fast, due to the high controller gains.

Figure II-35 shows the response of reactor inlet temperature to a reduction in power level from 400 to 300 MW with the temperature controller active. The controller settings used were relatively low ($K_p = 1. \text{ deg. valve displacement per } ^\circ\text{F}$, $K_I = 0.5$). Higher settings of the controller tend to cause instability in the system. In the future, studies will be made to determine the necessity for using compensation if the control loop is used.

Computer Code Development -

G. H. Dickerson

The four codes received from General Atomics, SAFE-SHELL, SAFE-PLANE, SAFE-PCRS, and AXI-SYM, have been converted to the UNIVAC 1107 and debugged and are available for general use. Two of the programs, SAFE-PCRS and AXI-SYM, appear to be the most general of the four. All programs are oriented towards analyses of heterogeneous structures such as reinforced or prestressed concrete vessels.

SAFE-PCRS is intended as a rapid and flexible design tool by which material constants, dimensions, and loading conditions may easily be varied. The program will numerically determine the stress and strain distribution within either homogeneous or heterogeneous, thick-walled bodies of revolution.

AXI-SYM differs from SAFE-PCRS in two aspects: the method of solution is iterative in AXI-SYM as opposed to the direct method used in SAFE-PCRS, and the capacity of AXI-SYM is approximately 40% that of SAFE-PCRS. This method permits alteration in design, such as changes in geometry and internal reinforcement, by simple addition to or subtraction from the input deck. This feature makes AXI-SYM a design-oriented program, whereas SAFE-PCRS is used as an analytic program.

Mechanical Loads and Material Requirements on FTR Tubes - M. T. Jakub

A study on mechanical loads in the various FTR tube concepts was completed. Conclusions were:

- Typical required S_m values (i.e., material design stress values as per ASME Nuclear Code) for the in-core tube material are shown in Table II-9.
- Fatigue loadings are shown on Table II-10; however, the paucity of fatigue knowledge about irradiated 304 SST makes the interpretation much more difficult.
- Stresses in Tube designs shown in Tables II-9 and II-10 do not present insurmountable problems in the proposed FTR concepts.
- From Table II-9, the most severe loading is the primary membrane plus primary bending stresses.
- For noncircular tubes, the corner bending stress due to internal pressure is the most significant factor in the primary bending stresses.

FIGURE II-34. Response of Power Level with Digital Controller to Step Reactivity ± 0.5 , ± 1.0 , ± 1.5 , and ± 2.0 MK.

FIGURE II-35. System Response to a Power Level Change of 400 to 300 MW with Digital Inlet Temperature Control

TABLE II-9 Typical S_m Values for FTR Tube Materials

S_m Criteria*	Simply Supported Tube**		Fixed Top Tube		Free Bowing Tube	
	5 in. Hex, 0.15 in. Wall	5 in. Round, 0.1 in. Wall	5 in. Hex, 0.15 in. =t	5 in. Round, 0.1 in. =t	3 in. Hex, 0.1 in. =t	3 in. Round 0.08 in. =t
Primary Membrane (Design $\Delta P = 100$ psi) $\sigma_{PM} = S_m$	3,300	4,950	3,300	4,950	3,000	3,210
Local Membrane $\sigma_{LM} = 1.5 S_m$	Estimated to be less than values in above line.					
Primary Bending Plus Primary Membrane ($\Delta P = 100$ psi) $\sigma = 1.5 S_m$	17,120	6,230	17,250	6,330	12,000	2,140
Primary Plus Secondary $\sigma = 3.0 S_m$	Estimated to be less than values in above line.					

* S_m is the material design stress given in ASME Nuclear Vessel Code--suitably modified for FTR conditions of temperature, irradiation and criticality of service.

**All hex tubes have corner radius to thickness ratio of 2.0.

TABLE II-10 Typical Fatigue Loadings for FTR Tube Materials

	Simple-Fixed Beams or Simple-Simple Beams		Free Bowing Tubes	
	5 in. Hex, t = 0.15 in.	5 in. Round, t = 0.1 in.	3 in. Hex, t = 0.1 in.	3 in. Round, t = 0.08 in.
1. Steady-State Thermal σ Negligible in drivers; loop values have maximum as shown at right, psi.	(14,400)	(14,400)	(14,400)	(14,400)
2. γ -Heating σ At Full Power	3,550	1,580	1,580	1,010
3. Scram Thermal Shock σ	39,500	36,700	18,350	26,600
4. Intermediate Shut-down Thermal Shock σ	1,100	5,470	~	3,800
5. Primary + Secondary σ ⁽¹⁾	33,500	12,250	23,400	4,170
6. Alternating Stress for Intermediate and normal FTR Cycles ^(2,5)	49,600	25,100	32,500	11,670
7. Cycles to Failure in 6. ⁽³⁾	140	1,700	700	∞
8. Alternating Stress for Scram Cycle ⁽⁴⁾	99,600	65,700	52,430	41,300
9. Cycles to Failure in 8. ⁽³⁾	10	48	115	280
10. $N_i = N_1 + N_2 \leq 0.8$ ⁽⁵⁾ where $N_i = \frac{\text{actual cycles}}{\text{cycles to failure}}$	$\frac{40}{140} + \frac{6}{10} = 0.886$	$\frac{40}{1700} + \frac{6}{48} = 0.15$	0.06	Negligible

Notes:

- (1) Assume 30% of stress from "primary + bending" is value of secondary stress.
- (2) For stress concentrations and other stresses in Section VII-D, assume their value is 30% of the sum of other stress loads. Assume that transient stresses during the normal startups are the same as in intermediate shutdown cycle (Row 4). Normal cycle thus has total $S_a = (\sigma_{\text{row 4}} + \sigma_{\text{row 5}} + \sigma_{\text{row 2}})$ 1.3.
- (3) Use S_a versus N curve in ASME Code Case 1331-1 at 1200 °F.
- (4) For stress concentrations and other stresses in Section VII-D, assume their value is 30% of the sum of other stress loads. Thus, the $S_a = 1.3 [\sigma_{\text{row 3}} + \sigma_{\text{row 5}} + \sigma_{\text{row 2}}]$ for scram cycles.
- (5) Consider the normal cycles as additional "intermediate" cycles.

- The outlet perforated tube regions of some FTR tube designs present natural high stress areas. The tube wall may need increased thickness to overcome the higher stresses around the holes.
- From Table II-10, the only tube design which may approach fatigue limits in the range of the 1200 °F curve of Code Case 1331-1 is with 5 in. hex tubes, 0.15 in. walls.
- R&D is needed in the following areas:
 - Fatigue design curves (S_a versus N) for irradiated 304 SST
 - Flow cycling rate fatigue data for 304 SST
 - Stress concentrations and stress loadings in perforated hexagonal and circular tubes.

For problems that may arise, the following design and operating alternatives can reduce the loadings on the tubes:

- Use another material (other than 304 SST). For example, Incoloy 800
- Change tubes more frequently to reduce fatigue stress cycles
- Use round tubes instead of hex tubes
- Use a free-bowing support method for the tube
- Reduce outlet temperature core delta-T, or core pressure drop
- Preload the tubes inward to obtain lower stresses at full power operation.

REACTOR SAFEGUARDS - D. E. SIMPSON

Scram Requirements for Single Channel Meltdown - D. E. Simpson

Analyses were performed to determine safety rod scram acceleration required to terminate a reactivity transient resulting from a central channel meltdown, without exceeding an overpower limit corresponding to the beginning of melting in adjacent fuel.

The meltdown model used in the analysis was the gravity drop of the top one-third of the fuel in a central subassembly to the position of the center one-third. The reactivity insertion in this model is about $\frac{62}{N}\%$, where N is the number of fuel assemblies in the core, assuming all assemblies to be identical. The average reactivity insertion rate is about $\frac{250}{N}\%/sec$.

The time for fuel temperature to rise an amount corresponding to an overpower factor of 1.2 was calculated as a function of reactivity ramp rate. From this, the safety rod scram acceleration was calculated for time delays

of 50 and 100 msec between existence of a high flux scram condition and initiation of rod motion. The resulting accelerations are plotted in Figure II-36 as a function of the number of driver fuel assemblies in the core. The upper broken curve in this figure includes a margin of safety over the calculated values; this curve was used in evaluating the ability of alternate conceptual designs to sustain a single channel meltdown.

On the basis of this study, it has been concluded that the concept selected for FFTF should be capable of core subdivision to at least 80 fuel assemblies without unacceptable penalties. This does not imply that 80 is a minimum or an optimum number, but only that 80 is an ample number. The final optimum may well be between 50 and 80. Less than 50 would appear to put impractical demands on the reactor safety system.

FIGURE II-36. Required Safety Rod Acceleration Versus Number of Driver Fuel Assemblies

INSTRUMENTATION AND CONTROL - C. D. SWANSON

Instrumentation Tasks - C. E. Huck,
G. J. Dau, T. Crocker, J. A. Christensen,
J. L. Jackson

Laboratory analysis and radiographic testing of the sheathed thermocouples for insertion in the first EBR-II capsule have been completed. Electrical response time and emf tests remain to be done. The first calibration furnace was energized and shown to satisfactorily control tube temperature from 300 to 1500 °F.

An experiment to detect the onset of central melting in UO_2 and $(U,Pu)O_2$ fuels was designed in support of FFTF requirements for in-pile instrumentation development. The tests consist basically of measuring the electrical resistance along a short segment of the fuel axis and correlating measurements with anticipated changes in resistivity on melting. Analyses of the equivalent electrical circuit and the expected change in resistivity on melting showed the concept to be potentially very sensitive for monitoring center melting. Changes in axial resistance will represent better than 90% of the total impedance change measured, and the limited data available for similar oxide systems suggest that a factor of ten change in resistivity on melting might be expected. Current modification of the experiment will include a centerline W/W 26 Re thermocouple. This was accomplished by designing one of the axial voltage probes as a bi-metal assembly which could serve the dual purpose of probe and thermocouple.

Reactor Environmental Effects on Signal Cables - G. J. Dau

Seebeck Coefficient

A study of the theoretical aspects of the Seebeck Coefficient was initiated to determine whether intense nuclear radiation can be expected to produce instantaneous side effects. This is not a study of long-term transmutation effects. Rather, it is aimed at an understanding of the so-called "prompt" effects of intense gamma and neutron flux on cables and insulators. Equipment is being fabricated to permit exploratory tests at the temperature of boiling water in high gamma fields.

Electromotive Force in Thermocouples

Mechanisms which govern the emf generated in thermocouples have been studied. This preliminary analysis indicates that a high radiation field (neutrons or gamma photons) can alter the calibration of a thermocouple. The theory as it is now formulated is not precise enough to predict the flux level at which a radiation induced calibration change becomes significant.

Experimental results on chromel/alumel and copper/constantan thermocouples indicate that a calibration shift does occur in a gamma field of 2.3×10^7 R/hr. These results also suggest that the shift depends on temperature, thermocouple size, and dose rate. All data should be considered as qualitative only at this time. Improved instrumentation is expected

to be in use very soon which will give the needed quantitative data.

The preliminary experiments were performed in the ^{60}Co gamma pit at several temperatures. The experimental technique used placed both junctions of a thermocouple in separate but identical constant temperature baths with a suitable meter in the circuit to measure the generated emf. Baths used include dry ice and alcohol ($-78\text{ }^{\circ}\text{C}$), liquid nitrogen ($-196\text{ }^{\circ}\text{C}$), ice points of water ($0\text{ }^{\circ}\text{C}$), and boiling water ($100\text{ }^{\circ}\text{C}$).

Higher temperature baths will be used in future work. If no outside forces are present, the net emf is zero. One of the constant temperature baths is then placed in the gamma photon field and the resulting emf measured. As both junctions are at the same temperature, the resulting emf is due to the radiation induced calibration shift.

Schedule for the Central Control and Data Handling System - J. H. Kinginger

An overall schedule was developed for the Central Control and Data Handling (CC&DH) system. The schedule is based on current FFTF milestones showing Plant Systems tests slated to begin in July 1972 and Reactor Critical in January 1973. The specification development for major digital components associated with CC&DH should be complete by April 1968 to allow sufficient time for bid and review, assembly, delivery and installation, and operational checks. This should present no problem because central control and data handling equipment specifications are not strongly dependent on detailed system design. A similar schedule for programming indicates that general

program design should begin about July 1967.

Initial control studies using the FFTF dynamic simulation model (analog computer) and the digital instrumentation prototype (PDP-7 computer) are being planned with the Simulation and Systems Analysis Unit. Hybrid computer runs are expected to be made in March.

Control System Design - G. L. Waldkoetter

The evaluation of candidate CC&DH concepts is in progress. Descriptions of two concepts have been completed, and the remaining two descriptions are in preparation. These descriptions are required to define the concepts which will be evaluated with the figure-of-merit model.

Each concept includes a digital data handling system. Based on an analysis of FFTF requirements involving data displays and calculations which should be available for operating decisions, a redundant digital data handling and display configuration appears to be the better choice. This configuration will reliably perform on-line calculations and display of operating parameters such as thermal and tube power calculation, total flow, flux distribution, etc. Redundant computers and input multiplexers provide a convenient method of checking the results of the on-line equipment as well as providing backup in case of failure.

Control room arrangement drawings were prepared and distributed for comment during the past month. Figures II-37 and II-38 show the approximate space requirements for both analog and digital control concepts. The final arrangement of the space

PLAN-EL(+32'-0")

0 5 10 10 FT
SCALE
(TYPICAL)

PLAN-EL(+16'-0")

EQUIPMENT LIST

PART NO.	DESCRIPTION
1	MAIN HEAT REMOVAL
2	NUCLEAR INSTRUMENTATION
3	ROD PISTON INDICATION
4	ANNUNCIATORS ABOVE
5	SAFETY CIRCUIT BY PASS
6	CORE INSTRUMENTATION
7	TEMPERATURE MONITOR
8	SHIELD COOLING
9	RG RECEIVING AND PROCESSING
10	FUEL HANDLING MACHINE
11	RABBIT FACILITY
12	INERT GAS
13	BUILDING RADIATION MONITORING
14	HEAT, VENT & CONFINEMENT
15	ELECTRONIC CONTROL PANEL
16	CLOSED LOOPS
17	REACTOR OPERATOR'S CONSOLE
18	COMMUNICATIONS CONSOLE
19	COMPUTER
20	DRUM
21	MAGNETIC TAPE
22	PRINTER
23	TABLE
24	DATA CONTROL CONSOLE
25	PROGRAMMING CONSOLE
26	INPUT MULTIPLEXERS

Neg 0670655-6

FIGURE II-37. Control and Electrical Equipment Rooms for Analog-1

PART	DESCRIPTION
1	SAFETY CIRCUIT
2	INPUT MULT. PLUGS
3	CONTROL OUTPUT INTERFACE
4	PROCESS COMPUTER
5	EXPERIMENT COMPUTER
6	SUPERVISOR COMPUTER
7	DISC
8	DRUM
9	OPERATOR CONSOLE
10	NUCLEAR INSTRUMENTATION
11	CONTROL EXPERIMENTAL CONTROL
12	SUPERVISOR DISPLAY
13	COMMUNICATION CONSOLE
14	PRINTER LAYOUT TABLE
15	PROGRAMMING CONSOLE
16	MAGNETIC TAPE
17	HIGH SPEED PRINTER

Neg 0670655-20

FIGURE II-38. Control and Electrical Equipment Rooms for DDC-1

will, of course, depend on the building design.

These drawings include provision for experimenters' equipment rooms which may be used to house experimenters' data equipment, at the experimenters' option, in addition to the capability provided by the FFTF facility.

A control room based on a Direct Digital Control (DDC) concept requires less than one-half the area needed by a control room based on an analog control concept.

Control System Development -

J. L. Deichman

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

A set of programs for the PDP-7 is being developed for general and FFTF simulation use. These programs will be used for direct digital control of the analog process and for additional simulation. The programs that have been written and tested successfully are:

1. Binary-to-decimal and decimal-to-binary conversion subroutines
2. Hybrid multiply and divide
3. Input subroutine (samples a given number of data lines from the analog computer)
4. Output subroutine (converts a number of words D to A).

The analog-to-digital and digital-to-analog converters are now ready for use in the hybrid facility. Some mod-

ification to the A/D converter overload circuit was necessary because it had previously failed to meet the requirements. Mode control set and sense circuits have been completed which enable the digital computer to control the analog computer in five modes (standby, hold, initial condition, compute, and pot set) and also provide a pot servo command.

Final analog addressing circuit fabrication and debugging have progressed during the month. When completed, these circuits will enable the digital computer to randomly address any variable on either of the analog computer consoles.

Work is in progress on fabrication of magnetic drum read-write circuits, drum field selection circuits, and optical timing circuitry. Initial design of the major units in the interface logic circuitry has been completed and necessary hardware has been ordered. Several cable systems have been evaluated. The most satisfactory read-write transfers were realized with the use of twisted pair ribbon cable.

Final mode control circuit fabrication and debugging is in progress. A real time (60 hertz) clock and a 5 kHz crystal controlled clock have been installed. The use of one of these clocks will enable the programmer to initiate program interrupts at present time intervals and also time various computer operations. The clock hardware addition increases the symbolic software package by three instructions. The new instructions are : CLSF, skip if clock flag equals one; CLOF, clear flag and disable clock; and CLON, clear flag and enable the clock.

FFTF REACTOR PHYSICS - P. L. HOFMANNEffect of BeO on FTR Neutronics -*R. B. Kidman and W. W. Little*

In large fast reactors, one can often increase the Doppler coefficient and ameliorate the Na void coefficient by adding a moderator, such as BeO, to the reactor core. A moderator increases the Doppler coefficient by enhancing the resonance flux, and decreases the Na void coefficient by leveling the adjoint flux spectrum.

A brief parametric survey has been performed to investigate the effect of BeO on the FTR neutronics. Items considered include the Na coefficient, Doppler coefficient, critical mass, and flux spectrum.

All calculations are based on the parameters listed in Table II-11. The principal variables are the volume-percent of stainless steel, fuel ($\text{PuO}_2 + \text{UO}_2$), and BeO.

Figure II-39 gives the critical ($k = 1.02$) Pu^{239} inventory for various concentrations of BeO. The corresponding core-averaged uranium to plutonium ratio and spectrum indices are shown in Figures II-40 and II-41, respectively. All calculations were performed using the DTF-IV transport code (cylindrical geometry) and a modified version of the Russian 26-group cross sections.

The Doppler coefficient as a function BeO concentration is given in Figures II-42. As predicted, note that BeO enhances the Doppler coefficient in cores containing a reasonable percent of fertile fuel.

The central Na perturbation worth in terms of ρ_p and δk is given in Figures II-43 and II-44, respectively, for various BeO loadings. The total Na void worth (for Case 5), as a function of the fraction of core voided, is described in Figure II-45. The

TABLE II-11. Effect of BeO on FTR Neutronics

800-liter core			
L/D = 0.9			
2 Equal-volume radial zones			
Pu: 85% 239, 15% 240			
F.P. Corresponding to a 2 a/o burnup			
$K_{\text{eff}} = 1.02$			
40 vol% Na in core			
<u>Case</u>	<u>Vol% SS</u>	<u>Vol% (Fuel + BeO)</u>	<u>Vol% BeO</u>
1	40	20	0
2	↓	↓	5
3			10
4	30	30	0
5	↓	↓	5
6			10
7	20	40	0
8	↓	↓	5
9			10

FIGURE II-39. The Change in Pu^{239} Inventory as a Result of Varying the Volume Percent of Fuel + BeO.

FIGURE II-40. The Change in U/Pu Ratios as a Result of Varying the Volume Percent of Fuel + BeO

FIGURE II-41. The Change in Energy Spectrum as a Function of the Volume Percent of Fuel + BeO

FIGURE II-42. Doppler Effect as a Function of Volume Percent Fuel + BeO

FIGURE II-43. The Central Na Perturbation Cross Section as a Function of the Volume Percent of Fuel + BeO

FIGURE II-44. Central Na Worth as a Function of Volume Percent Fuel + BeO

FIGURE II-45. Reactivity Change as a Function of Percent Core Voided

latter curve was generated using the DTF-IV code in spherical geometry.

It is clear from the preceding figures that BeO has a beneficial effect on both the Doppler and the Na void coefficient. The variations, however, are not pronounced. Furthermore, BeO may have a detrimental side effect on other reactor characteristics (fuel and thermal performance). These effects will have to be investigated in detail before the question of BeO addition to the FTR is resolved.

Considerations on the Fuel-Rod Plug Concept - A. E. Waltar

Because mixed oxide fuels are known to exhibit radial densification as a function of burnup, it is of interest to determine whether or not the resultant void in the center of a fuel rod presents a significant safety problem. Should melting occur in a rod containing a central void, fuel from a low reactivity worth region could flow unimpeded through the void into a high

worth region creating a more reactive configuration. One method of restricting such fuel movement is to insert plugs at selected axial positions along the fuel rod. Two aspects of this problem have been considered: the reactivity consequences of fuel melting with central voids present, and the effect on the Doppler coefficient of inserting plugs in the fuel rods.

Two idealized models were chosen to estimate reactivity changes associated with the melting of partially voided fuel rods. The first model (Case I) assumes an axial void along the entire length of the fuel as the initial condition. The worst possible condition in this model is for the entire fuel rod to melt and compact symmetrically toward the center of the core. The second model (Case II) assumes that initially only the upper two-thirds of the rod contains a void, and that only this portion of the rod melts in the accident. Fuel movement is then assumed to take place under the force of gravity. Plugs were located symmetrically above and below the fuel rod center line in Case I, whereas they were located only in the top half of the fuel in Case II. Figures II-46 and II-47 illustrate the two models chosen.

For the selected plug locations, the calculated results can be approximated by:

$$\text{CASE I : } \delta k = \frac{0.7 (V.F.) (\delta M/M_0)}{1 + N}$$

$$\text{CASE II : } \delta k = \frac{0.5 (V.F.) (\delta M/M_0)}{1 + N}$$

FIGURE II-46. Case I for 20 vol% Void

FIGURE II-47. Case II for 20 vol% Void

where

δk = reactivity change caused by fuel melting

V.F. = void fraction in the fuel rod

N = number of plugs in the top half of the element (Note that for the same N , the total number of plugs in the entire rod for Case I is twice that of Case II.)

$(\delta M/M_0)$ = fraction of fuel in the rod.

Figure II-48 illustrates these functional fits for the case of 20 vol% void. Optimal plug spacing was not considered in detail since it was found to be only a minor refinement.

As a numerical example, consider the melting of the entire central sub-assembly using the first model. If

FIGURE II-48. Comparison of Calculated Points with Functional Fits

CASE I : Worst Case; i.e., 100% Axial Void and all Fuel Collapses Toward Core Center

CASE II: Assume 67% Axial Void and all Fuel Collapses Under Gravity

the void fraction is 10% and the reactor is assumed to consist of 100 equally loaded subassemblies, then, for $\beta = 0.003$,

$$\delta k = 0.0007 \rightarrow 0.23$$

for the case of no plugs. One plug in the top half (two plugs total) would cut this value in half.

Since the introduction of plugs into the fuel rods would be done primarily to reduce the consequences of this type of accident, it is important to note the effect of the plugs on other parameters. Of paramount importance is the effect of plugs on the Doppler coefficient.

Four substances were considered as possible candidates for plug materials; Ta, W, SS, and BeO.

Based upon the models chosen for this investigation, the inclusion of plugs in the fuel rods would significantly reduce the reactivity insertion accompanying a fuel melting accident of the type postulated; however, the postulated accident with no plugs at all would not be very severe. Further, the effect on the Doppler coefficient of including such plugs is substantial, especially for Ta and W. In addition, the Ta and W plugs would adversely affect the sodium void coefficient and increase the critical mass. To the extent considered in this study, only BeO plugs would make a consistently favorable contribution to the safety problem.

Effect of Crystalline Binding in UO_2 on the U^{238} Doppler Coefficient

R. E. Schenter

The change in the Doppler coefficient due to crystalline binding effects has

been studied for U^{238} in a UO_2 lattice for fast reactor systems. Use has been made of the "weak binding," or the short compound nucleus lifetime (ISCNL) approximation,⁽¹⁾ where the U^{238} atoms in the UO_2 lattice are treated as a free gas with an effective temperature.* Calculations were made to determine the effective temperature versus temperature dependence using the experimental phonon distribution of Dolling, et al.⁽²⁾ and comparisons were made with results obtained assuming a Debye model for the crystal. It was found that for temperatures above 300 °K the effective temperature dependence was well represented by the Debye model ($\theta_D \sim 620$ °K). This result implies that the crystalline binding effects will be important only for temperatures less than or equal to the Debye temperature θ_D . Inclusion of the binding effects then leads to a reduction in the calculated value of the Doppler coefficient in the lower temperature region, which is of the order of 30% at room temperature (~ 300 °K). The temperature dependence of the reactivity was calculated for

1. C. R. Adkins, P. J. Persiani, R. N. Hwang, and J. J. Kaganove. "The Chemical Binding Effects on the Resonance Line Shapes of Uranium-238 in a UO_2 Lattice." Washington Conference on Neutron Cross Section Technology (March 1967).
2. G. Dolling, R. A. Cowley, and A. D. B. Woods. "The Crystal Dynamics of Uranium Dioxide," *Can. J. of Phys.* 43, 8, 1397 (August 1965).

* This should be a good approximation for neutrons with energies which are important for Doppler coefficient calculations in fast reactors (~ 2 keV). See Reference 1 for a discussion of this point.

the ZPR-III assembly 47 using the FCC fundamental mode code⁽³⁾ with a modified version of the Russian 26-group data compilation as input.⁽⁴⁾ Incorporation of the crystalline binding effects in this calculation caused the Doppler coefficient ($\frac{dK}{dT}$) to diverge from the " $\frac{1}{T}$ " rule in the low temperature region in qualitative agreement with the experimental results of Reynolds and Stewart.⁽⁵⁾

Effective Temperature Model

The effective temperature T_{eff} in the SCNL approximation is given in terms of $\bar{E}$ the average energy per mode of oscillation as⁽¹⁾

$$kT_{\text{eff}} = \bar{E} = \int_0^{\infty} d\nu g(\nu) (h\nu/2) \coth \left(\frac{h\nu}{2kT} \right), \quad (1)$$

where ν is the phonon frequency and $g(\nu)d\nu$ is the phonon frequency distribution, number per unit frequency interval. The results of using Equation (1) are given in Figure II-49 for the Dolling⁽²⁾ phonon distribution (Figure II-50) and for the Debye distribution, where $g(\nu)$ is given as⁽⁶⁾

$$g_D(\nu) = 3 \frac{\nu^2}{\nu_D^3} \text{ for } \nu \leq \nu_D \quad (2a)$$

$$= 0 \text{ for } \nu > \nu_D, \quad (2b)$$

and the Debye temperature θ_D is defined as

$$k\theta_D = h\nu_D \quad (3)$$

The description of a crystalline substance is very often given in terms of its Debye temperature so that crystalline effects in a fast reactor are sometimes characterized by a single number θ_D . This description is approximate, since in general the crystal does not have a Debye phonon distribution. One can, however, define an effective Debye temperature which varies with the medium temperature and gives the correct answer for a given physical property. The results of this type of prescription are shown in Figure II-51, where the effective Debye temperature versus temperature is given using the average energy $\bar{E}(t)$ and the specific heat $C_V(T)$.^{*} The results show (Figures II-49 and II-51), that to a good approximation, for temperatures above 300 °K, U^{238} in a UO_2 lattice is described by a Debye model with a Debye temperature of 620°.

3. W. W. Little, Jr., and R. W. Hardie. "FCC, A Fundamental Mode Code for Fast Reactor Analysis," BNWL-234, March 1966.

4. I. I. Bondarenko. "Group Constants for Nuclear Reactor Calculations," (Consultants Bureau, New York, 1964).

5. A. B. Reynolds and S. L. Stewart. "Application of the SEFOR Critical Experiments at ZPR-3 to SEFOR," International Conference on Critical Experiments and Their Analysis, P. 41 (October 1966).

6. Two-Dimensional Transport Code from the Los Alamos Scientific Laboratory, (unpublished).

* The specific heat curve $C_V(T)$ is given to indicate how the effective Debye temperature varies with a particular physical property. Also, the specific heat dependence is the experimental quantity which is usually used to determine the Debye temperature of a given substance (see Reference 1).

FIGURE II-49. Effective Temperature Versus Temperature for the Dolling Phonon Distribution and the Debye Distribution ($\theta_D = 620$ °K)

FIGURE II-50. Dolling Phonon Distribution for U^{238} in a UO_2 Lattice

FIGURE II-51. Effective Debye Temperatures Versus Temperature for the Dolling Phonon Distribution

Doppler Coefficient Dependence

In order to explicitly show the crystalline binding effects on the Doppler coefficient, a $1/T$ dependence in the free gas model has been assumed so that

$$\left. \frac{dK}{dT} \right|_{FG} = \frac{C}{T}, \quad (4)$$

where C is a constant. Integrating Equation (4) yields

$$k_{FG}(T) = A + C \ln T. \quad (5)$$

Using the effective temperature model then gives

$$k_{Bound}(T) = k_{FG}(T_{eff}) \quad (6a)$$

$$= A + B \ln T_{eff}. \quad (6b)$$

Finally, differentiating Equation (6) and substituting in Equations (4) and (5) produces the desired expression for the bound Doppler coefficient in terms of the free gas coefficient

$$\left. \frac{dK}{dT} \right|_{Bound} = \left\{ \frac{T}{T_{eff}} \frac{dT_{eff}}{dT} \right\} \left. \frac{dK}{dT} \right|_{FG}. \quad (7)$$

Results, using the above equations, showing the temperature dependence of the deviation from the free gas model due to crystalline binding effects are given in Figure II-52 and II-53.

FTR Phase-A Critical Experiments and Preliminary Analyses - R. A. Bennett, S. L. Engstrom, Q. L. Baird

FTR Phase-A Critical Experiments consisted of measurements of the worths of in-core and reflector control rods, measurements of spatial distributions of small sample worth, and reaction rate distributions. Rod worths were also measured in two and three rod arrays to investigate interactions. Heterogeneity and enrichment effects were also determined. This work was done in support of the design of the FTR control and safety systems. The results are being used in experiment-theory correlations to verify calculation models used in FTR neutronics design. This report summarizes the experiment-theory correlations completed to date.

Summary

Worths of heterogeneous rods at in-core positions can be calculated to within ~8% using relatively simple two-dimensional transport models. This is demonstrated in the comparisons of experimental and calculated results presented in Table II-12. Refinements of the model are expected to improve the agreement and are discussed in the body of this report.

Calculated worths of reflector rods and their interactions appear to be in better agreement (2-4%) than for the in-core rods as shown in Table II-13. However, the calculations completed so

FIGURE II-52. Crystalline Binding
Effects on the U^{238} Doppler Coefficient

FIGURE II-53. Reactivity Change Versus Temperature for the ZPR-III Assembly 47 Including Crystalline Binding Effects

TABLE II-12. Preliminary Experiment--Theory Correlation, In-Core Rods

Rod Position, in. Inserted	Rod No. and Description	Rod Worth		Correlation, E/C
		Experimental ^(a) (E) % $\Delta k/k$	Calculated ^(b) (C) % $\Delta k/k$	
8	1 (enr)	0.0029	--	1.011
12		0.0268	0.0265 ^(a)	
19		0.2870		
27		0.8920		
35		1.484		
42		1.720		
54		1.738		
54	1 (nat)	0.746	0.779	0.958
54	2 (enr lump)	1.528		
54	3 (enr)	1.734	2.023	0.857

(a) Experimental uncertainties are $\pm 2\%$.

(b) Calculational (numerical) precision is better than $\pm 0.005\% \Delta k/k$.

TABLE II-13. Enriched Reflector Control Rod Worths

Rod Locations ^(a)	Experimental ^(b) (E) % $\Delta k/k$	Calculated ^(c) (C) % $\Delta k/k$	Correlation, E/C
P9	0.417		
P23	0.412		
P9, P23	0.821	0.838	0.980
L10	0.392		
T10	0.382		
L10, T10	0.754		
L10, T10, P23	1.188		
L10, T10, P9	1.097		
N9	0.350		
R9	0.350		
N9, R9	0.674		
N9, R9, P23	1.092		
N9, R9, P9	0.939	0.977	0.961

(a) Rod locations are shown in Figure II-60.

(b) Experimental uncertainties are $\pm 2\%$.

(c) Calculational (numerical) precision is better than $\pm 0.005\% \Delta k/k$.

far do not correctly account for axial leakage nor rod heterogeneity. These effects do tend to cancel, however, and it is expected that the reflector rod calculations are at least as good as the in-core rod calculations. Heterogeneity and leakage effects are discussed in more detail in the main body of this report.

Experiments

All experiments were performed in ZPR-III in either Assembly 48 or in Assembly 48A, which was a modified version of Assembly 48. Assembly 48 as used for the in-core rod experiments is shown in Figures II-54, II-55, and II-56. A cross section of a typical core drawer is shown in Figure II-57. Cross sections of distributed and lumped rod assemblies are shown in Figure II-58 and II-59. Two heterogeneities that complicate the analysis of these experiments are to be noted in Figure II-54 and II-55. First, the safety rods are core drawers enriched in plutonium that are asymmetrically located and the asymmetry is different in each half of the reactor. Second, the distribution of materials in a unit cell of the core is asymmetric, which in effect shifts the neutronic center of the core off of the geometric center of the system. These effects may account for some of the disagreement between theory and experiment. The safety rod asymmetry has been evaluated experimentally⁽⁴⁾ and is discussed in more detail later on.

The reflector rods were measured in a modified version of Assembly 48 (48A) that had two diametrically opposite 100° sectors of the blanket re-

placed with a nickel-steel-sodium reflecting mixture. These reflector sectors were used to provide a neutron spectral environment for the reflector rod experiments that resembled that of an FTR more closely than would the spectrum of the U^{238} blanket. A cross sectional diagram of Assembly 48A is shown in Figure II-60. The horizontal section is similar to Figure II-56 with the inconel sectors extending over the core and end blankets. It should be noted that while safety rod asymmetry remains, here it is the same in both halves. The locations for experimental rods are indicated in Figure II-60 by letter-number combinations as used in Table II-12. These locations were selected to provide varying degrees of rod interaction.

Calculations

Calculations of the worths of both in-core and reflector rods were made with the transport codes DTF IV⁽⁵⁾ and 2DF,⁽⁶⁾ which are one and two dimensional models, respectively. Five group cross section sets were collapsed from a modified version of the Russian 26-group set⁽⁷⁾ using fundamental mode diffusion calculations of spectra in the code FCC.⁽⁸⁾ Cross section sets

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7. W. W. Little, R. W. Hardie, and L. L. Maas. "Multi-group Analysis of Selected Fast Critical Assemblies, BNWL-347.
 8. W. W. Little, Jr., and R. W. Hardie. FCC - A Fundamental Mode Code for Fast Reactor Analysis, BNWL-234. Pacific Northwest Laboratory, Richland, Washington, March 1966.

FIGURE II-54. Critical Assembly 48
(Half #1)

FIGURE II-55. Critical Assembly 48
(Half #2)

FIGURE II-56. Horizontal Cross Section of Assembly 48 with a Central Sodium Stainless Steel Column

FIGURE II-57. Assembly 48 Drawer Loadings

$1/4 \times 1/2 \text{ B}_4\text{C}$

- a. This rod (Rod #2) is to be compared to a rod similar to Rod #1 (Figure II-58) with the central sodium plate replaced by aluminum with 45% density (Rod #3).
- b. In this figure only the drawer contents are shown, the drawer and matrix tube are as shown in Figure II-58.

FIGURE II-59. Cross Section of Lumped Control Rod Drawer Contents (a,b) Rod #2

FIGURE II-60. Critical for Interaction Experiments, Assembly 48A

were averaged over 26 group spectra for each region of the assemblies as indicated in Table II-14.

The energy bounds of the five groups of the spectrum are 10.5, 0.8, 0.1, 0.01, 0.001 MeV, and 0.025 eV. The twenty-six group spectra are shown in Figure II-61 with these energy bounds illustrated.

In-Core Rod Calculations

Both R-Z and x-y calculations of the worths of in-core rods have been made to obtain some measure of both the axial leakage effects and heterogeneity effects. The safety rod asymmetries were removed by replacing them with the normal core composition. The corresponding necessary increase in core volume was added at the core-reflector interface. The additional amount of core required was established in a series of reactivity experiments performed earlier.⁽⁴⁾

For the R-Z calculations the system was cylindricized as illustrated in Figure II-56, which also shows the radial and axial mesh structure. The distributed rod of Figure II-58 was homogenized by conserving material as were the other zones of the assembly. These homogenized atom densities are presented in Table II-15.

The worth of this homogenized rod was deduced from the difference between the calculated effective multiplications of the assembly containing, first, a sodium-steel column and, second, the rod mixture. The value of 2.023% $\Delta k/k$ is quite large compared to the measured value 1.734% $\Delta k/k$. This difference is in part due to decreased self-shielding introduced by homogenizing the rod. An x-y heterogeneous calculation, to be discussed below, illustrates this.

To perform a good two-dimensional (xy) calculation of a three-dimensional system, one must properly account for axial leakage, which has spatial dependence. In the calculations reported here a single non-spatially dependent approximation has been used. This was established by adjusting the effective height of a one-dimensional (radial) model of the above R-Z configuration. This axial leakage was then applied directly in the xy calculations. The model for the xy calculations is shown in Figure II-62. The square core was chosen to give approximately the same k_{eff} as the unrodded cylindrical core and is considered adequate for these central rod calculations. The calculational models and mesh detail for the distributed and lumped rods are shown in Figures II-63 and II-64.

To test the axial leakage approximation the homogeneous rod calculation was repeated and the value of 2.001% $\Delta k/k$ was obtained, which is in good agreement with the R-Z result. Values of 1.843 and 1.645% $\Delta k/k$ were found for the distributed and lumped rods, respectively, which illustrates the effect of heterogeneity. While

TABLE II-14. Region Cross Section Sets

<u>Region</u>	<u>Spectrum</u>	<u>Criticality Condition</u>
Core	Core	$k_{\text{eff}} = 1$
Blanket	Blanket	$B^2 = 0$
In-Core Rods	Core	$k_{\text{eff}} = 1$
Reflector	Reflector	$B^2 = 0$
Reflector Rods	Core	$k_{\text{eff}} = 1$

TABLE II-15. Compositions for Core 48 and 48A Calculations

		Atom Densities - Atom/b-cm						
		Core	Safety Rod	Hom. Enr. Rod	Hom. Nat. Rod	Inconel 1 Zone 6	Inconel 2 Zone 7	Inconel 3 Zone 8
1	Pu ²³⁹	0.001645	0.003183					
2	Pu ²⁴⁰	0.000106	0.000152					
3	Pu ²⁴¹	0.000011	0.000014					
4	U ²³⁵	0.000016	0.000009					
5	U ²³⁸	0.007427	0.004682					
6	C	0.020767	0.020766	0.006350	0.005244			
7	Fe	0.009985	0.013611	0.026775	0.026775	0.013914	0.013921	0.013788
8	Cr	0.002681	0.003655	0.007189	0.007189	0.003684	0.003686	0.003650
9	Ni	0.001330	0.001798	0.003537	0.003537	0.047514	0.047514	0.047502
10	Mo	0.000206						
11	Al	0.000109	0.001682					0.000361
12	Na	0.006231	0.004154	0.006263	0.006263	0.004176	0.004152	0.003897
13	B10			0.015486	0.003736			
14	B11			0.001584	0.015736			

		Blanket	SS Mesh	Hom. Enr. Rod with Al	Het. Enr. ^(a) Rod with Al	Het. Nat. ^(a) Rod	Het. Enr. ^(a) Rod	SS-Na Column
1	Pu ²³⁹							
2	Pu ²⁴⁰							
3	Pu ²⁴¹							
4	U ²³⁵	0.000083						
5	U ²³⁸	0.039798						
6	C			0.006350	0.030162	0.024909	0.030162	
7	Fe	0.004444	0.004710	0.025831	0.050996	0.051826	0.051826	0.013080
8	Cr	0.001193	0.001264	0.006936	0.013693	0.013915	0.013915	0.003512
9	Ni	0.000587	0.000622	0.003413	0.006737	0.006846	0.006846	0.001728
10	Mo							
11	Al			0.002263	0.021499			
12	Na			0.004175	0.023498	0.023498	0.023498	0.016701
13	B10			0.015486	0.073556	0.017746	0.073556	
14	B11			0.001584	0.007524	0.074746	0.007524	

(a) For distributed or lumped heterogeneous rods.

FIGURE II-61. Calculated Spectra for
Assembly 48A

* See Figures II-63 and II-64 for detail

FIGURE II-62. Calculational Model--
Central Control Rod Experiments (Two
dimensional x-y geometry)

FIGURE II-63. Calculational Model of
Distributed Control Rod (No. 3)

these values still over-estimate the rod worths, they are reasonably close and the differences of 6 and 7%, respectively, can be used as correction factors in design calculations.

Further, while these heterogeneous calculations produce significant improvements in the experiment-theory correlation over that obtained from homogeneous models, the lack of agreement remains undesirably large. Inspection of the calculation model reveals three areas where improvement can be sought with additional effort.

First, a five group energy model of the neutron spectrum has been employed. While this has been shown to be reasonably accurate on fairly homogeneous full-reactor problems, no attempt has been made to demonstrate the same with respect to extremely heterogeneous problems. Group structure can be increased to 26 groups.

Second, an S2 transport scattering model has been used, which is known to be, at best, a weak approximation. Recent efforts of others indicate considerable sensitivity to the order of the Sn approximation. Again, the order of the approximation can be extended to even higher approximation (e.g., S_4).

Third, the choice of cross section averaging schemes is not too well established. Cross sections for a given zone of a heterogeneous system are a function of the energy distribution of neutrons within and outside the zone in question. While this problem is not new, it has been considered a second order problem in fast reactor analysis until recently. Since, as indicated, one is now interested in calculational precision of better than

a few percent, this problem must again be faced in fast reactor analysis as it has been faced in thermal reactor analysis.

A first step in such a refinement would be a fine group (26) calculation of the neutron spectrum in an idealized cylindrical model of the experimental rods and reactor. Using the spectra from the various zones of materials, one could then obtain average (five group) cross sections more closely approximating those required.

Reflector Rod Calculations

The calculations of the worths of reflector rods and combinations of rods is following essentially the same pattern as that established for the in-core rods. The xy geometric model and mesh detail is shown in Figure II-65. It is to be noted that the safety rod must be included in these calculations because no known method of increasing the core radius exists that will not displace the rods. The safety rod, for the 1/4 and 1/2-core calculations, has therefore been placed at the average safety rod position; the asymmetry has been ignored. This should introduce only a small error in the absolute multiplication and spatial flux distribution and only a second order error in rod worths.

Core and blanket compositions remain the same as for the in-core calculations. The reflector compositions are tabulated by zone in Table II-15.

Worths of the reflector rods are again deduced from k_{eff} calculations with and without rods present. With the rods out, the rod locations were filled with reflector materials. For

FIGURE II-64. Calculational Model of Lumped Control Rod (No. 2)

Zone	Material
1	Core "48A"
2	Safety Rod
3	Control Rod
4	Control Rod
5	Control Rod
6	Inconel "1"
7	Inconel "2"
8	Inconel "3"
9	U ²³⁸ Blanket
10	Stainless Steel Mesh

FIGURE II-65. Calculational Model-1/4 Core of APR-III

the two rod combinations computed to date, the agreement is quite good. However, the rods were homogenized and leakage was taken to be that of the unrodded core. Hence, the actual lengths of the rods have essentially been ignored. Heterogeneity and rod length effects have been investigated and found to be approximately 10%, opposite in sign, and hence tend to cancel. One may therefore conclude that this model for calculating reflector rod worths is as adequate as that used for the in-core calculations.

Axial effects of reflector rods are presently being investigated using three-dimensional 2 group diffusion models.

FTR Shielding Studies - *D. R. Marr, C. A. Mansius, and W. L. Bunch*

The attenuation characteristics of a number of alternate shield arrangements have been compared analytically to provide a basis for the conceptual design of the Fast Test Reactor. The removal-diffusion code MAC was used for these calculations. In all cases, the reactor was assumed to consist of an 800 liter cylindrical core with oxide fuel operating at 400 MW. Important considerations in evaluating a shield arrangement are: biological exposure rates in regions to be entered by operating personnel, radiation damage rates in structural components, heat generation rates in shield components, activation rates in the primary and secondary coolant, and radiation environment for sensors in the reactor instrument system.

Table II-16 presents the thickness and composition of four radial shield

arrangements. Results of the calculations are summarized in Figures II-66 and II-67. The first arrangement provides a base case in which the gas-cooled neutron shield is assumed to consist of graphite encased in steel tubes. The calculations indicate this arrangement to be inadequate because of excessive leakage of activation flux to the secondary sodium located outside the neutron shield. In the second case it was assumed that the graphite contained 1 wt% boron. Although the boron was calculated to depress the thermal neutron flux and gamma generation rate throughout the neutron shield region, total neutron leakage to the secondary sodium was still considered to be excessive because the effect of the boron was negligible on the intermediate and higher energy components of the spectrum. In the third case it was assumed that the graphite in the neutron shield region was replaced with serpentine, a mineral containing magnesium oxide, silicon dioxide and bound water of hydration. For the thickness of neutron shield considered, this arrangement would satisfactorily suppress activation in the secondary system. The fourth case simulates a pool concept in which the canned graphite neutron shield of the base case would be submerged in sodium. Although sodium is not an efficient shield material, its effect is quite apparent when used to fill the void regions in the shield.

Base case calculations have also been completed for the upper and lower axial shield regions. For these calculations it was assumed that skewed guide tubes would be used to position

TABLE II-16. Composition and Thickness of Postulated Radial Shield Regions

Region Number	Name	Thickness, cm	Composition, vol%
Base Case 1			
1	Core 1	27.1	{ PuO ₂ = 5.657, UO ₂ = 19.343, 304 SST = 25.0 } Na = 45.0, void = 5.0
2	Core 2	27.1	
3	Skewed Reflector	17.3	
4	Sodium Pool 1	10.0	304 SST = 55.7, Ni = 26.2, Na = 18.1
5	Sodium Pool 2	35.0	Na = 100
6	Tube Reflector	60.0	Na = 100
7	Sodium Pool 3	5.0	Na = 19.4, 304 SST = 17.6, Inconel X = 62.9
8	Vessel	7.0	Na = 100
9	Pipe Space	62.0	304 SS = 100
10	Neutron Shield	152.0	Void
11	Steel Wall	1.0	Steel = 25.7, C = 54.2 void = 20.2
12	Secondary	60.0	Steel = 100.0
			Na = 100
Boronated Case 2, same as base case except for following:			
10	Neutron Shield	152.0	Steel = 25.7, 1% boronated graphite = 54.1, void = 20.2
Serpentine Case 3, same as base Case 1 except for following:			
10	Neutron Shield	152.0	Steel = 25.7, Serpentine = 54.1, void = 20.2
Pool Case 4, same as base Case 1 except for following:			
9	Pipe space	62.0	Na = 100
10	Neutron Shield	152.0	Steel = 25.7, C = 54.1, Na = 20.2

FIGURE II-66. Neutron Flux Density as a Function of Distance into Shield

FIGURE II-67. Gamma Energy Flux Density as a Function of Distance into Shield

fuel within the core. Material densities in the shield regions are average values based on a skew angle of 2.75° and a triangular pitch. The lower and upper shield arrangements shown in Tables II-17 and II-18 are based on dimensions for concept PNL-III. Results of the neutron calculations for these two cases are summarized in Figures II-68 and II-69, where "fast" neutrons are those with $E > 0.06$ MeV. Additional calculations are in progress to determine the effect of alternate compositions.

Heat generation rates in shield regions have been estimated based on

the gamma distributions generated in the MAC calculations. These estimates are summarized in Table II-19 together with the composition and thickness of the shield regions employed in the calculations. Since MAC is a one-dimensional code, the geometric effect is not included in these estimates and the tabulated values should be conservative. Additional calculations are planned for alternate shield arrangements and methods of refining the code to incorporate the geometric effect are being investigated.

TABLE II-17. *Composition and Thickness of Postulated Lower Axial Shield Regions*

Shield Region	Name of Region	Region Width, cm	Region Materials and Effective Densities, g/cm ³			
			Sodium	Chromium	Iron	Nickel
1	Fuel Support	20.3	0.739	0.22	0.80	0.13
2	Stainless Tubes and Sodium	20.3	0.792	0.115	0.421	0.066
3	Stainless Tubes and Sodium	40.6	0.80	0.102	0.376	0.059
4	Stainless Tubes and Sodium	40.6	0.807	0.088	0.326	0.051
5	Stainless Tubes and Sodium	40.6	0.815	0.075	0.278	0.044
6	Stainless Tubes and Sodium	33.0	0.83	0.055	0.192	0.030
7	5 Inch Support Plate	12.7	0.17	1.15	4.23	0.66
8	Stainless Tubes and Sodium	33.0	0.80	0.073	0.27	0.042
9	5 Inch Support Plate	12.7		1.45	5.34	0.84
10	Sodium	38.1	0.84			
11	3 Inch Stainless Plate	7.6		1.45	5.34	0.84
12	Sodium	82.5	0.84			
13	2 Inch Stainless Vessel	5.08		1.45	5.34	0.84
14	Sodium	53.3	0.84			
15	1 Inch Primary Shell	2.5		1.45	5.34	0.84

TABLE II-18. *Composition and Thickness of Postulated Upper Axial Shield Regions*

Shield Region	Name of Region	Region Width, cm	Region Materials and Effective Densities, g/cm ³			
			Sodium	Chromium	Iron	Nickel
1	Fission Gas Plenum	61.0	0.42	0.426	1.57	0.25
2	Process and Hanger Tubes	30.5	0.601	0.387	1.426	0.224
3	Process and Hanger Tubes	71.1	0.634	0.329	1.212	0.190
4	Process and Hanger Tubes	200.7	0.720	0.177	0.651	0.102
5	Tubes and Nozzles	76.2	0.70	0.218	0.801	0.126
6	Gas Plenum Above Pool	50.8		0.203	0.75	0.118
7	Steel Ball Shield	76.2		0.87	3.2	0.50
8	Gas Plenum Above Ball Shield	73.7		0.116	0.43	0.067
9	304 Stainless Steel Plate	5.08		1.45	5.38	0.84

TABLE II-19. *Calculated Heat Generation Rates in the FTR Shield*

<u>Region (vol% Material)</u>	<u>Thickness, cm</u>	<u>Heat Generation Rate, MW</u>
Top Axial:		
Reflector (70 SS, 30 Na)	60	3.9
Plenum (3 SS, 97 Na)	140	0.3
Tube Sheet (97 SS, 3 Na)	7.5	0.04
Upper Pool (8 SS, 92 Na)	260	0.01
Bottom Axial:		
Reflector (70 SS, 30 Na)	60	3.9
Plenum (3 SS, 97 Na)	132	0.31
Intermediate Tube Sheet (97 SS, 3 Na)	12.8	0.03
Inlet (100 Na)	37.0	0.00
Lower Tube Sheet (100 SS)	12.8	0.00
Radial:		
Skew Reflector (56 SS, 26 Ni, 18 Na)	17.32	12.4
Sodium (100 Na)	45.0	1.3
Tube Reflector (18 SS, 63 Ni, 19 Na)	60.0	7.8
Sodium (100 Na)	5.0	0.03
Pressure Vessel (100 SS)	7.0	0.04

FIGURE II-68. Lower Axial Neutron Flux for PNL-III Concept

FIGURE II-69. Upper Axial Neutron Flux for PNL-III Concept

FAST REACTOR FUELS - W. E. ROAKEHydraulic Testing of Prototypic Fuel Pin Assemblies - M. K. Millhollen

A 24 in. long, 7-rod cluster of 0.200 in. diam fuel pins, arranged in a hexagonal array and spaced by 0.040 in. wire spirally wrapped (12 in. pitch) on each rod (see Figure II-70), was flow tested in a water flow loop. Pressure drop was measured with water temperature at 10 and 106 °C for a range of Reynolds Numbers from 55,000 to 170,000.

Previous calculations had yielded a Reynolds Number of 93,000 for sodium flow through the assembly at 900 °F and 35 fps. Test data, converted to sodium flow conditions, show a pressure drop of about 26 psi over the 24 in.-long assembly at FFTF flow conditions. This is well within the original design goal of less than 20 psi pressure drop per foot of fuel zone. Readings taken at six positions over the length of the test assembly indicated that end effects account for less than 3% of the total pressure drop.

Previous calculations on flow through a 331 rod bundle of 0.2 in.-diam fuel rods spaced by 0.040 in. wire wrap showed that a Reynolds Number of 76,500 would be obtained at a full sodium flow condition. When the 7-rod cluster is compared to the 331 rod cluster it can be shown that a sodium flow velocity of 28.8 ft/sec through the 7-rod bundle will provide the 76,500 Reynolds Number. At this flow velocity there will be a pressure drop across the 7-rod cluster of only 22 psi.

Fuel Pin Cladding Analysis - G. L. Fox

Creep data for both irradiated and unirradiated 304 SST were fit to empirical curves of the form $\dot{\epsilon} = a\sigma^B t^{a-1}$, where

$\dot{\epsilon}$ = creep rate, %/hr

σ = stress, psi

t = time, hr

c, B, a = material constants which depend on temperature.

A computer program was written to optimize the constants c , B , and a . Figures II-71 through II-73 show the data and curve fits.

To apply the creep data correlations to problems where the stress varies with time, it is necessary to assume the strain rate is dependent only on the applied stress and prior straining.

Combining the equations $\epsilon = c\sigma^B t^a$ and $\dot{\epsilon} = a c \sigma^B t^{a-1}$ yields $\dot{\epsilon} = a c^{1/a} \sigma^{B/a} \epsilon^{\frac{a-1}{a}}$.

The stress buildup in the fuel pin cladding was assumed to be a linear function of time as a first approximation, i.e., $\sigma = \frac{\sigma_{\max} t}{5000}$. This may be refined later for more accurate calculations. Integrating the creep rate equation over 500 hr provides the strain at prior maximum stresses shown in Figure II-74.

The scatter in data points among various metal specimens and the uncertainties in extrapolation to low stress levels requires a large margin of safety from the 0.2% allowable strain established in the basis of end of

*FIGURE II-70. Seven Rod Flow Cluster
and Basket with Tubes for Attaching
Pressure Sensing Probes*

FIGURE II-71. Creep Rate Data 304 SST, 550 °C

FIGURE II-72. Creep Rate Data 304 SST, 600 °C

FIGURE II-73. Creep Rate Data Type
 304 SST, 650 °C

FIGURE II-74. Allowable Stress with
 Linear Stress Build-up from 0 to Max-
 imum Stress over 500 hr Period

life ductility. This margin of safety is tentatively established to allow strains of only 0.02 to 0.0133%.

When applying Figure II-74, it should be remembered that the stress and strain data are from uniaxial tests. Therefore for a biaxial or triaxial stress system it is necessary to find the equivalent, or effective, uniaxial stress and strain. For a biaxial system such as a thin walled tube with closed ends, where the longitudinal stress is equal to half the tangential stress, the effective stress is 86.6% of the tangential stress. The effective strain is 1.155 times the tangential strain. These values were obtained from the "energy of distortion" condition:

$$S_o = \frac{1}{\sqrt{2}} \left[(S_1 - S_2)^2 + (S_2 - S_3)^2 + (S_1 - S_3)^2 \right]^{1/2}$$

S_o = effective stress
 S_1, S_2, S_3 = triaxial principal stress system and effective strain condition.

Fuel Motion Restrictor Development - *R. E. Bardsley and R. J. Lobsinger*

A mixed oxide FFTF core may require cylindrical restrictors in the fuel pins to limit axial fuel movement during excursions. Several potential restrictor materials are being tested to determine fuel-restrictor and cladding-restrictor compatibility. Two 9 in. long thin-wall (0.008 in.) stainless steel capsules containing UO_2 powder and W, Ta, Ta-10W, Mo, and Cb-10%-5Zr restrictor material specimens were tested at 1200 °F for 336 hr

in argon. Metallographic examination showed no evidence of reaction between these metals and the cladding but electron beam microprobe examination showed Ta and Cb diffusion as much as 0.00025 in. into the cladding.

Irradiation Tests - *J. E. Hanson and L. A. Pember*

Drawings of the fuel motion restrictor test specimens and capsules for irradiation in EBR-II were prepared and are being reviewed. Preparation of the experiment description and hazards analysis was begun.

The diameter of the floating restrictors for the VH-4 screening capsules was reduced to ensure that there will be no pressure contact between the restrictors and the fuel pin cladding. The restrictors were originally designed so that differential thermal expansion would cause contact between the restrictor and the pin cladding under transient conditions. These capsules are designed to generate at 28 kW/ft. At this power generation the fuel will be molten to approximately 50% of the radius.

ARGUS calculations were made to evaluate the expected performance of the thermal flux capsules during transient irradiation. Particular emphasis was placed on determining whether or not the heat sink surrounding the fuel pin would melt under the most severe TREAT transient expected to be required for the tests since the dimensions of the heat sink are based principally on the required steady-state cladding surface temperature. The ARGUS calculations indicate that melting of the aluminum heat sink

should not occur during the planned tests.

An analysis of the possible use of PRTR for thermal flux irradiation testing of FFTF fuel was made. The assumption was made that only peripheral positions and possibly the FERTF would be available for testing purposes. The peripheral positions must be eliminated from serious consideration due to the low available flux. Maximum power generation from a prototypic FFTF fuel pin in a peripheral position would be about 3.5 kW/ft. The FERTF has a flux level high enough to make it an attractive test facility. It appears feasible to irradiate multi-pin tests of full length FFTF pins at near design power generation in this facility. These tests could be either double encapsulated single pins with sodium heat transfer bonds, or multi-pin capsules cooled by sodium convection. Since the PRTR coolant is preheated above the melting point of sodium before power startup, there would be no problem of keeping the sodium molten in the test assembly.

Mixed Oxide Irradiations - L. A. Pember

The capsule designed for irradiation tests was approved for fabrication of mixed oxide fuels of various stoichiometries. Most of the capsule hardware is on hand. These test capsules are designed for irradiation in the MTR at a constant power generation of 16 kW/ft. Three fuel forms (solid pellets, annular pellets and vipac powder) of three stoichiometries, (1.92, 1.95 and 2.00) will be irradiated. One capsule

of each type will be irradiated to exposures of 50,000 and 100,000 MWd/tonne.

Design was completed for a series of mixed oxide fuel motion restrictor screening capsules using prototypic radial dimensions. These capsules will be irradiated in the VH-4 hydraulic rabbit facility of the MTR. Rapid insertion of these capsules into a high flux zone will simulate a transient during reactor operation. These capsules are designed to screen the effectiveness and compatibility of various restrictor candidate materials and designs. The tests will compare the effectiveness of both fixed and floating restrictors of various geometries. The restrictor materials to be tested include tungsten, molybdenum, nickel, 304 SST and depleted UO_2 . The advantages of these tests are the simplicity of fabrication, the short time required for irradiation and the low comparative cost.

Irradiation Testing of Cermet Pins -

L. A. Pember and J. E. Hanson

Two PuO_2 -SS cermet pins and one UO_2 -SS cermet pin that are being irradiated in EBR-II subassembly X011, have reached an exposure of 3.6 at.% burnup for the PuO_2 -SS pins and 3.0 at.% burnup for the UO_2 -SS pins. Goal exposure for these pins is 0 at.% burnup.

Neutron radiography of two PuO_2 -SS cermet pins that were irradiated in EBR-II subassembly X009 was completed. The pins were irradiated to burnup of

approximately 3.28 at.%. Comparison of the neutron radiographs with pre-irradiation radiographs reveal no change in the pins. Gamma scanning of these pins showed symmetrical activity along the fueled length except in two regions of the pellet interfaces. In these regions, the gamma activity was about 50% of the activity in the adjacent fuel. A specific check was made for ^{22}Na activity, but the activity was below the sensitivity range of the gamma scanning equipment.

Mixed Oxide Irradiation Testing -

J. E. Hanson

On January 24, 1967, W. E. Roake, B. R. Hayward, and J. E. Hanson met with members of the AEC-DRDT at AEC Headquarters to discuss PNL's proposed FFTF fuel development irradiation testing program in EBR-II. The significant points and conclusions made during these meetings were:

1. The EBR-II capacity for experimental subassemblies which have a negative reactivity effect has been reached insofar as current experiment loading plans are concerned. Two such subassemblies have been allocated for FFTF fuel development irradiation program requirements.
2. Approximately 10 to 12 experimental subassemblies, which have reactivity worths approximately equal to an EBR-II subassembly, can be accommodated.
3. As a result of these discussions, a revised and expanded program which would emphasize statistical testing as a basis for assuring performance reliability of the initial FFTF driver fuel was out-

lined. Priorities for the definition and evaluation of key fuel performance parameters were established as follows:

- Axial fuel motion restrictor: evaluation of necessity and effectiveness.
- Fuel Form: Solid pellet
Annular pellet
Vibrationally compacted powder
- Goal burnup range for irradiation experiments: 45,000 MWd/tonne average, 75,000 MWd/tonne peak.
- Fuel smeared density: 85 to 90% TD
- Stoichiometry: 1.95 to 2.00
- Composition (for mixed oxide test pins): 25% PuO_2 (total Pu)

4. It was agreed that the program outlined above would form the basis for detailed discussion during presentation, by PNL, to the EBR-II Evaluations Committee on February 1, 1967, at Argonne National Laboratory.

On February 1, 1967, B. R. Hayward and J. E. Hanson presented the revised EBR-II irradiations program proposed for FFTF fuel development to the EBR-II Evaluations Committee. The program details will be published as an informal report during March. In essence, the EBR-II irradiation portion of the FFTF fuel development program consists of three major tasks.

The first of these is devoted to statistical evaluation of key fuel parameters and performance under prototypic operating conditions to provide data for confident selection of the fuel design. The second task is concerned with demonstrating the reliability of the pin design on a statistical basis under prototypic operating conditions. The third task

is devoted to the demonstration of performance reliability, on a statistical basis, of the subassembly design.

Fuel Temperature Analysis - G. L. Fox

A fuel pin in the FFTF reactor will be subjected to temperature transients if there are sudden reactivity changes in the core. The magnitude of the temperature transients is important in evaluating the possibility of fuel melting and the effects of additional thermal stresses in the cladding. The following model for reactivity change was used:

1. A ramp increase in reactivity occurs at rates of 1, 2, 3, or 4\$/sec.
2. At 10% overpower, neutron detectors will initiate scram.
3. There is a 0.15 sec delay in scram circuitry.
4. The control rods remove as much as 30\$ reactivity according to equation

$$-\rho = R \left(\frac{\pi^2 c^3 g^3}{12H^3} \right) t^6$$

ρ = reactivity insertion

R = control rod worth (Used 30\$)

c = number of g's (Used 1)

g = acceleration due to gravity

H = extrapolated core height (Used 4.34 ft)

t = time

This reactivity change is applied to the average fuel pin using Nutiger-- a transient heat transfer computer code with reactivity inputs and feedbacks.

The particular reactivity feedbacks used in this analysis were those of Doppler, fuel expansion, and coolant density.

The reactivity change produced an average fuel pin power variation with time. Figures II-75 and II-76 show the results of a preliminary run at 2\$/sec ramp insertion. Note that although the power more than doubles momentarily, the hottest point in the fuel only rises about 200 °F. The greatest increase in cladding temperature was about 15 °F; this temperature was still increasing at the end of the time scale.

The power history for the hottest pin will be calculated from the average pin data. This will be re-submitted to the heat transfer computer code to obtain complete hottest fuel pin temperature histories. After the analysis of temperature histories resulting from scram at 10% overpower, the procedure will be repeated for scrams at 20% excessive coolant temperature.

Figure II-77 presents a temperature distribution for the hottest pin at steady state conditions. Note that the hottest outer surface cladding temperature is at the coolant exit.

The thermal hydraulics computer code MANTA is being converted for operation on the 1107 computer. Trial runs were completed which indicated that the code is ready for operation. MANTA is especially suited for determining steady state flow distributions in constant channel geometries with coolant mixing and variations in pin power generation.

Fuel Design Analysis - G. L. Fox

Fuel Sintering

During reactor operation, an oxide fuel element will undergo sintering

FIGURE II-75. Reactivity and Power Histories

FIGURE II-76. Temperature Histories
for Average Fuel Pin

FIGURE II-77. Hottest Pin at Steady
State Conditions

of the fuel particles in sufficiently hot regions. This sintering is a fuel densification which promotes a central void and raises the thermal conductivity of the sintered oxide. Both of these effects lower the central fuel temperature. The fuel temperature and extent of the central void are important factors in fuel melting and slumping analysis.

A mathematical model was developed to estimate the effect of sintering. It is based on a stress-free pellet inside a tube. The pellet may expand a desired amount upon heating. All oxide below the sintering temperature is assumed to remain at its initial density and thermal conductivity. All oxide above the sintering temperature is assumed to sinter to a higher density and conductivity. The hottest fuel temperature and core void diameter are evaluated and printed as outputs.

Preliminary results for the hottest fuel pin in a typical FFTF core at steady state operating conditions are shown in Figures II-78 and II-79. Figure II-78 compares the computed fuel temperatures with those obtained from a model that neglects the effects of sintering. Ignoring the sintering will result in overpredicting the temperatures by about 500 °F. Figure II-79 shows the size of core void developed by the sintering.

Fuel Compatibility

R. M. Crawford

Seven fuel-cladding compatibility test capsules were sodium bonded. The capsules are approximately 2 in. long and contain one fuel pellet each (Figure II-80).

The first three capsules used to check out bonding techniques contained UO_2 pellets clad in 304 SST. Destructive examination showed excellent wetting of both the fuel and cladding. The problem of insufficient sodium in the capsule experienced in earlier bonding attempts was eliminated by pressurizing the sodium column above the capsules with 12 psi of helium during the soaking and cooling part of the bonding cycle.

Four UO_2 -25 wt% PuO_2 /304 SST compatibility capsules were bonded and delivered to the Ceramics Research Unit for testing and evaluation. These capsules were similar to the three test capsules.

Thin Walled Tubing - *R. E. Sharp, R. L. Brown, Jr., C. A. Burgess, R. J. Lobsinger, and J. R. Ryden, Jr.*

A semiautomatic tube gaging machine has been developed for dimensional gaging fuel pin cladding (0.250 in. OD x 0.016 in. wall) used in the FFTF irradiation test program.

An existing thin wall tubing tester was modified and tests were performed to determine its capability in checking the FFTF stainless steel cladding tube for flaws.

The results indicated that the tester was not capable of performing the desired test. Additional modifications were then made to optimize the test. Various transducers were tried as well as different entry angles of the sound into the tube. The best results were obtained with a 1/8 in. diam, spherically focused, 12 MHz ceramic transducer at entry angles that do not correspond to that required for shear

FIGURE II-78. Sintering Effect on Hottest Fuel Temperature (preliminary data)

FIGURE II-79. Void Core Produced by Sintering (preliminary data)

FIGURE II-80. UO_2 -25 wt% PuO_2
Compatibility Capsule

wave generation. With this arrangement, simulated flaws 0.001 in. deep x 0.030 in. long could be detected.

Twenty-five tubes were tested with this equipment. Only eight were accepted. At the recommended entry angle for shear wave generation, the 0.001 in. deep flaw was masked by tube noise. Because this noise was thought to be related to grain size, two tubes were metallographically examined to determine the grain size. The tube that had presented the least noise had considerably smaller grains than the tube with the most noise.

Ultrasonic testing of about 100 lineal feet of 1.125 in. OD x 0.045 in. wall stainless steel tubes and of about 30 lineal feet of 0.568 in. OD x 0.035 in. wall stainless steel tubes was started.

A differential probe for eddy current testing of tubing was constructed consisting of two coils of 100 turns each of 0.002 in. diam magnet wire. This probe was used with a 100 kHz eddy current tubing tester to scan a section of tubing having artificial notch defects. The tubing was very non-uniform and the signals from the artificial defects could hardly be seen above the signals due to other variations, such as grain size. This indicates that the selection of probe design and operating frequency will be controlled principally by the need to discriminate between discontinuities and other variables rather than by a need for maximum sensitivity. An experiment is being performed to determine the response of the probe assembly to a standard defect at several frequencies in the range 10 kHz to 10 MHz.

This experiment should allow a better understanding of the way in which tester output varies with frequency and assist in the selection of the optimum operating frequency.

Vibrational Compaction of Fuel Pins -

R. E. Bardsley

Vibrational compaction tests were begun using pneumatically impacted, substoichiometric (O/M = 1.95) UO_2 -25% PuO_2 fuel material. Fuel was compacted to 82 to 84% TD in. pins 0.220 in. ID, 23 in. long, to an active fuel length of 13.5 in. Densities of the pneumatically impacted feed material was 94 to 97% of theoretical. Tests are being continued.

Two depleted UO_2 pellets were placed at the bottom of the fuel column to reduce the end cap temperature. During vibration, some of these pellets tended to rise or float. Limited tests with a single pellet showed no pellet movement. Shorter vibration times accomplished in part by using higher density feed material, will also reduce or eliminate upward displacement of the pellets.

Mixed Oxide Fuel Fabrication Process -

R. E. Bardsley and C. A. Burgess

Three hundred and fifty UO_2 -25 wt% PuO_2 pellets were fabricated for the MTR-ETR irradiation test series. Sintered densities were 85.5 to 87.5% of theoretical. The diameter variation within a process batch of sintered pellets was ± 0.001 in.; between batches, the diameter range was ± 0.001 in. Further characterization of the feed powders and other process variables will be required to reduce this

variation. Control of the sintered diameter of pellets from among powder batches will permit elimination of pellet grinding.

Four hundred depleted UO_2 pellets nominally 0.220 in. diam and 0.500 in. long were fabricated. These pellets will be used at each end of the fuel column in the irradiation test pins to reduce the end cap temperature. Previously, two 1/4 in. long pellets were used at each end, but X-ray examination showed that the pellets tended to separate and float during vibrational compaction.

Vibrational compaction and tap density tests are being conducted on a variety of fuel powder blends. Blends used in previous studies required specific particle sizes, resulting in poor utilization of the feed powder. These current tests are directed at optimizing both compactability and high feed powder yield.

Equipment is being modified and relocated to establish an integrated mixed UO_2 - PuO_2 pellet fabrication facility. The various 37 and 19 pin irradiation test subassemblies will require several thousand pellets which would be difficult to fabricate with present manual pressing facilities. An essential part of the current modifications include the hooding of an automatic Stokes pellet press for plutonium work.

Hot Channel Factors - M. P. Burns

A preliminary analysis of FFTF hot channel factors was completed. The 3σ (99.87% confidence) variance in the hot-test channel was calculated to

be 423 °F, resulting in an approximated engineering power multiple of about 1.14

To predict the highest fuel temperature likely to occur in the FFTF during normal operation, one must consider possible deviations from specified or predicted conditions. Deviations likely to cause elevated temperatures include coolant and flux maldistribution, dimensional and constituent variations in the fuel and cladding, and changes in heat transfer coefficients and thermal conductivity.

Predicted values for maximum deviations were statistically analyzed to yield a preliminary 3σ variance for the temperature in the hottest pin. Table II-20 lists these values. The temperature drops through the fuel, gap, cladding, and film were calculated by the Tiger V code for a 0.25 in. OD pin with 0.016 in. cladding, operating at an average power of 15 kW/ft. An engineering factor of 1.2 which was assumed in Tiger V calculations was eliminated from the temperature drops. The effects of sintering were neglected in the statistical analysis.

A normal distribution was assumed for each deviation. The maximum deviation was assumed to be the 3σ variance, which is equivalent to assuming that the probability of a particular uncertainty causing a temperature increase greater than that shown in Table II-20 is only 0.0013. Some deviations, such as flux maldistribution, tend to increase all temperature drops at one point in the pin; these factors are added before $\Sigma 3\sigma$ is

TABLE II-20. FFTF Hot Channel Factors

Uncertainty	Factor for Coolant Temp. Rise* $\Delta T_C = 300^\circ\text{F}$		Factor for Temp. Drop Through Fuel** $\Delta T_F = 2535^\circ\text{F}$		Factor for Gap Temp. Drop+ $\Delta T_G = 494^\circ\text{F}$		Factor for Temp. Drop Through Cladding $\Delta T_{cl} = 75.8^\circ\text{F}$		Factor for Temp. Drop Through Film $\Delta T_{fl} = 14^\circ\text{F}$		$\Sigma 3\sigma$	$(\Sigma 3\sigma)^2$
	F	3σ	F	3σ	F	3σ	F	3σ	F	3σ		
Coolant Maldistribution	1.20	60.0									60.0	3600.0
Deviations from Nominal Dimensions (Includes Pin Bowing)	1.03	9.0	1.01	25.4	1.05	247.0	1.05	3.8	1.10	1.4	9.0 25.4 247.0 3.8 1.4	81.0 645.2 610.1 14.4 2.0
PuO ₂ Maldistribution	1.03	9.0	1.03	76.1	1.03	14.8	1.03	2.3	1.03	0.4	102.6	10530
Flux Maldistribution	1.04	12.0	1.04	101.4	1.04	19.8	1.04	3.0	1.04	0.6	136.8	18710
Burnup	1.05	15.0	1.05	126.8	1.05	24.7	1.05	3.8	1.05	0.7	171.0	29240
Control Band Uncertainty	1.05	15.0	1.05	126.8	1.05	24.7	1.05	3.8	1.05	0.7	171.0	29240
Gap Heat Transfer Coefficient					1.30	148.2					148.2	21960
Thermal Conductivity			1.10	253.5							253.5	64260
							1.10	7.6			7.6	57.8

$$\Sigma (\Sigma 3\sigma)^2 = 178950.0$$

$$[\Sigma (\Sigma 3\sigma)^2]^{1/2} \cdot 3\sigma = 423^\circ\text{F}$$

* $\Delta T_C = 300^\circ\text{F}$ from inlet to hottest point on hottest pin.

** T_m
assuming $\int_0^T k dt = 62\text{W/cm}$
assuming
+ $h_G = 1500\text{ Btu/ft}^2\text{-hr-}^\circ\text{F}$

squared. The total 3σ value for the hottest pin is defined as $[\Sigma(\Sigma 3\sigma)^2]^{1/2}$.

For a nominal maximum fuel temperature of 3400 °F, a 3σ value of 423 °F implies a 0.9987 probability that the maximum temperature will not exceed 3823 °F. With a 2σ variance, there is a 0.977 probability that the maximum temperature will not exceed 3682 °F.

These values will be refined as additional information becomes available, particularly with respect to coolant flow maldistribution and fuel pin and subassembly manufacturing tolerances.

SUMMARY
SECTION III

DEVELOPMENT - F. W. WOODFIELD

Process and Materials Technology -
J. M. Davidson

Fast Reactor Corrosion and
Chemistry -
D. W. Shannon

The PNL nitrogen analysis procedure involves a modified Kjeldahl distillation in which the sodium sample is dissolved in alcohol and the nitrogen is distilled as ammonia for analysis. Calibration of the procedure with ammonium chloride standards is complete, and work with uranium nitride (UN) standards is now in progress. Initial results indicate the procedure is responsive to the presence of UN.

PNL analytical procedures for oxygen in sodium* are now considered satisfactory for concentrations of oxygen in excess of 5 ppm.

The apparatus to analyze total carbon by the combustion gas chromatograph method is almost complete.

The reaction system for hydrogen measurement has been designed and some of the required equipment has been obtained.

Design of a one gal/min sodium purification loop for cold trapping or hot trapping has been completed, and the detailed design is about 30% complete.

The feasibility of measuring oxygen in sodium by distilling the sodium, reacting the Na₂O with graphite to form

carbon monoxide, and measuring the carbon monoxide with a gas chromatograph has been demonstrated.

A commercial gas purification system was purchased for use in conjunction with an analytical drybox obtained from the CANEL equipment.

The PNL shop-built gas purification system was modified to utilize a heated zirconium bed for oxygen removal in place of the copper bed used previously.

Procedures for removing sodium from the FFTF piping systems are being evaluated. Ethyl alcohol is presently being evaluated as an agent for removing sodium from nondrainable piping system deadlegs.

In the work on radioisotope transport from the FFTF core, calculations have been performed on the generation of several long-lived isotopes in 304 SS.

Mechanical Metallurgy -
T. T. Claudson

The liquid-metal capsule GEH 22-3, containing 316 SS tensile specimens, has been irradiated. The liquid-metal capsule GEH 22-4, containing forty austenitic stainless steel specimens, has been placed in the ETR for irradiation. An inert (helium) atmosphere out-of-reactor capsule has been

* Using mercury amalgamation

assembled and is ready for its initial thermal cycle.

Optical metallography, electron microprobe analysis, and replica electron microscopy are being performed on AISI 304 SS tensile specimens.

Tensile tests on annealed AISI 304 SS irradiated to about 1.7×10^{22} total fluence at about 1000 °F (540 °C) in the EBR-II have been completed.* Optical metallography of the EBR-II specimens is in progress. Twelve MK-A-type, space-available pins have been charged into two subassemblies and inserted into the core of the EBR-II. Development of the biaxial sodium environment creep test apparatus for testing irradiated specimens and controls is continuing.

Stress-rupture tests were completed on eight pretreated Inconel 600 specimens which had been irradiated. X-ray diffraction phase analysis has also been completed. Microprobe analysis on a Nimonic 80-A sample has taken place.

A strain-versus-time curve was obtained for an in-reactor creep test conducted on Hastelloy X-280 at 1000 °C and 3500 psi stress. Further creep rupture tests on AISI 304 SS have been performed.

Construction and preliminary testing of a heat transfer apparatus that will be used to study the effect of temperature gradients and thermal diffusion on interstitial solute redistribution and bubble motion in cladding alloys has been completed.

The glow discharge technique is being refined for low temperature hy-

drogen permeation, effusion, and defect interaction studies of irradiated reactor alloys.

The monitor materials for the intercalibration test have been irradiated in the NTR-VH-2 rabbit facility. A two-dimensional transport-theory calculation of the MTR core and rabbit facilities has been initiated.

Reactor Metals - S. M. Gill

Documents and articles are being abstracted, cataloged and evaluated for the FFTF materials property handbook. An outline of materials of initial interest and a table showing property categories have been developed.

Each of the seven candidate FFTF reactor concepts was evaluated with respect to the metallurgical development requirements. In addition a review was conducted to assess the metallurgical problems associated with the use of austenitic alloys in the FTR core. A separate study on the effects of nonverticality of the non-fissile test specimens was concluded.

Materials Technology - Staff

An additional 215,000 Ci of Co-60 was added to the gamma irradiation facility. The total inventory of cobalt in the source is now 355,000 Ci.

Ceric sulfate dosimetry measurements were completed in the dry facility tube.

Measurements of the rate of oxidation of boronated graphite by 0.5% O₂ in a helium stream have been extended to include the oxidation of "black" boronated graphite containing 5% boron as B₄C.

Work is in progress to develop high-temperature fluence monitors useful at

* Below 900 °F uniform elongation was greater than 25%

temperatures in excess of 1200 °C that will not react with their environment.

Serpentine samples from Coalinga, California are being obtained from the Johns-Manville Company for comparison tests. Three serpentine irradiation capsules have been constructed. Tests on the water retention behavior of Quebec serpentine are continuing.

Calculations have been made to determine the rate of gamma activity buildup in the FFTF fuel cladding during irradiation.

Component Development - D. L. Hovorka

The first-phase results of the Hydraulic Core Mockup of core hydraulic performances, mechanical compatibility, and vibrational stability have been reported.

The program status and plans for the Hydraulic Core Mockup were presented to P. A. Halpine, of Headquarters RDT.

Model simulation for application to the Hydraulic Core Mockup program is being studied. Flow modeling techniques are being directed to cavitation phenomena and free convection.

Plans for a Fast Reactor Core Mockup Facility which provides non-nuclear sodium environment testing capabilities have been submitted as a PACE budget item.

In our attempt to explore the feasibility of reactor modeling PNL has contacted eight companies with modeling experience.

A control rod performance parametric study was made to evaluate the initial acceleration-spring length effect on poison-section insertion time.

Design of the major components to be used in the Sodium Pot Facility is 80% complete.

SECTION IIIDEVELOPMENT

F. W. WOODFIELD

PROCESS AND MATERIALS TECHNOLOGY - J. M. DAVIDSONFast Reactor Corrosion and Chemistry -

D. W. Shannon

Analysis - T. F. Demmitt, F. A. Scott

Oxygen. The PNL oxygen analysis procedure utilizes mercury amalgamation to separate the insoluble residue from the sodium sample. The residue is then hydrolyzed and analyzed by titrimetric and spectrophotometric methods.

The analytical procedure is now considered satisfactory for present PNL requirements, i.e., for oxygen concentration measurements of 5 ppm or greater. Standardized methods have been adopted for mercury purification for reuse and for cleaning the glassware used in the procedure. The oxygen "blank" value for the procedure is about 2 ppm, which is comparable to other sites. The following tabulation gives values indicative of the capability of this method.

<u>Sample</u>	<u>PNL Analysis</u>	<u>Vendor Analysis</u>
Vendor Standards (Group I)	5 ± 2	4 ± 5
Vendor Standards (Group II)	12 ± 3	11 ± 5

The PNL procedure was used to assess the oxygen concentration in a stainless steel capsule used as a sodium reservoir for addition of fuel-clad bonding material. Two samples were removed

from the capsule for analysis, and results of 6 ppm and 9 ppm were obtained.

The PNL procedure was used to assess the longitudinal distribution of oxygen in a 15 in. length of 1/2 in. diameter stainless steel tube of sodium. This tube was filled while in a vertical position in an argon-atmosphere drybox by pouring molten sodium into the tube. The tube cooled slowly and the sodium did not completely solidify until about 30 minutes after pouring was complete. The diagram below indicates the oxygen distribution measured.

Discarded	21 ppm	9 ppm	7 ppm	8 ppm	7 ppm	10 ppm	Discarded
1 1/2 in.	2 in.	2 in.	2 in.	2 in.	2 in.	2 in.	1 1/2 in.

This tube was not marked to indicate the top and so the reason for the high value (21 ppm) at the left is not definitely known. It is likely that diffusion of the oxide and/or reaction with the impurities in the argon atmosphere are responsible. However, excluding this one value, it is evident that the balance of the tube was homogeneous (8 ± 2 ppm) with respect to oxygen concentration. The unmarked end sections of the tube were discarded without being analyzed.

Carbon. The apparatus to analyze total carbon by the combustion-gas chromatograph method is almost complete. The chromatograph has been checked out and calibrated and first analysis attempts on sodium should begin in April.

Hydrogen. The PNL hydrogen analysis procedure will involve amalgamation of the sodium sample followed by refluxing of the amalgam to release hydrogen. Operation at different reflux temperatures will permit measurement of hydroxide and nonhydroxide hydrogen present.

The reaction system for hydrogen measurement has been designed and some of the equipment required has been obtained. The gas chromatograph that will be used for hydrogen measurement is currently being calibrated.

Sodium Purification Facility. Scope design of a 1 gal/min sodium purification loop for cold trapping or hot trapping has been completed, and the detail design is about 30% complete.

Instrumentation and control equipment requirements have been defined; required units have been located and tentatively allocated from the CANEL equipment. The hot trap strip heaters have been ordered.

Distillation-Vacuum Fusion

Apparatus. The feasibility of measuring oxygen in sodium by distilling the sodium, reacting the Na_2O with graphite to form carbon monoxide, and measuring the carbon monoxide with a gas chromatograph has been demonstrated. High oxygen results were observed on consecutive samples so the system was dismantled for examination. An accumulation of oxidized sodium was observed near the sample inlet valve. Apparently

when the samples were introduced some of this oxide was dislodged and caused the high observed concentrations. The system was cleaned and reassembled for further testing.

The gas chromatograph detection system was recalibrated for oxygen, nitrogen, hydrogen, methane, and carbon dioxide.

Drybox Gas Purification. A commercial gas purification system was purchased for use in conjunction with an analytical drybox obtained from the CANEL equipment. The gas purification system will remove oxygen, water, carbon dioxide, and heat from the glovebox atmosphere. Assembly and checkout of the complete drybox system will begin next month.

The PNL shop-built gas purification system was modified to utilize a heated zirconium bed for oxygen removal in place of the copper bed used previously. The zirconium bed operates at 600 °C. Following this change, the gas atmosphere in the drybox attached to this system improved considerably. The zirconium bed is functioning as a highly efficient moisture getter; it reduces water to below 0.1 ppm. It is now possible to maintain a freshly exposed sodium surface in this atmosphere for 6 to 8 hr before visible sodium tarnishing occurs. Previously, oxidation occurred as soon as 30 min.

Decontamination Studies - G. R. Bloom, T. F. Demitt, V. P. Kelly, T. Owen, D. W. Shannon

Radioisotope Transport Rate from FFTF Driver Core. Calculations on the generation of radioisotopes in 304 SS predicted these long lived isotopes (see "Fuel-Clad Activity Calculations"):

	$T_{1/2}$	Ci/cm^3 (saturated)
Ni ⁵⁸ (N,P)Co ⁵⁸	71 days	31
Fe ⁵⁴ (N,P)Mn ⁵⁴	313 days	10
Co ⁵⁹ (N, γ)Co ⁶⁰	5.3 years	2
Cr ⁵⁰ (N, γ)Cr ⁵¹	28 days	3
Ta ¹⁸¹ (N, γ)Ta ¹⁸²	115 days	1
		<hr/> 47 Ci/cm ³

With the assumption of an 800 liter FFTF driver core containing 91 sub-assemblies each with 169 pins (0.25 x 36 in.), and the use of the GE mass transfer data (10 to 15 ppm oxygen), the radioisotope transport rate was calculated for fuel clad temperatures of 900 to 1400 °F. These data (plotted in Figure III-1) indicate the radioisotope transport rate at a clad temperature of 1200 °F is about 5500 Ci/month, which is a large amount of activity. The matter of where this activity will deposit is subject to considerable speculation.

The basis for the calculation was: A fuel pin was assumed to have a 400 °F ΔT from bottom to top, and the corrosion was averaged over this gradient resulting in an average corrosion rate of 0.20 mils/yr for a pin with a 800 °F inlet and 1200 °F outlet. This converts to $7.7 \times 10^{-3} \text{ cm}^3/\text{month}$ loss x 15,350 pins = $118 \text{ cm}^3/\text{month}$. Thus, $118 \text{ cm}^3/\text{month} \times 47 \text{ Ci/cm}^3 = 5550 \text{ Ci/month}$. The activity will take more than a year to saturate.

Sodium Cleaning. Sodium removal procedures for FFTF piping systems and components are being evaluated. Ethyl alcohol is presently being evaluated as an agent for removing sodium from nondrainable piping system deadlegs. Parameters which will be considered include deadleg length-to-diameter ratio, water concentration in the alcohol, alcohol flow, alcohol volume to sodium volume ratio, and deadleg orientation.

The ethyl alcohol experimental tests are being conducted in glass tubing sections. Tests performed have demonstrated that 1/4 in. ID tubing sections with a length-to-diameter ratio of 10/1 can be completely cleaned. Tubing sections filled with sodium were positioned vertically with the open end up. Ethyl alcohol containing 5 wt% water was contacted with the sodium.

The time required for complete removal of sodium decreased with increased alcohol to sodium volume ratio. An alcohol to sodium volume ratio of 10/1 required 2 1/2 hr for complete sodium removal while an alcohol to sodium volume ratio of 125/1 required only 1/2 hr for complete sodium removal.

Continuation of these tests is expected to result in graphical tabular design data to define "cleanable" deadlegs and crevices.

FIGURE III-1. Calculated FFTF Driver Core Radioisotope Transport

Mechanical Metallurgy* - T. T. Claudson

Liquid-Metal Capsule Irradiations - K. R. Wheeler

The liquid-metal capsule GEH 22-3 containing 316 SS tensile specimens, was discharged January 30, 1967, after 99.9 full power days and a fast fluence of 1.38×10^{21} nvt. Arrival of the irradiated capsule at PNL is expected the week of March 6, 1967, for radio-metallurgical examination and specimen retrieval.

Liquid-metal capsule GEH 22-4, containing forty austenitic stainless steel specimens, has been charged in the G-6 position of ETR cycle 87 and is now accumulating fluence. Certain design modifications are incorporated into GEH 22-4 to permit higher capsule operating temperatures and to extend capsule life. During reactor startup, the temperature of GEH-22 capsules is controlled with 100% helium. After reaching full reactor power, the range of temperature control through gas mixing is established. The following temperature distributions were observed for capsule GEH 22-4 in achieving a design operating temperature of 1400 °F (760 °C).

GEH 22-4 Capsule Temperature (°F) Distribution						
Thermo-couple	Position	He	Gas Mixture			
			He-80 Ar-20	He-70 Ar-30	He-60 Ar-40	He-50 Ar-50
1	Bottom Na Column	825	945	990	1035	1080
2	Center Na Column	1065	1175	1220	1265	1310
3	Na Annulus	830	950	1000	1040	1080
4	Top Na Column	1175	1275	1315	1360	1400

The capsule, having met design expectations, is being operated at the 1400 °F (760 °C), 50-50 mixture, control point.

An inert (helium) atmosphere out-of-reactor capsule has been assembled and is ready for its initial thermal cycle. This capsule contains annealed AISI 304 and 316 SS and will duplicate the thermal history of capsule GEH 22-3. Operating conditions will be a temperature of approximately 1100 °F (595 °C) through four simulated ETR cycles, totaling 99.9 days.

Various heater geometries and capsule dimensions are being explored in an effort to produce liquid-metal out-of-reactor capsules at considerably less cost than that required for their in-reactor counterparts. The envisioned out-of-reactor capsules would simulate in-reactor thermal gradient and flow characteristics.

Stainless Steels - J. J. Holmes, R. E. Robbins, and E. R. Gilbert

The purpose of this phase of the program is to determine the combined effects of irradiation and environment on the mechanical properties of stainless steels. Radiation-induced property changes will be determined from irradiations and tests conducted at various temperatures on several alloys. Particular emphasis will be placed on determining the existence of metallurgical instabilities and the mechanisms by which they are enhanced in a nuclear environment.

Optical metallography is being performed on AISI 304 SS tensile specimens irradiated at 650 °C in NaK environment to about 3.5×10^{20} fast fluence and subsequently tested at various temperatures. Electron microprobe analysis, as well as transmission and

* Although related to the FFTF, this work is not funded under the FFTF program.

replica electron microscopy, is also in progress. Several encapsulated stainless steel tensile specimens have been charged in the ETR G-7 hot water loop. The purpose of these irradiations is to explore the possibility of obtaining specimen irradiation temperatures between 400 and 600 °C in the loop.

Structural Materials and Fuel Cladding Studies - J. J. Holmes and R. E. Robbins

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

Tensile tests on annealed AISI 304 SS irradiated to about 1.7×10^{22} total fluence at about 1000 °F (540 °C) in the EBR-II have been completed. Tests conducted over the temperature range, room temperature to 1500 °F (815 °C), provided the following significant information:

1. At all test temperatures below 900 °F (480 °C), the uniform elongation was greater than 25% and total elongation was never less than about 27%. Between 900 °F (480 °C) and 1100 °F (595 °C), the uniform and total elongation decreased to 6.5 and 11.5%, respectively. Above 1100 °F (595 °C), uniform ductility decreased continuously to a minimum observed value

of about 0.5% at 1500 °F (815 °C). The total elongation, however, passed through a minimum of 5% at 1300 °F (705 °C), then increased to 9.2% at 1500 °F (815 °C).

2. A slight increase in ultimate strength due to irradiation was observed at temperatures above 900 °F (480 °C). Below this temperature, ultimate strength was not affected. A marked increase in yield strength due to irradiation was found at all test temperatures, with the yield strength approaching the ultimate strength with increasing temperature. Above 1300 °F (705 °C), the ultimate and yield strength are practically identical.

The high temperature tensile data from the EBR-II irradiations closely approximate high exposure (6×10^{21} nvt) data from ETR, 290 °C irradiations, but exhibit somewhat higher strength properties above 1100 °F (595 °C). In the ETR studies, however, the ultimate and yield strengths are nearly identical for all test temperatures. Ductility losses in both EBR-II and ETR studies are similar above 1100 °F (595 °C) with somewhat greater loss in the ETR experiments

Optical metallography of the tested EBR-II specimens is in progress. Below 900 °F (480 °C) fracture is transgranular, while at 1100 °F (595 °C) and above intergranular fracture prevails. Electron microscopy of the as-irradiated specimens shows a complex defect state in which the dislocation loop density is so high that the loops have interacted to form a network. The thermal stability of the network is thought to account for persistence of irradiation hardening at the higher temperatures.

Twelve MK-A-type, space-available pins have been charged into two sub-assemblies and inserted into the core of the EBR-II. Subassembly X-019 contains eight (8) of these pins and is scheduled for a fast fluence of 6.0×10^{22} nvt. Subassembly X-020 contains four (4) pins and is scheduled for a fast fluence of 6.0×10^{22} nvt. These pins contain 124 tensile and creep test specimens of AISI 304 and 316 SS, evenly divided, of which half of each alloy is exposed to bulk sodium.

Development of the biaxial sodium environment creep test apparatus for testing irradiated specimens and controls is continuing. A prototype of the stressing system, sodium filling, and sodium recovery system has been completed. With the use of alcohol in place of sodium, this system has been successfully operated at room temperature.

Nickel-Base Alloys - I. S. Levy

The purpose of this program is to determine the effects of modified microstructures on the irradiation stability of nickel-base alloys. Microstructural modifications are made by preirradiation thermal or thermo-mechanical treatments and are evaluated by tensile tests, stress-rupture tests, and microstructural examinations.

Stress-rupture tests at 1350 °F (735 °C) and at the 1000 hour rupture stress of 9800 psi were completed on eight pre-treated Inconel 600 specimens which were irradiated at 540 °F (280 °C) to an approximate fluence of 1×10^{20} nvt (>1 MeV).

An experimental treatment increased postirradiation rupture life 65% over that of the commercial treatment. This

treatment also gave the best postirradiation 1350 °F (735 °C) tensile ductility; however, irradiation still caused a marked reduction in rupture life (approximately 5-fold) even for the best treatment. Stress-rupture tests on the unirradiated control specimens have been initiated.

X-ray diffraction phase analysis of Inconel 600 and Inconel X-750 tensile specimens has also been completed. The analysis of Inconel 600 shows that more M_7C_3 is formed as the aging temperature of the experimental treatments is increased. The higher postirradiation ductility imparted by these experimental treatments was hypothesized to be due to the removal of matrix chromium from the vicinity of the grain boundaries to form M_7C_3 in or near the boundaries. This mechanism would provide a more plastic zone adjacent to the grain boundaries, improving resistance to grain boundary crack propagation. The phase analysis data confirms that such a mechanism occurs. The better treatments from a ductility standpoint corresponded to greater M_7C_3 and, therefore, less chromium adjacent to the boundaries.

The phase analysis of Inconel X-750 showed, surprisingly, no M_7C_3 or $M_{23}C_6$ phases. The carbide phases were (Cb, Ti)C only, which can at present be explained by assuming little carbon being available after the formation of this carbide to allow formation of other carbide phases. A chemical analysis is in progress to elucidate this point. A microprobe analysis of grain boundary carbides is also being undertaken.

Microprobe analysis of a Nimonic 80-A sample shows that the microprobe

can detect chromium concentration gradients in the vicinity of grain boundaries. There is a denudation of chromium to the extent of about 2.5 wt% near grain boundaries (compared to the general matrix concentration) as the chromium combines with carbon to form grain boundary carbides. This reduced concentration value agrees with similar data in the literature. Microprobe analysis will now be used to study the effect of experimental treatments on grain boundary concentration gradients in the alloys being studied in this investigation.

In-Reactor Measurements of Mechanical Properties - *E. R. Gilbert*

The purpose of the in-reactor measurements program is to determine the effects of irradiation on the mechanical properties of reactor structural materials. The effort is involved in measuring creep in AISI 304 SS and nickel-base alloys during neutron irradiation.

A strain versus time curve was obtained for an in-reactor creep test conducted on Hastelloy X-280 at 1000 °C temperature and 3500 psi stress. The data show very little first-stage creep. The total elongation was greater than 11% as measured by the in-reactor extensometer system. This value will be verified by postirradiation specimen examination. Rupture occurred after about 75 hr of actual test time.

In-reactor creep-rupture test results on AISI 304 SS have indicated a much greater effect of irradiation on rupture life at 840 °C than at lower temperatures. Rupture indication in an in-reactor test was received

after 6 hr under 3.52 kg-mm^{-2} (5000 psi) stress. In contrast, an unirradiated control specimen tested under the same conditions of temperature and stress did not rupture until 926 hr. Further metallographic studies will be pursued in an attempt to determine the cause of this difference between in-reactor and unirradiated-control creep tests.

Fast Neutron Mechanisms - *R. P. Allen*

Construction and preliminary testing of a heat transfer apparatus that will be used to study the effect of temperature gradients and thermal diffusion on interstitial solute redistribution and bubble motion in cladding alloys has been completed. With a few minor modifications, it will be possible to readily achieve the temperature gradients and mean specimen temperatures required for this work.

Substantial progress has been made in perfecting glow discharge techniques for low-temperature hydrogen permeation, effusion, and defect interaction studies of irradiated reactor alloys. Hydrogen permeation rates that would require temperatures in excess of 400 °C with conventional gas phase hydrogen charging methods have been readily obtained at room temperature for steel test specimens using these discharge techniques. In addition, a hollow cathode electrode arrangement that significantly reduces specimen heating by the glow discharge has been developed for non-steady-state permeation experiments where maintenance of isothermal conditions is essential and has been difficult to achieve using other hydrogen charging methods.

In some initial room temperature effusion studies using 0.005 in. steel

specimens and a glow discharge technique as the hydrogen source, it has been determined that hydrogen desorption into a vacuum follows a second order rate law, which suggests that surface desorption and not matrix diffusion is the rate controlling effusion mechanism for these conditions.

Some interesting observations have also been made which indicate that a glow discharge in an inert gas can significantly enhance the normal low-temperature surface desorption rate of hydrogen; this effect will be explored further as it could provide a needed solution to the problem of rate-obscuring, phase-boundary processes in low temperature hydrogen desorption studies of irradiated alloys.

Intercalibration Test of Counting Laboratories - *R. E. Dahl*

The monitor materials for the intercalibration test have been irradiated in the MTR-VH-2 rabbit facility. The materials have been returned to PNL and are being counted prior to distribution to participating laboratories. The distribution has been scheduled for early February.

A two-dimensional transport-theory calculation of the MTR core and rabbit facilities has been initiated. Results of this calculation will provide spectral-averaged cross sections for the monitor materials. These average cross-sections will allow the participating laboratories to determine the fluence from the activated monitors.

Reactor Metals - S. M. Gill

FFTF Materials Property Handbook -

S. M. Gill, J. W. Helm

The initial literature search on the properties of nonmetallic materials for inclusion in the FFTF Materials Property Handbook has produced some 100 documents and articles. These have been abstracted and are being cataloged and evaluated.

The FFTF Materials Property Handbook will provide a unified source of information on materials properties required in the design of the FFTF. The initial intent was to adequately cover metallic structural and cladding materials; now it is intended to include nonmetallic materials.

Table III-1 outlines the materials initially of interest for the FFTF Materials Property Handbook. Table III-2 details the property categories. The prime purpose of the handbook is explicit coverage of the variation of these properties with any or all of the following parameters:

Effects of Irradiation:

Especially, but not limited to, fast flux conditions.

Effects of Temperature:

Ambient temperature up to 1600 °F (870 °C) for metals and 4000 °F (2200 °C) for nonmetallic materials.

Effects of Media:

Gas, vacuum, water, molten sodium, and containment material.

Effects of Prior Thermal Treatments or Fabrication Methods.

An initial literature search on stainless steels has produced over 150 documents on irradiation effects on properties. These have been abstracted

and cataloged and are waiting evaluation.

The American Society for Metals sent information about their Mark II Information Retrieval System. This is a computer based system designed to

TABLE III-1. Materials

Austenitic Stainless Steels

Prime Interest

304 (or AISI 304)	FV548
316 (or AISI 316)	321 (or AISI 321)
348, 347 (or AISI 348, 347)	

Others

25 Cr-20 Ni	19-9 DL
20 Cr-25 Ni	316L
20 Cr-25 Ni-1 Cb	304L or 304 ELC

Any other 300 series or "18-8" series stainless steel

Nickel-Base Alloys

Inconel 600, 625, 702, 718, X750

Inconel 800

Hastelloy X-280

Any alloy with Inconel, Incoloy or Hastelloy designation

Shield or Shielding Materials

Graphite

Carbon

Serpentine (3MgO·2SiO₂·2H₂O)

Boron carbide (B₄C)

Boronated or borated graphite or carbon

Calcium hydroxide (CaOH)

Iron oxide

Ferric oxide (Fe₂O₃, Fe₃O₄)

Ferrous oxide (FeO)

Hydrides

Zirconium (ZrH₂)

Barium (BaH₂)

Calcium (CaH₂)

Beryllium (Be)

Beryllium oxide (BeO)

Control Rod Materials (control poison)

Boron carbide (B₄C)

Tantalum

Plastics

Plastics

Elastomers

Rubber

Polymers

Adhesives

Sealants

Coatings

Insulating Materials

Alumina (Al₂O₃)

Magnesium oxide (MgO)

TABLE III-2. Properties

Mechanical Properties

Tensile Properties

Ultimate strength
Yield strength
Ductility (elongation, reduction in area)
Strain rate

Elastic Properties

Modulus of elasticity (Young's, shear, bulk)
Poisson's ratio
Viscoelasticity

Compressive Properties

Creep and Stress Rupture
Impact
Fatigue
Hardness
Shear or Torsional Strength

Physical Properties

Thermal Properties

Thermal expansion
Thermal conductivity
Specific heat

Electrical Properties

Electrical resistivity
Insulation characteristics

Magnetic Properties

Emissivity
Density
Dimensional Behavior (dimensional changes)

Chemical Properties

Reactions with Gases (air, oxygen, inert gases)

Reaction with Liquid Metals

Sodium, NaK

Mass transport effects

Aqueous Corrosion (water, steam, acids, alkalis)

Organic Compounds Reactions

Corrosion Film Properties

Gas Evolution

Gas

Water

Chemical Analyses

store and retrieve materials and metallurgical information from all publications. Potential use of the system in preparation of the handbook is being studied.

Metallurgical Assistance - S. M. Gill,
and J. E. Irvin

Each of the seven candidate FFTF reactor concepts was evaluated with respect to the metallurgical development requirements. It was concluded that the bulk of the metallurgical development program was concept independent and that the only differences were minor ones in fabrication development.

An additional review was conducted to assess the metallurgical problems associated with the use of austenitic alloys in the FTR core. AISI 304 and 316 SS have been proposed as reference and backup materials, respectively, for all of the components in contact with the primary coolant. It was found that the use of these and other materials for core components (fuel cladding and closed loop tubes) under proposed FFTF design conditions represented an extrapolation from current technology. It was further concluded that:

- There is little data now available which may be used directly by the designer for sodium cooled fast breeder reactor requirements.
- Extrapolations of data from thermal reactor materials technology will be required.
- A moderate amount of data from fast reactor irradiation studies will not be forthcoming until FY-1968. It will be FY-1969 before longer exposures to 1×10^{23} will be available from United States investigators.
- It may be expected that a relatively large amount of information on the tensile properties of commercial austenitic stainless steel will be forthcoming, at test temperatures

to 1200 °F. Smaller amounts will cover the 1200 to 1400 °F range.

- There is a noticeable lack of data being generated for materials to back up the austenitic stainless steels. More effort should be expended on nickel-base, vanadium, and refractory metal alloys. Studies related to the chemical and thermal/mechanical modifications of the austenitic stainless alloys should be extended.
- Creep and creep-rupture properties of commercial alloys must be determined as well as thermal stability of metallurgical structure. This work should cover neutron exposures somewhat above 1×10^{23} nvt and temperatures to 1600 °F (mostly to 1200 °F).
- Based on present programs, it can be expected that there will be a lack of low cycle and high cycle fatigue data. In addition, fracture studies related to sodium environ-

ments will not be adequately covered.

- Any design concept for the FFTF will expose in-core components to a fast flux exposure greater than that previously experienced or evaluated, within a few months time. This requires a core design that will permit replacing all components subject to exposures greater than those established as safe. As long as this criterion is followed, all alternate concepts are essentially independent as far as metallurgical considerations are concerned.

In a separate study on the effect on testing results of nonverticality of nonfissile test specimens, it was concluded that there would be no effect due to an overall angularity with vertical. It was determined, however, that problems of internal alignment and axiality are quite important, but these are problems which affect all tests whether vertical or not.

Materials Technology - Staff

Gamma Irradiation Facility -

D. E. Baker

An additional 215,000 Ci of Co-60 was added to the gamma irradiation facility. The total inventory of cobalt in the source is now 355,000 Ci. Twelve cobalt pieces were transferred into the 13.5 ft deep water-filled pool facility from two 14 ton casks. A capsule with melt wires was shipped in one of the casks which contained about 100,000 Ci of Co-60. The indicated temperature in the central cavity of the cask near the cobalt elements was 300 ± 15 °C after 3 hr. The cobalt was encapsulated in stainless steel cans in the form of hollow cylinders 1-7/16 in. OD x 1 in. ID x 16 in. long.

Ceric sulfate dosimetry measurements were completed in a 2 in. diam dry facility tube which extended from the source to 3 ft above the surface of the water. With the cobalt positioned around four 2 in. diam facility tubes, the maximum dose rate was 2.34×10^7 R/hr. The variation in dose rate over an 8 in. length was less than 5%. Two of the four facility tubes are equipped with heaters and gas purge lines to perform irradiations at temperatures up to 500 °C. The other two facility tubes are for irradiations near room temperature.

Oxidation of Boronated Graphite -

R. E. Woodley

Measurements of the rate of oxidation of boronated graphite by 0.5% in a helium stream have been extended to include the oxidation of "black" boronated graphite containing nominally

5% boron as B_4C . The difference between "black" boronated graphite and the "grey" material previously employed is that there is no microscopic evidence that melting of the B_4C occurred during manufacture of the "black" material, whereas melting did occur in the "grey" material. "Black" boronated graphite has a specific surface area that is a factor of 3 greater than that of the "grey" material.

In general, the initial oxidation rate of the "black" material is faster than that of the "grey" material because of its higher surface area. However, the actual ratio of the rates varies with temperature, because the two materials exhibit different activation energies. As previously reported, the activation energy for the "grey" material is about 32 kcal/mole at temperatures between 580 and 780 °C. At temperatures up to about 680 °C, the activation energy observed for the "black" material is about 38 kcal/mole, the difference being attributed to the greater ease with which the oxidant can penetrate the "black" boronated graphite.

Three separate series of rate measurements at a temperature of about 730 °C have yielded rates which are greater when calculated from weight measurements than when calculated from the gas-flow rate and the concentration of gaseous products. Because B_2O_3 is formed in the boronated graphite sample, the reverse should be true. We are trying to resolve this discrepancy.

An interesting feature of the oxidation rate at 730 °C is the presence of a small initial hump in the rate

versus time curves, indicating the occurrence of a fast initial reaction which subsequently decreases and is then overshadowed by the normal rate. This fast initial rate may result from the presence of machining dust on the samples, but it has never been detected previously.

Serpentine Irradiations -

J. H. Cox, Jr.

Serpentine samples from Coalinga, California are being obtained from the Johns-Manville Company for comparison with the Quebec serpentine being studied. Only one grade is available and the size distribution will be measured when the material arrives.

Construction has been initiated on the three capsules for irradiation of serpentine. The capsule design is complete and shop drawings are being made. The capsule consists of an inner tube containing the five specimen cans of serpentine and an outer tube concentric with the inner tube and separated from it by stainless-steel heat transfer fins. The cans are vented to the inner tube of the capsule which in turn is vented to the reactor stack. Each specimen can contains a sheathed thermocouple located in the center of the can. One of the five cans also has a thermocouple located at the surface to measure the temperature differential across the can, permitting calculation of heat generation in the serpentine. Resistance heaters on the inner tube will maintain the preselected sample temperatures. The heat flow will be adjusted by varying the size and number of heat transfer fins to pro-

vide specimen temperatures of 400, 550, 700, 800, and 900 °F.

Water Retention and Outgassing of Serpentine Shield Material -

P. G. Pallmer

Tests on the water retention behavior of Quebec serpentine are continuing. Successive out-gassings of the fine grade at 300 °C show a logarithmic time dependence of water loss (Figure III-2). Extrapolation of this rate gives an estimated water loss of 4% after 1000 hr, 4 1/2% after 10,000 hr, and approximately 5 1/2% over the FTR design lifetime of 20 yr. Additional work is under way to refine these data as a function of time, temperature, pressure, and grade of serpentine.

Heating of Quebec serpentine samples to temperatures up to 500 °C results in a water loss of 3 to 4% of the weight of the sample, and additional heating to 900 °C results in a total water loss of about 13%. Since the theoretical amount of water of hydration for pure serpentine ($3\text{MgO} \cdot 2\text{SiO}_2 \cdot 2\text{H}_2\text{O}$) is 13%, it was concluded earlier that a portion of the water of hydration was lost even at temperatures below 500 °C. However, based on additional information, it is now believed that the water lost below 500 °C is not the water of hydration of the magnesium silicate.

As reported above, long-term heating of serpentine at 300 °C results in total water losses comparable to those for samples heated at 480 °C, but the rate of water loss is much slower. Additional tests have also been made to determine water loss as a function of temperature. Serpentine samples were suspended from a recording balance

FIGURE III-2. Outgassing Behavior of Serpentine

and heated in flowing argon. The temperature was increased at a 7 °C/min rate to 900 °C. The samples lost from 0.3 to 2.3 wt% below 500 °C, and an additional 10 to 12 wt% above 500 °C, mostly in the range from 650 to 850 °C. Small amounts of the fine material were heated in a differential scanning calorimeter at 20 °C/min to 500 °C. Only one endothermic peak was observed. This peak occurred at approximately 410 °C and corresponded to a weight loss of 3%. The heat adsorbed during this loss of 7.6 to 7.9 kcal/mole of H₂O is comparable to the heat required for vaporization of water. Tests were conducted in a pressurized system, and the water lost at temperatures below 500 °C is regained when the samples are exposed to steam pressure of 1200 psi at 400 °C. The chemical analysis has also been completed on samples of the Quebec serpentine and has provided further information on the water of

hydration. Analyses were made on the coarse material (Grade 34) and on the finely divided material (Grade 36). Table III-3 lists the results of the analyses.

As noted in the data, there are substantial quantities of FeO in the material. Serpentine samples commonly have varying quantities of iron in place of the magnesium. Also listed in the table are the number of moles of constituents per 100 g of sample. These data are used to estimate the SiO₂ and H₂O in excess of the amount required for the composition of serpentine, 3[(MgO)_x, (FeO)_{1-x}]·2SiO₂·2H₂O. If the FeO/MgO ratio for the fine material is used, 0.30 moles of serpentine with a formula (MgO_{0.94}FeO_{0.06})₃·2SiO₂·2H₂O are contained in 100 g of the mineral. The excess SiO₂ is 0.021 moles/100 g, and the excess H₂O is 0.110 moles/100 g. For the coarse material a formula of

TABLE III-3. Analyses of Quebec Serpentine

	Fine (Grade 36)		Coarse (Grade 34)	
	wt%	moles/100 g	wt%	moles/100 g
MgO	34.9	0.843	28.2	0.699
FeO	3.99	0.056	8.65	0.120
SiO ₂	37.3	0.621	37.2	0.619
H ₂ O	12.8	0.710	12.4	0.688
Aluminum	0.2			
Calcium	0.2			
Chromium	0.1			
Manganese	0.02			
Nickel	0.2			
Titanium	0.005			
Zirconium	-			

(MgO_{0.85}FeO_{0.15})₃·2SiO₂·2H₂O is derived yielding 0.27 moles of serpentine in 100 g. The excess SiO₂ and H₂O is therefore 0.073 and 0.142 moles/100 g, respectively. Thus, about 0.13 moles/100 g or about 2% of the water in the material is not combined as water of hydration of the serpentine.

The results of the water losses (with time and as a function of temperature), the calorimeter measurements, the re-adsorption of water under pressure, and the chemical analysis all indicate the water loss below 500 °C is not due to dehydration of the serpentine. Possible sources of this water released below 500 °C are inter-crystalline adsorbed water, water of hydration of impurity minerals (such as iron oxide, quartz, and mica), or water produced by reactions of serpentine with magnesium hydroxide or with quartz. Therefore, it appears the material under investigation contains only 9 to 10% water of hydration for the serpentine.

Fuel-Clad Activity Calculations -

A. DePino, Jr.

Calculations have determined the rate of gamma activity buildup in the FFTF fuel cladding during irradiation. Estimates of the induced activity and mass transfer data have been used to determine the activity transferred to the coolant.

The nuclear reactions that could occur in 304 SS to produce gamma radiation have been computed. A computer code, ACT-II, which has an extensive cross-section library, calculated the gamma activity produced by the (n,γ), (n,p), (n,α), and (n,2n) reactions of constituents and common impurities. Then the reactions which produce most of the activity, as determined by ACT-II were hand calculated using more refined energy-group structure and cross-section data than were available in the program. The neutron spectrum for the FFTF core used in these analyses was calculated with the DTR transport-theory code.*

* Spectral calculations were performed by FFTF Nuclear Design Analysis

The reactions analyzed and their approximate saturation activities are listed in Table III-4. Approximately 50 Ci/cm^3 are generated within a day after the start of an irradiation because the products of the Mn^{55} , Fe^{56} , and Cr^{52} reactions have short half-lives (all less than 3 hr). Activity from the Ni^{58} and Fe^{54} reactions gradually increases and requires more than a year to reach saturation.

The principal sources of error in this type of an analysis are cross-section data and the failure to recognize possible reactions. In order to

test the calculations, analyses are being made on activation rates in the EBR-II. Calculations are then compared with measured activities to determine overall uncertainty. So far, reasonable agreement has been obtained on very limited experimental data.

Component Development - D. L. Hovorka

Program and Facility Requirement
Study on the Hydraulic Core Mockup -
 K. G. Toyoda

The Hydraulic Core Mockup (HCM) provides capabilities for prestartup de-

TABLE III-4. Induced Gamma Activity in FFTF 304 SS Fuel Clad

Reaction	Saturation Activity, Ci/cm^3	Reaction	Saturation Activity, Ci/cm^3
$\text{Al}^{27}(\text{n},\gamma)\text{Al}^{28}$	<1	$\text{Co}^{59}(\text{n},\gamma)\text{Co}^{60}$	2
$\text{Al}^{27}(\text{n},\text{p})\text{Mg}^{27}$	<1	$\text{Co}^{59}(\text{n},\gamma)\text{Co}^{60\text{m}}-\text{Co}^{60}$	<1
$\text{Al}^{27}(\text{n},\alpha)\text{Na}^{24}$	<1	$\text{Co}^{59}(\text{n},\text{p})\text{Fe}^{59}$	<1
$\text{Al}^{27}(\text{n},2\text{n})\text{Al}^{26}$	<1	$\text{Co}^{59}(\text{n},\alpha)\text{Mn}^{56}$	<1
$\text{Si}^{29}(\text{n},\text{p})\text{Al}^{29}$	<1	$\text{Cu}^{63}(\text{n},\gamma)\text{Cu}^{64}$	<1
$\text{Si}^{30}(\text{n},\gamma)\text{Si}^{31}$	<1	$\text{Cu}^{65}(\text{n},\gamma)\text{Cu}^{66}$	<1
$\text{Si}^{28}(\text{n},\text{p})\text{Al}^{28}$	<1	$\text{Cu}^{63}(\text{n},\alpha)\text{Co}^{60}$	<1
$\text{Si}^{30}(\text{n},\alpha)\text{Mg}^{27}$	<1	$\text{Zn}^{67}(\text{n},\text{p})\text{Cu}^{67}$	<1
$\text{Cr}^{50}(\text{n},\gamma)\text{Cr}^{51}$	3	$\text{Zn}^{64}(\text{n},\text{p})\text{Cu}^{64}$	<1
$\text{Cr}^{52}(\text{n},\text{p})\text{V}^{52}$	18	$\text{Zn}^{64}(\text{n},\gamma)\text{Zn}^{65}$	<1
$\text{Ni}^{58}(\text{n},\text{np})\text{Co}^{57}$	<1	$\text{Zn}^{68}(\text{n},\gamma)\text{Zn}^{69\text{m}}-\text{Zn}^{69}$	<1
$\text{Ni}^{58}(\text{n},\text{p})\text{Co}^{58}$	31	$\text{Zn}^{70}(\text{n},\gamma)\text{Zn}^{71}$	<1
$\text{Ni}^{58}(\text{n},\text{p})\text{Co}^{58\text{m}}-\text{Co}^{58}$	<1	$\text{Fe}^{58}(\text{n},\gamma)\text{Fe}^{59}$	<1
$\text{Ni}^{60}(\text{n},\text{p})\text{Co}^{60}$	<1	$\text{Fe}^{54}(\text{n},\text{p})\text{Mn}^{54}$	10
$\text{Ni}^{62}(\text{n},\alpha)\text{Fe}^{59}$	<1	$\text{Fe}^{54}(\text{n},\alpha)\text{Cr}^{51}$	<1
$\text{Ni}^{64}(\text{n},\gamma)\text{Ni}^{65}$	<1	$\text{Fe}^{56}(\text{n},\text{p})\text{Mn}^{56}$	5
$\text{Mn}^{55}(\text{n},\gamma)\text{Mn}^{56}$	30	$\text{Mn}^{55}(\text{n},2\text{n})\text{Mn}^{54}$	<1
$\text{Mo}^{98}(\text{n},\gamma)\text{Mo}^{99}$	<1	$\text{Mo}^{97}(\text{n},\text{p})\text{Nb}^{97}$	<1
$\text{Mo}^{100}(\text{n},\gamma)\text{Mo}^{101}$	<1	$\text{p}^{31}(\text{n},\text{p})\text{Si}^{31}$	<1
$\text{Mo}^{92}(\text{n},\text{p})\text{Nb}^{92}$	<1	$\text{p}^{31}(\text{n},\alpha)\text{Al}^{28}$	<1
$\text{Mo}^{92}(\text{n},\alpha)\text{Zr}^{89}$	<1	$\text{Ta}^{181}(\text{n},\gamma)\text{Ta}^{182}$	1
$\text{Mo}^{95}(\text{n},\text{p})\text{Nb}^{95}$	<1	$\text{S}^{34}(\text{n},\alpha)\text{Si}^{31}$	<1

sign verification and for evaluation of the reactor vessel and core hydraulic and mechanical characteristics in water and hot gas environments.

The first-phase results of the Hydraulic Core Mockup study of core hydraulic performances, mechanical compatibility, and vibrational stability have been reported. The objectives of the study were:

- Establish in a qualitative manner the need for FFTF core mockup(s)
- Determine the optimum mockup(s) to achieve the required data
- Define specific experimental programs.

Four general facility approaches were considered for resolving the types of problems encountered in the FFTF reactor core-vessel system: 1) a combination of analytical-individual component tests, 2) a full-scale water mockup, 3) a scale model water mockup, and 4) a full-scale sodium mockup. The approach that meets FFTF needs best required two facilities: a full-scale, partial core water mockup, and a scale-model water mockup for early hydraulic studies. The former facility will contain a reactor vessel mockup simulating the FTR design and accommodating various core configurations, internal heaters to provide temperature gradients in the vessel and internal components, a water circulating system, an inert gas system to provide dry heated environment in the vessel, controls and instruments for steady state and transient operations. The internal components will be prototypes of actual reactor components. The full size vessel will be capable of containing part of the core components, including

tubes as needed for closed loop testing. This facility will provide mechanical compatibility testing capabilities for full size components and hydraulic testing in a suitable segment of the reactor vessel. The latter facility will provide early hydraulic data for design verification which cannot be provided by the former in timely fashion, and which reduces the experimental load on the former facility to minimize overlap with the FTR core-vessel design effort.

The required data can be obtained from a sodium mockup, but satisfactory data can be obtained from a water mockup in a more timely manner at less cost. The need for a sodium mockup for FTR startup and operational support will be evaluated further.

The first-phase study reported above was concerned with qualitative assessment of the significant problems. These qualitative means are sufficient to define the basic test requirements and hence the needs for a core mockup(s). However, quantitative assessments of specific tests are necessary to formulate the more detailed program requirements which considers the trade-offs of risks, schedules, and costs. These studies are now in progress.

HCM Program Status Presented at AEC Headquarters - K. G. Toyoda

The program status and plans for the HCM were presented to Mr. P. A. Halpine, RDT Headquarters, in a meeting on February 2, 1967. Some of the important points brought out are as follows:

- HCM program has identified four major areas of the FFTF vessel and

core which require mockup evaluation: hydraulic performance; vibrational stability; mechanical compatibility; and thermal expansion effects.

- Results of a site survey performed by PNL on the modeling experience of others are being tabulated.
- The nonconcept dependent aspects of the HCM has permitted detailed studies of the building and support equipment requirements. From these studies, site preparation, acquisition of excess support equipment (pumps, tanks, and electrical switchgear from the Hanford production reactors) and support equipment arrangement designs are ready to commence (preliminary project proposal and criteria were reviewed).

Discussions on HCM program and project scope resulted in the conclusion that the more detailed program requirement studies presently underway should continue to provide the complete technical justification. However, it is also concluded that much of the supporting project activities which are not concept dependent should proceed. The immediate start could save about 4 months from the overall HCM schedule, as well as permit a more orderly transition of personnel effort from the nonconcept dependent support activities to the future concept dependent vessel and core mockup activities.

Hydraulic Modeling - D. S. Trent

Study of flow modeling techniques is being directed to cavitation phenomena and free convection. This is part of the study to review the theory of model similitude for application to the HCM program.

Cavitation, or formation of vapor pockets, is a phenomena which occurs in a hydraulic system when the fluid pressure drops to the vapor pressure. The cavitation number N_c , involves introducing the fluid vapor pressure P_v , and is stated symbolically as

$$N_c = \frac{P_v}{\rho v^2}$$

where ρ is the liquid density and v is the local velocity. The flow in model and prototype must be dynamically similar in every respect, at least in the region of interest, to correctly simulate cavitation. This means that the trajectories of and forces on all homologous fluid elements must be similar in model and prototype.

In the FFTF hydraulic core model, fluid flow will be characterized by the pressure coefficient ($\Delta P / \rho v^2$) and the Reynolds Number (vL/ν). If the Reynolds Number ratio for prototype and model is unity and is combined with the cavitation number ratio, the following relationship between model and prototypic vapor pressure results:

$$P_{vm} = \frac{\rho_p v_m^2}{L_r^2 \rho_m v_p^2} P_{vp}$$

where L_r is the scale ratio, L_m/L_p , and the subscripts m and p refer to the model and prototype, respectively. Also, combining the pressure coefficient ratio and cavitation number ratio yields

$$\frac{P_{vm}}{P_{vp}} = \frac{\Delta P_m}{\Delta P_p}$$

Since the vapor pressure of water in the probable model temperature

range is grossly different from sodium in the 800 to 1200 °F range, exact cavitation modeling will not be possible.

Nevertheless, valuable qualitative cavitation information can be gained with dissimilar model and prototypic vapor pressures. Since prototypic vapor pressures will probably be much less than model vapor pressures, and if local velocities and system pressure are reasonably simulated, cavitation will occur in the model before in the prototype. Hence, if no vapor pockets form in the model, it may be safely assumed that none will form in the prototype for similar flow conditions. Pressure distortion may also be used to obtain qualitative data. If the pressure drop is correctly simulated in a geometrically similar model, the absolute pressure scale may be translated to induce cavitation and at least indicate where vapor pockets will first form. It may also be possible to obtain some valid qualitative data through pressure scale translation.

Although free convection occurs in most all flow systems where thermal induced density variations are present, it will be significant in the FFTF hydraulic vessel only at very low flow ratio or if a pumping failure occurs. A true free convection model cannot be achieved in the FFTF hydraulic core mockup unless heat generation in the fuel assemblies can be simulated. The model criteria has not been fully developed for this experiment as yet.

Fast Reactor Core Mockup Facility -
K. G. Toyoda

Plans for a Fast Reactor Core Mockup Facility (FRCMF) which provides non-

nuclear sodium environment testing capabilities have been submitted as a FY-1969 PACE budget item. The facility will consist of prototypical vessel and core components and a flowing 1200 °F sodium system (partially provided by Hallam excess equipment). Final proof testing of multiple fuel assemblies, and other core components will be demonstrated under prototypical FFTF environment (except nuclear). A prototypical closed loop, including tube, piping, heat exchangers, pump, etc., will be procured on expense funds for installation during FY-1971. The completed Fuel Handling System will be demonstrated under prototypical interface environmental conditions (except radiation). Mockup cells, including simulated radiation shielding walls, lead glass viewing windows, various lighting schemes and techniques, traveling cranes, manipulators, jigs, fixtures, etc., will be installed during FY-1970 and 1971.

Reactor Modeling - K. D. Hayden

Eight companies were contacted to determine specific modeling experience. Since some of the desired information was not immediately available from the site representatives interviewed or was not determined, and to substantiate our understanding of their experience, written concurrence and supplementary data are being requested.

In general, complete reactor models or portions thereof have been used in the design and construction of complex hydromechanical systems and equipment. Table III-5 lists the sites, reactors, model size, and experimental studies conducted for design support of various

reactors constructed by or for the sites surveyed.

Reactor Nuclear Control - *J. C. Noakes*

A control rod performance parametric study was performed to evaluate the initial acceleration-spring length effect on poison-section insertion time. It was determined that a short accelerating spring (5 in.) was almost as effective as a long spring (20 in) for the first few inches of rod insertion. A gravity drop safety rod will not meet the current anticipated system design requirements.

Flow and Pressure Sensors for the Sodium Pot Facility - *D. E. Rasmussen, T. J. Crocker*

Design of the major components (e.g., storage tank, glove boxes and sodium pot) to be used in the heated,

nonflowing Sodium Pot Facility is 80% complete.

A specimen carrier and twenty-six instrumentation sensor test specimens were prepared for sodium compatibility testing. The specimens are fabricated from 304 and 316 SS and have been submitted to the Sodium Studies Facility personnel for exposure in the 314 Building Sodium Facility.

Five commercial pressure transducers and associated transmitters are presently being procured. The transducers and transmitters will be tested to determine their applicability to high temperature sodium operation.

TABLE III-5. Reactor Modeling

	Reactor	Testing On	Scale	Fluid	What Studied
General Electric	SEFOR (Na)	Reactor Vessel	1/2	Water	Surface Wave, ΔP , Distribution Vibration
Argonne	EBR-II (Na)	Reactor Vessel	(Work Done by Franklin Institute)		
Oak Ridge	HFIR (H_2O)	Reactor Vessel	Full	Water	Flow Distribution
	MPRE (K)	Reactor System	Full 1/2 Sector	Water	System Operating Cycle
	MPRE (K)	Reactor System	Full	Potassium	System Operating Cycle
	HRT (D_2O)	Reactor Vessel	Full	Water	Residence Time, Mixing, Dist.
	MSRE (UF_4)	Reactor Vessel	Full	Water	Dist., ΔP , Mixing, Remote Maint.
	MSRE (UF_4)	Reactor Vessel	1/5	Water	Inlet Orifices, Anti-Swirl Vanish
Westinghouse APD	Yankee (H_2O)	Reactor Vessel	1/12	Water	Flow Dist., ΔP And Mixing
	Ind. Point-2 (H_2O)	Reactor Vessel	1/7	Water	Flow Dist., ΔP And Mixing
Babcock & Wilcox	Keowee Dam PWR (H_2O)	Reactor Vessel	1/6	Water	Mixing, Pressure Drop and Dist.
Atomics International	PIQUA (Organic)	Reactor Vessel	Full 1/4 Sector	Water	Flow Distribution
	SNAP-8DS (NaK)	Reactor Vessel	Full	Water	Flow Distribution
Battelle-Columbus	PWR (H_2O)	Lower Plenum	1/4	Air	Mixing
	PWR (H_2O)	Thermal Shield	1/4	Air	Flow Restrictors
	PWR (H_2O)	Core	1/4	Air	Flow Distribution
	PEGCPR (He)	Core	1/4	Air	Flow Patterns, Mixing and ΔP
	PWR (H_2O)	Core 2	0.4	Air	Flow Dist, Mixing and Pressure Drop
Franklin Institute	FERMI (Na)	Outlet Plenum	1/4	Water	Mixing
	EBR-II (Na)	Reactor Vessel	0.6	Water	Flow Distribution
	Westinghouse (H_2O)			CO_2	

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