**FAST FLUX TEST FACILITY
QUARTERLY PROGRESS REPORT
APRIL, MAY, JUNE 1968**

OCTOBER 1968

**AEC RESEARCH &
DEVELOPMENT REPORT**

J. Ryden, Jr. 30835 326 200 5 1970

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By

E. R. Astley
Project Manager

October 1968

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INTRODUCTION

This report was prepared by Pacific Northwest Laboratory (PNL) under Contract No. AT(45-1)-1830 for the Atomic Energy Commission, Division of Reactor Development and Technology. It is a report on progress in conceptual design and development, and in program administration and management for the Fast Flux Test Facility, during the period April, May, and June 1968. Prior progress is cited where appropriate for reader orientation.

This report is arranged in four chapters. Chapter I presents a brief description of FFTF overall program progress. Chapter II Systems describes systems management progress made by the FFTF Project Engineering Department during the reporting period. Chapter III (General Technology) reports progress made by the FFTF Reactor and Plant Technology Department; this chapter also includes results of work by all departments on reactor concepts. Chapter IV presents work by FFTF Development Department. Contributions from a number of off-site government organizations and industrial firms who are supporting PNL in the FFTF Program are also summarized in the various chapters.

To facilitate interpretation of progress for individual systems, a Cross Index is included following the Table of Contents. This indicates the arrangement of the work by Systems.

I. FFTF OVERALL PROGRAM PROGRESS

This chapter presents a summation of work, by all FFTF departments, directed toward systems Engineering Administration, Design and Development.

The chapter summarizes overall FFTF Program progress during the last quarter of FY-1968. Where appropriate, prior progress is reviewed to assist the reader in evaluating the significance of the current results.

A. ADMINISTRATION OF DESIGN CONTRACTORS

1. PNL Functions

PNL was requested by USAEC directive to manage the construction budget funded FFTF Program (AEC Budget Item 04-67-3-a). Design is to be performed by responsible organizations under contract to Battelle-Northwest, with some local on-site assistance.

2. Selection of Design Contractors

Through agreement with the AEC, it was determined that a General Plant Architect-Engineer (GPAE) and a Reactor Plant Designer would be selected with the intent of subsequently expanding the scope of the GPAE contract to include construction management.

After a formal process of issuing requests for proposals and evaluating bids, PNL provided information to the AEC on capabilities of candidate organizations. The AEC selected Bechtel Corporation as the GPAE (and construction manager), Westinghouse Corporation as the Reactor Designer, and Atomics International as the principal sub-contractor. A design contract has been signed with Bechtel Corporation for preliminary design and negotiations with Westinghouse were in progress at the end of the reporting period.

B. REACTOR PLANT CONCEPTUAL DESIGN

1. Selection of Reactor Concept

a. Interim Reference Concept History

Work on an "Interim Reference Concept"⁽¹⁾ for a fast neutron flux irradiation test facility (FFTF) was initiated late in FY-65 by PNL, under sponsorship of the USAEC, Division of Reactor Development and Technology. Recommendations on the Interim Reference Concept were submitted by PNL to the AEC in mid-FY-66, and formed the basis for an FY-67 AEC budget submission of \$87,500,000 for the proposed facility. The Interim Reference Concept for the FFTF was designed to provide a maximum hard neutron flux (approximately 70% above 0.1 MeV) of about 1.35×10^{16} n/cm²-sec.

The design was based on a "skewed tube pattern"⁽¹⁾ to provide maximum access for instrumentation and test specimen handling. In this skewed tube pattern, each tube was tilted about four degrees from vertical. PuO₂-SS cermet fuel was employed in a 600 liter sodium-cooled tubed-core.

b. Reference Concept History

After submission of the Interim Reference Concept, PNL continued conceptual design studies at the request of AEC-DRDT to establish a "reference concept." A major goal for the "reference concept" was a design providing "plumb vertical" closed loop tubes in the reactor, while at the same time retaining sufficient physical dispersion of coolant piping to permit convenient handling of test specimens and adequate instrumentation of test positions.

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1. Fast Flux Test Facility Quarterly Progress Report, January, February, March 1967, BNWL-501. Pacific Northwest Laboratory, Richland, Washington.

A $\text{PuO}_2\text{-UO}_2$ ceramic fuel was selected for development as the "reference concept" driver fuel. Total peak neutron flux requirements were concurrently reduced to 0.7×10^{16} n/cm²-sec.

Conceptual design studies, to meet the criteria outlined above, were performed by PNL, Atomics International, General Electric, Phillips Petroleum, and Westinghouse. A total of seven conceptual designs so generated were evaluated by PNL. These included three variations of the skewed design of the interim reference concept, three variations of a vertical-tube core and a hybrid design (split core) which provided vertical closed loops and slightly inclined driver tubes.

Through agreement with the AEC, it was determined that the hybrid "split core concept" appeared to be a suitable reference concept choice for the reactor, provided that required flux levels could be obtained and mechanical design problems could be solved. In view of the uncertainties associated with the reference concept, PNL sponsored conceptual design work for a vertical "backup core." This work, performed by General Electric Company, was initiated early in FY-68.

During FY-68, PNL conducted extensive studies on the reference concept. As a result of these studies, it was determined that the reference core did not offer reasonable potential for meeting the criteria for test position flux of 0.7×10^{16} n/cm²-sec.

In FY-68, PNL established two task forces, one to investigate methods for improving the reference core design, and one to evaluate the backup core and make any needed improvements. Results of the work of these task forces are summarized below and described in Chapter III, Section A. Further studies are continuing in FY-69.

c. Current Core Concept Studies

(1) Dispersed Core. A PNL "Improved Dispersed Core Task Force" analyzed, during the last quarter of FY-68, both the reference concept core and a number of other "dispersed" cores.

The task force selected an annular conical configuration containing three central vertical closed loops as the prime improved core configuration.

Areas requiring further investigation include further definition and sophistication of a radial restraint design, development of simple, assured fuel handling methods, and further study of the closed loop nozzle and piping access requirements.

(2) Vertical Core. A PNL task force carried out, during the last quarter of FY-68, design studies on a vertical core concept, using the backup core design as the basis of the work.

The resultant vertical core concept consists of an open lattice vertical core with open plug refueling. Six closed loops and 12 open loops are provided.

The core is rated at 400 MW_t with a total sodium coolant flow of 35,000 gal/min and a core delta-T of 300 °F. Peak total neutron is computed as $0.88 \times 10^{16} \text{ n/cm}^2\text{-sec}$ at start of a 90 day cycle.

The open and closed loops penetrate the refueling plug and are sealed with a cap containing the instrument connections.

Areas requiring further investigation include instrumentation arrangement, control rod drives, and closed loop piping.

2. Definition of Design Basis Accident

Studies defined the Design Basis Accident (DBA) for the FFTF as a nuclear excursion terminated by core disassembly with a maximum theoretical work energy release estimated to be 1500 MW-sec. For further details see Chapter III, Section D.

3. Reactor and Cooling System Conceptual Design

a. Reactor Vessel Conceptual Design

Babcock and Wilcox has recommended a number of design features for incorporation into the FFTF vessel conceptual design. Recommendations accepted by PNL are listed below:

- Piped bottom entry
- A high pressure inlet plenum
- A single wall vessel
- A surrounding guard vessel without penetrations
- Vessel-cover integrity that ensures that the cover remains intact under DBA conditions
- Six inlet and three outlet nozzles
- A design philosophy of ease and reliability of inspection during fabrication, erection, and usage
- Top support of vessel
- Vessel base material to be Type 304 SS

For further details see System 30, Chapter II.

b. Reactor Safety System Conceptual Design

The reactor safety system planned for the FFTF will use standard instrumentation to monitor the gross neutron flux and initiate safety action when predetermined limits in flux level, flux rate of change, or reactor period are exceeded. The safety circuitry will be independent of the control sensors and circuitry.

Neutron sensors, located externally to the vessel, will cover the entire operating power level range of the reactor.

The safety circuitry will comprise three, low-level safety channels using compensated ionization chambers, and three, high-level safety channels using compensated or uncompensated chambers. For further details, see System 90, Chapter II.

c. Inert Gas Selection

It was agreed with the AEC that inert gas systems conceptual design would not be required to make provisions for testing vented fuel in the driver core or open test positions. Following this agreement, a re-evaluation of gas selection was made. Use of helium was eliminated. Argon was selected for all sodium cover gas and fuel handling applications where contact with the reactor primary coolant is possible. For further details see System 80, Chapter II.

4. Fuel Handling and Examination Conceptual Design

a. Fuel Handling for Dispersed Core Concepts

Fuel handling conceptual design studies for dispersed core reactor concepts have been performed by Atomics International. Details are given in Chapter II under System 40.

b. Inert Gas Examination*

(1) Conceptual Design. A conceptual design for an inert gas cell examination facility was submitted to the AEC. This facility will permit final disassembly and detailed examination of irradiated fuel. A purified argon atmosphere, shielded confinement system (cell bank) and necessary operational support areas are provided.

(2) Design Evaluation.* The feasibility of decay heat removal from FFTF fuel assemblies by the forced circulation of an inert gas stream during examination operations was established by demonstration tests. The tests demonstrated the practicability of removing 20 kW of decay heat while limiting maximum fuel cladding surface temperatures to 600 °F.

c. Underwater Examination*

A decision was made to eliminate the Underwater Cell Examination Facility because of difficulty in ensuring that

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* For further details see System 70, Chapter II.

underwater handling of sensitized stainless steel FTR core components would not affect data collected during subsequent metallurgical examination.

C. STRUCTURE, SITE, AND UTILITIES CONCEPTUAL DESIGN

1. Containment

The Reactor Containment System conceptual design incorporates the following principal structural components: a steel vessel, a reactor cavity vault, a reactor hot cell housing the fuel handling machine, primary sodium equipment cells and facilities, closed loop cells, and inert gas equipment cells. Nitrogen gas was selected for the cell atmosphere for all cells containing radioactive piping and equipment.

- The reactor cavity vault and the reactor hot cell function as a first containment barrier in a DBA or sodium air reaction accident.
- The Reactor Containment Vessel functions as the final containment barrier in any accident, including the DBA, to limit release of radionuclides to meet the guidelines of AEC Site Criteria 10 CFR 100.

For further details see System 20, Chapter II.

2. Nuclear Proof Test Facility Conceptual Design

The Nuclear Proof Test Facility (NPTF) includes the Nuclear Proof Test Reactor (NPTR), the reactor cell, control room, fuel storage, vault, work areas, instrumentation, fuel handling equipment, and containment system. The facility is located in the FFTF Building adjacent to the nonirradiated fuel handling area. A separate containment system is used. The arrangement is shown in Drawing SK-3-14142 (System 30, Chapter II). The reactor cell is a cylindrical, shielded vault approximately 35 ft in diameter and 60 ft high having

access ports for transferring equipment, fuel, etc., and air lock doors for personnel access. For further details see System 30, Chapter II.

D. FFTF FUELS DEVELOPMENT*

The current objectives of the FFTF Fuel Development Program are to develop $\text{PuO}_2\text{-UO}_2$ driver fuel for the FFTF and to qualify fuel vendors for fabrication of FTR fuel cores.

1. Model Testing

a. Heated Fuel Pin Model

A major interim goal was achieved in the development of electrically heated fuel pin models for use in bundle geometry heat transfer studies. A single pin achieved a heat flux of 500,000 Btu/hr-ft² at a sodium temperature of 800 °F and maintained this condition for 6 hr. This pin was subsequently found capable of 650,000 Btu/hr-ft² heat flux in 1100 °F sodium before failure occurred because of electrical arcing between the sheath and the central core.

b. Flow Testing of Subassemblies

The laboratory tests to determine the effect of fuel pin spacing, wire wrap pitch, and pin diameter on the pressure drop-flow behavior of FFTF fuel subassemblies were completed. Preliminary results showed that the friction factors determined for 7-pin, 37-pin and 217-pin assemblies of similar geometry agreed at corresponding Reynolds Numbers, thus substantiating the validity of hydraulic tests with small partial assemblies.

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* For further details see Chapter IV, Section A.

2. Irradiation Testing

a. PNL-1 Subassembly

The first PNL fast reactor irradiation experiment, PNL-1, a 19-pin encapsulated subassembly, completed its goal burnup of 10,000 MWd/tonne in the EBR-II. The subassembly is now being examined.

b. Prototypical Length Fuel Columns

The first four of six capsules containing prototypical length fuel columns were irradiated to the goal burnup of 10,000 MWd/tonne in GETR. The capsules are now being examined.

3. LMFBR Fuel Supplier Prequalification Program

FFTF has the responsibility to develop and evaluate the capabilities of commercial plutonium fuel suppliers to fabricate FTR fuel cores. The first step in this development is the recently initiated LMFBR Fuel Supplier Prequalification Program in which potential FTR vendors will be evaluated as to their capabilities to handle plutonium bearing fuel material and to fabricate a limited number of finished fuel pins.

II. SYSTEMS

W. B. McDonald

HIGHLIGHTS OF SYSTEM 10. ELECTRICAL

G. H. Strong

The proposed FFTF site is an isolated arid region with no natural low resistance soil so that a conventional ground rod grid system may not provide the required low resistance. Studies indicate that a two or more deep well configuration may be necessary to obtain the proper ground. A computer program is being considered which prepares wire run tests or tabulations, selects wire routes in the FFTF building on the basis of desired end points and allowable paths, optimizes the tray and wire way system and provides an isometric diagram of the building and wire way system. Results of a study of the nuclear irradiation effects on television camera lens show that zoom lens drives become inoperative after gamma irradiation to 1×10^8 R. All lenses of conventional glass, both fixed focus and zoom types, became opaque at 1×10^6 R gamma exposure, and a 1-in. nonbrowning lens retained 89% of its transmissivity after 1×10^9 R gamma exposure.

10. ELECTRICAL

G. H. Strong

The Electrical Systems consist of Primary Electrical Power, Building Electrical Power, Communications, Lighting and Plant Protection.

1. Primary Electrical Power System

D. T. Thonn and M. F. Wiitala

The Primary Electrical Power System provides normal power to the FFTF. The system includes the connections to the local area utility system, transmission lines, and step-down transformers located in the FFTF substation. The reference concept provides two separate 115 kV lines from the

Bonneville Power Administration's 115 kV network. Each line terminates in a transformer rated at 12/16/20 MVA.

Outage Times of the Primary Power System for Line Faults
with Normal Breaker Operation

D. T. Thonn and T. M. Shay

A study is being made to determine the effects of momentary phase-to-phase and line-to-ground faults on the primary electrical system and on the primary and secondary sodium pumping systems. Indications are that the phase-to-phase faults will result in approximately a 30 cycle (1/2 sec) outage if the fault has cleared upon reclosure of the circuit breakers. A time delay of 30 cycles in the low voltage protection circuit of the pump starters would allow the pumps to continue running without effect on reactor cooling.

The outages caused by line-to-ground faults are not as easily defined. The difference in impedance for each fault causes an unpredictable amount of time to occur before the relays actually open the line to clear the fault; however, the operation after the breakers open is the same as for phase-to-phase faults.

During the time before the breakers open (1/2 sec), indications are that the faulted line will deliver enough power to the pump motors to maintain about 85% of their normal full load torque. No pumping problems are anticipated at this time.

2. Building Electrical Power System

M. F. Wiitala

The Building Electrical Power System receives power at 13.8 kV from the Primary Electrical Power System and steps down and distributes the power at the utilization voltage levels of 2.4 kV and 480 V. Special power supplies such as

emergency power, instrument power, zero time outage power and 125 V dc and 24 V dc are provided. Included in the concept are electrical control, electrical heating of sodium piping and vessels, conduit, and raceways and wiring.

a. Instrument Power Supplies

M. F. Wiitala and J. P. Thomas

Special power supplies were inspected at a vendor's plant in Los Angeles. These special power supplies are referred to as "uninterruptible" supplies because they are solid-state inverters operated in synchronism with the normal ac supply and will assume the load, without interruptions, upon loss of normal power. These supplies have been furnished to the U.S. Navy for their communications systems. This kind of power supply is a candidate for use with the FFTF instrumentation and control systems.

b. Electrical Grounding Study for the FFTF

C. S. Bucholz

The proposed FFTF site is in an isolated arid region of fine surface soil, and there are no natural surface areas of lower resistivity soil or water within a reasonable distance. Considering the intrinsic hazards and exposure of the site to lightning, a grounding system having 1 to 2 ohms to earth should be provided. A grounding cable, buried about 18 in. deep, around the periphery of the reactor buildings will be necessary for personnel protection, and an interconnected grid of cables is convenient for grounding equipment both inside and outside the buildings. However, studies show that the combined resistance of the conventional grid with ground rods is approximately 4 ohms. Therefore, to provide the desired low resistance ground, it was recommended that two or more 100 to 125 ft deep grounding wells be used. The resistance of the grid (approximately 7 ohms) combined with the

resistance of the two deep wells (approximately 1.16 ohms) gives a total resistance of slightly less than 1 ohm. Although mutual effects will tend to increase this figure, other interconnected grids and underground piping will decrease it.

c. Development

(1) Electrical Control (J. H. Kleinpeter). Reed switches have been tested to aid in the procurement of switches to be used in the reactor to sense control rod position.

Testing done during the report period indicates that reed switches operate satisfactorily up to 750 °F, but operation is erratic at 800 °F. Total failure usually occurs after the switches have been at 800 °F for about 1 hr. Normal function resumes as the temperature is lowered.

(2) Computerized Wire Run List (J. P. Thomas, C. W. Lindenmeier, B. H. Duane, and H. S. Zwibel). Studies have been made on the development of a computer program for automatic wire routing which can be used in the electrical design, construction and maintenance of the FFTF. A preliminary state-of-the-art survey indicated that computer programs for making wire run lists are used by several design groups and Architect Engineer (A-E) firms. However, it was determined that some of the programs are limited in scope and are too specialized to use in making wire run lists for a complex wiring job such as that needed for FFTF. None of the programs surveyed had the feature of automatic wire routing which is most important in reducing design time in the last stage of a project.

Any program developed for FFTF must be suitable and available for use after construction is completed for preparing "as-built" drawings. It must also easily accommodate changes and additions made to the wiring to match changes in reactor operation or the addition of equipment for new experiments

during the life of the reactor. This latter feature may easily be as important as the use of the program in the initial design and construction. Proprietary programs that cannot be made available for use throughout the reactor lifetime would not be suitable.

A PNL request to the AEC for a survey of Government-sponsored work on computer programs for making wire run lists disclosed that the TVA has done some work on this problem for large central power stations. However, currently available funds have permitted only a modest effort at Knoxville, Tennessee. It is planned to maintain contact with the TVA Engineering Office to keep informed as their work progresses.

The development of the computer program for automatic wire routing has been tentatively broken into the following three major phases with the goal of completing Phases I and II by mid-1970 with Phase III to be completed as time and funds permit.

(a) Phase I. This phase is the "bookkeeping" part of the program and takes care of the following information normally found on wire run lists; cable numbers, numbers of conductors in cables, wire lengths, routing information, insulation, type of wire, and other more general information.

(b) Phase II. This phase consists of the automated computer selection of wire routes in the FFTF building given the desired end points and the allowable wire paths. Path selection at a junction, where up to five exit possibilities exist, would include selecting the shortest (or least costly) path given the straight line distance between the wire end points. This can be done by determining which path has the smallest angle of departure from the straight line direction. A "look-ahead" feature can possibly be included so that at a

wiring junction, the computer will look ahead to the next junction for all junction exits to determine which exit path is the optimum.

Constraints will be built into the wire routes allowable for certain kinds of wire runs (control, instrument, safety circuits, etc.).

(c) Phase III. This phase would provide optimization of the tray and wire way system in the building and also provide isometric diagrams of the wire way systems.

Recently, Computer Sciences Corporation at Richland worked on this program development. Their major effort was directed toward investigating the most efficient method of solving the shortest route through a network. Approximately 20 different methods have been studied falling into the following general categories:

- 1) Dynamic Programing Using Matrix Methods. Although these methods are very fast, they require a great deal of computer memory and are therefore unfeasible for a project the size of the FFTF.
- 2) Fanning-Out Process. Most of the methods studied fall into this category. This method is somewhat slower than the matrix method, but is more efficient in its use of computer memory, and can be used on the FFTF project.
- 3) Decomposition Techniques. Several schemes have been developed to decompose the network into small parts and solve each part using a matrix method and then combining the parts into a total solution. This technique seems to be the best and by far the fastest for our problem.
- 4) Analog Methods. Two analog methods were also studied briefly. These techniques are very efficient but not applicable to the digital problem at hand.

3. Communications System

R. C. Hoffman

The Communications System provides audio and visual communications, paging capability, and necessary alarms within the FFTF. Included are audio and visual communications between the control room and all important work locations, remote observation of important objects or areas and audio communication between restricted access areas during special maintenance and inspection. Necessary alarms include critical radiation alarm, alert attack and warning alarm, and annunciation of selected undesirable or abnormal conditions.

A meeting was held with PNL, AEC-RL and ITT/PPSU to discuss the extension of the AEC general area dial telephone system to the proposed FFTF site four miles northwest of the 300 area. ITT/PPSU is the contractor responsible for operating the AEC telephone system. It was decided to include the installation of an underground 50 pair cable from FFTF to the 300 area exchange and a 125 line PABX in FFTF. This arrangement would permit approximately 20 simultaneous conversations within the FFTF complex.

a. Television Camera Lens Testing

R. C. Hoffman, M. F. Wiitala, and J. H. Kleinpeter

A state-of-the-art survey indicated that data on the effects of nuclear radiation on television camera lens are not available. A study to obtain the data to be used in the preparation of television camera purchase specifications is being conducted.

Lenses of the following types have been irradiated in Hanford gamma radiation facilities:

- Zoom type of nonbrowning glass
- Fixed focal length of nonbrowning glass

- Fixed focal length of conventional glass
- Lens blanks of both conventional and nonbrowning glass.

Preliminary information obtained from the study shows the following effects:

- Zoom lens drive mechanisms became inoperative after irradiation to 1×10^8 R when the nylon gears became embrittled and broken.
- In all movable lens systems, the lubricants became stiff after exposure to 1×10^8 R making the lens adjustments inoperable.
- Nonbrowning lens glass was still usable after exposures to 1×10^8 R although light transmission was reduced by approximately 50%. The lens was placed in a TV camera before and after irradiation and actual picture quality was compared on a monitor. The lens opening had to be increased from the f 4 setting used with the lens prior to irradiation to an f 2.5 setting following irradiation to obtain comparable picture quality.
- A 1 in. nonbrowning lens retained approximately 89% of its original light transmission capability after exposure to 1×10^9 R. Irradiation of this lens is being continued until it shows appreciable deterioration.
- A 4 in. nonbrowning lens lost some transmissivity, but was still usable as a camera lens after 1×10^8 R exposure. Irradiation of this lens is being continued.
- All lenses of conventional glass, both fixed focus and zoom type, lost approximately 95% of their original transmissivity after 1×10^6 R exposure and became useless as camera lens.

4. Plant Protection System

J. P. Thomas

The Plant Protection System includes the reactor scram subsystem and the actuation subsystems for containment and

emergency core cooling. The present concept for protection system circuitry includes redundant sensing channels used in majority logic coincidence schemes (2-out-of-3 or 2-out-of-4) implemented with relays which are energized during normal plant operation. The protection system instruments are connected to a very reliable power source (the zero time outage bus) and the logic elements are operated from redundant 125 V dc sources.

A drawing, Figure 2.1, shows the input to the scram circuit from the reactor and process system variables being monitored.

II. SYSTEMS

W. B. McDonald

HIGHLIGHTS OF SYSTEM 20. STRUCTURES, UTILITIES, AND CONTAINMENT

F. H. Shadel

The Structures, Utilities, and Containment Facilities for FFTF incorporate the following Systems and Facilities:

- Structures Facility
- Site Facilities
- Service Piping System
- Radioactive Waste System
- Heating and Ventilation System
- Plant Fire Protection System
- Reactor Containment System

The conceptual designs for the Site Facilities, Radioactive Waste System and the Reactor Containment System were completed and issued to DRDT during the quarter. The site facilities described are independent of the site location. The Radioactive Waste System incorporates a Radioactive Waste Facility separate from the FTR Building and includes interim storage tanks, a control room, and loadout and transfer facilities.

The Reactor Containment System described incorporates such principal structural components as a steel containment vessel, a reactor cavity vault, a reactor hot cell housing the fuel handling machine and other refueling components, primary sodium equipment cells and facilities, closed loop cells, and inert gas equipment cells.

20. STRUCTURES, UTILITIES AND CONTAINMENT

F. H. Shadel

The conceptual designs for the Site Facilities, Radioactive Waste System, and the Reactor Containment System were completed and issued to DRDT during the quarter. The following sections summarize the information presented.

1. Site Facilities

J. R. Samuel

The FFTF site facilities arrangement is shown in Figure 2.1-a. The plant layout takes into consideration individual building (or facility) location with respect to prevailing winds, access, utilities, safety, and future building requirements. The civil engineering features and ground improvements facilitate safe and efficient operation and maintenance of the FFTF.

The utilities provide the required plant sanitary and process sewers, steam, normal and emergency electrical substation, and other utilities as required for plant operation.

The site facilities arrangement provides for the following FFTF structures and facilities:

- Main reactor building
- Sodium receiving and disposal facilities
- Inert gas storage support structures
- Heat dump support structure
- Electrical substations
- Maintenance materials storage
- Radioactive waste facility
- Exhaust filter building
- Exhaust stack
- Facility heat source
- Water supply facilities
- Visitors' facility
- Radiometallurgy Facility (future)

2. Radioactive Waste System

L. J. Nitteberg

The FFTF Radioactive Waste System provides the equipment and facilities for the routine collection, transfer, segregation, temporary storage, and disposal of radioactive liquid and solid

FIGURE 2.1-a. Building Plan XVIII Plot Plan

SK-3-12981

wastes (except alkali metal which is disposed of by the Sodium Receiving and Processing System).

Radioactive liquid wastes are disposed of by the Radioactive Waste System via the Radioactive Liquid Waste Facility. Radioactive liquid wastes handled by this system are classified as follows:

- Low level (disposal and controls in accordance with AECM-0524)
- Intermediate level ($\leq 100 \mu\text{Ci}/\text{mL}$)
- High level ($> 100 \mu\text{Ci}/\text{mL}$)

Radioactive liquid wastes are received from systems within the FTR Building, the filter vault, and the sodium disposal building. The outside waste lines are shown in the plot plan Figure 2.1-b

a. Radioactive Liquid Wastes

The collection and interim storage of liquid wastes within the FTR building is shown schematically in Figure 2.1-c. A receiving tank is provided inside the containment vessel for potentially radioactive liquid wastes. Receiving tanks are also provided for radioactive aqueous and solvent wastes in the process area of the structures outside of the containment vessel.

The wastes are routed to the Radioactive Liquid Waste Facility for interim storage in low, intermediate, or high activity level tanks or a solvent waste tank in accordance with waste composition as determined by sample analysis. Waste flow routing and shielded liquid waste transfer cask and tank car loadout facilities are shown schematically in Figure 2.1-d.

The Radioactive Liquid Waste Facility shown in Figure 2.1-e houses the interim storage tanks, neutralizing equipment, shielded cask and railroad tank loadout stations, a control room, and supporting facilities.

2.16

BNWL-880

SK-3-14121
 FIGURE 2.1-c. Flow Diagram Radioactive Liquid Waste FTR Bldg.

2.17

BNWL-880

FIGURE 2.1-d. Flow Diagram Radioactive Liquid Waste Facility

SK-3-14116

The conceptual design of a shielded liquid waste cask is shown in Figure 2.1-f.

b. Radioactive Solid Wastes

Routine radioactive solid wastes are packaged by the system generating the waste and are then collected and transported to the 200 Area burial ground by the Radioactive Waste System. The waste recipient, Atlantic Richfield Hanford Co. (ARCHO), has established segregating and packaging requirements for critical mass and chemical hazards control necessary for safe and proper burial ground management.

The conceptual design of a waste container and shielded cask is shown in Figure 2.1-g. The cask will be used in removing solid wastes from spaces having an inert gas atmosphere. The cask design, therefore, incorporates features for maintaining cell atmosphere purity during the waste container loading operation.

3. Reactor Containment System

N. Vieli

The Reactor Containment System is an integral part of the FFTF Reactor Containment Concept. It consists of the containment structures and the Containment System Instruments and Controls. It is designed to minimize, in conjunction with the Heating and Ventilation System, the Inert Gas Receiving and Processing System, the Plant Fire Protection System, and the Radioactive Waste System, the consequences of any accident which might release radioactive material from the reactor and its components. These accidents include:

- The Design Basis Accident (DBA)
- Sodium air reactions (sodium fires)
- Irradiated fuel accidents in the core, in transit or in storage within the containment vessel; and
- Accidents involving handling of contaminated equipment.

PLAN - OUTER SPHERE CUT AT HORIZONTAL C

LIFTING YOKE

ELEVATION

LIQUID WASTE TRANSFER CASK - 500 GAL. CAPACITY

ELEVATION - INNER SPHERE

REFERENCE DRAWINGS:
SK-3-9578 SHTS 1 THRU 9
H-3-21110, 111 & 112
H-3-21264 SHTS 1 THRU 3

CONCEPTUAL ■ 1/27/15 UG 2-18-GH

2.20

BNWL-880

SK-3-14117

FIGURE 2.1-f. Typical Shielded Liquid Waste Transfer Cask

2.21

BNWL-880

FIGURE 2.1-g. Typical Shielded Solid Waste Transfer Cask

SK-3-14119

The reactor containment concept provides the following nuclear safety functions:

- Contains the effects of the full range of accidents up to and including the DBA and limits the release of radionuclides in any accident to meet the Guidelines of AEC Site Criteria 10 CFR 100. The DBA is defined (see Chapter III, Section D) to be a major reactivity insertion or a loss of coolant accident followed by core meltdown. The maximum nuclear excursion energy release is estimated to be 1500 MW-sec of work energy. The initial shock and blast energy released in the core is absorbed by plastic deformation in the core structure, the reactor vessel, and the reactor blast shield. Following deformation of the vessel or the vessel head, a quasi-static pressure is expected to build up in the first containment barrier zone which consists of the reactor cavity vault and the reactor hot cell. Under this pressure, radioactive gases leak out into the equipment cells which function as a containment barrier zone, and/or to the containment space bounded by the containment vessel which constitutes the final containment barrier.
- Minimizes occurrence of sodium air reactions by providing an inert gas atmosphere in radioactive sodium system equipment cells.
- Reduces temperatures, pressures, and airborne contamination after a DBA or other accident, by cooling and condensing gases and vapors released in the accident with ventilation and clean-up systems.
- Reduces residual atmospheric activity in sodium system cells by simultaneously drawing in clean inert gas while discharging contaminated gas to the inert gas purification system.
- Reduces residual containment air activity by simultaneously drawing in clean air while discharging contaminated air through the stack and filter system.

The specific functions of the Reactor Containment System are:

- The reactor cavity vault and the reactor hot cell function as a first containment barrier in a DBA or sodium air reaction accident. (See Figures 2.2 and 2.3 for descriptive purposes.)
- The reactor containment vessel functions as the final containment barrier in any accident, including the DBA, to limit release of radionuclides to meet the guidelines of AEC Site Criteria 10 CFR 100.
- The reactor cavity vault and the reactor hot cell contain the initial pressures, temperatures, and dynamic impacts resulting from the DBA or sodium air reaction accident.⁽¹⁾ They protect containment structures and other system components located outside the cavity vault and the hot cell from being damaged by missiles originating in the reactor assembly and from accident initiated temperature and pressure transients.
- The reactor hot cell functions as a shielded space for reactor refueling operations. It has the capability to confine the effects of a radioactivity release accident in the reactor hot cell and the capability to facilitate recovery from such an accident.
- The reactor cavity vault and the reactor hot cell structures function as biological shields to adjacent cell and work areas. Shielding requirements are determined from occupancy ratings and applicable safety standards.
- The Primary Sodium and Inert Gas System cells function as the first barrier for accidents originating within these cells. They have the capability to confine and control cell accidents involving radioactivity release.

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1. *The reactor cavity vault has an inert gas atmosphere. If the hot cell also has an inert gas atmosphere, no sodium fire can occur. The space under the secondary gamma shield will have an inert atmosphere to preclude sodium fires in*

FIGURE 2.2. *Transverse Section Through the Reactor Hot Cell*

FIGURE 2.3. Longitudinal Section
Through the Reactor Hot Cell

They are provided with purge and exhaust filtration facilities to aid in the recovery from radioactivity release accidents. The cell structures function as biological shields to meet the criteria requirements.

- Work areas and galleries within the containment vessel provide the space for normal reactor operations and maintenance activities.
- The instruments and controls for the reactor containment system:
  - (a) Continuously monitor pressures, temperatures and radioactivity in individual cells, work areas and the containment vessel effluent gases, and initiate partial or total containment in case of an accident involving the release of radionuclides.
  - (b) Initiate automatic ventilation containment (exhaust and supply valves close simultaneously) upon indication of radionuclide concentration in the exhaust air which exceeds a preset limit.
  - (c) Initiate the automatic closure of all containment valves not essential to safe reactor operation upon an increase in pressure following ventilation containment.
  - (d) Initiate a controlled reactor shutdown upon ventilation or total containment.
  - (e) Initiate closure of containment valves isolating the individual cells whenever the cell pressure, temperature, or airborne activity exceeds preset limits.

- In the event of an accident, partial or total containment is maintained until the accident has been evaluated and conditions have been corrected. The Containment System is returned to normal by manual control. If a ventilation closure occurs and there is no indication of a pressure increase, the incident is evaluated and the system returned to normal by manual control after corrective action has been completed.
- Personnel access to the containment vessel is controlled during reactor operations. All space inside the containment vessel is treated as a controlled work zone. Where the dose rate exceeds 10 rem/hr or the concentration of radionuclides in air is >1000 times the control concentration, access to the space will be controlled by locked physical barriers. The locations where the dose rate is lower than 10 rem/hr or the concentration of radionuclides in air is <1000 times the control concentration, will be posted as controlled work zones.

The principal structural components of the Reactor Containment System are:

- The containment vessel
- The reactor cavity vault which houses the reactor vessel and shield
- The reactor hot cell which houses the fuel handling machine and other refueling equipment components
- Primary sodium equipment cells (This includes the cells for the Primary Reactor Heat Transport System, the Irradiation Test Facility, the fuel decay storage system, the Sodium Receiving and Processing System and the Fuel Failure Monitoring System.)
- Inert gas system cells (These include the cells for the Inert Gas Receiving and Processing System.)

Other FFTF systems provide the following functions in support of the Reactor Containment System:

- The Heating and Ventilation System provides for air recirculation and exhausting the air through filters and a stack system. Provisions will be made to continuously monitor and record gaseous and particulate radioactivity in the exhaust gases.
- The Inert Gas Receiving and Processing System, which maintains the atmosphere in the inerted cells also provides for hold up and decay of radioactive gases and the controlled release of gases to the atmosphere via the exhaust filter and stack systems.
- The Plant Instrumentation System provides the instruments and controls necessary to meet the functions and requirements of the Reactor Containment System.
- The Radioactive Waste System provides for the safe collection, storage, and disposal of radioactive liquid and solid wastes generated in the FFTF facility.
- The Plant Fire Protection System detects and annunciates fires within the FFTF facility and provides automatic or manual fire extinguishing capability.

II. SYSTEMS

W. B. McDonald

HIGHLIGHTS OF SYSTEM 30. REACTOR PLANT

R. C. Walker

The FTR reference core is a split conical design having five vertical closed loops in the split zone. Task force studies have indicated, however, that an annular conical core with three vertical central closed loops have higher neutron flux capability than the reference core. Parallel work is being continued on the design of a vertical core. Investigations of three reactor vessel concepts have identified several design features which will be incorporated in the vessel conceptual design. A conceptual design description has been completed for the Nuclear Proof Test Facility based upon an NPTF reactor which is to be a nuclear mockup of the FTR. The NPTR will utilize FTR fuel pins canned in frozen sodium and arranged in the same geometry as in the FTR to provide a neutronic environment which is prototypical of the shutdown FTR at normal refueling temperature.

30. REACTOR PLANT

R. C. Walker

The Reactor Plant is comprised of the reactor core, fuel elements, reactor vessel, core support and positioning components, control and safety rods, and shielding. The Fast Test Reactor (FTR) furnishes the fast neutron flux for the irradiation testing facilities. This group also includes work on the Nuclear Proof Test Facility (NPTF), a critical facility for verifying physics calculations and test fuel loadings planned for the FTR.

1. Improved Reference Core Task Force Study

Results of the subject task force are currently being documented in a separate topical report. Although a significant amount of conceptual design work remains relative to

strongly interfacing systems such as the fuel handling systems and hot cell design (including evaluation of containment and refueling DBA requirements), the task force has concluded that the annular conical core design, with a central arrangement of three vertical closed loops, represents the best overall dispersed core configuration for the FTR. This conclusion is based on the judgement that this core concept offers acceptable performance capability, conservative thermal and nuclear design, a desirable complement of testing facilities, and full instrumentation capability.

Further details of the subject study results are presented in Chapter III of this document.

2. Vertical Core Studies

- a. General Electric Company Contract BDR-341, Backup Core Design
F. J. Kempf

(1) Core Design. Recent studies of the two zone enrichment core, initiated as a result of the possible need to use tantalum reflector control rods, indicate several substantial advantages. The major advantages are:

- (a) A flattening of the power gradient across the core.
- (b) A minimal reduction in peak neutron flux. (Center flux remains approximately 0.9×10^{16} n/cm²-sec.)
- (c) Reduction in core size from 139 subassemblies to 127 subassemblies with a concurrent increase in control worth margin. The calculated 24% increase in reflector control worth should be sufficient to allow tantalum as an alternate control material for the backup design.
- (d) Possibly reduction in number of subassembly orifices from 7 to 3.

- (e) Increase in average power or flux can be achieved in the outer zone. A flux increase from 0.55×10^{16} to 0.65 or 0.7×10^{16} n/cm²-sec appears attainable at the periphery of the core.

(2) Control Rods and Drives. Optimization of the two-zone core will allow the reflector control configuration to attain a hexagonal array surrounding the 127 subassembly core. Nineteen control rod drive mechanisms actuate 38 reflector control rods which are laterally offset with respect to the drive vertical axis. Essentially half of the reflector control drives activate primary safety rods. Three in-core backup safety rods are also provided.

Control rod and drive parametric studies have resulted in the current selection of a 10-in. offset between drive and rod centerlines. This selection allows drive nozzles on the reactor vessel head to be displaced outward as far as possible and maximizes space utilization on the head. Problems associated with the 10-in. offset which have been identified and are being investigated include:

- Dashpotting the drive mechanism and control elements near the end of the scram stroke
- Clearance requirements around damping pistons
- The usual friction, galling and hangup considerations
- The mechanical problems of maintaining the selected offset without excessive flexure from the eccentric loading.

(3) Driver Fuel Instrumentation. Emphasis on alternative concepts for detecting failed drive fuel elements have included the development of layout drawings that show how 0.5 in. sodium sample lines can be accommodated in the refueling plug structure. These lines extend from the fuel elements to the underside of the refueling plug and terminate at its periphery. At this point the sodium is transferred, across a 100 mil gap

between the refueling and annular plugs, to corresponding lines in the annular plug. Several methods of making this sodium transfer between the lines in the two plugs remain under investigation.

(4) Closed Loop Design. The six closed loops are arranged as follows:

- One U-tube type loop remains located at the center of the core.
- Five re-entrant type loops are now located at equal radial positions from the core center.

The external configuration of both types of closed loop is essentially identical at elevations above the loop coolant piping connections. Relocation of the mid-core closed loops to equal radial positions from the core center has improved accessibility to the driver fuel.

3. Nuclear Proof Test Facility

Q. L. Baird, R. A. Bennett, S. L. Engstrom, C. W. Higby, and H. C. F. Ripfel

A conceptual design description for the Nuclear Proof Test Facility (NPTF) has been completed and is being reviewed for approval. The Nuclear Proof Test Reactor (NPTR) is to be a nuclear mockup of the FTR. The concept described in this section is based upon the reference split conical FTR core and will be revised to match the final FTR core configuration. The basic design as described herein, namely the use of close packed core modules containing fuel pins potted in frozen sodium, is expected to be applicable to the final core configuration.

a. Summary Description

The NPTF comprises the NPTR and its appurtenances which include the reactor cell, control room, fuel storage vault, work areas, instrumentation, fuel handling equipment, and containment

system. The facility is located in the FFTF Building adjacent to the nonirradiated fuel handling area.

The NPTR is a nuclear mockup of the FTR. The reference concept uses pin type fuel canned in frozen sodium within tapered hexagonal modules which are packed together to form the reactor core. The core elements are arranged in the same geometry as the FTR. The reference concept provides for testing of actual FTR fuel and test assemblies in all core positions.

b. Facility Description

The NPTF arrangement is shown in Figure 2.3-a. The reactor cell is a cylindrical, shielded vault approximately 35 ft in diameter and 60 ft high having access ports for transferring equipment, fuel and other components, and air lock doors for personnel access. Cell equipment includes a rotating bridge crane, service platforms, a service elevator, and a shielded stairway for emergency egress. Space is available for experimental apparatus not yet defined.

The NPTR cell utilizes a containment system which is independent of the FTR containment. Separate containment is provided to preclude an accident or operating restriction in either the NPTF or FTR resulting from an accident or malfunction in the other.

The NPTF fuel storage vault is adjacent to the work area. The dimensions (lateral distances between holes) are chosen such that subcriticality is maintained even when water-flooded. The storage vault is estimated to require approximately $700 \text{ ft}^3/\text{min}$ of room air to remove 12 kW of decay heat from stored fuel.

c. Reactor Description

The Nuclear Proof Test Reactor (NPTR) is the major component of the Nuclear Proof Test Facility. The NPTR is a critical

FIGURE 2.3-a. N.P.T.F. Arrangement Concept I

SK-3-14142

assembly designed for a maximum power level (fission energy) of approximately 2 kW, and normally operates at essentially zero power. Occasional power ascensions to levels up to 2 kW are required for reactor period measurements and/or extended irradiations.

The NPTR simulates the FTR in having a like number of core and reflector elements arranged in the same geometry as the FTR. The NPTR reference concept is shown in Figures 2.3-b, 2.3-c, and 2.3-d. This design has hexagonal modules which contain reflector and core fuel elements in frozen sodium. NPTR fuel is in the form of FTR type driver fuel pins. The fuel module shown in Figure 2.3-c simulates a 217 pin assembly. One hundred sixty-two pins are cast in sodium; the remaining 55 positions have aluminum tubes for ready changeout of about 25% of the fuel pins to simulate different core compositions (e.g., burnup simulation in the whole core). The replacement of sodium by the aluminum tubes has a relatively small effect on the neutron spatial and energy distribution. The aluminum tubes with fuel pins removed provide access for neutron counters, flux wires and foil assemblies for experimental purposes.

All hexagonal modules are interchangeable, permitting placement of fuel reflector elements, poison rods (safety, control, or shim rods), or tests in all core positions. Full length or partial FTR driver fuel or test assemblies may be inserted in any core position.

The α -decay heat generation of a core loading containing 700 kg of plutonium has been estimated to be about 3 kW based on a representative FTR isotopic composition. Together with a short-term fission power level of 2 kW, a total maximum heat generation of 5 kW results. Approximately 300 ft³/min of room air is forced through the core to remove this amount of heat. The radius of curvature on the corners of the hexagonal driver

REFERENCE CONCEPT

PRELIMINARY

SK-3-12821

FIGURE 2.3-b. Nuclear Proof Test Facility Reactor Arrangement I

2.37

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FIGURE 2.3-c. Nuclear Proof Test Facility Driver Fuel Module

SK-3-12823

PRELIMINARY 1/24/68
1/17/80 R/12

2.38

BNWL-880

SK-3-12827

FIGURE 2.3-d. Nuclear Proof Test Facility Safety Control Rod II

fuel subassemblies provides channels for this purpose. Passing the air downwards through the core and releasing it through filters to the stack or to the building protects against spread of contamination in the case of fuel cladding failure.

The reactor instrumentation (Figure 2.3-e) consists of two channels of low level (startup) instrumentation, two channels of intermediate level instrumentation, two high level safety channels, and two channels of wide-range linear instrumentation.

Instrumentation, in addition to the nuclear instrumentation, includes thermocouples to monitor temperatures in the reactor core and in inlet and exit air streams. The air coolant flow is also monitored to provide surveillance of the reactor cooling system operation. Reactor cell instrumentation includes means for monitoring cell temperature, absolute pressure, and radiation level. All instruments read out in the NPTF control room, and operate independently of other instrumentation systems.

Neutron flux monitoring channels with chamber traversing mechanisms are provided for neutron flux mapping. Small diameter fission chambers, e.g., ^{235}U and ^{239}Pu , are inserted in reactor core channels and traversed through the core by a remotely actuated traversing mechanism with a remote position indicator. Flux wires and foil assemblies may also be used for reaction rate measurements. Other special equipment such as for oscillator and Doppler experiments requires additional definition.

4. Babcock and Wilcox Contract No. AT(11-1)-1280, FFTF Reactor Vessel Studies

O. W. Priebe

As a method of establishing reactor vessel features, Babcock and Wilcox Co. has been performing material studies, flange analysis, failure models, seismic analysis, and hydraulic consultation.

FIGURE 2.3-e. Reactor Instrumentation Block Diagram

Studies of selected features of three reactor vessel concepts with different inlet piping methods were studied. The three concepts are identified as the annular inlet concept (Figure 2.3-f), the piped downcomer concept (Figure 2.3-g), and the bottom entry concept (Figure 2.3-h). Depth of study was limited principally to steady-state, full power temperature and pressure conditions. Computations were restricted to provisions of the "ASME BOILER AND PRESSURE VESSEL CODE (1965) SECTION III NUCLEAR VESSELS".

Structural considerations show less difficulty with the downcomer core support than with the annular core support. Weld studies show that weld size, filler metal, and welding repair is about the same for Type 304 and 316 welds. Sigma phase is slightly less detrimental in Type 316 welds. Irradiation data are not available on weld metals studied. Guidance and head cooling studies indicate a temperature of about 700 °F is satisfactory. Slip joint data is virtually nonexistent. The annular design is more dependent on a slip joint provision than is the downcomer design.

The bottom entry concept appears to be the best choice at this time because it eliminates the need for slip joints that are required for the other two concepts, and it avoids the large uplift forces and collapse considerations due to hydraulic pressure on the core support cylinder of the annular and piped inlet concepts.

As a result of the Babcox and Wilcox effort, recommendations have been made of a number of features that the conceptual vessel configuration should include. Those features agreed upon to form characteristics of the reactor vessel are:

- Piped bottom entry
- A high pressure inlet plenum
- A single wall vessel

FIGURE 2.3-f. Annular Inlet Reactor Arrangement VII

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FIGURE 2.3-g. Piped Inlet (Helices) Reactor Arrangement IX

FIGURE 2.3-h.

- A surrounding guard vessel without penetrations
- Vessel-cover integrity that assures that the cover remains intact under DBA conditions
- Six inlet and three outlet nozzles
- A design philosophy of ease and reliability of inspection during fabrication, erection, and usage
- Vessel base material to be Type 304 SS
- Top support of vessel
- Provisions for reactor components.

II. SYSTEMS

W. B. McDonald

HIGHLIGHTS OF SYSTEM 40. FUEL HANDLING AND RADIO-
ACTIVE MAINTENANCE

W. E. Cawley

An interim Reference Concept was established for the Irradiated Fuel Handling Facility (IFHF) which receives, transports, cools, processes, stores, and provides for the shipment of various irradiated core components.

Atomics International (AI) performed analyses of the types of fuel handling machines and equipment required for the split conical core.

The feasibility of a gas-cooled fuel handling machine for driver fuel subassemblies and experimental test trains of similar geometry both with decay heat loads of 30 kW was investigated.

Concepts were developed for use of removable secondary gamma shields and nozzle preparation methods. Practical methods for remote gamma shield removal using the hot cell crane and special hoisting and jacking equipment were developed.

40. FUEL HANDLING AND RADIOACTIVE MAINTENANCE

W. E. Cawley

1. Irradiated Fuel Handling Facility

C. L. Peckinpaugh and D. E. Bloomfield

An interim Reference Concept was established for the Irradiated Fuel Handling Facility (IFHF) which receives, transports, cools, processes, stores, and provides for the shipment of various irradiated core components. These irradiated components are received from the Reactor Refueling System and are delivered to either the Inert Gas Cell Examination Facility or fuel reprocessing facilities. Packaged pins, specimens, and hardware are received from the examination facilities and shipped to radio-metallurgy facilities. Core components are received from the Nonirradiated Fuel Handling System and transferred through reactor containment to the Reactor Refueling System.

Figure 2.4 shows a plan view of the reference facility. Figure 2.5 shows the reference concept, facility layout, and mechanical process operations.

The Irradiated Fuel Handling Facility consists of the following major areas or subsystems:

- Transfer cell
- Irradiated fuel cleaning cell
- Fuel storage basin and loadout basin
- Irradiated fuel shipping area.

a. Transfer Cell

The transfer cell is a multipurpose cell about 12 ft wide by 55 ft deep by 78 ft long. All handling, transferring, and other activities are normally performed by remote methods in an inert atmosphere. The cell is normally sealed from the reactor containment vessel, cleaning cell, and inert gas examination cells. The transfer cell equipment includes manipulators, hoists, or cranes, remote tools, fixtures, and a fuel transfer cell mechanism(s) containing cooling devices. Direct visual control of operations is provided by remote viewing equipment including viewing windows, TV, and periscopes. A waste processing area is provided to cut components, package waste, and temporarily store components, waste, and waste containers. Equipment located in this area consists of a shear, work table, manipulators, remote tools, and containers.

b. Cleaning Cell

The irradiated fuel cleaning cell provides facilities for the safe removal of sodium and final cleaning of components before they are transferred to inert gas examination or underwater storage.

Two cleaning vessels are enclosed sealed chambers in which controlled reactions between the cleaning mediums and sodium

Neg 0681546

SK-3-14149

FIGURE 2.4. Irradiated Fuel Handling System (Alternate I)

2.49

BNWL-880

PLAN VIEW
@ EL (+)20'-0"

ELEVATION

CONCEPTUAL

Neg 0681538-7

SK-3-14150

FIGURE 2.5. Close Coupled Fuel Handling and Examination

2.50

BNWL-880

contained on the component surfaces take place. Remotely operated equipment is used for handling the irradiated components and for performing the cleaning operations.

Remote viewing equipment including shielding windows, periscopes, and TV is utilized for viewing the handling operations required for inserting and removing the elements from the vessel and viewing the cleaning area and equipment. The liquid and gaseous wastes generated from the cleaning operations are handled by other FFTF systems. Equipment is designed to remove the decay heat and heat of reaction and to control temperatures during cleaning operations.

c. Fuel Storage and Loadout Basin

The clean components are transferred from one of the two cleaning vessels into the fuel storage and loadout basin. Fuel elements which are to be put into the fuel storage basin are inserted into a fuel handling basket contained within the basin entry transfer dolly. This dolly transports them to a leak monitoring area where the individual fuel basket is removed and inserted into a leak monitoring vessel.

The driver and test sections are checked for leakage. Leakers are encapsulated and subjected to special handling. Normally, fuel elements are removed from the leak monitoring cans and transported by a monorail system to a position in the basin storage racks.

The basins contain demineralized water. A water recirculation system is utilized to clarify, cool, decontaminate, and remove soluble fission products from the water. Filter(s), recirculation pump(s) and demineralizer(s) are required for the water recirculation system.

A cask loadout area is located at one end of the fuel storage basin for handling and loading the shipping casks.

d. Irradiated Fuel Shipping

The cask monitoring, decontamination, and storage area provides space and facilities for receiving, storing, monitoring, decontaminating and performing tests upon incoming or outgoing shipping casks.

Shipping casks are provided by this facility for offsite shipment of driver fuel sections and pins to reprocessing facilities. Shipping casks for experiments are furnished by the experimenters.

2. Atomics International Fuel Handling Studies

Atomics International (AI) performed analyses of the types of fuel handling machines and equipment required for the split conical core. Results and conclusions as summarized by AI are given below.

a. Secondary Gamma Shield Study

Various concepts of secondary gamma shields were studied to define the requirements and to establish alternative designs. Several concepts of both rotating and stationary shields were developed. A double rotating gamma shield offered advantages for indexing to reactor nozzles and required the least number of floor valves. The stationary concepts were simpler because of fewer moving parts; however, the stationary concepts required floor valves to be moved from nozzle to nozzle by external means and this results in extending the refueling cycle time.

b. Single Purpose Multi-FHM Study

The benefits derived from using two or more fuel handling machines was assessed. Duplicate machines offer the potential of a faster refueling cycle, equipment redundancy, and versatility. A transporter system for the FHM's was identified. The time savings resulting from nozzle preparation in advance of refueling rather than during refueling were investigated. It

was found that (1) fuel handling operations were simplified, and (2) a definite time savings could be realized.

c. Gas-Cooled FHM Study (Shielded-Continuous Cooling)

The feasibility of a gas-cooled fuel handling machine for driver fuel subassemblies and experimental test trains of similar geometry both with decay heat loads of 30 kW was investigated. A machine was conceived to force helium gas down through the fuel subassembly as it was lifted from the reactor sodium pool into the fuel handling machine. The 30 kW decay heat load could be removed indefinitely with the fuel subassembly at any stage of the lift. This concept could have the adverse effect of introducing gas bubbles into the primary sodium during the early stages of lifting the fuel subassembly. The study was conducted in sufficient detail to permit direct comparison with sodium-cooled fuel handling machine concepts.

d. Sodium-Cooled FHM Study (Shielded - NaK 100 kW Capacity)

A sodium-cooled fuel handling machine for handling driver fuel assemblies and experimental test trains of similar geometry developing 30 to 35 kW of decay heat was studied. In this concept, reactor sodium would be transferred into the FHM to act as the primary coolant and either NaK or the reactor hot cell atmosphere could be used as the secondary coolant system. The NaK or forced convection cell atmosphere secondary system would limit the fuel temperature to 1200 °F and below emergency conditions, natural cell atmosphere convection would limit the fuel temperature to 1600 °F. The FHM forced convection secondary NaK cooling system could be designed to remove up to about 100 kW of fuel element decay heat.

e. Fuel Subassembly Length Effects on FHM Study

The effects of driver fuel subassembly length on sodium-cooled and gas-cooled fuel handling machines was evaluated. Two lengths, 37 ft and 16 ft, were given by PNL as the comparison

cases. Short fuel subassemblies permit reduction in FHM weight and height (estimated at 150 tons to 110 tons and 60 ft to 32 ft for the 37 ft and 16 ft lengths, respectively). A slight saving in cycle time is possible by use of shorter cells (reduced hoist time) and a simpler sodium level control for the sodium-cooled FHM. No discernible benefits were found in FHM alignment, indexing, or overall manipulation.

f. Minimum-Maximum FHM Shielding Study

The effects of shielding thickness on the fuel handling machines were investigated. Two thicknesses, 15 and 35 cm of lead, were given by PNL as the comparison cases. There were no significant advantages for partially (15 cm of lead) shielded fuel handling machine over a fully shielded machine relative to cost or development. Operation and maintainability were adversely influenced by partial shielding for cases of stuck fuel in the fuel handling machine.

g. Fuel Transfer Cell Study

Transfer cell concepts were identified for use by PNL. Methods for transfer out of containment, disassembly of fuel subassemblies, and cleaning, and transferring to storage were developed. No concept recommendation was made because of the need for integrating the transfer cell with the overall containment design. However, transfer cell interfaces with the reference gas-cooled and sodium-cooled fuel handling machines were developed.

h. Decay Heat Determination Study

An investigation of the fission product decay energy of the reference driver fuel subassembly established that a reduction in decay heat load is permitted because of the high energy gamma radiation escaping from the fuel bundle. About 84% of the energy is retained as thermal energy. Coefficients for

adjusting the PNL decay heat valves were developed for various periods of full power operation.

i. Gas-Cooled FHM Study (Unshielded-Noncontinuous Cooled)

The feasibility of using a noncontinuous gas-cooled fuel handling concept was investigated. This extrapolation of HNPF fuel handling technology for FFTF application was found to be feasible up to 30 kW of the decay heat load. In this concept, the fuel subassembly would be lifted from the reactor sodium pool to the fuel handling machine prior to initiating gas cooling thereby eliminating problems of gas entrainment in the primary system. Only during emergency operation, such as fuel stuck in the nozzle region, is gas introduced into the primary system. This design would provide the reference gas-cooled FHM concept.

j. Sodium-Cooled FHM Study (Unshielded-Simplified Cooling)

Potential improvements to the sodium-cooled fuel handling machine cooling system design through elimination of the shielding were studied. These concepts utilize a ducted chimney for direct gas-cooling of the FHM cask and would result in substantial increases in system reliability.

k. Removable Secondary Gamma Shield Study

Concepts were developed for use of removable secondary gamma shields and nozzle preparation methods. Practical methods for remote gamma shield removal using the hot cell crane and special hoisting and jacking equipment were developed. Storage of the gamma shield after removal was also considered in the study. Remotely operated nozzle preparation machines were shown to be practical, both with and without the gamma shield in place. Designs resulting from this study would be utilized in an unshielded reactor refueling system.

1. Sodium FHM Level Control Study

Alternative methods for raising and lowering the liquid level in the sodium-cooled fuel handling machine were developed. Four concepts were evaluated based on unweighted criteria of reliability, performance, state-of-the-art, and interface complexity. The simple vacuum system was found to be superior and was recommended.

II. SYSTEMS

W. B. McDonald

HIGHLIGHTS OF SYSTEM 50. HEAT TRANSPORT

J. H. Westsik

The PNL concept for the Heat Transport System consists of four parallel circuits each having a primary and secondary sodium loop and a modular heat dump. The Conceptual System Design Description has been submitted to DRDT for approval. Pump design studies have started under LMFBF Phase II contract. Procurement of IHX design studies is under way. Concept selection for the Emergency Core Cooling System is scheduled for late 1968.

50. HEAT TRANSPORT SYSTEM

J. H. Westsik

1. Main Heat Transport System

C. M. Cantrell

a. Heat Transport System Concept

The Heat Transport System concept as described in the Conceptual Design Description is based on a total heat removal capacity of 400 MW_t . Four loops are provided, each with 133 MW_t heat removal capacity, which makes available one redundant loop. Sodium primary coolant loops, piping, and components are located in shielded cells within the containment sphere. Components of the sodium secondary loops and most of the piping are located outside the containment sphere. Final heat dissipation is through a sodium-to-air heat dump.

Pumps in both the primary and secondary loops are located in the cold leg sections. Primary and secondary sodium pump concepts are based on submersible vertical free surface type pumps.

The intermediate heat exchangers are vertical shell and tube similar in general concept to the units used at the Fermi

plant. The units are designed to permit removal of tube bundles. The concept is arranged with primary sodium flow on the shell side and secondary sodium flow on the tube side. The units are designed for maintaining a high pressure on the secondary side to prevent leakage of the contaminated primary radioactive sodium into the secondary system.

The air heat dump is of a modular construction consisting of tube banks carrying the secondary sodium, ducts, casing and stack for air flow, and dampers and louvers for adjustment of cooling rate and of electrically driven blowers. Each secondary coolant loop has four heat dump modules.

b. Status of the Heat Transport System Design

The Conceptual System Design Description (CSDD) for the HTS was approved by PNL and forwarded to DRDT for approval action in January 1968. The document described the concept as outlined above.

Further conceptual studies are being conducted to improve definition of the concept and to investigate wider ranges of operating parameters. These include:

- Study of economics of redundant loops and their effect on plant availability goals
- Effects of lowering the primary hot leg temperature from 900 to 800 °F while maintaining 300 °F core temperature rise
- Need for and benefit of loop isolation devices and methods to provide isolation
- Need for and means to provide strainers in the system
- Functions of the HTS in providing emergency core cooling.

Design studies for the HTS pumps are in progress as part of Phase II of the LMFBR pump study effort. These studies of critical parameters affecting the pumps such as shaft length,

NPSH requirements, vibration problems, and suction configuration will lead to the start of preliminary design in the second half of FY-69.

Emergency core cooling is part of the HTS functions. Conceptual studies thus far indicate that with appropriate controls and power backup the HTS will be able to provide coolant to the core in the event of most emergencies. The analysis of cooling needs in the event of large pipe break is now in progress. Initial studies show three sequential phases. In the first phase immediately following the break, events occur before protective measures can be initiated. Passive features (e.g., pipe size and configuration) designed into the system must prevent core starvation in this phase. In the second phase the reactor controls have responded to the emergency, and a reduced but adequate coolant flow to the core is maintained until the coolant nozzles are uncovered. The final phase calls for adequate cooling until the reactor can be defueled.

Current efforts are directed toward detailed definition of the allowable core damage, analysis of core temperature patterns, level of cooling needs, and selection of the conceptual features to provide emergency core cooling during the three sequential phases just described. Concepts under consideration include double containment of parts of all of the primary boundaries, elevation of the primary system above the reactor to give natural convection cooling, isolation of defective loops, and the inclusion of a separate emergency core cooling system. Concept selection resulting from these studies is scheduled for late 1968.

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II. SYSTEMS

W. B. McDonald

HIGHLIGHTS OF SYSTEM 60. IRRADIATION TESTING

M. A. Vogel

The Closed Loop Conceptual System Design Description was completed and submitted for PNL Evaluation Board approval. The Closed Loop Design is based on a 6 MW loop permitting testing in 1200 °F coolant with up to 600 gal/min flow. To provide a proven in-reactor closed loop, a prototype out-of-reactor closed loop will be built to test individual loop components and system functions. The request for a building to house the prototype and other test facilities has been submitted for AEC approval.

Studies are being performed offsite to establish the feasibility and any preference between short-term irradiation facilities of either a trail cable (mechanical) or "rabbit" (pneumatic) type sample insertion method. Analyses thus far indicate that a liquid metal cooled system will be required.

60. IRRADIATION TESTING

M. A. Vogel

The FFTF will accommodate irradiation testing in the following:

- Closed loops
- Open test positions
- Short-term irradiation facilities.

The closed loops are to be located in specified core positions. The closed loops provide an isolated, recirculating coolant to an in-reactor tube for testing at conditions other than those provided for or permitted by the reactor coolant. Experiments involving testing to meltdown or purposely defected fuel would be conducted in the closed loops.

Open test positions are directly cooled by the reactor primary coolant. All driver fuel positions are potential open

test positions. Open tests are restricted to operation under conditions permitted by the reactor coolant. Meltdown or defective fuel element testing would not be permitted in open test positions.

Short-term irradiated facilities for testing small specimens for less than one reactor cycle may include a pneumatic or hydraulically operated rapid transit (rabbit) facility and/or a mechanically driven trail-cable facility. Heat removal from these facilities will be either by reactor coolant or by an auxiliary liquid metal or gas system.

1. Closed Loops System Management Plan

M. A. Vogel

An overall system Management Plan for the closed loops has been developed. The plan requires completion of conceptual design and the preparation of a conceptual system design description. A development and testing plan will be formulated based on the CSDD, coincident with preliminary design. During final design of the system, a closed loop test facility will be designed and constructed. After final design, a prototype loop will be constructed for testing within the closed loop test facility. Components which have been previously proof tested will be utilized to the maximum extent possible. Preliminary test results will be used to modify the final design and make any modifications in loop construction. Reactor loop fabrication will continue in parallel with final system testing to define operational limits and train operators for in-core operation. The reactor loops will be constructed and installed at the reactor site ready for checkout at reactor criticality.

The problem areas requiring investigation in a prototypical closed loop are:

- Seals for the in-reactor test tube which must withstand temperatures to 1200 °F

- Piping and component joints (mechanical or welded) to allow replacement, if required. The joints must seal radioactive sodium at temperatures to 1200 °F and withstand high thermal rates of change during reactor or loop startup or shutdown.
- Integrity of the in-reactor tube design considering differential thermal gradients between the closed loop coolant on one side and the reactor coolant on the other.
- Operating complications concerned with loop control, test charge-discharge, possible mass transfer, maintenance, and loop component performance.

2. System 61-P - Prototype Closed Loop

R. R. Wyer

The design and development program for closed loops includes design testing of individual loop components and system testing of a complete prototype loop prior to construction of a closed loop for the FFTF. System testing of the prototype loop will:

- Demonstrate the validity of the loop design
- Provide a suitable test bed for reliability and endurance testing
- Provide the experimental data required for the development of a workable remote maintenance system
- Provide a low risk experimental verification of the suitability of the FFTF closed loop operating procedures
- Train operators for the FFTF closed loops
- Provide means to evaluate future reactor loops and experiments.

Suitable facilities of system testing of the prototype loop do not exist at the present time. In order to satisfy this need and other current needs of PNL, a line item request for a combined engineering test and office building has been submitted to the AEC. The building, to be located in the 300 Area, will have a gross floor area of approximately 65,000 ft². Of

this space 27,000 ft² are for offices, 13,000 ft² are for high temperature sodium studies, and 12,000 ft² are for shop use. The prototype loop and its support equipment will be contained in the high temperature sodium studies area.

3. System 61 - Closed Loops

M. K. Mahaffey

The closed loop conceptual design description (CSDD System 61) was completed and submitted for PNL evaluation board approval. This design description provides the functions and requirements which will serve as a basis for preliminary design of the in-reactor tube and heat transport systems. The document presents the conceptual design including maintenance and operating principles and safety considerations. The appendices provide supporting studies and define the analytical and development effort required to complete the design.

Design and operating conditions specified in the conceptual design description are:

Heat removal capability	6 MW
<u>Primary Loop</u>	
Coolant	Sodium
Maximum Flow	600 gal/min
Pumping Head	250 psi at 600 gal/min
Maximum Temperature	1200 °F
Maximum Pressure	300 psig
<u>Secondary Loop</u>	
Coolant	Sodium
Maximum Flow	1000 gal/min
Pumping Head	50 psi at 1000 gal/min
Maximum Temperature	1200 °F
Maximum Pressure	300 psig
Material of Construction	Type 304 SS

4. System 68 - Short-Term Irradiation Facility (STIF)

R. R. Derusseau

A feasibility study for the Short-Term Irradiated Facilities was initiated during the report period. The objectives of the study are:

- To provide feasibility studies for STIF concepts.
- To provide a conceptual facility design for irradiating fuel and material samples for short time periods. The capability to insert and remove test specimens from the reactor during full power operation is a basic requirement for the conceptual design.

Layouts of a trail cable reflector tube were made to establish a basis for hydraulic and flow orificing calculations in a trail cable assembly. Heat transfer analyses were made for the following rabbit loop concepts:

- An all-gas system. This system uses gas for coolant and gas as the motive fluid to insert and remove test specimens.
- A gas carrier with liquid metal heat transport system. This system uses liquid metal for coolant and gas as the motive fluid to insert and remove test specimens.
- A single liquid metal loop system. This system uses liquid metal both for coolant and as the motive fluid to insert and remove test specimens.
- A dual liquid metal loop system. This system consists of two liquid metal loops. One loop is used to cool the test specimen. The other loop provides the means to insert and remove the test specimens.

II. SYSTEMS

W. B. McDonald

HIGHLIGHTS OF SYSTEM 70. FUEL EXAMINATION FACILITIES

C. L. Boyd

Fuel examination development work has begun on equipment for neutron radiography; and disassembly-reassembly; and studies have begun on inert gas atmosphere impurity control, fuel subassembly sodium removal, and sodium vapor control.

70. FUEL EXAMINATION FACILITIES

C. L. Boyd

1. Conceptual Design

D. J. Meyers and E. B. Ramsey

The conceptual design for the Inert Gas Cell Examination Facility (Figures 2.6 through 2.10) submitted to RDT for approval is based on the examination facility functions and design requirements as applicable to the split conical core FFTF Reactor Concept. This facility concept provides the operations staff and experimenters with a means for retrieving (under controlled environmental conditions) postirradiation data on fuels, materials, and components irradiated in the FTR. The facility also provides means for servicing fuel, material and component assemblies such as replacing test specimens, and repairing or replacing defective instrumentation.

Fuels, materials, and components to be examined or serviced in the facility will be representative of those for use in liquid metal fast breeder reactors and include vented and aqueous reactive fuels which may be tested to failure. Since a controlled inert environment is required for processing such materials, a facility containing a purified argon atmosphere, shielded confinement system (cell bank) and necessary operational support areas is provided.

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CONCEPTUAL

J-4-12	/ISSUE	2-26-68
J-3-14	/ISSUE	2-16-68
J-2-13	/ISSUE	11-13-67
J-1-12	/ISSUE	9-27-67

2.68

BNWL-880

Neg 0681538-5

FIGURE 2.6. Inert Gas Cell Examination Facility Arrangement

SK-3-12858

CONCEPTUAL

34 ISSUE A-28-62
250 ISSUE 2-16-68
R 12 ISSUE 11-8-67

Neg 0681538-4

FIGURE 2.7. Inert Gas Cell Examination Facility Plan Views

SK-3-12965

2.69

BNWL-880

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SK-3-12966

FIGURE 2.8. Inert Gas Cell Examination Facility Plans and Sections

SECTION G-G

SECTION F-F

SECTION H-H

SECTION A-A

■ 1/2" / 135W 2-28-67
 ■ 1/2" / 135W 2-28-68
 ■ 1/2" / 135W 11-8-67

CONCEPTUAL

Neg 0681538-2

FIGURE 2.9. Inert Gas Cell Examination Facility Sections

SK-3-12967

14 ISSUE 11-17-67

Neg 0681538-1

SK-3-12992

FIGURE 2.10. Process Flow Chart of Inert Gas Cell Examination Facility

The cell bank is equipped for completely remote operations and safe handling of highly radioactive materials. The facility is designed to ensure that reliable data is obtained from the postirradiation examination of fuels, materials, and components irradiated in the FTR.

a. Functions and Requirements

Space and handling, and examination equipment or means are provided in the cell bank to meet all functions and examination processing requirements as follows:

- Limited nondestructive interim examination of open test position and closed loop test assemblies irradiated in the FTR that can be removed from, examined, and returned to the reactor during a single reactor outage.
- Extended nondestructive interim examination of open test position and closed loop test assemblies, driver fuel assemblies, and other core components irradiated in the FTR that require one or more operating cycles for removal from, examination, and return to the reactor. This equipment must have the capability for repair and replacement of defective instrumentation and for replacement of individual fuel pins or specimens.
- Disassembly of assemblies, capsules, and components irradiated in the FTR whose irradiation has been completed.
- Final examination of driver fuel and test assemblies, capsules, fuel pins, specimens (fuel and materials), and components irradiated in the FTR whose irradiation has been completed.
- Packaging of fuel pins, capsules, specimens (fuel or materials) and components irradiated in the FTR whose irradiation is complete and have been examined.

- Critically-safe, cooled and shielded storage of assemblies, capsules, fuel pins, specimens and components irradiated in the FTR and being examined.
- Identification of assemblies, fuel pins, specimens, and components irradiated in the FTR during handling, storage, examination and packaging.
- Service the cell bank.
- Transfer fuels, materials, and components irradiated in the FTR to and from the fuel transfer cell, and to and from stations within the cell bank.
- Portable subassembly cooling and vapor control during transferring and processing of fuel assemblies.

b. Arrangement

The cell bank is arranged as a sealed, shielded, confinement system consisting of four in-line cells surrounded by operating galleries and support areas. All four cells of the sealed cell bank operate with a filtered, cooled, and purified argon atmosphere, under slightly negative pressure, which will assist in avoiding loss of experimental data particularly for those aqueous reactive fuels that may be tested to failure. These four cell areas are the cleaning and storage cell, the interim examination and disassembly cell, the equipment storage and transfer cell, and the detailed examination cell. The cleaning and storage cell and the interim examination and disassembly cell are sized for handling and processing core component assemblies 35 to 40 ft long in vertical orientation.

(1) Cleaning and Storage Area. The cleaning and storage cell provides space and equipment for handling, cooling, cleaning, and storage of fuel, material and component assemblies transferred (cooling of fuel subassemblies

is required during processing) from the FTR via the fuel transfer cell. The cell is sized to permit vertical handling of complete core component assemblies during transfer, cooling (fuel only), cleaning, and storage.

(2) Interim Examination and Disassembly Area. The interim examination and disassembly cell next in line is sized and equipped to accommodate fuel, material, and component assemblies designated for interim or final examination which must be handled, cooled, disassembled, examined, and reassembled in a vertical position.

(3) Equipment Storage and Transfer Area. The equipment storage and transfer cell next in line provides a shielded storage and remote maintenance area for all remotely operated in-cell equipment. It also acts as a transfer area for equipment to be remotely installed in the cell bank, for equipment being transferred from the cell banks, and for equipment being transferred from the cell bank for decontamination, contact maintenance, repair, or storage.

(4) Detailed Examination Area. The detailed examination cell, last in line, is sized to accommodate material capsules to be disassembled, and fuel pins, fuel and material specimens, and component parts to receive final examination, and to be stored, and packaged for shipment.

(5) Support Areas. The operating galleries surround the cell bank to provide visual and handling control of in-cell operations. Space is provided to accommodate installation and removal of manipulator master ends, for the movement of operational and FFTF experimenter personnel, and for the installation and management of equipment and consoles.

2. Alternative Concepts for Fuel Examination

E. B. Ramsey

PNL established two task forces to develop improved core designs for the FTR with the first working on an improved dispersed core geometry, and the second working on an improved vertical core arrangement. As part of the task force studies to define an overall fuel handling hot cell configuration compatible with the improved dispersed core, the following studies were made to identify and compare alternative concepts of an integrated fuel examination facility based upon the assumption that full length driver, test, and component assemblies would not be removed from reactor containment.

a. Concept Number I - Fuel Examination Within Containment

This scheme considers complete nondestructive fuel examination as well as the preparation of driver fuel assemblies for shipment to be reprocessed within containment. The facility is separate from the reactor refueling hot cell.

b. Concept Number II - Fuel Examination Within Containment

This scheme considers complete nondestructive fuel examination within containment with driver fuel assemblies having only their hanger rods removed for transfer from containment. The facility is separate from the reactor refueling hot cell.

c. Concept Number III - Interim Examination Within Containment

This scheme considers only interim nondestructive examination of full length test component assemblies within

containment. Final disassembly and detailed examination is performed in a separate fuel examination facility outside of containment. Hanger rods are removed from all assemblies prior to their transfer from containment. The facility is separate from the reactor refueling cell.

d. Concept Number IV - Interim Examination and Disassembly Within Containment

This scheme considers interim nondestructive examination of full length test and component assemblies as well as final disassembly within containment. Only shortened driver fuel subassemblies destined for reprocessing and capsules, fuel pins and specimens for detailed examination, and waste materials are transferred through the fuel exit confinement lock. The facility is separate from the reactor refueling cell. A separate detailed examination facility is provided outside of containment.

e. Concept Number V - Interim Examination Within Containment

This scheme is similar to Concept No. III. The facility shares part of the reactor refueling hot cell.

f. Concept Number VI - Examination Within Containment

This scheme is similar to Concept No. II. The facility shares the reactor refueling hot cell. The reactor refueling cell is located above an air reactor head access room of the under sodium refueling scheme.

Concept No. IV, Interim Examination and Disassembly within Containment, was selected as the reference method for integrating examination capability within containment, as discussed in the Task Force Report.

Criteria such as development risks, safety, reliability, performance, maintainability, and cost were

considered during preparation of these concepts. No attempt was made to establish weighting factors for the criteria used or the potential benefits and potential problem areas. Further evaluation of Fuel Examination Facility Concepts will be performed as part of an overall fuel handling-hot cell evaluation study in progress.

3. Gas Cooling of Fuel Assemblies During Examination

E. B. Ramsey

The objective of the FY-1968 activity in this program is to establish the feasibility of decay heat removal from FFTF fuel assemblies by the forced circulation of an inert gas stream during examination operations. Data from test results are required for input to subassembly gas cooling equipment development, inert gas system preliminary design and for planning future tests on fuel subassemblies of varying configurations.

Gas cooling demonstration tests were completed which demonstrated the feasibility of removing 20 kW (axial flow) and 10 kW (transverse flow) of decay heat while limiting maximum fuel cladding surface temperature to 600 °F. Test data are being reduced for incorporation into a preliminary report.

4. Fuel Examination Equipment Development

E. B. Ramsey

Neutron Radiography

Agreement was reached with General Electric (IPO) on the scope of work for conceptual design of Fuel Examination Facility neutron radiography equipment. The initial effort which includes preparation of a work plan and a

state-of-the-art survey is in progress. The equipment concept selected must also meet the requirements of the ANL Hot Fuel Examination Facility (HFEF) in accordance with agreements between the FFTF/HFEF design co-ordination groups.

The conceptual design of the neutron radiography equipment is required by the Architect-Engineer to establish the preliminary design of the overall Fuel Examination Facility. This includes the space, structural, and shielding requirements which have a large impact on the facility configuration.

Disassembly-Reassembly

Agreement was reached with Union Carbide Corporation for Oak Ridge National Laboratory (ORNL) on the scope of work for conceptual design of FFTF Fuel disassembly-reassembly equipment. The initial effort which includes preparation of a work plan and the technology survey phase of the design study is underway.

The conceptual design of the disassembly-reassembly equipment is required by the Architect-Engineer to establish the preliminary design of the overall Fuel Examination Facility. This includes the space, structural, and shielding requirements which have a large impact on the facility configuration.

5. Inert Gas Atmosphere Impurity Study

D. J. Meyers

A study is being performed by the PNL Metallurgy Department to provide the technical basis for selecting impurity control limits for the FFTF Fuel Examination Facility inert gas atmosphere.

Interim examinations of irradiated core components and experimental fuels are to be made in this facility. Disassembly and nondestructive examination will be done in an argon atmosphere. The components, after being removed from liquid sodium, will have surfaces wetted with sodium, and because of heat generation from fission product decay, the surfaces of fuel elements will be at elevated temperatures. The cell atmosphere must be sufficiently inert to preclude any harmful reactions with the surfaces of the core components which would interfere with the examination or impair the fuel performance after replacement in the reactor. The objective of this study is to define the impurities and the impurity levels that are known to be significant on the basis of existing knowledge and experience, and to define the areas where experimental work is needed to establish impurity level limits. The study anticipates probable material selection and operating temperatures for experimental core components which will be the subject of closed loop irradiations.

The report for this study will be issued shortly.

6. Fuel Subassembly Sodium Removal

E. B. Ramsey

Agreement was reached with General Electric (IPO) on the scope of work (LMFBR funded) required to develop a method for removing sodium from FFTF fuel components. GE has started the initial work effort which includes a state-of-the-art survey, comparison of existing techniques against FFTF examination requirements, selection of existing or new candidate cleaning methods, and preparation of a detailed work plan to perform the needed development programs. Further work will be performed under a separate contract to be negotiated upon PNL/RDT approval of the development work plan.

7. Sodium Vapor Control

E. B. Ramsey

A contract was negotiated with General Electric (IPO) to perform the initial work required to develop a method to control sodium vapor deposition on in-cell surfaces and examination equipment. GE has started the initial work effort which includes the identification and evaluation of sodium vapor problems which may be applicable to FFTF fuel examination cells and preparation of a work plan for performing the needed development program. Further work will be performed under a separate contract to be negotiated upon PNL/RDT approval of the development work plan.

II. SYSTEMS

W. B. McDonald

HIGHLIGHTS OF SYSTEM 80. AUXILIARY SYSTEMS

J. A. Westsik and R. V. Dulin

Design effort on the Auxiliary Systems has been directed principally toward development of conceptual designs of Systems 81 and 82 (Sodium and Inert Gas Receiving and Processing) and establishment and definition of additional conceptual effort required in the Auxiliary Systems area.

Selection of inert gases for the various FFTF applications has been reevaluated and reviewed with RDT, and a revised gas selection listing and supporting rationale submitted for their information.

Conceptual definition of additional auxiliary services has resulted in establishing the following two additional Auxiliary Systems:

System 84 - Decontamination and Cleaning

*System 85 - Sodium and Inert Gas Sampling
and Analysis Laboratory*

80. AUXILIARY SYSTEMS

J. H. Westsik and R. V. Dulin

1. Sodium Receiving and Processing System

R. G. Baumgartel

The approved Conceptual System Design Description for the Sodium Receiving and Processing System, which includes the services of sodium handling, purification and storage for all systems, was updated and revised, internally approved, and submitted for final printing.

2. Inert Gas Receiving and Processing

C. J. Foley

Discussions were held with RDT on May 2-3 to present the impact of the use of vented fuel in the driver core on

cover gas system conceptual design and gas selection. As a result of that meeting, it was agreed that inert gas systems conceptual design would not be required to make provisions for testing vented fuel in the driver core or open test positions.

Following this agreement, a reevaluation of gas selection was completed in which argon was selected for all sodium cover gas and fuel handling applications where contact with primary sodium is possible and nitrogen was selected for the cell atmospheres in cells containing radioactive sodium piping and equipment. These selections and supporting rationale were submitted to RDT for their information. Based on that selection, a revised Conceptual System Design Description is being prepared.

3. Additional Auxiliary Systems

Further project definition provided by studies of possible project breakdown structures has established the following additional systems which may be required:

a. Decontamination and Cleaning

Prior to performing contact maintenance on any of the radioactive sodium systems, the radiation levels around the instrument, component, or system have to be reduced to a tolerable level. This operation consists of one or more of the following steps:

1. Draining of sodium
2. Removal of sodium from surfaces
3. Decontamination of contaminated surfaces.

Since no other system provides the necessary functions of sodium removal and decontamination, System 84 - Decontamination and Cleaning was established.

This system includes centralization of the service functions of radioactive decontamination and sodium removal from components and from related reactor systems such as closed loops, reactor heat transport system, and sodium receiving and processing.

The system will provide not only the hardware required for supporting decontamination and sodium removal work but will act as a focal point for establishing the necessary technology in these areas.

4. Sodium and Inert Gas Analytical Laboratory

This proposed system consists of the radioactive sodium and gas analytical equipment required for making immediate laboratory analysis of radioactive sodium and gas from the closed loop or reactor primary systems, and to aid in providing system process control.

Since these functions cannot be done practically in existing laboratory facilities because of the high radiation involved, it appears that it may be necessary to incorporate specific provisions (e.g., hot cells) into the FFTF complex for their performance. Studies will be made to define the facility and firm up costs.

II. SYSTEMS

W. B. McDonald

HIGHLIGHTS OF SYSTEM 90. PLANT CONTROL AND DATA ACQUISITION

C. D. Swanson

The Central Control and Data Handling System concept includes a multiple configuration of digital computers and a plant simulation facility for operator training, operational standards studies, and checkout of the Central Control and Data Handling System during installation.

Design support studies for the Central Control and Data Handling System are being made by two independent design organizations.

Handling of plutonium bearing fuels dictates the use of a unique radiation monitoring technique to detect the alpha activity associated with measuring 1 MPC of plutonium in radon and thoron alpha background.

The following reactor channel instrumentation is being considered:

- In-core flowmeters are being planned for installation in each driver and open test position.*
- Each open test and closed loop assembly will have the capability for the attachment of sensors to the test assembly.*
- Each driver and open test assembly will be designed to have a remotely operated leak-tight signal connector and cable connections which provide flexibility for exchange of assemblies between driver and open test positions.*

90. PLANT CONTROL AND DATA ACQUISITION

C. D. Swanson

1. Plant Instrumentation and Control

J. W. Mitchell

a. Central Control and Data Handling System

G. L. Waldkoetter

The reference concept for the Central Control and Data Handling System (CC&DH) provides the capability for plant

coordination, data acquisition and display, analysis of variables, and implementation of improved control techniques. It is a safe and reliable system which will enable the FFTF to operate at its maximum potential, provide required data, and be enabled to accommodate changes in operating procedures and test requirements.

A multiple configuration of digital computers is planned for data handling to provide extensive acquisition, recording, and display of operating and experimental data. An estimated 3200 inputs will be scanned to provide data in support of operations and the experiments. Manual and analog control modes will be provided for initial operation of the facility with the capability for expansion to digital control at a later date. Emphasis has been placed on data handling and display to provide reliable test data and information to aid the operator. A plant simulation facility is included in the system to train operating personnel, check operational standards, and to perform studies and checkout of the CC&DH system.

The present conceptual design of the CC&DH system is based on the following technical considerations:

- Initial checkout and startup of the plant and experimental facilities will be made from the central control room.
- After initial startup, all control, data handling, and display will be performed by the operating personnel in the central control room.
- The plant protection system functions shall be independent of the CC&DH system functions.
- Sufficient analog displays and alarms will be provided for critical variables in event the computer directed displays fail.

- The CC&DH system shall integrate and coordinate all process and experimental reporting, and supervisory and control functions to ensure safe, continuous plant operation.
- No failure of either a computer or peripheral device shall require a plant shutdown.

Data will be acquired by the system through several devices including the input scanning system and manual entry devices such as keyboards, paper tape readers, and punched card readers. The selection of the multiplexer for input scanning is one of the critical items because of the variable scan rates and control of low level signals. A state-of-the-art survey indicates that the two types of low-level multiplexers which show the most promise are mercury-wetted reed relay and field effect transistor (FET) types. While these two types show the most promise from an operational standpoint, studies during preliminary design will also consider the application of both types on the basis of cost and detailed application.

A state-of-the-art survey⁽¹⁾ of computer-generated displays evaluated devices such as cathode ray tube displays, direct digital readouts, electroluminescent devices, and projection techniques. Cathode ray tubes appear best for FFTF use, based on the following considerations:

- Format flexibility to present data and operational aids in alphanumeric and graphic forms
- Simplification of the control room layout
- Fast acquisition of data from many sources, correlation, and presentation of the data to the operators.

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1. R. E. Mahan. A State-of-the-Art Survey of the Data Display Field, BNWL-725. Pacific Northwest Laboratory, Richland, Washington, May 1968.

A commercially available three-color (red, blue, and green) CRT display unit is being purchased for evaluation. This evaluation will study refresh and storage problems, selection of character and function generation, and the human engineering aspects to optimize the response of the operator.

Since the data handling and display portion of the CC&DH system is essentially computer-based, stored real time programs are required to gather, organize, coordinate and display data to the operators and experimenters. The real time programming design will have three major features:

- Conservative timing design to permit cycle stealing
- Modularity for flexibility in functional organization
- Memory protection in event of malfunction or failure.

Industrial experience with computer-based systems has indicated that the major difficulties in achieving a satisfactory design have been caused by problems in programing design. This is particularly true in the areas of executive program design. Hardware design generally precedes programing design. As a result, the programs have often been inadequate and impossible to implement with the hardware. To prevent this, programing design has been started early with particular emphasis on executive or monitoring programing design definition. Along with this, a programing design organization has been established at PNL, the center which has the most current information about the FFTF process systems. To ensure proper integration and verification of the programing design, the Architect-Engineer will monitor the efforts of the programing design staff.

The programing design will be done on a modular basis. The programs will be designed in order of their priority. Those required for minimum operation will be designed first followed by lower priority tasks.

To obtain independent appraisals of the conceptual design of the CC&DH system, two design study contracts were initiated with Autonetics Division of North American Rockwell Corporation and LTV Electro-Systems Incorporated. Autonetics has submitted their final report for PNL review, while LTV will submit their final report in September. The results of a third study, independently undertaken by Aerojet-General Corporation, will also be submitted in September. A summary of the three studies will be included in the next report.

b. Plant Instrumentation System

M. O. Rankin and C. R. F. Smith

The Plant Instrumentation System provides both parameter and system instrumentation and control for the out-of-vessel process systems, fuel handling systems, and the fuel examination cells. An estimated 250 control loops will be served by the Plant Instrumentation System.

State-of-the-art principles are being applied wherever possible in the design of the FFTF Instrumentation and Control Systems. However, special problems arise as a result of the environment in which the systems must operate. As a result, study programs and development tasks have been undertaken to solve these problems. In the following paragraphs, some of the problems and findings are described.

At present, magnetic flowmeters are being considered for flow measurements in the FFTF sodium process systems because of the relative advantages over conventional flow measuring devices. While much experience exists for the use of flowmeters up to 6-in. sizes, little experience has been acquired for large diameter flowmeters (14 to 24 in.) that may be required for the FFTF Sodium Process Systems. The uncertainties associated with the use of large flowmeters are related to the effects of nonuniform flow

velocities or calibration and the signal-to-noise ratio as a function of flowmeter-piping configurations. Work has been initiated to study these effects at the Liquid Metal Engineering Center (LMEC) and the data are presently being evaluated by PNL.

An assessment of possible in-line sodium impurity monitoring techniques such as the plugging meter, electrochemical oxygen meter, carbon meter, and hydrogen meter was made to determine if any additional development is required.

The plugging meter was found to be the best in-line method available at the present time since it avoids contamination problems inherent in sodium sampling and is more rapid in comparison with sampling techniques. A potential alternative to the plugging meter is the United Nuclear Corporation's electrochemical oxygen meter. However, further tests are required to resolve problems associated with thermal shock, time and temperature dependent effects, and polarization effects caused by sodium attacking impurities in the electrolyte.

United Nuclear Corporation has developed a potential technique for measuring the index of carburizing potential in sodium which is a function of the various carbide concentrations. Further evaluation and proof testing of this technique is necessary to determine the feasibility of this method.

Two experimental techniques are currently under development for in-line measurement of hydrogen in sodium. One technique involves the diffusion of hydrogen through a nickel membrane whose opposite face is plated with palladium to separate the hydrogen. The hydrogen is reacted with oxygen to form water with the concentration measured by the current

required for electrolysis. The other technique being developed by United Nuclear Corporation consists of an electrolytic cell arrangement which will produce an electromotive force signal as a function of hydrogen activity in sodium. Both techniques require additional testing to establish full feasibility.

c. Radiation Monitoring System

M. O. Rankin

The Radiation Monitoring System detects, warns, and initiates control actions when radiation levels or radionuclide concentrations exceed predetermined levels. To perform these functions, the following radiation monitoring subsystems are required:

- Airborne Particulate Monitors
- Laboratory Equipment
- Gamma Activity Monitors
- Liquid Monitors
- Radioactive Gas Monitors
- Criticality Monitors.

All of these subsystems except the airborne particulate monitor, laboratory equipment, and the criticality monitors are found as standard practice within the industry.

The handling of plutonium-bearing fuels imposes a unique design requirement on the airborne particulate monitors and is related to separate signals due to radon and thoron radioactivity which exceeds those from hazardous airborne plutonium alpha activity. Previous work at PNL has resulted in a separation technique to provide satisfactory detection levels. Separation of these signals can be accompanied by application of coincidence techniques, alpha energy analysis, rate of change techniques, subtraction

techniques and mechanical separation techniques. In the laboratory, automatic handling equipment and electronic equipment will be supplied to analyze filters for plutonium alpha, gross alpha, gross beta-gamma and detection of any one selected gamma emitter up to 2.5 MeV.

Neutron sensitive criticality monitors will be installed in all facilities which contain more than one-third of a minimum critical mass of fissionable materials. The criticality monitor and alarm system is powered from an emergency power supply. The criticality alarm system from each building is connected to the criticality annunciator at a central criticality monitoring station.

2. Reactor Instrumentation and Control

R. R. Cone

a. Reactor and Vessel Instrumentation System

W. Dalos

The Reactor Instrumentation System provides both instrumentation and control for inside-the-vessel systems. It includes all of the reactor temperature, pressure, and flow instrumentation, and the vessel and shield instrumentation. It provides the interface between the reactor and the Central Control and Data Handling System and the Plant Protection System. A survey⁽¹⁾ has indicated experimenters wish to measure and/or control fuel pin pressure, coolant pressure, fuel cladding and meat temperatures, subassembly vibration, inlet coolant temperature, sodium boiling, and in-core flux. Capability for making these measurements is being provided.

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1. J. J. Regimbal. Technical Basis for FFTF Open Test Position and Closed-Loop Instrumentation, BNWL-779. Pacific Northwest Laboratory, Richland, Washington, April 1968.

In addition: (1) open test assemblies will have the capability for installing a thimble to contain in-core flux measuring devices such as activation wires, balls, or traversing miniature chambers; and (2) sensors and leads for open test, and closed loop assemblies are being designed to facilitate interim examination of the test section and to permit removal and replacement of assembly without the loss of instrumentation.

A compilation of the preliminary requirements for the conceptual design of reactor core instrumentation is shown in Table 2.I.

Wherever possible, standard instrumentation will be used. A survey of the state-of-the-art for reactor core instrumentation has indicated the need for development of an in-core flowmeter and signal lead connector. The fabrication of coolant thermocouples for high temperature liquid metal service alone imposes no problem, but exposure to both the high temperature and radiation anticipated in the FTR core presents problems in obtaining thermocouples with known high reliability.

At present, ANL has undertaken an in-core flowmeter development to establish technology to ensure that magnetic eddy current or other type flowmeters can be obtained for use in the FTR core. Out-of-pile and EBR-II test programs have been established to determine the material and irreversible temperature effect of various magnetic materials. The design of a preliminary prototype permanent magnet flowmeter has been initiated. A high temperature version of an eddy current flowmeter is being fabricated and flowmeter characteristics in sodium will be studied.

ANL has also undertaken the development of a signal lead connector to operate in the FFTF environment. The fabrication

TABLE 2.I. Preliminary Requirements for Reactor Core Instrumentation

	Poison Rods	Driver	Open Test	Closed Loop	For Bulk Measurement (in some removable reflectors)
<u>Temperature Monitoring</u>					
No. Coolant Inlet TC's/assy	0	0	0	3	8 (2 at 4 locations)
No. Coolant Outlet TC's	2	5	5	3	8 (2 at 4 locations)
No. Cover Gas TC's	N.A. (a)	N.A.	N.A.	N.A.	4 (1 ea. in 4 places)
No. Bulk Outlet TC's	N.A.	N.A.	N.A.	N.A.	8 (2 at 4 locations)
TC Response, max	2 sec (1 sec if 1/16 in. size)	1 sec	1 sec	1 sec	2 sec (1 sec if used for protective circuit)
TC Type	Cr/Al	Cr/Al	Cr/Al	Cr/Al	Cr/Al
TC Sheath OD, max	1/16-1/8 in.	1/16-1/8 in.	1/16-1/8 in.	1/16-1/8 in.	1/8 in.
TC Insulation	MgO	MgO	MgO	MgO	MgO
Mixer in Assembly	No	Yes	Yes	No	No
Accuracy (TC only)	±1%	±1%	±1%	±1%	±1%
Sheath Material	Incoloy 800	Incoloy 800	Incoloy 800	Incoloy 800	Incoloy 800
<u>Flow Monitoring</u>					
No. of Sensors	1 ^(b)	1	1	1 (For By- pass flow)	0
Type of Sensor	-	Magnetic	Magnetic	Magnetic	-
Repeatability, max	-	±2%	±2%	±5%	-
Accuracy	-	±10%	±10%	±15%	-
Response, max	-	0.1 sec	0.1 sec	0.1 sec	-
Sensor Length, max	-	12 in.	12 in.	6 in.	-
Flow Straightener	-	Yes	Yes	No	-
Sensitivity	-	1% of full flow	1% of full flow	-	-

a. Not Applicable

b. Either flow sensor or pressure sensing (see Pressure Monitoring).

TABLE 2.I. (contd)

	Poison Rods	Driver	Open Test	Closed Loop	For Bulk Measurement (in some removable reflectors)
<u>Pressure Monitoring</u>					
No. for Inlet Coolant	1 ^(a)	0	1 (if needed)	1	3 (1 at each of 3 locations)
No. for Outlet Coolant	1 ^(a)	0	1 (if needed)	1	3
No. for Hyd. Holddown (if used)	0	0	0	0	3
No. for Fission Gas Pressure	0	0	1 (if needed)	1 (if needed)	0
Coolant Pressure Bellows, diam, max	-	-	1 in.	1 in.	3 in.
Fission Gas Bellows, diam, max	N.A.	-	1/2 in.	1/2 in.	-
Coolant Pressure Accuracy	-	-	±3%	±3%	±3%
<u>Coolant Level Monitoring</u>					
No. Operating Level Sensors	0	0	0	0	3 (may be in thimble through vessel top)
No. Drain-Fill Sensors	0	0	0	0	1 (18 TC's in 1 reflector)
Op. Level Type	N.A.	N.A.	N.A.	N.A.	Induction
Op. Level Accuracy	N.A.	N.A.	N.A.	N.A.	±1/2 in.
Drain-Fill Accuracy	N.A.	N.A.	N.A.	N.A.	±3 in.
Op. Level Response	N.A.	N.A.	N.A.	N.A.	0.5 sec
<u>Rupture Location</u>					
Fission Gas Sampling in Assembly	No	Yes	Yes	No	No
Na Trap in Assembly (to remove Na from gas sample lines)	No	Yes	Yes	No	No
Na Sampling in Assembly ^(b)	No	No	No	No	No

a. Alternate for flow sensor. Delta P would be measured across orifice.

b. Not desirable and it is the last choice for failed fuel location, but it may be necessary for driver and open test assemblies if other methods (gas sampling, etc.) cannot be used.

TABLE 2.I. (contd)

	Poison Rods	Driver	Open Test	Closed Loop	For Bulk Measurement (in some removable reflectors)
<u>Flux Measurement</u>					
Thimbles Initially Installed in Assembly	No	No	Yes (2)	No ^(a)	Yes (6)
Thimble ID Capability	-	-	3/8-1/2 in.	-	3/8-1/2 in.
<u>Capabilities (Cable, Conn., etc.)</u>					
Coolant Flow Control	No	No	Yes	No	No
Contact Sensors (TC's on Fuel, etc.)	No	Yes ^(b)	Yes	Yes	No
Future Flux Thimble	No	Yes ^(b)	Yes	Yes	Yes
Vibration and Strain	No	Yes ^(b)	Yes	Yes	Yes
<u>Nozzle Signal Connector</u>					
No. Total Pins	36	36	36	36	36
No. Gas Lines	0	2 (gas sample)	3 ^(c)	1 (fuel pin P)	0
Flux Thimble ID, max	-	-	1/2 in.	1/2 in.	1/2 in.
Shell Diameter, max	3 in.	3 in.	3 in.	3 in.	3 in.
Remote Connection and Disconnection	Yes	Yes	Yes	Yes	Yes
Min No. Pins for Experimenter	-	-	16	16	-
Sodium Sample Through Conn. ^(d)	No	No	No	No	No

- a. Could install flux wires in closed loop subassemblies without thimbles, if desired and if planned during fabrication.
- b. No installation planned for driver assembly. But capability exists because open test assemblies must have this capability and driver and open test assemblies are to be interchangeable as to position.
- c. One for fission gas pressure, two for gas sampling for failed fuel location.
- d. Not desirable and it is the last choice for failed fuel location, but it may be necessary for driver and open test assemblies if other methods (gas sampling, etc.) cannot be used.

and assembly of a driver channel coupling device has been completed and the checkout in a high temperature gas environment initiated.

The evaluation of thermocouples for the FTR is being carried out by a development activity at PNL.

b. Fuel Failure Monitoring System

R. R. Cone, W. Dalos, and C. R. F. Smith

This system will provide equipment which will detect and locate a fuel cladding failure. Detection of a cladding failure is necessary to prevent any further subassembly damage. The capability to quickly and accurately identify the subassembly having the failed fuel pin is needed to prevent reducing reactor availability.

Failure detection will be accomplished in the FTR reactor by detecting fission products in both the cover gas and the reactor coolant using techniques similar to those at EBR-II. EBR-II draws a continuous sample of the cover gas through traps and other sample conditioning equipment and deliver the gas to a charged wire type detector which collects the positively charged radioactive daughters of the xenon and krypton fission products.

Fission products in the reactor coolant are detected by a delayed neutron monitoring system in which a portion of the coolant flow is diverted to the detector and then returned to the loop. This detection system consists of moderating material, several thermal neutron detectors, and preamplifiers. Unlike the EBR-II the application of these systems to the FTR must consider a lower neutron-to-gamma ratio and a higher temperature environment. Development efforts are under way to meet these more stringent conditions.

Fuel failure location presents a more difficult problem since no such system is used in any present sodium

cooled reactor. The major problem is to obtain an adequate sample of the coolant or fission gas from each subassembly and transfer this sample to the detector location without sample line plugging from cooling the sodium or oxide formations.

A firm decision has not yet been made on the type of fuel failure location system. However, for the conical core concept, a method of separating fission gas from the sodium within the reactor and delivering this sample to detectors located in adjacent shielded cells shows the most promise. The following methods are also under study and are being investigated through development efforts:

- Sodium sampling with delayed-neutron detection
- Sodium sampling with fission gas separation and detection of fission products outside of reactor
- Coolant or fission gas sample grabbing for laboratory analysis
- Fission gas separation and detection within the subassembly
- Detection of fission gas bubbles by means of the sub-assembly magnetic flowmeter
- Xenon isotope tagging in the fuel pin.

c. Flux Monitoring and Control System

L. W. McClellan

The Flux Monitoring and Control System provides the instrumentation and control associated with the reactor system and comprises the safety circuit and control instrumentation subsystem and the in-core flux instrumentation subsystem.

(1) Safety Circuit and Control Instrumentation Subsystem.

The safety circuit and control instrumentation consists of the sensors, electronic circuitry and associated equipment needed to:

- Monitor the gross neutron flux
- Initiate protective system actions

- Control and indicate rod position
- Estimate reactivity.

The neutron sensors, located external to the vessel, cover the entire operating power level range (10 to 11 decades) of the reactor.

The safety circuitry will comprise three low-level safety channels using pulse chambers, three intermediate-level safety channels using compensated ionization chambers, and four high-level safety channels using compensated or uncompensated chambers. The safety circuitry will be independent of the control sensors and circuitry and will automatically provide trips to the Plant Protection System when predetermined limits in flux level, flux rate of change, or reactor period are exceeded. The Plant Protection System will provide the logic system to initiate a scram signal to the rods.

The control circuitry will comprise three high-level control channels for manual and automatic control, and two wide range linear power channels for manual and automatic control. If multidecade continuous automatic control is required, two or three logarithmic channels may be provided to obtain an average control signal. The control circuitry will provide a reactant for such variables as flux level, flux rate of change and period for manual control. Provisions will also be made to control the reactor power level using closed loop control of selected regulator rods. The application of automatic closed loop control, however, will be made only after its feasibility has been established by simulation studies and actual test operation. The control circuitry will also respond to controlled power reduction or setback commands from the Central Control and Data Handling System and initiate the required command signal to the rod drive system.

All rod positions will be indicated continuously to monitor the reactivity status of the reactor. The rod position sensor will be an integral part of the rod drive system. The control circuitry channels will also provide input signals to the malfunction detection or anomalous reactivity detection program provided by the Central Control and Data Handling System.

One of the most difficult problems in determining reactor gross flux level by measuring the leakage flux in the shield is the effect of the inherent gamma background on the instrumentation. In the most limiting situation this gamma background is not in proportion to the reactor power level because of the following possible contributions from other gamma sources:

- Neutron-gamma reactions in materials near the detector
- Decay of ^{22}Na and ^{24}Na
- Fission product decay
- Decay from activation of structural materials near the detector positions
- Decay from the possible activation of detector materials.

Preliminary calculations indicate a high gamma background together with a low thermal neutron flux creates the most severe neutron detection problem, particularly after the fission product buildup has occurred. It has been estimated for the FTR that a gamma background approaching 1×10^5 R/hr will exist for a minimum thermal neutron flux of 1 to 10^1 nv. At the present time developmental evaluation and testing of detectors, electronic components, and instruments for the FFTF low-level neutron startup channel are being pursued by PNL.

III. GENERAL TECHNOLOGY

P. L. Hofmann

HIGHLIGHTS OF DESIGN EVALUATION

L. M. Finch

Early in 1967, the split conical core configuration was chosen as the reference concept for the FTR. The selection of a core arrangement that differed from the typical LMFBR vertically oriented cores was based upon the superior access for irradiation tests in the dispersed conical geometry; however, to ensure project continuity in the event the dispersed core proved unsuitable, a vertical core arrangement was selected as a backup conceptual design. A design review team met in December, 1967, and January, 1968, to review progress on the FFTF; particularly, design progress on the split conical and the vertical backup cores. The design review team concluded that design progress for the reference core was deficient in meeting these requirements:

1. Neutron flux levels greater than 7×10^{15} n/cm²-sec require compact core geometries. Core space in the split zone of the split conical core must be utilized efficiently for tests or fuel if flux requirements are to be met.
2. Requirement for high degree of interchangeability for core components required.
3. Core subassembly length must be as short as possible.
4. Preferred fuel burnup, power density, and location of nuclear control rods (such as reflector control) must be treated as secondary requirements to the testing requirements. They may be less than optimum if testing capability is enhanced thereby.
5. Refueling methods must be simple, assured, and demonstrated, by experiment if necessary.

6. The performance of the FTR must be studied for a wide range of testing conditions, including the performance expected with advanced core and fuel design.
7. Core radial restraint methods are not sufficiently developed.

Concurrently, the design review team offered the following criticism of the backup core design progress.

1. Design exhibits insufficient commonality with reference design with respect to the following design considerations:
  - a. Number and size of closed loops
  - b. Driver fuel and open loop instrumentation
  - c. Control rod worth and control margins
  - d. Refueling safeguards
2. Severe technical problems were identified with instrumentation installation, control rod drives, and closed loop piping connections.
3. B_4C may not be a suitable control rod material.
4. The interfaces of the interchangeable backup core with the reference plant were not well defined.

Two task forces were formed to permit intensive effort upon core design to correct the deficiencies noted in the design review report. Task force efforts for improving the performance of the reference core were successful principally by the modification of the conical core geometry from the split configuration to an annular configuration. The fuel handling problem is the subject of continuing conceptual design work. The task force efforts for the backup core did result in a high degree of commonality with the reference core design and in well defined interfaces with the FTR. Task force work upon the backup core technical problems of instrumentation arrangement, control rod drives and closed loop piping confirmed that these were problems that will require careful and ingenious design; the task force developed feasible design concepts for each problem.

The Dispersed Core Task Force work was headed by D. L. Condotta (currently Manager, FFTF Development Department). Contributions were made by members of all FFTF departments. Results of this work are summarized in Section A.1.

The vertical core task force was led by E. M. Johnston, (currently Assistant Department Manager, Project Engineering). Contributions were made by members of all FFTF departments. Results of this work are summarized in Section A.2.

III. GENERAL TECHNOLOGY

P. L. Hofmann

A. DESIGN EVALUATION

L. M. Finch

1. Reactor Evaluation Dispersed Core Task Force Activities

(This work done for System 30. Reactor)

D. Marinos

a. Potential Improvements in the Bisected Core

G. R. Waymire

(1) Introduction. Previous work on the FTR reference design, the split conical configuration, indicated significant difficulty in achieving a tightly packed core with the design criteria proposed. Changes to these design criteria, however, warranted an additional study. Major criteria changes affecting previous studies included:

<u>Characteristic</u>	<u>Previous Bases</u>	<u>Recent Change</u>
Number of Vertical Positions Required	5	Three minimum
Clearance for Closed Loop Piping	2 to 3 in. corridor	Normal clearance between cover nozzles
Tubesheet Location	54 in. below the core	36 in. below the core
Minimum Fuel Pin Spacing	0.03 in.	0.04 in.

The first three items tend to improve the core packing efficiency while the fuel pin spacing significantly decreases the packing efficiency. Since the criteria changes primarily involved the packing potential in the split gap, additional work was concentrated in this area of interest.

(2) Requirements and Criteria. To ensure common bases with other candidate core configurations, common criteria for studying split gap utilization were adopted.

These criteria included:

- Components for utilization in the gas should be easily fabricated utilizing standard techniques.
- Components must be handled by standard techniques utilized for the remainder of the core components. This requirement does not preclude the use of adapters to the basic fuel handling scheme.
- Gap components need not conform to a standard design. The number of different configurations, however, should be minimized.
- Fuel design (for gap components) should be consistent with the bases established for the driver fuel region.

(3) Evaluation of Split Conical Core (Reference as of January 1968). To achieve a core packing consistent with the minimum total flux goal of 0.7×10^{16} nv, "the packing efficiency" for the split core must be significantly increased. The increased pin spacing (minimum increased from 0.030 to 0.040 in.) results in a degradation of flux below 0.5×10^{16} n/cm²-sec. The opportunity for improvement lies in the utilization of the large gap around the vertical closed loops. Several options exist for utilizing this gap; these include:

- Rearrange the core. Erect a minimum of three vertical loops and place fueled positions around these vertical positions.
- Decrease the gap width: Utilize the reference concept and attempt to reduce the split width, utilizing the criteria changes noted.
- Utilize the gap for other purposes: Trade-off fueled for unfueled positions in the core for fuel by placing the unfueled components in the gap region.

(4) Core Rearrangement. Earlier parametric studies,⁽¹⁾ indicate that the greatest improvement in centerpoint flux is obtained by utilizing the central gap for driver fuel. Before arriving at a feasible design, the maximum theoretical potential for flux improvement by core rearrangement was investigated utilizing the above observation as a starting point. To evaluate a practical degree of gap utilization against the theoretical potential, a brief investigation was performed on how to construct fueled positions practically. The study was then completed in two phases.

The first phase was an analysis for the central flux assuming the gap was filled with fuel to an equivalent density consistent with the 0.03 in. spaced reference fuel design. As shown in Figure 3.1, if the gap could be filled, the central flux would increase from about 0.53×10^{16} to about 0.66×10^{16} n/cm²-sec. As a matter of practical design, however, an 80% utilization appeared to be a maximum value, with 60% being a more realistic design.

For the second phase, design layouts were constructed to determine precisely what theoretical fuel density could be achieved with the 0.04 in. fuel pin spacing. These layouts indicated that about 1140 pins could be placed in the defined envelope of the split zone, or the equivalent of 5.25 additional driver assemblies. These added assemblies which were assumed to replace lower worth positions in the core periphery, gave a maximum theoretical flux of 0.56×10^{16} n/cm²-sec. Since the theoretical maximum utilization fell short of the required

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1. L. M. Finch, R. E. Peterson, and G. R. Waymire. Geometry Problems Encountered in Design of the FFTF Split Conical Core, BNWL-SA-1323. Pacific Northwest Laboratory, Richland, Washington, November 1967.

Total Void Space	98.02 liter
Maximum Filler Fuel	73.3
Probable Volume	58.4
Driver Fuel	943
Test Volume	68.9

Driver Fuel Center Void
 Total Core Volume $\neq$ Constant
 Total Driver Fuel = Constant

FIGURE 3.1. Gap Utilization - Theoretical Study

minimum flux requirement (0.7×10^{16} n/cm²-sec) by 20%, further efforts in rearrangement of the split core were curtailed.

(5) Gap Closure. This study involved moving the two conical halves of the core inward until a mechanical interference developed. Core components were altered to correct this interference and split zone closure continued until another interference developed. The study was initiated with the five vertical loop reference design. With this design, interference occurs first at the reactor cover between the intermediate loops and adjacent driver fuels. The 5-loop cover configuration is shown in Figure 3.2. Average core gap at this point has increased by 1/4 in. Elimination of the two vertical intermediate loops fulfills the minimum vertical position criteria and allows the gap closure to continue.

Continued movement of the core halves results in an interference at the two outer vertical positions. Since the minimum number of vertical loops has now been reached, further closure requires outward radial shifting of these outboard vertical positions. Lateral movement continues until two adjacent drivers contact as shown in Figures 3.3 and 3.4. To avoid interference during insertion of positions, the projections of nonvertical drivers (see dashed projection lines, Figure 3.3) have forced the outer vertical loops into the radial reflector region.

As a result of this gap closure exercise, the resulting core has only one vertical loop with a flux improvement of about 10%.

(6) Non-fuel Gap Utilization. Since the most promising gap utilization, i.e., filled with fuel, does not attain the minimum flux goal, identification of potential nonfueled core components that could be placed in the gap was undertaken.

Plan at (+) 28 ft 0 in.
Scale: 1 1/2 in. = 1 ft 0 in.

FIGURE 3.2. Five Loop Minimum Clearance Design

3.7

Note: Loop position outside core

BNWL-880

FIGURE 3.3. Three Loop Design at Bottom of Core

FIGURE 3.4. Three Loop Design at Cover

Possible nonfueled uses include:

- Downcomers for U-tube type vertical loops (also considered in relation to closing the gap.)
- Structural material (such as materials to improve Doppler coefficient)
- Transposition of control rods from the driver region into the gap
- Additional test space such as rabbit or trail cable facilities.

(7) Conclusions. In addition to improvements which would apply to other conic geometries such as a tapered fuel bundle, the best potential for improving the bisected core involves more efficient utilization of the gap region in the core. The largest flux improvement results when the split region is utilized for driver fuel. A maximum flux of 0.56×10^{16} n/cm²-sec may be attained by practically utilizing a major portion of the gap packed with fuel at a density equivalent to a driver position. Since this value is 20% below the required minimum, it may be concluded that it is virtually impossible to achieve the flux minimum requirement with the bisected core. Even with the previous 0.03 in. spacing, the maximum flux is 11% below the required minimum.

b. Comparison of Alternative Core Concepts

G. R. Waymire

(1) Introduction. In accomplishing its objectives, the Improved Dispersed Core Task Force evaluated characteristics of three basic types of dispersed core geometries with variations. The evaluation served to (a) identify any geometric possibilities not previously considered for improving core performance (b) confirm assessments of the reference core design, and (c) identify the most promising dispersed core concept.

This discussion is concerned primarily with the mechanical design considerations of the core. Basically, the goal is to achieve a minimum core dispersion while maintaining access for refueling, instrumentation, and tests to each lattice position in the core. Desired trends in dispersed core configurations have also been discussed.⁽¹⁾

(2) Design Criteria. A set of common design criteria was established as a guide for concept development. Since it was evident that the criteria were somewhat contradictory, inability of a design to satisfy the initial criteria in some degree was anticipated. Each concept was developed in a manner to exploit its advantages, and the criteria violation was noted.

Criteria for the improved reference core for the FFTF were based on those characteristics essential to meeting project objectives for an irradiation testing facility for LMFBF fuels and materials. The characteristics of prime importance are (a) available fast flux delivered to adequately sized and controlled closed loops, (b) operation of these loops at appropriate thermal conditions, (c) plant availability, and (d) sufficient instrumentation to ensure unambiguous test results. Acceptable safety characteristics are mandatory. Of lesser importance is the criterion for design flexibility to accommodate changing fast flux test missions.

Nontechnical criteria of major significance are the availability of FFTF to initiate testing within prescribed schedules and the cost of the facility. The specific criteria utilized for concept selection are listed in Table 3.I, p. 3.23.

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1. L. M. Finch, R. E. Peterson and G. R. Waymire, Geometry Problems Encountered in Design of the FFTF Split Conical Core, BNWL-SA-1323. Pacific Northwest Laboratory, Richland, Washington, November 1967.

(3) Specific Designs Investigated. Three basic dispersed geometries and variations on each, were investigated, as shown in Figure 3.5. Although some flexibility exists for selecting the numbers of vertical closed loops, the various geometries are compatible with the range of vertical loops noted below:

<u>Variation</u>	<u>Number of Vertical Positions</u>
True Array (Skew and Cone)	1
Annular Array (Skew and Cone)	3
Split Arrays (Skew Cone and Wedge)	4 to 7

Specific designs are discussed in the following sections. A brief performance summary is included with each design.

(4) Conic Geometries. The true cone violates a basic criterion by having only one vertical position. The design shown in Figure 3.6 utilizes a 10 in. lattice spacing at the top cover. Routing of closed-loop piping with this design is difficult; however, this is true of most of the conic designs.

(a) Annular Cone. The annular cone meets the minimum requirement for vertical test positions with a minimum loss of flux by utilizing the unused volume on a "shared" basis, as shown in Figure 3.7. The central position must be eliminated from this design to allow adequate room at the cover for piping connections to the three vertical loops. The basic design for an annular cone is shown in Figure 3.8. The design shown is feasible but represents the minimum spacing which can be tolerated for above-the-cover fuel handling and closed-loop pipe routing.

(b) Bisected Cone. Task Force effort on the bisected design concentrated on reducing the unused volume in the gap

FIGURE 3.5. *Dispersed Geometries Investigated*

PERFORMANCE AND DESIGN TABULATION

CLOSED LOOP TESTING:

POSITION NOTED ON DRAWING FACE	TEST SECTION ID	LOOP TILT ANGLE	UW
1	2.6"	0	1.6
2		4.32°	
3		4.32°	
4		2° 16'	
5		2° 16'	

SUBASSEMBLY CHARACTERISTICS-SEE SK-3-14269
 CONTROL ROD CHARACTERISTICS-SEE SK-3-14297
 OPEN TEST CHARACTERISTICS-SEE SK-3-14303
 CLOSED TEST CHARACTERISTICS-SEE SK-3-14303

LATTICE ALLOCATION (SEE MID-CORE SECTION FOR SPECIFIC POSITIONING)

- DRIVER SUBASSEMBLIES—72
- IN-CORE CONTROL RODS—6
- OPEN TEST POSITIONS—8
- CLOSED LOOPS—5
- REFLECTORS/MATERIAL TESTS (2 + 42 = 54)
- REFLECTOR CONTROLS—24

PERFORMANCE CHARACTERISTIC:

CENTER POINT FLUX AT 400 MW: 0.69×10^{18} NV.
 PU₂₃₉ LOADING 587 KG.
 INCLINATION PER ROD — 1° 9'
 COOLANT AT 400 MW(1) & 65 PSI DROP THROUGH CORE 360°.

VOLUMES AND VOLUME FRACTIONS:

FUEL LENGTH REQUIRED FOR 7.3 KW/FT AVERAGE
 LINEAR POWER — 35.5 INS.
 VOLUME/DRIVER LATTICE POSITION — 12.05 L

OTHER NA FRACTION/DRIVER POSITION — 18.5 L
 TOTAL CORE VOLUME — 1096 L
 TOTAL OTHER NA — 19.4 L

REMARKS:

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SK-3-14137

FIGURE 3.6. True Conic Design

OTHER N_a VOLUME
ASSOCIATED WITH
DISPERSION OF A
SINGLE LATTICE
POSITION

ADDITIONAL OTHER
 N_a FOR A VERTICAL
PLANAR ARRAY OF
CLOSED LOOPS

SHARED OTHER N_a
FOR BASIC DISPERSION

IDEALIZED VERT.
PLANAR ARRAY

IMPOSSIBLE TO ACHIEVE
DUE TO INTERFERENCE
WITH NEIGHBORING
DRIVER POSITIONS

FEASIBLE CLUSTER
ARRANGEMENT

FIGURE 3.7. Means of Arranging Vertical Loops

PERFORMANCE AND DESIGN TABULATION

CLOSED LOOP TESTING

POSITION NOTED ON DRAWING FACE	TEST SECTION ID	LOOP TILT ANGLE	WW
1	2.6	0°	1.8
2		0°	
3		0°	
4		3° 44'	
5		3° 44'	

DRIVER ASSEMBLY SEE SK-3-14184 SK-3-14269
 CONTROL ROD—SEE SK-3-14257
 OPEN TEST—SEE SK-3-14270 SK-3-14313
 CLOSED TEST—SEE SK-3-14353

LATTICE ALLOCATION (SEE MID-CORE SECTION FOR SPECIFIC POSITIONING.)

DRIVER SUBASSEMBLIES	71
IN-CORE CONTROL RODS	8
OPEN TEST POSITIONS	8
CLOSED LOOPS	5
REFLECTORS/MATERIAL TESTS	54
REFLECTOR CONTROLS	24

PERFORMANCE CHARACTERISTICS

CENTER POINT FLUX AT 400 MW: 0.370 X 10¹⁸ NV
 P₂₃₈ LOADING: 874 KG
 INCLINATION PER ROW: 0° 56'
 COOLANT AT 400 MW: 1.6 PS: DROP THROUGH CORE: 3.80°

VOLUMES AND VOLUME FRACTIONS

FUEL LENGTH REQUIRED FOR 5.3 KW/FT AVERAGE
 LINEAR POWER: 36 IN.
 VOLUME/DRIVER LATTICE POSITION: 11.7 %
 OTHER NA FRACTION/DRIVER POSITION: 16.6 %
 TOTAL CORE VOLUME: 1064 %
 TOTAL OTHER NA: 16.3 %

REMARKS

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3.15

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SK-3-14306

FIGURE 3.8. Annular Conic Core

region. The goal was to determine if a split cone had the potential for achieving the minimum required flux. Several alternatives were considered for utilization of the central gap of the reference bisected core as noted in the preceding section.

(c) Trisected Cone. The trisected design is actually a variation of the bisected concept. Its chief advantages lie in its more symmetric pattern and its ability to accommodate a larger variation in the number of vertical test positions (4 to 7). In splitting the cone three ways, however, additional leakage is generated which reduces the flux to a value slightly below the bisected design. A representative trisected design is shown in Figure 3.9.

(5) Wedge. The unique feature of the wedge geometry is that it disperses in only one plane. This feature results in an elongated core section. The gap around core subassemblies at any point is also elongated with the maximum elongation occurring along the two sides perpendicular to the plane of dispersion. The gap along the other four sides is smaller and produces a higher overall packing efficiency for a given dispersion rate than do comparative conical designs. The lattice and spacing at the cover nozzles; however, is not equilateral. The dispersion in only one plane also reduces the gap at the vertical slab section for a given nozzle clearance at the top cover. This reduced gap is the result of the single angle of dispersion away from the vertical plane. (The conical designs with angular dispersion components in two planes requires an additional central zone gap to provide clearance for the dispersion component nearly parallel to the edge of the vertical plane at the split zone.)

Disadvantages for the wedge include its nonsymmetric subassembly cross section at support pads and possible problems

PERFORMANCE AND DESIGN TABULATION

CLOSED LOOP TESTING:

POSITION NOTED ON DRAWING FACE	TEST SECTION ID	LOOP TILT ANGLE	MW
1	2.6	0°	1.8
2		0°	
3		0°	
4		0°	
5		5.33°	

SUBASSEMBLY CHARACTERISTICS-SEE SK-3-
CONTROL ROD CHARACTERISTICS-SEE SK-3-
OPEN TEST CHARACTERISTICS-SEE SK-3-
CLOSED TEST CHARACTERISTICS-SEE SK-3-

LATTICE ALLOCATION (SEE MID-CORE SECTION
FOR SPECIFIC POSITIONING)

○ DRIVER SUBASSEMBLIES	77
○ IN-CORE CONTROL RODS	6
○ OPEN TEST POSITIONS	8
○ CLOSED LOOPS	5
○ REFLECTORS/MATERIAL TESTS	22
○ REFLECTOR CONTROLS	24

PERFORMANCE CHARACTERISTIC:

CENTER POINT FLUX AT 400 MW: 0.54×10^8 NV
 P_{238} LOADING 597 KG.
 INCLINATION PER ROW 0° 58'

VOLUMES AND VOLUME FRACTIONS:

FUEL LENGTH REQUIRED FOR 7.8 KW/FT AVERAGE
 LINEAR POWER: 32.0'
 VOLUME/DRIVER LATTICE POSITION: 10.4
 DRIVER SECTION VOLUME: 92.0
 OTHER NA FRACTION/DRIVER POSITION: 10.7
 TOTAL CORE VOLUME: 104.4
 TOTAL OTHER NA: 14.6

REMARKS:

CONCEPTUAL 30% 155/112, 6-12-68

SK-3-12883

FIGURE 3.9. Trisected Cone Design

with application of radial restraint loads. The core is also nonaxisymmetric and the angular dispersion penalty is greater for a given center-to-center spacing at the cover nozzles. A preferred wedge design is shown in Figure 3.10.

(6) Skewed Geometries. The skewed cores may most easily be visualized by a bundle of pencils which are skewed by rotating the two ends in opposite directions. The twisting results in a 3-dimensional dispersion of the elements in the bundle. By placing the reactor core center at the pinch plane of such an array, a minimum dispersed core geometry results.

Chief advantages for the skew include its higher packing efficiency and the increased dispersion below the core which gives more space at a lower tubesheet and eliminates the safety concern with slumping fuel in a convergent geometry. In addition, the skew packing characteristic is less sensitive to changes in lattice spacing at the top cover.

Primary disadvantages for the skew lie in the elliptical duct cross section at any horizontal plane. Each row of subassemblies cut by a plane perpendicular to the core vertical axis has a different elliptical shape. This implies differences in contact pad geometry for each row of subassemblies.

(a) True Skew. If one vertical position were adequate, the true skew would offer the highest flux of any of the dispersed geometries investigated. Figure 3.11 shows the basic true skew design.

(b) Annular Skew. Because of the 3-dimensional dispersion of the skew, an annular configuration with three vertical closed loops requires that the central cluster of seven lattice positions be removed. The resultant geometry is shown in Figure 3.12.

PERFORMANCE AND DESIGN TABULATION

CLOSED LOOP TESTING:

POSITION NOTED ON DWG FACE	TEST SECTION ID	LOOP TILT L	MW
1	2.6	0°	1.6
2		0°	
3		0°	
4		2° 25'	
5		2° 25'	

SUBASSEMBLY CHARACTERISTICS-SK-3

CONTROL ROD CHARACTERISTICS-SK-3

OPEN TEST CHARACTERISTICS-SK-3

CLOSED TEST CHARACTERISTICS-SK-3

LATTICE ALLOCATION:

(SEE MID-CORE SECTION FOR SPECIFIC POSITIONING)

○	= DRIVER SUBASSEMBLIES	72
○	= IN-CORE CONTROL RODS	6
○	= OPEN TEST POSITION	8
○	= CLOSED LOOPS	5
○	= REFLECTORS/MATERIAL TESTS	10
○	= REFLECTOR CONTROLS	24

PERFORMANCE CHARACTERISTIC:

CENTER POINT FLUX @ 400 MW O.B. $\times 10^{16}$ NV
P₂₃₅ LOADING 565 KG
INCLINATION PER ROW — 1.34° (UNIDIRECTIONAL)

VOLUMES AND VOLUME FRACTIONS:

FUEL LENGTH REQUIRED FOR 7.5 KW/FT
AVERAGE LINEAR POWER 34.4 INS
VOLUME/DRIVER LATTICE POSITION: 10.23 %
DRIVER SECTION VOLUME: 9500 %
OTHER NA FRACTION/DRIVER POSITION: 8 %
TOTAL CORE VOLUME: 1038 %
TOTAL OTHER NA: 9.8 %

REMARKS:

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FIGURE 3.10. Wedge Core Configuration

PERFORMANCE AND DESIGN TABULATION

CLOSED LOOP TESTING:

POSITION NOTED ON DWG FILE	TEST SECTION ID	LOOP T/L L	HW
1	2.6"	0	1.6
2		3'54"	
3			
4			
5	2.6"	3'54"	1.6

SUBASSEMBLY CHARACTERISTICS - SK-3-

CONTROL ROD CHARACTERISTICS - SK-3-

OPEN TEST CHARACTERISTICS - SK-3-

CLOSED TEST CHARACTERISTICS - SK-3-

LATTICE ALLOCATION:

(SEE MID-CORE SECTION FOR SPECIFIC POSITIONING)

- = DRIVER SUBASSEMBLIES - 72
- ⊖ = IN-CORE CONTROL RODS - 6
- = OPEN TEST POSITION - 8
- ⊖ = CLOSED LOOPS - 5
- ⊖ = REFLECTORS/MATERIAL TESTS - 54
- = REFLECTOR CONTROLS - 24

PERFORMANCE CHARACTERISTIC

CENTER POINT FLUX @ 400 MW 0.78×10^{14} NV
 P_{425} LOADING 518 KG
 INCLINATION PER ROW 1°57'

VOLUMES AND VOLUME FRACTIONS:

FUEL LENGTH REQUIRED FOR 73 KW/FT
 AVERAGE LINEAR POWER: 35.5 MW
 VOLUME/DRIVER LATTICE POSITION: 10.2 2

OTHER NA FRACTION/DRIVER POSITION: 3.5%
 TOTAL CORE VOLUME: 920.2%
 TOTAL OTHER NA: 3.5%

REMARKS:

- 0.007" NOMINAL CLEARANCE IS INCLUDED TO PREVENT INTERFERENCE BETWEEN ADJACENT SKEW ELEMENTS.

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FIGURE 3.11. True Skew Core

PERFORMANCE AND DESIGN TABULATION

CLOSED LOOP TESTING:

POSITION NOTED ON DRG FACE	TEST SECTION ID	LOOP TILT Z	MIN
1	2.6"	0	1.6
2		0	
3		0	
4		6° 56'	
5	2.6"	6° 56'	1.6

SUBASSEMBLY CHARACTERISTICS- SK-3

CONTROL ROD CHARACTERISTICS- SK-3

OPEN TEST CHARACTERISTICS- SK-3

CLOSED TEST CHARACTERISTICS- SK-3

LATTICE ALLOCATION:

(SEE MID-CORE SECTION FOR SPECIFIC POSITIONING)

- = DRIVER SUBASSEMBLIES - 68
- = IN-CORE CONTROL RODS - 6
- = OPEN TEST POSITION - 8
- = CLOSED LOOPS - 5
- = REFLECTORS/MATERIAL TESTS - 54
- = REFLECTOR CONTROLS - 24

PERFORMANCE CHARACTERISTIC:

CENTER POINT FLUX @ 400 MW: 0.68 x 10¹⁹ NV
P_u 233 LOADING: 535 KG
INCLINATION PER ROW: 1° 44'

VOLUMES AND VOLUME FRACTIONS:

FUEL LENGTH REQUIRED FOR 7.6 KW/FT
AVERAGE LINEAR POWER: 36 INS.
VOLUME/DRIVER LATTICE POSITION: 10.35 L
OTHER NA FRACTION/DRIVER POSITION: 4 %
TOTAL CORE VOLUME: 942 L
TOTAL OTHER NA: 0.2 %

REMARKS:

* 0.007" NOMINAL CLEARANCE INCLUDED TO
PREVENT INTERFERENCE BETWEEN ADJACENT
SKEW ELEMENTS.

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FIGURE 3.12. Annular Skew Core

The increased unused space around vertical positions, compared with the annular conic core, offsets the improved skew packing efficiency in the driver portion of the core.

(7) Core Evaluation and Selection. Several other core designs were considered but appeared to be beyond the state-of-the-art or had little or no advantage over the more conventional approaches. The additional designs are listed below with a brief explanation for their elimination.

Other Configurations Considered

Dispersed geometry deflected into a vertical array in the core region.	Excessive duct stresses under deflection.
Bent tube array with flexible inserted rod for inserting fuel.	High stresses in rod and high potential for fuel hangup during handling.
Under-the-head, slightly dispersed design with instrument leads through the cover.	Lack of testing flexibility. Vertical core disadvantages with lower flux.
Diamond-shaped modulized subassemblies to reduce sodium gap.	Swelling between diamond elements within a module.

After the concepts noted above were eliminated, the remaining basic configurations were evaluated against the specific selection criteria shown in Table 3.1. This table also summarizes the approach utilized in selecting the final concept. Critical areas utilized for selection are noted below:

Bisected Cone/Trisected Cone:

Both of these designs may be eliminated on the basis of low flux. Potential for improving the flux does not exist, as noted previously.

True Cone/True Skew:

Although the true skew and cone offer the highest flux, neither design meets the requirement for three plumb vertical positions.

TABLE 3.I. Dispersed Core Selection Criteria

Criteria	True Skew	Annular Skew	True Cone	Annular Cone	Bisect Cone	Trisect Cone	Wedge
1* Total flux available to closed loops should approach 10^{16} nv-minimum value is 0.7×10^{16} nv.	A	B	B	B	C	C	B
2* A minimum of three vertical loops with usable test diameter of at least 2.5 inches.	C	A	C	A	A	A	A
3. Open loops can be replaced by drivers -- flexibility in open loop positions for positioning throughout core.	A	A	A	A	A	A	A
4* Concept consistent with availability no later than 1974. Development monies also consistent with this date.	}						
5* Core concept consistent with allocated capital funds.							
6* Capabilities consistent with a 75% plan availability goal.							
7. Closed loop melt-down (w/o major damage) repairable in less than one week.	A	A	A	A	A	A	A

* Indicate major criteria

A Definitely Meets Criteria
 B Marginally Fulfills Criteria
 C Does Not Meet Criteria

TABLE 3.I. (contd)

Criteria	True Skew	Annular Skew	True Cone	Annular Core	Bisect Cone	Trisect Cone	Wedge
8. Components and features have demonstrated feasibility by end of preliminary design. Fallback feature adoption results in a maximum delay of one year.	B	B	A	A	A	A	A
9* Closed loop in-pile tubes removal with or without fuel possible during a normal outage.	A	A	A	A	A	A	A
10* Like core components identical and interchangeable to a maximum practical degree.	C	C	A	A	A	A	A
11* Open position and closed loop contact instrumentation required.	A	A	A	A	A	A	A
12. Capability to install single channel failure detection system required.	A	A	A	A	A	A	A
13* Interim examination of closed and open positions required.	A	A	A	A	A	A	A

* Indicates major criteria.

The three remaining candidates, the wedge, annular skew, and annular cone, were then investigated in more detail against specific criteria. Table 3.II outlines the results of the final analysis for the three remaining concepts.

Based on the prime criterion of flux, all the remaining cores are approximately equal. The annular designs, however, appear to have a greater potential for improved flux with advanced cores. A degree of flexibility exists with the skew design as a result of its insensitivity to nozzle lattice changes.

Two other criteria which tend to show a measurable difference for the three cores under consideration are subheadings under Criterion 8. They are:

- Radial Constraint - Which may be discussed with Criterion 10, Design Simplicity.
- Stainless Steel Swelling.

The radial restraint design involves the application of an external load to the outer periphery of the core. Any tendency toward nonsymmetry (which is an acute problem with the skew, and a problem to a lesser degree with the wedge) increases the difficulty of applying a uniform load. Unique pad designs for various selections of the skew are a fabrication and a logistics problem for fuel handling and shuffling.

Data on stainless steel swelling indicate a relationship between temperature and swelling for a given fluence level. Decreased space between subassemblies in the upper, hot portion of the skewed design lessens its capability to accommodate radial expansion resulting from swelling.

(8) Conclusions. Final choice for the Improved Dispersed Core Task Force design is based on the items outlined above, specifically:

TABLE 3.II. *Comparison of Annular Skew, Annular Core and Wedge Against Criteria*

	<u>Annular Skew</u>	<u>Annular Cone</u>	<u>Wedge</u>
Criteria 1 - Centerpoint Flux*	0.68×10^{16}	0.66×10^{16}	0.68×10^{16}
Future Flux Potential*	$\sim 0.82 \times 10^{16}$ (using a true skew)	$\leq 0.79 \times 10^{16}$ (using the tapered fuel assembly or a true cone)	0.68×10^{16}
Effect of Top Nozzle Spacing	Little flux change	Significant flux change	Flux change intermediate with cone and skew
Criteria 2 - Closed Loop Testing			
Flux in position away from Core Center	Higher flux in other loops - especially at edge of core	Less gradient in a given channel	Insufficient data available to judge
Test Tilt Angle	Less than 10°	Less than 5°	Less than 10°
Test Loop OD	All are about equivalent, the skew has more room below the core for blankets, etc.		
Criteria 3 - Open Loop	Equivalent with all designs.		
Criteria 4 - Schedule	Insufficient data for complete differentiation.		
Criteria 5 - Cost	Larger tilt of skew and wedge may increase hot cell cost. Nonsymmetry of wedge and skew would require additional development costs.		
Criteria 6 - Plant Factor	Insufficient data available for differentiation.		
Criteria 7 - Test Element Failure	Problems equivalent with all designs.		

* *Relative numbers only - not based on detailed core design.*

TABLE 3.II. (contd)

	<u>Annular Skew</u>	<u>Annular Cone</u>	<u>Wedge</u>
Criteria 8 - Feasibility and Backup Design			
Ability to Increase Lattice Spacing	Can easily be increased to accommodate piping or refueling machine with little flux penalty	Great incentive to decrease spacing may require routing of closed loop piping over nozzles.	Less sensitive to spacing than cone but adequate spacing available only in parallel paths across cover due to single plane dispersion.
Radial Constraint	Problem with achieving adequate pad thickness at contact plane. Pads are different for each row.	No problems, easier than the bisected design.	Should present no problems. Non-symmetry could introduce nesting problems under load.
Duct below Core	Can make transition to circular section or remain hexagonal.	Taper must begin directly below core, more difficult to fabricate, few options.	Tight spacing may require eccentric design more difficult to fabricate.
Lower Tubesheet Design	Larger centerline spacing-reduced ligament problem but larger overall area-improved inlet flow distribution.	Smaller diameter and close spacing result in higher velocities and inferior inlet flow distribution.	Same as cone but less symmetric therefore harder to analyze.

TABLE 3.II. (contd)

	<u>Annular Skew</u>	<u>Annular Cone</u>	<u>Wedge</u>
Stainless Swelling	Tighter spacing in hot region therefore more susceptible	Superior due to maximum dispersion in hot zone.	Similar to the cone although spacing is slightly tighter.
Criteria 9 - Closed Loop Fuel Handling	Coupled with first item under Criteria 8, skew may have a slight advantage.		
Criteria 10 - Simplicity of Design	Subassemblies are not symmetric therefore are unique to a given core area.	No problems envisioned. Subassemblies are relocatable.	Subassemblies interchangeable but pad thickness difference requires orientation by <u>+180°</u> .
Criteria 11 - Contact Instrumentation	Equivalent problems with all concepts.		
Criteria 12 - Failure Detection	Similar problem with all concepts.		
Criteria 13 - Interim Examination	Similar problem for all three.		
<u>Safety Criteria</u>			
Sodium Void Coefficient	Minimum value.	Slightly larger.	Slightly larger.
Doppler Coefficient and DBA	No clean differentiation between concepts.		
Control Rods Required	Minimum in-core control required.	Additional in-core rods required to meet safety margins.	Slightly more control required due to nonsymmetry.

TABLE 3.II. (contd)

	<u>Annular Skew</u>	<u>Annular Cone</u>	<u>Wedge</u>
Core Meltdown	Limited positive feedback due to dispersion below pinch plane (center of core).	Inherent positive feedback due to convergence.	Inherent positive feedback due to convergence.

Wedge: Eliminated on the basis of potential for flux improvement and nonsymmetry of overall configuration.

Annular Skew: Eliminated on the basis of nonuniform subassemblies, possible associated radial restraint problems, and a higher potential for duct swelling problems.

An annular conical core appears to be the best overall design.

c. Core Analysis

R. B. Rothrock

(1) Performance Evaluation of Alternate Core Geometries.

Driver fuel subassembly designs and performance calculations for alternate dispersed core arrangements are required for comparison and choice among alternates. To provide a fair performance comparison, common design bases must be used for all cores.

(2) Criteria and Requirements. The core designs will be optimized to provide the highest possible flux at the core center, while meeting the other requirements listed below.

(a) Common Testing Requirements. Comprehensive testing requirements will be imposed on all core designs. The requirements and related calculational assumptions which directly affect the performance comparison are listed below.

1) Open Test Position Requirements and Assumptions

- The number of open test positions within the core boundary is equivalent to 10% of the total number of fueled core positions (within the core boundary), rounded to the nearest integer. In determining the number of fueled core positions, it is assumed that all in-core test positions contain fueled tests.

- Each open loop is assumed to contain 75% of the fissile loading of an average driver subassembly.
- The open loops are assumed to be dispersed throughout the driver region such that each contributes, on the average, 75% of the reactivity and power generation of an average driver.

2) Closed Loop Requirements and Assumptions

- The dispersed cores will be assumed to have 5 closed loops, each with a minimum test section diameter of 2.5 in. and a preferred diameter of 3.0 in.
- Loops with 2.5-in. test sections will be assumed to contain 37 pin fueled tests. Those with 3.0-in. test sections will be assumed to contain 61 pin fueled tests. They are weighted at 17% and 28% of the fissile loading capability of a driver, respectively, for 217 pin drivers.
- Where closed loops (or other tests) are clustered near the core center, their worth and effects on flux peaking are accounted for through the use of nuclear parametric calculations for annular-zoned cores of varying dimensions. Any closed loops not clustered near the center are homogenized into the driver region.
- The minimum OD's for reentrant closed loops with 2.5 and 3.0-in. test sections are assumed to be 4.4 and 5.1 in., respectively, in the active core. Lattice dimensions for dispersed cores are required to be large enough to accommodate the 4.4 in. loop.

(b) General Core Design Requirements and Assumptions.

In-core control rod positions are allocated generally on the basis of total core volume,* as shown in the following tabulation.

<u>Core Total Volume, liters</u>	<u>Number In-Core Rods</u>
<825	5
825-875	6
875-925	7
925-975	8
975-1025	9
>1025	10

The in-core rod positions are homogenized into the driver region of the core or the test region, depending on the specific core arrangement.

Dispersion or skew angles of the dispersed cores are adjusted to maintain a constant clearance between enclosing diameters of adjacent driver subassemblies at the reactor cover. This corresponds to 4.75 in. nominal clearance (based on 5.25 in. enclosing diameter and 10 in. pitch for the Reference design) at an elevation of 28 feet above the bottom of the fuel. The wedge geometry was adjusted to yield a comparable "mean" clearance, resulting from the different dispersion rates in the two directions.

The minimum clearance between adjacent subassemblies is 0.030 in. for all arrangements. This is assumed to occur

---

* This allocation of in-core rod positions was consistent with the split-core design approach of reducing the number of reflector rod positions in low-leakage cores so as to maintain greater freedom in location of reflector tests and reduce interference of reflector rod drives with piping, instrument runs and general core access. Subsequent evolution of the Task Force annular core design resulted in a change of emphasis to utilize more reflector control resulting in fewer in-core rod positions and a more peaked flux shape.

at the bottom of the fuel pin bundle (2.50 in. below the bottom of the fuel) in the cone and wedge arrangements, and at the twist plane (center of fuel column) in the skew.

The minimum acceptable fuel length is 32.0 in.

(c) Driver Fuel Design Bases. Fuel pin cladding thickness is intended to be chosen on a basis consistent with 0.014 in. cladding for 0.235 in. (OD) pins. This was represented in the core survey calculations as a constant clad thickness-to-OD ratio, equal to $(0.014)/(0.235)$.

Smeared density of fuel = 0.89 times theoretical density. The theoretical density of $(U, Pu)O_2$ in the enrichment range of interest (20 to 25% Pu) is assumed to be 11.06 g/cm^3 independent of composition.

The fuel length is adjusted to yield an average fuel pin linear heat deposition rate of 7.8 kW/ft at a total (driver plus open loop) power of 400 MW, assuming 91% heat deposition in the fuel pins. This yields a peak linear power of 18 kW/ft at 125% power when hot channel factors typical of the (zoned) reference design are used (1.14 engineering factor, 1.61 overall peaking factor).

The fuel enrichment calculated by the core survey model is set at a maximum of 19.5% ^{239}Pu . This yields a $U/^{239}\text{Pu}$ ratio of 4.0 for plutonium with a 12% ^{240}Pu content. The fissile loading correlation on which the enrichment calculation is based was determined from calculations for a uniform (unzoned) driver region.

The total fuel pin length is assumed to be twice the active fuel length plus 9.5 in. This is assumed to accommodate a fission gas plenum of length equal to the fuel, plus one (upper) 6 in. axial reflector.

The total fuel pin bundle ΔP is calculated based on a friction factor of $0.20 (\text{Re})^{-0.2}$ and the mean flow velocity existing in the in-core portion of the subassembly. For cases where an increase in the flow duct cross section is possible above the active core, this is taken into account by using an effective hydraulic length which is less than the physical lengths of the pins. This effective hydraulic length is taken as the sum of the first 42 in. of the fuel pins plus 80% (for skew and wedge geometry) or 25% (for cone geometry) of the excess length beyond the first 42 in. The calculated ΔP for the fuel pin bundle is limited to 65 psi maximum.

The minimum fuel pin spacing is 0.040 in. If average coolant ΔT 's greater than 300 °F are calculated at 65 psi total fuel pin bundle ΔP for 0.040 in. spacing, then the spacer diameter is increased as needed to yield a 300° average coolant ΔT , based on 95% of 400 MW removed by the driver plus open loop coolant.

(3) Results. Comparative results calculated by consistent methods are listed in Table 3.III for the various core arrangements, with the exception of the trisected core. Peak flux estimates are based on minimal use of reflector control, and conditions expressed in Table 3.IV.

TABLE 3.III. Fluxes of Various Core Arrangements^(a)

Core Arrangement	Flux Estimates at Center at 400 MW
Split Cone with 2.6 vol% PuO ₂ in Gap	0.56 x 10 ¹⁶
Wedge Study III Modified	0.65
Annular-cone, Study I	0.66
Annular-skew, Study I	0.70
True Cone, 4 closed loop tests in central 7-cluster	0.66
True Skew, 4 closed loop tests in central 7-cluster	0.85

a. Fluxes valid only under conditions expressed in Table 3.IV.

(4) Evaluation.(a) Basic Features Distinguishing Different Geometries.

The performance differences between various core geometries tabulated in Section III arise from differences in the following characteristics:

- Test region geometry (i.e. slab or cylinder) and size relative to driver region.
- Homogenized fuel fraction within test region.
- Packing efficiency within driver region.

The mechanical studies performed for each core arrangement provide this necessary information, in the form of shape and size of test region (and its relation to the driver region lattice, if any); feasibility of locating driver subassemblies within the test region boundary; and in-core dispersion of subassemblies needed to satisfy nozzle clearance requirements at the reactor top head. The information in Table 3.IV summarizes the mechanical input data used for this study.

TABLE 3.IV. Core Geometries

Core Arrangement	Test Region	PuO ₂ Fraction ^(a) in Test Region, vol%	Mean Clearance in in. Between S/A's in Driver Region
Split cone with fuel in gap	Slab, ~6 in. width, across full diameter	2.6	0.326
Wedge, 10 in. ϕ spacing at reactor cover	Slab, ~2.7 in. width, across full diameter	2.0	0.240
Annular-cone, 3 vertical closed loops in central 7	Central cluster of 7 positions	3.5	0.344
True cone, 4 closed loops in central 7-cluster	Central cluster of 7 positions	3.5+	0.333
Annular-skew, 3 vertical closed loops in central 7 twist plane at middle of core	Central cluster of 7 positions	0.6	0.050
True skew, 4 closed loops in central 7-cluster, twist plane at middle of core	Central cluster of 7 positions	3.5+	0.050

a. Based on a driver fuel PuO₂ fraction of 7.0 vol%.

(b) Relative Accuracy of Calculations. In the annular (and true cone and true skew) core calculations, the required fissile loading and the estimated flux were obtained from nuclear survey calculations which represented the central test region and the outer driver annulus as homogenous regions with different PuO_2 fractions. Estimates of flux and ^{239}Pu loading for the wedge and split cone were based on previous nuclear studies made for the split cone, with greater geometric detail, but with a low PuO_2 fraction in the gap. Consequently the comparisons among the annular cone, annular skew, true cone, and true skew are believed to be mutually consistent and represent the true relative ranking of the cores. The comparisons between the annular and slab-type cores are subject to some bias arising from the use of different nuclear models. Finally, the flux estimates for the bisected cone and wedge are also subject to some error due to homogenization of the test region. This is somewhat an academic question with regard to the split cone, since the placement of driver fuel assemblies in the gap (to yield the PuO_2 fraction of 2.6 vol%) represents a substantial mechanical design problem and was included in the analysis only to represent an "ultimate" condition. However, the location of drivers within the gap adjacent to the vertical closed loops in the wedge geometry appears to be mechanically feasible, and their effects on the flux at the test locations may not have been adequately represented by smearing the drivers into a homogenized slab test region of rather low (2.0 vol%) PuO_2 fraction. The wedge appears conclusively to have the highest flux of the class of bi- or trisected cores studied, independent of these considerations. For cores of similar geometry (e.g. annular), the items in Table 3.IV affecting the achievable flux are the mean subassembly clearance, which determines the packing efficiency of the driver region; and the PuO_2

fraction in the test region, which affects the flux shape, particularly the degree of peaking or depression within the test region. Among the annular designs listed in Table 3.IV only the true skew appears capable of combining a high degree of driver region compaction with a capability for maintaining a fairly high smeared fuel fraction within the central 7-cluster without the use of special fuel assemblies, and it therefore has the highest ultimate flux capability. The annular-skew with special fuel assemblies permitting the location of three drivers within the test region (not shown in Table 3.IV) would have virtually the same flux capability as the "true skew" with four closed loops in the central 7-cluster. In regard to the skewed designs, the mean clearance between subassemblies of 0.050 in. used in the core calculations did not properly take into account the variation of subassembly clearance from row to row radially, and it appears that a larger mean sodium gap should have been used to be representative of the entire core.

(5) Core Analysis of Annular Core. The major "free variables" which must be selected to establish the core design are the fuel pin and spacer wire diameters. Once these dimensions are chosen the remaining core parameters are fixed, or narrowly limited, by the design bases and requirements described in the preceding Section. The consequences of various choices for fuel pin OD and pin spacing are illustrated in Figure 3.13(a), in terms of calculated variables of core volume and flux, average $^{238}\text{U}/^{239}\text{Pu}$ ratio, and core hydraulic (ΔT , ΔP) performance. The curves shown are based on the annular cone design with the fuel length and smeared density, thickness to diameter ratio of the fuel pin cladding and subassembly duct, and allocation of core lattice positions between drivers, control rods and tests, all held fixed. The

FIGURE 3.13. Core Performance Map (Parameter Survey)

core lattice dimensions are adjusted as necessary to correspond with the various fuel pin and spacer sizes, and the core dispersion rate (inclination per row) is held fixed at 0.016 in./in. From these results a 0.235 in. fuel pin was chosen because it provides the smallest core (highest flux) that can be achieved with a spacer diameter ≥ 0.040 in. and a $^{238}\text{U}/^{239}\text{Pu}$ ratio ~ 4.0 . The average unorificed coolant temperature rise is about 260 °F at 65 psi pressure drop across the fuel pin bundle. Based on a subassembly peaking factor of 1.35, the hot channel coolant temperature rise is about 350 °F. If it should be found necessary to orifice the subassemblies for equal temperature rise, then the lowest bulk coolant temperature rise obtainable (assuming no orificing of the hottest subassemblies) would be 350 °F. The hydraulic capability for an orificed coolant temperature rise of 300 °F could be obtained by limiting the average (unorificed) temperature rise to 225 °F. The various core designs to achieve this can be interpolated from the graph; for example, if it were desired to retain the 0.235 in. fuel pin and provide an orificed ΔT of 300 °F, the spacer diameter would be increased to about 0.043 in. and the $^{238}\text{U}/^{239}\text{Pu}$ ratio would become about 3.85 at 89% smeared density.

A nominal cladding thickness of 0.014 in. was selected for the 0.235 in. fuel pin. This selection meets the requirement of resisting internal fission gas pressure at end-of-life, while also being within the range of fast ceramic fuel pin irradiation experience. It is expected that the fission gas pressure criterion alone may permit thinner cladding in the range of 0.011 to 0.012 in. The additional thickness margin is included to accommodate possible harmful fuel-cladding interaction effects, pending further irradiation experience.

d. Nuclear Control

D. Marinos

The control system selected for the dispersed core by the task force consists of poison rods moving axially into the core from their "full out" position above the core. These poison rods occupy one lattice position per rod in the core and reflector. The control system is divided into three subsystems; (1) the primary safety system in the core, (2) the shim-safety system in the core, and (3) the shim-regulating system in the inner reflector row.

The three primary safety rods, worth 6 to 8\$ each, will reside above the core during operation to be driven into the core upon indication of an operating abnormality. The selected poison material is enriched B_4C to 26.2 at.% in a 61-pin array. The drive mechanism has not been selected pending clarification of safety requirements on the use of a different mechanism in the primary system from that used in the secondary system. This requirement, if imposed, will minimize a failure of a generic type that might result in the failure of both safety systems from the same fault. A pneumatic cylinder drive which is most compatible for scram speed changes is also the most compact drive for this system, Figure 3.14 (SK-3-14327). The use of piston and rod seals in a sodium environment puts this drive outside the "state-of-the-art" category, and a development program will be required. The magnetic jack drive used on PWR reactors is also a candidate for the safety rod drive. The large jog step of 0.375 in. is too great to meet the requirements for a control rod but can be used for a safety rod. This drive has some "test" time in a sodium environment with good results. One of the two drives will be recommended only if they satisfy the "different mechanism" requirement for safety purposes.

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FIGURE 3.14. Primary Safety Rod: Dispersed Core Task Force

The shim and regulating rods which comprise the control and secondary safety system are apportioned between two subsystems from the control worth standpoint. The shim safety rods, of which there are three, are located in the core and will be worth 3 to 4\$ each. They will be used to compensate for fuel burnup, temperature coefficients, and poison burnup. The tentative reference poison material for this system is tantalum. Analysis of the temperatures with rods partially inserted in the core during operation indicates that the use of B_4C may result in temperatures exceeding 1000 °F; and results suggest that the behavior of this material at these temperatures is not well enough defined. Tantalum temperatures of the order of 1600 °F have been calculated and no known problems have been found. The poison sections of B_4C (primary safety rods) and tantalum (shim-safety rods) are shown in Figure 3.15 (SK-3-14328).

There are 24 regulating rods located in the first reflector ring around the core. These rods are of low individual worth and are used for fine control. The poison selected will be natural B_4C and the calculated temperatures are below 900 °F.

The drive mechanism selected for both the shim-safety and regulating rods is a roller nut drive. Five candidate drives were chosen to be evaluated for their potential use in the FTR. Three of the drives are operating units with several years experience, not necessarily in sodium service. The other two drives were conceptual in nature but specifically adapted for the FTR. The evaluation criteria were formulated to indicate operating requirements of the FTR. Although none of the drives completely met all the requirements, the roller nut drive showed the greatest promise and was thus selected as the reference drive. See Figure 3.16 (SK-1-14257).

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Scram requirements for the primary safety system indicate accelerations greater than 1 g and possibly up to 3 g. The pneumatic scram feature is selected because of the ease in altering scram forces to any desired value by changing the cylinder pressure. One of the alternate primary drives is a pneumatic drive. The other, the magnetic jack, requires scram assist for accelerations of 0.8 g or greater. Again, a pneumatic scram assist is the most logical choice because of its adaptability.

The scram requirement for the secondary safety system results in less rapid response times than the primary system. The secondary system is not required to function until failure of the primary system is indicated. This requirement may allow use of a poison system that is not readily removable from the core. If, however, a poison rod shim control system also is used for the secondary safety system, it is not unreasonable to use the same response times for both safety systems and to scram both systems simultaneously. In this case, if a pneumatic system is used for the primary safety system, it would be reasonable to use the same type of system for the secondary safety system scram assist. Power supplies such as signal sources and accumulators, can be separate and redundant to ensure against generic failure.

e. Core Design Parameters

The design parameters developed for the annular conical core configuration by the dispersed core task force are shown in Table 3.V.

TABLE 3.V.   *Core Design Parameters - Improved Dispersed Core*GENERAL

Reactor Type	Fast Fuel and Material Test
Reactor Power	400 MW _t
Driver Fuel	(U,Pu)O ₂ Pellets, SS Cladding
Reflector	Nickel, SS Cladding
Primary Coolant	Sodium

CORE CONSTRUCTION

Geometry	Annular Cone
Dispersion Angle - Per Row	0 degree 56 min
Height - Active Zone	36.0 in.
Effective Radius - Active Zone	47.9 in.
Volume - Active Zone	1064 liters
H/D Ratio	0.75
Lattice Geometry	Hexagonal, 4.77 in. Mean Pitch
Core Subdivision	
Total	91
Driver Fuel	71
Test Positions	13
Control Elements	6
Optional (Center)	1
Radial Reflector	
First Ring Elements	7
First Ring Control	24
First Ring Test	5
Second Ring Elements	42
Test Facilities	
Closed Loops	5
Open Test Positions	8 In-Core 5 First Row Reflector

TABLE 3.V. (contd)FUEL ASSEMBLY

Duct Dimension (Across Flats) OD	4.420
Duct Wall Thickness	0.130
* Fuel Pins Per Subassembly	217
Fuel Pin OD	0.235
* Fuel Pin Spacing	0.275
Spacing Method	Spiral Wire Wrap
Pitch/Diameter	1.17
Cladding Material	TP-316 SS
Cladding Thickness	0.014
Fuel Length	36.0 in.
Fission Gas Plenum Length (Top)	36.0 in.
Axial Reflector Length	
Top (Integral)	6.0 in.
Bottom (Non-Integral)	6.0 in.
Fuel Form	Solid Cylindrical Pellets
Smear Density	89% TD
$^{238}\text{U}/^{239}\text{Pu}$ Ratio	3.9
Goal Exposure (Peak)	80,000 MWd/MT
Assembly Length (To Disconnect)	18 ft

CONTROL ASSEMBLY

Type	Movable Poison Rod
Number - Poison	
In-Core Safety	3, Enriched B_4C
In-Core Shim	3, Tantalum
Reflector	24, Natural B_4C
B_4C Design (Reflector Shim)	
Pins/Assembly	61
Pin OD	0.461 in.
Clad Thickness - Material	0.040 in.
B_4C Pellet Density	80 Percent

TABLE 3.V. (contd)CONTROL ASSEMBLY (contd)B₄C Design (Reflector Shim)
(contd)

Volume Fractions

Coolant	41.1 vol%
Poison (B ₄ C Plus Void)	28.2 vol%
Stainless Steel	30.7 vol%

Tantalum Design (In-Core Shim)

Pins/Assembly	7
Pin OD	1.300 in.
Clad Thickness - Material	0.030 in. - Stainless Steel

Volume Fractions

Coolant	35.89 vol%
Poison	43.58 vol%
Stainless Steel	20.11 vol%
Void	00.42 vol%

Shim Rod Drive Type	Roller Nut-Lead Screw, With Scram Assist

THERMODYNAMIC DATA

Thermal Core Power - Total	400 MW, Plus Closed Loop Tests
Driver Fuel	369 MW
Tests (Open Loops)	31 MW
Coolant Inlet Temperature	550 °F
Average Coolant Outlet Temperature (Orificed)	910 °F
Maximum Coolant Outlet Temperature	910 °F*
Maximum Coolant Velocity	30 ft/sec
Pressure Loss	
Across Fuel Subassembly	100 psi
Vessel Inlet to Outlet	120 psi

* Clean Core

TABLE 3.V. (contd)THERMODYNAMIC DATA (contd)

Maximum Fuel Temperature	
Maximum Heat Flux	
Maximum Clad Temperature	
Hot-Channel Factor (Eng.)	1.14
Hot-Channel Factor (Radial Peak)	1.40*
Hot-Channel Factor (Axial Peak)	1.24
Average Linear Heat Rating	7.3 kW/ft
Maximum Linear Heat Rating	14.4 kW/ft*
Maximum Linear Heat Rating (With OPF)	18.0 kW/ft*
Overpower Factor (OPF)	1.25

PHYSICS DATA

^{239}Pu Loading in Fresh Core	574 kg
Total Peak Neutron Flux at Central Closed Loops	0.7×10^{16}
Effective Neutron Lifetime	3.5×10^{-7} sec
Delayed Neutron Fraction	0.003
Doppler Constant ($T \frac{dk}{dT}$)	-0.004
Control Requirements	
Temperature Defect	1.0%
Fuel Burnup	2.0%
Experiment Variation	2.0%
Fractions of Neutrons above 0.1 MeV	67%
Operating Cycle	80 Full Power Days

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* *Clean Core*

2. Vertical Core Task Force Activities (This work done for System 30. Reactor)

Work of the Vertical Core Task Force is summarized below.

a. Vertical Core Design

A. F. Lillie

A task force was established within PNL and assigned the responsibility for developing certain features of the vertical core concept in sufficient detail to demonstrate a firm backup design. Features that had previously been identified as insufficiently developed to constitute a backup position for the concept included:

- Mechanical arrangement of core instrumentation
- Capability for operating and maintaining instrumentation
- Instrumentation for open and closed loops
- Making and breaking sodium piping below the top head
- Suitability of control rod drives
- Fuel handling and experimental loop removal techniques compatible with the reference concept.

In addition to the above design problems, the study also included requirements for commonality with the reference dispersed core in the areas of fuel pin design, vessel and test loop capability. The design concept, as developed by the Task Force, consists of an open lattice vertical core with open plug refueling. An illustration of the concept is shown in Figure 3.17. The core itself consists of 139 lattice positions comprising 12 positions allocated for closed loop facilities, 12 positions for open test positions, 3 safety rod positions, and 112 driver fuel positions. Shim control-primary shutdown is accomplished with 44 peripheral control rods using enriched B_4C absorber material.

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FIGURE 3.17. Backup Core Concept

The core, 36 in. in height and with a 42 in. equivalent diameter, is rated at 400 MW_t with a total sodium coolant flow of 35,000 gal/min and a core ΔT of 300 °F. Peak total neutron flux is calculated to be 0.88×10^{16} nv with a core loading of 2950 kg of PuO₂-UO₂ fuel, enriched to 18.5% fissile Plutonium.

The core fits within the reactor vessel presently being considered for the reference core. Changeout of the core support plate would be required to accommodate the vertical core in the reference vessel. The 12.5 ft long fuel assemblies (see Figure 3.18) are compatible with the under-sodium refueling mode chosen by the task force.

b. Vertical Core Control Mechanisms

C. A. Munro (Development Department)

The control mechanism as envisioned for the vertical core consists of 44 peripheral control rods worth \$0.47 each and three central region safety rods worth \$3.71 each. Thirty-four of the peripheral rods are driven in tandem by one drive unit. Therefore, there are 27 drives required for peripheral control and three for the central safety rods.

Because of the severe diametrical space limitations imposed by the vertical core design, the control mechanism envisioned for all 30 drives is a modified General Electric design. This consists of a pneumatic assisted electric motor, screw-nut arrangement.

Gas pressure on the under side of a piston attached to the extension rod drives the upper end of the piston rod into engagement lugs on a positioning nut which in turn is driven by a motor-screw combination. The scram function is accomplished by venting the gas pressure from the under side of the piston thereby allowing the poison section and

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FIGURE 3.18. Vertical Core Fuel Assembly Configuration

extension rod to fall into the reactor core. Scram acceleration assist is provided by simultaneously applying gas pressure to the upper side of the piston driving the rod down. The last 4 in. is used for deceleration by a snubber. Two gas pressures would be required: a 150 psi supply for the rod lifting function and a higher pressure source for scram assist (500 to 600 psi for 3 g). The problems of excess acceleration-induced compression and bending moment loads in the long extension rod was investigated and the design was found to be feasible.

The poison section consists of enriched B_4C pellets clad in stainless steel. Because of the high costs of ^{10}B , enrichment was held to minimum values consistent with the poison strengths needed. Peripheral rods contain 30% ^{10}B while the safety rods contain 45% ^{10}B . Peripheral control rods include a nickel follower in the lower pin array for a reflector. Tantalum was considered; however, additional control rod positions would be required because of its lesser poison strength. Central safety rod assemblies would weigh ~180 lb each. Single peripheral control rods would weigh ~291 lb.

The only portion of the control mechanism exposed to sodium vapor is the extension rod. A cover gas would be supplied to the shaft seal to preclude the necessity of and associated problems with a sodium vapor-tight axial moving seal.

Ten other control rod designs were reviewed for application to the Vertical Core Concept, however, most will not fit into the very limited envelopes available for peripheral control mechanisms.

c. Vertical Core Testing Facilities

M. Vogel (Project Engineering Department)

(1) Closed Loop Test Positions.(a) Summary of Design Parameters.

Type	Hairpin in two adjacent lattice positions (3-3/8 in. across flats)
Tubes in core	6
Lattice positions	12
Usable test diameter	2.75 in.
Pin per loop	37 (0.25 in. diam each)
Instrumentation	1 in. diam tube containing 19 electrical leads.

(Contact or channel thermocouples, or electrical leads, pressure measurements converted to electrical signals inside tube.)

(b) Operating Capabilities.

37-Pin Experiment Total Power, MW	Linear Power, kW/ft	Flow Rate Through Tube, gpm	ΔP Across Tube ^(a) psi	Bulk Outlet Temperature, °F
2.5	22.5	300	100	1200
4.0	36	400	178	1200
2.5	22.5	400 ^(b)	178	1400

a. With 70 psi ΔP across test bundle at 300 gpm

b. 1400 °F outlet attainable using 50% bypass around test, i.e., 200 gpm through test, 200 gpm around test

(2) Open Test Positions.(a) Summary of Design Parameters.

Size	Same as driver fuel duct loaded with up to 91 fuel pins
Lattice positions	12 provided (instrument and sodium sampling layouts indicate the potential for 26 in-core positions)

(b) Operating Capabilities.

Channel Power	0.3-4.4 MW
Channel Flow	30-440 gpm
Inlet Temperature	600 °F initial - 900 °F final
Outlet Temperature	600 °F to 900 °F initial - 900 °F to 1200 °F final
$\Delta P^{(a)}$ Across Test Bundle	0 to 70 psi
$\Delta T^{(b)}$ Across Test Bundle	40 °F to 400 °F

-
- a. ΔP across test position plenum-to-plenum = 100 psi
 b. ΔT across test position plenum-to-plenum = 300 °F,
 i.e., bypass flow required for 400 °F test ΔT

(c) Open Test Position Instrumentation. An instrument stalk, consisting of a 1 1/2 in. diam tube, is attached to the test assembly and extends upward through the refueling plug. A remotely operable instrumentation connector is mounted on the top end of the stalk. The instrumentation stalk accommodates 37 electrical wires allowing both contact and channel instrumentation for the test assembly. Provision for open test position failure detection is provided by a 1/2 in. diam sodium sampling tube, integral with the refueling plug- instrument harness assembly, which dips into the top end of the test assembly flow duct.

(d) Vertical Core Instrumentation (G. H. Strong, Project Engineering Department). Driver fuel instrumentation consists of:

- Failed element detection and location system (FEDAL)
- Individual fuel channel coolant outlet temperature measurement
- Individual fuel channel coolant flowrate measurement (at a minimum, a go-no-go indication of coolant flow).

The FEDAL system selected was the delayed neutron monitor approach, based upon experience at EBR-II and the current design philosophy embraced by PFR. The system consists of 1/2 in. tubes routed to each driver fuel and open test position channel. A portion of the effluent coolant from each channel is removed from the reactor through these tubes by means of a Toepler pump and routed to appropriate delayed neutron monitor stations. The sampled coolant is returned to the primary system by an EM pump located at the Toepler pump discharge, being returned either to the purification system inlet or to the suction side of the primary sodium pumps.

Flow measurements are made using EM flowmeters on each and every sodium sampling tube used in the FEDAL system. The system forms a bypass flowmeter arrangement for measuring individual channel flowrates.

Individual channel coolant outlet temperatures are measured using redundant thermocouples positioned inside a protective 1/2 in. diam tube which is positioned in the outlet region of each channel similar to the sodium sampling tube arrangement.

The sodium sampling and thermocouple tubes are suspended from an instrumentation harness assembly affixed to the underside of the refueling plug. All electrical and sample coolant

connections are made above the plug in the hot cell using remotely operable connectors. A spare identical refueling plug-instrument harness rig is available for use if repairs are required on the other rig.

(e) Reactor Component Handling (G. H. Salzano). The fuel handling system is an open pool type employing a finned pot for driver fuel and for open and closed loop transfers. Because the refueling plug is removed for fuel transfers, an inert gas hot cell is required.

The open loop instrument stalks and closed loop tube extensions penetrate the refueling plug and are sealed with a cap containing the instrument connections.

A lobe large enough to provide removal of a 7-cluster group of reflectors is designed into the refueling plug. The resulting open space allows insertion of a finned pot for transporting in-core components.

Remotely operated cranes, manipulators, special tools, television viewers and shielded windows are provided for hot cell operations. An air-to-gas lock is provided for manned entry during emergency operations.

(f) Plant Availability (G. A. Halseth, *Operations Section*). Based on past experience with operating personnel under similar conditions, the crew efficiency was assumed to be 75%. Because the remotely operated equipment is made up of complex mechanisms and because some of it will require spares, a 20% loss of time was assessed against equipment failures.

Using the above assumptions and three crews on a 24-hr work day, the results of four time-sequence cases are as follows:

(1) Outage for a single test loop replacement	9.5 days
(This is the minimum time including disconnecting jumpers, removing seals, unlocking the plug, removing the plug, reversing the procedure, and checking out the jumper connections.)	
(2) Normal quarterly outage with the NPTF in operation	42 days
(3) Normal quarterly outage with NPTF in operation and a dummy seal plug used in place of the refueling plug	31 days
(4) Normal quarterly outage without NPTF in operation and regular closure plug used.	61 days

In Case 1, studies indicate a plant availability of 32% to 42%. In Case 2, studies indicate a plant availability of 44% to 52%. In Case 3, studies indicate a plant availability of 27% to 34%.

Thus it becomes evident that the NPTF and a special seal plug must be used during normal outages if acceptable plant availability is to be achieved.

(g) Core Design Parameters. The design parameters developed for the vertical core configuration by the vertical core task force are shown in Table 3.VI.

TABLE 3.VI.   *Design Parameters of Vertical Core Configuration*

<u>Item</u>	<u>Value</u>	<u>Basis</u> (See Table footnotes)	<u>Notes</u>
<u>Core Configuration</u>			
Type	Vertical, open lattice open pool refuelling	G	
Lattice configuration	Hexagonal, ducted	A	1
Number of core lattice positions	139	A	1
Number of lattice positions for closed loops	12	CR	2
Number of lattice positions for open test positions	12	CR	2
Number of safety rod lattice positions	3	A	1
Number of driver fuel lattice positions	112	C	
Number of control rod positions	44	C	
Location of control rods	Core periphery	A	1
Refueling mode	Fully remotized, inert gas hot cell, open pool core access	A R	
<u>Overall Core Performance</u>			
Nominal operating power (not including closed loops), MW _t	400	G	3
Coolant conditions:			
Coolant	Na	G	3
Total reactor flowrate, gal/min	35,000	C	

TABLE 3.VI. (contd)

<u>Item</u>	<u>Value</u>	<u>Basis</u>	<u>Notes</u>
<u>Overall Core Performance (contd)</u>			
Coolant conditions: (contd)			
Core inlet temperature (initial), °F	600	A	4
Core outlet temperature (initial), °F	900	C	3
Peak total flux (nv x 10 ¹⁶):			
Start of cycle	0.88	C	5
End of cycle	0.83	C	5
Plant availability (assuming an NPTF, a dummy refuelling plug, and about a 90-day reactor cycle), %	50	C	
Testing Facilities:			
Number of closed loops (maximum)	6	G	2
Number of open test positions (maximum)	12	G	2,6
<u>Dimensions</u>			
Active core volume, liters	885	C	5
Core size:			
Height, in.	36.0	C	5
Equivalent diam, in.	42.0	C	5
Overall minimum fuel assembly length, in.	150.0	C	

TABLE 3.VI (contd)

Item	Value	Basis	Notes
<u>Dimensions</u> (contd)			
Reflectors:			
Lower axial (nickel), in.	6.0	C	7
Upper axial (nickel), in.	6.0	C	7
Radial (nickel), in.	10.5	A	1
Shielding requirements:			
Lower axial (SS) (integral with fuel assembly - see Figure 3.18), in.	30.0	C	
Upper axial (integral with reactor cover):			
50% ρ thermal shielding (SS), in.	18.0	C	
Refuelling plug (SS), in.	6.0	C	
Reactor vessel requirements	Backup core to fit within reference reactor vessel	G	
<u>Compositions</u>			
Fuel assembly:			
Fuel material	PuO ₂ -UO ₂	G	1
Cladding material	Type 316 SS	A	1
Duct material	Type 316 SS	A	8
Axial reflector material	Nickel	A	1

TABLE 3.VI. (contd)

Item	Value	Basis	Notes
<u>Compositions (contd)</u>			
Fuel assembly (contd):			
Lower axial shield material	Type 316 SS	C	
Fuel-cladding thermal bond	Helium	A	8
Volume fractions, %			
Fuel (also-very nearly reflector and lower shield material)	34.5	C	
Stainless steel	25.3	C	
Coolant	38.6	C	
Helium	1.6	C	
Safety rod assembly:			
Absorber material	45% enriched B ₄ C	C	
Cladding material	Type 316 SS	A	8
Duct material	Type 316 SS	A	8
Follower material	None		
Volume fractions, %			
Absorber	42.3	C	
Stainless steel	24.9	C	
Coolant	29.6	C	
Control rod assembly:			
Absorber material	30% enriched B ₄ C	C	
Cladding material	Type 316 SS	A	8

TABLE 3.VI. (contd)

Item	Value	Basis	Notes
<u>Compositions</u> (contd)			
Control rod assembly (contd):			
Duct material	Type 316 SS	A	8
Follower material	Nickel	A	
Volume fractions, %:			
Absorber material	42.3	C	
Stainless steel	24.9	C	
Coolant (in absorber region)	29.6	C	
Follower (nickel; cladding and duct included)	70.4	C	
Coolant (in follower region)	29.6	C	
Radial reflector:			
Reflector material	Nickel	A	1
Cladding material	Type 316 SS	A	8
Duct material	Type 316 SS	A	8
Volume fractions, %:			
Nickel	34.5	C	9
Stainless steel	25.3	C	9
Coolant	38.6	C	9
Helium	1.6	C	9
Radial shield:			
No design effort placed on this item during the task force work.			

TABLE 3.VI. (contd)

<u>Item</u>	<u>Value</u>	<u>Basis</u>	<u>Notes</u>
<u>Compositions (contd)</u>			
Closed loops (for physics calculations purposes)			
Volume fractions, %:			
Fuel	15% of standard driver fuel in both downcomer and riser sections	G	2
Stainless steel and coolant	Remainder	G	2
Open test positions (for physics calculations purposes)			
Volume fractions, %:			
Fuel	75% of standard driver fuel assembly	G	2
Stainless steel and coolant	Remainder	G	2
<u>Fuel Assembly Design and Performance</u>			
Design (see Figure 3.18)			
Fuel pin OD, in.	0.235	G	
Cladding wall thickness (nominal), in.	0.015	G	3
Fuel column length, in.	36.0	C	3
Overall fuel pin length (including integral axial reflectors), in.	84.0	C	3,7

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TABLE 3.VI. (contd)

Item	Value	Basis	Notes
<u>Fuel Assembly Design and Performance</u> (contd)			
Design (see Figure 3.18) (contd)			
Number of fuel pins per assembly	127	A	1
Fuel pin pitch, in.	0.290	C	
Fuel pin P/D	1.233	C	
Wire wrap spacer diam, in.	0.055	C	
Fuel duct outside dimension across flats, in.	3.51	C	
Fuel duct nominal wall thickness, in.	0.100	A	1
Fuel assembly pitch, in.	3.54	C	10
Fuel assembly weight (approximate), lb	225	C	
Overall fuel assembly length, ft	12.5	C	
Core loading:			
Total PuO ₂ -UO ₂ , kg (assuming 88% smear density)	2950	C	5
Total fissile Pu, kg	475	C	5
U/fissile Pu ratio (assuming 12% ²⁴⁰ Pu)	4.33:1	C	5
Fissile enrichment, %	18.3	C	5

TABLE 3.VI. (contd)

<u>Item</u>	<u>Value</u>	<u>Basis</u>	<u>Notes</u>
<u>Fuel Assembly Design and Performance (contd)</u>			
Performance:			
Peaking factors (1 zone core, end of cycle)			
Core radial	1.4	A	1
Core axial	1.2	A	1
Local	1.1	A	1
Overpower factor	1.2	A	1
Peak linear heat rating at nominal full power, kW/ft of pin:	14.0	G	3
Peak cladding surface heat flux, Btu/hr-ft ² -°F	780,000	C	
Peak operating temperatures at nominal full power, °F			
Cladding ID, 900 °F coolant outlet, hot spot	1100	C	3
Fuel center, 900 °F coolant outlet	4500	C	
Available coolant ΔP (plenum to plenum), psi	100	G	3
Coolant velocity through fuel pin section, ft/sec	31	C	
Coolant temperature rise (orificed core assumed), °F	300	A	1,3

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TABLE 3.VI. (contd)

Item	Value	Basis	Notes
<u>Fuel Assembly Design and Performance (contd)</u>			
Performance (contd):			
Flow to peak driver fuel assembly, gpm	370	C	
Power output at nominal full power operation:			
Peak driver assembly, MW _t	4.41	C	
Average driver assembly, MW _t	3.15	C	
Target peak fuel burnup, MWd/tonne of metal	80,000	G	3
Maximum reactivity worth of fuel assembly, dollars	3.00	C	
<u>Safety Rod Assembly Design and Performance</u>			
Design			
Absorber pin OD, in.	0.352	C	
Cladding wall thickness, in.	0.015	C	
Absorber form	B ₄ C pellets, 99% TD		
Number of pins per assembly	61	C	
Absorber pin pitch	0.382	C	
Absorber pin P/D	1.09	C	

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TABLE 3.VI. (contd)

<u>Item</u>	<u>Value</u>	<u>Basis</u>	<u>Notes</u>
<u>Safety Rod Assembly Design and Performance (contd)</u>			
Design (contd)			
Wire wrap spacer OD, in.	0.030	C	
Absorber length, in.	36.0	C	
Assembly duct outside dimension across flats, in.	3.13	C	
Assembly duct wall thickness, in.	0.065	C	
Clearance between assembly duct and channel duct, in.	0.090	C	
Type of drive mechanism	G.E. pneumatic	A	1
Weight of safety rod assembly (total moving weight), lb	180	C	
Performance:			
Maximum reactivity worth per safety rod assembly, dollars	3.70	C	
Maximum permissible scram initiation delay, msec (with 1 g rod acceleration)	200	C	
Minimum required rod acceleration, g's, (with scram delay of 200 msec)	1.0	C	
Total worth in safety rod bank, \$	11.10	C	11
Peak operating absorber temperature, °F	Nearly coolant temperature	C	

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TABLE 3.VI. (contd)

<u>Item</u>	<u>Value</u>	<u>Basis</u>	<u>Notes</u>
<u>Control Rod Assembly Design and Performance</u>			
Design:			
Absorber pin OD, in.	0.352	C	
Cladding wall thickness, in.	0.015	C	
Absorber form	B ₄ C pellets, 99% TD		
Number of pins per assembly	61	C	
Absorber pin pitch	0.382	C	
Absorber pin P/D	1.09	C	
Wire wrap spacer OD, in.	0.030	C	
Absorber length, in.	36.0	C	
Follower length, in.	30.0	C	
Assembly duct outside dimension across flats, in.	3.13	C	
Assembly duct wall thickness, in.	0.065	C	
Clearance between assembly duct and channel duct, in.	0.090	C	
Type of drive mechanism	G.E. pneumatic	A	1
Number of control rod drive mechanisms	27	A	1

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TABLE 3.VI. (contd)

Item	Value	Basis	Notes
<u>Control Rod Assembly Design and Performance (contd)</u>			
Design (contd):			
Weight of control rod assembly lb:			
Single assembly (total moving weight)	290	C	
Double assembly (total moving weight)	460	C	
Performance:			
Worth per control rod assembly, dollars	0.47	C	
Maximum scram initiation delay, msec	200	C	
Minimum rod acceleration, g's	0.5	C	
Total worth in control rod bank, \$	21.60	G	11
Peak operating absorber temperature, °F	1090	C	

Basis designations:

- G Groundrule for task force
- A Assumption by task force
- CR Commonality basis or requirement established for both dispersed core task force and vertical core task force efforts
- C Calculated value determined by task force
- R Design requirement established by task force.

TABLE 3.VI. (contd)

Notes

1. Data provided by General Electric Co., Sunnyvale, California.
2. Study bases were as follows: (1) number of open test positions will be 10% of total number of fueled core subdivisions, to the nearest whole number; (2) open test position will consist of 75% of equivalent driver fuel assembly; (3) provisions will be included in plant design for a total of six closed loops, one loop being located in the peak flux region; and (4) closed loop sizes will be chosen to provide a minimum test envelope diameter of 2.5 in. to 3.0 in., with the larger dimension preferred.
3. Study bases were as follows: (1) a peak linear heat rating at nominal fuel power of 14.4 kW per ft of fuel pin, (2) a cladding wall thickness-to-diameter ratio given by a 0.016 in. wall for a 0.250 in. fuel pin OD; (3) a maximum cladding temperature of 1100 °F; (4) fission gas plenum length equal to fuel column length; (5) a target peak fuel burnup of 80,000 MWd/ton; (6) reactor nameplate rating of 400 MW_t; (7) core average coolant $\Delta T \leq 350$ °F; (8) a fuel column length in the 32 to 36 in. range.
4. Coolant inlet temperature was assumed by task force to be 600 °F, equal to the coolant inlet temperature being used for calculational purposes by the dispersed core task force.
5. Based on a 0.235 in. fuel pin (commonality requirement).
6. A total of 24 open test positions are actually possible in the core design generated by the task force. For calculational purposes and on a commonality basis with note 2, twelve open test positions were assumed.
7. PNL calculations indicated that a 6 in. long axial reflector directly above and below the core is a reasonable design value. This length was used by the task force.

TABLE 3.VI. (contd)

Notes (contd)

8. Design assumption by task force.
9. Computed values.
10. Clearance between fuel assemblies was arbitrarily assumed to be 0.030 in., no allowance for possible stainless steel swelling has been included as of this writing.
11. Study bases were as follows: the combined control-primary safety system for the vertical core shall have a worth of 6.5% $\Delta k/k$ and the secondary shutdown safety system for the vertical core shall have a worth of 3.0% $\Delta k/k$.

III. GENERAL TECHNOLOGY

P. L. Hofmann

HIGHLIGHTS OF REACTOR AND SYSTEMS ANALYSIS

R. E. Peterson

- Transient analysis for manual reactor scrams has identified heat transport system characteristics to minimize reactor coolant-temperature transients. A 2 sec time constant for an eddy-current clutch primary drive is ideal, and the time between reactor scram and start of pump coastdown must be minimized.
- Analysis of loss of plant power accidents for the first core shows that adequate cooling can be provided by emergency pony motor pump drives or by natural circulation.
- Transient analysis of the reactor and heat transport system following a 2 ft² break downstream of the pump in one primary loop shows that the main heat transport system can be used to hold the reactor coolant temperatures safely below the boiling point provided the check valves and process controllers and switches operate properly. Furthermore, the analysis is limited to the first core with the initial bulk coolant reactor outlet temperature at 900 °F, and it was assumed that a backup cooling method is available to remove decay heat after the main heat transport system becomes inoperative due to loss of primary coolant.
- The FFTF hybrid system simulation was greatly improved including the following accomplishments:
  1. Multiloop model capability. Two complete heat transport circuits (including pumps, pipes, IHX and DHX) are simulated with each capable of simulating one, two, three or four main coolant loops.

2. Pony motor models. Pony motors may automatically be switched into the pump drive simulation when the shaft speed reaches a preset value, corresponding to a mechanical override clutch.
  3. Digital printout in addition to normal stripchart recording. Selected variables may be monitored during a simulated transient at a wide choice of time intervals.
- A mathematical model of the Heating and Ventilation (H&V) System for use on a digital dynamic systems analyzer computer code is approximately 50% completed. The significant air-atmosphere spaces in the FFTF complex are being included in a transient study to determine the H&V System capacity required to prevent excessive space temperatures during a loss of normal electrical power. From this result the emergency electrical power requirements for the H&V System may be determined.
  - Initial design analysis for the FTR closed test loops to evaluate the testing capability of several alternate designs was completed. Functional testing requirements, identification of critical components, and associated research and development necessary to provide the technology for design and fabrication are objectives of the study. Results to date consist of tradeoff analysis of six variables--power; primary, secondary, and air flow rates; as well as intermediate and air blast heat exchanger sizing. The base conceptual design consists of a 3 in. diam re-entrant test section, counterflow intermediate heat exchanger and mixed crossflow-counterflow air blast heat exchanger.
  - Detailed thermal analysis of the temperature peaking underneath the wire wrap fuel pin spacer in the tricuspid pinch has shown that area to be 20 to 25 °F above the

nominal cladding surface temperature for the hot channel peak power point in the core. It may be concluded that the presence of the wire wrap does not cause severe temperature peaking in the fuel cladding.

- Combined stress calculations indicate that end-of-life stress levels for the cladding may be severe, thus limiting burnup. The CYGRO computer program is now operational for steady-state mechanical analysis of FTR fuel pins. The PECT computer program is being modified for transient mechanical analysis.

B. REACTOR AND SYSTEMS ANALYSIS

R. E. Peterson

1. Systems Analysis

H. G. Johnson

- a. Digital Simulation of the Reactor and Heat Transport System [This work performed for Systems 30 (Reactor) and 50 (Heat Transfer)]

C. D. Flowers and H. G. Johnson

(1) Manual Reactor Scrams and Loss of Plant Electrical Power. A digital simulation program of the reactor and primary heat removal loops was used to calculate the manual scram and loss of plant electrical power (BPA power) transients. One of the two modeled heat removal circuits accounts for two of the system loops, while the other simulated loop accounts for the third loop. The reactor model simulates inlet and outlet mixing plenums and three coolant flow paths through the reactor; tube coolant. Leakage coolant and radial reflector coolant. Multinode simulation of the fuel, cladding, and coolant regions is used with finite difference techniques applied in the tube and leakage coolant regions. The reactor simulation includes point kinetics with reactivity feedbacks accounting for Doppler, sodium temperature, and structural effects. Fission product decay power is simulated by a three-group approximation to the estimated FTR core decay curve. Fuel and tube coolant temperatures are calculated for an average channel in the reactor. Hot channel temperatures are calculated by the following equation:

$$(\Delta T)_{\text{Hot Channel}} = \alpha_H (\Delta T)_{\text{Avg}}$$

$$\alpha_H = \text{hot channel factor}$$

$$\alpha_H = 1.4, \text{ assumed}$$

$$(\Delta T)_{\text{Avg}} = \text{core } \Delta T \text{ for the average channel}$$

The simulation of the primary heat removal circuits accounts for the complete thermal and hydraulic description of all components in the system. More specifically, the primary simulation includes:

- Pump characteristics relating flow, speed, head, torque, and efficiency
- Transport time delays between components
- Eddy-current clutch characteristics relating slip, torque, and clutch excitation
- Process controllers for weight flow control under normal operation
- Process controllers for pump speed control for emergency operation using pony drive motors
- Multinode intermediate heat exchanger model
- Natural circulation head due to component elevation differences.

Since the secondary and tertiary models are not included, we have assumed a constant secondary coolant cold leg temperature entering the IHX. This is a valid assumption for short transients (less than approximately 100 sec) because the coolant transport time in the secondary pipe runs is quite long in a flow coastdown situation. At the loss of power or the initiation of the flow decay following manual scram, the secondary flow rate through the IHX is decayed in accordance with data obtained from the secondary loop digital simulation model.

The manual reactor scram transients consist of a 1 g rod acceleration scram (15\$ total reactivity), a time delay of magnitude zero to one second, and a flow decay initiated by either deenergizing both the pump motors and the clutches or by just deenergizing the clutches. Clutch time constants from 0.5 to 3.0 sec were investigated. Emergency controller and automatic pony motor operation, as explained below for the loss of BPA transients, was employed for manual scram transients.

The loss of BPA electrical power transients with eddy-current clutch pump drive results in:

- Loss of BPA power
- 3/4 sec delay to scram of core, eddy-current clutch deenergized at scram
- Reenergized clutch causes emergency pump speed controller to take over at measured pump speed of 300 rpm; Emergency controller setpoint for pump speed is 100 rpm.
- Emergency cooling provided by:
  - Pony motors either engaged automatically at motor speed of 125 rpm or 20 seconds after reactor scram, or
  - Flow decay to natural circulation cooling.

The clutch was continuously energized until the emergency controller took over for approximate simulation of a wound rotor motor drive. Reactor core to IHX centerline elevation differences (Z) of 6, 12, and 18 ft were investigated for the natural circulation cases. Reactor scram strengths of 15\$ and 6\$ were studied also. An eddy-current clutch time constant of two seconds was assumed for all of the loss of BPA transients.

The results of the manual scram transients indicate that a 2-sec eddy-current clutch time constant (time for clutch current to decay approximately 63% for a step input change) provides the most desirable flow decay, and that the flow decay should be initiated as soon as possible after the scram occurs (immediately, if possible). Also, it appears desirable to initiate the flow decay by deenergizing the clutch and maintaining power to the pump motor. Average reactor temperature results for a manual scram with a 2-sec clutch time constant and immediate flow decay are given in Figure 3.19.

FIGURE 3.19. Reactor Temperature Responses for Manual Scram with Immediate Flow Coastdown; Coastdown Initiated by Clutch Deenergization; Emergency Controller Gain is 0.25; 2-sec Clutch Time Constant

For the range of parameters and the transients studied, the loss of BPA power studies indicates that an emergency core cooling system is not required to prevent reactor hot-channel coolant boiling (1620 °F) for the initial FTR core with a 900 °F outlet temperature. Pony motors or natural circulation (reference concept reactor centerline to IHX centerline height difference of 12 ft) will provide adequate core cooling for the loss of power transient. Typical loss of BPA transient results (reactor temperatures) are given in Figures 3.20, 3.21, 3.22 and 3.23.

(2) Loss of Primary Coolant

(a) Step Flow Changes in Coolant to Reactor. Preliminary calculations were made to determine reactor coolant temperatures for various step changes (downward) in primary coolant flow to the reactor core. The digital code reactor model described in the preceding section of this report was used for simulation of the first core. The analysis indicates that without a reactor scram the reactor hot channel coolant will not reach boiling (1620 °F) in the initial FTR core (900 °F outlet temperature) for flow steps of 45 and 55%, but hot channel boiling will occur for a flow step of 65%. An additional run, however, indicated that hot channel boiling can be prevented by scrambling the reactor (1 g, 15%) in less than 3/4 sec after a 65% flow step.

(b) Computer Code for Loss of Coolant Analysis. A digital computer code is being developed for transient analysis of loss of coolant accidents for pipe breaks of any size downstream of the cold-leg-located pump in one of the two, three, or four primary loops. A check valve may or may not be located downstream from the break.

The model includes a complete description of primary loop hydraulics, coolant transport delay time in pipe runs,

FIGURE 3.20. Reactor Temperature Responses for Loss of BPA Power, 3/4 sec Scram Delay, 1 g (\$15) Reactor Scram, Pony Motors Engaged Automatically at 125 rpm.

FIGURE 3.21. Reactor Temperature Responses for Loss of BPA Power, 3/4 sec Scram Delay, 1 g (\$15) Reactor Scram, Pony Motors Engaged 20 sec After Scram

FIGURE 3.22. Reactor Temperature Responses for Loss of BPA Power, 3/4 sec Scram Delay, 1 g (15) Reactor Scram, Flow Decay to Natural Circulation with $\Delta Z = 6, 12,$ and 18 ft

FIGURE 3.23. Hot Channel Core Outlet Temperature Responses for Loss of BPA Power, 3/4 sec Delay, 1 g (15 ϕ) Reactor Scram Flow Decay to Natural Circulation with $\Delta Z = 6, 12,$ and 18 ft

pump and drive characteristics, IHX heat transfer, primary coolant flow controllers, emergency pump speed controllers, trip switches for scram and initiation of pump and flow coast-down, and reactor kinetics, heat transfer and mixing.

Results are shown in Figures 3.24, 3.25 and 3.26 for a 2 ft² break in one of three primary loops, with the break downstream of the pump but upstream of the check valve. The break is relatively large, with the opening larger than the cross-sectional area of the 18 in. pipe which ruptures. The break occurs just outside of the vessel. The sequence of events is as follows:

1. The break occurs at 2 sec. The break is instantaneous to the full 2 ft² size.
2. The reactor is scrammed at 4 sec; the scram is 1 g, 15\$.
3. Flow coastdowns on all three pumps are started at 5 sec.
4. The check valve in the broken loop starts to close at 2.05 sec, the instant of flow reversal.
5. The check valve is fully closed at 5.05 sec; 3 sec closing time because of a dashpot.

Some key assumptions are noted as follows.

- The system is initially at 400 MW, with 900 °F hot leg and 600 °F cold leg temperatures, and with three loops in operation.
- The normal coolant level in the reactor vessel is 12 ft above the outlet nozzles.
- The pumps are double suction, which do not become NPSH-limited and are adequately submerged to prevent gas entrainment. The pump barrels provide an ample coolant reservoir until the pump coastdowns start, after which there should be no NPSH or submergence problems. A pump shaft length of 30 ft is used, with the impeller 24 ft below the normal coolant level in the reactor vessel.

FIGURE 3.24. Primary Coolant Flow Rates

- The primary coolant is on direct weight flow control in each loop.
- The reactor power is under manual control. The control rod position is fixed.
- The reactor core hot-channel factor is 1.4.

The results show that coolant boiling will not occur for a two square foot break if the system functions as outlined.

FIGURE 3.25. Reactor Temperatures and Power Data for Loss of Coolant Accident with 2 ft² Break in One of Three Primary Loops. Break Occurs Downstream of Pump and Upstream of Check Valve.

FIGURE 3.26. Reactor Coolant Level and Pressure at Break Data for Loss of Coolant Accident with 2 ft² Break in One of Three Primary Loops. Break Occurs Downstream of Pump and Upstream of Check Valve.

Future. Work is continuing on model improvements. For example, the coolant levels in the pump barrels are to be calculated and a reactor power process controller is to be available.

- b. Analog-Hybrid Simulation of the Reactor and Heat Transport System [This work performed for System 30 (Reactor) and System 50 (Heat Removal)]

A. L. Gunby and R. D. Benham

(1) Model Development Completed

(a) Pony Motor Models. Pony motors for each sodium pump have been simulated and added to the hydraulics portion of the simulation. Pony motor simulation is achieved by switching into the simulation of the sodium pump motor a different torque-slip characteristic corresponding to the smaller motor. Switching occurs at a preset motor speed which corresponds to a mechanical override clutch. Runs have been made for loss of motor power to establish flow decay curves for both primary and secondary coolant loops, as shown in Figures 3.27 and 3.28. Motor power was lost at time zero and when the motor speed reached 125 rpm the pony motors began providing torque to the system.

(b) Multiloop Hybrid Simulation. Addition of a second coolant loop to the hybrid simulation has been completed. The simulation is now capable of operating in either three- or four-loop operation with either coolant loop capable of simulating one, two, three, or all four of the loops. The improved simulation represents a reactor vessel and kinetics, and two parallel independent coolant loops capable of single or mixed loop operation and complete with time delays and pony motor backup. The system has been operated successfully with constant heat transfer coefficients; incorporation of variable heat transfer coefficients are scheduled for early completion.

FIGURE 3.27. Flow Transient Hybrid Simulation for Primary Coolant Loop

FIGURE 3.28. Flow Transient Hybrid Simulation Secondary Coolant Loop

Application of extensive hybridization techniques has resulted in approximately 60% of the system operating digitally, as well as supplying beneficial time saving functions during setup and data readout. The functional responsibilities of each mode are summarized in Figure 3.29 and itemized below.

1) Analog Portion

- a) Reactor vessel heat transfer and neutron kinetics
- b) Coolant Loop A
  - Intermediate Heat Exchanger (IHX)
  - Dump Heat Exchanger (DHX)
  - Primary and Secondary Motor Speed and Flow Integrations
- c) Coolant Loop B
  - Primary and Secondary Motor Speed and Flow Integrations

1) Digital Portion

- a) Reactor Vessel
  - Inlet plenum mixing and lower support metal heat transfer
  - Variable heat transfer coefficient for fuel to coolant (to be added).
- b) Coolant Loops A&B
  - IHX and DHX variable heat transfer coefficients (to be added)
  - Pump, motor, and clutch torques; pump speed; inertial and hydraulic inputs for motor speed and flow integrations (performed by analog)
  - Flow transport time delays
  - Tertiary (air) flow calculations.
- c) Coolant Loop B Only
  - Intermediate Heat Exchanger
  - Dump Heat Exchanger

FIGURE 3.29. FFTF Dynamics Hybrid Simulation

(c) Model Improvements. Improvements on the simulation output were added to print selected variables with time in such a manner that any desired time interval between printings can be selected. This digital printout provides a valuable addition to the standard analog strip chart output. For example, output values could be printed at 1/5 sec intervals for 1 sec, 1 sec intervals for the next 5 sec, 1/10 sec intervals for 1 sec, and the remainder at intervals of 5 sec. A maximum of 25 points in time plus the initial conditions are available for printout now. This could easily be expanded with the use of more memory space.

Improvements on the simulation input were added so that operator instructions to perform any type of transient may be used. Any combination of actions occurring at any particular time can be achieved simply. For example, the simulated reactor may be scrammed at time zero, the primary flow shut off 1 sec later, and the DHX fans shut off 20 sec later. The program then prints out in tabular form a description of what has occurred on the same output sheet that has the digital printout.

(2) Heat and Transfer Coefficients. Calculations are complete for the heat transfer coefficients of the core, IHX, and DHX as functions of temperature and flow. New programs in FORTRAN IV were written for this purpose, with emphasis on ease of changing design data (tube size, etc.). Function generators for the heat transfer coefficients are now under development with the following bases:

- Core heat transfer based on fuel temperature only
- IHX heat transfer based on both primary and secondary flow and temperature
- DHX heat transfer based on air flow and air temperature.

(3) Hybrid Subroutines. All operator action programs previously in use in the system have been placed on magnetic

tape. They can be called into a buffer area of memory (about 600 locations) during operation and provide increased flexibility for the simulation.

The program being used to initiate scram actions has been rewritten to enable it to monitor any selected variable giving a signal to initiate a particular action when that variable reaches a particular trip point. With this program, it is possible to set up in a matter of 2 or 3 min any transient condition with as many as 10 different actions taking place on separate trip signals.

The programs presently on magnetic tape are capable of the following:

- (a) Printing out continuously coolant flow rates, pump speeds, motor speeds, or pump heads. A separate program exists for each set of parameters. Headings are typed defining the variables.
- (b) Printing out continuously any number of arbitrary variables, (maximum of six at present). The desired variables are merely typed directly into an intermediate program (DDT--"digital debug tape") with the teletype.
- (c) Calculating an energy balance of the heat transported through the heat removal system. This is useful in determining proper operation of the simulation..
- (d) Listing the steady-state values of all digital and analog variables which need "initial condition" values.
- (e) Initiating ten successive actions, each determined by a trip point established for a particular variable. For example: At time zero, some action; e.g., A, occurs. When variable X reaches trip point Q, action B occurs. When variable Y reaches trip point R, action C occurs, etc. The action programs presently available are capable of: 1) Initiating reactivity insertions (either positive or negative) at any desired ramp rate to any desired final value; 2) Simulating loss of electrical power to any magnetic clutch or pump motor; 3) Stepping the desired value

of magnetic clutch current to any value and at any time constant; 4) Initiating a ramp closure of the tertiary flow valve on either coolant loop; 5) Initiating a partial loss of tertiary flow (simulates loss of one fan, etc.); 6) Simulating loss of power to tertiary fans leaving only natural convection flow.

Future. Three efforts will receive major emphasis during the next quarter:

- Reworking the reactor model for greater refinement in the fueled region. It appears that multiple nodes in the core region will provide greater accuracy, particularly during flow transients.
- Addition of variable heat transfer coefficients for the fuel, IHX's, and DHX's. Heat transfer for each component will be a function of flow and temperature, as needed.
- Addition of control loops to the total system simulation. Various configurations of process and reactor control will then be evaluated.

c. Heating and Ventilating System Transient Study (This work performed for System 20. Structures)

R. N. Madsen

The Heating and Ventilation (H&V) System is designed to control the air atmosphere portions of the FFTF complex to acceptable temperature and humidity conditions during normal operation. The H&V System must also be effective in preventing excessive ambient temperatures during a prolonged loss of normal electrical power or a containment incident. (Maximum space temperatures during a loss of normal power have been tentatively set at 100 °F; for accident or other conditions involving containment, a maximum temperature of 120 °F has been tentatively selected.)

subject of this study. The results of this study will make possible an accurate determination of the H&V System emergency power requirements.

The initial phase of this study was an evaluation to determine the essential portions of the FFTF complex which required study. An evaluation of a typical building concept resulted in the selection of eleven spaces to be included in the study. These spaces, in order of priority of analysis, were: 1) Containment Work Area, 2) Refueling Hot Cell, 3) Fuel Handling and Examination Work Area--Four Levels of the Secondary Sodium Building, 4) Pump Level, 5) Piping Level, 6) Cleanup Level, and 7) Sodium Storage and Auxiliary Equipment Level, 8) Gas Systems Space, 9) NPTF Reactor Cell, 10) NPTF Equipment and Fuel Storage, and 11) Sodium Laboratory.

The next phase has been the creation of a detailed mathematical model using a digital dynamic systems analyzer computer code. This phase is approximately 50% complete. This model represents all of the significant thermal processes occurring in each space: heat sources, including solar heat gains, lighting, equipment, etc.; heat sinks, including both the main air-conditioning system and space recirculating air-conditioning units; and thermal storage effects of the building mass, air mass, and equipment within the spaces. The model of the Containment Work Area, Refueling Hot Cell, and Fuel Handling and Examination Area has been completed. Detailed simulations of the main and recirculating air-conditioning systems, together with simulation of their automatic process and temperature controls, have also been completed. These air-conditioning system simulations have included and been closely coordinated with manufacturers' typical performance ratings to ensure the accuracy of the model.

Many additional analyses have been performed in the course of the model development. These analyses have included:

- Study of transient temperature behavior in concrete shielding walls, including gamma heating effects

- Damping of diurnal solar heat gain in a concrete shielding wall
- Computation of thermal phase shifts (time lags) for each of the anticipated types of building construction.

Future. The digital simulation should be completed during the ensuing quarter at which time a program of analysis using the model will be initiated. The following 6-step analysis is planned for two units in the main air-conditioning system, three recirculating units in the Containment Work Area (CWA), two units in the Refueling Hot Cell (RHC), and three units in the Fuel Handling and Examination Area (FHE).

- 1) Shutdown of one main air-conditioning (AC) fan and one main exhaust fan.
- 2) Shutdown of all previous items, plus the second main AC fan. (One exhaust fan to remain on to maintain a minimum air flow.)
- 3) Shutdown of all previous items plus one recirculating AC unit in CWA and FHE, and RHC.
- 4) Shutdown of all previous items plus one more recirculating AC unit in CWA and FHE.
- 5) Shutdown of all previous items plus reduction in the speed of the remaining main exhaust fan.
- 6) Shutdown of all previous items plus one more recirculating AC unit in CWA, RHC and FHE. This shuts down all recirculating units. (The other spaces will be added to this program as the systems in these spaces are more exactly defined and included in the model.)

These reductions in AC system performance will be in conjunction with a simulated loss of normal electrical power accompanied by a reactor scram and associated reductions in system heat gains. Since the behavior of the AC system is dependent on the time of day and year because of solar heat loads and outdoor air temperature, this transient study was scheduled for the worst case design conditions at mid-afternoon on August 1.

The most important results of these transient studies will be the determination of the average room temperature versus time following the outlined shutdowns. From these results it is expected that the optimum emergency H&V system operation may be determined.

d. Computer Code Development

G. H. Dickerson

(1) Digital Simulation Code. The DYNASAR method analyzes the dynamic response of an arbitrary linear or nonlinear system, including feedback controls. Mathematical representations of physical systems are built up using 50 different types of boxes. These can be interconnected arbitrarily on the input to the program to duplicate the physical system as would be accomplished on an analog computer. This gives extreme versatility in the system that can be analyzed. Boxes may include simple logic, general transfer functions (or differential equations), summaries, tables, limits, etc. The program will handle a maximum of 1250 function boxes which may be in combinations of up to 200 of the input and/or transfer function types and the remaining functions comprised of algebraic logic functions. The code has been extensively debugged and is ready for general use with the locally available Univac-1108 computer.

(2) Piping Flexibility Analysis Program. This particular version, entitled MEL-21, is a modification of the original MEC-21/7094 with the provision of the Nuclear Piping Code B31.7 added. Contact has been established with personnel at the Naval Ship Research and Development Company, Annapolis, Maryland, for additional information regarding the program and its inputs. When the required information is received, the program will be placed on the Univac-1108 for general use.

- e. Closed Test Loop Analysis (This work performed for System 60. Irradiation Testing)
C. E. Leach and D. L. Koreis

The purpose of the closed test loop analysis is to evaluate the testing capability of several alternative designs in light of functional testing requirements and to identify critical components and associated Research and Development necessary to provide the technology for design and fabrication. The analysis considers all major heat transfer and fluid flow components on the closed loops.

During the quarter the majority of information necessary to accomplish the study was gathered. This includes:

- Preliminary functional testing requirements--a description of power level, fuel pin size, coolant flow and temperature for four potential LMFBR test fuels, and one materials test
- Description of a preliminary closed loop conceptual design--piping runs, test section, heat exchanger components and insulation
- Preliminary safety requirements
- Applicable pressure vessel and piping codes
- Reactor operational requirements including lifetime, number of scrams, etc.

A computer thermal-hydraulic model of the loop consisting of about 180 temperature points has been formulated. Both primary, secondary and air blast heat exchanger have been programmed and separately checked out on a newly modified code, the Thermal Analyzer Program - Loop (TAP - Loop). The model includes the reentrant test section, test fuel, intermediate heat exchanger, primary pumps, primary heater, secondary pumps, air blast heat exchanger, interconnecting piping, and parasitic heat losses to atmosphere, reactor sodium-pool and cover gas. Trace heating is also included. The model may be used for either steady-state or transient analysis.

A conceptual design of 4 MW, 300 °F temperature rise at 800 °F coolant bulk outlet has been selected to ensure testing capability of low temperature, high power density fuels. This is the most demanding condition for high-power heat rejection.

Preliminary to parametric analysis of this design with TAP - Loop, several studies have been performed to size the IHX (intermediate heat exchanger) and ABHX (air blast heat exchanger) as well as to assess the accuracy of the ABHX computer model. The latter comparison is shown in Figure 3.30 and portrays the heat exchanger performance in terms of effectiveness and number of transfer units (NTU_{max}). The continuous curve is for an infinite number of passes (approximately four) in a cross-counterflow exchanger with equal capacitance ratio. The three data points are from the computer model of the same type of exchanger with four hairpin passes and with capacitance ratio of 0.47. The points should fall above the curve as the effectiveness of all exchangers improve with capacitance ratios of less than 1. We believe the discrepancy is induced by the coarse temperature mesh used in the exchanger model.

The second preliminary study was performed to define the relative sizing of the ABHX and IHX. Figure 3.31 portrays the effectiveness tradeoff between ABHX and IHX to achieve the 4 MW, 300 °F delta-T and 800 °F fuel outlet test for 120 °F air inlet to the ABHX. Also shown is the effect of selection of other than a counterflow exchanger for the IHX. The counter, parallel, and crossflow curves also show the effect on IHX areas. Figure 3.32 portrays the critical nature of the ABHX effectiveness for sizing the ABHX. To illustrate, consider the following three examples:

FIGURE 3.30. Heat Transfer Effectiveness as a Function of Number of Transfer Units for an Infinite Number of Passes; Multipass Cross-Counterflow Exchanger

FIGURE 3.31. Air Blast Heat Exchanger (ABHX) Heat Transfer Effectiveness as a Function of Intermediate Heat Exchanger (IHX) Heat Transfer Effectiveness for Counter, Parallel and Crossflow Designs

FIGURE 3.32. Number of Transfer Units as a Function of Air-Blast Heat Exchanger Effectiveness

<u>Example No.</u>	<u>Heat Exchanger</u>	<u>NTU_{max}</u>	<u>E, %</u>
1	IHX	3.5	90
	ABHX	2.5	71.5
2	IHX	2.5	82.5
	ABHX	2.8	76
3	IHX	1.5	70
	ABHX	Large	88

These data are derived from the curves by selecting an IHX size and intercepting the counterflow curve on Figure 3.31 at Point (A), determining its effectiveness on the abscissa, thence to Point (B) and determining the required ABHX effectiveness on the ordinate. The corresponding ABHX NTU is found on Figure 3.32. Two conclusions are: The IHX must be of counterflow design and must be of sufficiently high effectiveness to achieve reasonable ABHX sizing. It is possible to select a sufficiently small IHX to preclude successful design of an ABHX capable of operating at the selected design condition. An approximate 80 percent effective ABHX has been selected for the base case. It will consist of about 18 extended surface tubes with four hairpin passes.

Future. Three different sized loops all with capability for 1200 °F coolant exit temperature from the test fuel will be investigated and 1400 °F coolant exit temperature with bypass attemperation will also be investigated.

2. FFTF Fuel Design and Analysis (This work performed for the Driver Fuel Development Program)

P. D. Cohn

a. Temperature Peaking Under the Wire Wrap Spacer

D. C. Kolesar

The circumferential variation of the coolant temperature (temperature "peaking") in the neighborhood of the wire wrap (30 mil) of the hottest FFTF fuel pin was investigated using

MANTA⁽¹⁾ (PNL modified version), a code written at General Electric (APED). The temperature peaking is a consequence of lower coolant flow rates in relatively confined spaces. Calculations were made in three dimensions to determine an upper limit on the temperature peaking. Results disclose only a modest circumferential variation of temperature. The seriousness of this temperature peaking depends upon the particular related phenomenon to be investigated, e.g., thermal stress, reactor safety analysis, and pin life.

The primary parameters employed in this analysis were:

Pin OD, in.	0.250
Wire Wrap OD, in.	0.030
Inlet Coolant Temperature, °F	550
Average Core Delta-T, °F	350
Peak Pin Linear Power, kW/ft	14.4
Active Fuel Length, in.	32.0
Axial Power Peaking Factor	1.24
Radial Power Peaking Factor	1.30

The composition and power distribution of the fuel pin were predicted by the SINTER Code.

The physical problem was approximated by a model consisting of 1/12 of a "unit cell" (30° slice) centered on the hottest fuel pin (Figure 3.33). The unit cell is composed of a single fuel pin and a fraction of the coolant established by symmetry

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1. S. F. Armour and D. L. Smith. MANTA-Mixing Analyzed Nodal Thermal-Hydraulic Analysis, GEAP-4805. General Electric Company, Schenectady, N. Y., February 1965.

FIGURE 3.33. Unit Cell Model Analyzed by MANTA

based upon geometric considerations. One-twelfth of a unit cell is the smallest part that can be isolated for analysis because of its nearly adiabatic "boundaries." Only the active fuel region was considered in the analysis. The coolant velocity distribution was characterized by fully developed parallel flow as predicted by Koreis⁽¹⁾ with a mean velocity as predicted by Kolesar.

1. T. J. Bennett et al. Thermal Hydraulic and Mechanical Aspects of FTR Fuel Pin Spacer Design, BNWL-750. Pacific Northwest Laboratory, Richland, Washington, September 1968.

Figure 3.34 illustrates the nodal mesh employed in the geometric model. Three regions were considered--mixed oxide fuel, 316 SS, and flowing sodium. The region in and around the wire wrap was divided for the convenience of the computer code. The active fuel was divided into 16 equal lengths each possessing the nodal arrangement displayed in Figure 3.34.

Both a restriction of MANTA and limitations imposed by the film coefficient library required that the wire wrap be oriented parallel to the pin and be placed as far as possible from the centers of the tricus channels. The actual configuration is one complete revolution about the pin per foot of length. However, the hypothetical arrangement should lead to conservative (larger than expected) temperature peaking results. On one hand, the tendency to enhance the peaking resulting from the longer path traveled by fluid "trapped" by the wire (thereby increasing exposure to a higher energy flux) as compared to fluid channeling down the center of the tricus region is disregarded. Neglecting this effect should be more than compensated by confining the wire wrap to the region where its influence is the greatest.

Most transport and thermodynamic property input data are those employed in similar calculations at PNL with the significant exception of the film coefficient. To predict the temperature variation about the pin circumference properly, it is necessary to employ a local heat transfer coefficient dependent upon polar angle with origin at the center of the fuel pin (see Figure 3.33). Koreis⁽¹⁾ has calculated such a heat transfer coefficient employing a variation of the method of Dwyer.⁽²⁾ Dwyer's method is a graphical determination by

1. T. J. Bennett et al. Thermal Hydraulic and Mechanical Aspects of FTR Fuel Pin Spacer Design, BNWL-750. Pacific Northwest Laboratory, Richland, Washington. September 1968.
2. O. E. Dwyer. "Analytical Study of Heat Transfer to Liquid Metals Flowing In-Line Through Closely Packed Rod Bundles." Nucl. Sci. and Eng., vol. 25, p. 347. 1966.

FIGURE 3.34. Subdivision and Nodal Network for 30° Slice of Cell Model

a complicated series of related plots of various field quantities and transport constants. Thereby, the necessary heat transfer coefficient (a function of polar angle) is deduced from the computed temperature field. This heat transfer coefficient was assumed here to be independent of axial position.

MANTA employs mass, momentum, and energy balances in conjunction with the equivalent diameter concept. However, the equivalent diameter concept is not strictly valid for coolant channels of the type displayed in Figure 3.34. This did not introduce any conceptual difficulty since the velocity profile was to be fixed and therefore only the energy balance in MANTA was needed (essentially independent of equivalent diameter). Use of only the energy balance could not be accomplished directly so an iterative process was employed as follows:

- (1) The pressure drop was calculated by hand in one channel.
- (2) The equivalent diameters were adjusted to give this pressure drop in each channel.
- (3) The equivalent diameter obtained in Step (2) based upon MANTA results was corrected to adjust the input velocity profile identical to the output velocity profile. This sequence was repeated until velocity profiles converged sufficiently.

Steps (1) and (2) were based upon the fluid being relatively incompressible and turbulence not affecting interchannel mass transfer. Step (3) was required because of natural computer inaccuracies as well as the temperature dependence of some properties; e.g., viscosity. The iteration in Step (3) was considered complete when errors in the mass flow in each channel were of the order of 5% (0.1% for channels nearest the wrap) or smaller.

MANTA employs a diffusional mixing model with a multiplying factor, K , inserted to permit correlation with experimental data. Here results are reported for no mixing ($K = 0$), molecular diffusion ($K = 1$), and mixing characteristic of the conservative mixing parameter ($\beta = 0.01$) employed in earlier

COBRA code⁽¹⁾ calculations ($K = 5$). The COBRA calculations attempted to duplicate tightly packed bundle phenomena through this value of β ; however, the presence of wire wrap could require a β four times larger. The use of a K greater than unity probably implies eddy diffusion and this is certainly true for $K = 5$. The turbulent mixing parameter, β , used in COBRA is predicted from experimental results based upon full interchannel mixing and not for transfer between arbitrarily defined subchannels as illustrated in Figure 3.34. Whether a gross parameter such as can be employed for the local effects investigated here is not certain.

Figure 3.35 displays the temperature peaking in terms of the circumferential coolant temperature distribution at an axial location near the center of the fuel region. Figure 3.36 illustrates the temperature field in the neighborhood of the wire wrap at the same axial location and for the same degrees of mixing as provided in Figure 3.35. Figure 3.37 displays this peaking effect in terms of the temperature difference between the hottest and coldest regions of the sodium as a function of axial position. These ranges of temperature extremes are located immediately adjacent to the wire wrap and in the center of the tricuspid channel, respectively, and are the temperature extremes of Figure 3.35.

b. FFTF Fuel Subassembly Dynamics

F. E. Bard

The CYGRO computer program is now operational for prediction of the steady-state stress distribution in fuel elements.

1. D. S. Rowe. Cross-Flow Mixing Between Parallel Flow Channels During Boiling, Part 1. COBRA-Computer Program for Coolant Boiling in Rod Arrays, BNWL-371, Part 1. Pacific Northwest Laboratory, Richland, Washington, March 1967.

FIGURE 3.35. Circumferential Temperature Distribution in the Coolant near the Center of the Fuel Region for Various Degrees of Mixing

FIGURE 3.36. *Temperature Field near the Center of the Fuel Region for Various Degrees of Mixing*

FIGURE 3.37. *Temperature Peaking as Represented by the Maximum Circumferential Temperature Difference of the Coolant as a Function of Axial Distance Along the Fueled Portion of the Hottest Pin*

The PECT computer program is being modified to predict the transient stress distribution in fuel elements. The FTR fuel will experience large temperature changes during its operational life. Where temperature gradients are small, the following stress-strain relation can be used.

$$\sigma_{ij} = 2\mu \epsilon_{ij} + \sigma_{ij} \left[\lambda \epsilon_{kk} - (3\lambda + 2\mu) \alpha \Delta T \right] \quad (1)$$

where

σ_{ij} are the component of the stress tensor

ϵ_{ij} are the components of the infinite simal strain tensor

ϵ_{kk} is the first principle strain in variant

σ_{ij} are the components of the Kronecker delta λ and μ are the Lamé constants

α is the linear coefficient of thermal expansion

ΔT is the temperature change from zero stress state.

Note that the constants λ , μ , and α are independent of temperature. With large temperature gradients, however, the material properties are temperature dependent. Hence, the stress-strain relation must be reformulated as follows:

$$\dot{\sigma}_{ij} = 2\bar{\mu} \dot{\epsilon}_{ij} + \sigma_{ij} \left[\bar{\lambda} \dot{\epsilon}_{kk} - \left\{ 3\bar{\lambda} + 2\bar{\mu} \right\} \bar{\alpha} \dot{T} \right] \quad (2)$$

Note that the material constants are now a function of temperature. Equation (2) is the basic equation that will be used in the modified PECT program.

c. Cladding Stress Analysis

R. J. Jackson

The combined stress in the fuel pin cladding results from interactions of several stress contributions. There is no known technique for simultaneously calculating all stress values resulting from these interactions. One technique for

assessing the net effect of combining several stresses is described in ASME Code Section III. This code does not have jurisdiction over the fuel pin design but the techniques which have been developed may be helpful. The results of preliminary calculations are shown in the following table for peak pin steady-state operation. The stress categories used are as defined in the code. The reference pin geometry, i.e., 0.250 OD with 30 mil spacer wire, was the basis of the stress determinations.

	Axial Stress, <u>psi</u>	Hoop Stress, <u>psi</u>
<u>General Primary Membrane Stress</u>		
Fission Gas Pressure	+3,000	+7,000
Fuel Swelling (estimated)	+6,000	+~29,000
<u>Secondary Membrane Plus Bending</u>		
Thermal Stress	+16,800	+16,800
Thermal Camber	+29,400	0
Swelling Induced Bending (est)	+33,000	0
<u>Peak Stress</u>		
Differential Thermal Expansion	-1,080	0
Temperature Peaking Under Spacer	-20,800	-20,800

+ indicates tension

- indicates compression

Stresses resulting from fission gas pressure, fuel swelling, and swelling-induced bending are those associated with the end of pin life. Remaining stress values are assumed to remain constant throughout the pin life. The stress associated with fuel swelling has been estimated. Based upon extrapolated values for creep rates, a stress value of 30,000 to 40,000 psi was calculated to attain the predicted end of life total creep.

Therefore, the membrane stress was estimated to be 36,000 psi at end of life. Work is continuing to obtain a better estimate of the fuel-cladding interaction stress.

A comparison of the calculated (and estimated) stress to the allowable design stress intensity is shown in the following tabulation.

Stress Category	Calculated Stress Intensity, psi	Allowable Stress Intensity, psi	
		ASME Code	End of Life ^(a)
General Primary Membrane, P_m	36,000	15,600	32,200
Secondary Membrane plus Bending, $P_L + P_b$	88,200	23,400	48,300
Peak Stress, $P_L + P_b + Q + F$	66,220	60,000	--

a. Includes radiation hardening

The general primary membrane stress is not relieved by yielding; therefore, the allowable stress intensity is established as a percentage of the yield strength. The secondary membrane and bending stresses are relieved by deformation; therefore, the allowable stress can be greater than the yield strength. The allowable design stress intensity for the peak stress is determined by the fatigue strength of the material. The allowable stress intensity shown in the table for end-of-life takes credit for radiation enhanced yield strength.

The calculated stresses exceed their allowable stress intensity. In the absence of all other stresses, failure is not indicated by one of these stresses exceeding the allowable stress, which is somewhat conservative. The primary stress may cause failure if it exceeds the yield strength. The secondary stress by itself would have to exceed greatly the

yield strength before it would cause failure in a ductile material. The major component of the primary membrane stress is the cladding stress resulting from the swelling fuel. Within a practical range of cladding thickness little can be done to reduce cladding stress. The predominant components in the calculated secondary stress are the stresses resulting from thermal camber and swelling-induced bending. The thermal camber stress could be reduced by use of flow control inside the duct. The swelling-induced bending stress can be reduced only by eliminating it. This can be accomplished by limiting the fuel burnup or increasing the cladding strength in such a manner that the fuel pin diameter increase is less than 1%. If this is done, the calculated secondary stress will be 55,200 psi. This is near the allowable stress based upon the end-of-life material properties.

It is concluded that the reference conceptual fuel pin design, while not subject to code requirements, is much less conservative than ASME Code Section III requirements. If burnup is limited or cladding strength is increased such that the diametral increase is less than 1%, the reference design almost meets code requirements based upon end-of-life strength values. To meet the code requirements based upon beginning-of-life strength it would be necessary to reduce burnup to the extent that there is no fuel-cladding interaction force. It is believed that the realistic goal is to meet the code requirements based upon end-of-life material strength. Based upon the strain predictions reported in the February progress report (BNWL-712) p. 3.62 this corresponds to a peak burnup of 50,000 to 60,000 MWd/ton.

III. GENERAL TECHNOLOGY

P. L. Hofmann

HIGHLIGHTS OF REACTOR PHYSICS AND OPERATIONS

R. E. Heineman

The effects of sodium remaining in the core of a fast reactor during a large power excursion was investigated. There are effects which tend to both increase and decrease the magnitude of the excursion.

The 2-dimensional, diffusion theory code, 2DB, used for fast reactor criticality and burnup studies is described.

Calculations have been completed which indicate that there are no serious changes in the neutronics of the Nuclear Proof Test Reactor if the fuel module cans in the NPTR are tapered, so that sodium and stainless steel are relocated.

The Phase B FTR critical experiments in ZPR-III have been analyzed and the critical masses are consistently over-estimated by the calculations.

It has been found that the calculated attenuation of lower energy neutrons through sodium is significantly larger than that obtained experimentally.

The neutron flux in the inlet plenum upper tube sheet has been calculated in order to estimate the radiation damage to this structural component of the FTR.

The decay heat from irradiated plutonium, calculated with the code RIBD, is found to compare favorably with experimental measurements.

An estimate has been made of the nuclear heat generation in candidate control material poisons for the FTR.

C. REACTOR PHYSICS AND OPERATIONS

R. E. Heineman

1. Nuclear Analysis

W. W. Little

a. Considerations of Sodium in Conditions for Fast Reactor Large Excursion Calculations (This work performed for System 30 Reactor)

A. E. Waltar and R. A. Harris

The consequences of the presence of sodium during major nuclear excursions in fast reactors have traditionally been ignored. The argument usually advanced for disregarding the sodium effects is that a sodium voided core will result in a larger calculated energy release than one containing a normal sodium inventory because of the early disassembly pressures associated with sodium-in conditions. An investigation in which the effects of sodium were approximately accounted for during the entire excursion, however, indicates that the presence of sodium might cause a net increase in the potential destructive energy in such accidents.

There presently appear to be three phases of a major power excursion where the inclusion of sodium significantly changes the analysis--before, during, and following core disassembly. Emphasis will presently be focused on the first two phases, although results of a postnuclear excursion sodium mixing study are mentioned to enable conclusions to be drawn.

Prior to the disassembly phase, massive fuel melting could result in a positive reactivity feedback due to fuel compaction⁽¹⁾ and/or sodium voiding. If the normal sodium

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1. A. E. Waltar, W. W. Little and R. A. Harris. "Preliminary Analysis of Core Meltdown Transients in a Fast Test Reactor," Trans. Am. Nucl. Soc., vol. 11, No. 1, p. 333. 1968.

inventory is present at the time cladding failure occurs, a violent molten fuel-sodium interaction could take place resulting in extremely rapid sodium voiding. The magnitude of this voiding rate is of particular concern because: (1) the position of initial sodium voiding (cladding failure) will probably be near the axial center of the core--just the region in which the sodium void coefficient has its maximum positive value, and (2) in order for such voiding to be initiated, the core is most likely to be near prompt critical, so that even a few cents of rapid positive feedback is highly influential.

If the molten fuel-sodium interaction can approach peak thermodynamic efficiency, sodium pressures in the range of several hundred atmospheres might be attained. Using a simple piston ejection model and early estimates of the sodium worth in the FTR, ramp rates of the order of 200\$/sec just prior to core disassembly have been calculated.

The disassembly process itself is highly sensitive to the presence or absence of sodium, since this directly affects the volume available for fuel expansion. In a sodium voided core, fuel vapor constitutes the only disassembly driving force until temperatures near the critical point (~ 8000 °K) are reached. With sodium present, however, very large pressures characteristic of a heated, confined liquid are established early in the transient, thus leading to rapid disassembly and a relatively small energy release.

An equation of state for UO_2 applicable to the temperatures (3000 °K to ~ 6000 °K) and fuel densities (8.7 to ~ 7 g/cm³) encountered in "sodium-in" calculations has been generated. This equation was based on the liquid fuel density measurements of Christensen⁽¹⁾ and the vapor pressure measurements of

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1. J. A. Christensen. *Thermal Expansion and Change in Volume on Melting for Uranium Dioxide*, HW-75148. Available from Clearinghouse for Federal Scientific and Technical Information, Springfield, Virginia, 1962.

Ackemann.⁽¹⁾ Estimates of the pressure-temperature relationships in the pure liquid phase were obtained from the corresponding states tables of Hougen et al.⁽²⁾

This equation was used in the MAX⁽³⁾ disassembly code to determine the effects of sodium on the energy generated in the transient. In these calculations, corrections to the volume available to the fuel (and thus the pressures existing in the core) were made to reflect:

- (a) Fuel and core material movement,
- (b) Absence or pressure of sodium, and
- (c) the compressibility of sodium.

Table 3.VII contains a brief comparison for a class of accidents in a core similar to the FTR in which the initiating ramp rate was assumed to be 100\$/sec. The inclusion of sodium is seen to give rise to a significantly higher ramp rate at the time of disassembly (caused by rapid sodium voiding), a sizeably smaller total energy generation in the nuclear excursion--despite the larger ramp rate, but an overall available work which is perhaps higher than that for the sodium-out case. The latter range of numbers provide thermodynamic brackets for the manner in which the molten fuel-sodium mixture might expand down to one atmosphere.⁽⁴⁾

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1. R. J. Ackermann et al. "High Temperature Thermodynamic Properties of Uranium Dioxide," *Chemistry and Physics*, vol. 25, no. 6, p. 1089. 1956.
  2. O. A. Hougen et al. *Chemical Process Principles, Vol. II, Thermodynamic*, 2nd Ed., John Wiley and Sons, 1959.
  3. W. W. Little, Jr. *Max-A One-dimensional Maximum Hypothetical Accident Code in FORTRAN-IV*, BNWL-612, Pacific Northwest Laboratory, Richland, Washington, March 1968.
  4. A. E. Waltar, W. W. Little and R. A. Harris. "Preliminary Analysis of Core Meltdown Transients in a Fast Test Reactor," *Trans. Am. Nucl. Soc.*, vol. 11, no. 1, p. 333. 1968.

TABLE 3.VII. *Comparison of Sodium-in and Sodium-out Energy Release Calculations*

Sodium Conditions Assumed	Ramp Rate at Time of Disassembly, \$/sec	Energy in Molten Fuel at end of Nuclear Excursion, (a) MW-sec	Available Work, MW-sec
Out throughout excursion	102	4000	360 ^(b)
In throughout excursion	250	1200	250+600 ^(c)

a. *Energy equals zero at melting point in solid phase.*

b. *Fuel is assumed as the working fluid.*

c. *Sodium is assumed as the working fluid.*

b. 2DB, A 2-Dimensional Fast Reactor Burnup Code

W. W. Little and R. W. Hardie

A 2-dimensional, multigroup, diffusion theory computer code, 2DB,⁽¹⁾ was recently developed at PNL for use in fast reactor criticality and burnup analysis. Subsequent to the original version, significant improvements have been incorporated into the code. The following paragraphs give a brief description of the present operation and capability of 2DB.

The eigenvalue and flux profiles are computed by standard source-iteration techniques in which group rebalancing, successive over-relaxation with line inversion, and fission source over-relaxation are used to accelerate convergence. Adjoint solutions are obtained by inverting the input data and redefining the source terms.

Code options include the use of reflective, vacuum, and periodic boundary conditions. Criticality searches, permissible

1. W. W. Little, Jr. and R. W. Hardie. 2DB, A Two-Dimensional Diffusion-Burnup Code for Fast Reactor Analysis, BNWL-640. Pacific Northwest Laboratory, Richland, Washington, 1968.

during burnup as well as for fresh core conditions, can be performed on buckling, time absorption, material concentrations, and region dimensions. Both α and k_{eff} can be used as parametric eigenvalues.

Variable dimensioning is used to make maximum use of the available fast memory. Since only one energy group is in the fast memory at any given time, the storage requirements are insensitive to the number of energy groups.

Representative 2DB running times for problems involving typical fast reactor compositions are shown in Table 3.VIII. All problems are for an initial flat flux and a k_{eff} convergence criterion of 10^{-5} .

TABLE 3.VIII. *Representative Running Times for k_{eff} Calculations on a UNIVAC 1108*

<u>Case</u>	<u>Geometry</u>	<u>No. of Groups</u>	<u>Mesh Size</u>	<u>Approximate Time, min</u>
1	R-Z	2	20 x 20	0.2
2	R- θ	4	20 x 30	0.7
3	X-Y	5	40 x 40	2.3
4	R-Z	20	20 x 20	3.0
5	R-Z	20	33 x 29	6.1
6	R-Z	26	44 x 18	4.1

c. Numerical Study of Selected Acceleration Techniques for 2-Dimensional Fast Reactor Diffusion Calculations
R. W. Hardie and W. W. Little, Jr.

In the process of writing a 2-dimensional diffusion code, 2DB,⁽¹⁾ the relative merits of various acceleration schemes as applied to representative fast reactor problems have been investigated. Among the schemes studies are:

1. W. W. Little, Jr. and R. W. Haride. "2DB, A Two-Dimensional Diffusion Burnup Code," Computer Code Abstracts, Nucl. Sci. Eng., vol. 32, pp. 275-276. 1968.

- Line inversion
- Group rebalancing
- Flux over-relaxation
- Fission source over-relaxation.

Because the flux cannot be calculated explicitly at each space point in a 2-dimensional problem iteration by line⁽¹⁾ or by point must be used to compute the group-flux profiles. Line inversion has been found to be markedly superior to point inversion for R-θ problems (when the inversion is along areas of equal radius)--or, indeed, any problem with tight mesh spacing in one dimension. In addition, line inversion was found to be particularly attractive for problems involving periodic boundary conditions even though the matrix, in these cases, is not tridiagonal.

Group rebalancing has been found to be a simple scheme for increasing the eigenvalue convergence rate. It consists of balancing the total source and loss rate in each group before each group-flux calculation. With group rebalancing, note that a one-region problem with reflective boundaries would be solved exactly in one outer (fission-source) iteration.

Flux over-relaxation, by the simple and well-known expression,

$$\phi_2^v = \phi_2^{v-1} + \beta(\phi_1^v - \phi_2^{v-1})$$

where

$$\phi_2^v = \text{relaxed flux for } v\text{th iteration}$$

$$\phi_1^v = \text{unrelaxed flux for } v\text{th iteration}$$

$$\beta = \text{relaxation factor,}$$

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1. E. L. Wachspress. Iterative Solution of Elliptic Systems, Prentice-Hall, Inc., Englewood, N. J. 1966.

significantly enhances the inner iteration, or group-flux, convergence rate. Similarly, fission source over-relaxation, using an identical expression, has been found to significantly accelerate eigenvalue convergence provided that the fission source is renormalized after each relaxation. If the fission source is not renormalized, severe oscillations in eigenvalue can be encountered.

Table 3.IX shows representative running times with and without the four aforementioned accelerating techniques. The numbers shown in Table 3.IX are based on a typical fast reactor configuration in R- θ geometry with four energy groups. Observe that each of the four accelerating schemes induce an appreciable savings in running time.

TABLE 3.IX. *Effect of Various Acceleration Schemes on Running Time*

<u>Case</u>	<u>Description</u>	<u>Number of Outer Iterations</u>	<u>Running Time on a UNIVAC-1108, min</u>
1	Base Case, Using All Four Acceleration Schemes	9	0.62
2	No Line Inversion	113	7.24
3	No Group Rebalancing	10	0.67
4	No Flux Over-relaxation	16	1.10
5	No Fission Source Over-relaxation	14	0.94

2. Experimental Physics

R. A. Bennett

- a. Effects of Tapered Module Cans in the NPTR (This work performed for System 30. Reactor)
Q. L. Baird and J. V. Nelson

The FTR shroud tubes or ducts are of constant cross section at all axial positions in the active core region. Fuel module cans in the NPTR are tapered in order to contain all of the sodium (or sodium simulant) associated with the dispersed conical core.⁽¹⁾ The effects of this relocation of stainless steel and sodium have been investigated. To carry out the calculations, the fundamental mode code FCC-IV⁽²⁾ and the one-dimensional transport code DTF-IV⁽³⁾ (S-8) were used. The cross sections were taken from a PNL modified 26 group Bondarenko cross section set.⁽⁴⁾

The various geometries and compositions used for the calculations are given in Table 3.X. The NPTR subassembly model, as taken for this analysis, represents the maximum dispersion of the module can (i.e., at the top of the active core zone). In changes of the stainless steel

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1. Progress Report, Fast Flux Test Facility Reference Concept, BNWL-470. Pacific Northwest Laboratory, Richland, Washington, August 1967.
  2. R. W. Hardie and W. W. Little. FCC-IV, A Revised Version of the FCC Fundamental Mode Test Reactor Code, BNWL-450. Pacific Northwest Laboratory, Richland, Washington, August 1967.
  3. K. D. Lathrop. DTF-IV, A Fortran-IV Program for Solving the Multigroup Transport Equation for Anisotropic Scattering, LA-3373. Los Alamos Scientific Laboratory, Los Alamos, New Mexico, July 1965.
  4. W. W. Little, R. W. Hardie and L. L. Maas. Multigroup Analysis of Selected Fast Critical Assemblies, BNWL-347. Pacific Northwest Laboratory, Richland, Washington, December 1966.

TABLE 3.X. Geometry and Composition Calculation

			Geometry		Composition, vol%			
Zone			Inner Radius, cm	Outer Radius, cm	Fuel	SS	Na	Inconel
Description								
<u>Fuel Pin Cell^(1,2)</u>								
Fuel Pin (NPTR/FTR)	1	Fuel	0.0	0.27559	100	-	-	-
	2	Stainless Steel	0.27559	0.31623	-	100	-	
	3	Na	0.31623	0.35687	-	-	100	
<u>Subassembly Cell^(1,2)</u>								
FTR S/A	1	Fuel, Na, and SS	0.0	5.627	52	16.5	31.5	
	2	Stainless Steel	5.627	6.008	-	100	-	
	3	Na	6.008	6.704	-	-	100	
NPTR/S/A	1	Fuel, Na, and SS	0.0	5.627	52	16.5	31.5	
	2	Na	5.627	6.365	-	-	100	
	3	Stainless Steel	6.365	6.704	-	100	-	
<u>Reactor^(1,2,3)</u>								
	1	Open Test Loop	0.0	7.14	-	-	-	-
	2	Stainless Steel	7.14	7.46	-	100	-	-
	3	Na	7.46	10.0	-	-	100	-
	4	Core	10.0	22.5	36.5	21.4	42.1	-
	5	Na	22.5	30.41	-	-	100	-
	6	Core	30.41	63.31	36.5	21.4	42.1	-
	7	Reflector	63.31	78.51	-	14	30	56

1. Fuel Composition: PuO_2 - 22 vol%; (plutonium - 10% ^{240}Pu), $^{238}\text{UO}_2$, 78 vol%.
2. Stainless Steel Composition: Fe, 74 wt%; Cr, 18 wt%; Ni, 8 wt%.
3. Inconel Composition: Fe, 10 wt%; Cr, 18 wt%; Ni, 72 wt%.

location the total mass of stainless steel was conserved rather than the actual thickness of the stainless annulus. Thus no material volume fractions were changed between the two cases. Flux-volume weighted cross sections were first obtained from a fuel pin cell calculation in which the input cross sections were core averaged ones taken from an FCC-IV computation. Subsequently, cell calculations were made for a typical NPTR and FTR subassembly cell. Averaged cross sections from these cases were then used in core calculations, utilizing the model as shown in Table 3.X. As indicated in Table 3.XI, the effect of removing the duct from the fuel boundary is about 0.15% $\Delta k/k$. A comparison of spectra for these two models shows only very minor differences. The spectral difference between the NPTR and FTR subassemblies at the edge and at the center of the fuel zone were examined. Figure 3.38 shows the effect at the edge of the fuel zone. The results at the center of the subassembly were even less pronounced.

TABLE 3.XI. NPTR/FTR Calculated Reactivities

	k_{eff}	Δk (%)
FTR S/A Cell ^(a)	1.00152	0.152
NPTR S/A Cell ^(a)	1.00000	
FTR Reactor Model	1.02508	0.145
NPTR Reactor Model	1.02363	
FTR S/A Cell ^(b)	1.01021	0.87
FTR S/A Cell ^(a)	1.00152	
NPTR S/A Cell ^(b)	1.00804	0.80
NPTR S/A Cell ^(a)	1.00000	

a. Using fuel pin cell flux weighted cross section

b. Using fundamental mode core averaged cross section

As a subsidiary to the foregoing results the subassembly cell calculation were repeated with core averaged (from FCC-IV) cross sections being used instead of the fuel pin cell flux-volume weighted values. This gave the results

FIGURE 3.38. Spectrum at Fuel Zone Boundary (FTR S/A) and Spectrum Difference Between NPTR and FTR Subassemblies

as shown in Table 3.XI. The reactivity difference between the two NPTR and FTR subassembly cell cases gives some indication of the effect of neglecting fuel heterogeneity. The differences obtained here are somewhat higher than those obtained by Harris earlier.⁽¹⁾ The disparity may be caused by improper convergence of calculations in the cases where very small cell size is used.

The primary result of the calculations that were performed is the indication that no serious changes were effected in the NPTR core neutronics characteristics by the choice of tapered module cans.

- b. FTR Experimental Program, Phase B (This work performed for System 30. Reactor)
W. R. Young

The Fast Test Reactor (FTR) will have an in-core irradiation test zone in which the fuel concentration will be less than it is in the driver zone. Critical masses for six reflected fast plutonium cores with fuel free test zones of varying sizes and shapes have been determined in the ZPR-III by Argonne National Laboratory, for the Battelle Northwest Fast Flux Test Facility Project. The reflector, test zone, and driver compositions of each assembly remained nearly constant and represent similar zones in the FTR. These experiments are being analyzed to test computational methods and data sets.

Core volumes approximated right circular cylinders 86 cm high. Nickel-sodium reflectors were about 30 cm thick.

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5. R. A. Harris. A Preliminary Study of the Heterogeneous Effects in the FTR and FTR Critical Mockup, Pacific Northwest Laboratory, Richland, Washington, December 15, 1966.
(Private Communication)

Simulated test zones were centered on the core axis and extended through the core and axial reflector. Table 3.XII gives the nominal composition of each zone of the cores. Figure 3.39 shows a transverse section through each core.

Three of the test zone shapes represented central cylindrical volumes and could be approximated in R-Z geometry. Since the ZPR-III is a square matrix reactor, these test zones were mocked up as volumes that were, in turn, three drawers wide by three drawers high; 5 x 5, less three drawers at each corner; and 7 x 7, less three drawers at each corner. Each drawer is 2.176 in. high by 2.184 in. wide.

TABLE 3.XII. *Composition of Zones of ZPR-III Assembly 52 Series, vol%*

	<u>PuO₂, (10% ²⁴⁰Pu)</u>	<u>UO₂</u>	<u>Na</u>	<u>304 SS</u>	<u>Ni</u>
Driver Zone	7.6	28.8	41	25	
Test Zone			80	20	
Radial Reflector			18	12	61
Axial Reflector			45	17	31

Measured critical masses and calculated masses for the cylindrical test zones are compared in Table 3.XIII. The calculations were done with specification of compositions and boundaries of each zone except for the radius of the core. This parameter was determined in the critical calculation.

All cores had spiked safety rods; however, for 52d, e, and f increased spiking was necessary. In these last assemblies the safety rod fissile material was increased an additional 5.5 kg. Critical calculations show that an additional 5.5 kg fissile at 20 cm radius in 52f allows 17.5 kg to be removed at the core edge; i.e., the critical mass decreased by about 4%.

The cores were analyzed using a modified 26 group Boundarenko Cross Section Set.⁽¹⁾ Energy self-shielding factors

1. W. W. Little, R. W. Hardie and L. L. Maas. *Multigroup Analysis of Selected Fast Critical Assemblies*, BNWL-347. Pacific Northwest Laboratory, Richland, Washington, December 1966.

FIGURE 3.39. FTR Hole/Slot Comparison Assemblies
(ZPR-III Experiment-Assemblies 52)

and cross-section preparation was carried out with the computer code FCC-IV.⁽¹⁾ The cross sections were flux-volume weighted to correct for heterogeneities. Critical masses were determined from 2-D reactor calculations. The calculated masses are consistently overpredicted.

TABLE 3.XIII. *Critical Masses of Some Fast Plutonium Assemblies Having Central Fuel-Free Volumes (Test Zones)*

Assembly Number	Test Zone Areal Profile	Test Zone Volume, liter	Critical Mass kg of Fissile		E/C
			Experiment	Calculated	
52a		0	200.7 $\pm$ 0.5	210	0.95
52b	Square, 3 x 3	24	235.8 $\pm$ 0.5	249	0.94
52c	Square, 5 x 5 - less corner drawers	66	280.0 $\pm$ 0.5	290	0.96
52f	Square, 5 x 5 - 3 drawers each corner	98		323	
52d	Rectangle, 3 x 11	88	275.2 $\pm$ 0.5		
52e	Rectangle, 3 x 15	119	282.0 $\pm$ 0.5		

3. Radiation and Shield Analysis

W. L. Bunch

a. Neutron Attenuation Through Sodium (This work performed for System 30. Reactor)

D. R. Marr

Design of the shielding for the FFTF requires a capability to calculate neutron attenuation in sodium. To evaluate calculational methods, a neutron attenuation experiment in sodium was analyzed.⁽²⁾ The experiment consisted of measuring the activity induced by the reactions shown in Table 3.XIV at different depths in the sodium. The sodium was canned in thin walled aluminum tanks 5 ft square by 1 ft thick. Ten of these cans plus one 6 in. thick were stacked next to an enriched

1. W. W. Little, Jr. and R. W. Hardie. FCC-IV, A Revised Version of the FCC Fundamental Mode Fast Reactor Code, BNWL-450. Pacific Northwest Laboratories, Richland, Washington, August 1967.
2. A. W. Thiele. NEUTRON and GAMMA RAY ATTENUATION in SODIUM SHIELDS. NAA-SR-MEMO-12467. *Atoms International*, June 30, 1967.

uranium fission plate which was driven by thermalized neutrons from the STIR (Shield Test and Irradiation Reactor).⁽¹⁾

TABLE 3.XIV. Foil Reactions

Au ¹⁹⁷	(n,γ) Au ¹⁹⁸	Thermal
In ¹¹⁵	(n,γ) In ¹¹⁶	1.46 eV Resonance
Au ¹⁹⁷	(n,γ) Au ¹⁹⁸	4.91 eV Resonance
Na ²³	(n,γ) Na ²⁴	2.85 keV Resonance
Co ⁵⁹	(n,γ) Co ⁶⁰	132 eV Resonance
In ¹¹⁵	(n,n') In ^{115m}	1.2 MeV Threshold
S ³²	(n,p) P ³²	3.0 MeV Threshold
Al ²⁷	(n,α) Na ²⁴	8.1 MeV Threshold

Two calculational methods were used. The first was a kernel generation by a 1-dimensional transport code ANISN,⁽²⁾ followed by an integration of the kernel over the fission plate radial distribution for each dose point. The second method was the use of the 2-dimensional transport code DOT.⁽³⁾

The 1-dimensional method gave very good agreement for the attenuation characteristics for all the threshold energies, but gave absolute saturated activities which were approximately 50 to 60% high. Neutrons at the various resonance energies were all calculated by this method to have attenuation lengths of about 50 cm, whereas, the experimental values were about 60 cm.

1. A. W. Thiele. NEUTRON and GAMMA RAY ATTENUATION in SODIUM SHIELDS. NAA-SR-MEMO-12467. *Atoms International*, June 30, 1967.
2. W. W. Engle, Jr. A Users Manual for ANISN--A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering, K-1693. *Union Carbide Corporation*, March 30, 1967.
3. F. R. Mynatt. A Users Manual for DOT--A Two-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering, K-1694. *Union Carbide Corporation*, (In Process)

In addition, the calculated absolute saturated activities were much higher (up to 500% higher depending on the cross sections used). The 2-dimensional calculation gave absolute saturated activities which were also high for the resonance detectors, but comparable to the disagreement at threshold energies. This indicates that the 2-dimensional treatment is more realistic for handling the fission plate-shield interface with its geometrical complexities. The 2-dimensional calculation also gave an attenuation length of about 50 cm for the resonance energies.

The attenuation length at the threshold energies is about 25 cm, whereas the lower energy detectors indicate an attenuation length of about 60 cm. The range of energies in which the change in attenuation lengths occurs is sufficiently low so that anisotropy in the elastic scattering process is not important, because diffusion theory gave the same results for the resonance energy detectors as P_3 , S_{16} calculations. Since the calculation yield a significantly shorter attenuation length (50 cm versus 60 cm), it is possible that incorrect sodium cross-section data exist at some energies between the resonance and threshold values of the experiment.

b. Neutron Flux to FTR Tube Sheet (This work performed
System 30. Reactor)

C. A. Mansius

One of the key structural components within the FTR vessel is the inlet plenum upper tube sheet. The intensity and spectral distribution of the neutron flux in the tube sheet is required to estimate radiation damage. A rigorous treatment of the problem would require a 3-dimensional analysis, which is beyond our present capabilities; therefore, a preliminary estimate has been made based on 1-dimensional calculations. Results of the calculations are summarized in Table 3.XV; the spectral data are plotted in Figure 3.40 for the transport solution in spherical geometry.

TABLE 3.XV. Calculated Flux Spectra at Tube Sheet-Surface, Flux/MeV

Group	Energy - Ev (Upper Bound)	Spherical (Transport)	Spherical (Diffusion)	Finite Slab (Transport)	Infinite Slab (Transport)	Infinite Slab (Removal Diffusion)	Δ Energy, MeV	Δ Lethargy
1	1.0 x 10 ⁷	2.9 x 10 ⁶	1.4 x 10 ⁵	7.8 x 10 ⁶	1.2 x 10 ⁷	9.5 x 10 ⁶	3.935	0.50
2	6.065 x 10 ⁶	1.9 x 10 ⁷	1.5 x 10 ⁶	4.9 x 10 ⁷	7.4 x 10 ⁸	3.0 x 10 ⁸	2.386	0.50
3	3.679 x 10 ⁶	8.7 x 10 ⁷	1.4 x 10 ⁷	2.0 x 10 ⁸	3.1 x 10 ⁹	7.0 x 10 ⁹	1.448	0.50
4	2.231 x 10 ⁶	9.2 x 10 ⁸	2.9 x 10 ⁸	1.9 x 10 ⁹	3.0 x 10 ⁹	4.0 x 10 ⁹	0.878	0.50
5	1.353 x 10 ⁶	5.7 x 10 ⁹	2.5 x 10 ⁹	1.1 x 10 ⁹	1.8 x 10 ¹⁰	2.0 x 10 ¹⁰	0.532	0.50
6	8.209 x 10 ⁵	5.2 x 10 ¹⁰	3.5 x 10 ¹⁰	8.4 x 10 ¹⁰	1.6 x 10 ¹¹	2.5 x 10 ¹¹	0.433	0.75
7	3.877 x 10 ⁵	4.6 x 10 ¹¹	3.8 x 10 ¹¹	6.6 x 10 ¹¹	1.4 x 10 ¹²	2.5 x 10 ¹²	0.205	0.75
8	1.832 x 10 ⁵	3.6 x 10 ¹²	3.3 x 10 ¹²	4.4 x 10 ¹³	1.0 x 10 ¹³	2.0 x 10 ¹³	0.116	1.0
9	6.738 x 10 ⁴	9.6 x 10 ¹³	9.0 x 10 ¹³	1.1 x 10 ¹³	2.7 x 10 ¹³	4.5 x 10 ¹³	4.26 x 10 ⁻²	1.0
10	2.479 x 10 ⁴	3.3 x 10 ¹³	3.1 x 10 ¹³	3.7 x 10 ¹³	9.1 x 10 ¹³	1.5 x 10 ¹⁴	5.5 x 10 ⁻³	0.25
11	1.930 x 10 ⁴	4.0 x 10 ¹³	3.9 x 10 ¹³	4.4 x 10 ¹³	1.1 x 10 ¹⁴	2.2 x 10 ¹⁴	1.018 x 10 ⁻²	0.75
12	9.119 x 10 ³	4.0 x 10 ¹³	3.9 x 10 ¹³	4.4 x 10 ¹³	1.1 x 10 ¹⁴	2.6 x 10 ¹⁴	5.76 x 10 ⁻³	1.0
13	3.355 x 10 ³	1.7 x 10 ¹⁵	1.65 x 10 ¹⁴	1.8 x 10 ¹⁵	4.8 x 10 ¹⁵	6.0 x 10 ¹⁵	2.12 x 10 ⁻³	1.0
14	1.234 x 10 ³	1.0 x 10 ¹⁵	1.0 x 10 ¹⁵	1.0 x 10 ¹⁵	2.8 x 10 ¹⁵	4.5 x 10 ¹⁵	7.8 x 10 ⁻⁴	1.0
15	4.540 x 10 ²	2.3 x 10 ¹⁵	2.3 x 10 ¹⁵	2.3 x 10 ¹⁵	6.4 x 10 ¹⁵	1.0 x 10 ¹⁶	1.79 x 10 ⁻⁴	0.50
16	2.754 x 10 ²	3.4 x 10 ¹⁵	3.4 x 10 ¹⁵	3.5 x 10 ¹⁵	9.7 x 10 ¹⁵	1.5 x 10 ¹⁶	6.09 x 10 ⁻⁵	0.25
17	2.145 x 10 ²	6.2 x 10 ¹⁵	6.2 x 10 ¹⁵	6.0 x 10 ¹⁵	1.7 x 10 ¹⁶	2.6 x 10 ¹⁶	1.132 x 10 ⁻⁴	0.75
18	1.013 x 10 ²	1.2 x 10 ¹⁶	1.2 x 10 ¹⁶	1.2 x 10 ¹⁶	3.4 x 10 ¹⁶	5.0 x 10 ¹⁶	4.0 x 10 ⁻⁵	0.50
19	6.144 x 10	2.4 x 10 ¹⁶	2.4 x 10 ¹⁶	2.2 x 10 ¹⁶	6.6 x 10 ¹⁶	1.0 x 10 ¹⁷	3.2 x 10 ⁻⁵	0.75
20	2.902 x 10	4.8 x 10 ¹⁶	4.9 x 10 ¹⁶	4.3 x 10 ¹⁶	1.4 x 10 ¹⁷	2.1 x 10 ¹⁷	1.53 x 10 ⁻⁵	0.75
21	1.371 x 10	7.4 x 10 ¹⁶	7.5 x 10 ¹⁶	6.5 x 10 ¹⁶	2.1 x 10 ¹⁷	3.0 x 10 ¹⁷	3.03 x 10 ⁻⁶	0.25
22	1.608 x 10	1.2 x 10 ¹⁷	1.2 x 10 ¹⁷	1.0 x 10 ¹⁷	3.5 x 10 ¹⁷	4.2 x 10 ¹⁷	6.75 x 10 ⁻⁶	1.0
23	3.928	2.3 x 10 ¹⁷	2.3 x 10 ¹⁷	1.8 x 10 ¹⁷	6.5 x 10 ¹⁷	8.0 x 10 ¹⁷	2.48 x 10 ⁻⁶	1.0
24	1.445	2.9 x 10 ¹⁷	3.0 x 10 ¹⁷	2.4 x 10 ¹⁷	8.5 x 10 ¹⁷	1.25 x 10 ¹⁸	5.69 x 10 ⁻⁷	0.5
25	0.8764	3.1 x 10 ¹⁷	3.2 x 10 ¹⁷	2.5 x 10 ¹⁷	9.0 x 10 ¹⁷	1.4 x 10 ¹⁸	1.94 x 10 ⁻⁷	0.25
26	0.6826	3.1 x 10 ¹⁷	3.2 x 10 ¹⁷	2.5 x 10 ¹⁷	9.1 x 10 ¹⁷	1.3 x 10 ¹⁸	2.69 x 10 ⁻⁷	0.5
27	0.414	2.5 x 10 ¹⁷	2.5 x 10 ¹⁷	2.0 x 10 ¹⁷	7.2 x 10 ¹⁷	1.2 x 10 ¹⁸	1.36 x 10 ⁻⁷	0.4
28	0.2775	2.0 x 10 ¹⁷	2.1 x 10 ¹⁷	1.6 x 10 ¹⁷	5.9 x 10 ¹⁷	1.0 x 10 ¹⁸	9.15 x 10 ⁻⁸	0.39
29	0.186	1.6 x 10 ¹⁷	1.7 x 10 ¹⁷	1.2 x 10 ¹⁷	4.8 x 10 ¹⁷	7.5 x 10 ¹⁷	6.13 x 10 ⁻⁸	0.4
30	0.1247	1.1 x 10 ¹⁷	1.1 x 10 ¹⁷	8.0 x 10 ¹⁶	3.1 x 10 ¹⁷	5.0 x 10 ¹⁷	4.87 x 10 ⁻⁸	0.5
31	0.076	4.3 x 10 ¹⁶	4.4 x 10 ¹⁶	3.2 x 10 ¹⁶	1.3 x 10 ¹⁷	2.1 x 10 ¹⁷	7.595 x 10 ⁻⁸	--

FIGURE 3.40. *Calculated Neutron Spectrum Incident on Tube Sheet*

The two computer codes ANISN and MAC were used for the calculations. Either diffusion theory or transport theory can be employed in ANISN using spherical, cylindrical, or slab geometry, with buckling corrections to approximate finite dimensions where necessary. The MAC code is limited to using removal-diffusion theory and infinite slab geometry. Because transport theory is the most rigorous of the calculational techniques employed, and since spherical geometry can be represented mathematically without approximation, the results presented in Figure 3.40 should be most realistic. The infinite slab geometry using removal-diffusion theory yields the most conservative results.

The difference in results between diffusion theory and transport theory is most apparent in the spectrum above 0.39 MeV, with relatively minor differences between 0.39 MeV and 0.0012 MeV. For this reason, diffusion theory was used in all ANISN calculations below 0.0012 MeV and the computer running time was reduced by about 40%.

The accuracy of the finite slab calculation is limited by the introduction of a constant buckling factor for all points in the shield. This approximation should be realistic near the core, but it becomes progressively less valid at greater distances into the shield. On the other hand, the spherical approximation should be valid at large distances but becomes progressively less valid close to the core. The disagreement between these methods, which can be noted in Table 3.XV, gives an indication of the uncertainty introduced by the 1-dimensional treatment.

All calculations are based on the shield data in Table 3.XVI. The distance from the core to the tube sheet and average region material composition are based on conceptual design studies. The purpose of these calculations is to provide a basis for estimating the effect of radiation

TABLE 3.XVI. *Shield Data*

Region Number	Region Name	Region Thickness, cm	Volume Percent			Material Composition, g/cm ³			
			Stainless	Sodium	Nickel	Nickel	Chromium	Iron	Sodium
1	Axial Reflector	30.5	31.7	55.83	0.2147	1.31	0.452	1.86	0.469
2	Stainless Tubes and Sodium	111.8	21.93	78.07		0.139	0.313	1.28	0.656
3	Upper Tube Sheet	12.7	60.0	40.0		0.38	0.856	3.52	0.336
4	Stainless Tubes and Sodium	55.9	7.0	93.0		0.044	0.10	0.41	0.781
5	Lower Tube Sheet	10.2	100.0	0		0.633	1.43	5.86	--
6	Sodium	60.0	0	100.0		--	--	--	0.84

3.137

BNWL-880

damage to determine whether this is a satisfactory arrangement. Total flux at the core-shield interface was normalized to a value of 2.5×10^{15} n/cm²-sec.

- c. Decay Heat from Irradiated Plutonium (This work performed for Systems 40. Fuel Handling and 70. Fuel Examination W. L. Bunch

Knowledge of the amount of decay heat associated with plutonium fuel irradiated in a fast reactor environment is required for the design of fuel handling, storage, and examination facilities in the FFTF. An experimental determination of the decay heat generated by fission products in plutonium oxide test samples irradiated in the Dounreay Fast Reactor is reported by K. Johnston.⁽¹⁾ The computer code RIBD⁽²⁾ was used to calculate the heat generation rate associated with these samples. Results of the experiment and calculations are shown in Figure 3.41. It can be seen that total heat generation rates are in good agreement. The calculated gamma values are about 10% higher than the measured values, whereas, the calculated beta results are about 10% lower. It is concluded that RIBD with its present nuclear library provides a realistic method of estimating decay heat rates from FTR fuel in the time interval covered by the experiment.

Two samples were irradiated intermittently in the DFR over a period of 125 days to an integrated exposure determined to be 1.89×10^{21} n/cm², which yielded 2.68×10^{17} fission events per sample. Calorimetric measurements were made over a period of about 70 days, as shown in Figure 3.41. By

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1. K. Johnston. "A Calorimetric Determination of Fission Product Heating in Fast Reactor Plutonium Fuel," *J. Nucl. Energy, part A/B*, vol. 19, pp. 527-539, Pergamon Press Ltd., New York.
  2. R. O. Gumprecht. Mathematical Basis of Computer Code-RIBD, DUN-4136. Douglas United Nuclear, Richland, Washington, June 17, 1968.

FIGURE 3.41. *Fission Product Decay Heat from Plutonium Sample Irradiated in Fast Reactor to Integrated Exposure of 2.68×10^{17} Fissions (125 days of cyclic operation of Dounreay Fast Reactor)*

using blank capsules for background determination and uranium absorbers for gamma shields, the decay energy was divided between the beta and gamma components.

The computer code RIBD calculates the inventory of 450 different fission product isotopes by considering direct yield factors from fission, mode of decay and half-life, and the effect of neutron induced transmutations of the fission products. One option includes conversion of the time dependent inventory into decay heat. In the calculations, the exposure history paralleled that received by the test samples and was normalized to the same number of fissions (2.68×10^{17}).

Because of the large number of isotopes involved and the complex relationship between the inventory and the fuel history, agreement with this single experiment does not validate all of the nuclear data incorporated in the RIBD library, but it does give confidence in the ability to treat relatively complex exposure histories.

d. Heat Generation in Control Material (This work performed for System 30. Reactor)

W. L. Bunch

An estimate was made of the nuclear heat generation in candidate poisons for control of the FTR. Results are summarized in Table 3.XVII for both in-core and peripheral locations. The tabulated values, which are maximum heat generation rates based on an 875-liter oxide-fueled core, are realistic for the range of conceptual cores being considered for FTR.

Heat is generated in any material inserted into a nuclear reactor because of the radiation that is present. Three basic radiation energy sources exist: gamma energy, the kinetic energy of the neutrons, and the binding energy released when a neutron is absorbed. Because of the large neutron capture

cross section associated with control materials, the latter process tends to be dominant in poison rods. However, an estimate was made of the contribution of each of the three sources.

TABLE 3.XVII. *Calculated Heat Generation Rates in Control Materials (watts per gram)*

Energy Source	In-Core		Peripheral	
	B ₄ C	Ta	B ₄ C	Ta
Capture	48.8 ^(a)	11.4	26.6	6.2
Gamma	9.5	9.5	4.6	4.6
Kinetic	10.4	1.7	5.0	0.8
Total	68.7	22.6	36.2	11.6

a. Capture values for B₄C assume natural boron containing 19.78 at.% ¹⁰B. For enriched boron carbide, the capture rate must be increased by the ratio (% enrichment)/(19.78%). Other values would not be influenced materially by the enrichment level.

Heat generation resulting from neutron capture in a relatively small volume of material is given by:

$$H = hN\phi\sigma e$$

where h is the energy release per capture, N is the number of atoms in the region with a spectrum weighted capture cross section σ , ϕ is the volume average flux in the material, and e is the fraction of the capture energy absorbed in the material in which the neutron is captured. For ¹⁰B, h is about 0.447×10^{-12} W-sec, whereas, in ¹⁸¹Ta it is about 1.25×10^{-12} W-sec. Values of ϕ and σ at the two positions were obtained from neutronics calculations for the reference core. It was assumed that all of the capture energy would be self-absorbed ($e = 1.0$). This is a realistic assumption for the kinetic energy associated with the (n, α) reaction of ¹⁰B and is conservative for the predominantly gamma radiation associated with ¹⁸¹Ta.

It was assumed that gamma energy would be lost in the reactor primarily as a result of Compton scattering events. Under this assumption, equal masses of material will absorb equal amounts of gamma energy when placed at the same location in the reactor. The gamma intensity within a core is inversely proportional to its size and directly proportional to the power level. For the reference concept it is estimated that about 35 MW of gamma energy (per 400 MW of total reactor power) would be deposited within the core, giving an average deposition rate of about 6.8 W/g. Values at the control positions are based on the power distribution within the FTR.

The kinetic energy of the neutrons can be lost by both elastic and inelastic scattering events. Since only about 8.35 MW of neutron kinetic energy (per 400 MW of total reactor power) is estimated to be deposited within the reference core, it was assumed that an equal energy loss could be ascribed to each interaction, regardless of the type of scattering nucleus. Under this assumption, the watts of neutron kinetic energy deposited in a small sample located at an average power location is:

$$W = \frac{\sum(\text{sample})}{\sum(\text{core})} 8.35 \times 10^6$$

where $\sum(\text{sample})$ is the macroscopic interaction cross section of the sample and $\sum(\text{core})$ is the total macroscopic interaction cross section for all of the materials comprising the core. As in the case of the gamma energy, it was assumed local variations could be related to the power distribution.

III. GENERAL TECHNOLOGY

P. L. Hoffman

HIGHLIGHTS OF SAFETY AND RELIABILITY ANALYSIS

D. E. Simpson

Preliminary studies were carried out to determine coolant flow through the core assuming a major coolant pipe break. The results point out the importance of subdivision of the high-pressure piping in maintaining core flow during the period immediately following a high pressure pipe break, for cases in which devices such as check valves are omitted.

The FFTF Reliability Analysis Program has been developed. This program includes effectiveness goals prepared for the various systems within the facility. These goals set out reliability and/or availability targets for each system which are consistent with a net reactor availability of 75%. The program also describes the relationship between PNL and subcontractors; monitoring of subcontractor reliability analysis during the design phase will be accomplished through use of the following reports from the subcontractors.

Analysis of a 2-barrier containment system was performed. For a system with an inner containment leak rate of 10%/day and an outer containment of 1%/day, the following control distances would be required.

Exclusion Area Boundary	0.3 miles
Low Population Zone Boundary	0.8 miles
Population Center Distance	1.0 miles

D. SAFETY AND RELIABILITY ANALYSIS

D. E. Simpson

1. Safety Analysis

D. D. Stepnewski

a. Preliminary Safety Study of the Loss-of-Coolant Accident

R. D. Peak

Two studies pertaining to the loss-of-coolant accident have been performed. The first study (Hydraulic Study, below) examined the hydraulic effects of a large pipe break and suggested provisions to mitigate these effects. The second study (Pipe Break Considerations) examined the basis for assuming a large pipe break as the initiator of the accident.

(1) Hydraulic Study. The hydraulic study concentrated on a pipe break between the primary sodium pumps and the reactor because a break along this pressurized piping would result in greater leakage, owing to pumping action, than a break elsewhere in the primary system. Also, a pipe break between the pumps and reactor would cause a reduction in coolant flow through the core as the pumps in unaffected loops continue to supply coolant to the reactor inlet plenum and thence, either (a), up through the core for recirculation or (b), backward through the affected loop and out the break.

The study explored the reduction in core flow following a break, as a function of the following parameters;

- A 3-loop system versus a 4-loop system, each designed optimally for the initial 400 MW core. Pertinent characteristics of these two systems are noted below.⁽¹⁾

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(1) R. A. Harvey, H. G. Johnson, M. A. McLaughlin, and E. F. Witeck. Technical Basis for Main Heat Transport System Concept and Concept Configuration Selections, BNWL-554, p. 63, Table VI; p. II-10, Table XXIV; and p. II-28, Table XLII. Pacific Northwest Laboratory, Richland, Washington, September 1967.

<u>Characteristic</u>	<u>3-Loop System</u>	<u>4-Loop System</u>
Primary pipe sizes:		
Reactor-to-pump, diam	24 in.	18 in.
Pump-to-reactor, diam	18 in.	14 in.
Primary pump flow, each	11,200 gpm	8,430 gpm

- Subdivision of pressurized piping from pump-to-reactor: the 18 in. diam pipe into two 12 in. or three 10 in. pipes; the 14 in. diam pipe into two 10 in. diam pipes.
- Reactor pressure losses of 95, 115, and 135 psi.
- Primary pump speeds of 870 and 1050 rpm.
- Flow resistances from the reactor inlet plenum, back to, and out the break of 1.0 to 4.0 velocity heads ($V^2/2g$).

The analysis included flow limiting effects of pump cavitation based on predictions of "Net Positive Suction Head" (NPSH) requirements for the primary pumps. The analysis assumed normal operating conditions prior to the break and also disregarded the time necessary to effect pipe breakage, overcome fluid inertia, and change flow patterns in the primary system to "psuedosteady" conditions prior to reactor scram and pump shutdown. During these psuedosteady state conditions, the pump of the affected loop forces coolant out the break and the other pumps supply coolant to the reactor inlet plenum of which only part flows through the core, the rest flows out the break.

The results of this parametric study are summarized in Figure 3.42 to illustrate some trade-offs that may be considered in primary system design in order to sustain core flow rates of, for example, 34 or 55% of normal following a pipe break. Digital simulation⁽¹⁾ studies indicate that, without scram, reduction of core flow to less than 34% of normal may cause boiling in the hot channel of a 600 to 900 °F core and reduction to less than

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(1) H. G. Johnson and C. D. Flowers, this report, Section III B.

FIGURE 3.42. Parametric Study of Core Flow Rates Following a Break in the Pressurized Pipes Between Pumps and Reactor

55% of normal may cause boiling in a 900 to 1200 °F core. With these interpretations of flow effects, Figure 3.42 indicates that subdivision of piping from the pumps to the reactor is the most important design means for sustaining core flow following a break, primarily because the break is then in a smaller pipe. For example, core flow of at least 34% of normal would not be realized following a break in one 18 in. pipe of a 3-loop system unless the flow resistance from the reactor inlet plenum through the break was very high, but if these 18 in. pipes were replaced by pairs of 12 in. pipes, core flow of at least 34% of normal would be realized following a break in one of the 12 in. pipes. Sustaining core flow of at least 55% of normal requires more extensive piping subdivisions, three 10 in. pipes from each pump in the 3-loop system or two 10 in. pipes from each pump in a 4-loop system, plus more than nominal flow resistance from the plenum through the break. In comparison to piping subdivision, pump speed and reactor pressure loss have only small effects on core flow reduction which could be traded for devices providing higher flow resistance from the plenum through the break.

(2) Pipe Break Considerations. The basis for assuming a large pipe break as the initiator of the loss-of-coolant accident is being explored in experiments on initiation and extent of ductile pipe rupture at Battelle Memorial Institute.⁽¹⁾ These experiments suggest that the large leak is not itself the initiator of accident but rather the consequence of an accident.

The BMI experiments will include full-scale rupture tests on austenitic stainless steel pipe in addition to tests performed

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(1) Investigation of the Initiation and Extent of Ductile Pipe Rupture, Battelle Memorial Institute, Columbus, Ohio, Progress Reports as follows: BMI-1793, May-December, 1966; BMI-1813, January-June, 1967; BMI-1817, July-September, 1967; BMI-1828, October-December, 1967; BMI-1836, January-March, 1968.

to date on A106B carbon steel pipe. The cracks for these pipe tests have been longitudinal saw or mill cuts. Each "cracked" pipe has been subjected (at temperatures of 550 to 690 °F) to increasing water or steam pressure until the crack propagates sufficiently to cause leakage and major pressure reduction. Three types of crack propagation have been observed in these tests: complete rupture, short crack extension, and leak. Complete rupture and short crack extension types of failure are believed to be dependent upon the state of the pressurizing medium, geometry of the crack, and toughness of pipe material but the leak type of failure is believed to be dependent only on geometry and toughness.

The theoretical formulations used by BMI in correlating conditions causing failure of cracked piping have been used to survey pressures that might be required to cause leakage or rupture failures in the sizes of piping that may be used in the primary sodium system of the FTR. For this survey and in lieu of experimental data, values of the following properties of Type 304 SS had to be assumed:

K_c Stress-intensity factor, a measure of the toughness of the pipe material, defined at the critical stress at which a defect becomes unstable and starts to propagate.

S_c Rupture stress for unflawed pipe.

Figure 3.43 shows pressures, calculated from BMI formulations, which might cause rupture or leakage failures in FTR sodium piping having nominal 0.375-in. thick walls. The results for pipes of 10.75 in. to 20.0 in. OD are based on 800 °F material properties because such pipes may be used to connect primary pumps to the reactor; results for pipes of 24.0 and 30.0 in. OD are based on 1200 °F properties because such pipes may be used to connect the reactor to intermediate heat exchangers. In Figure 3.43a, pressures to cause pipe rupture

Properties assumed for Type 304 SS

Temp.	K_c	S_c
800 °F	300 ksi $\sqrt{\text{in.}}$	40.5 ksi
1200 °F	300 ksi $\sqrt{\text{in.}}$	30.0 ksi

(See text for definitions of K_c and S_c)

FIGURE 3.43. Pressures to Cause Rupture or Leakage of Longitudinal Surface Cracks in Piping

(rapid crack extension) are graphed for short surface cracks penetrating (because of fatigue, creep, or other mechanisms) less than about 52% through the wall before the pipe becomes pressurized; pressures for zero length surface cracks being for unflawed pipes. The failure mode for surface cracks deeper than 52% of the wall is leakage because the crack will extend through the wall rather than propagate at its ends. In Figure 3.43b, pressures to cause pipe leakage are graphed for short cracks penetrating about 98.7% through the wall (leaving only 0.005 in. unpenetrated) before the pipe becomes pressurized.

The significant but tentative conclusion to be drawn from Figure 3.43 is that sodium pressure must be greatly in excess of the design pressure of (at most) 175 psig for primary sodium piping before this piping would rupture into a large leak; the more likely response to overpressure is no leakage or, at most, small leakage when surface cracks propagate through the remainder of the wall.

2. Reliability Analysis

R. A. Harvey

a. Reliability Analysis Program

R. A. Harvey

The overall Reliability Analysis Program is indicated on Figure 3.44. An allocation of reactor unavailability chargeable to the various systems was made based on the 75% plant availability goal. The unavailability allocations, safety, and engineering judgements were used as bases for the definition of system effectiveness goals. The program requires reliability studies by the contractors during the design phases so that reliability considerations can influence the design.

The long range plan calls for an updating of reliability predictions and estimates through the fabrication and installation

FIGURE 3.44. Schematic Representation of Effectiveness Analysis Steps Relating to FTF Design, Fabrication, and Operation. (RAN signifies FTF reliability analysis function.)

phases, and measurement of effectiveness during the initial operational phase of the FFTF. Throughout the design, fabrication, installation, and initial operation phases, the reliability status will be made visible through charts that compare status with goals for the FFTF and its systems.

The reactor unavailability goal allocation for each system group as shown in Figure 3.45 and for each system is shown in Table 3.XVIII. The monitoring of the contractor's Reliability Analysis efforts during the design phase will be accomplished by monitoring reports from the contractor that fall into the following categories:

- Reliability Analysis Program Plan Report
- Progress Reports
- Separate Reliability Design Analysis and Prediction Reports
- Reliability Analysis Program Plan
- Periodic Progress Reports
- Reliability Design Analysis and Prediction Reports.

A preliminary reliability prediction for the FFTF Conceptual Building Electrical Power System was completed. Determinations were made to predict the availability of power, mean-time-between-failures of power, and the mean-time-to-restore power to the various busses of System 12. No reliability problems were revealed in the analysis.

The reliability required for each process channel of the in-channel instrumentation was calculated for two different criteria. The first is for the case where the reactor must be shut down if the instrumentation functions are lost in any one channel, and the second is for the case where instrumentation functions of two adjacent channels can be lost before reactor shutdown is required. A table of overall reliabilities (for 91 channels) was prepared for each criterion versus the single channel reliability.

FIGURE 3.45. Unavailability Time Allocations (Numbers are based on a 1000 hour reference time).

TABLE 3.XVIII. *Reactor Unavailabilities*

<u>Systems</u>	<u>Unavailability hr per 1000 hr</u>
Electrical Systems	
11 Primary Power System	0.1
12 Building Electrical Power System	0.1
15 Communications System	0
16 Lighting System	0
Structures and Utilities	
21 Structures	0.2
22 Site Facility	0
23 Service Piping System	0.3
24 Radioactive Waste System	0
25 Heating and Ventilating System	0.5
26 Plant Fire Protection System	0
27 Reactor Containment System	0.2
Reactor Systems	
31 Reactor Core	15
32 Reactor Vessel and Shielding	0
33 Reactor Nuclear Control Components	6
34 Nuclear Proof Test Facility	0
Component Handling Systems	
41 Reactor Refueling System	12.4
42 Non-Irradiated Fuel Handling	0.1
43 Irradiated Fuel Handling	2.5
44 Radioactive Maintenance System	5.0
45 Inert Gas Examination Facility	0
Reactor Heat Transport	
51 Heat Removal System	17
Irradiation Test Systems	
61 Closed Loop Test Systems	20
68 Short Term Irradiation System	2
Sodium and Inert Gas Receiving and Processing Systems	
81 Sodium Receiving and Processing System	4
82 Inert Gas Receiving and Processing System	1

TABLE 3.XVIII. (contd)

<u>Systems</u>	<u>Unavailability hr per 1000 hr</u>
Instrumentation, Control, and Data Handling Systems	
91 Control and Data Handling	3
92 Reactor Channel and Vessel Instrumentation	3.5
93 Plant Instrumentation Systems	3.0
94 Fuel Failure Monitoring System	1.5
95 Flux Monitoring and Control	2.5
96 Radiation Monitoring System	0.5
99 Plant Protection System	0.1
TOTAL	100.5

Calculations were made which relate the required number of installed thermocouple circuits in one channel with the failure rate of each thermocouple and with the number of required operating thermocouple circuits per channel. A table was prepared which relates this information with the probability of having at least the required number of circuits operating for one year. The purpose of the table is to provide information for use in determining how many circuits are to be installed in each process tube.

- b. Reliability Analysis of the Building Electrical Power System (Work Performed for System 10. Electrical)
O. B. Monteith

A preliminary reliability analysis of the FFTF Building Electrical Power System was completed in this quarter. The purpose of this analysis is to obtain an early prediction of the Building Power System reliability based on the Conceptual System Design Description dated December 8, 1967.⁽¹⁾ The Building

(1) M. F. Wiitala. Conceptual System Design Description of the Building Electrical Power System, BNWL-500-12. Pacific Northwest Laboratory, Richland, Washington, (Preliminary release, December 1967).

Electrical Power System is a combination of several subsystems with various interconnections. The conceptual design of System 12 is shown in the Functional Block Diagram of Figure 3.46. Results of the analysis yield the availability of electrical power to the bus, the mean-time-between-failure of power to the bus, and the mean-time-to-restore the power in event of failure.

A preliminary examination of System 12 performed here reveals no significant reliability shortcomings. There are no suggestions to be gleaned from this analysis for changes in equipment configurations to improve reliability. Any improvements to be achieved would seem to lie in the materials and methods of construction which are beyond the scope of this review.

(1) Analysis of the 2.4 kV System, Busses B5,6,7,8, & 9.

The 2.4 kV system is made up of five busses, each supplied from the 13.8 kV system through a circuit breaker, cable, and 13.8 kV/2.4 kV, 5 MVA transformer. Its major load is the FFTF Heat Transport System, System 50. The 2.4 kV system supplies four primary coolant loop pump motors, four secondary coolant loop pump motors, 16 heat dump blower motors, and three sodium heating loads. These loads are connected to four of the five busses, with the fifth bus energized but supplying no load.

In the event of power loss to any one of the four loaded busses, the fifth bus can be switched to pick up the load of the power deficient bus.

The results of the analysis are as follows:

For four out of five 2.4 kV busses

Availability of power = 0.999999785

Mean-time-between-failure = 73,000,000 hr

Mean-time-to-restore = 15.6 hr

FIGURE 3.46. System 12 Functional Block Diagram

(2) Analysis of the 480 V System, Busses B13,14,15,16,17, 18,19 & 20. The normal 480 V system comprises eight busses each supplied from the 13.8 kV system through a circuit breaker, cable, and 13.8 kV/480 V, 1.5 MVA transformer. After providing for the loads of the heat transport system, System 12 supplies the remaining FFTF electrical loads by the 480 V system as a normal source. The planned operation of the 480 V system provides for switching of the loads from a failed bus to an adjacent bus, through bustie breakers. In the event switching is required, some noncritical loads may be disconnected to adequately carry both loads on one bus.

All of the busses B13, 14, 15, 16, 17, 18, 19, and 20 are considered to be identical. The results of the analysis are as follows:

For one 480 V bus

Availability of power = 0.999999867

Mean-time-between-failure = 130,000,000 hr

Mean-time-to-restore = 17.5 hr

(3) Analysis of the Emergency Power Busses B11 and B12. The emergency power busses supply critical reactor auxiliaries and building service loads which include the pony motors for the primary and secondary sodium pumps, selected ventilation equipment, instrument and control systems, and others that are listed in the CSDD for System 12. There are two separate sources of power for the emergency busses: (a) normal power is provided from the 13.8 kV system by redundant paths of circuit breaker, cable, and 13.8 kV/480 V, 1.5 MVA transformer, and (b) emergency power from a redundant combination of a 1500 kVA diesel driven generator and a 1500 kVA gas turbine driven generator.

The two emergency power busses are considered to be identical for this analysis. The results of the analysis are as follows:

For the emergency bus

Availability of power = 0.999999999999565

Mean-time-between-failure = 15,100,000,000,000 hr

Mean-time-to-restore = 6.6 hr

(4) Analysis of the "Zero Time Outage" Busses. The "zero time outage" (ZTO) busses supply instrumentation and control loads which cannot tolerate momentary power loss. There are two such busses (each with the same arrangement) which have as their normal source the 480 V ac emergency power bus. In addition, each bus is connected to a synchronous alternator that is floating on the line and will deliver power through flywheel inertia in event of momentary outages. The alternator is also driven by a dc motor and clutch from the 125 V dc system. The dc drive will provide alternator power for extended outages.

The results of the analysis are as follows:

For one ZTO bus

Availability of power = 0.9999999615

Mean-time-between-failure = 106,000,000 hr

Mean-time-to-restore = 5.12 hr

(5) Analysis of the 125 V dc and 24 V dc Busses. The dc power systems contain three 125 V dc busses and two 24 V dc busses. Each of the busses is supplied from the normal 480 V system and is automatically switched to the emergency bus if the normal supply fails. A third source for the dc busses is provided by batteries that are paralleled with the busses and are kept in the charged state by the normal power and a dc power supply. The load on the dc systems include critical instrumentation, emergency lights, and controls.

The sources of power and equipment arrangements are the same for the 125 V dc and the 24 V dc systems. The only difference occurs in the dc power supply and the battery. It is possible to use a lower voltage circuit breaker for the 24 V dc system but the effect on the reliability analysis is insignificant.

The results of the analysis are as follows:

For the 125 V dc bus

Availability of power = 0.999999966

Mean-time-between-failure = 54,000,000 hr

Mean-time-to-restore = 1.83 hr

For the 24 V dc bus

Availability of power = 0.999999983

Mean-time-between-failure = 109,000,000 hr

Mean-time-to-restore = 1.85 hr

(6) Analysis of the 120 V ac Busses. There are two 120 V ac busses that serve instrumentation and control loads and are associated with the ZTO busses. However, they do not have the alternator source of the ZTO busses and are supplied by the 480 V ac emergency bus. The reliability logic for each of these 120 V ac busses is a series arrangement of circuit breaker, voltage regulator, and transformer connected to the emergency bus. In effect, this is a connected load of the 480 V emergency bus with the additional components mentioned above.

For one 120 V bus

Availability of power = 0.9999558

Mean-time-between-failure = 100,000 hr

Mean-time-to-restore = 4.4 hr

(7) Analysis of the Station Service. Power for the Station Service is utilized at 125 V dc. The normal supply for the Station Service is redundant circuits each with a circuit

breaker, cable, and 13.8 kV/480 V 300 kVA transformer. The 480 V is then converted to 125 V dc by a power supply. An alternate source for Station Service is a 125 V battery which is floating on the bus and kept in the charged state by the dc power supply. The load on the Station Service includes the operating coils of the 13.8 kV, 2.4 kV, and 480 V switchgear, interlocks, relay and substation lighting.

The results of the analysis are as follows:

For the Station Service dc Bus

Availability of power = 0.999999966

Mean-time-between-failures = 55,000,000 hr

Mean-time-to-restore = 1.88 hr

(8) Effects of Building Electrical Power System Failures on Reactor Availability. The Emergency Power Busses represent the only significant equipment in System 12 that will have an effect on reactor availability. For determining reactor unavailability it is assumed that any failure in either of the two normal sources to the emergency busses will cause the reactor to be shut down. The expected reactor downtime per 1000 hr operation due to failures in the emergency busses is:

$$\sum \lambda_i D_i (1000) = 0.76 \text{ hr}$$

where D_i is the component repair time plus 1 hr for the reactor to recover after a shutdown.

- c. Reliability Requirements for In-Channel Instrumentation
(This Work Performed for System 90. Instrumentation and Control)

R. A. Harvey

The reliability required of the in-channel instrumentation is sensitive to safety and operating philosophy. A comparison is presented here which shows how the operating philosophy affects the in-channel instrumentation reliability requirements.

The requirements are compared for defining a failure as (1) failure of the instrumentation function in a single tube, and (2) failure of the instrumentation function in two adjacent tubes.

(1) Single-Tube Failure Criterion. A single-tube failure criterion implies that if the instrumentation function is lost in any of the channels the reactor will be shutdown and the defective instrumentation will be repaired or replaced. It is assumed that there are 91 independent identical channels and that the probability of failure is the same for all channels. The reliability for all of the channels is

$$R_{(\text{reactor})} = R^{91} \quad (1)$$

where R is the single-tube reliability and $R_{(\text{reactor})}$ is the composite reliability for all of the channels. The reliability as used here is defined as the probability of performing without failure the required functions under the actual in-use conditions for a period of one year. The relationship between the single-tube instrumentation reliability and the reactor in-channel instrumentation reliability as determined by Equation (1) is listed in Table 3.XIX and is shown graphically in Figure 3.47 (single-tube criterion).

The required single-channel reliability can be determined if both the allowable reactor unavailability resulting from the in-channel instrumentation and the associated lost time per failure are known. A reactor unavailability of two hours per thousand hours is used as the basis for the reliability requirement. This number was determined by engineering judgement to be a reasonable allocation based on 75% reactor availability. The lost-time-per-failure is predicted as follows to be 56 hours:

TABLE 3.XIX. *Reactor Reliability Versus Single Tube Reliability for Two Failure-Criteria*

Single-Tube Reliability	Reactor Reliability	
	Single-Tube Criterion	Adjacent-Tube Criterion
0.9		
0.91		
0.92		
0.93	0.001	
0.94	0.003	0.533
0.95	0.01	0.696
0.96	0.024	0.837
0.97	0.063	0.942
0.98	0.16	0.988
0.985	0.253	
0.99	0.40	0.999
0.993	0.527	
0.996	0.694	
0.997	0.761	
0.998	0.833	
0.999	0.913	
1.0	1.0	

FIGURE 3.47. *Reactor Reliability Versus Single Tube Reliability for Two-Failure Criteria*

Initial fuel cooldown	8 hr
Additional cooldown so fuel handling machine can handle	4 hr
To get subassembly out and transferred to the gas exam facility	8 hr
To replace the instrument components	4 hr
To replace the fuel	8 hr
To reach reactor criticality	16 hr
To reach full power	<u>8 hr</u>
Total	56 hr

The allowable mean-time-between-failures (MTBF) can be determined from ⁽¹⁾

$$A = \frac{MTBF}{MTBF + MTR} \quad (2)$$

where A is availability (=0.998), and MTR is the mean-time-to-repair (= 56 hr). From Equation (2) the MTBF was determined as 28,000 hr, which is equivalent to a 1-year reliability of 0.731.

From Figure 3.47 for the single-tube failure criterion, it is noted that the single-channel instrumentation reliability must be approximately 0.997 in order for the corresponding overall reliability to be 0.731.

(2) Adjacent-Tube Failure Criterion. The adjacent-tube failure criterion implies that the reactor will not be shut down for failure of the instrumentation function in any one tube, but will be shut down for failure of the instrumentation function in two adjacent tubes. Operation can continue with single-tube failures by dropping down slightly in power and using the adjacent-channel instrumentation to calculate failed channel conditions. It is assumed that the average number of adjacent tubes per channel is five.

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(1) *Handbook - Reliability Engineering, NAVWEPS 00-65-502.*

It can be reasoned roughly that on the average, five channels will fail before two of the failed channels are adjacent. The probability that the second failed tube is not adjacent to the first failed tube is 85/90. The probability that the third one is not adjacent to either of the first two is 79/89. The joint probability that five tubes fail without being adjacent to each other is thus determined to be 0.535; hence five tube failures is used as the base for calculating adjacent channel probabilities.

The relationship between the reliability of a single channel instrumentation function and the probability of having five channels fail, when there are a total of 91 channels that can fail, is given by

$$R_{(\text{reactor})} = \sum_{K=87}^{91} R_K^{91} \quad (3)$$

where R_K^{91} , the probability that exactly K of the 91 channels operate without failure for a year, is given by the binomial distribution.

$$R_K^{91} = C_K^{91} P^K Q^{(91-K)} \quad (4)$$

where P is the single channel reliability.

Q is the single channel unreliability ($=1-P$), and

C_K^{91} is the number of combinations of 91 things taken K at a time.

$$C_K^{91} = \frac{91!}{K!(91-K)!} \quad (5)$$

The results of the calculation for various values of single channel reliabilities are listed in Table 3.XIX and shown graphically in Figure 3.47 (Adjacent Tube Criterion).

It is assumed that on the average, three tube instrumentation functions are restored each time that a 5-tube failure occurs. Some tubes will not be repaired because they will be

removed at the next scheduled shutdown for routine refueling. The mean-time-to-repair (MTR) for this type of failure includes the 56 hr as for single channel repair plus 20 hr for each of the two additional channels being repaired. The total MTR thus is determined to be 96 hr. From Equation (2), and still with an availability, A, of 0.998, the MTBF is 48,000 hr, which corresponds to a 1-year reliability requirement of 0.833 for the in-channel instrumentation. From Figure 3.47 it can be seen that the corresponding single channel instrumentation reliability is approximately 0.96.

d. Reliability Requirements for In-Channel Thermocouple Circuits (This Work Performed for System 90. Instrumentation and Control)

R. A. Harvey

The number of installed thermocouple circuits in each of the reactor channels is determined by the safety circuit philosophy, the operation philosophy, the reactor availability requirements, and the reliability of a thermocouple circuit. The circuit includes the sensing element, the wires, the connector, and anything else that can cause the circuit to fail which requires removal of the in-channel subassembly to repair.

Since little information is available, a calculation result is presented which can be used to determine the number of thermocouple circuits to install in each channel if the following are known:

- The channel required reliability
- The number of circuits required to operate in each channel
- The reliability of a thermocouple unit.

The binomial distribution was used as follows:

$$P_i^n = C_i^n P^i Q^{n-i} \quad (6)$$

where

P_i^n is the probability that exactly i out of n circuits in the channel survive for 1 yr

C_i^n is the number of combinations of n things taken i at a time

P is the 1 yr reliability of each circuit, and

Q is the 1 yr unreliability of each circuit ($= 1 - P$).

The reliability, P , of each circuit is determined from the failure rate, λ (failures per year), by

$$P = e^{-\lambda} \quad (7)$$

The channel reliability is determined for n installed circuits of which at least K are required to operate by

$$R_K^n = \sum_{i=K}^n P_i^n \quad (8)$$

Failure rates of 0.01, 0.1, and 1 failures per year were used. The numbers of both required operating circuits and installed circuits per channel were varied from one to seven to scan the range of interest. The reliabilities of the channel thermocouple function for these various conditions are listed in Table 3.XX.

As an example of the use of Table 3.XX, assume that the only required instrumentation function in the channel is the thermocouple function. Assume also that the failure rate of 0.1 failures per year is applicable, and that the safety circuits require that two thermocouple circuits be operable in order for the function to be considered successful. If the single-channel failure criterion is applicable, the channel reliability must be 0.997 or higher (from Figure 3.47). It can be

TABLE 3.XX. *Probability of Having at Least the Required Number of Circuits Operating for One Year (R_K^n)*

Number of Installed Circuits, <u>n</u>	Number of Required Operating Thermocouple Circuits per Tube (K)						
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>
<u>$\lambda = 0.01$ Failures/year per circuit</u>							
1	0.99						
2	1	0.98					
3	1	1	0.97				
4	1	1	0.999	0.961			
5	1	1	1	0.999	0.951		
6	1	1	1	1	0.998	0.941	
7	1	1	1	1	1	0.998	0.932
<u>$\lambda = 0.1$</u>							
1	0.905						
2	0.991	0.819					
3	0.999	0.975	0.741				
4	1	0.997	0.953	0.671			
5	1	1	0.993	0.926	0.607		
6	1	1	0.999	0.986	0.895	0.549	
7	1	1	1	0.998	0.978	0.863	0.498
<u>$\lambda = 1$</u>							
1	0.368						
2	0.600	0.135					
3	0.748	0.306	0.050				
4	0.763	0.468	0.114	0.018			
5	0.896	0.602	0.264	0.064	0.007		
6	0.993	0.710	0.386	0.136	0.028	0.003	
7	0.954	0.792	0.506	0.226	0.068	0.012	0.001

$$P_i^n = C_i^n P^i Q^{n-i} \text{ and } R_K^n = \sum_{i=K}^n P_i^n$$

where

C_i^n is the number of combinations of n things taken i at a time

P is the 1 yr survival probability of one circuit, and

Q is the 1 yr failure probability of one circuit ($P = e^{-\lambda}$).

seen from Table 3.XX that for $\lambda = 0.1$ and for two circuits per tube required, there must be four installed circuits. If the adjacent channel failure criterion is applicable, the channel reliability must be 0.96 or higher and from Table 3.XX it can be seen that three installed circuits satisfy this condition.

3. Radiological and Environmental Analysis (This Work Performed for System 20. Structures)

J. P. Hale

a. Siting Studies

M. M. Hendrickson

A site has been proposed for the FFTF on the Hanford AEC reservation which is about five miles from projected residential areas southwest of the reactor. The nearest area of significant population is the Hanford Laboratory 300 Area about five miles to the southeast where about 3000 employees work for several Hanford contractors.

This discussion gives the projected radiation dose (following a postulated FTR Design Basis Accident) at existing plant boundaries and at arbitrary points, including those which would correspond to the Exclusion Area Boundary (EAB) and Low Population Zone Boundary (LPZB), as defined in the Code of Federal Regulations.

For comparison purposes we have chosen a basic case which is composed of a 2-barrier containment system, the inner system having a leak rate of 10% per day and the outer 1% per day. This discussion is not recommending either 2-barrier containment or the leak rates mentioned.

Summary and Conclusion. For a low leak rate containment system, the hypothetical exclusion area boundary and low population boundary for the FFTF are determined by the external whole body dose from radionuclides in the containment vessel. These

boundaries are found to be Exclusion Area Boundary, 0.3 miles and Low Population Zone Boundary 0.8 miles (see Table 3.XXI). Actual plant boundaries are substantially greater than this and are in the range of 3.5 and 5.5 miles to the east and south. Low leak rate systems are considered in this report to be single containment systems of 0.1% per day leak rate or less, or double containment systems where the hot cell leaks 10% per day or less and the outer vessel 1% per day or less. For systems with significantly higher leak rates, the bone, thyroid, and lung dose are of significance in that order. Depending upon the comparison criteria chosen, the bone, thyroid, or lung dose may be so significant that one of these determines allowable leak rates rather than noble gas leakage.

Data on aerosol behavior have been applied in these studies. The application of these data indicate a reduced significance of the bone, thyroid, and lung dose as compared to the direct gamma dose from fission products suspended or deposited in the containment system. If these aerosol data are not applied, very low leak rate double containment systems are indicated in order to meet bone dose values which have been arbitrarily selected for site comparison purposes. These site comparison values are 30 rem to bone first year dose, 150 rem to bone lifetime dose, 15 rem to lung first year, and 75 rem to lung lifetime dose.

Table 3.XXI shows the dose to the total body and to the bone, lung, and thyroid from exposure at 0.3, 0.8, 3.5 and 5 miles from the proposed FTR. The tabulated results are for a postulated design basis accident. The consequences are well below established criteria and arbitrary site comparison criteria for both the single and double containment cases.

TABLE 3.XXI. Dose at Hypothetical and Actual Boundary Distances

Base Case ^(a)	Dose in rem at			
	EAB, ^(c) 0.3 mi	LPZB, ^(d) 0.8 mi	Actual Plant Boundaries ^(e)	
			East, 3.5 mi	S.W., 5 mi
External Whole Body				
Dose from contained radio-nuclides and plume	25	25	0.001	0.001
Thyroid Dose (Inhalation)	5	5	1	1
Bone Dose (Inhalation)	13	14	3	2
Lung Dose (Inhalation)	0.5	0.6	0.1	0.1
Single Containment ^(b)				
External Whole Body	25	25	0.05	0.03
Thyroid Dose (Inhalation)	240	80	10	7
Bone Dose (Inhalation)	540	180	30	18
Lung Dose (Inhalation)	22	8	1	0.7

(a) Double Containment Base Case - 10%/day leakage hot cell and 1%/day leakage from outer containment.

(b) Single Containment - 0.1%/day leakage

(c) Exclusion Area Boundary as defined in 10CFR100, dose from two hours exposure

(d) Low Population Zone Boundary as defined in 10CFR100, dose from 30 days exposure

(e) Lifetime dose resulting from exposure for the duration of the accident

IV. DEVELOPMENT

D. L. Condotta

HIGHLIGHTS OF FFTF FUELS

W. E. Roake

Analytical computer codes successfully predicted fuel pin swelling by using data obtained from the DFR MK-1 fuel pin.

Simulated fuel pin swelling experiments have verified accommodation of fuel pin diametral growth by pin deformation into the spiral pattern of the wire wrap without denting or deforming the cladding from its round cross-section when restrained in a fuel bundle.

Good correlations have been obtained for fuel pellets between green density and sintered pellet shrinkage, between green density and sintered density, and between press pressure and green density.

Prototypical FTR driver fuel subassemblies were fabricated for thermal, hydraulic, and mechanical testing in flowing sodium and water and for process development studies.

An improved lower grid design was developed to minimize radiation exposure to personnel during assembly and to eliminate spot welding as a primary means of attaching strip layers to locking strips.

Based on a series of screening capsules a side-grooved 304 SS floating restrictor was chosen for use in the long fuel column thermal transient tests.

The first four of six capsules containing prototypical length fuel columns were irradiated to the goal burnup of 10,000 MWd/tonne in GETR.

One of a series of capsules to study the effects of fuel stoichiometry failed as a result of operation at 50% above design power. Destructive examination showed a fuel-cladding reaction about 0.002-in. thick.

The first PNL fast reactor irradiation experiment, PNL-1, a 19-pin encapsulated subassembly, completed its goal burnup of 10,000 MWd/tonne in the EBR-II.

Flow testing of a mockup 37-pin subassembly using 0.250-in. diameter pins and 0.0625-in. diameter wire wrap indicated slightly higher flow rates than were calculated for positions in rows 1 through 6 of EBR-II.

The effect of oxygen-to-metal (O/M) ratio on melting heat rating was measured. Correlations between postirradiation fuel structure and O/M ratio were noted. Pore morphology, second phase content, and void distribution varied systematically with oxygen content of the fuel.

IV. DEVELOPMENT

D. L. Condotta

A. FFTF FUELS

W. E. Roake

This section reports driver fuel development work within sub-department. A cross index of other related work is given on page 4.37.

1. Fuel Fabrication Process Demonstration

C. A. Burgess

FFTF Fuels has the responsibility for developing and evaluating the capabilities of commercial fuel suppliers to fabricate FFTF fuel cores. The first step in this development is the LMFBR Fuel Supplier Prequalification Program in which potential FTR vendors will be evaluated according to their capability to handle plutonium-bearing fuel material and to fabricate a limited number of finished fuel pins. In addition, a Fuel Fabrication Demonstration Facility capable of fabricating FTR fuel assemblies and demonstrating an acceptable process will be built in support of the FTR.

a. LMFBR Fuel Supplier Prequalification Program

C. A. Strand and R. E. Bardsley

The necessary documents for starting the LMFBR Fuel Supplier Prequalification Program were prepared. These include the Management Plan for LMFBR Fuel Fabrication Prequalification, the FFTF Driver Fuel Pin Produce Specifications, and the supporting Standards for Nondestructive Testing and Analytical Chemistry Procedures.

b. FFTF Fuel Fabrication Demonstration Facility

J. W. Thornton and M. D. Jackson

The pin portion of the Fuel Fabrication Demonstration Facility, is proceeding on schedule. The design for all of the

fuel fabrication gloveboxes was completed. Plywood mockups of the hood designs were built to confirm design dimensions, to test equipment placement, and to test material handling systems. The engineering specifications were completed for all but two items (pellet gaging and fuel pellet pin loading) required for this project.

2. Mechanical Analysis and Testing

P. D. Cohn

The purpose of this task is to determine analytically and experimentally the mechanical behavior of the FTR fuel pins and subassembly. It incorporates all phases of mechanical analysis for the FTR fuel members and includes correlation and application of experimental data.

a. Fuel Pin Diameter Increase Calculations

J. B. Shafer and K. R. Merckx

Comparisons were made of the calculated and measured radial growth of the DFR MK-1 fuel pin. The SINTER code was used to calculate the radial temperature distributions at various locations along the fuel pin. These temperature distributions were then used as input information for the DEFORM⁽¹⁾ program. The comparison of measured values with calculated values for radial deformation is shown in Figure 4.1. The radiation dependence of creep in the cladding and the coefficient of viscosity was adjusted in DEFORM to obtain these results. It is apparent that except at the top of the fuel pin the calculated diametral growth agrees closely with actual measurements.

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(1) K. R. Merckx. A Program for Calculating the Steady-State Deformation Rates for Cylindrical Fuel Elements, BNWL-663. Pacific Northwest Laboratory, Richland, Washington, January 1968.

FIGURE 4.1. Calculated and Measured Values of Diametral Growth of DFR MK-1 Fuel Pin

b. Wire-Wrap Behavior During Fuel Pin Swelling

R. J. Lobsinger

Simulated swelling tests were performed to determine the effect of diametral changes on wire-wrapped fuel pins. In the initial burst tests empty, wire-wrapped cladding tubes were internally pressurized with gas; Figure 4.2 shows a typical result. The failed tube burst at a pressure of 10,200 psig. Release of contained gas caused the failure to propagate as shown. The second tube was pressurized to 8,000 psig for 1 min. No deformation of the tubes resulting from the wire wrapping occurred in either case.

Subsequent tests were performed on simulated fuel pins using water-filled compartments in the fueled region. Fuel expansion was simulated by freezing or heating. These tests demonstrated that axial shortening of the cladding tubes occurred during unrestrained (unzoned) pressurization. This condition promoted loosening of the wire wrap and tended to obscure interaction effects. This type of expansion is not typical of reactor behavior so zoned expansion tests were run.

A seven-pin subassembly was tested. It consisted of nonfueled 304 SS cladding, 0.250 in. OD x 0.016 in. wall thickness, containing a water-filled compartment in the

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FIGURE 4.2. *Wire-Wrapped Cladding Tubes After Burst Tests*

simulated fuel region. The expansion of the water upon heating was designed to produce swelling. The pins were wire-wrapped with 0.030 in. wire, assembled in a prototypical array, and constrained in a tight-fitting cylindrical shroud tube to prevent radial movement. Axial shortening was prevented by application of tension on the threaded end fittings. Heating in an electric furnace to 260 °C produced 2 to 2.5% diametral expansion on each pin. No interaction occurred with the shroud tube or wire wrap. Accommodation for swelling was achieved through distortion of the array. This distortion produced bending of the fuel pins in the region of the swelling as shown in Figure 4.3.

3. Fuel Process Development and Irradiation Testing

B. R. Hayward

The objectives of the fuel process development task are to (a) define optimized mixed oxide fuel fabrication unit operations, (b) develop fabrication data and procedures for selecting a reproducible and reliable fuel fabrication process, (c) develop feasible and reliable unit operations for manufacturing full-scale FTR driver fuel pins and subassemblies, and (d) define product requirements for FTR driver fuels, pins, and subassemblies through close interaction with FTR fuel design and analysis studies.

Fuel performance under both steady-state and transient conditions is a complex interrelationship of key parameters such as density, fuel form, void deployment, burnup level, and cladding characteristics. The irradiation testing program focused on development of the initial FTR driver fuel is providing information on these parameters.

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FIGURE 4.3. *Bending of Fuel Pins Subjected to Swelling
with Radial and Axial Restraint*

a. Mixed Oxide Fuel Process Development

H. G. Powers, R. J. Evans, M. J. Barr

Fuel pellet parametric studies included: (1) comparison of the effect of a dry lubricant (zinc behenate), and the standard lubricant (Carbowax 20M in water) upon fuel pellet sinterability, (2) determination of the effect of calcination temperature of PuO_2 derived from the oxalate precipitation process upon sintered density of $(\text{U}_{0.75} + \text{Pu}_{0.25})\text{O}_2$ fuel pellets, and (3) correlation of green density with press pressure percentage shrinkage, and sintered density. These experiments showing basic relationships produced useful data for a comprehensive factorial experiment designed to determine both main and interacting effects of the more important process variables. Feed materials selected for the experiment were oxalate precipitated calcined PuO_2 and sintered grade Eldorado UO_2 .

The dry binder was evaluated as a potential means of eliminating the long drying time required for small critically-safe batches of powder containing wet binders. Results of the test are shown in Table 4.I. Although the data indicate consistent densities with pellets made from burned metal PuO_2 , ceramography indicated those pellets made with dry binder (zinc behenate) had numerous transverse cracks with some running from the center of the pellet to the surface. The process used during these tests was granulation, preslug, regranulation, final pellet press, and sinter. Adding dry binder before the preslug step made the final press very difficult. Many pellets were scored or chipped indicating insufficient die lubrication. The cracks observed in the sintered pellets may be due to inhomogeneity of binder and oxide or perhaps to a high preslug pressure.

TABLE 4.I. *Mixed Oxide Sintering Test*
(PuO_2 from Burned Metal,
Batch ME-26)

Binder	Sintered Density, % TD
10 wt% Solution Containing 2 wt% Carbowax 20M	93.3
2 wt% Zinc Behenate (a)	93.5
2 wt% Zinc Behenate (b)	93.6

(a) Binder added before preslugging.

(b) Binder added after preslugging.

The calcination temperature of oxalate source PuO_2 was shown to interact with sintered density of mixed oxide fuel pellets. Table 4.II lists the effect of calcination temperature on sintered density of mixed oxide pellets containing 25 wt% PuO_2 . All samples including the control samples indicated lower densities in pellets which used zinc behenate binder.

TABLE 4.II. *Effect of Calcination Temperature on Sintered Density - Oxalate Source PuO_2*

Calcination Temp., °C	Binder	Sintered Density, % TD
450 (as received)	Carbowax	97.4
600 (a)	Carbowax	93.9
	Zinc Behenate	91.7
700 (a)	Carbowax	94.5
	Zinc Behenate	92.6
800	Zinc Behenate	96.2
1/3 950 2/3 800 (mixed batch)	Carbowax	88-90

(a) Control sample indicated approximately 2% TD lower than previous runs.

Good correlation between green density and sintered pellet shrinkage has been observed. The relationship shown in Figure 4.4 indicates that the amount of shrinkage is largely dependent upon the blend (particulate) characteristics. Once the relationship is known for a particular batch, a specific sintered diameter may be obtained by selecting a specific die size and green density.

FIGURE 4.4. Relationship Between Green Density and Diameter Shrinkage of Fuel Pellets

The relation between green density and sintered density is shown in Figure 4.5. The curves show that there is an optimum green density range for maximum sintered density and that green densities below and above this optimum will produce a slightly lower sintered density. It should be noted that a 13% range in green density produced only a 4% variation in sintered density.

FIGURE 4.5. Relationship Between Green Density and Sintered Density of Fuel Pellets

The curve in Figure 4.6 indicates that the press pressure-green density relationship is not linear.

FIGURE 4.6. Relationship Between Green Density and Press Pressure Experimental Batch #1 - No Binder

b. Subassembly Process Development

H. G. Powers, J. W. Thornton

Subassembly process development activities included the design and development of process equipment and the modification of assembly techniques to provide improved reliability.

A wire wrap machine previously used for wrapping PRTR fuel pins was modified to permit wrapping small diameter (0.250-in.) fuel pins without incurring distortion. Fuel pin support frames were designed and fabricated to permit strip layers to be handled without damage. Drill and machine weld fixturing for pinning fuel pins to strip layers was designed and fabricated. Equipment for assembly of the 217-pin fuel bundle and installation of an 8-foot section of flow duct was designed and fabricated. The equipment was designed for vertical assembly but permitted lowering the completed subassembly to the horizontal position and rotation of the subassembly in the horizontal position (Figure 4.7).

The fuel bundle of the subassembly consists of a hexagonal array of fuel pins formed by stacking strip layers varying in width from 9 to 17 pins (Figure 4.8). Strip layers are attached to the lower grid which in turn is attached to the flow duct (Figures 4.9 through 4.11). An upper grid is attached to three central pins of the fuel bundle to maintain lateral positioning without restricting independent axial expansion of fuel pins (Figure 4.12). A flow duct is installed over the completed fuel bundle (Figure 4.13) and a hexagonal array of encapsulated neutron reflector pins is installed below the fuel bundle. A completed subassembly for mechanical, thermal, and hydraulic testing in flowing sodium is shown in Figure 4.14. Two of these subassemblies were completed, and a third was fabricated for developmental and demonstration purposes.

FIGURE 4.7. *Assembly Fixture for Fuel Subassembly Showing the Horizontal and Vertical Positioning Capability*

4.13

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FIGURE 4.8. Assembling Strip Layers

4.14

BNWL-880

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FIGURE 4.9. *Welding of Strip Layer Retaining Pins to End Cap*

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FIGURE 4.10.

*Positioning Strip Layer in Bottom Grid
Tray Support*

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FIGURE 4.11. *Welding Strip Layer to Bottom Grid*

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FIGURE 4.12. *Closeup of Top Grid in Place on the Subassembly*

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FIGURE 4.13. Completed Z17-Pin Subassembly Held in Assembly Fixture

4.19

BNWL-880

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FIGURE 4.14. Examination of Completed Fuel Subassembly

A technique was developed for pinning and welding the bottom end caps of the fuel pins to the grid strips. Two metallographic sections taken transversely through a bottom end cap, in the region of the pin and weld are shown in Figure 4.15. The photographs show the end cap (a) which is slotted to fit over the grid strip (b), and the pin (c) placed through a hole in both the cap and the grid strip and spot welded at both ends. Fifteen assemblies were shear tested. The average shear force required to separate the fuel pin and grid strip was found to be 299 lb with a range of 53 lb.

Measurements taken to determine the effect of accumulated dimensional tolerances on the distance across the flat dimensions of the fuel bundle are tabulated below. These data indicate a minimum clearance of 0.025 in. between the fuel bundle and flow duct except in the lower grid section. A slight misalignment of grid sections is responsible for the irregular dimensions at the lower grid position.

Across Flat Dimensions of Fuel Bundle, in. (a)

<u>Position</u>	<u>(1)</u>	<u>(2)</u>	<u>(3)</u>
Lower Grid	4.200	4.200	4.180
+1 ft	4.186	4.190	4.190
+2 ft	4.187	4.188	4.188
+3 ft	4.188	4.188	4.188

(a) *Design nominal, 4.190 in.*

Design dimension across flat dimensions of flow duct = 4.220 ± 0.005 in.

The existing assembly technique for fastening grid strips to the bottom subassembly grid structure by spot welding is shown in Figure 4.16. This technique could present problems in the areas of weld quality certification and personnel radiation

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20X

Neg 568-5143A

30X

FIGURE 4.15. Transverse Metallographic Sections Welded Through the Bottom End Cap Where It Is Pinned to the Grid Strip. A: End Cap; B: Grid Strip, C: Pin

4.22

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FIGURE 4.16. Bottom Grid, FTR Driver Fuel Subassembly in Assembly Fixture Showing Spot Welded Grid Strips

BNWL-880

exposure during assembly. An alternative grid design to overcome these problems was completed and is shown in Figures 4.17 and 4.18. The concept does not require welding and should facilitate the disassembly of test fuels.

c. Thermal Flux Irradiation Tests

G. E. Culley, L. A. Pember, J. W. Weber

The objective of this task in the FTR driver fuel development program is to evaluate through steady-state and transient thermal flux irradiations those performance characteristics of prototypical FTR mixed oxide fuel pins that are not strongly dependent upon the flux spectrum. The irradiation tests being conducted are designed to provide data that will contribute to the assurance that safe operation of a long length fast reactor core using mixed oxide fuel can be achieved. Prototypical FTR fuel pin characteristics being evaluated include:

- The propensity of mixed oxide fuel to relocate during steady-state and transient irradiation conditions
- The mechanical and chemical behavior of cladding and fuel in contact with sodium
- The plutonium and fission product migration characteristics in mixed oxide fuel
- The characteristics of various axial fuel motion restrictors.

The irradiation of a series of 18 capsules in the MTR to evaluate the performance of several candidate restrictor designs was completed. These restrictor screening capsules (Figure 4.19) were operated for approximately 30 min with 50% of the fuel radius molten. Restrictor materials subjected to the rigors of molten fuel included tungsten, 304 SS, molybdenum, nickel, zirconium oxide, thorium oxide, and depleted uranium oxide. The geometry of the restrictors was either annular, solid, or solid with longitudinal grooves machined into the circumference.

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FIGURE 4.17. *Alternative Design of Lower Grid, FTR Driver Fuel Subassembly*

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FIGURE 4.18. *Alternative Grid Partially Disassembled*

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FIGURE 4.19. *Graphic Illustration of Restrictor Screening Capsules*

In addition, the restrictors were either free to move axially or fixed to the cladding by crimping. The postirradiation evaluation of the restrictor performance was based on:

- Prevention of molten fuel motion
- Mechanical interactions with the cladding
- Structural integrity of the restrictors
- Short-term restrictor-cladding-fuel compatibility.

The detailed results of the postirradiation examination are reported in BNWL-623.⁽¹⁾ Based upon these results, a fuel motion restrictor design was selected for use in the long fuel column thermal transient tests. This design was a side-grooved 304 SS floating restrictor chosen because of its capability to

1. A. E. Waltar, J. W. Hagan, J. E. Hanson, and L. A. Pember. Considerations on the Use of Fuel Motion Restrictors in the FTR Fuel Pins, BNWL-623. Pacific Northwest Laboratory, Richland, Washington, November 1967.

stop the axial flow of molten fuel by solidification while allowing the passage of fission gas past the restrictor.

The second type of thermal irradiations is carried out in the GETR. The instrumented capsule design is shown in Figure 4.20. The fuel pin under irradiation is prototypical of the FTR fuel pin in design and operating conditions. This series of tests consists of 12 capsules, PNL-59-1 through -12, with the odd-numbered capsules containing fuel pins with axial fuel motion restrictors. The nominal operating characteristics of these fuel pins are shown in Table 4.III.

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FIGURE 4.20. *Irradiation Capsule for Prototypical Length Fuel*

TABLE 4.III. *Nominal Operating Characteristics of GETR Capsules*

Maximum Power	14.5 kW/ft
Peak Power/Average Power	1.45
Maximum Fuel Cladding OD Temperature	829 °F
Maximum Fuel Surface Temperature	1808 °F
Maximum Fuel Centerline Temperature	4046 °F

The fuel in each pin is 25% PuO_2 + 75% UO_2 , 93% of TD pellets, 0.212-in. diam by 0.250-in. long, cladding with 304 SS 0.250-in. OD by 0.218 in. ID which gives a smeared density

88% of TD. The active fuel column is 31 in. long with 0.50 in. of depleted UO_2 pellets on both ends. The upper portion of the fuel pin contains a 5 in. nickel reflector and an extensometer device.

The capsule irradiations were initiated on February 19, 1968, and the irradiation of capsules PNL 59-1, 2, 3, and 4 to the goal burnup of 10,000 MWd/tonne was completed on June 6, 1968.

The third type of irradiations to be conducted is transient irradiations in the TREAT facility. These tests are conducted to evaluate the transient behavior of prototypical length fuel columns. Tests are conducted on fuel pins of three categories: pins preirradiated to 50,000 MWd/tonne, pins preirradiated to 10,000 MWd/tonne, and pins having no preirradiation. The fuel pins that are preirradiated are irradiated under steady-state conditions in GETR as described above and the GETR capsule is inserted into the TREAT capsule as shown in Figure 4.21. There will be four capsules at each burnup level available for TREAT testing.

During this report period, three of the four capsules not to be preirradiated were fabricated, four capsules were preirradiated to 10,000 MWd/tonne, and four capsules began an irradiation exposure to 50,000 MWd/tonne.

d. Thermal Flux Irradiations of Hypostoichiometric UO_2 - PuO_2 Materials

J. E. Hanson, L. A. Pember

The objective of this task is to evaluate the high exposure performance of mechanically blended low-density solid pellet and high-density annular pellet UO_2 - PuO_2 fuel as a function of stoichiometry. The oxygen-to-metal (O/M) ratios being evaluated are 1.94, 1.96 and 2.00. Twelve capsules designed as shown in Figure 4.22 are being irradiated.

4.29

BNWL-880

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FIGURE 4.21. *Transient Irradiation Capsule*

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FIGURE 4.22. Capsule for Oxygen-to-Metal Studies

One capsule suffered a lead tube failure and was removed. A second capsule, BNW-1-8, was identified as operating at about 50% above design power. This capsule contained annular pellets 95% TD with an O/M of 1.94 and it failed after a burnup of 21,000 MWd/tonne. First indications of failure were signaled by erratic thermocouple readings. Subsequent neutron radiography showed a bulge in the fuel pin near the bottom and another at approximately the axial center of the fuel column. Destructive examination showed that there was fuel-cladding reaction about 0.002 in. thick visible in most sections of the cladding. There appeared to be very little fuel-NaK reaction.

e. Fast Flux Irradiations in Support of FFTF Driver Fuel Development

J. E. Hanson and L. A. Pember

The objectives of this irradiation program are to:

- Provide statistically significant steady-state fast flux irradiation experience with reference and alternative UO_2 - PuO_2 FTR driver fuel at the anticipated minimum, average, and peak FTR linear power ratings of 5 to 7 kW/ft, 9 to 11 kW/ft, and 15 to 17 kW/ft, respectively (Table 4.IV shows the reference and major alternative fuel design parameters for the test program);
- Establish fuel performance limits and determine failure mechanisms under nonstandard (transient) operation conditions;
- Provide fast flux irradiation data for correlation with thermal-flux irradiation data to assess fuel performance as a function of neutron energy spectrum.

TABLE 4.IV. *Reference and Alternative FTR Fuel Design Parameters for Irradiation Testing Program*

<u>Design</u>	<u>Pellet Density, % TD</u>	<u>Fuel Cladding Diametral Gap, mils</u>	<u>Smeared Density, % TD</u>
Reference	93	6	88
Alternative No. 1*	96	6	85
Alternative No. 2	92	8	85
Alternative No. 3	91	6	85

* Annular pellet: ID = 0.054 in.

Two subassemblies of 19 encapsulated fuel pins are being irradiated in the EBR-II as PNL-1 and PNL-2, designated by ANL as X031 and X032 respectively. Table 4.V lists the detailed parameters for each of the 19 pins in the two subassemblies. PNL-1 was discharged at the end of EBR-II Run 28 at the goal exposure of 10,000 MWd/tonne of heavy atoms. Disassembly and examination will start after a short cooling period.

Five encapsulated fuel pins in X039 subassembly are being irradiated to provide information on the following parameters:

- Kinetics of fuel cladding gap closure
- Effect of initial gap size on the fuel radial temperature profile
- Fission gas release as a function of temperature
- Extent of fuel restructuring at low and medium power levels.

The fuel pins contain $\text{PuO}_2\text{-UO}_2$ fuel at a constant smeared density 88% of TD with fuel cladding diametral gaps of 0.004, 0.006, and 0.008 in. The parameters for the pins and the irradiation conditions are shown in Table 4.VI.

TABLE 4.V. Fuel Pin Parameters for EBR-II Irradiations of PNL-1 and PNL-2 (304 SS Cladding)

Encapsulated Pin ^(a) Designation	Fuel Column Length, in.	Fuel Pellet OD, in.	Fuel Pellet ID, in.	Fuel Pellet Density, % TD	Fuel Smeared Density, %TD	Fuel Weight, g	Fuel- Cladding Diametral Gap, in.	Cladding OD, in.	Cladding Thickness, in.	Axial Motion Restrictor	Transient Irradiation ^(b)
PNL-1-1	13.5	0.212	-	93	88	80.62	0.006	0.250	0.016	no	no
2	↓	↓	-	↓	↓	80.62	↓	↓	↓	no	yes
3	↓	↓	-	↓	↓	79.62	↓	↓	↓	no	yes
4	↓	↓	-	↓	↓	79.12	↓	↓	↓	yes	no
5	↓	↓	-	↓	↓	79.12	↓	↓	↓	yes	yes
6	↓	↓	-	↓	↓	79.12	↓	↓	↓	yes	yes
7	↓	↓	0.054	96	85	77.87	↓	↓	↓	no	no
8	↓	↓	↓	↓	↓	77.87	↓	↓	↓	no	yes
9	↓	↓	↓	↓	↓	77.87	↓	↓	↓	no	yes
10	↓	↓	↓	↓	↓	76.43	↓	↓	↓	yes	no
11	↓	↓	↓	↓	↓	76.43	↓	↓	↓	yes	yes
12	↓	↓	↓	↓	↓	76.43	↓	↓	↓	yes	yes
13	↓	↓	-	93	88	80.62	↓	0.238	0.010	no	no
14	↓	↓	-	93	88	80.62	↓	↓	↓	no	no
15	↓	↓	0.054	96	85	77.87	↓	↓	↓	no	no
16	↓	↓	0.054	96	85	77.87	↓	↓	↓	no	no
17	↓	↓	-	89.9	85	77.87	↓	↓	↓	no	no
18	↓	↓	-	89.9	85	77.87	↓	↓	↓	no	no
19	↓	0.210	-	91.6	85	77.87	0.008	↓	↓	no	no

(a) Pins in subassembly PNL-2 are identical to those in PNL-1

(b) Subsequent to steady-state irradiations

PNL-1 goal exposure - 10,000 MWd/tonne;
specific power - 10.8 kW/ftPNL-2 goal exposure - 50,000 MWd/tonne;
specific power - 10.8 kW/ft

TABLE 4.VI. *Fuel Pin Parameters for EBR-II Irradiations of PNL-X (304 SS Cladding)*

<u>Pin Number</u>	<u>Pellet Diam, in.</u>	<u>Pellet Density, % TD</u>	<u>Fuel-Cladding Diametral Gap, in.</u>	<u>Smeared Density, % TD</u>	<u>Peak Pin Power, kW/ft</u>
PNL-X-1	0.2135	91.3	0.004	88	~ 6
PNL-X-3	0.2115	93	0.006	88	~ 6
PNL-X-4	0.2095	94.8	0.008	88	~ 6
PNL-X-6	0.2115	93	0.006	88	~10
PNL-X-7	0.2095	94.8	0.008	88	~10

Flow testing is being conducted at ANL on mockup bare pin subassemblies in preparation for irradiation of unencapsulated 37-pin subassemblies in EBR-II. Flow testing of mockup a 37-pin subassembly with 0.250 in. OD cladding and 0.0625 in. wire wrapping was completed. The tests showed slightly higher flow rates than were calculated for positions in rows 1 through 6 of the EBR-II. The flow rates and pressure drops are shown below for 800 °F sodium.

<u>Row</u>	<u>Pressure Drop, psi</u>	<u>Flow, gpm</u>
1	36	124
2	36	124
3	38	112
4	34.5	85
5	34	72
6	40.8	67

Fabrication of components for a mockup subassembly using 0.250 in. OD cladding and 0.040 in. wire wrap have been completed. This subassembly will be flow tested at ANL.

The 25% PuO₂-75% UO₂ fuel columns are 13.5 in. long. Cladding is 0.250 in. OD x 0.016 in. thick. Pins X-1, -3 and -4 contain normal uranium and pins X-6 and -7 contain uranium enriched 45% with ²³⁵U.

f. In-Reactor Thermal Performance of Mixed Oxide Fuels
as a Function of Oxygen-To-Metal Ratio

J. A. Christensen

The objectives of this task are to (1) determine the effect of departure from stoichiometry in the interval $1.94 \leq O/M \leq 2.00$ on the melting heat rating for $(U/25 \text{ wt\% Pu})O_2$ fuels and (2) correlate postirradiation fuel structure and composition gradients with oxygen-to-metal (O/M) ratio.

Irradiated $(U/25 \text{ wt\% Pu})O_{2-x}$ was examined by metallography, electron microprobe and X-ray diffraction. Results show a correlation between certain structural features and oxygen-to-metal ratio. Twelve nominally 0.25-in. diameter fuel pins (three each containing fuel having O/M ratios of 2.00, 1.99, 1.97, and 1.94, respectively) were previously irradiated for short times at heat ratings which caused incipient melting in approximately half of the specimens. The most significant features revealed during postirradiation examination of cross sections from each fuel pin were:

- (1) Small quantities of etch resistant, spherical inclusions ranging in size to 10μ were observed in each fuel pin. These were identified by electron microprobe analysis as stainless steel. This may have been derived from fuel processing vessels or from the cladding, although no fuel-cladding reaction was evident.
- (2) All specimens with O/M ratios < 2.00 contained fine acicular inclusions ($0.1 \mu \times 10 \mu$), the concentration of which increased with increasing fuel temperature and decreasing average fuel O/M. Large quantities of the inclusions (ranging near 100% near the fuel center) were present in the $MO_{1.94}$. In this material, inclusion precipitation was initiated near the 1400°C isotherm where the inclusions appear to have pinned grain boundaries causing a decrease in fuel grain size in this region as

compared with the colder fuel. Near the fuel center, plutonium concentration was inversely proportional to the concentration of the precipitate phase; thus the inclusions were enriched in plutonium and the matrix depleted, both relative to the original homogeneous $\text{MO}_{1.94}$. The inclusions are thought to be an oxygen-depleted phase, probably $\beta\text{-Pu}_2\text{O}_3$, which precipitates when oxygen migrates to the colder fuel regions. Oxygen migration was inferred from the change in chemical etching characteristics of the cooler fuel regions from the rapid attack typical of MO_{2-x} to the slower attack seen for $\text{MO}_{2.00}$.

- (3) Pore morphology was correlated with O/M ratio. Stoichiometric specimens contained only spherical voids in the non-molten regions. Between 1400 and 1900 °C these were uniformly 5 μ in diameter and appeared to be migrating up the thermal gradient. Migrating spheres of this type are predictable from a surface diffusion controlled transport process. As O/M ratio decreased below 2.00, chevron shaped voids (1 μ across and 50 μ wide) replaced the spheres in this temperature interval. These all had the apex pointing toward the fuel center but nothing in the structure indicated that the chevrons were mobile. The results suggest a stoichiometry effect on surface diffusion with lower O/M oxides having lower diffusivities.

Methods for micro-core drilling of irradiated samples were developed. A series of micro-cores taken across the fuel cross-section is required for measurements of radial and axial oxygen and plutonium migration during irradiation. The technique employs diamond core-drills for removing 0.6 mm diam x 5 mm long cores from selected regions in the fuel. The cores are subsequently analyzed for O/M and Pu/U using a microbalance weight change method and mass spectrometer, respectively.

CROSS INDEX

Given below is a cross index to other sections of this report containing data directly related to the fuel development program.

<u>Section</u>	<u>Title</u>
IV-C-2-a	Thermal Hydraulics - Fuel Assembly Heat Transfer Characteristics
IV-C-2-d	Hydraulic Testing
IV-D-1-f	Clad Characterization
IV-D-1-g	Cladding Biaxial Creep
IV-D-2-a	Fabrication Development - Vendor Evaluation
IV-D-2-b	Fuel Duct Fabrication Development
IV-D-2-c	Wear Resistant Coatings
IV-D-2-d	Fuel Duct Creep Evaluation

IV. DEVELOPMENT

D. L. Condotta

HIGHLIGHTS OF ENGINEERING DEVELOPMENT

J. R. Carrell

Based on a state-of-the-art survey it has been concluded that centrifugal, free surface, sump-type pumps are best suited to FFTF service conditions.

Component tests conducted in water verified the operating principle of the shim-scram control rod mechanism concept based on the Fermi design.

Scope drawings of the reactor cover mockup were completed.

A creep compensating, quick-disconnect connector was selected as the reference concept for nozzle closures.

A reference concept was selected for the closed-loop hardware which includes a welded closure for the spool piece.

B. ENGINEERING DEVELOPMENT

J. R. Carrell

1. Sodium Components Development

L. A. Whinery

a. Management Plan

An overall Sodium Components Development Management Plan was formulated which establishes guideline procedures, controls, and near-term schedule commitments to be followed in implementing, managing, and monitoring the development and testing activities for major FFTF sodium system components.

b. Sodium Pumps (Work Performed for System 50. Heat Transport)

W. Babcock

A state-of-technology survey was completed covering existing sodium pumps having features applicable to the FFTF main heat transport system and closed loop systems. Conclusions derived from this survey are that centrifugal, free-surface, sump-type pumps are best suited to FFTF service conditions for both the main and closed loop sodium systems. The following extrapolations require further design studies and proof testing:

- (1) The 440 ft total dynamic head (at 12,000 gpm) dictates an impeller diameter larger than previously used for single-stage pumps for high temperature sodium service, and peripheral velocities will likely exceed technology limits. Impeller disk and hub stresses may be incompatible with material structural properties at the high temperatures involved. Impeller strength-to-weight ratios are quite low, and possibilities exist for damage resulting from creep.

- (2) To achieve adequate net positive suction head (NPSH) and impeller submergence for a "cold leg" pump installation, a shaft length of approximately 30 ft is required. This is roughly twice the length used in existing large sodium pumps.
- (3) Because interfacing limitations are imposed by a maximum reactor cover gas pressure of 15 psia, a maximum pump shaft length of 30 ft, and a "cold leg" system pump location, the primary pump must be especially designed for a NPSH of approximately 25 ft, which is less than that of comparable extant technology.
- (4) Improvements in pump shaft seal designs are required to satisfy FFTF operating life goals.

2. Equipment Development

C. A. Munro

- a. Control Mechanism Development (Work Performed for System 30. Reactor)

J. C. Hoakes and P. K. Telford

A shim-scam control rod mechanism concept based on the Enrico Fermi Nuclear Power Facility design was studied for FTR application. In this concept the drive motor, position readout devices, limit switches, motion translation mechanisms, scram magnet, and seals are located on the reactor vessel cover. The scram release fingers, initial accelerating spring, poison element, and decelerating device are located under the sodium. The drive motor is a Slo-Syn HS 400B-1003 stepping motor, which will provide a maximum force to the poison section of 1000 lb at 9 in./min translation rate. The motor also functions as a static brake and clutch. Position readout sensors are provided by a digital transmitter and a multiturn potentiometer. In addition, limit switches are provided to indicate the limits of drive travel and also to indicate that the poison section is

coupled to the scram fingers. The main seal to prevent sodium vapor from coming in contact with the drive motor assembly, ball lead screw, and scram magnet is a labyrinth backed up by a bellows. A decelerator located below the active core is used to minimize mechanical stresses at the end of the scram cycle. Sodium is forced out of a dash pot attached to the control rod duct by the ram attached to the poison section.

The scram latch fingers were modified from the Fermi control rod design by increasing the size of the latch to reduce stress levels and by altering the angular gripping arrangement to allow self-release in the event of spring failure. Analysis was performed on the dynamics and kinematics of the poison section to determine the drop forces during the scram cycle, and a preliminary spring design was prepared. Component tests conducted in water verified the operating principle.

b. Pneumatic Control Mechanism Concept (Work Performed for System 30. Reactor)

C. A. Munro

In connection with the vertical core task force studies, a pneumatic mechanism for control rods (Figure 4.23) was evaluated. Details are given in Chapter III, Section A, Paragraph 2, pp. 3.45.

c. Reactor Hardware Development (Work Performed for System 30. Reactor)

J. W. Kolb, B. G. Smith, R. Kolowith

Scope drawings of the reactor cover mockup were completed. The cover mockup will consist of a 150° sector of the reactor cover containing 57 nozzles, three of which will be prototypical in design permitting sodium to be picked up with the FHM snorkel simulator.

BNWL-880

FIGURE 4.23. *Pneumatic Control Mechanism Concept*

Conceptual designs of various reactor nozzle closures were prepared, and a creep compensating, quick disconnect connector was selected as the reference design. This type of closure is shown on Figure 4.23a (SK-3-14000) for driver fuel and has been adapted for all other top face nozzle closures, thereby standardizing the maintenance tooling required.

d. Closed Loop Tube and Hardware Development (Work Performed for System 60. Irradiation Testing)

J. W. Kolb

The reference concept for the closed loop was selected as shown on Figure 4.24 (conceptual drawings SK-3-12956, sheet 1), and Figure 4.25 (SK-3-12936). A backup design for the top face hardware (spool piece) was selected as shown on Figure 4.26 (SK-3-14127). The reference concept has a welded closure for the spool piece to ensure a positive sodium seal. Detail drawings were prepared for fabrication of a remote welder and remote cutter for this concept.

SK-3-14000, Sheet 1

FIGURE 4.23a. Driver Fuel Creep Compensating Connector Assembly (Connector Latched Position)

Neg-3-14000, Sheet 2
 FIGURE 4.23b. Driver Fuel Creep Compensating Connector
 Assembly (Connector Release Method)

4.46

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SK-3-12956, Sheet 1

FIGURE 4.24. Closed Loop Subassembly

SK-3-12936

FIGURE 4.25. Closed Loop In-Pile Tube Immersed Section (Study I)

4.48

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SK-3-14127

FIGURE 4.26. Closed Loop Top Face Hardware
(Arrangement "C")

IV. DEVELOPMENT

D. L. Condotta

HIGHLIGHTS OF COOLANT AND PROCESS DEVELOPMENT

B. M. Johnson

In the study of the transport and deposition behavior of radioisotopes formed by irradiation of core components, ^{51}Cr and ^{54}Mn were found to behave quite differently. The ^{51}Cr was deposited by isotopic exchange in the near vicinity of the sample while the ^{54}Mn had a greater tendency to be carried to the cold leg of the loop and to show a deposition rate more sensitive to velocity and temperature.

The reaction of methane and sodium was found to be a second order reaction with an activation energy of about 9 Kcal/g mole. The gas plenum volume appeared to be the controlling variable for the reaction.

Ethylene glycol has shown the most promise as a reagent for sodium removal from piped systems. No good correlation has been found between deadleg diameter and required cleaning time.

Based on a survey, the plugging meter, electrochemical oxygen meter, carbon diffusion meter, and hydrogen diffusion meter were found to be the only in-line sodium analysis instruments with sufficient merit to warrant use in the FTR.

The amalgamation method was selected for determination of oxygen in sodium.

A major interim goal was achieved in the development of electrically heated fuel pin models for use in bundle geometry heat transfer studies. A single pin achieved a heat flux of 500,000 Btu/hr-ft² at a sodium temperature of 800 °F, maintained for six hours. This pin was subsequently found capable of

650,000 Btu/hr-ft² heat flux in 1110 °F sodium before failure occurred as a result of electrical arcing between the sheath and the central core.

The laboratory tests to determine the effect of fuel pin spacing, wire wrap pitch, and pin diameter on the pressure drop-flow behavior of FFTF fuel subassemblies were completed. Preliminary results showed that the friction factors determined for 7-pin, 37-pin, and 217-pin assemblies of similar geometry agreed at corresponding Reynolds Numbers, thus substantiating the validity of hydraulic tests with small partial assemblies.

The feasibility of cooling FFTF fuel subassemblies during inspection by axial flow of gas was demonstrated.

C. COOLANT AND PROCESS DEVELOPMENT

B. M. Johnson

1. Corrosion and Chemistry

D. W. Shannon

a. Radioisotope Transport and Deposition (Work Performed for System 80. Auxiliary Systems)

W. F. Brehm, E. A. Kovacevich

Radioactive species produced by irradiation of the stainless steel cladding in FFTF will be removed from the reactor core by corrosion and transported by the flowing sodium to other parts of the heat removal system. A recent calculation indicates that up to 10,000 Ci of active species can exist in out-of-core piping, and that under certain conditions the radiation levels from deposited species can exceed 100 R/hr. The principal species are ^{51}Cr , ^{58}Co , ^{60}Co , and ^{54}Mn . While a significant amount of information has been gathered on the factors affecting the corrosion and deposition process, there is no reliable method to predict the magnitude and location of deposited radioactive material from flowing sodium in stainless steel piping, as a function of temperature, flow rate, sodium purity, and system geometry. The PNL effort in this area is, therefore, concentrated on observing the migration of radioactive species released from irradiated source tabs in flowing sodium. A pumped loop, known as the radioisotope transport loop (RTL) and small thermal convection loops (TCL) are used for these studies.

(1) Radioisotope Transport Loop Operation. The radioisotope transport loop in which deposition studies are being conducted is a closed loop constructed from Type 304 SS tubing. The design of the loop is based upon the reference design closed

loop as stated in BNWL-CC-400. The details of the loop design have been reported previously.⁽¹⁾ The loop consists of a primary loop and a secondary purification loop. The primary loop contains the various test sections in which the desired sodium velocities are achieved and a provision for inserting a radioactive test specimen. The secondary system bypasses part of the primary loop sodium flow to provide a means of purifying the sodium during loop operation by either cold trapping or hot trapping. A plugging meter is also incorporated into the secondary system to determine the oxygen content of the sodium during operation.

In addition to the test loop and its associated plumbing, a loop enclosure consisting of a Plexiglass panel with glove parts is employed to provide an inert atmosphere during the elevated temperature loop operation. A nitrogen cover gas containing an oxygen content of less than 4 vol% and a moisture content of less than 4500 ppm is maintained during loop operation. The nitrogen cover gas pressure is maintained at slightly subatmospheric (-0.25 in. of water) pressure to ensure complete control of radioactive products in the event of a loop failure and a subsequent sodium leak. Instrumentation for measuring the oxygen, moisture and hydrogen (which would result from a sodium leak) content in the nitrogen cover gas is incorporated into the alarm system.

The first experiment employed a Type 304 SS specimen irradiated in a thermal reactor. The specimen, which measured 92 R/hr.

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(1) D. W. Shannon. Fast Reactor Corrosion and Chemistry, Interim Technical Progress Report No. 1, October 1966 - April 1967, BNWL-483, pp. 5.1-5.10. Pacific Northwest Laboratory, Richland, Washington, July 1967.

at the surface, was loaded and sealed into the loop system under inert gas conditions provided by the loop enclosure. Lead shielding was installed around the irradiated specimen area to reduce the radiation levels for the operating personnel. The loop was operated for 468 hr under the following conditions:

Maximum Temperature, °F	1300
System Thermal Gradient, °F	500
Primary Sodium-Flow, gal/min	1.7
Oxygen Content, ppm	<10

After loop shutdown, the radioactive specimen was removed and the loop was scanned with a sodium iodide crystal detector. The gamma survey readout was accomplished by a 4096 multi-channel analyzer.

Following the gamma scan, the radioactive test source was replaced and loop operation was continued at the above test conditions until a total of 1212 hours was accumulated. A second gamma scan was then made.

The second series of tests involved a Type 304 SS specimen which had been irradiated in the EBR-II. A velocity section was fabricated and inserted into the loop which permitted determination of velocity effects on deposition rates in an isothermal condition.

The specimen was inserted into the loop and the loop was operated under the following conditions:

Maximum Temperature (Test specimen), °F	1300
System Thermal Gradient, °F	400
Primary Sodium Flow, gal/min	1.7
Oxygen Content, ppm	<10

The maximum system thermal gradient obtainable with the existing water-cooled radiator was 400 °F and not the desired 500 °F. As before, the radioactive source was removed from the

loop after 351 hr of operation and the loop was scanned for radioactivity. Following the completion of the gamma scan, the loop was restarted with a nonradioactive Type 304 SS specimen instead of the EBR-II irradiated specimen. Loop operating parameters were as stated above.

(2) Experimental Results. The first experimental results were obtained with stainless steel specimens irradiated in a thermal flux, so that the primary active species were ^{51}Cr , ^{59}Fe , and ^{60}Co . After exposure to flowing sodium at 1300 °F in both the pumped RTL and the TCL, only ^{51}Cr was found deposited on the pipe walls to any extent, and only for short distances downstream. This can be seen in Figures 4.27, 4.28 and 4.29. Virtually no activity was found in the heat exchanger or cold trap of the radioisotope transport loop, nor was ^{51}Cr transported to the cold leg of the thermal convection loop test. Note the preferential deposition of ^{51}Cr at the two top corners of the thermal convection loop in Figure 4.29. Analysis of the sodium taken from the thermal convection loops showed ^{51}Cr , ^{60}Co , ^{110}Ag , ^{124}Sb , and ^{54}Mn . The latter three isotopes were not present in significant amounts in the irradiated tab, and therefore must be trace impurities preferentially leached by the sodium. Only trace amounts of ^{59}Fe were seen on the tubing. No radioactivity was found in sodium samples taken from the radioisotope transport loop; however, this loop contains a very large volume of sodium compared with the convection loops.

Known amounts of material were removed from the inside of a piece of tubing from one of the thermal convection loops by electropolishing, and the polishing solutions were counted for ^{51}Cr . At a depth of 0.5 mil into the tubing, the activity level was about one-fourth that of the surface activity. A plot of the logarithm of activity versus square of the distance into wall yielded a straight line. From the slope of this line, a diffusion coefficient of about $10^{-11} \text{ cm}^2/\text{sec}$ was estimated for chromium in stainless steel at 1300 °F.

FIGURE 4.27. Relationship of Radioactive Scan of the Radioisotope Transport Loop with Fluid Velocity, Fluid Temperature, and Distance from Radioactive Source

FIGURE 4.28. Count Rates of ^{51}Cr in Thermal Convection Loop, (counts/min)

FIGURE 4.29. Thermal Convection Loop Temperature and Counting Rate Distribution of the No. 2 Convection Loop

Similar activity-distribution tests were run in the radio-isotope transport loop with a specimen irradiated in EBR-II, in which the principal species formed are ^{54}Mn and ^{58}Co . Under similar conditions as in the previous experiment, the deposition pattern was vastly different, as large amounts of activity were found in the downstream exit of the heat exchanger and in the cold trap, as well as immediately downstream from the source. The distribution of activity around the loop is shown in Figure 4.30. The counting rates are much higher with the manganese source, even allowing for the difference in specimen specific activity.

A section of the RTL is constructed in such a manner as to induce step changes in velocity within this section. This region is isothermal, and is downstream from the region near the source. Changes in deposition rate within this section are shown in Figure 4.31. It is apparent that more activity per unit area is deposited in the higher velocity sections. The counts-per-minute data obtained can be converted into microcuries per square inch by dividing by 2.2.

Experiments in which a loop containing deposited ^{51}Cr is run without a source tab to observe activity migration are in progress. Preliminary information indicated that ^{51}Cr tended to be transported from its original hot-leg deposition sites toward lower temperature regions of the loop. This finding is in accord with behavior of Cr in tests at General Electric Advanced Products Operation.⁽¹⁾

(3) Transport of ^{51}Cr . Figures 4.27, 4.28, and 4.29 show the ^{51}Cr is moved primarily by isotopic exchange, which dominates

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(1) G. P. Wozadlo et al. "Mass-Transfer Studies and Related Metallurgical Effects for Austenitic Stainless Steel in High-Temperature Pumped Sodium Systems," paper presented at 1968 AEC Corrosion Symposium, Columbus, Ohio. May 1958.

the gross material removal (corrosion) process. The apparent limiting step in the material deposition process is collision with the downstream pipe wall. The increased deposition at corners where the flow is more turbulent reinforces this suggestion. The release part of the exchange process is controlled by the usual factors: either surface reaction or liquid boundary layer diffusion. These, in turn, should be functions of temperature and flow conditions. Similar exchange processes have been reported in sodium-stainless steel loop tests.^(1,2) The activity migration observed in the loop test run without specimen indicates that the system will slowly approach classical "equilibrium" (corroded active material finally deposited in the cold leg) but that the transport process is the slow isotopic exchange and deposition, rather than rapid solution, convective-transport, and deposition.

(4) Transport of ^{54}Mn . Figure 4.30 shows that the transport of ^{54}Mn is dependent on other parameters besides simple isotopic exchange. The fact that an extensive amount of activity remains near the specimen shows that the exchange process is still important, even though it is not as dominant as in the case of ^{51}Cr transport.

The presence of activity on the downstream side of the heat exchanger suggests a classical solution-and-precipitation process, with perhaps a slight amount of supersaturation. In the cold trap, this reaction is possible, assisted by the chemical reaction:

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- (1) F. G. Haag. *Nucleonics*, vol. 15, p. 58. 1957
 (2) R. W. Lockhart. *Sodium Mass Transfer II, Screening Test Data and Analysis, Vol. 1. Mass Transfer Results*, GEAP-3726. General Electric Company, Atomic Products Division, Schenectady, N. Y., April 1962.

A phenomenon of this sort seems feasible in light of the fact that the cold trap, with about one-eighth of the flow, has at least twice as high a counting rate as any other spot in the loop; and that nowhere else in the loop are substantial amounts of oxygen present.

The increased deposition rate of ^{54}Mn with increasing velocity, as shown in Figure 4.31, shows that the deposition mechanism for ^{54}Mn , when there is no temperature dependent process operating, is dependent upon fluid boundary-layer thickness (higher velocity—smaller boundary layer—less resistance to deposition). A similar increase in deposition occurs at the outlet of the heat exchanger where the velocity increases. This phenomenon is somewhat analogous to the ^{51}Cr isotopic exchange, where the increased turbulence at pipe bends led to higher deposition rates.

The higher counting rate obtained with ^{54}Mn sources, as well as the previously mentioned presence of ^{54}Mn in sodium taken from a test loop where only a trace amount was present in the source, shows that manganese is more preferentially leached from stainless steel than is chromium. The absence of iron and the small amount of cobalt detected indicate that these elements are not preferentially removed by the sodium.

These results are a start toward making reliable predictions about the location and magnitude of radiation levels in FFTF piping.

- b. Behavior of Fission Products in Sodium (Work Performed for System 60. Irradiation Testing)
P. L. Koehmstedt

Cladding failures within FFTF closed loops are expected to release significant amounts of radioisotopes and possibly fuel. Releases of fission products into sodium in earlier sodium reactors have shown extensive deposition of these products onto

pipng surfaces. Soluble and particulate fuel is expected to deposit or settle in unwanted regions of FFTF closed loops. In addition to potential plugging problems, fuel particulates must be removed to minimize radiation for personnel entry and to enable fuel failure detectors to function with useful sensitivity involving subsequent tests.

A fission product control screening loop which will allow the introduction of fission products into sodium of predetermined purity was designed and fabrication initiated. Capability to vary sodium velocity, pressure, temperature, and impurity introduction were included. One modular section enables the bypass testing of three trapping configurations, whereas another section allows bypass introduction of two sources of fission products or impurities. The loop is built in a single plane to facilitate counting. A flow diagram is shown on Figure 4.32 (H-3-27140).

c. Sodium Cover Gas Interactions (Work Performed for
System 80. Auxiliary Systems)
H. P. Maffei

Experimental studies on the reactions of cover gases and their impurities with molten sodium are being conducted to establish allowable impurity levels for FFTF cover gases. This will be accomplished by determining the equilibrium values and kinetic reaction rates between the cover gas impurities and molten sodium. The reaction rates of methane in helium with molten sodium have been studied to date. The effect of sodium volume and surface area on the reaction rate has also been investigated. The experimental apparatus is shown in Figure 4.33.

The reaction of methane with molten sodium was measured at 800, 1000, 1200, and 1400 °F. All of the test capsules contained 100 cm³ of sodium at the particular test temperatures.

Neg 0673846-1
FIGURE 4.33.

Experimental Equipment for Measuring Reaction Rates of Cover Gas Impurities with Sodium

The sodium surface for each test was 2.47 in.². Two liters (760 mm pressure) of cover gas containing nominally 100 ppm of methane were recirculated over the sodium and the resultant decrease in methane concentration was measured by a gas chromatograph.

The reaction of methane with sodium proceeds most probably as follows

The reaction, being bimolecular, suggests a second order kinetic reaction rate which may be expressed as:

$$\frac{dx}{dt} = -k'x^2 \quad (1)$$

where x is equal to the methane concentration at time t. If this equation is then integrated from zero time when x = a, the original methane concentration, to any time t, the following expression is derived:

$$\frac{a - x}{x} = ak't \quad (2)$$

a resulting straight line plot of $\frac{a - x}{a}$ versus t verifies a second order reaction for the process.

Figure 4.34 is a plot of the data at various temperatures. Typical plots of Equation 2 are shown in Figures 4.35 and 4.36 and verify that the reaction of methane and sodium is a second order reaction. The measured rate constants for the four test temperatures are:

Temperature	k'
800 °F	3.45×10^{-4}
100 °F	4.83×10^{-3}
1200 °F	8.1×10^{-3}
1400 °F	5.0×10^{-2}

These results indicate an activation energy of about 9.1 kcal/g mole.

FIGURE 4.34. Reaction of Methane With Sodium at Various Temperatures

FIGURE 4.35. Reaction of 100 ppm Methane Over Sodium at 800 °F

FIGURE 4.36. Reaction of 95 ppm Methane Over Sodium at 1200 °F

The establishment of methane impurity limits for FFTF cover gas requires not only the determination of methane-sodium reaction rates but also the determination of the controlling variable(s) for the reaction and the behavior of the carbon added to the sodium. Various investigations have been performed at 1200 °F to establish the reaction variables. The parameters investigated included sodium volumes, sodium surface areas, gas plenum volumes, gas plenum surface areas, and the disposition of the sodium carbide formed from the reaction of methane and sodium.

It was found that the gas plenum volume of the test capsules appeared to be the controlling variable for the reaction of methane with sodium. The reaction rate is probably controlled by the amount of sodium vapor present and not the amount or geometry of liquid sodium. This is shown in Table 4.VII. It can be seen that the reaction rates increase with an increasing volume of sodium vapor in the gas plenum whereas there is no apparent correlation between increasing reaction rates and total liquid sodium volumes or surface areas.

TABLE 4.VII. Reaction Rates of Methane and Sodium at 1200 °F

<u>Gas Plenum Volume, moles of Na vapor</u>	<u>Liquid Na Surface Area, in.²</u>	<u>Total Volume Na in Capsule, cm³</u>	<u>Reaction Rate, moles CH₄/hr</u>
4.8 x 10 ⁻⁴	2.2	500	9.3 x 10 ⁻⁴
6.8 x 10 ⁻⁴	2.2	250	2.2 x 10 ⁻³
7.3 x 10 ⁻⁴	1.5	100	1.3 x 10 ⁻³
7.7 x 10 ⁻⁴	0.68	100	7.5 x 10 ⁻³
7.9 x 10 ⁻⁴	2.2	100	6.1 x 10 ⁻³
3.5 x 10 ⁻³	11.6	200	3.9 x 10 ⁻²
3.5 x 10 ⁻³	11.6	200	4.8 x 10 ⁻²

If the above hypothesis was correct, the sodium carbide formed during the reaction of methane and sodium should be found in higher concentrations in the condensed sodium on the capsule walls and probably on the surface of the sodium pool. This point was investigated by reacting 10 mg of methane with 100 g of sodium in a capsule heated to 1200 °F. Sodium samples were taken from different levels of the sodium pool and from the condensed sodium on the capsule wall for carbon analyses. A longitudinal strip of the capsule wall was also sectioned and analyzed for carbon. It was found that the condensed sodium from the capsule walls contained 370 ppm of carbon, whereas the top portion of the sodium contained 45 ppm of carbon as compared to 0 to 20 ppm for the remainder of the sodium pool. There was no apparent carburization of the 304 SS capsule walls, since the carbon analysis on the stainless steel ranged from 578 to 620 ppm as compared to 585 ppm for a standard sample. These results confirmed the conclusion that the reaction occurs in the gas phase and hence the gas plenum volume was the controlling variable.

d. Selection and Testing of Cover Gas Instruments and Sampling Systems

G. B. Barton

The objectives of this activity are to select and proof test gas analysis instrument systems, define design criteria for associated equipment, select an instrument type, identify important specifications, and verify that the sampling and analysis system selected will work satisfactorily in the FFTF. The various applications of inert gases for blanketing the sodium and components of the FFTF system have been examined and the probable range of impurities estimated. This will provide a basis for selecting instruments for evaluation.

(1) Oxygen Monitors (This Work Done for System 90. Instrumentation and Control). The measurement of trace oxygen in the inert atmosphere of glove boxes has provided an opportunity to evaluate several oxygen monitors of two major types including: (a) those utilizing the Hersch electrochemical cell of a silver cathode and a lead or cadmium anode in an alkaline electrolyte, and (b) those using a calcia-stabilized zirconia electrolyte at high temperatures. Although no decision has been made as to the preferred instrument for FFTF application, it has become apparent that the calcia-stabilized zirconia instrument will measure only the oxygen in excess of that which will combine with reducing agents at the 750 to 850 °C operating temperature. The high temperature zirconia-calcia cell instrument may give misleading results for trace oxygen measurements (particularly where hydrogen or hydrocarbons could be nondeleterious components of the gas sample).

e. Evaluation of FFTF Sodium Cleaning Requirements
(This Work Done for System 80. Auxiliary Systems)
G. R. Bloom

The objective of this study is to evaluate and select sodium removal processes for the FFTF fuel assemblies, components, and piping systems. Sodium loop pipe systems must have all sodium removed prior to decontamination with aqueous solutions including complete removal of the sodium from all crevices, deadlegs, and components in the pipe system. Candidate reagents for sodium removal from pipe systems are steam, moist gas, alcohols, and ammonia.

Ethylene glycol has shown the most promise of the alcohols tested to date as a reagent to remove sodium from piped systems. The alcohols tested include ethanol, methanol, batanol, and Dowanol EB. Principal advantages of using ethylene glycol are: (1) a very slow, controlled reaction with sodium at room

temperature, (2) increased reaction rate which can be controlled between the temperatures of 35 and 50 °C, and (3) a relatively high flash point of 115 °C resulting in a lower fire hazard.

Parameters in the deadleg cleaning process which have been investigated include: the type of cleaning agent, length of deadleg, deadleg diameter, and time using a nonflow system.

Test results to date have shown no good correlation between deadleg diameter and required cleaning time. Experimental error and variations caused by other parameters, such as temperature fluctuations, probably masked the effect of deadleg diameter. Thus a 1/6 in. diam deadleg proved about as easy to clean as a 3/4 in. diam deadleg.

The sodium-filled glass deadlegs were prepared and then cleaned with ethylene glycol and Dowanol EB at 25 °C in a non-flow system. The ethylene glycol-to-sodium volume ratio was 100:1, and the Dowanol EB-to-sodium ratio was 70:1. The ethylene glycol and Dowanol EB had removed sodium from the various deadlegs as shown below after the periods shown.

Removal of Sodium from Deadlegs with Alcohols at 25 ± 3 °C

<u>Deadleg Diameter,</u> <u>in.</u>	<u>Ethylene Glycol</u> <u>Cleaned Length</u> <u>After 13 Days, in.</u>	<u>Dowanol EB</u> <u>Cleaned Length</u> <u>After 7 Days, in.</u>
1/6	10.5	6.4
1/4	7.2	7.0
3/8	8.5	2.3
1/2	5.75	12.0
3/4	6.8	10.5

The test results infer that a 6 in. sodium deadleg with diameters between 1/6 and 3/4 in. can be cleaned with ethylene glycol in 13 days at 25 °C. The Dowanol EB cleaning rate for all deadlegs remained constant during the first three days.

The cleaning rate for the 1/6, 1/4, and 3/8 in. diam then slowed and had essentially stopped at seven days. The cleaning rate for the 1/2 and 3/4 in. diam deadlegs stayed fairly constant throughout the entire test. Conclusions are that the pure Dowanol EB would not be a good sodium removal agent for sodium deadlegs which are both under 3/8 in. in diameter and more than 2 in. long. However, 95% Dowanol-5% water solution will remove sodium from a 1/4 in. diam deadleg in 1 1/2 hr. The pure Dowanol could be used first to clean the pipe walls and then 95% Dowanol-5% water solution used to clean the sodium from the deadlegs and crevices.

A small stainless steel loop with two deadlegs was assembled to evaluate effects of flow and temperature on the deadleg cleaning process. One deadleg is positioned vertically with the open end up and another deadleg is positioned horizontally. The vertical deadleg is placed in a constant temperature bath of ethylene glycol during the 50 and 75 °C tests. The horizontal deadleg is heated with an insulated nichrome heater. The first set of tests was conducted at different temperatures and a constant flow rate. The temperature had a large effect on the reaction rate as would be expected. The relative cleaning rates at 35, 50, and 75 °C for 1/4 in. diam vertical deadlegs were 1.0, 2.11, and 434 as shown in Table 4.VIII.

TABLE 4.VIII. Ethylene Glycol Pumped Loop Sodium Deadleg Cleaning Tests

Deadleg Mouth Position	Temperature, °C	Ethylene Glycol Flow Rate	N _{Re} at Deadleg Mouth	Deadleg Cleaned Length, in.	Cleaning Time	Inch/Day Cleaning Rate	Relative Cleaning Rate
Vertical up	35 ± 3	1.0	670	5.4	4 days	1.35	1
Vertical up	50 ± 3	1.0	1000	11.4	4 days	2.85	2.11
Vertical up	75 ± 5	1.0	2300	7.3	18 min	585	434
Horizontal	35 ± 3	1.0	670	3.5	4 days	0.875	0.65
Horizontal	50 ± 3	1.0	1000	4	4 days	1.0	0.74

f. Selection and Testing of In-Line Sodium Analysis

Instruments (This Work Done for System 90.
Instrumentation and Control)

J. J. McCown

The objectives of this task are to: (1) identify FFTF in-line instrument requirements; (2) review offsite instrument development and evaluation programs to assess their applicability to FFTF needs, and recommend redirection of these programs where necessary; and (3) coordinate the selection, proof testing, procurement, and pre-startup testing of the FFTF in-line instruments. The scope of this work is limited to those instruments which measure a chemical contaminant (carbon, hydrogen, oxygen) present in the sodium.

FFTF Instrumentation Studies. The status of the various instrument development programs being performed under LMFBR sponsorship was discussed in detail in BNWL-562.⁽¹⁾ This survey concluded that the only instruments having sufficient merit to warrant use in the FTR are the plugging meter, electrochemical oxygen meter, carbon diffusion meter, and hydrogen diffusion meter. Of these, only the plugging meter, a general purity type instrument, has been sufficiently tested that it could be used in all major FTR systems without further change. The other three instruments are still in various stages of development and/or field testing and will require improvements prior to use in the FTR.

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(1) T. F. Demmitt. FFTF In-Line Instrumentation of Sodium Purity Measurement: Status Report, BNWL-562. Pacific Northwest Laboratory, Richland, Washington. September 1967.

g. Selection and Testing of FFTF Sodium Sampling Methods

(This work done for System 80. Auxiliary Systems)

V. P. Kelly

The objectives of this activity are to: (1) select methods of sampling radioactive sodium, (2) build and proof-test prototype FFTF samplers, (3) provide the design criteria and specifications for the FFTF sampler, and (4) demonstrate that the sampler is adequate to take repetitive samples of radioactive sodium and capable of being remotely maintained and operated by reactor technicians.

(1) By-Pass Loop Sampler. The by-pass loop sampler must produce a representative sample and must be remotely operable. In addition, the following requirements have been established.

- (a) The sampling technique must not affect the impurity level in the sodium within the limits of the analytical method.
- (b) The system must provide 1 sample/hr to an analytical laboratory repetitively and reliably, with a normal conditioning time of 4 hr.
- (c) The sample loop must be a horizontal length of pipe or tubing which can be readily crimped to close off the ends and which can be sectioned in the analytical facility. A minimum sample of 30 g must be provided.
- (d) A means of cooling the sample tube rapidly and uniformly is required to prevent segregation and transfer of impurities. A 3 min cooling time for a 1/2 in. tube has been recommended.
- (e) The loop must be heated and insulated to maintain the flowing sodium at the sampling point temperature (1400 °F maximum) and without a change in temperature between the sampling point and the point of discharge from the removable sample section.

- (f) The by-pass loop must be designed so that in-line sample systems, distillation samplers, or other sample systems can be readily connected into the loop remotely.

The arrangement of the by-pass loop sampler designed to meet these criteria is shown in Figure 4.37. The sampler has two removable-section hairpin sample tubes which are connected in parallel in the primary sodium sampling circulating system. The primary sampling system is in turn connected to the reactor loop so that sodium flows from the main loop, and through the removable-section sample tubes returning to some point downstream from the sample point. Each one of the removable-section sample loops consists of a flowmeter, inlet and outlet block valves, hairpin sample tube, air quench header, and mechanical connectors. Types of connectors which will be installed on the loop for evaluation include compression tube fittings, Marman conoseals®, metal O-rings, and metal K-seals.

(2) Freeze-Seal Sampler. The freeze-seal sampler, Figure 4.38, must meet the same general design criteria as the by-pass loop sampler with respect to remote handling capability and the capability of yielding representative samples without introducing significant amounts of contamination. The freeze-seal sampler is envisioned as functioning in a main loop or by-pass loop and with sodium flowing either perpendicularly or parallel with the sample carrier. The latter case is represented in Figure 4.38. As shown, the sampler consists of a U-bend in the sodium piping. The bottom of the "U" is the sample chamber. Eight-inch long sample tubes are loaded into a magazine on a tee-branch off the sample chamber and are pushed into the chamber along with dummy rods. Rods and tubes are pushed in sequence through inlet and outlet freeze ports. Ports are 16 in. long, based on a length of 1 in./100 °F required to freeze the sodium from the design operating temperature of 1400 °F. Ports have

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FIGURE 4.37. Removable-Section By-Bass Loop Sodium Sampler

FIGURE 4.38. Parallel-Flow Freeze-Seal Sampler

axial external fins and shrouds for forced-convection air cooling. When the sampler is loaded, two dummy 1/2 in. diam rods occupy each one of the ports and a single 1/2-in. diam rod occupies each one of the ports and a single 1/2-in. diam sample tube occupies the sample changer. A repeated feed cylinder operation pushes the end rods and sample tubes into the receiving chamber, and at the same time pushes in new tubes and rods. The sampler is designed to operate continuously without interrupting the sodium flow.

h. Selection and Testing of Radioactive Sodium Analyses Methods (This Work Done for System 80. Auxiliary Systems)

F. A. Scott, J. F. Jarosch, G. B. Barton

(1) Objectives. The objectives of this activity are to:

- (a) Establish PNL capability for the chemical analyses of oxygen, hydrogen, total carbon, nitride nitrogen, corrosion products, fission products, uranium and plutonium in sodium, both in the presence and absence of plutonium contamination.
- (b) Provide proven analytical techniques and equipment for the analyses of highly radioactive sodium in the FFTF.
- (c) Provide design criteria and requirements for the chemical analyses laboratory of the FFTF--notably the hot cell facility.

(2) High Level Analytical Facility Design. The results of a design study of the chemical analysis laboratory of the FFTF were completed and published.⁽¹⁾ This document describes a design concept for the laboratory and outlines the initial

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(1) F. A. Scott, J. S. Jarosch, and G. B. Barton. Fast Flux Test Facility Chemical Analysis Laboratory, Preliminary Design Study, BNWL-821. Pacific Northwest Laboratory, Richland, Washington. August 1968.

methods and equipment to be used. Plans to design and construct mock-up facilities for remote testing of equipment and analytical methods were made.

(3) Sodium Studies Facility Analytical Laboratory. Laboratory facilities were established for the analysis of nonradioactive sodium metal. Equipment and methods were installed for the determination of oxygen by amalgamation, hydrogen by amalgamation, and metallic impurities by emission spectrography.

(a) Oxygen Determination (L. F. Lust, E. L. Osborne, R. L. Patteson). Procedures in use at other laboratories throughout the USA were reviewed, and the amalgamation method was selected as the one best suited to PNL needs.⁽¹⁾

The method described by Shannon was modified in three details:

- The sample size was reduced from 5 to 2 g.
- The amount of mercury used for the amalgamation was reduced from 100 to 50 ml.
- The size of the amalgamation vessel was reduced to accommodate the smaller amount of amalgam.

Equipment including a recirculating argon atmosphere glove box necessary to analyze sodium metal by the amalgamation method was installed. An indication of precision of customer samples run in duplicate is given in the following tabulation.

Sample	Oxygen Analyses		Sample	Oxygen Analyses	
	Analyses, ppm	Range, ppm		Analyses, ppm	Range, ppm
A	29, 31	2	F	3, 4	1
B	6, 5	1	G	6, 8	2
C	58, 55	3	H	5, 6	1
D	2, 3	1	I	2, 3	1
E	1, 3	2	J	1, 3	2
			K	3, 4	1

(1) D. W. Shannon. Fast Reactor Corrosion and Chemistry Interim Technical Progress Report No. 1, October 1966-April 1967, BNWL-483. Pacific Northwest Laboratory, Richland, Washington. June 1967.

An experiment was performed to determine if there was a detectable linear segregation in a sodium sample contained in a 1/2-in. OD stainless steel tube. Three initial samples were used, each divided into six samples approximately as illustrated (the shaded sections were discarded).

The results of oxygen determination of these samples is shown in the following tabulations.

Sample Tube	Oxygen Analysis, ppm					
	A	B	C	D	E	F
1	10	11	10	10	10	12
2	29	28	15	15	14	16
3	18	17	18	15	16	16

These data showed that the precision of the method was well within 3 ppm and that there was no significant linear segregation. The two high values in Sample 2 were believed to have resulted from contamination.

(b) Determination of Metal Impurities (R. Koy, P. Anderson). A method was developed for the determination of trace amounts of metals in sodium. Methanol was used to dissolve the sample, water was added, and the solution acidified with hydrochloric acid. The excess methanol was removed by evaporation. The metals were removed from the solution by carrier precipitation with lanthanum hydroxide and were measured by emission spectrography. By this method it was possible to determine 31 elements semiquantitatively (within a factor of 2) and five elements (Fe, Cr, Ni, Co, Mn) quantitatively ($\pm 13\%$).

(c) Hydrogen Determination (R. F. Keough). An adaptation of Meacham and Hill's⁽¹⁾ amalgamation method for the determination of hydrogen in sodium was nearly completed. Significant modifications in the apparatus included an attachment of the system to an inert atmosphere glove box to permit direct introduction of the sample into the reflux vessel, and elimination of the diffusion pump.

Hydrogen from hydride and dissolved hydrogen was released on refluxing the amalgam at 200 °C under vacuum. A second refluxing at 370 °C and atmospheric pressure released hydroxide hydrogen. Actual hydrogen measurements were made by injecting aliquots of the reflux vessel atmosphere into a gas chromatograph. A detection limit of 0.2 ppm was obtained.

A protracted single refluxing at each of the two temperatures rather than the brief, multiple refluxes used by Meacham and Hill reduced operator time. Sodium hydroxide tagged with ^{22}Na was used for hydroxide standardization. Sample size could then be determined by correlating the ^{22}Na activity of the refluxed amalgam with the known specific activity of the standard material. Since a similar technique could not readily be used for hydride, sodium hydride was blended with potassium chloride powder. Either potassium or chloride analysis was then used to establish sample size. Recovery of hydrogen from standards is shown in the following tabulation.

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(1) S. A. Meacham and E. F. Hill. The Determination of Hydrogen in Sodium Metal, APDA-183. Atomic Power Development Associates, Inc. Detroit, Michigan. June 1966.

<u>Form</u>	<u>μg Taken</u>	<u>μg Found</u>	<u>Recovery, %</u>
Hydroxide	147	144	98
	250	205	82
	81.2	84	103
	19.6	18.8	96
Hydride	84	81.5	96
	68	69.5	102

Frequently the stainless steel tubing used for sampling was found to evolve more hydrogen than the sodium it contained (Figure 4.39). An extruder, to remove the sodium from the tubing prior to the analysis, was designed.

FIGURE 4.39. Evolution of Hydrogen from Stainless Steel Tubing on Refluxing with Mercury

2. Thermal Hydraulics

J. M. Yatabe

a. Heat Transfer Characteristics of FFTF Fuel Assemblies

(Work Done for Driver Fuel Development Program)

V. R. Hill

The major purpose of this laboratory program is to determine the heat transfer characteristics of FFTF fuel assemblies in direct support of the design of driver fuel assemblies and for the FTR safeguards analyses. To perform this program, electrically heated and instrumented models of these fuel assemblies must be developed. In local cladding temperature tests, heat fluxes of $500,000 \text{ Btu/hr-ft}^2$ at 800°F coolant temperature will be required, whereas for thermal-hydraulic design verification tests, heat fluxes as high as $2,000,000 \text{ Btu/hr-ft}^2$ at 1500°F will be necessary.

Design evaluation and verification tests of single pins were performed in water and sodium. In the water tests a major program milestone was completed when a fuel pin model operated for six hours at a heat flux of $1,500,000 \text{ Btu/hr-ft}^2$ in flowing water with an exit temperature of 316°F . On the completion of this verification test, a further test was performed to demonstrate the maximum capabilities of this pin. Power was increased progressively until a failure occurred at a heat flux of $2,020,000 \text{ Btu/hr-ft}^2$. The test pin operated at heat fluxes at or in excess of $1,500,000 \text{ Btu/hr-ft}^2$ for more than 7 hr. The failure point occurred 4 in. from the upstream end of the rod with gross visual evidence indicating cladding failure caused by local overheating.

Two electrically heated pins were tested in the Small Heat Transfer Loop, which provides a heated sodium environment for the performance of pin verification tests. The first pin consisted of a $1/8$ in. diam tantalum heater element, and $1/8$ in.

diam molybdenum electrical connections, insulated with vipacked boron nitride. This pin was tested by slowly increasing sodium temperature concurrently with increasing heat flux to minimize differential expansion between sheath and core. A heat flux of 522,000 Btu/hr-ft² with a sodium outlet temperature of 800 °F was successfully achieved. However, the pin could not maintain this flux, and the tantalum core melted 6 in. from the inlet end with no rupture of the sheath. The analysis of test data and the failed assembly indicates that the boron nitride was not compacted to a sufficiently high density, and the overall radial thermal resistance was excessive.

The second pin evaluated employed a 1/8 in. diam molybdenum heater element and electrode ends. The boron nitride was compacted to a higher density than was the previous pin. In the evaluation test, a heat flux of 500,000 Btu/hr-ft² with a sodium outlet temperature of 800 °F was maintained for 6 hr. During the second hour, serious fluctuations in building voltage occurred for 45 min, which resulted in repeated heat flux fluctuations of 63,000 Btu/hr-ft². Although such fluctuations are considered detrimental to pin life, the pin survived and continued to operate satisfactorily for the remainder of the test.

After this first test, sodium temperature was progressively increased, to investigate the ultimate capabilities of this fabrication technique. At 860 °F, it was necessary to reduce temporarily the heat flux to 188,000 Btu/hr-ft². Sodium temperature was increased to 1000 °F with pin heat flux of 545,000 Btu/hr-ft² and maintained for 1 hr. As the heat flux was increased further to 650,000 Btu/hr-ft² at 1110 °F a failure occurred 10 in. from the inlet end of the pin. Subsequent examination of the pin revealed that failure occurred because of arcing between the sheath and the core. The performance of the pin was judged to be satisfactory considering the severe thermal shock experienced during testing.

b. Hydraulic Testing of FFTF Fuel Assembly and Components
T. C. Reihman

The overall objective of this program is to provide data concerning flow, pressure drop, and flow mixing required for the thermal hydraulic design evaluation of FFTF fuel assemblies and cooling systems, and in support of the establishment of operating limits. This work is currently divided into three subtasks which include:

- Hydraulic Testing
- Parametric Testing
- Instrumentation and Analytical Development for Flow Mixing Tests.

The purpose of the hydraulic tests is to provide pressure drop-flow data for candidate FFTF fuel assembly designs. These tests are being performed in the water flow facilities located in the PNL Thermal Hydraulics Laboratory.

The purpose of the parametric tests is to extend the range of these data to future or alternative fuel assembly configurations by determining the effect of changes in the independent parameters of pin spacing, spacer pitch, and pin diameter on fuel assembly pressure-drop flow behavior over the following ranges:

Pin Diameter	0.196 to 0.300 in.
Wire Wrap Size	0.0160 to 0.0792 in.
Wire-Wrap Pitch	3 to 24 in.
Length of Pin	36 in.

Several 37-pin bundles were tested. In addition, a 19-pin bundle and a 217-pin bundle were tested to determine how adequately the hydraulic behavior of the complete fuel assembly (217-pin bundle) was represented by the 37-pin data, and to determine the proper extrapolation of the parametric data if it should prove necessary.

Each test section was run for a range of water flow rates representing equivalent sodium flow rates from 10 to 200% of conceptual reactor design flow. The tests were performed with both room temperature and with 190 °F water to maintain dynamic similarity while remaining within apparatus limitations of the hydraulic flow loop. The preliminary results of these tests are as follows:

- The experimentally determined friction factors at the Reynolds Number corresponding to conceptual reactor design conditions were found to be within 3.5% of each other for 7-pin, 37-pin, and 217-pin fuel assemblies of otherwise similar geometry.
- The 19-pin data tentatively appeared to be about 6% above that for the other three bundle sizes. A recheck of pertinent geometric variables for this latter test section is in progress. The good correspondence in this friction factor data for these four bundle sizes leads to a tentative conclusion that the bundle size effect observed by other investigators⁽¹⁾ will not be an important factor in the study of FFTF fuel assembly models.
- The wire wrap pitch was found to have a strong effect on the pressure-drop behavior. The increase in friction factor over smooth tube values is given for Reynolds Numbers corresponding to sodium flow conditions at the tentative reactor operating levels in the following tabulation.

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(1) K. Rehme. The Measurement of Friction Factors for Axial Flow Through Rod Bundles with Different Spacers, Performed on the INR Test Rig, EURFNR-14ZP. U.S.-EURATOM Fast Reactor Exchange Program, November, 1965.

Effect of Wire Wrap Pitch on Reynolds Number

<u>Wire Wrap Pitch, in.</u>	<u>Increase in Friction Factor over Smooth Tube at Same Re No.</u>
24	2%
12	8%
6	19%
3	56%

- One of the flow ducts for these parametric hydraulic tests was instrumented to investigate apparent static pressure deviations around the periphery of the flow duct at an axial station due to dynamic effects. This dynamic effect was shown to be appreciable, amounting to as much as 7% of the static pressure drop per foot of test section. This indicates that careful attention must be paid to the location of static pressure taps on the flow duct for all future tests.

The purpose of flow mixing subtask is to develop and adapt the best of currently available analytical and experimental techniques to enable the study of coolant mixing and flow distribution in both normal and distorted FFTF fuel assemblies. The "noise" analysis technique to determine velocity distribution and mixing will be investigated and demonstrated.

c. Gas Cooling of FFTF Fuel Subassemblies During Inspection
(Work Done for System 70. Examination)

T. C. Reihman

The objectives of this program are to:

- Demonstrate the feasibility of decay heat removal from FFTF fuel subassemblies by the forced circulation of an inert gas stream.
- Establish the thermal hydraulic criteria for the design and operation of the inspection process decay heat removal system.

- Obtain data necessary to permit the establishment of operating limits for the specific heat removal system developed for the FFTF.

The major accomplishments in the past quarter were the completion of the gas heat transfer facility and the multipin, heated fuel assembly model, both of which are required for the performance of the entire test program. In addition, the major experimental portion of Subtask 1, demonstration of the feasibility of decay heat removal by the forced circulation of an inert gas stream, was completed.

The gas heat transfer facility consists of two separate gas blowers providing an axial and transverse flow to an electrically heated model of a proposed FFTF driver fuel assembly. These blowers may be operated simultaneously or independently. Air is used as the cooling gas; test conditions are established using accepted similarity principles to simulate the cooling capabilities of the proposed inert gas. The operating capabilities of this facility are as follows:

Flow rates	(70 °F air to 1 atm)
Axial	0 to 500 ft ³ /min
Transverse	0 to 300 ft ³ /min
Available pressure differential at test section	
Axial	15 psi maximum
Transverse	3 psi maximum

The test section consists of 169 heated pins of 1/4 in. OD with a heated length of 3 ft. It is capable of producing 40 kW and is instrumented to provide pressure drop-flow and temperature data.

The testing to demonstrate the feasibility of decay heat removal by the axial flow of gas was completed. Six test series were performed considering six different flows. For each series, the test section power was increased until a

maximum temperature of 600 °F was attained in the test section. This temperature corresponds to tentative maximum allowable fuel temperatures in the inspection facility. In all cases, this maximum temperature was observed at the downstream end of the center pin in the bundle. The mixed gas outlet temperature was found to be from 100 to 150 °F below this maximum rod temperature for all six flows.

Preliminary inspection of the data provided the following information.

Air Flow, scfm	Power to Produce 600 °F Maximum Surface Temperature,	$\Delta P/ft$, psi
	kW	
29.6	4.20	0.071
80.0	9.85	0.212
104.4	13.3	0.320
169	20.4	0.880
239	29.5	1.15
321	37.7	1.78

The experiments to demonstrate the feasibility of decay heat removal by the transverse flow of an inert gas during inspection and disassembly were completed for spacings of 0.030 and 0.255 in. between the heated fuel bundle and the flow baffles at the sides of the bundle. The bundle was oriented so that flow was toward one of the hexagonal corners (i.e., line contact between the hexagonal corners and the flow baffles). The 0.030-in. spacing corresponded to baffle contact with the wire wraps. The 0.255-in. spacing corresponded to the normal distance between the fuel bundle and the outer surface of the hexagonal flow ducts and represents minimum baffle clearance during liner removal. For the 0.030 in. gap spacing, approximately 250 scfm of air was required to remove 10 kW of heat while all fuel bundle temperatures were held

below 600 °F. For the 0.255 in. gap approximately 750 scfm of air were required to achieve the same total cooling of the heated bundle.

The feasibility of removing the decay heat by a combined axial and transverse flow such as would be experienced during flow duct removal was also studied for one configuration. A 12 in. axial gap in the flow duct near the downstream end (axial flow) was examined. The flow baffles were brought into contact with the outer corners of the portion of the hexagonal flow tube that remained to create a gap of approximately 1/4 in. on each side in the exposed portion. To cool every point of the heated bundle to below 600 °F, an axial air flow of approximately 100 scfm and a transverse air flow of about 400 scfm were required at a power of 10 kW. For 20 kW of generated heat, the axial and transverse air flows required were about 180 scfm and 700 scfm, respectively.

3. Instrumentation Development (This Work Performed for System 90. Instrumentation and Control)

M. R. Wood

a. In-Reactor Coolant Temperature Sensors

C. E. Huck and J. L. Jackson

Thermocouple Calibration. The objective of this effort is to develop and test thermocouples to provide reliable, commercially-available thermocouples to measure coolant temperature in the FTR thermal and nuclear environment. The post-irradiation electrical characteristics of all thermocouples irradiated to date in EBR-II (see illustrated Table 4.IX) is complete except for one 304L SS-sheathed W/W-26% Re unit and an unsheathed W-5% Re/W-25% Re unit now undergoing tests. Calibration data reveal an emf shift approximating 1 to 2% at 1400 °F and 2.5 to 4.5% at 600 °F for Chromel/Alumel thermocouples irradiated to a neutron fluence of about 2.7×10^{21} ($E > 0.18$ MeV). Figure 4.40 illustrates the relatively constant

TABLE 4.IX. Capsules BT-1 and BT-2 Pin Loadings

Loading in Pin	Group	Thermocouple No.	Description
<u>BT-1</u>			
	I (Top)	17	Ch/Al, beaded
	I	19	Geminol, beaded
	I	25	Ch/Al, beaded
	I	21	Geminol, beaded
	II	15	Ch/Al, beaded
	II	7	Ch/Al, Al ₂ O ₃ , 304L
	II	11	Ch/Al, Al ₂ O ₃ , Inconel
	II	29	Geminol, beaded
	III (Bottom)	13	Ch/Al, beaded
	III	9	Geminol, MgO, 304L
	III	23	W 5% Re/W 26% Re, stranded, beaded
	III	28	Geminol, beaded
	<u>BT-2</u>		
	I	16	Ch/Al, beaded
	I	20	Geminol, beaded
	I	26	Ch/Al, beaded
	I	22	Geminol, beaded
	II	14	Ch/Al, beaded
	II	8	Ch/Al, Al ₂ O ₃ , 304L
	II	12	Ch/Al, Al ₂ O ₃ , Inconel
	II	6	W/W 26%, Re, Al ₂ O ₃ , 304L
	III	10	Geminol, MgO, 304L
	III	18	Geminol, beaded
	III	24	W 5% Re/W 26%, stranded, beaded
	III	4	Pt/Pt 10% Rh, Al ₂ O ₃ , 304L

Note: All beaded TC's used 0.020 wire. The sheathed TC's were 0.100 in. diam and used 0.010 in. wire.

positive emf shift existing above a temperature of 600 °F. Thermocouple 13, irradiated in a higher and more uniform flux region of the reactor (Capsule Level III), evidenced a decreasing offset with increasing temperature in contrast to the sensors irradiated at Capsule Level II.

Figure 4.41 shows data for Chromel/Alumel units irradiated out of the EBR-II core region in Capsule Level I. These exhibited deviations varying about +0.7% at 600 °F to the range of -0.05 to -0.9% at 1400 °F.

Each pin contained melt wires for determination of irradiation temperature and various foils such as iron, nickel, titanium, and cobalt for fluence and neutron spectrum determination. The melt wires indicated the junctions of Group I reached a maximum of 405 °C while the junctions of both Groups II and III reached 440 °C. The irradiation temperature was lower than the designed level which was apparently caused by the actual gamma flux being lower than that which had been claimed for EBR-II. The foils have been counted but the fluence and energy spectrum has not yet been determined.

Thermocouple wire rather than extension lead wire was used to connect the 11-in. long sensors to the cold junction compensator, to provide for a minimum extraneous error during pre- and postirradiation calibrations. Each thermocouple had its own extension lead set which was used on both the post- and preirradiation calibration tests.

The wire attachment consisted of welding or crimping together the wires of like polarity. Use of Chromel/Alumel crimp-type connectors for other types of thermocouples dictated the use of special care in placing all connections in a uniform temperature zone.

FIGURE 4.40. *Chromel Alumel Thermocouple Postirradiation Deviations BT-1 and BT-2 Capsule Level II (Not corrected for preirradiation calibration)*

FIGURE 4.41. *Chromel Alumel Thermocouple Postirradiation Deviations BT-1 and BT-2 Capsule Level I (Not corrected for preirradiation calibration)*

During postirradiation calibration a portion of the 11 in. length of the thermocouple was placed in the maximum possible furnace thermal gradient as a means of detecting any possible distributed transmutation or other reactor-caused effects. Thus, these data will be directly comparable with all future calibration data for longer sensors which will be irradiated subsequently since future couples will extend through the region of maximum furnace gradient.

The Chromel/Alumel calibration offset shown in Figure 4.40 is caused either by irradiation effects, irradiation temperature, or more likely a combination of these two variables. About 5 °F of this offset probably results from the short term thermal aging effect recently described in the literature.⁽¹⁾ The remainder of the offset for sensors at the Level II position between 600 and 1400 °F must result from fast neutron irradiation effects because short term aging effects could not account for such a large deviation.

Postirradiation wire resistances agreed reasonably well with preirradiation values and it is concluded that irradiation causes negligible resistance changes. Wire temperatures were not recorded during preirradiation resistance measurements so a precise comparison of the pre- and postirradiation data cannot be made. However, sets of data were taken in air-conditioned laboratories and variations of more than 10 °F in the laboratory temperatures would affect only the fourth decimal place.

Thermal time constants of irradiated sensors, as determined using a stirred ice bath and a boiling water bath, agreed reasonably well with preirradiation data as shown below.

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(1) F. S. Sibley, N. F. Spooner, and B. F. Hall, Jr. "Aging in Type K Couples," Instrumentation Technology, pp. 53-55. July 1968.

Candidate Thermocouple Number	Time Constant, sec	
	Preirradiation	Postirradiation
8	0.72	0.75
10	0.61	0.70
11	0.44	0.52
12	0.42	0.41

Because all sheathed thermocouples incorporated grounded hot junctions, very little change in thermal time constant should be expected for the identical pre- and postirradiation test conditions. The slightly higher postirradiation time constants for two of the sensors may result, in part, from a dark oxide film occurring on the outside of the thermocouple sheath during irradiation and subsequent calibration. However, the estimated experimental accuracy of about ± 0.1 sec precludes any conclusion that fast neutron irradiation alters the time constant significantly.

b. Reactor Environmental Effects on Thermocouples

R. R. Bourassa and J. L. Stringer

The objective of this task is to determine instantaneous or dose rate effects on thermocouple calibration. The experimental work was conducted in a ^{60}Co gamma facility with a maximum dose rate of 2.3×10^7 R/hr. The freezing point of tin (232 °C), and the boiling points of liquid nitrogen (-196 °C), and water (100 °C) were used as the reference temperatures. The measurements at -196 and 100 °C were made differentially; the measurements at 232 °C were absolute. Bare Chromel-Alumel and iron-Constantan thermocouples with grounded, and ungrounded, junctions were immersed directly into the reference bath to ensure adequate heat transfer. A calibration shift of 4 ± 2 μV was observed for the grounded junction sheathed thermocouples while a shift of 8 ± 5 μV was measured for the ungrounded sheathed junctions. The results are given in Table 4.X.

Differences between voltage shifts observed in Chromel/Alumel and iron-Constantan thermocouples were too small to be detected in the experiments. Experiments indicated that the 8 μV shift was not caused by changes in the electrochemical potentials alone, but were caused by gamma heating of the thermocouple bead and the high resistance to heat transfer between the bead and sheath.

TABLE 4.X. *Increase in Thermocouple Output in a Gamma Field of 10^7 R/hr*

<u>Trial</u>	<u>Shift in Grounded or Bare TC, μV</u>	<u>Ref. Bath</u>	<u>Shift in Ungrounded Sheathed TC, μV</u>	<u>Ref. Bath</u>
1	0	N ₂	2	H ₂ O
2	9	CO ₂	10	H ₂ O
3	4	CO ₂	10	H ₂ O
4	4	CO ₂	3	H ₂ O
5	3	CO ₂	8	H ₂ O
6	3	CO ₂	8	H ₂ O
7	4	N ₂	1	N ₂
8	0	N ₂	2	N ₂
9	3	N ₂	0	N ₂
10	0	Tin	0	N ₂
11	0	Tin	10	N ₂
12			30	CO ₂
13			7	Tin
14			15	Tin
Avg, μV	3 $\pm$ 2		9 $\pm$ 5	

In an effort to obtain high neutron flux levels (which simulate as closely as possible the reactor environment expected in the FTR), an experiment was designed and conducted in a TRIGA pulsed reactor facility. This reactor can be pulsed

to 1500 MW with an 11 msec pulse width at half of the maximum pulse value. The gamma dose rate at the maximum pulse peak is more than 10^{10} R/hr while the neutron flux is about 10^{16} nv with the fast-to-thermal ratio being approximately 1:1. This facility can also be operated at a steady-state power level of 1 MW with a flux greater than 5×10^{12} nv.

The constant temperature bath used in this experiment consisted of a container of high-purity tin surrounded by an electrical heater with the melting point being the constant temperature reference for in-reactor thermocouple calibration. Four thermocouples, two 1/16 in. and one 1/8 in. of Chromel/Alumel and one 1/8 in. of iron-Constantan, were inserted into this bath. The capsule, heater, and thermocouples are shown in Figure 4.42. A multichannel brush oscillograph was used to record the emf generated in the thermocouples during the reactor pulse. Results of the pulsed data on the four thermocouples indicated an apparent temperature increase of less than 2.5 °C in every case. The results are given in Table 4.XI.

In addition to the pulsed experiment, two tests were carried out at a steady-state reactor power of 1 MW. In the first test, the reactor was brought to maximum steady-state power and maintained for 30 min. No change in voltage of thermocouples was detected, but during this time the instrumentation noise level increased. After 30 min of irradiation, the reactor was scrammed. Again no voltage change was detected. In the second test, an attempt was made to reproduce the experimental results of Leonard and Hunkar⁽¹⁾ who found a shift of 0.42 mV or about 10 C° following a reactor shutdown. The net neutron flux in that experiment was given as 6.3×10^{12} nv fast and 4.5×10^{11} nv thermal. In the present experiment the

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(1) J. H. Leonard and D. B. Hunkar. *Nucl. Appl.*, vol. 3, p. 718. 1967.

Neg 0642296-1

FIGURE 4.42. Capsule, Heater and Thermocouple Arrangement
Before Final Assembly

neutron flux was about 7×10^{12} nv fast and thermal. The power level was brought to 1 MW, held at that level for 20 min, and then scrammed. One Chromel/Alumel and one iron-Constantan thermocouple were monitored during this time and for 1 hr after shutdown. Essentially no changes were observed. The calibration shifts during the 80 min of the run are indicated in Figure 4.43. The random change of up to ± 1.0 C° is believed to be consistent with the error inherent in this experiment caused by reference temperature or instrument drifts.

TABLE 4.XI. *Increase in Thermocouple Output in a Radiation Field of 10^{16} nv Fast and Thermal and 10^{10} R/hr at 232 °C*

<u>Trial</u>	<u>1/16 in. Chromel/ Alumel, μV</u>	<u>1/16 in. Chromel/ Alumel, μV</u>	<u>1/8 in. Chromel/ Alumel, μV</u>	<u>Constantan, μV</u>
1	84	86	92	129
2	91	89	75	128
3	87	87	74	141
4	90	93	96	
5	84			
6	77			
7	91			
8	103			
9	103			
Avg, μ V	90 ± 6	89 ± 4	84 ± 10	133 ± 8
Avg, C°	2.3 ± 0.2	2.3 ± 0.1	2.3 ± 0.3	2.4 ± 0.2

c. Pressure Sensors for Sodium Service

T. J. Crocker and R. D. Crosier

The objective is to develop pressure instrumentation for in-core and process piping applications. The in-core sensors must operate immersed in sodium. Activities include evaluation of present technology, procurement and testing of commercially available products, component modification and improvement, and final proof testing.

FIGURE 4.43. *Change in Thermocouple Calibration as a Function of Time for a 1/16 in. Iron-Constantan and 1/8 in. Chromel/Alumel Thermocouple Together with Reactor Power Level*

The Static Sodium Facility was designed and built to test sensors immersed in sodium. The facility consists of four sodium pots (with a helium gas pressurization system) capable of testing up to eight sensors at pressure ranges to 250 psig and temperatures to 1200 °F. The transducers under test are compared to a reference transducer measuring the gas pressure to the sodium pots.

Four pressure sensors have been tested. Two strain-gage type sensors have failed. One failure was caused by faulty fabrication where the joint between the sensor housing and the sheathed cable was brazed rather than welded. The other failure was a broken tab junction at a strain gage. A total of 120 days

test time was accumulated on the other two sensors. One unit was a NaK filled capillary system with the diaphragm seal sensing 1200 °F sodium. The other unit was a strain-gage type immersed in sodium at 600 °F. The following tabulation lists the initial characteristics and the results of the conclusion of testing. Based on these results the most promising sensor for in-core use appears to be a NaK-filled capillary system connected to an electromechanical transducer in the thermal shield region.

Summary of Test Results

Type Max Operating Conditions:	Strain Gage 600 °F, 200 psig		NaK Capillary 1200 °F, 200 psig	
	Initial	120 day	Initial	120 day
Thermal Zero Shift (% Full Range/°F)	0.0065	0.0158	Negligible	Negligible
Thermal Sensitivity (% Full Range/°F)	0.0056	0.0186	Negligible	0.001
Linearity (% Full Range)	0.8	0.7	0.5	0.6

An industry survey was made to determine the availability of high temperature pressure sensors. Although several manufacturers listed electromechanical sensors for 900 °F service in their catalogs, further contact with these vendors disclosed that this temperature rating was valid only for a few hours of operation. Creep in the diaphragm material is the limiting factor at the present time.

d. Low Level Neutron Detectors

C. N. Jackson and N. C. Hoitink

This work focuses on evaluation and testing of detectors, cables, electronic components, and instruments for a low level neutron flux startup channel system and a failed fuel delayed neutron monitor. The technical requirements for selecting candidate detectors and associated equipment emphasize high neutron

sensitivity, and capability for operation under a combination of high gamma background and elevated temperatures.

Results achieved include completion of the gamma irradiation test facility in the PNL ^{60}Co gamma pit; acquisition, evaluation, and calibration of the fast pulse processing and measurement system; preliminary testing of the specified characteristics; preamplifier and the detector, and continuation of the first phase of the testing program to determine optimum parameters.

The PNL ^{60}Co gamma facility provides a convenient testing environment for simulation of FFTF gamma background conditions, but lack of experiment positioning apparatus necessitated design and fabrication of such a feature. Simulation of the FFTF environment required inclusion of a capability for operation to temperatures of 700 °F simultaneously with the high gamma background of 10^6 R/hr maximum. The completed hardware for the gamma facility, shown in Figure 4.44a, enables precise radial positioning of the 4 in. diameter tube which holds the detectors. A neutron source provides a thermal neutron flux of approximately 3.5×10^3 nv at the detector location. The temperature test jig, shown in Figure 4.44b, provides a test position for a 2 in. diameter detector and adapters for smaller detectors.

The measurement system adopted incorporates the fastest commercially available pulse handling equipment with maximum pulse shaping versatility. Semiautomatic data retrieval presents the data in the form of plotted curves. Integral and differential bias curves identify the neutron sensitivity and simultaneously reveal effects of gamma irradiation on the sensitivity and performance of the detector. Figure 4.45 shows the complete instrumentation system. Figure 4.46 illustrates an example of the integral and differential bias curves

Neg 0681451

FIGURE 4.44a. *Gamma Test Facility*

4.103

BNWL-880

Neg 0682423-2

FIGURE 4.44b. Temperature Test Holder

Neg 0682423-1

FIGURE 4.45. Measurement System in Operation

FIGURE 4.46. Typical Integral and Differential Bias Curves

plotted by the system. These curves were obtained in a zero gamma field so the background count, as noted in the figure, represents system noise.

Since the preamplifiers can be operated at some distance from the detectors, current efforts center on determining preamplifier performance outside the hostile environment imposed on the detectors.

Table 4.XII summarizes the important characteristics including laboratory noise measurements for the three preamplifier types. The 1 μ sec clip shown for the TC-133 represents a modification made to the amplifier which had an unacceptably long 100 μ sec clip as supplied by the manufacturer.

TABLE 4.XII. Preamplifier Characteristics

Model	Type and Sensitivity	Risetime, nsec	Pulse Shaping	Maximum Input Capacitance	Equivalent Noise Referred to Input Open	Ion Pair rms 30 ft RG62/U Cable
PA-4	0.25 V/pC (Charge)	16	0.3 μ sec	3000 pF	6×10^3	18×10^3
TC-133	1 V/pC (Charge)	50-500	1 μ sec clip	500 pF	1.5×10^3	12.5×10^3
RA-801	0.5 V/ μ A (Current)	<25	None	N/A	12.5×10^4	25×10^4

Table 4.XIII lists the important characteristics of the six fission chambers and fourteen ^{10}B lined proportional counters being tested. Selection of detectors reflects the necessity to achieve high neutron sensitivity and low gamma sensitivity.. Fission chambers typically provide excellent gamma tolerance but neutron sensitivities are an order of magnitude less than proportional counters with equivalent volumes.⁽¹⁾ Boron-10

(1) W. J. Price. Nuclear Radiation Detection. McGraw-Hill, Inc., New York, N.Y. 1964.

lined proportional counters exhibit reasonable gamma tolerance and recent reports⁽¹⁾ indicate that satisfactory operation might possibly be achieved using small diameter detectors in gamma fields above 10^5 R/hr. Figure 4.47 shows representative multiplication characteristics for each detector type, obtained in accordance with MIL-D-22558A (SHIPS) Section 4.8.5

TABLE 4.XIII. *Neutron Detector Characteristics*

Quantity	Identification	Type	Neutron Sensitivity, (Counts/sec)/nv	Dimensions	
				Length in.	Diameter in.
2	RSN 10A	Fission	0.7	12	2
2	WL-23033	Fission	0.7	12	2
2	NA-29	Fission	1	12	2
2	RSN 17A	^{10}B Prop	4	15	1
3	RSN 251	Prop	5	*	*
5	SK 410	Prop	0.6	11	0.25
4	RSN 321	Prop	1.5	12	0.5

* RSN 251 is a cluster of nine RSN 250 detectors in an assembly 11 in. long by 1 in. diam

The curves shown in Figure 4.48 depict the effects of pulse shaping time constants on the system gain employing the various preamplifier types. The effect of different risetime input pulses on the curves illustrates the importance of the detector collection characteristics which control the pulse risetime. The RA-801 and the PA-4 preamplifiers are not significantly affected by the input capacitance but the TC-133 risetime deteriorates from 50 nsec with no input capacitance to

(1) Irwin Bluestein. "Neutron Measurements in High Gamma Backgrounds," ANS Transactions, vol. 8, supplement, p. 24. 1965.

300 nsec with the 20 ft input cable (270 pF). This reduction in high frequency response explains why the TC-133 response falls rapidly at faster pulse shaping time constants. Based on these curves, the faster detectors will probably operate well with fast shaping time constants (preferable for gamma pulse-pileup rejection using either the RA-801 or the PA-4 preamplifiers. The TC-133 preamplifier will work better slower risetime fission chambers and with slower pulse shaping time constants.

FIGURE 4.47. Gas Gain Characteristics For Various ^{10}B Lined Proportional Counters

FIGURE 4.48. Preamplifier Characteristics as Functions of Time Constants

IV. DEVELOPMENT

D. L. Condotta

HIGHLIGHTS OF METALS AND MATERIALS DEVELOPMENT

J. C. Tobin

An estimated fluence limit for the FTR vessel material (304 SS) based on short-term tensile ductility properties was found to vary with temperature.

The effects of irradiation on the low-cycle fatigue properties of austenitic stainless steels threatens the conservatism of the ASME code more than do the effects of hold-time between cycles on unirradiated material.

Irradiation to relatively high fluences has very little effect on the water retention characteristics of serpentine; however, it has been determined that nuclear graphite will be used for the FFTF shield material, and development work on all alternative materials has been stopped.

All of the candidate FFTF shield materials (graphite, carbon, and serpentine) react in some degree with both the stainless and mild steels which would be used as containers.

Subassembly X-022, containing more than 500 stainless steel test specimens, was removed from the EBR-II after a peak exposure of about $1 \times 10^{22} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$).

Type 304 SS exhibits a 4-fold reduction in creep-rupture life after irradiation to fluences varying from 0.5 to $2 \times 10^{22} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$).

After a fluence of about $5 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$) samples of 304L SS from an EBR-II control rod thimble showed a maximum volume increase of 1.17 percent. At even higher fluences approaching 10^{23} n/cm^2 the swelling appears to approach 10 percent.

Procurement of sections of fuel duct for testing in the core Components Test Loop pointed out the major difficulties of purchasing these items within a detailed specification.

Initial tests indicate that Metco 405 and Stellite 1016 have good wear properties in molten sodium at 700 °F.

D. METALS AND MATERIALS DEVELOPMENT

J. C. Tobin

1. Materials Properties

J. E. Irvin

The primary purpose of this effort is to provide property data on all materials of interest to the FFTF project. Effects of irradiation, temperature, environment, and other major parameters on materials properties will be determined under conditions as closely related to FFTF requirements as possible.

a. Recommendation of Fluence Limit for FTR Vessel

R. A. Moen

A fluence limit based on the retention of adequate mechanical properties at end-of-life for the vessel material (Type 304 SS) was estimated using existing irradiation data.

The fluence limit for the FTR vessel, based on short-term tensile ductility properties, was found to vary with temperature according to the following tabulation:

Fluence Limits, n/cm^2 for $E > 0.1$ MeV
Based on Reduction in Total Elongation

<u>Temp., °F</u>	<u>10% reduction</u>	<u>30% reduction</u>	<u>50% reduction</u>
1200	2.7×10^{17}	2×10^{18}	1.6×10^{19}
1050	3.7×10^{17}	4.3×10^{18}	6.4×10^{19}
900	1.3×10^{18}	3.8×10^{20}	Not determined
570	6.2×10^{17}	2.3×10^{19}	2.7×10^{20}

Short-term tensile data for material irradiated and tested at approximately the same temperatures were compiled from ETR 6-7 loop and EBR-II Row 7 irradiations. Fluence values were converted to include only neutrons with energies greater than 0.1 MeV.

The fluence limits established are thought to be conservative from the standpoint of the FTR vessel, because of the use of thermal reactor data. Recent work has shown that for a given fluence ($E > 0.1$ MeV) one would expect about twice as much displacement type damage in ETR G-7 samples than in EBR-II Row 7 samples. According to estimates, the number of displacements expected in the FTR vessel will fall 10 to 15% below those occurring in EBR-II Row 7 specimens.

- b. Code Applications (This work performed for System 30.
Reactor)
R. A. Moen

The degree of conservatism of applicable codes, under the unique conditions to be experienced by materials in the FFTF, is being identified. A review of certain areas of interest for high temperature systems such as the FTR vessel provided the following results:

- Stress-Rupture - The ASME Code allowable design stress for Type 304 SS at 1200 °F, the ultimate design temperature of the FTR vessel in the vicinity of the outlet nozzles, is controlled by stress-to-rupture. The design stress value at 1200 °F corresponds to 80% of the minimum stress to cause rupture in 100,000 hr. The only data available thus far on irradiated stainless steel are for rupture times of 1000 hours or less. These data indicate a degradation in stress rupture properties, as determined by comparison with average properties for unirradiated material. However, at 1000 hr, the rupture strength of the irradiated material is still 50% above the value dictated by 80% of the minimum stress to produce rupture on the code allowable design stress.
- Low-Cycle Fatigue - Low-cycle fatigue data on irradiated stainless steel is even more scarce than the stress-rupture data discussed above. The Manson-Halford

relationship has shown that the effects of irradiation on the low-cycle fatigue properties of austenitic stainless steels threaten the conservatism of the Code (fatigue curves from Code Case 1331-4) more than do the effects of hold-time between cycles on unirradiated material.

c. Irradiation Behavior of FTR Shield Materials (This work performed for System 30 Reactor)

D. E. Mahagin and D. E. Baker

Materials selected for shielding applications in the FTR are required to maintain their capacity for neutron moderation and absorption for the expected plant lifetime. In addition, irradiation-induced changes in the shield material's dimensions or its physical and mechanical properties must be characterized as a function of fluence in order that the shield can be designed to accommodate changes. Early conceptual studies of the FTR shield identified the mineral serpentine, ungraphitized carbon, and electrode graphite as potential shielding materials. The major consideration in the selection of these materials was an economic comparison between them and the more expensive (and more highly characterized) nuclear grades of graphite. This task was undertaken for the purpose of providing data concerning the effects of neutron irradiation on these materials at temperatures and fluences representative of FTR conditions.

During this report period, irradiation of samples of all three classes of materials was completed. Specimen identification and irradiation conditions are shown in Table 4.XIV.

The critical parameter with respect to the irradiation behavior of serpentine is its water retention. Dehydration of the mineral reduces its capacity for neutron moderation and, thus, decreases its shielding effectiveness. Measurements of the water content of irradiated serpentine samples

have been completed. The results are shown in Table 4.XV. Also shown are measurements of the water content of unirradiated specimens which were heat treated to provide thermal histories identical with those of the irradiated samples. The measured weight gain at 500 °F in the out-of-reactor test was caused by water vapor condensation in the furnace vent tube.

TABLE 4.XIV. Irradiation Conditions of Candidate Shield Materials

<u>Material</u>	<u>Type</u>	<u>Irradiation Facility</u>	<u>Temperature Range, °F</u>	<u>Fluence Range</u>
Carbon	CIX	ETR	200-800 °F	$0.5-5 \times 10^{20}$ nvt (E > 0.18 MeV)
	BC	ETR		
	250	ETR		
	GA	ETR		
	785	ETR		
Graphite	HBX	ETR		
	AGR	ETR		
	580	ETR		
Serpentine	QUEBEC	ETR	200-600 °F	$1.4-3.0 \times 10^{20}$
		Hanford	500-900 °F	

TABLE 4.XV. Water Retention of Serpentine Samples

<u>Sample</u>	<u>Irradiation Temp., °F</u>	<u>Fluence</u>	<u>Water Released^(a) During Irradiation, wt%</u>	<u>Water Released During Thermal Control Run</u>
1	500	1.42×10^{20} nvt (E > 0.18 MeV)	0.80	-0.63
2	550		0.82	0.47
3	670		2.19	1.38
4	825		3.35	3.24
5	890		7.57	7.48
6	500	2.2×10^{20} nvt (E > 0.18 MeV) (b)	0.61	--
7	570		1.28	
8	670		2.20	--
9	825		4.74	--
10	890		9.10	--

a. Natural serpentine contains 13 to 14 wt% H₂O

b. Estimated

These results are in excellent agreement with values calculated from a model of the mineral's dehydration kinetics during thermal treatment in the absence of irradiation. They indicate that the irradiation to relatively high fluence levels induces very little effect on serpentine's water retention characteristics and that the shield design could be based on considerations of its operating temperature. A recent decision has been made to use nuclear grade graphite for the FTR shield, consequently, all development effort with alternative materials has been stopped.

d. Irradiation Effects on Absorbing Materials for FTR Control Rods (This work performed for System 30 Reactor)

D. E. Mahagin, D. C. Carlson and A. L. Pitner

The candidate neutron poisons for use in the FTR control rods are boron carbide and tantalum. Their selection was based on nuclear considerations relative to their absorption characteristics in the FTR spectrum and the necessary reactivity control requirements of the reactor. Neither of these materials has been irradiated at the conditions under which they will be utilized in the FTR. In consequence, irradiation testing of both poisons is necessary.

During this report period, seven experimental capsules were designed, fabricated, and shipped to EBR-II for irradiation. A description of the rods and their calculated operating conditions are shown in the following tabulation.

As the irradiation progresses, selected rods will be removed and destructively examined. They will be replaced with identical rods in order that data can be collected on three discrete boron burnup levels and three tantalum fluence levels.

Operating Conditions for Experimental
EBR-II Neutron Absorber Rods

Rod	Absorber	Cladding Mat'l	Cladding ID, in.	Cladding OD, in.	Cladding Temp, °F	Poison Temp, °F	Goal Exposure
1	VIPAC B ₄ C, (80% TD)	304L SS	0.37	0.50	1130-1250	1140-1260	1-10% of B-10 Burnup
2			0.37	0.50	1130-1250	1140-1260	Burnup
3			0.43	0.50	1170-1320	1185-1340	Burnup
4			0.43	0.50	1170-1320	1185-1340	Burnup
5	Ta		0.30	0.38	1230	1350	0.7-5 x 10 ²² nvt
6			0.30	0.38	1230	1350	0.7-5 x 10 ²² nvt
7			0.30	0.38	1230	1350	0.7-5 x 10 ²² nvt

e. Chemical Compatibility of Shield and Control

Rod Materials (This work performed for System 30 Reactor)

D. E. Mahagin, G. L. Tingey and T. M. Fish

The materials used for shielding and control rod poisons in the FTR must be compatible with their containers and with the reactor environment in which they are utilized. Their reaction rates with cladding and container materials must be well defined to ensure that the container lifetimes are not limited by chemical interaction with the shield or neutron absorbers. It is also necessary to characterize the behavior of the poison materials in contact with high temperature sodium to determine the consequences of a cladding rupture.

Tests to determine the interaction of candidate shield materials with potential container alloys were completed. All tests were conducted by packing the shielding materials around tensile specimens of austenitic stainless steel and mild steel and maintaining contact between the materials for 336 hr at a temperature of 940 °F. In addition to the compatibility specimens, control samples of the steels were annealed at 940 °F in helium for the same time period. Following their heat treatment, all the steel samples were evaluated by measurement of their mechanical properties, by metallographic examination, and by microhardness traverses across their diameters. The

results are shown in Tables 4.XVI and 4.XVII. It is apparent that all the candidate shield materials react to some degree with both stainless and mild steel.

TABLE 4.XVI. *Tensile Properties of Steel After Treatment with Graphite and Carbon for 336 Hours at 500 °F*

AISI-1015 Low Carbon Steel	Average Weight Change, %	Average Modulus, 10^6 psi	Average Yield Strength, 10^3 psi	Average Tensile Strength, 10^3 psi	Average Total Elongation, %
After Helium Anneal	-0.04	26.2	29.4	54.2	33.2
In Graphite	+0.01	28.7	58.9	63.9	24.2
In Carbon	+0.07	27.4	53.7	63.9	23.2
In Serpentine	+0.08	31.1	59.8	--	--
304 Stainless Steel	0.2% Offset Yield Strength				
After Helium Anneal	+0.07	27.6	53.3	93.5	63.6
In Graphite	-0.01	25.9	55.9	95.2	60.8
In Carbon	0.00	26.4	53.0	94.7	60.4
In Serpentine	-0.02	27.5	56.3	97.6	61.4

TABLE 4.XVII. *Microhardness from Edge to Center of Tensile Specimens (DPH Values, 1 kg Load)*

Distance in from Sample Edge, in.	As-Received	After Heating in Helium	After Treatment with Serpentine	After Treatment with Graphite	After Treatment with Carbon
<u>304 Stainless Steel</u>					
0.005	264	151	253	216	258
0.010	245	158	247	258	273
0.020	247	153	245	250	258
0.040	235	144	230	232	232
0.080	221	148	184	198	208
0.160	194	162	182	191	196
<u>AISI-1015 Carbon Steel</u>					
0.005	122	112	111	107	136
0.010	119	123	120	112	131
0.020	114	119	120	118	124
0.040	114	109	122	111	123
0.080	120	104	125	120	130
0.160	123	101	129	120	133

f. Cladding Characterization (This work performed for fuel development program)

M. M. Paxton

The purpose of this program is to determine the variation in two lots of AISI 304 SS, seamless tubing 0.250 in. x 0.016 in. wall, and in any subsequent lots that will be used in future fuel pin experiments. Room temperature and 1800 °F burst tests were completed. Room temperature uniaxial tensile tests were completed and elevated temperature tests are in progress.

Room temperature burst tests were completed on tube lots "E", and "F". The results are summarized in Table 4.XVIII. Room temperature burst tests were conducted with non-defective tubing.

TABLE 4.XVIII. *Room Temperature Test Burst Data for Non-defective "E" and "F" Tubing*

<u>Tube Lot</u>	<u>Burst Pressure, psi</u>	<u>Hoop Stress, psi</u>	<u>Measured Posttest OD, in.</u>
E	10,700	72,870	0.300
	10,700	72,870	0.305
	10,700	72,870	0.298
	10,800	73,550	0.304
	10,800	73,550	0.305
F	9,950	67,760	0.303
	9,950	67,760	0.303
	9,900	67,420	0.301
	9,950	67,760	0.303
	9,950	67,760	0.310

Conducting uniaxial tensile tests at elevated temperatures on 0.016-in. wall tubing presented two problems. The grips used for room temperature tests could not be used at elevated temperature due to their size. This problem was circumvented by using Swageloks. Because of the thin wall of the tubing it was felt necessary to do the elevated temperature tensile tests in an inert atmosphere. An apparatus was designed and constructed to satisfy this requirement.

- g. Biaxial Creep and Fuel Cladding Studies (This work performed for Fuel Development Program)

R. W. Barker

The objective of this program is to determine the combined effects of environment and fast reactor irradiation on the biaxial stress-to-rupture properties of fuel cladding material. The effects of fluence, coolant-cladding interactions, fuel-cladding interactions, time at temperature, and stress will be evaluated with respect to the biaxial creep properties. Fuel cladding materials will be irradiated and tested in gaseous and liquid metal environment. These tests will be conducted at temperatures from 800 to 1400 °F and for rupture times to 10^4 hr.

A Mk B-7 type subassembly, (Subassembly X-022) containing over 500 stainless steel test specimens was discharged from the EBR-II after a peak exposure of $\sim 1.1 \times 10^{22}$ n/cm² (E > 0.18 MeV). This assembly contained uniaxial tensile and creep specimens and biaxial stress-to-rupture specimens. All of the pins were opened and the loose (irradiated in a gaseous environment) uniaxial specimens, 124 total, were placed in containers according to experiment designation. Thirty-two of the sodium filled subcapsules were opened. These contained a total of 370 uniaxial test specimens. Sodium was removed from the deionized water. The specimens were water rinsed followed by an additional rinse in acetone to facilitate drying. The balance of unopened capsules, including all capsules containing biaxial test specimens, were placed in dry storage at the Idaho Nuclear Radioactive Materials Laboratory. All flux and thermal monitors were recovered from opened capsules and are being processed. Due to difficulties encountered in the cleaning of biaxial test specimens irradiated in EBR-II, it was decided that no attempts would be made to remove the sodium from the biaxial test specimen cavity. Rather, the biaxial test specimens

would be extracted from their sodium-filled irradiation capsule in an inert atmosphere. The specimen exterior may then be easily cleaned. Prior to test, the cap would be removed and the specimen will be attached to the biaxial test rig in an inert atmosphere--with the pure sodium remaining in the specimen cavity.

The replacement in Row 7 for X-022 was prepared and shipped to EBR-II in April, 1968. This subassembly was assigned the designation X-041 and was placed in position 7A3 (X-022 had been located in position 7C4). The same orientation with respect to the reactor core center was utilized in X-022 and X-041.

h. Uniaxial Creep (This work performed for System 30. Reactor)

A. J. Lovell and J. J. Holmes

The objective of this phase of the program is to provide postirradiation creep-rupture data after irradiations in sodium at elevated temperatures to various fast neutron fluences for reactor vessel and core structural materials. Creep-rupture specimens irradiated in space available pins in the EBR-II were tested at five test temperatures. The Type 304 SS specimens were irradiated in four different pins to the approximate fluence range of 0.5 to 2×10^{22} n/cm² in a helium atmosphere at temperatures near 900 °F. Specimens in the irradiated condition have approximately a 4-fold reduction in rupture life when compared with the as-fabricated material, as shown in Table 4.XIX. Thermal aging of control specimens is partially complete.

TABLE 4.XIX. *Postirradiation Creep Rupture of Type 304 XX*

<u>Specimen No.</u>	<u>Test Temp., °F</u>	<u>Stress, psi</u>	<u>Rupture life, hr</u>
A-2-2	1300	20,000	5.9 (25) (a)
A-6-5	1250	20,000	21.9 (83)
A-38-5	1000	45,000	25.3 (88)
A-38-6	1800	40,000	57.1 (215)

a. Values in ( ) are as-fabricated control specimens.

i. Damage Analysis (This work performed for System 30.
Reactor)

J. J. Holmes and H. R. Brager

The objective of this study is to establish the mechanisms responsible for fast reactor induced mechanical property changes develop an understanding of swelling effects in candidate FFTF metals, and establish data extrapolation techniques. Emphasis was placed on swelling studies during the quarter. Density changes in an EBR-II control rod thimble (304L stainless steel) irradiated to a peak fluence of 4.8×10^{22} n/cm² (E > 0.1 MeV) were measured. At the peak fluence a maximum volume change of 1.17% was measured. The rate of swelling (in terms of fluence) at this point appears to be sufficiently great that volume change will need to be considered in component design.

Electron microscopy was completed on samples cut from the control rod thimble. The specimens were irradiated over a fluence range from 0.03 to 5×10^{22} n/cm² (E > 0.1 MeV). The irradiation-produced cavity size, number density, and the material microstructure, were a strong function of the irradiation temperature (704 to 866 °F) and fast fluence.

Volume changes based on immersion density measurements are given in Table 4.XX. The base (unirradiated) density

TABLE 4.XX. *Volume Change Results for EBR-II Control Rod Thimble*

Sample No.	Fluence, n/cm ² (E > 0.1 MeV)	Volume Change, $\Delta V/V$ %	Irradiation Temp., °F
A	1.1×10^{22}	0.07 ± .06(a)	704
M	1.7	0.30 ± .02	704
B	2.8	0.29 ± .03	704
C	4.0	0.54 ± .05	728
D	4.8	1.17 ± .04	792
E	3.1	0.91 ± .04	866
Q-1	2.2	0.39 ± .09	866
Q-2	1.0	0.17 ± .02	866
F	0.7	0.09 ± .02	866
R-1	0.4	0.08 ± .04	866
R-2	0.25	0.05 ± .03	866
R-3	0.10	0.08 ± .07	866
G	0.03	0.15 ± .03	866

a. Later corrected to 0.14 ± 0.01 on basis of annealing studies

was estimated by annealing seven specimens for one hour periods at temperatures to 1900 °F to remove the voids which cause the swelling. All specimens except sample D approached a common density of about $7.900 \pm 0.003 \text{ g/cm}^2$ (confidence limits describe total scatter band). This value agrees very well with the preirradiation value of 7.895 g/cm^2 measured at ANL on archive material. It is expected that sample "D" will approach the base density at annealing temperatures in excess of 1900 °F. These data are shown in Figure 4.49 together with previously reported PNL data on an EBR-II experimental capsule and Dounreay data on Type 316 SS reported by Cawthorne and Fulton.⁽¹⁾ The Dounreay data have been converted to equivalent EBR-II fluence by multiplying the Dounreay fluence by a factor of 1.6. The data can be described by a concave upward curve which seems to be increasing at both low and high fluence, going through a minimum at about $2.5 \times 10^{21} \text{ n/cm}^2$ $E > 0.1 \text{ MeV}$. At high fluences swelling is occurring very rapidly with increasing fluence and appears to be approaching values in the 10% range near 10^{23} n/cm^2 . The magnitude of the volume increase together with the rate of increase at 10^{23} n/cm^2 suggests that volume change should be taken into account in FFTF design.

The increase in volume with decreasing fluence below $2.5 \times 10^{21} \text{ n/cm}^2$ relies on the validity of sample "G". The density decrease in sample "G" was completely recovered after annealing to 1900 °F but electron microscopy revealed no voids in this sample. In all other specimens examined (A, B, C, D, E, and F), density decrease was always associated with internal void formation. This indicated that the density decrease is due to second phase distribution differences or some surface effect which disappears on annealing. In any case, the density decrease in sample "G" was caused by a different mechanism

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1. C. Cawthorne and E. J. Fulton, Nature, vol. 216, p. 575. 1967.

FIGURE 4.49. Fast Reactor Induced Volume Changes in Austenitic Stainless Steels

than void formation. This suggests that all volume changes measured at fluences less than about $7 \times 10^{21} \text{ n/cm}^2$ may need to be considered as being, in part, due to the same process as that which caused the density decrease in "G" rather than void formation per se.

The low fluence swelling behavior will be further studied by examining a second sample from the "G" position and by performing additional swelling studies on specimens taken from the structural materials subassembly X-022 and specimens from a second EBR-II thimble. Data at exposures greater than $4.5 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$) will be obtained on an EBR-II safety rod when this item is removed from the reactor.

Transmission electron microscopy studies on austenitic stainless steel are being conducted to provide information for interpreting fast reactor irradiation effects and to assist in both the formulation of material irradiation behavior models and improvement of the performance of austenitic stainless steel in reactor environments.

Transmission electron microscopy examination has been completed on specimens cut from the EBR-II control rod thimble. Quantitative analysis of the micrographs has just started.

Sample G, Figures 4.50 and 4.51, which received about $3 \times 10^{20} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$) at 866 °F, should exhibit the condition of the microstructure which is closest to that of the unirradiated steel. Three distinct features should be noted about this section of the steel:

1. Preliminary stereoscopic examination showed about $10^{12}/\text{cm}^3$ of 100 to 200 Å diameter cavities within the foil. Two of these cavities are shown near the center of Figure 4.50. The number and size of the observed cavities are consistent with about one-tenth the amount of helium that would be produced by transmutation of a nominal boron impurity level found in commercial austenitic stainless steel. Therefore, the cavities are probably helium filled bubbles.

FIGURE 4.50. *Sample G, AISI 304 SS Irradiated on EBR at 866 °F to 3×10^{20} n/cm² ($E > 0.1$ MeV)*

Neg 5774-C

24,000X

FIGURE 4.51. *Sample G, AISI 304 SS Irradiated in EBR-II at 866 °F at 3×10^{20} n/cm² ($E > 0.1$ MeV)*

2. About $10^{12}/\text{cm}^3$ of 2000 to 5000 Å M_{23}C_6 precipitates were observed.
3. The dislocation density of sample G is significantly greater than fully annealed material. This observation indicated that the unirradiated steel had at least a small amount of cold work, about 5 to 10% reduction of area, in its unirradiated condition.

The microstructure of sample G differed significantly from samples A through F which contained:

1. Large cavities whose size and concentration were strong functions of irradiation temperature and fluence.
2. A lower density of large precipitates and dislocation density than sample G.
3. Displacement damage whose morphology, over the fluence range investigated, was a strong function of irradiation temperature.

Samples B, Figure 4.52 and sample E, Figure 4.53, received about the same fluence [$3 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$)] and were irradiated at the lowest and highest temperatures investigated, 704 and 866 °F, respectively. The thicknesses of the electron microscopy foils have not yet been determined. Assuming a 1500 Å foil thickness, very preliminary analysis indicates that sample B contained about $2 \times 10^{15}/\text{cm}^3$ of 100 Å average diameter cavities, while sample E contained about $7 \times 10^{14}/\text{cm}^3$ of 250 Å average diameter cavities. This trend toward a lower concentration of larger cavities at higher irradiation temperatures, over the temperature range investigated, is also shown by sample D, Figure 4.54. This steel, which received the highest fluence, $5 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$) at 792 °F, contains about $10^{15}/\text{cm}^3$ of 275 Å average diameter cavities. The microscopic observation showed that sample D had the greatest amount of irradiation swelling of the samples examined. This observation is consistent with the macroscopic density measurements.

Neg 5743-B

93,600X

FIGURE 4.52. *Sample B, AISI 304 SS, Irradiated in EBR II at 704 °F to 2.8×10^{22} n/cm² ($E > 0.1$ MeV), Microstructure out of contrast*

Neg 5729-D

93,600X

FIGURE 4.53. *Sample E, AISI 304 SS, Irradiated in EBR-II at 866 °F to 3.1×10^{22} n/cm² ($E > 0.1$ MeV) Microstructure out of contrast*

Neg 5674-B

93,600X

FIGURE 4.54. *Sample D, AISI 304 SS Irradiated in EBR-II at 792 °F to 4.8×10^{22} n/cm² ($E > 0.1$ MeV), Microstructure out of Contrast*

Samples A through F, which received high fluences, have a lower concentration of both large carbide precipitates and dislocations than sample G, which received a relatively low fluence of about 3×10^{20} n/cm² ($E > 0.1$ MeV). Very preliminary X-ray analysis of samples G and D substantiate the observed difference in precipitate morphology of the specimens. Examination of sample G produced two strong $M_{23}C_6$ (X-ray peaks while sample D produced weak) X-ray signals just above the background noise level. The metal carbide precipitate concentration could have been reduced by displacement damage. The damage mechanism could displace carbide atoms into the matrix, thereby increasing the effective solubility of the carbon in the steel at the irradiation temperature. Similar results have been previously observed by dissolving and re-forming small bubbles of inert gas in irradiated and post-irradiation annealed UO_2 .⁽¹⁾

It appears that the microstructure of the examined austenitic stainless steel irradiated to high fluence (about 10^{22} n/cm²) is effectively independent of the preirradiation dislocation network, but is strongly dependent on irradiation temperature. Sample G, Figures 4.50 and 4.51, irradiated to the relatively low fluence of 3×10^{20} n/cm² ($E > 0.1$ MeV) contained a higher dislocation density than annealed material. However, no dislocations were observed in the microstructure of sample C, irradiated to a high fluence of about 4×10^{22} n/cm² ($E > 0.1$ MeV), and shown in contrast in Figures 4.55 and 4.56. Either the displacement damage of the steel irradiated to high fluence is great enough to mask the dislocations, or the existing dislocations were able to climb and be annihilated by interaction with other dislocations and/or grain boundaries during the irradiation. In either event, the dislocation structure of sample G was not observable in the microstructure of the other samples irradiated to a high fluence [about 10^{22} n/cm² ($E > 0.1$ MeV)] over the temperature range investigated.

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(1) A. D. Whapham and B. E. Sheldon, *Nucl. Appl.*, 2, 123 (1966)

Neg 5824-E

93,600X

FIGURE 4.55. *Sample C, AISI 304 SS, Irradiated in EBR-II at 728 °F to 4.0×10^{22} n/cm² ($E > 0.1$ MeV), Microstructure in Contrast*

Neg 5824-C

2,3400X

FIGURE 4.56. *Sample C, AISI 304 SS, Irradiated in EBR-II at 782 °F to 4.0×10^{22} n/cm² ($E > 0.1$ MeV), Microstructure in Contrast*

The microstructure of these high neutron fluence specimens varied with irradiation temperature. At the 704 °F, the microstructure consisted of a small amount of ordering of structure along crystallographic planes in addition to the normal low temperature displacement damage. At 728 °F irradiation temperature, a low concentration of ordered structure is observed which appear to be small faulted loops, Figure 4.56. At 866 °F irradiation temperature, the size and concentration of the faulted loops have significantly increased with a corresponding decrease of the low temperature displacement damage, sample F, Figure 4.57.

2. Fabrication Development

R. N. Johnson

a. Vendor Development and Evaluation (This work performed for Fuel Development Program)

R. C. Aungst

The tubes used in tests conducted in the Core Components Test Loop (CCTL) were in the form of 6 ft of straight hexagonal section blending through a 5 in. transition zone into a 4 ft length of round. This configuration is prototypical of that section of the fuel duct containing the fuel bundle. Procurement of these tubes has served to highlight some problem areas to be anticipated in production of the more complex tubes needed for the FTR and NPTF. Some of the difficulties were as follows:

- Seam welded tubing had to be accepted instead of the desired seamless.
- Only two bids were received, one to be explosively formed and the other drawn. Explosive forming was more costly by a factor of 3 so the drawn tubing was ordered.
- It was necessary to relax the minimum elongation requirements in the tensile tests from 40 to 30% and to eliminate grain size requirements in the welds.

Neg 5654-E

93,600X

FIGURE 4.57. Sample F, AISI 304 SS, Irradiated in EBR-II at 866 °F to 7×10^{21} n/cm² (E > 0.1 MeV), Microstructure in Contrast

- Most welds had an undercut groove along the edge as much as 0.010 in. deep.
- None of the tubes fully met the diameter requirements. This may have been partially due to the unpredictability of the effects of the annealing treatment.

Requests to bid were also sent to four potential fabricators for sections of round tubing incorporating a taper at one end. These were needed as preforms for developing a tapered hexagonal section. No responsive bids were received.

Vendors appear extremely reluctant to bid on small quantities of tubes of unusual geometry. Those tubes that have been received have been of marginal quality at best. Not only is considerable fabrication development needed but a competitive situation must be created among tube fabricators before acceptable tubes can be produced for the FTR and NPTF.

b. Fuel Duct Fabrication Development (This work performed for Fuel Development Program)

F. B. Quinlan, J. O. Vining and W. F. Brown

Preliminary work on forming a tapered hexagonal section of a fuel duct was completed. Three sections of tubes 6 in. in length were reduced to a conical shape using a hydraulic press. Two of these parts were successfully formed into a tapered hexagonal section using a tapered hexagonal die and mandrel. Due to the experimental nature of the tooling, trouble was encountered in stripping the parts from the die and off the mandrel which resulted in end damage. One part was scrapped and the other two were reduced in length approximately 50% to remove the damaged ends.

- c. Wear Resistant Coatings for Tube Components [This work was performed for the Fuel Development Program and for Systems 30 (Reactor) and 60 (Irradiation Testing).]

R. V. Bowersock and R. N. Johnson

Tubular components in the FTR such as the fuel ducts, the closed and open loops, and the control rod ducts will be subject to wear and abrasion in certain areas. These areas must be capable of withstanding the charge-discharge and operating cycles in the FTR environment without excessive wear, galling, or self-welding.

Candidate materials for wear testing were selected which could be used as coatings, wear pads, or in some cases even as the basic material of construction to protect components from excessive wear. Some materials which are of interest have been or are being tested at LMEC. To supplement the data that have already been generated there, wear plates and pins were fabricated to LMEC size specifications.

Type 304 SS pins are being tested on contact with flame sprayed coatings of Metco 405 (nickel aluminide), Inconel 750, Stellite 1016, and Inconel 718. The wear tests will be conducted at LMEC in flowing sodium at temperatures ranging from 700 to 1200 °F.

The Metco 405 coating is not normally used as a wear resistant coating, but is often used as an undercoating for better adhesion of the other hard facing alloys. Since nothing was known about the compatibility of the nickel aluminide with high temperature sodium, it was included in order to test both its wear properties and its corrosion resistance.

The initial test results, shown below, indicate that at 700 °F, at least, the nickel aluminide has good wear properties. Further tests will be run at 900 and 1200 °F.

Performance of Various Material Combinations in Sliding Contact

Pin Material	Plate Material	Sodium Temperature, °F (°C)	Pin Sliding Velocities, cm/sec	Coefficient of Sliding Resistance at Test Temperature		Specific Wear on Pins [Average ^(b)], cm ³ /cm ² -kg
				Initial	Final ^(a)	
304 SS	Metco 405	700 (371)	3.1,3.9,4.7	0.28	0.48	4.5×10^{-11}
304 SS	Stellite 1016	700 (371)	3.1,3.9,4.7	0.28	0.33	6.4×10^{-11}

a. Test period ~200 hours

b. Contact pressure for each of 3 pins involved in each test = 8.65 kg/cm^2 (123 psi)

Personnel at Fermi and LMEC were contacted to discuss their experiences with wear resistant materials in sodium. The following is a summary of their comments and recommendations:

- Nitrided 304 SS has shown very poor performance. Untreated 304 SS performs much better.
- Cobalt base alloys, such as Stellite 6B, have shown good wear and corrosion properties, but should be avoided, if possible, in high flux regions since any material abraded from such surfaces may collect in external piping and cause extremely high radiation levels outside the core regions. This problem will require further evaluation since the nickel base alloys, under consideration, also create radioactive cobalt under irradiation, but to a much lesser degree. The cobalt base alloys may have better corrosion resistance, however, which could mitigate the potential radiation levels.
- Inconel 718 has shown exceptionally good wear resistance and overall performance and was highly recommended. Some erosion tests in high velocity sodium will be conducted for further evaluation of this material.
- Colmonoy-4 has given good service in nonnuclear applications, but was not recommended for a high flux environment due to its high boron content and potential helium formation.

The stability of precipitation hardened alloys, such as Inconel 750, in a high neutron radiation field is questionable, and therefore not recommended.

d. Fuel Duct Creep Evaluation (This work performed for Fuel Development Program)

E. B. Schwenk

The relatively complex geometry of the hexagonal section fuel ducts preclude a simple analytical analysis of general creep behavior based on results from uniaxial creep specimens. In the absence of such an analysis experimental evaluation is the only feasible way to ascertain that behavior.

Stress risers exist in the fuel duct corners due to the noncircular cross section of the ducts and can vary significantly depending on the magnitude of the corner radius. Anticipated reactor pressure and temperature levels thus could induce time-dependent deformation of the corners leading to an overall swelling or increase in duct dimensions.

Based on the dimensions of hexagonal section tubing fabricated for hydraulic evaluation tests in the CCTL and an anticipated pressure level of 85 psi at the bottom of the active core, the following equations⁽¹⁾ predict a maximum corner stress of 17,000 psi.

$$\frac{\sigma_{\max}}{P} = K \left[2 \left(\frac{L}{R} \right) \left(\frac{R}{t} \right) + \left(\frac{R}{t} - \frac{1}{2} \right) \right] + \left(\frac{L}{R} \right) \left(\frac{R}{t} \right) \left[\frac{1}{2} \left(\frac{L}{R} \right) - \left(\frac{L}{R} \right) \frac{M_o}{PL^2} + 2 - \sqrt{3} \right] \left[-1 + 12 \left(\frac{R}{t} \right)^2 \frac{1}{2 \frac{R}{t} - 1} \right] \quad (1a)$$

$\sigma_{\max}$ = Maximum corner stress (psi)

K = Correction factor

(1) C. D. Flowers. Structural Analysis of a Hexagonal Tube or Ring Under Internal Pressure, BNWL-263, Pacific Northwest Laboratory, Richland, Washington, June 1966.

P = Pressure (psi)
 L = 1/2 wall length (in.)
 R = Average corner radius (in.)
 t = Thickness (in.)
 M_O = Bending moment at mid section of wall

$$\begin{aligned}
 &= \left[\frac{PL^2}{6 + \frac{\pi}{L/R}} \right] \left[1 + \frac{1.57}{L/R} + \frac{0.558}{(L/R)^2} \right] \\
 \frac{\Delta_{tot}^E}{PL} &= \frac{2R}{t} \left[0.03 \left(\frac{R}{e} \right) (1 - \nu^2) - 0.292 (1 - \nu^2) + 0.262 \right. \\
 &\quad + 0.262 \left(\frac{\alpha E}{G} \right) \\
 &\quad + 0.134 (1 - \nu^2) \left(\frac{R}{e} - 1 \right) \left(\frac{L}{R} \right) \left(\frac{1}{2} - \frac{M_O}{PL^2} \right) \left. - \left(\frac{L}{R} \right)^3 \left(\frac{R}{T} \right)^3 (1 - \nu^2) \right] \\
 &\quad \left(\frac{1}{2} - 6 \frac{M_O}{PL^2} \right) \tag{1b}
 \end{aligned}$$

where Δ_{tot} = Total defection across the flats (in.)
 E = Youngs Modulus (psi)
 e = Distance between centroidal and neutral axis at the corner (in.)
 α = Nonuniform shear stress distribution factor
 ($\alpha = 1-L$ for rectangular cross section)
 G = Shear modulus = $\frac{E}{2} (1 + \nu)$
 $\frac{\alpha E}{G} = 2 (1 + \nu)$

This level of stress could induce yielding of the tube corners at 400 °F based on minimum yield strength values of AISI 304⁽¹⁾ or a creep rate in the order of 0.0001% per hour at 1000 °F.⁽²⁾ In addition, the equations are based on a

-
- (1) *W. F. Simmons and J. A. Van Echo. The Elevated Temperature Properties of Stainless Steels, ASTM-DS5-51, Fig. 18, p. 61. 1955.*
 (2) *W. F. Simmons and H. C. Cross. The Elevated Temperature Properties of Stainless Steels, ASTM-STP-124, p. 612. 1953.*

perfect hexagonal geometry which may not be economically feasible or practical to produce in large tubes.

To evaluate the effect of hexagonal geometry on creep, two specimens were fabricated from CCTL tubes; Figure 4.58 shows one of the specimens. Measured values of room temperature wall-to-wall displacement at a single pressure level agreed fairly well with elastic displacements predicted by Equation (1b); for 120 psi the measured values averaged 0.0027 in. against 0.0022 in. predicted by the equation. Thus, at higher temperatures the corresponding wall deflection could increase due to the temperature dependence of Young's Modulus, and yielding and subsequent creep of the tube corners.

Experimental evaluation of the effect of static and dynamic loading on creep behavior has started. A dynamic load creep test was initiated to simulate reactor operation and because intermittent loading can accelerate creep in austenitic stainless steels.⁽¹⁾ The static or constant pressure-constant temperature specimen was suspended within a vertical electric furnace and maintained at 85 psi pressure and 1000 °F temperature. The dynamic or intermittent pressure-constant temperature specimen was tested similarly but was pressure-cycled. A given cycle consisted of one minute 0 to 85 psi, 6 minutes at 85 psi, one minute to reach 0 psi and two minutes at 0 psi. Antipodal flat and corner 'diameters' were measured at three points along the axis of the tube using a micrometer for a total of 18 measurements per test exposure.

Changes in both the average corner-to-corner (ΔC) and average flat-to-flat (ΔF) dimensions with time are shown in Figure 4.59. After about 2500 hr exposure the constant

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(1) G. J. Guarviere. The Creep-Rupture Properties of Aircraft Sheet Alloys Subjected to Intermittent Load and Temperature Application, ASTM STP 165, pp. 105-148. 1955.

4.142

Neg. 068075

FIGURE 4.58. Fuel Duct Creep Specimen

BNWL-880

FIGURE 4.59. Hexagonal Tube Creep

pressure-constant temperature ΔF dimension increased about 0.1% while the average ΔC dimension increased about 0.02%. The reason for the rapid increase in ΔF between 1200 and 1700 hours is not clear but might be related to a geometry effect.

Following about 1600 hr exposure at temperature the intermittently pressurized specimen showed an increase in the ΔF dimension of about 0.03% with virtually no change in ΔC . It should be noted that if the intermittent pressure specimen data were plotted against time at pressure instead of time at temperature the exposure times would be about 60% of the values shown. Thus, further exposure of the dynamically pressurized specimen is necessary to determine if a similar rapid increase occurs in the ΔF dimension. The data indicate that creep expansion of the fuel ducts under the noted conditions is possible.

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A. W. de Mershand	C. R. Nash
G. E. Driver	R. E. Nightingale
K. Drumheller	D. P. O'Keefe
J. F. Erben	M. G. Patrick
E. A. Evans	C. L. Peckinpaugh
L. M. Finch	L. T. Pederson
J. C. Fox	J. A. Perry
R. C. Free	R. E. Peterson
R. J. Gandy	O. W. Priebe
E. E. Garret	H. L. Pringle

Battelle-Northwest (contd)

W. D. Richmond	D. M. Walley
W. E. Roake	J. H. Westsik
R. L. Raske	J. Wett
J. M. Seehuus	R. G. Wheeler
F. H. Shadel	L. A. Whinery
D. W. Shannon	R. D. Widrig
D. E. Simpson	T. W. Withers
C. R. F. Smith	N. G. Wittenbrock
R. J. Squires	M. R. Wood
D. D. Stepnewski	F. W. Woodfield
G. H. Strong	D. C. Worlton
C. D. Swanson	J. M. Yatabe
J. C. Tobin	H. H. Yoshikawa
K. G. Toyoda	W. R. Young
M. A. Vogel	FFTF File (25)
R. C. Walker	Technical Information (5)
	Technical Publications (2)