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FAST FLUX TEST FACILITY STUDIES

APRIL, 1965

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PROGRESS REPORT
FAST FLUX TEST FACILITY STUDIES

April, 1965

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PROGRESS REPORT
FAST FLUX TEST FACILITY STUDIES
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1.0 INTRODUCTION

On April 1, 1964, a Hanford document⁽¹⁾ reviewing the status of the United States fuel development program for fast reactors was issued by F. W. Albaugh and P. G. Holsted. The document described U. S. progress and technical problems in development of fast fuels, and presented two major recommendations: (1) increased emphasis should be placed on fast fuel development, and (2) a fast flux test facility consisting of a fast test reactor and supporting auxiliaries should be constructed to serve as a primary irradiation test facility for the U. S. fast reactor development program.

The conceptual scope of the proposed fast flux test facility was described in the document. The fast test reactor was visualized as a tube-type 400 Mwt sodium-cooled unit, providing a hard neutron flux of 10^{16} n/cm²-sec with 75% of the flux above 0.1 Mev. Fuel testing capabilities were provided for specific powers up to 3.0 Mw/kg and outlet temperatures up to 1400 F.

In the period of April 1964 through the first quarter of 1965 conceptual design studies continued on an incidental and intermittent basis, and other programs in progress at Hanford and elsewhere resulted in significant inputs of supporting technology. Hanford work during this period was done first within Hanford Laboratories-General Electric Company and then continued by the successor organization, Pacific Northwest Laboratory of Battelle Memorial Institute (BNW).

(1) F. W. Albaugh and P. G. Holsted. Fast Fuel Test Reactor Study. April 1, 1964.

Section 2.0 of the report summarizes the results of these activities and includes schematic illustrations of the reactor concept, the skewed reactor tube proposal, the heat removal system, and the building layout.

Section 3.0 of the report presents the results of the conceptual design studies and includes engineering drawings of the core concept, the cylindrical tubular arrangement, several candidate types of reactor tubes, and building plans and elevations. In order to provide a complete description of the current fast flux test facility concept in one document, Section 3.0 also includes some descriptive information which was presented in the initial report.

Section 4.0 of the report presents the supporting technology findings including physics data curves and tabular data on time requirements for reactor design and construction.

In the spring of 1965, the USAEC requested BNW to perform a major conceptual design study and analysis of the "Fast Flux Test Facility." This major study is now in progress and detailed results will be reported at a later date.

2.0 SUMMARY

2.1 Abstract

During the period April 1964 through the first quarter of 1965, limited study work was performed on the conceptual design of the proposed fast flux test facility and information was obtained in important areas of supporting technology. The studies for the FFTF were based on: (a) maximum use of components already proven in sodium-cooled reactor service, (b) maximum utilization of known sodium technology and (c) planned maximum possible utilization of existing sodium technology facilities and sites for all required development.

Additional conceptual design data were obtained on the tube-type reactor core, on the instrumentation system, on the type of building required and on the scope of Engineering Test Facility. In the area of supporting technology, information was obtained on physics, materials selection, driver fuel development, sodium technology, analytical chemistry requirements, program management technology, and fast-flux irradiation testing needs for structural materials.

2.2 Conceptual Design Studies

The Fast Flux Test Facility concept will provide a versatile test facility for high energy neutron irradiation of fuels and structural materials for fast breeder reactors. The test facility is centered about a fast test reactor which is equipped with a variety of instrumented open and closed loops and specimen irradiation stations for tests under conditions which closely simulate the actual core environments of fast breeder power reactors.

The test facility complex includes fuel and specimen handling and examination equipment and cells, a reactivity test facility, and a supporting engineering test facility. Maximum use will be made of components already proven in sodium-cooled reactor service.

FIGURE 2. 1
Fast Flux Test Facility (Reactor Elevation)

FIGURE 2.2
Skewed Tube Elevation

2.4

2.2.1 Reactor Core Concept

The fast test reactor is conceived as a 400 Mw_t, fast neutron reactor (Figure 2.1) with fuel and test channels arranged in a vertical, skewed array of tubes (Figure 2.2). The reactor uses liquid sodium as the primary coolant for the nuclear fuel with successive interchanges of waste heat to the secondary circuit and finally to either the atmosphere or to river water (Figure 2.3).

Test specimens of fuel will be inserted into the reactor core tubes; material specimens will be inserted into test holes in the reflector and shields. The conceptual design is based on six "closed" (i.e., independently cooled) instrumented test loops, six "open" instrumented test loops cooled by the reactor primary coolant system, and 15 to 25 test holes. Details are given in Sections 3.2 and 3.3.

2.2.2 Instrumentation

Limited studies were performed on instrumentation requirements for the FFTF and on conceptual design of the system. Two general requirements have been established, namely (a) Reactor Control and (b) Data Recording, Computation, and Retrieval. In meeting these requirements sensors, monitors, and related equipment will be used which employ proven sodium reactor technology whenever possible. Details are given in Section 3.7.

2.2.3 Structural Concept

During the reporting period preliminary scoping was performed on a below-grade design for reactor confinement (Figure 2.4). In this design a special below-grade vessel houses the reactor and primary coolant system. The design, which will satisfy all applicable safety requirements, will have a number of advantages over conventional above-grade confinement domes. These advantages include increased operating flexibility, improved fuel handling capability, reduced operating and maintenance costs, and probable reduction in construction costs. Details are given in Section 3.11.

2.5

FIGURE 2.3
Schematic of Heat Removal System
(Showing Four of Six Closed Loops)

2.2.4 Engineering Test Facility

The scope of engineering testing research and development activities in support of the FFTF was studied. It was tentatively concluded that much of the feasibility testing and the component testing can be adequately performed in existing AEC sodium research and development facilities. Integrated performance testing (or systems testing) under proposed operating environments and procedures is required prior to startup to ensure systems reliability, and following startup to evaluate proposed in-reactor experiments, equipment modifications, and operating modes. These two capabilities (prestartup systems testing and operations support systems testing) will be provided in an Engineering Test Facility (ETF). The systems testing facility in the ETF will be a nonisothermal flow system of greater than fuel assembly flow and heating capabilities. Peripheral to the systems testing facility will be several smaller specialized supporting facilities for electrically heated thermal-hydraulics studies, defected fuel testing, and small component development and evaluation. Details are given in Section 3.12.

FIGURE 2.4
Cutaway of FFTF Building

2. 3 Supporting Technology

2. 3. 1 Introduction

Design of the FFTF will be based, to the maximum extent possible, on use of equipment and technology which has already been proven in sodium-cooled reactors. However, there are some areas of supporting technology which will require specialized application in the FFTF program. During the reporting period information was obtained in some of the more important areas of supporting technology, including physics, materials selection, driver fuel development, sodium technology, analytical chemistry requirements, project management techniques, and materials irradiation test requirements for the long-range fast breeder reactor program. These findings are summarized below.

2. 3. 2 Physics

Neutronic data were obtained on a series of conceptual fast test reactor designs. Critical sizes and reflector control worth for a range of reactor geometries and compositions were obtained. Preliminary estimates of sodium coefficients and core expansion coefficients were made.

The sodium void coefficient of the FFTF was reviewed and, as previously reported, ⁽¹⁾ this coefficient was found to be quite negative because of the predominance of leakage effects in the test reactor geometry.

Further analysis was made of the physics aspects of control of the reactor by variations in the reflector albedo. It was found that reduction of reactivity by removal of portions of the reflector offers in principle a greater control span than control by rotating asymmetrical poisons drums in the reflector. However, when very large control margins must be compensated for, supplementary in-core control may also be employed. Details are presented in Section 4. 2.

(1) Same as Introduction.

2. 3. 3 Materials Selection Considerations for FFTF

Information was obtained on the selection of metallic materials of construction for the FFTF. Details are given in Section 4. 3. Some of the more pertinent conclusions were as follows:

1. In the FFTF (and in other fast reactors) the effect of (n, α) and (n, p) reactions on the properties of materials in fast neutron environments will be substantial and potentially limiting to component service life, and this factor must be carefully evaluated in materials selection.
2. Representative heats of commercially available austenitic stainless steels exhibit marked variability in chemical composition, grain size, homogeneity of structural constituents, and mechanical properties. Normal heat-to-heat variations in properties may exceed variations caused by a service variable. Well-written procurement specifications, coupled with closely monitored inspection and quality control procedures will be required in the FFTF program.
3. For nickel base alloys, both solution hardened and precipitation hardened, vacuum melting is recommended to eliminate low melting point impurities such as tin, lead, and bismuth, and gaseous impurities such as oxygen and nitrogen.
4. Preconditioning thermal-mechanical treatments tailored for reactor service conditions can markedly improve the properties and irradiation stability of reactor structural materials, and advantage should be taken of this approach when selecting FFTF materials.
5. All materials chosen for exposure to sodium must be carefully examined for mutual compatibility at FFTF conditions, and corrosion tests run where required.

2. 3. 4 Cermet Driver Fuel Development

A cermet fuel, PuO_2 dispersed in a stainless steel matrix, is being considered as the driver fuel for the FFTF. Recently, within the BNW

Plutonium Utilization Program, work on cermet fuel development has been performed, with emphasis on methods for fabricating cermets and methods for preparing PuO_2 with properties suitable for cermets. In addition, irradiation tests are under way in the MTR and other tests are planned to determine irradiation behavior of cermets prepared by various fabrication methods using PuO_2 having various physical properties.

Three methods of fabricating cermets, namely coextrusion, impaction, and cold-pressing, are being investigated. Coextrusion has given very encouraging results in fabrication of high density cermets; pneumatic impaction is a satisfactory backup process. Low density cermets may be routinely produced by cold-pressing. Details are given in Section 4. 4.

Cermet studies to date have used stoichiometric plutonium dioxide. Plutonium suboxide ($\beta\text{-Pu}_2\text{O}_3$) offers promise for improvement of cermet stability, and specimens of this material have been prepared.

2. 3. 5 Sodium Technology

Information was obtained on the status of sodium technology with regard to corrosion, coolant purity control, and special engineering problems such as self-welding, fretting, vibration, and cavitation. Potential problems in these areas are anticipated in connection with the FFTF design; these problems will be identified early in the design stage and supporting development applied for their solution. Details are given in Section 4. 5.

2. 3. 6 Analytical Chemistry Service Requirements for FFTF

A review was made of the analytical chemistry services required for development, design, and operation of the Fast Flux Test Facility. It was determined that BNW has available most of the basic facilities for analytical services required for development and design support. For operation of the reactor some analytical development and some additional equipment and facilities will be required. Details are given in Section 4.6.

2. 3. 7 Program Management Technology

To achieve maximum effectiveness in program direction, the Program Evaluation and Review Technique ("PERT") has been adopted for planning

and follow-up; this technique will also be employed at a later date for cost control. Initial "PERT" program charts have been developed; these charts identify key areas of the program, and serve to point out potential future problem areas. Details are given in Section 4. 7.

Information was obtained on time requirements for design and construction of liquid-metal cooled reactors and of associated major problem areas. Time requirements from start of Title I Design to achievement of full power operation ranged from 48 months (EBR-I) to 115 months (Dounreay). Major problem areas with liquid metal cooled reactors included:

- (a) Entrainment of gas in coolant -- 3 reactors
- (b) Heat exchanger failures -- 1 reactor
- (c) Steam generator failures -- 1 reactor
- (d) Pump failures -- 2 reactors
- (e) Moderator can leaks -- 1 reactor
- (f) Reactor mechanisms difficulties -- 2 reactors
- (g) Graphite shielding damage -- 1 reactor
- (h) Coolant purity control problems -- 1 reactor.

To minimize these problems and their attendant delays in the FFTF program, components proven in sodium reactor service will be used to the maximum possible extent. Extensive component testing and sound engineering practice and management will be employed throughout the entire FFTF program.

2. 3. 8 Materials Irradiation Test Requirements

Since the FFTF will be a primary test facility for the fast breeder reactor program, it is essential that the functional capabilities of the reactor be carefully conceived to meet demands for future materials development and irradiation testing. To this end information was obtained on materials irradiation test requirements for the fast breeder reactor development program. Details are given in Section 4. 8.

2. 3. 9 Hanford Reactor Technology Experience

The design of the FFTF will be based, to the maximum extent possible, on use of components and technology already proven in sodium-cooled reactor experience. However, it has been found that there are many critical areas of supporting technology, such as physics, fuel development, design analysis, nuclear safety, analytical chemistry, corrosion monitoring, irradiation testing, project management, as well as design practicability "know-how, " which will be required in order to carry out a successful FFTF program. Hanford reactor technology experience has been extensive in all these areas of supporting technology. Details are given in Section 4. 9.

3.0 CONCEPTUAL DESIGN OF THE FAST FLUX TEST FACILITY

3.1 General

The essential requirement of any test facility is that it provide a controlled environment in which the behavior of experimental objects may be observed. The FFTF must provide neutronic, thermal, chemical, and hydraulic environments at least as severe as those which will be created in the cores of the first several generations of fast power reactors*.

Physics calculations for the FFTF indicate that the minimum power core which would provide the proper test environments is about 400 Mwt. The corresponding core volume is about 600 liters.

The general testing approach envisioned for the FFTF is conventional; fuel assemblies will be inserted into the reactor core region via instrumented "loops" and irradiation samples will be inserted into selected stations in the reflector and shielding regions. Fuel assemblies may or may not be power reactor fuel prototypes. "Loops" consist of two general types: "closed loops" which are completely isolated from the reactor primary coolant fluid and which include separate provisions for pumping, heat transfer, coolant purification, control, and instrumentation; and instrumented "open loops" which utilize reactor primary coolant fluid. Current plans envision 6 "closed loops", 6 "open loops", and 15 to 25 test positions in the reflector and shield.

-
- * (1) Large Fast Reactor Design Study, ACNP-64503, Allis-Chalmers, Atomic Power Development Associates, Babcock and Wilcox Company, January 1964.
- (2) Liquid Metal Fast Breeder Reactor Design Study, CEND-200, Combustion Engineering, January 1964.
- (3) Liquid Metal Fast Breeder Reactor Design Study, GEAP-4418, General Electric Company, January 1964.
- (4) Liquid Metal Fast Breeder Reactor Design Study, WCAP-3251-1, Westinghouse, January 1964.
- (5) An Evaluation of Four Design Studies of a 1000 MWe Ceramic Fueled Fast Breeder Reactor; COO-279, Reactor Engineering Division, Chicago Operations Office, USAEC, December 1, 1964.

3.2

Assuming a core volume of 600 liters and assuming a nominal fuel assembly size of 36 inches long by 5 inches equivalent diameter, it is apparent that 12 test loops will occupy a volume of about one-fifth of the reactor core. In the face of this commitment of a significant fraction of the core volume to test loops, should not the full reactor core be assembled from individual reactor tubes so arranged as to provide optional use as "driver" fuel channels or as open or closed loops? This question started a comprehensive examination of tubed reactor concepts resulting in the present proposal.

3.2 Summary Description of the Fast Flux Test Facility

3.2.1 Facility Scope

This conceptual proposal describes a versatile tube-type fast test reactor and supporting auxiliaries, conceived for fast neutron irradiation testing of fuels and structural materials for fast breeder reactors. The fast test reactor is equipped with open loops, closed loops, and materials specimen irradiation stations for in-core and proximate core tests under conditions which closely simulate the actual core environments of power reactors. Specialized facilities are provided for handling, storage, and limited examination of driver fuel, test fuel and material samples. Modern instrumentation is included for recording, retrieval, and computation of data to provide maximum capability for obtaining necessary experimental data and information.

In addition to the reactor and its direct auxiliaries, the proposed test complex also includes a reactivity test facility for evaluation of the nuclear characteristics of fuel test samples prior to actual charging of the test reactor; and a supporting Engineering Test Facility.

3.2.2 Reactor

The test reactor is conceived as a 400 Mwt, fast neutron reactor with fuel and test channels arranged in a vertical, skewed array of tubes. (Fig.3.1, p.3.27) The reactor uses liquid sodium as the primary coolant for the nuclear fuel with successive interchanges of waste heat to the secondary circuit and finally to either the atmosphere or to river water. Since this test reactor will not have a breeder zone blanketing the fast core zone, the neutron reflector zone (a high nickel content annular ring) is immediately adjacent to outer row of reactor core tubes. Reactor control for safety and operational control will be obtained by movable components in the annular reflector zone and by movable in-core components. Extensive study is needed to assure adequate control for the large core reactivity changes resulting from large-scale fuel tests. The reactor core and primary sodium circuits (including pumps, heat exchangers, and primary coolant auxiliaries) will be contained in successive pressure vessel shells to prevent release of radioactive material to the environs.

Although the out-of-reactor components of large closed test loop systems will not be included within the same enclosure with the reactor, the primary coolant parts of the loops will be contained in separate isolated containment systems.

3.2.3 Specimen Testing

Specimens for fast flux irradiation testing will be inserted into the reactor core tubes, or into specimen tubes in the reflector zone, by the transfer machine. Although the reactor design will permit a large number of irradiation tests of structural materials, the unique feature of this reactor concept is the facility with which both open and closed loops may be installed in the core, repeatedly charged and discharged with fuel and other test specimens, and finally removed from the core. This facility for testing is obtained from the reactor tube concept which permits full isolation and full accessibility to each channel in the reactor core.

The reactor tubes extend above and below the core zone in an expanding array as dictated by the skewed geometry. The tube extensions below the core penetrate a tubesheet (which functions as a combined

3.4

heat barrier, neutron shield, and tube constraint) for connection to the primary coolant supply plenum. Tube extensions above the core pass through a similar barrier-shield-restraint tubesheet and thence on up through some ten feet thickness of thermal barrier and neutron shielding. Reactor tubes containing driver fuel or open loop tests, (exposed to and cooled by reactor primary sodium coolant), will discharge coolant through orifices in the reactor tubes above the core zone and below the neutron shield. Above the neutron shield where the expanding tube array provides clearance between the reactor tube extensions, special coolant piping connections will be made to closed loop tubes. Above the piping connection zone, the reactor tube extensions will enlarge to accommodate a fuel transfer container as part of the charge-discharge sequence for fuel assemblies (including test fuel specimens and assemblies). At the upper ends of the reactor tube extensions, the tubes will extend through a gamma shield and seal plate to permit direct access to the reactor tubes by the fuel transfer machine.

3.3 Reactor Core

3.3.1 General

As presently visualized, the core of the FFTF is composed of 61 hexagonal cross-section reactor tubes, tightly compacted and slightly skewed (2.7°) about their long (vertical) axis. (Fig. 3.1, p. 3.27) Although there are no physical limitations on the number of tubes which may contain experimental fuel assemblies, initially only 12 of the 61 tubes are planned to contain experimental fuel assemblies; the balance will be driver fuel. The tubes are slightly tapered in the core region, measuring 4.25 inches across flats at the core bottom and 4.58 inches across flats at the core top. The driver fuel tube wall thickness in-core is 0.1 inch; the tube overall length is 40 feet.

3.3.2 Reactor Tube Configuration

In general, tube systems for nuclear reactors are difficult design problems because of the opposing requirements (1) a close-packed tube

array in the reactor core zone, and (2) an open, well-separated tube array away from the core to facilitate fueling and coolant piping connections. The reactor tube concept proposed for the FFTF is developed as described in Figure 3.2, p. 3.28. For the sake of clarity, only seven of the 61 process tubes in the reactor core are considered in this explanation.

The obvious advantage of the proposed skewed array is its characteristic voidless intercept zone at the reactor core extending to an open symmetrical array by straight, nonintersecting extensions of core tubes. The skewed array permits a full tube system for reactor fuel positioning and cooling.

Coolant fluid may be supplied to, and discharged from the skewed tube system by a variety of methods including plenum, grouped tube services, and individual tube services. Reactor tubes in the skewed array may be single pass or (reentrant) multiple pass. Depending upon the design of flow channels in the reactor tube, the inlet and discharge points may be at the same or opposite ends of the tube.

Individual Reactor Tube Monitoring

The skewed reactor tube system is compatible with monitoring of all coolant flow channels. In addition to fluid measurements such as pressure, temperature, and flow rate, special apparatus such as fuel rupture detectors and other test devices may be installed when available. Since reactor fuel assemblies may be monitored for individual process and performance variables, the entire reactor may be operated at higher performance levels than would be possible only with bulk measurements of fuel and coolant conditions. In addition, because of the individual tube monitoring, safe reactor operation can be obtained during deliberate or inadvertent atypical operation such as irradiation of defected fuel elements.

Fuel Handling

The skewed array presents a simple, consistent geometrical system to facilitate mechanical fueling handling. The use of a reactor tube which extends in a straight line from the reactor core to an accessible location remote from the core permits simple tooling for the fuel handling apparatus; the fuel and fueling tool are fully guided in the tube from core to charging port.

Close Packed Core

The skewed array combined with a skewed hexagonal reactor tube creates a core without voids between tubes. This feature imparts a high degree of dimensional stability to the core. Nuclear stability is enhanced by eliminating uncertain void size and "filler material". The elimination of core voids is particularly desirable for fast nuclear reactors because of their inherent sensitivity to core dimensional instability and because typical high power densities require efficient utilization of core volume for coolant, fuel and structure.

The skewed array of skewed hexagonal elements can be forced to interelement contact by axial translation of the elements when the array is constrained to permit only axial movement. This feature permits less stringent dimensional tolerances to obtain a given condition of core packing.

Core Geometry

The close packed hexagonal skewed array yields a close approximation of a right circular cylindrical core zone. Since the core axis is vertical and the core is located at the skew intercept plane, the diverging lower tube configuration inherently provides desirable distribution of

the core critical mass in the event of an incident resulting in core melting. Further, when the skew intercept plane is at the bottom of the core zone, the neutron flux profile along the core axis is modified favorably for upward coolant flow.

Reactor Tube

The skewed array allows the use of reactor tubes to contain the reactor fuel and to provide coolant fluid conduits in the reactor core region. Individual tubes for each fuel subassembly provide general performance monitoring as discussed previously. In addition, the reactor tube concept is compatible with the requirements for instrumented in-core loops for fuel and irradiation experiments utilizing a variety of materials and coolants. The immediate accessibility of each individual tube for inspection or recharging is a prime advantage of the reactor tube concept.

The high degree of accessibility also reduces the cumulative operator radiation hazard when compared to large core and coolant openings of typical non-tubed power reactor designs.

The tube concept generally permits the use of smaller (more numerous) coolant conduits. The resulting thinner metal thicknesses improve the thermal stress-strain limitations on high performance systems.

3.4 Coolant Systems

3.4.1 Introduction

The coolant fluid selected for the primary heat transfer circuit of the FFTF is sodium. The selection of this fluid is dictated largely by the certainty that most fast power reactors to be designed in the next twenty years will use sodium as the primary circuit coolant. Considerations of radiation stability, good specific heat, high boiling temperature, high thermal conductivity, and chemical characteristics which make sodium a preferred power reactor coolant also favor its use in the FFTF.

3.8

The nominal design operating conditions of the reactor including the heat transfer apparatus will be 400 Mw_t power and 1100 primary coolant outlet temperature. Startup conditions of 350 Mw_t power and 750 F outlet temperature represent conservative initial conditions.

In general, closed loops will serve to reduce heat transfer requirements for the primary system since the closed loops will have separate secondary and tertiary heat exchange circuits.

On the other hand, open loops do not reduce the heat transfer load on the primary system; indeed they may increase the heat load if high power fuel tests are conducted.

3.4.2 Primary Coolant System

The primary heat transfer system uses molten sodium as coolant directly for the fuel core (via tubes) and directly or indirectly for the core reflector and shields. The reactor concept calls for three coolant processing modules connected in parallel to inlet and outlet plenums which in turn connect to coolant channels in the core components. The processing modules each consist of a pump, an intermediate heat exchanger, and a purity control and indicating unit. The modules will be sized adequately so any two modules may sustain the reactor indefinitely at a minimum but useful power level (about 300 Mw_t). The primary coolant system is contained by multiple vessels and jackets.

Primary coolant system components and piping will be austenitic stainless steel. Generally, the components will be fully replaceable and fully interchangeable. It is expected most piping connections may be completed by manual welding or joining; however, each piping unit will be

arranged for remote welding by specialized welding equipment in the event the pipe is inaccessible for manual work.

Primary System Pumps

The primary pumps are vertical shaft, single stage, centrifugal units with electric motor drive. Nominal rating values for a single pump are 12,000 gpm of sodium with a total head of about 150 psig. The pump will be a sump-type, immersed unit to allow pump removal (via airtight transport cask).

Since the pumps will be downstream of the intermediate heat exchangers, they will handle coolant at core inlet temperature. The pump barrel will serve as pressurized, level controlled chamber to collect entrained gas and to maintain a gas fluid interface at the pump shaft seals. The pump motor and flywheel will be mounted above the pump and isolated from the primary coolant by gas seals and shielding. Direct maintenance access above the main floor of the reactor building will be provided for pump drive equipment.

Intermediate Heat Exchangers

The intermediate heat exchangers provide for heat transfer between the primary coolant and the secondary coolant. The heat exchangers will be shell and tube types using a primary coolant shell side and a removable (vertical via airtight transport cask) tube bundle for the secondary coolant. Removal technique for the heat exchangers for replacement or repair will correspond to the primary pump removal technique.

3. 4. 3 Emergency Operation

In all cases of emergency operation, the reactor is assumed to scram and the primary circuit heat capacity is computed to be adequate for absorption of the initial decay heat release. If electrical power fails to the primary and secondary system pumps, the arrangement of the systems will allow sufficient natural (thermally induced) circulation in the primary and secondary circuits to dispose of the decay heat generated. Water or

air supply failure in the tertiary isolates the heat rejection to the primary and secondary circuits, and loss of coolant in the secondary circuit isolates heat rejection to the primary circuit; to meet both conditions, the primary coolant system will include a high temperature natural circulation loop adequately sized to allow decay heat rejection to the atmosphere via its own secondary heat transfer fluid.

3.5 Fuel Transfer and Handling

3.5.1 Driver Fuel Transfer

The reactor tube is the key element in the fuel charge-discharge concept for the FFTF. The tube functions at once as a receptacle for the fuel assembly, a guide tube for the fuel during charge and discharge, a locating guide for the fueling tool prior to discharge, a container for the fuel transfer pot, and an access port to the core.

To discharge a fuel assembly, the fueling machine is aligned first above the upper end of the reactor tube. After removing the tube cover flange and shielding plug the fuel transfer pot is placed in the tube. The fueling machine then extends a fueling tool down the tube to engage the fuel assembly for withdrawal. The fuel assembly is lifted from the core portion of the tube and placed in the fuel transfer pot. (Sodium coolant flooding is maintained over all components during this sequence.) The fuel transfer pot with its charge of spent fuel and sodium bath is now ready for discharge from the tube and up into the fueling machine. (Depending upon the intensity of decay heat release, supplementary cooling in addition to inert gas forced convection may be required for the pot during discharge. Supplementary cooling may be provided by a liquid metal heat exchange probe carried by the pot handling tool and inserted into the sodium melt in the pot.)

The fuel transfer pot next is withdrawn from the upper end of the reactor tube to the transfer tube of the fueling machine. Pot cooling is maintained during subsequent transfer operations by either forced gas convection or probe heat exchanger. After the transfer pot is removed from the reactor tube, both the reactor tube and the lower end of the

fueling machine transfer tubes are sealed and disengaged.

The next step in the transfer sequence is to move the fueling machine into alignment with the spent fuel storage port. The transfer pot is inserted through the port into a vacant station in the storage apparatus. Since the spent fuel storage vessel contains liquid sodium with appropriate heat removal equipment, individual transfer pot cooling is not required during the fuel "cooling" period. Final fuel discharge occurs after the "cooling" period by transfer of the pot to the cleaning, decontamination, inspection, and storage facilities outside of the reactor enclosure.

3. 5. 2 Test Fuel Transfer

Test fuel assemblies normally will not be permitted the long "cooling" period of the driver fuels. To facilitate early inspection and disassembly (or recharge to the reactor) the test fuel from open loops (and closed loops containing compatible coolants) will be discharged in the same sequence with the same transfer equipment as used for driver fuel except that the storage delay will be eliminated or reduced. Test fuel assemblies from closed loops with coolant and/or materials not compatible with sodium will be transferred out of the reactor by similar sequence as described for driver fuel using sealed fuel transfer pots and separate transfer apparatus mounted on the fueling machine.

3. 5. 3 Reactor Tube Installation and Removal

The same general procedures for fuel handling will be used to insert and remove reactor tubes. It is planned to use transport casks for the tubes (including test loop tubes); the casks will attach to ports on the containment vessel. Subsequent manipulation of the reactor tube from the cask into the core (and vice versa) will be performed by the fueling machine.

3. 6 Test Loops

3. 6. 1 General

The test loop function, of course, is the real purpose of the FFTF. In general, loops will be of "open" and "closed" types which respectively have been defined as loops using reactor primary coolant and loops using

other than reactor primary coolant. The various test requirements have been summarized previously in this study; however, it is important to recognize that any particular fuels testing experiment may involve design of a highly specialized loop which probably will not resemble any present prediction. Recognition of this uncertainty is implicit in the FFTF design concept wherein a wide range of choice is allowed for preferred mechanical, thermodynamic, and nuclear test parameters.

Most tests will be performed in open and closed loops which use reactor tubes of the same size and shape as the standard reactor tubes. When required, much larger loop tubes may be installed in the center and peripheral tube positions; it is expected that full scale fuel assemblies for large power reactors will be tested in the FFTF after initial development work on fuel components leads to testing of fuel assembly prototypes. Assembly sizes up to nine inches diameter by four feet long (fast fuel core zone) may be accommodated without drastic core modifications.

A large number of test positions will be provided in the tubed core of the FFTF. Since each tube may be instrumented for coolant flow and pressure, coolant and fuel temperature, mechanical performance, and fission product release or other damage criterion, then each tube potentially is an open loop. Indeed, complete monitoring of the driver fuel performance will furnish valuable fast fuel test data. Further, many fuel designs may be used as "driver" fuel depending upon the then current requirements for core specific power, neutron flux concentration and spectra, and coolant temperatures. However, it must be recognized that some open loops and most closed loops will be expensive, complex, and highly temperamental. Reliable reactor operation may not be obtained if large numbers of complex loops are operated simultaneously. Obviously, the number of such loops permitted in the FFTF must be limited to allow all test programs to proceed effectively, safely, and within available funds.

3.6.2 Loop Arrangement

Three "standard" reactor tube geometries are presented in the conceptual proposal. These types are shown on Fig.3.3, p. 3.29. Reactor tubes for "driver" fuel normally will be the single pass type. Open loop reactor tubes will also be of the single pass type. Closed loop reactor tubes may be of the U-tube or annular reentrant types. Supply piping connections to closed loops will be made above the neutron shielding in the reactor pipe gallery. Discharge piping for closed loop reactor tubes will be connected in the same gallery. Closed loop supply and discharge piping will extend out of the reactor vessel to remote containment enclosures for the primary coolant components of the individual test loops.

Open Loops

The standard open loop arrangement is an instrumented exposure test of nuclear fuel components installed in the standard reactor tube. Such a loop is a passive test in the sense that no test parameters can be varied independently for the loop during any run.

Closed Loops

The closed loop is the ultimate technique in fuels testing. It is a necessity for any nuclear fuels test which probes beyond normal operating limits into the range of probable failure. The closed loop is also essential for tests of materials and coolants differing from the reactor "standard". The closed loops will require separate, out-of-reactor coolant processing facilities for pumping, heat transfer, testing, coolant purity, and pressurizing, unless the closed loop consists of a complete, integral process tube assembly or "loopsule".

3.7 Instrumentation System

3.7.1 Introduction

The concept of the instrument system reflects the primary purpose of the reactor to provide a flexible high-flux fast-neutron test facility. To this end, the basic functions of the system are:

1. To provide control of the reactor and the experiments, including general surveillance of the Fast Flux Test Facility
2. To provide a complete and readily retrievable record of all data associated with the operation of the reactor and the experimental facilities, perform on-line computation to evaluate accuracy of data being generated, and perform data reduction and correlation. Extensive use of digital computation techniques will be required to meet this objective.

3.7.2 Reactor Control

A highly reliable safety circuit will be utilized to shut the reactor down whenever required. However, the primary safety protection of the reactor, as explained in more detail in Section 4.2, derives from its inherent strong negative temperature coefficient of reactivity. This negative coefficient will always arrest any surge of reactivity and give the safety system time to shut the reactor down safely.

In general, the FFTF safety circuit will be patterned after the latest best designs of safety circuits which have been developed at Hanford over the past 20 years. Wherever possible, components proven in sodium-reactor experience elsewhere will be employed. Safety circuit trips will include nuclear flux level and period and perhaps coolant flow, temperature and level, depending upon the results of physics and core mechanics calculations.

Reactor neutron flux level will be monitored by multiple chambers with signals shared by the panel-mounted instruments and the digital computer. The nuclear instrumentation system will provide interpretative information for operator control of the reactor through startup, steady-state operation, and shutdown. Reactor period will be determined from the various flux level signals. Positions of the reactor control and safety elements will be sensed, and will be displayed.

Reactor temperatures will be sensed, probably by resistance temperature detectors (RTD's), in the gamma shields, neutron shields, and control elements.

Instrumentation for the primary and secondary loops will be similar and includes the following:

- Magnetic flow meters are being considered for monitoring coolant flow. However, three other types of candidate liquid metal flow meters will be investigated before a final determination of the type of sensor is made. These include (1) turbine, (2) venturi, and (3) heater and thermocouple type instruments.
- Resistance temperature detectors appear to be superior to thermocouples for measuring coolant temperatures. A new development in RTD's at Hanford has potential for being superior to previous designs in this field. Testing is currently being initiated on prototypes of this new design.
- Coolant pressure will be monitored by extant instrumentation germane to the appropriate FFTF conditions. Either force-balance transducers or strain-gage sensors can be used for pressure sensing. Coolant level can be determined by resistance probes, calculated differential pressure, or perhaps even by magnetic floats.
- Fission product activity will be monitored in the primary loop continuously or periodically to sense results of fuel rupture. Fast rises in activity can be detected by gross counting or wide window pulse-height discrimination. More sensitive measurements may be provided by energy analysis in gas samples. Bulk coolant activity will be monitored at a remote location from the reactor core to improve the sensor environment. An intensive program is planned to develop a process channel sensor. If a reliable device is not developed by startup time, provision will be made for later installation of individual process tube sensors.
- The secondary coolant will be monitored for high radioactivity (gamma) to detect conditions indicative of heat exchanger leaks. Monitoring the secondary loop will be much easier than for the

primary loop and can be done with standard instrumentation.

- An intensive investigation is currently under way to determine the best means of monitoring primary coolant purity. Details of this study are covered in Section 4. 6. The target of this pursuit is in-line continuous monitoring of carbon and oxygen. Provision will be made for eventual installation of in-line continuous monitoring instruments even though sampled batch monitoring may be used at startup.
- Engineering effort will be required for the selection of the necessary instrumentation and transducers required for the primary and secondary loops to assure that the reactor is adequately and safety instrumented. The goal is to select transducers with high inherent reliability to assure continuity of reactor operation. Redundancy will be used where required for reactor safety considerations on critical parameters.

Tertiary coolant flow and temperature will be monitored by proven instrumentation. Heat exchanger integrity will be checked by monitoring for secondary loop reaction products in the tertiary loop. Almost all of the tertiary loop and miscellaneous instrumentation is standard and requires little or no development.

Instrumentation for closed loops will be somewhat similar to that for the open loops in order to standardize as much as practical for ease of maintenance. However, there is more latitude in locating certain sensors for the closed loops external to the reactor. Therefore, they will be more accessible.

Other instrumentation will be required to monitor and sometimes control such items as primary and secondary coolant purification, building radiation monitoring, vibration, gas purity and pressure, charging machine position, heating and ventilation, and others.

3.7.3 Data Recording, Computation, and Retrieval System

The basic function of the FFTF will be to provide reliable experimental data in an area of new technology. To this end complete records must be maintained of all reactor operating conditions and all test loop

conditions. The information from these records must be readily available so that experimenters can obtain on demand the full record of all conditions which existed during their experiments. Such information, covering not only average conditions but also transients, is critically necessary in evaluating and interpreting the results of experiments. In order to provide full data covering transients, the system will automatically record not only the transient conditions but also the pertinent detailed data preceding the transient.

In addition to recording and retrieving data, it is necessary for the data to be "reduced" and correlated promptly into useable form in order to guide the conduct of the experiment. Part of the required correlation must consist of computations to evaluate the status of the accuracy of the data sensing system, so errors in input data can be determined on line and corrected.

Due to a vast volume of data required, the rapidity which it must be recorded, and the complexity of computations and retrieval requirements, a digital computer data recording, computation, and retrieval system is required.

During the reporting period, further conceptual design and development work was carried out on the Data Recording, Computation and Retrieval System. The system will be based on the use of on-line digital computer equipment of standard manufacture to obviate computer development requirements. Information will be fed to the computer from all of the reactor, coolant, and closed and open loop instruments or sensors, and any other instrument which is important to the operation of the reactor.

Instrument and sensor inputs will utilize analog to digital converters, and multiplexers will be utilized to serialize the inputs from the array of data continuously offered to the computer.

Data will be stored on drums, tapes, and cores. The type of storage will be determined by the required volume of data and maximum access time permitted by the interrogating signal. Data logging will be done on typewriters, printers, magnetic tape, and high-speed tape punches.

Data display from the computers will be by means of about four cathode ray tubes (CRT). The CRT can generate at least three colors and can present readout data by alpha-numeric, graphic and traced curve displays. Any reactor, coolant, or test loop data and information can be displayed on the CRT to give immediate assistance to the test, project, or program at hand. Typical readouts include control element position, reactor period, power level, coolant flow, temperature, curves showing reactivity or flow trends, patterned displays indicating relative process tube temperatures by brightness, overall annunciation, and so on.

3.7.4 Automatic Process Control Concepts

In view of the fact that the Data Recording, Retrieval, and Computation System will be provided with all plant operating and experimental data, consideration is being given to expanding the system applications to include automatic process control functions. The rapid progress which has been made in the electronics and computer industry in recent years in automatic computer control is a strong indicator that many benefits can be derived from this type of control. Incentives for progressing in this direction accrue from the following worthwhile objectives:

1. Optimum performance of heating and cooling cycles and minimization of thermal shock both to the reactor and to the coolant equipment and appurtenances
2. Ability to maintain optimum conditions for a controlled and stable testing environment
3. Reduce the number of spurious and unnecessary scrams
4. Enhance plant safety by adding a scram trip controlled by the computer
5. Enhance plant safety and operability by maintaining more stringent surveillance of less critical systems.

Adoption of computer control of a process is usually an evolutionary development. The computerized control systems are normally initially installed in an "open-loop" configuration with the computer providing information to the operator. At a later date, as the computer demonstrates its capabilities, portions of the system (or the entire system) may be placed under direct computer control.

The applicability of these concepts to the FFTF is being analyzed in depth. In particular, the reliability of such systems for reactor control is being carefully evaluated. As described above, any initial applications could be "open-loop" with the operators retaining control of the process. (Sufficient analog type panel-mounted instrumentation will be installed so as to be able to start up and operate the reactor entirely by manual control with proven automatic trip protection. This instrumentation will be adequate both as to quantity and type to ensure ease of operation and complete safety.)

3.7.5 Physical Design of Instrumentation

Integrated terminal panels will be installed for each closed loop (and for the reactor) to provide wide flexibility in instrumenting and adding and changing wiring to match varying test requirements. Low level signal wiring, such as for nuclear flux monitors, will be well protected using techniques which have been highly successful in NPR. Likewise, radiation and flame retardant wires and cables will be used of the same general types which were developed and used successfully in NPR.

Overall instrumentation design will be guided by two main considerations:

1. Desire for a large amount of instrumentation throughout the plant to provide as complete a record of operating characteristics and test results as possible
2. The use of either standard equipment or required adaptations of standard equipment wherever possible to ensure good operability and to minimize cost.

A strong effort will be made to design the various types of instrumentation so that each instrument, module, or device can be replaced with a minimum of cost and effort. Thus, if some instrument becomes troublesome, or if a superior type becomes available, it can easily be changed out.

3.8 Irradiated Specimen Examination Facility

3.8.1 Purpose and Scope

An irradiated specimen examination facility will be included as part of the FFTF building. The primary purposes of the facility will be the nondestructive preliminary examination of irradiated fuel and materials specimens, and the disassembly of fuel assemblies for shipping to laboratories equipped for more comprehensive evaluations. In addition, the irradiated specimen examination facility may be used to a limited extent for repair of test specimens, and for some destructive testing when required.

This closecoupled irradiated specimen examination facility will permit early examination of specimens, and is one of the facilities required to permit removal, examination, and return of a fuel assembly to the reactor in a minimum time.

The examination facility will include cells both for underwater operations and for nonaqueous operations. Further details are given below.

3.8.2 Underwater Examination and Repair Stations

Several underwater examination and repair stations will be provided. The following operations may be performed at the underwater stations:

- Optical viewing (underwater periscopes)
- Length and diameter measurements
- Warp or camber measurements
- Ellipticity or ovality
- Weight measurements
- Photography
- Gamma scanning.

One examination station will provide equipment for cleaning, disassembly, repair, and assembly of fuel assemblies.

The underwater examination and repair stations will be constructed as cells within the fuel storage basin. The cells in which fuel assembly repair and welding operations are performed will be provided

with watertight access doors and pump-out facilities designed to minimize the spread of contamination within the basin. Common access will be provided to the examination cells at the operating floor level to permit maximum utilization of specialized examination equipment.

3.8.3 Dry Examination and Fuel Sectioning Stations

The facility will have one dry examination station where the fuel assemblies will be air- or gas-cooled. One fuel sectioning cell will be equipped for cleaning and sectionalizing or cutting up fuel assemblies while being cooled by a cooling media other than water (such as inert gas or inert transparent hydrocarbons).

The following operations may be performed in the dry examination station:

- Optical viewing
- Measure length, diameter, and warp
- Translate and rotate so that all exterior surfaces can be viewed
- Decontamination.

The dry examination station will be adjacent to the underwater examination facilities. Fuel assemblies may be transferred from the storage basin by means of a cask or by an underwater transfer system through a transfer lock utilizing pneumatic sealed doors in the common wall between the wet and dry cells. Access to the cell will be provided by removable cover blocks at the top and through the sidewall near the bottom of the cell.

The fuel sectioning cell will be located adjacent to the fuel cleaning cell. Fuel assemblies will be transported to the cell by means of the fuel handling cask. Equipment will be provided for cleaning fuel assemblies with agents other than water. The cell will be equipped to safely handle ruptured fuel assemblies.

3.9 Additional Examination Facilities

In addition to the rudimentary postirradiation examination facilities provided as part of the FFTF complex, a highly developed specimen examination and testing technology should be available to both the test reactor operator and experimenter. The types of facilities needed for the more

sophisticated postirradiation examinations are typified by a listing of the facilities available at Hanford. Facilities such as these should be available on or near the test reactor site to minimize the handling and transportation problems associated with irradiated specimens.

PRTR Fuel Examination Facility (Air-Cooled)

- a. Optical viewing (5X magnification)
- b. Length and diameter measurements
- c. Warp measurements
- d. Translate and rotate to view all exterior surfaces
- e. Photography
- f. Borescope examination
- g. Decontamination

PRTR Storage Basin (Existing)

- a. Underwater viewing
- b. Optical viewing
- c. Assembly and disassembly
- d. Gamma scanner

PRTR Fuel Examination Basin

Project Proposal CAH-119 (under construction)

- a. Underwater disassembly, repair, reassembly of the following types of fuel elements: 19-rod cluster, 7-rod cluster, 4-rod cluster, nested tubular, single rod, Fermi and inverted cluster type. Specific operations which may be performed on a depth-adjustable underwater work table are: horizontal or vertical positioning, rotation in either direction, milling, drilling, sawing, and tungsten inert gas welding.
- b. Optical viewing (5X magnification)

105-C Fuel Element Examination Facility

- a. Optical viewing
- b. Length and diameter measurements
- c. Warp or camber measurements
- d. Ellipticity or ovality

- e. Still photography with slit camera
- f. Borescope (with camera)
- g. Radiation intensity
- h. Tensile testing
- i. Weighing (in air or water)
- j. Decladding aluminum clad elements to take core measurements

327 Building - Radiometallurgy and 325 Building - Radiochemistry

- a. Optical viewing
- b. Photography
- c. Length and diameter measurements
- d. Warp measurements
- e. Metallography (In this operation metal samples are cut with a silicon carbide saw; ground, polished, and etched; and photographed at high magnification.)
 - (1) Microstructure
 - (2) Phase stability
 - (3) Grain structure
 - (4) Phase transformation
 - (5) Fission product diffusion
 - (6) Fuel-cladding-coolant interactions
 - (7) Cladding erosion
 - (8) Cladding corrosion
 - (9) Effects of fuel melting
- f. Hardness testing
- g. Tensile testing at room and elevated temperatures
- h. Fatigue and impact testing
- i. Dilatometry
- j. Density
- k. Annealing studies
- l. X-ray diffraction studies
- m. Electrical resistivity
- n. Chemical analysis for burnup
- o. Chemical stability studies

- p. Fission gas release
- q. Mass spectrographic gas analysis
- r. Fission product and corrosion product analysis

3.10 Reactivity Test Facility

The reactivity test facility will provide a test environment for obtaining reactivity and power density measurements on test fuels before insertion in the FFTR. A prediction of the heat production from a given test fuel in a given test position will ensure the most efficient use of test space, assure the test sponsor that the power density and gradient are known, and provide a means of making neutron spectrum index measurements for use in correlating the damage studies.

Information on the effects of sodium will be obtained in experiments using static sodium contained in removable containers. These experiments, together with other perturbation experiments, will be of great importance in assessing cold reactivity coefficients and various components of the temperature coefficients of the fuel assemblies to be irradiated in the FFTF.

It is proposed to incorporate the Reactivity Test Facility, shielding room, supporting services, and office space in the main FFTF building.

The flexibility will be such that the basic core arrangement may be changed quickly so that fast reactor tests may be accomplished to support the operation of the FFTF.

Auxiliaries will include a fuel storage vault, a mock-up room equipped to handle plutonium fuels, a control room, equipment and counting rooms, and office space for supporting personnel.

3.11 Fast Flux Test Facility Building

The fast flux test facility building will consist of three major parts: (1) the reactor hall, (2) the auxiliary equipment area, and (3) the administration and service area. The facility will have a ground floor area of approximately 60,000 sq ft and a total floor area of about 172,000 sq ft. A preliminary arrangement is shown in Figures 3.4 through 3.8 (pp. 3.30 through 3.34).

The reactor hall will be a reinforced concrete canyon type structure housing the reactor proper; the experimental (loop) cells; the specimen cleaning, handling, and storage facilities; the reactivity test facility; hot cells for hot equipment decontamination, maintenance, and storage; tube storage cells; and hot waste handling and storage facilities.

The auxiliary equipment area will join the reactor hall on one side. It will be a multi-story structure and will house the heat dump facilities, liquid metal handling and storage, the gas system, heating and ventilation equipment, reactivity test control room, storage basin cooling and clean-up equipment, analytical laboratory, an irradiated fuel storage vault, and other support facilities.

The administration area will be attached to the reactor hall on the opposite side and will house the control room, lunch room, offices, change room, electrical gear room, cable room, an auxiliary equipment room, and other facilities.

As shown, the building is arranged to separate areas of lower from areas of higher potential for radioactive contamination. Personnel entry to all potentially radioactive areas would be through change rooms where facilities for changing to protective clothing, radiation survey and personnel decontamination would be available.

The building structure in general will be of reinforced concrete construction with a limited use of structural steel and masonry construction. Building airtightness will be of prime importance for all areas. Bridge cranes will be provided for the reactor hall and the secondary heat exchanger building. The reactor hall crane may be equipped for remote operation. Light cranes, hoisting equipment, and monorails will be provided for shops and other specific handling needs.

The fuel handling area will provide an operating area at the main floor level with access to the fuel storage basin and access via removable

shielding plugs to the specimen examination facility, liquid waste handling cell, equipment decontamination and maintenance cells, tube storage, and reactivity test facility, which are all located below. Operations within the cells will be conducted remotely and viewed through lead glass shielded windows and optical devices from operating galleries at the lower levels. Contact maintenance will be used when radiation levels permit. New reactor fuel will be stored in a vault adjacent to the fuel handling area at the main floor level. Materials and equipment will be moved into the reactor hall by rail or truck via an air lock.

The heating and ventilating systems will be designed in accordance with Hanford practice for effective control of airborne radioactive contamination. Air flow will be from regions of lower to regions of higher potential for radioactive contamination. The potentially contaminated exhaust air will be cleaned of radioactive particles by passage through high efficiency filters or scrubbers and discharged to the atmosphere through an exhaust stack.

FIGURE 3.1
Core Concept

FIGURE 3. 2
Cylindrical Tubular Arrangement

FIGURE 3.3
Preliminary Study-Reactor Tube Types

FIGURE 3.5
Building Plan, Elevation 17'0"

FIGURE 3. 6
Building Plan, Elevation 34'0"

FIGURE 3. 7
Building Section

FIGURE 3.8
Building Section

3. 12 Engineering Test Facility

3. 12. 1 Purpose and Scope

The scope of engineering testing research and development activities in support of the FFTF was studied. It was tentatively concluded that much of the feasibility testing and the component testing can be adequately performed in existing AEC sodium research and development facilities. Integrated performance testing (systems testing) under proposed operating environments and procedures is required prior to startup to ensure systems reliability, and following startup to evaluate proposed in-reactor experiments, equipment modifications, and operating modes. These two capabilities (prestartup systems testing and operations support systems testing) will be provided in an Engineering Test Facility (ETF). The systems testing facility in the ETF will be a nonisothermal flow system of greater than fuel assembly flow and heating capabilities. Peripheral to the systems testing facility will be several smaller specialized supporting facilities for electrically-heated thermal-hydraulics studies, defected fuel testing, and small component development and evaluation. Details are given below.

3. 12. 2 Engineering Testing Requirements

Engineering testing programs in support of the design, construction, and operation of the FFTF must be performed in four distinct phases, namely feasibility testing, component testing, prestartup systems testing, and operations support systems testing. These four phases of work, as related to the FFTF Program, are discussed in the following paragraphs.

Feasibility Testing and Component Testing. Much of the feasibility testing and component testing for the FFTF can be performed in AEC Sodium Components Development Program facilities. The use of facilities such as the Sodium Components Test Installation and the planned Sodium Pump Test Facility is required to meet the proposed schedule for the FFTF design and construction.

Despite the extreme importance and necessity of feasibility testing and component testing, these two initial steps in the testing sequence do not provide adequate assurance of plant operability. Even when feasibility

testing and component testing are performed in an optimum manner, employing the most modern facilities and highly skilled personnel, the value of the results in an actual system is limited as described below.

Feasibility Testing. The purpose of feasibility testing is to establish the practicality of certain basic concepts of the reactor. Feasibility testing often takes the form of "quick-and-dirty" mockups, and is intended to determine as rapidly as possible that the underlying concepts of the reactor are practical, or can be engineered around. Feasibility testing must be performed very early in a reactor project to avoid substantial disruptions in schedules and subsequent cost increase. Emphasis is not generally placed on exact geometric similtude, operating parameters, materials of construction, and so on. Important as feasibility testing is to a project, it should not be considered a substitute for the more thorough testing required to ensure safety and acceptable operability of the facility.

Component Testing. The purpose of component testing is to develop and to evaluate performance of specific components, under environmental conditions which approximate plant operating conditions. Component testing follows feasibility testing as components are designed and fabricated or specified and procured. This phase of testing is generally manifested by several characteristics:

- a. Testing in a "limited environment" with a minimum of other prototypical equipment involved in the test.
- b. Testing with only nominal control over environmental conditions such as coolant quality, flow, temperature, thermal gradients, and so on.
- c. "Stop-and-start" operations, where components are tested for short periods, removed and modified, and committed for retest. Components may be under development for many months or years, yet it is not unusual that when the development is completed, the final product may have been tested only a few weeks or months at operating conditions.

Component behavior as it relates to other systems components is generally absent from this phase of R and D testing. Nevertheless, component testing is a necessary requisite to the next step in engineering research and development systems testing.

Systems Testing. To assure proper operation of the FFTF initially, and to provide continuing assurance that satisfactory long-term test-reactor operation can be achieved with a variety of test loadings and closed loop cooling systems, it is necessary for extensive systems testing to be performed. This systems testing will take place (a) before initial startup, and (b) after startup to verify the operational practicability and safety of proposed tests and related specified test operating conditions.

Systems testing for the FFTF will be performed in the Engineering Test Facility.

3. 12. 3 Engineering Test Facility Application

Systems Testing

The prime functions of the Engineering Test Facility (ETF) will be systems testing of FTR systems prior to startup, and operations support systems testing following startup. Prestartup systems testing will be performed under prototypical conditions with fully controlled environments to ensure reliable, integrated operation of reactor systems. Problems which may arise can be corrected before reactor startup. Operations support systems testing is required because the FFTF is a test reactor, designed to explore unknown regions of technology; for assurance of reliable operation under these extreme conditions it is necessary to carry out extensive out-of-reactor systems tests of proposed fuels and test loop equipment.

Prestartup systems testing, frequently slighted in many reactor research and development programs, is the key to satisfactory reactor startup and operation. It includes system evaluation and component interaction under prototypical operating regimes (temperatures, flows, pressures, coolant quality, thermal gradients, and proposed operating and maintenance modes), and under possible off-standard conditions. Since component performance can be affected by other components in the system (for instance,

component performance in sodium systems can be significantly affected by carbon mass transport arising from the presence of dissimilar materials), systems testing requires the combined functional and performance testing of core components (fuel, sodium service hardware, control assemblies), control components (for coolant conditions and quality control), and out-of-reactor components under projected standard and off-standard operating conditions.

Operations support systems testing provides operational support for the test reactor subsequent to startup to evaluate proposed in-reactor experiments, new modes of operation required for the tests, and related equipment modifications. Unique experiments can be evaluated prior to commitment to the reactor environs. Failure to provide this capability for a test reactor exploring new technology would seriously compromise reactor operating efficiency and safety.

ETF Systems Testing Capabilities

As presently conceived, the "systems" testing portion of the ETF will have flow, heating, and cooling capabilities of 4000 gpm and 14.6 Mw. The facility will contain a "subassembly" mockup of a reactor tube and its attachments and appurtenances, as well as a significant portion of the core so that the unique features of the FTR system can be adequately developed and tested. The facility will have capabilities to test at operating temperatures the hydraulic characteristics and performance of fuel elements, test assemblies, control assemblies, instruments, and other equipment. The subassembly and core mockup will be located in a physical environment similar to the reactor so that the critical equipment, piping, and jumper loops, etc., can be installed, tested, and maintained in prototypical operating conditions. This systems facility will be constructed to function as an operational tool for proof-testing advanced testing loops and test specimens prior to irradiation in the reactor core.

Other ETF Functions

The ETF will also contain facilities for basic engineering support in specialized areas which are fundamental to, or characteristic of the Fast Test Reactor, and where existing facilities and experience have not been sufficiently developed. These include (a) heat transfer testing capabilities for full-scale, internally heated, simulated fuel elements, and (b) fuel washout studies with irradiated defected fuel elements. Because the systems test facility should not be used for "start and stop" type development, two other peripheral loops for development and testing will be provided. The need for the latter two facilities is based on previous experience that shows (a) a number of integral parts will require last-minute testing prior to installation, and (b) a certain few integral parts will require unusually close design-development testing.

These facilities are in the 50 to 100 gpm flow range and 1400 to 1500 F operating temperature, and will be used to provide basic engineering data to support the operation of the systems facility and for FTR design, construction, and operational support. They will be "packaged" types in their own enclosure. The systems facility and the peripheral facilities will use common services wherever feasible, including the systems for sodium unloading, component cleaning, inert gas, and so on.

Long Term Applications

While continuing direct support of FTR operation is the prime anticipated use for the ETF facilities following startup, they will also find use in the AEC's growing liquid metal, fast breeder program of the 1970's and beyond, much as existing facilities will play a vital role in the immediate efforts for the successful implementation of the FTR.

4.0 SUPPORTING TECHNOLOGY

4.1 Introduction

Design of the FFTF will be based, to the maximum extent possible, on the use of equipment and technology which has already been proven in sodium-cooled reactors. However, there are a number of areas of technology which will require unique application in the FFTF program. During the reporting period information was obtained in a number of the more important areas of technology, including physics, materials, analytical chemistry requirements, driver fuels and project management.

4.2 Physics of the Fast Test Reactor (FTR)

In the following section the results of some recent neutronic calculations for the FTR are summarized. This work is an extension of that reported in HW-81940.* The present work deals primarily with reactor statics and control. Critical sizes and reflector control worth for a wide range of reactor geometries and compositions are presented. In addition, estimates of sodium coefficients and core expansion coefficients are also included.

Since the FTR design is in a very early stage, neutronic analyses are by no means complete. In fact, many important topics, particularly in the area of reactor kinetics and stability, have as yet not received sufficient attention.

4.2.1 Reactor Statics

Reactor Model and Method of Calculation

The critical size (determined at a k_{eff} of 1.05) of various Fast Test Reactor configurations was computed as a function of the core length to diameter ratio (L/D) and the fuel volume fraction. For computational purposes, the FTR was assumed to consist of a homogeneous, cylindrical

* J. J. Regimbal and R. E. Peterson. Nuclear Design Calculations for a Fast Fuel Test Reactor, HW-81940. April 21, 1964.

4.2

core surrounded by a 25.4 cm Inconel radial reflector. A one dimensional diffusion theory code (HFN) using 16 energy groups* was employed for most criticality calculations.

The axial neutron leakage was simulated by a constant buckling expression with a reflector savings of 11 cm (22 cm total).

Results of Criticality Calculations

Figures 4.1 and 4.2 give the critical core volume and diameter as a function of fuel loading for various core length to diameter ratios. All pertinent input information is listed in Table 4.1. Figures 4.3 and 4.4 are simply cross plots of Figure 4.1. Perhaps the most significant conclusion that can be drawn from these graphs is that the critical core size is rather insensitive to small changes in L/D. This insensitivity is due, in part, to the relatively thick axial and radial reflectors.

TABLE 4.1

NUMERICAL DATA FOR PHYSICS CALCULATIONS

	<u>Molecular Weight</u>	<u>ρ</u>	<u>Molecules/cc x 10^{-24}</u>	
PuO ₂ (93 a/o 239, 7 a/o 240)	271.12	11.46	0.0254587	
Na	22.99	0.872	0.022844958	
304 SS (26 a/o Ni, 74 a/o Fe)	56.59	7.92	0.02191653	Ni
			0.0623778	Fe
Inconel (94 a/o Ni, 6 a/o Fe)	56.019	7.449	0.07434	Ni
			0.00575	Fe
BeO	25.0116	2.86	0.0688711	

ATOM CONCENTRATIONS FOR VARIOUS LOADINGS (40 v/o Na, v/o SS = 50 - v/o Fuel)

5 v/o PuO ₂	1.18435-3	8.91450-5	2.54699-3	1.14225-2	2.86817-2	1.00773-2
7 v/o PuO ₂	1.65809-3	1.24803-4	3.56579-3	1.14225-2	2.74070-2	9.62946-3
9 v/o PuO ₂	2.13183-3	1.60461-4	4.58459-3	1.14225-2	2.61322-2	9.18158-3
11 v/o PuO ₂	2.60558-3	1.96119-4	5.60339-3	1.14225-2	2.48575-2	8.73370-3

A few check calculations of the diffusion theory results have been made using LASL's DTK transport theory code. In general, the transport code gives a multiplication constant 2 to 3% higher than the diffusion theory model.

* G. E. Hansen, et al. Los Alamos Group-Averaged Cross Sections, LAMS-2941. July, 1963.

FIGURE 4.1

Effect of Fuel Density on Core Volume

FIGURE 4.2

Effect of Fuel Density on Core Diameter

FIGURE 4.3

Effect of Geometry on Core Volume

FIGURE 4.4

Plutonium Inventory

Radial and Axial Neutron Distributions

In order to obtain some estimate of the neutron distribution in a typical FTR configuration, the radial and axial neutron source rate was calculated in a reactor with 9.5 v/o PuO_2 , 50 v/o Na, and 40.5 v/o stainless steel. The results are shown in Figures 4.5 and 4.6.

Note that although the spectrum is slightly more thermal at the core-reflector interface than in the core center no significant thermal spike is present at the interface. Furthermore, the axial source distribution can be represented by a cosine with a reflector savings of about 11 cm.

Neutron Spectra and Average Cross Sections

The neutron flux spectrum in the center of the core is shown in Figure 4.7 for 5, 7, 9, and 11 v/o PuO_2 cores. The corresponding spectrum weighted cross sections are listed in Table 4.2. All calculations were performed using the 16 group Hansen-Roach cross section set and the densities tabulated in Table 4.1.

TABLE 4.2
SPECTRUM AVERAGED CROSS SECTIONS
FOR VARIOUS FUEL LOADINGS*

<u>Cross Section</u>		<u>v/o PuO_2</u>			
		<u>5</u>	<u>7</u>	<u>9</u>	<u>11</u>
Pu^{239}	σ_a	2.38	2.20	2.11	2.06
	σ_f	2.01	1.92	1.87	1.85
Pu^{240}	σ_a	0.918	0.877	0.883	0.901
	σ_f	0.467	0.544	0.605	0.654
Fe	σ_a	0.00979	0.00936	0.00912	0.00895
Ni	σ_a	0.0153	0.0139	0.0131	0.0126

* v/o Na = 50

v/o SS = 50 - v/o Fuel

FIGURE 4.5

Radial Fission Neutron Source Distribution

FIGURE 4.6

Axial Fission Neutron Source Distribution

FIGURE 4.7

Flux Spectrum for Various Fuel Loadings

As one would expect, the spectrum becomes harder as the fuel volume fraction is increased. This hardening causes the fissile fuel and structural material cross sections to decrease, and the fertile fuel fission cross section to increase. Note that the FTR spectra are slightly harder than those found in most fast breeder reactors. For 5 v/o PuO_2 , more than .7 of the total flux is above .1 MeV; for 9 v/o PuO_2 , .8 of the flux is over .1 MeV.

Effect of BeO Addition

The unmoderated FTR spectrum will probably be slightly harder than the spectrum in first generation fast power reactors. In order to achieve a better spectrum match, or to increase specific power, a small amount of BeO could be added to the FTR core. For the reactor sizes and compositions of interest, the addition of BeO to the core will increase reactivity. In fact, BeO has a quite large positive reactivity value for a wide range of fuel volume fractions.

The effect of BeO on the spectrum in the center of the core is shown in Figure 4.8 for a reactor with 9 v/o PuO_2 , 41 v/o SS, and 30 v/o Na. Table 4.3 gives the corresponding variation of spectrum averaged microscopic cross sections. Observe that the specific power for a given flux, or σ_f , increases by a factor of about 1.25 in going from 0 to 20 v/o BeO.

TABLE 4.3
SPECTRUM AVERAGED CROSS SECTIONS*

Cross Section		v/o BeO				
		0	2.5	5	10	20
Pu^{239}	σ_a	2.06	2.14	2.21	2.38	2.71
	σ_f	1.85	1.89	1.93	2.02	2.21
Pu^{240}	σ_a	0.891	0.900	0.918	0.976	1.16
	σ_f	0.639	0.612	0.588	0.551	0.500
Fe	σ_a	0.00903	0.00913	0.00923	0.00949	0.0102
Ni	σ_a	0.0126	0.0133	0.0140	0.0153	0.0178

* 9 v/o PuO_2 , 30 v/o Na, 41 v/o SS

FIGURE 4.8

Effect of BeO on Central Core Spectrum

4. 2. 2. Analysis of Reflector Control

Introduction

Reflector control of the FTR could have some design advantages. In this section, the reactivity swing achievable by varying the reflector albedo is estimated for various core and reflector configurations. Because the reflector worth is rather sensitive to the flux behavior at the core-reflector interface, the DTK transport code with 16 energy groups was employed for all control calculations.

Total Reactivity Worth of Reflector

The total amount of reactivity tied up in the radial reflector is quite large. This is demonstrated in Figure 4. 9 for reactors with 7 and 9 v/o PuO_2 . In practice, it is difficult to design a control system which utilizes an appreciable fraction of the total reflector worth. Two possible reflector control designs have been proposed. One design is based on actual reflector removal, while the other relies on the motion of reflector poison elements (rotating control drums or translating control elements). Both of these systems are considered in the following sections.

Control by Reflector Removal

The removal of part or all of the radial reflector permits more neutrons to leak out of the reactor, which reduces reactivity. For structural and mechanical reasons, it is impractical to consider dropping the entire radial reflector. However, sizeable reactivity variations can be achieved by removing the inner portion of the reflector. Figure 4. 10 shows the control span achievable by discarding the entire 25. 4 cm reflector as compared to removing only the inside 15 cm. For this comparison, an infinitely long reactor was assumed.

The effect of dropping the inside portion of the reflector can be enhanced by placing an absorbing material in the outside portion of the reflector. This procedure reduces the backscattering from the stationary part of the reflector. Figure 4. 11 shows the effect of surrounding a 15 cm movable nickel reflector with 10. 4 cm of B_4C . Although the gain in control span is obviously not startling, it is significant.

FIGURE 4.9

Worth of Nickel Reflector

FIGURE 4.10

Effect of Backscattering on Reflector Worth

FIGURE 4.11

Effect of Reflector Composition on Control Span

Figure 4.11 also demonstrates the effect of replacing nickel by sodium in the movable portion of the reflector. This exchange would occur if sodium were to flood the void left by removing the nickel reflector. Such flooding reduces the control effectiveness by roughly a factor of 2 for the case considered.

Control Using Poison Drums

The albedo of the reflector can also be regulated by rotating asymmetrical poison drums in the reflector. As the absorbing side of the drum moves toward the core, the reactivity decreases because more neutrons are captured by the poison. Although this method of control is mechanically straightforward, it is rather difficult to simulate by mathematical models. An adequate representation would require a 3-dimensional transport code with many energy groups. In order to make the problem tractable, it was assumed, optimistically, that the control drums are equivalent to a thin annulus of poison material. The reactivity worth of a set of control drums can then be estimated by computing the reactivity as a function of the position of the poison annulus.

Figure 4.12 shows the reactivity as a function of the position of a 1 cm B_4C annulus for reactors with 7 and 11 v/o PuO_2 . Note that even under the above quite optimistic assumptions, the estimated control drum worth is less than 10%. Moreover, a small beryllium ring, added to enhance the poison worth, has essentially no effect on the control span. The principal reason for the small control swing is that the neutron spectrum, even in the reflector, is quite hard, hence most of the neutrons pass through the B_4C poison.

The error introduced by using a homogeneous annulus in place of discrete drums depends on many factors. If many tightly spaced drums are employed, the poison annulus approximation should be permissible. On the other hand, the approximation will yield very poor results if the drums are spaced widely apart. Drum worth estimates from other studies indicate that the achievable control span is roughly proportional to the fraction of arc subtended by the poison drums.

FIGURE 4.12

Reactivity Swing Achievable Using B_4C Annulus

Discussion of Results

It is clear from the preceding graphs that reflector removal offers, in principle, a greater control span than reflector poison control. Furthermore, the reflector removal calculations are probably more realistic than the poison annulus computations. Since a large control capability is highly desirable in a fuel test reactor, control by reflector removal appears to be more advantageous than control by drum rotation.

When very large control margins must be compensated for, reflector control alone may be marginal. For these conditions, supplementary in-core control is being planned.

4.2.3 Sodium Loss and Volume Expansion Reactivity Coefficients

Sodium Reactivity Coefficient

In general, a decrease in sodium density in a fast reactor causes:

1. A reactivity increase because fewer neutrons are captured in sodium
2. A reactivity decrease because more neutrons leak out of the reactor
3. A reactivity increase due to the harder spectrum and consequently higher source to capture ratio ($\nu\Sigma_f/\Sigma_a$).

In most cases, the capture in sodium is negligible. Large, low leakage, fast breeder reactors often have a positive sodium void coefficient owing to spectrum hardening. The high leakage FTR, in contrast with large breeder reactors, has a negative sodium void coefficient because the effect of leakage variation predominates over the effect of spectrum variation. This argument is demonstrated in Table 4.4 for a FTR with 9.5 v/o PuO_2 , 50 v/o sodium, and 40.5 v/o stainless steel. Observe that almost all neutrons either leak out of the reactor or get captured in Pu^{239} .

The effect of sodium voiding on reactivity is given in Table 4.5 for various fuel volume fractions. For all cases, sodium is assumed to initially occupy 50 v/o of the core. Stainless steel is assumed to have a void fraction of .5 minus the volume fraction of fuel.

TABLE 4.4

NEUTRON BALANCE IN A CORE WITH 9.5 v/o PuO₂*

	Na at Full Density		Na at 1/2 Density	
	Source	Sink	Source	Sink
Pu ²³⁹	97.22	36.86	88.81	32.97
Pu ²⁴⁰	2.78	1.17	2.86	1.12
O		0.08		0.08
Na		0.19		0.09
Ni		1.79		1.63
Fe		0.91		0.79
Leakage		59.00		63.32
	100.00	100.00	91.67	100.00

* 50 v/o Na, 4.5 v/o SS

TABLE 4.5

EFFECT OF SODIUM LOSS ON REACTIVITY*

v/o PuO ₂	δk for Complete Sodium Loss in Core	
	Fundamental Mode	DTK Transport Code
5	-0.184	
7	-0.177	-0.0925 (D = 80, L = 80)
9	-0.171	
9.5	-0.169	-0.06377 (D = 54, L = 90)
11	-0.166	

* 50 v/o Na, (.5 - v/o Fuel) v/o SS

It is clear from Table 4.5 that a sodium loss causes an appreciable decrease in reactivity for all cases considered. Moreover, since Pu²³⁹ has a small delayed neutron fraction, the reactivity change in dollars is enormous. For example, a reactivity change of .09 corresponds to about \$40.00.

The fundamental mode approximation greatly overestimates the core sodium coefficient because it does not properly consider the reflector. If

sodium were removed from both the core and the reflector, the reactivity change would be roughly that given by the fundamental mode approximation.

Although it is not shown here, it was found that the effect of sodium removal is almost exactly a linear function of the amount removed. In other words, if only one-tenth of the sodium is removed from the core, the reactivity change would be one-tenth of that given in Table 4.5.

Volume Expansion Coefficient

The reactivity change caused by homogeneous expansion of the reactor core was estimated using the one-dimensional DTK transport code. For a given volumetric core expansion, two expansion coefficients were calculated.

1. Radial expansion only, i.e., $\frac{\delta V}{V} = 2\frac{\delta R}{R}$. This computation was performed in cylindrical geometry with the axial leakage approximated by an energy independent buckling. The density of the 25.4 cm Inconel radial reflector was not varied.
2. Axial expansion only, i.e., $\frac{\delta V}{V} \sim \frac{\delta L}{L}$. Slab geometry was employed, with the radial leakage simulated by a constant buckling. The density of the 30 cm axial reflector, consisting of 20 v/o SS and 80 v/o sodium, was not altered.

Table 4.6 demonstrates the effect of volume expansion in a core composed of 9.5 v/o PuO_2 , 50 v/o sodium, and 40.5 v/o stainless steel. The axial expansion coefficient is slightly greater in magnitude than the radial expansion coefficient. The difference stems from the fact that the radial leakage is greater than the axial leakage ($D = 54$, $L = 90$).

TABLE 4.6

EFFECT OF VOLUME EXPANSION ON REACTIVITY

$\delta V/V$	Radial δk	Expansion $\delta k (\beta = 0.0021)$	Axial δk	Expansion $\delta k (\beta = 0.0021)$
.01	-0.00478	-\$2.28	-0.00766	-\$3.66
.02	-0.00966	-\$4.60	-0.0154	-\$7.34

It should be emphasized at this point that the assumption of a homogeneous volume expansion may not give a true picture of the actual reactor behavior. In particular, if the individual fuel pins expand, but the entire reactor does not expand, the coefficient shown in Table 4.6 would be totally misleading. However, the reactivity change for this situation could be estimated from the sodium coefficients given in Section 4.2.3.

4.3 Materials Selection Considerations for FFTF

4.3.1 Introduction

In the design of the FFTF, maximum application will be made of proven equipment and technology. However, in some parts of the design, particularly in the closed loops, extrapolation of known technology will be required. The discussion in this section summarizes available pertinent information on candidate metallic materials, points out guidelines for materials selection for severe FFTF service conditions, and outlines recommendations for practices to be employed during procurement of FFTF components.

4.3.2 Applications of Materials

Current technology involves the use of austenitic stainless steels and nickel-base alloys in liquid sodium at temperatures ranging from 450 to 1000 F. These operating conditions are not considered stringent for readily available, commercial alloys. However, marked reductions in the stress-rupture properties of these materials have been observed at approximately 1100 F after irradiations near that temperature.

Potential materials for sodium up to the 1200 -1400 F temperature region such as for FFTF closed loops include 300 and 400 series stainless steels, nickel alloys such as Inconel or Hastelloy, cobalt base alloys such as Haynes 25, and refractory metals such as columbium, molybdenum and tungsten. Such candidate materials would be expected to have acceptable resistance to sodium corrosion, although there are far fewer data on which to base this judgment than is the case for the parallel judgment relating to austenitic stainless steels in the neighborhood of 1000 F.

While these materials all find application in liquid sodium systems they are not necessarily compatible with each other in the same system. Although it would be an admirable goal to have a whole reactor built of one material, this is rarely possible in practice. Each new material introduced

complicates the successful corrosion performance of the reactor and perhaps compromises the fuel element test results obtained. Thus all multi-material systems must be carefully screened by laboratory testing to assure mutual compatibility.

In general, radiation is not thought to affect liquid metal transport processes. However, actual data are quite limited especially transport data at FFTF flux rates of 10^{16} nv. There are several mechanisms by which sodium might become more corrosive under the unusually high fluxes of high-energy neutrons in the FFTF.

The transport and deposition of radioactive species is potentially a serious problem and must be factored into materials choices. For example the cobalt base Haynes 25 which has been used out-of-reactor in sodium up to 1850 F might prove unacceptable because of Co^{60} deposition.

In the FFTF (and in other fast reactors) the effect of (n, α) and (n, p) reactions on the properties of materials in fast neutron environments will be substantial and potentially limiting to component service life, and this factor must be carefully evaluated in materials selection and plant design.

Austenitic Steels

Austenitic stainless steels have been employed almost exclusively in the present fast reactor primary systems. These steels also were specified in the four, one-thousand Mw Fast Reactor Studies.* The high temperature properties of the austenitic series are greatly dependent upon grain size and composition. Normal heat-to-heat variability of both grain size and composition in austenitic stainless steels often causes a broader range of properties than is practically realized in selecting one composition in preference to another. For example, a relatively clean heat of 304 stainless steel with well-controlled, postfabrication heat treatment may exhibit better resistance to creep than a poorly controlled heat of 347 or 316 stainless steel. Adequate specifications, quality control, and inspection are required for the FFTF.

Nickel-Base Alloys

Two general classes of nickel-base alloys have application for fast breeder reactors: solution hardened alloys, such as Inconel 600 and Hastelloy-X, and precipitation-hardened alloys, such as Inconel X-750,

*See Page 3. 1.

Inconel 718, and Hastelloy R-235. Problems associated with these alloys include cracking during fabrication and welding, improper heat treatment for the intended application, and embrittlement and loss of strength during service. The resistance of stressed-components to failure is highly sensitive to service environment as well as metallurgical structure and grain size.

The melting practice used to produce nickel-base alloys has direct influence on ductility at elevated temperatures. Vacuum melting is recommended to eliminate low melting-point impurities such as tin, lead, and bismuth and gaseous impurities such as oxygen and nitrogen.

Preconditioning Treatments

Investigations under way at Battelle-Northwest, Naval Research Laboratories, and Oak Ridge National Laboratories have demonstrated that preconditioning thermal-mechanical treatment tailored for reactor service conditions can markedly improve the properties and irradiation stability of reactor structural materials. This approach to materials development has the economic advantage of requiring only minor modifications to existing commercial alloy compositions and fabrication and heat treatment schedules, and this approach should be explored for the FFTF.

Procurement Specifications

Experience gained at Battelle-Northwest in procuring materials for the USAEC Irradiation Effects to Reactor Structural Materials Program has shown that representative heats of commercially available alloys exhibit marked variability in chemical composition, grain size, homogeneity of structural constituents and mechanical properties. Consequently, mechanical properties established on small quantities of materials employed for laboratory studies may not be representative of the mechanical behavior of components fabricated from large ingots and representing plant fabrication schedules. Normal heat-to-heat variations in properties may exceed variations caused by a service variable. These problems must be considered in writing procurement specifications. Substantial improvements in service life and design limits can be obtained for the FFTF by closely monitored inspection and quality-control procedures.

4. 4 Cermet Fuel Development

A cermet fuel, PuO_2 dispersed in a stainless steel matrix, is being considered as the driver fuel for the FFTF.

4. 4. 1 Methods for Fabricating Cermets

Three methods for fabricating cermets--extrusion, impaction, and cold-pressing--are being investigated.

Coextrusion of Cermet Pins

Coextrusion is probably the simplest way to make integrally clad, high density cermet fuel pins and is currently the preferred method for fabricating FFTF driver fuel. Potentially, integrally clad and capped small diameter fuel pins can be extruded to any desired length.

Results have been very encouraging. Stainless steel-20 v/o PuO_2 fuel pins clad in stainless steel have been co-extruded to the approximate dimensions specified for FFTF driver fuel pins. These pins were extruded from high density billets obtained by pneumatic impaction. Recently co-extrusions of stainless steel-20 v/o UO_2 fuel pins were made to determine the effect of cermet density in the billet on the quality of extruded cermet pins. Preliminary results indicate that low density billets (82% TD) may produce better extruded cermet pins than high density billets (97. 5% TD). The extruded core density (93 to 95% of theoretical) from the low density billet was about the same as obtained from the high density billet. Cermet pellets for low density billets can be made by cold-pressing, thereby eliminating the pneumatic impaction step used to obtain high density billets. Other variables being studied related to the extrusion process are ceramic distribution, bond integrity, clad thickness variations, and postfabrication condition of the ceramic phase.

Extrusion studies are in progress to more conclusively establish the effect of billet density on the quality of extruded cermet fuel pins. Studies are also underway to establish extrusion conditions for producing cermet fuel pins to FFTF specifications.

Impaction of Cermet Pins

Pneumatic impaction is being used to produce integrally-clad and capped cermet pins. These pins, about 3 to 4 inches long, could be welded

end for end to produce any desired length fuel pin. This process is being considered only as a backup to the extrusion process. An advantage of cermet made by impaction is the relative absence of stringing. Initial evaluation of the process will consider pin straightness and eccentricity, uniformity of clad thickness, core-to-clad and core-to-cap bonding, and core density.

An impaction mandrel designed to produce nineteen pins per impaction was used to make the first set of pins and these pins are currently undergoing examination. Although no measurements have been made, a few of the pins have been visually examined. With the exception of a slight flaring at the ends, these pins appear to be straight with no gross diametrical variations.

Because a large number of pins can be made with each impaction, this process offers an excellent method for studying the effects of different fuel and matrix materials (and variations in the properties of these materials) on the irradiation behavior of cermets. Mandrels can be designed that are capable of producing as many as 30 to 35 pins per impaction, thereby providing cermet specimens produced under identical fabrication conditions.

Cold-Pressed Pellets

Cermet pellets, 81 to 83% of theoretical density, can be routinely produced by cold-pressing at 70 psi. If low density cermets are satisfactory for fast reactor fuel, then cold-pressing is probably the simplest way of making such cermets.

Low density cermets have been heated to 900 C under vacuum for 1000 hours with only a negligible decrease ($< 1.0\%$) in density occurring.

High density cermets (97.5% TD) tested at the same conditions for 409 hours decreased in density 2.18%. Apparently the porosity of the low density cermets accommodates any swelling which occurs. Low density and high density cermets are being irradiated in the MTR. The first capsules of each type have been discharged after attaining about 0.8×10^{20} fissions/cm³. These capsules are to be returned to Hanford for postirradiation examination. Other capsules are being irradiated to higher burnups.

4. 4. 2 Preparation and Characterization of PuO₂

PuO₂ properties of primary interest for cermets are particle size and shape, particle density, and stoichiometry.

Particle Size and Shape

The desired ceramic particle size for cermets is generally considered to be in the range of -60+150 mesh (250 to 100 microns). This size range is a good compromise for obtaining fission fragment retention and cermet integrity.

To obtain the desired particle sizes, -325 mesh PuO_2 powder produced by controlled, low-temperature burning of plutonium metal has been pneumatically impacted and then crushed to size. More recently, PuO_2 of the desired particle sizes has been produced "directly" by high-temperature burning of the metal. With this "high-fired" PuO_2 , the pneumatic impaction step can probably be eliminated.

Particle density of high-fired PuO_2 is about 95% of theoretical. The oxygen-to-plutonium ratio is 2.00. Gas content is 0.133 cc/gm. About 75% of the particles are greater than 100 mesh.

Ideally, spherical (ceramic) particles are the most desirable for cermets; spherical particles may not be necessary for the high integrity cermets produced by impaction and extrusion. Angular particles have been used in most of the work done to date. Spherical PuO_2 particles, produced by Mound Laboratory, are to be evaluated for cermet use. All the cermet fabrication studies performed to date have been conducted with uncoated particles. With rolling or pressing processes, coated particles apparently enhance the quality of the cermet. With the high integrity cermet that can be obtained by impaction, coated particles may not be necessary. The distribution of the PuO_2 in the impacted cermets, while not ideal, may be satisfactory.

Particle Density

There are opposing ideas as to whether high density or low density ceramic particles are most desirable for cermets. High density particles are thought to be good because of their strength and ability to retain fission gases. Low density particles are considered desirable because void space is available in the particle to accommodate fission product buildup. Currently, very little effort is being applied to determining optimum particle density. PuO_2 particles used in current studies have densities ranging from 95 to 99% of theoretical.

PuO₂ Stoichiometry

Cermet studies to date have used stoichiometric plutonium dioxide. β -Pu₂O₃ has been prepared for cermet use to determine if the sub-oxide improves cermet integrity.

4. 4. 3 Matrix Material

Currently the matrix material of interest is stainless steel. Cermets fabricated to date have all used 304 stainless steel. Because of its higher strength at high temperature, 316L stainless steel will also be tested as a matrix material and projected studies will include other alloys.

4. 4. 4 Irradiation Tests

Irradiation tests underway and planned include capsule irradiations in the MTR and ETR, fuel pin irradiation in the ETR P-7 Loop, and fuel pin irradiations in the EBR-II.

Capsule irradiations are being used primarily to screen the various fabrication processes under study. High density and low density cermets are currently being irradiated in the MTR.

Fuel pins, approximately 1/4 in. diameter x 12 in. long, are scheduled for ETR P-7 Loop irradiation. The following types of cermets are to be irradiated.

1. Extruded PuO₂-SS
2. Impacted PuO₂-SS
3. Cold-pressed PuO₂-SS
4. Cold-pressed β -Pu₂O₃-SS
5. Extruded U²³⁵O₂-SS
6. Impacted U²³⁵O₂-SS
7. Cold-pressed U²³⁵O₂-SS

About six cermet fuel pins are to be irradiated in the EBR-II as a joint effort with Argonne National Laboratory.

4. 5 Sodium Technology

4. 5. 1 Corrosion and Coolant Purity Control

Corrosion

In liquid metal systems, corrosion and coolant impurities are closely related since impurities often increase corrosion. The successful corrosion performance of most alloys in sodium depends on elimination and control of oxygen and carbon in the sodium. Facilities must be included in a design to cold trap and hot trap the sodium to maintain desired purity.

It is well known that corrosion of steels by sodium is aggravated by oxygen contamination, which promotes attack and mass transfer in a polythermal loop. While the mechanisms of these effects are still the subject of much study and controversy, the problem has been minimized in practical systems by using a bypass cold trap to keep the oxygen in the sodium at about 25 ppm or less.

Another potentially undesirable effect is the transfer of carbon by sodium to the austenitic stainless steel. Bulk carburization embrittles the steel and reduces its ultimate strength. If there is an adequate source of carbon available to the sodium, the loop walls will be carburized. The rate at which this will occur depends on the temperature and concentration of carbon in the sodium.

In spite of intensive studies, the nature of carbon in sodium and its chemical interactions with other contaminants and container materials are little understood. However, the carbon in practical sodium systems can be controlled by diverting part of the flow through a stainless steel filled hot trap maintained at a temperature well above that of the normal operating system.

Cover Gas Purity

The cover gas over the sodium is the first line of defense in preventing contamination of the sodium. The ability to maintain pure sodium may largely depend on the success in providing and maintaining high purity cover gas. This means high quality gas purification equipment will be needed.

Control and Regulation of Oxygen and Carbon in Closed Loop Facilities

In most experimental situations involving the closed loop facilities, low contamination levels will be desired to prevent aggravated sodium corrosion owing to relatively high operating temperatures and to dissimilar metal effects. For others, considerations relating to corrosion may be of lesser importance than the regulation of contamination at specific levels to create the environment of the customer's reactor. In either type of situation, control and monitoring of oxygen and carbon levels will assume appreciable importance. In fact, it is reasonable to expect that the loop problems along this line will be more demanding than those associated with the main cooling circuits of the reactor.

For contamination control, the techniques of cold trapping for oxygen and hot trapping for carbon can be utilized to achieve and maintain high-purity sodium in the loop circuits. Oxygen control below 5 ppm can be obtained with zirconium hot traps. If called for by the environmental criteria for particular experiments, upward adjustments of contamination levels (from the minimum of about 25 ppm of oxygen achievable by cold trapping and about 15 ppm of carbon achievable by hot trapping) can be made by purposely adding the contaminant and appropriately adjusting the trap temperatures. However, supplemental laboratory loop experimentation will be required to provide the ground rules for contamination-level manipulations such as these.

4. 5. 2 Self-Welding and Fretting

Close-fitting austenitic stainless steel core components immersed in sodium may be subject to self-welding. Alternatively, it is possible that vibrations induced in the core by flowing sodium will lead to fretting of contacting surfaces. Fretting, as used here, is defined as accelerated wear occurring at contacting surfaces undergoing low-amplitude vibratory relative motion.

Austenitic stainless steel is well known for its tendency to gall and seize when sliding against itself. Under static-contact conditions, 300-series stainless steel can be made to self-weld at elevated temperatures when the surfaces are clean and in a high-vacuum environment. Penetration or evaporation of native oxide films on the surfaces permits metal-to-metal contact and subsequent adhesion through diffusion. In a reducing medium like molten sodium, particularly at high temperatures, native oxide films would tend to be reduced, so that self-welding of sodium-immersed contacting surfaces is highly likely.

Recognizing this potential problem, designers of the Fermi reactor conducted static and rubbing adhesion tests⁽¹⁾ with various candidate materials for sodium-immersed mating surfaces. These led to the adoption of Stellite 6, the Colmonoy hard-facing alloys, and nitriding for assorted bearing and contact applications in the sodium system, the choice depending on design life and ease of replacement.

Extended studies at Battelle-Columbus^(2, 3) have uncovered a phenomenon which may prove very useful in the control of fretting, wear, and self-welding of the sodium-immersed exterior surfaces of the nested hexagonal tubes forming one core concept of the FFTF. This refers to the fact that in many sodium-oxygen-metal (M) systems, thermodynamics

1. R. B. Jerman, R. C. Williams, and D. O. Leeser. "Evaluation of Material Wear and Self-Welding in Sodium-Cooled Reactors," J. of Basic Engineering, June 1959.
2. J. W. Kissel, W. A. Glaeser, and C. M. Allen. "Frictional Behavior of Sodium-lubricated Materials in a Controlled High-Temperature Environment," ASME Paper No. 61-LUBS-16, presented at Lubrication Symposium, Miami, Florida, May 8-9, 1961.
3. J. W. Kissel, C. W. Melton, and W. S. Glaeser. Behavior of Rubbing Molybdenum Surfaces in Sodium Environments, BMI-1405. January 4, 1960.

favor the formation of a double oxide ($M_xO_y \cdot nNa_2O$). Furthermore, in a number of cases this double oxide forms a stable, adherent protective, lubricating surface film which replenishes itself by further reaction if it is scratched or worn away.

4. 5. 3 Heat-Exchanger Vibration and Fatigue

In any type of liquid-filled heat exchanger, there is the possibility of fatigue and fretting corrosion, as well as increased pressure drop, from flow-induced vibrations of the elements.^(1, 2) Fatigue-induced failures of heat exchanger components in sodium-cooled reactors have been documented.⁽³⁾

Experience has shown that heat exchangers are not ordinarily designed to preclude the occurrence of such failures. In the case of gaseous heat exchangers, vibration problems usually produce a high noise output which is often unbearable in itself and warns of impending disaster. However, in the case of sodium-filled units, the massive walls and absence of acoustic modes often muffle this telltale noise. Thus, this source of trouble may not be discovered until well into the shakedown program, or even later when fatigue failures occur. At this point, even minor changes become costly.

Each heat-exchanger component in the FFTF design should be examined as a possible source of vibration problems, and required prototype tests should be performed.

4. 5. 4 Cavitation in the FFTF Facilities

Because of possible adverse nuclear, thermal, and mechanical effects of cavitation in the FFTF sodium-flow circuits, precautions must be taken at the design stage to minimize the chances for cavitation to occur.

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1. A. A. Putnam. "Flow-Induced Noise and Vibration in Heat Exchangers," ASME Paper No. 64-WA/HT-21. November, 1964.
 2. J. O. Smith, W. A. Boelter, J. H. McCarthy, and J. H. Brennan. "Sodium Flow Induced Vibration in Steam Generator Tubes," ASME Paper No. 64-WA/PWR-4. 1964.
 3. "Proceedings of Na Components Meeting," SAN-8002. August 20-21, 1963.

This is particularly advisable because the high sodium-flow velocities and complex flow-passage geometries in the FFTF loops may be conducive to cavitation. It has been observed that the rate of erosion of steel by cavitation in liquid sodium at 400 F is about 50% greater than that in water at 80 F for a given amplitude and frequency.⁽¹⁾

An analytical study, followed by laboratory work to check the areas of greatest uncertainty, is desirable. However, radiation could influence cavitation behavior by affecting the nucleation and condensation of subcooled sodium vapor. Thus, the final checks on cavitation will have to await loop operation under prototype conditions of flow and radiation field.

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1. H. S. Presier, A. Thiruvengadam, and C. E. Couchman, III. "Cavitation Damage in Liquid Sodium," Technical Report 235-1, Hydronautics, Inc., April 1, 1964 (Report NASA-CR-54071 on Contract NASr-105).

4. 6 Analytical Chemistry Service Requirements for FFTF

It has been determined that BNW has available most of the basic facilities for analytical services required for support of FFTF development and design. For operation of the reactor some additional facilities will be required.

4. 6. 1 Fuel Development

The reactor fuel will operate under extremely severe conditions, and trace impurities may be of vital importance in the fabrication of a satisfactory fuel. Emission spectrographic techniques are used for trace metallic impurities, and neutron activation analysis may be an important supplementary technique. BNW has both capabilities. Trace adsorbed gaseous impurities are important in the performance of fuels at high temperatures, and hot extraction techniques are used for their measurement. BNW has an active analytical program for these measurements.

The isotopic composition of the fuel must be known at the time of fabrication. Mass spectrometers are used for the measurement. BNW has full capability for making the measurements.

Burnup of fuel will be measured to some extent by measurement of selected fission products by radiochemistry and mass spectroscopy, but mainly by measurement of the isotopic composition of the fuel by means of mass spectrometry. Shielded analytical cells and a mass spectrometry and radiochemistry laboratory will be needed. BNW has the mass spectrometer and radiochemical facilities.

The mass analysis of noble gas fission products will be needed on gases extracted from spent fuel elements. BNW has experience in gamma spectrometry and mass spectrometry of fission products.

4. 6. 2 Cover Gas Analysis

For cover gas analysis, analytical equipment of the highest sensitivity is required especially for oxygen and water.

Gas chromatographic techniques have been developed by BNW for the Advanced Test Reactor high-temperature gas loop to measure helium impurities below 1 ppm. In addition a zirconia-calcia electrolytic oxygen probe is being developed which promises oxygen and water determinations in the parts per billion level.

4. 6. 3 Alkali Metal Analysis

The analytical chemistry of the alkali metals, although improving, still offers much incentive for further development. Techniques for oxygen and carbon analyses are not yet acceptable at the level of a few parts per million. Most of the methods suffer from sampling problems in getting a representative sample to the analytical laboratory.

As in most industrial processes the desired result would be in-line instrumentation, or at least methods capable of being operated close to the process stream. Some work has been done in this area but more is needed. Sampling systems in particular need to be closely integrated with both loop design and the analytical methods.

In the case of oxygen-level monitoring, plugging-indicator devices can be employed to obtain semicontinuous information over the range where cold trapping is effective. There is no proven continuous monitoring scheme although there are some in the laboratory stage of evaluation. One which is an especially likely prospect, is essentially a galvanic cell formed by containing a reference oxygen-containing electrode material in a closed-end ceramic tube fabricated of a solid electrolyte. When the tube is dipped into oxygen-containing sodium, a small emf which is a function of this oxygen level is developed and is measurable with sensitive instrumentation.

The monitoring of carbon levels in flowing sodium systems has received relatively little developmental attention, and no effort at all has been directed toward techniques for continuous monitoring.

4. 6. 4 Other Analyses

During development and design, analyses for minor constituents and impurities in cladding and structural materials will be required in support of corrosion and other studies. These will include analyses for nonmetals such as for carbon, nitrogen, hydrogen, and oxygen, using amalgamation, combustion, distillation, or vacuum fusion techniques. Analysis for metals requires gravimetric, spectrophotometric, and emission spectrographic techniques. BNW has equipment available for these analyses. Also analyses on films formed on metals under nonradioactive conditions will be required. X-ray fluorescence and diffraction and emission spectrography will be the principal techniques used. Radioactive tracers will be used to study some metal transport phenomena. BNW has equipment and is competent in all these fields.

For reactor operation, analyses for metallic and nonmetallic impurities in materials of construction will be required, similar to those required in support of reactor development and design. The materials exposed to the neutron flux will require shielded analytical cells for much of the analysis work. Also analyses of films formed on metals during radioactive conditions will be required, and such analyses will also be performed in shielded analytical cells. BNW is active in the analysis of radioactive materials in shielded analytical cells. Additional cells may be needed for reactor support.

4. 7 Program Management Technology

4. 7. 1 PERT Applications

In the past several years program management technology has been enhanced by adoption of two somewhat similar techniques involving program planning and follow-up in depth.

These techniques have been adopted for program management of the FFTF program. Initially, the PERT technique has been adopted for planning the engineering aspects of the program, since the momomial nodes of the PERT network are particularly useful for engineering planning. Initial PERT program charts have been developed. Also, for planning and management of the FY 1965 study program a modified form of the PERT - COST System has been adopted. This will be expanded as the program develops.

4. 7. 2 Critical Path Method Applications

For detailed management of procurement, fabrication and construction, the "Critical Path (ambiguous node) Method" is usually preferable to PERT because the substantially smaller number of network nodes reduces computer memory requirements. Hence, the initial PERT programs, and the computer programs for the plant cost estimates, are being set up to permit easy transition to the Critical Path Method for control of the procurement, fabrication, and construction aspects of the program.

4. 7. 3 Survey of Time Requirements for Reactor Design and Construction

In order to obtain preliminary data for FFTF program planning, a literature survey was made of time requirements for design and construction of typical reactors, and of problem areas encountered in the course of these activities. It is to be noted that this survey was necessarily general in nature. Data were correlated from a number of different sources, and some minor inconsistencies are presumably present. However, within these limitations

the survey provided useful preliminary data on time requirements for the various phases of reactor design and construction, and also provided useful guidance on problem areas and sources of more detailed information on the problems.

Time Requirements

The time requirements data obtained for reactor design and construction are summarized in Table 4.7. It is clearly apparent that liquid-metal cooled reactors have in the past consistently required more time than other reactors.

Major Problem Areas

The survey included a review of major problem areas in liquid metal cooled reactors and other reactors. These problems are summarized in Tables 4.8 and 4.9. Clearly the most severe problems were in the liquid-metal cooled reactors.

Comments on Problems

The problems encountered in the various reactor programs appear to have resulted from two main causes: (1) process technology limitations and (2) difficulties in design and/or procurement and installation of conventional components.

In the area of Process Technology, it is planned that the FFTF design will employ to the maximum possible extent components already proven in sodium reactor service. The design will likewise reflect the benefits of knowledge gained at the other reactors both in their successful designs and their trouble areas. Extensive testing will be performed on all critical components.

TABLE 4. 7

REACTOR TIME SCHEDULES

(Based on Best Available Data as of 12-1-64)

Legend:	I	Title I Design (scoping)	N	Nuclear Testing
	II	Title II Detailed Design	A	Ascension to Full Power
	C	Construction (until start of testing)	S	Total from Start Title I to Full Power
	T	Testing (normally pre-operational)	X	Poststartup Failures

Reactor	Months of Controlling Time Along Critical Path							
	I	II	C	T	N	A	S	X
<u>Large Light-Water or D₂O Cooled</u>								
Dresden			← 32 →		← 7 →		54	12
Yankee			← 36 →		← 3 →		54	a
Indian Point			38	10	7	1	71	
Oyster Creek Plans			← 44 →				67 ^b	
Niagara Mohawk Plans			41	5	← 7 →		68 ^b	
Shippingport	7	14	28	1	3	1	54	a
Elk River		2	32	19	12	6	76	
PRTR (D ₂ O)	4	10	32	4	← 9 →		59	
Approximate Mean	5	10	35	5	5	3	63 ^c	
<u>Liquid-Metal (Na or NaK) Cooled</u>								
EBR-I			← 44 →			4	48	
SRE (Thermal)		10	← 23 →		4	10	52	14
Fermi		← 51 →		31	11 ^b	4 ^b	102 ^b	
EBR-II			28	← 52 →		2	97	
Hallam (Thermal)	10	7	28	12	12	4	73	14 ^b
Dounreay			57	← 24 →		19	115	
Approximate Mean	5	10	35	15	9	7	81 ^c	
<u>Organic Reactor</u>								
Piqua	12	4	28	18	6	3	71	
<u>Test Reactors</u>								
MTR (1950-52)			← 24 →				39	
ETR (1955-57)		6	← 18 →				29	
EBWR (1955-56)			← 19 →				34	
VBWR (1956-57)			← 16 →				31	
Brookhaven (1947-51)			← 36 →			8	59	
Approximate Mean							38	

a - Time data not available

b - Plans

c - Totals assume mean values of missing data.

TABLE 4. 8MAJOR PROBLEM AREAS IN LIQUID-METAL COOLED REACTORS

Entrainment of gas in coolant - Hallam, Dounreay
(and also BR-5 in Russia)

Heat Exchangers - Hallam

Steam Generators - Fermi

Pumps - EBR-II (and SRE after startup)

Moderator Can Leaks - Hallam (after startup)

Reactor Mechanisms - Fermi, EBR-II

Graphite Shielding - Fermi

Coolant Purity - Dounreay

TABLE 4. 9MAJOR PROBLEM AREAS IN OTHER REACTORS

Procurement of Reactor Vessel - Elk River, PRTR, BONUS,* LaCrosse*

Poststartup - Control Blades - Dresden

Other - Leaks - Piqua

- Fire - Piqua, Peachbottom*

*Time Schedule data for these reactors were not included in Table 4. 7.

In the area of problems with design, procurement and installation of conventional components, special efforts will be taken to utilize sound engineering and administrative practices, and advanced methods of program management technology.

Thorough operability reviews will be carried out. A thorough operability review requires examining each component and each system by assuming it will fail, assessing the consequences of failure, changing the design if the consequences of failure are too great, and making sure there is a practical way to overcome the results of failure.

A thorough operability review also examines all equipment to make sure it conforms to two basic Hanford "operability rules": (1) make it simple to operate and (2) make it strong.

The operability review examines maintenance characteristics of proposed equipment and systems, and incorporates maximum "maintainability" into the designs.

Finally, the operability review must assure that specifications reflect true transient conditions to be encountered, such as temperature, pressure, flow, vibration, and power failures and fluctuations.

In the area of procurement and installation a very rigid quality control, inspection, and testing program will be carried out. This program will be based on carefully prepared specifications which reflect all USA liquid-metal cooled reactor experience.

4.8 Materials Irradiation Test Requirements

4.8.1 Introduction

Since the FFTF will be a primary test facility for the fast breeder reactor program, it is essential that the functional capabilities of the reactor be carefully conceived to meet demands for future materials development and testing. To this end, information was obtained on irradiation induced materials problems anticipated in future fast breeder reactors and on the nature of the irradiation test programs required for the long-range future.

It is believed that this information will be of considerable value in defining FFTF functional requirements to satisfy long-term materials development needs for fast breeder reactors.

4.8.2 Irradiation Effects on Materials Properties

Data concerning the effects of a predominantly fast neutron flux environment on the properties of reactor structural materials are currently rare. However, some of these effects can be deduced from experiments conducted in semimoderated spectra and from known nuclear properties. The types of damage can be classified as follows:

- (a) Mechanical property changes caused by atomic displacements in the lattice.
- (b) Acceleration of diffusion-controlled reactions by vacancy generation, and
- (c) Gas generation and property changes caused by transmutation reactions.

All three types of reactions are expected to be significant in candidate materials for fast breeder reactors over the temperature range and neutron fluxes of interest. Some types of reactions and instabilities are specific to the particular alloy and are discussed separately.

Atomic displacements result from energy exchanges between neutrons and the lattice culminating in displacement spikes. If the lattice damage is truly "frozen-in" at the temperature of irradiation, then property changes usually saturate after all of the atoms have been statistically displaced two or three times. Most of the changes in properties caused by this "in-situ" damage can be annealed at temperatures below the cold-work recovery range, leaving little or no residual damage.

Recent experiments with stainless steels and other alloys reveal that neutron-induced property changes at certain elevated irradiation temperatures exceed those at lower temperatures after sufficient exposure. The peak increase in yield strength and decrease in ductility for austenitic stainless steel occurs at about 300 C (570 F) and failure occurs with low energy absorption and only minor amounts of uniform plastic strain. This form of damage is attributed to defect agglomeration, and the more complex defects require temperatures within the cold-work recovery range for annihilation.

Some residual damage due to atomic displacements occurs in austenitic stainless steels at irradiation temperatures ranging to 600 C (1100 F); however, the amount of damage above 300 C decreases markedly with temperature due to instantaneous defect recombination.

The effect of irradiation at low temperatures on fatigue resistance is uncertain. Postirradiation measurements indicate a strain dependence in that irradiated stainless steels are less resistant at higher strains, but are more resistant at lower strains than the unirradiated controls used in the experiments.

The generation of vacancies by neutron bombardment in excess of the equilibrium temperature concentration accelerates such diffusion-controlled processes as precipitation, cold-work recovery, creep, and phase transformations. The general effect is for processes normally expected to occur out-of-reactor to occur at much lower sensible temperatures in-reactor. The effect on time-dependent plastic deformation (creep) depends upon temperature. At irradiation temperatures below the range for rapid self-diffusion (dislocation climb) where creep is stress assisted, neutron

bombardment lowers creep rates. At temperatures within the self-diffusion range where creep is thermally activated, neutron bombardment might increase creep rates. Service temperatures of 1200 to 1500 F are within the rapid self-diffusion range for most candidate structural materials (except refractory materials). Other potential radiation-assisted reactions are discussed later under the particular alloy system.

Advanced fast reactors will operate at fast flux intensities in the 10^{16} nv range with neutron energies approaching a fission spectrum. Structural alloys have high activation cross sections for (n, p) and (n, α) reactions in these neutron environments, and will generate large quantities of hydrogen and helium gas atoms as well as significant quantities of new metallic constituents. The quantities of gas atoms generated are expected to be greater than those known to have serious deleterious effects at both high and low temperatures. For example, a 2-year irradiation in a flux of 10^{16} nv results in about 25 ppm helium and 125 ppm hydrogen in an 18/8-type, austenitic stainless steel. At low temperatures this gas might promote delayed failure or accelerate cyclic fatigue failure. At higher temperatures internal gas generation can be expected to greatly reduce stress-rupture properties. Although hydrogen has a high permeability in both stainless steel and nickel-base alloys at normal service temperatures and should escape by diffusion, the effects of internal defects and surfaces on hydrogen retention must be considered.

Marked reductions in the stress-rupture properties of austenitic stainless steels and nickel-base alloys have been observed at approximately 600 C (1100 F) after irradiations near that temperature. This type of damage is not recovered by postirradiation recovery and appears to increase when the thermal component of the flux is increased. This damage has been linked to the thermal-flux burnout of B^{10} . It is still uncertain whether the loss in creep ductility and time to rupture is caused by helium or lithium production from the boron, and direct evidence of the presence of helium is lacking. However, helium generation from normal boron concentrations in stainless steels or nickel-base alloys in fast breeder reactors is insignificant compared to that produced from (n, α) reactions in the basic alloy constituents.

Transmutation reactions can cause significant changes in physical properties used for in-reactor instrumentation. An example is the change in thermoelectric properties in iron-, nickel-, or tungsten-base alloys used for thermocouples resulting in calibration drifts.

Stainless Steel

Since 18/8 stainless steels are metastable alloys, the formation of such equilibrium constituents as ferrite, carbides, and sigma phase might be accelerated by neutron irradiation at normal service temperatures. The formation of these constituents normally results in embrittlement, reduction of stress-rupture properties, and reduction of fatigue life. Some of these problems can be averted by selecting stable austenitic alloys containing higher nickel contents. Such alloys will also resist the formation of martensite with cold work and the possible subsequent increase in martensite composition with irradiation.

Nickel-Base Alloys

Whether the use of precipitation-hardened nickel-base alloys over solution-hardened alloys offers any strength advantage for long-term reactor service is questionable based on available data. Irradiation appears to eliminate the effectiveness of precipitate phases in strengthening such alloys as Inconel X-750. Similar effects have been noted for precipitation-hardened stainless steels such as A-286. Overaging of these alloys near normal aging temperatures is likely. The acceleration of both these reactions and grain boundary/matrix interactions by neutron bombardment can cause transients in creep behavior.

The benefit of boron and zirconium additions in increasing time to rupture at a given stress cannot be expected in a neutron environment. The number of defects "healed" at the grain boundaries by these off-size atoms would probably be small relative to the number of defects generated by neutron bombardment.

A change in the phase equilibria of nickel-base alloys will occur from the $\text{Ni}(n, \alpha) \text{Co}$ reaction. The increased concentration of cobalt in the nickel matrix will decrease the solubility of titanium and aluminum and increase

the solubility of carbides. These changes might be beneficial from the standpoint of high-temperature strength of precipitation-hardened alloys. Although only about 0.4 w/o cobalt is generated in 2 years of exposure at 10^{16} nv, this concentration may cause significant transport of Co^{60} in the coolant system.

4.8.3 Irradiation Effects on Corrosion by Liquid Metals

In general, radiation is not thought to affect liquid metal transport processes. However, actual data are quite limited especially transport data at flux rates of 10^{16} nv. There are several mechanisms by which sodium might become more corrosive under the unusually high fluxes of high-energy neutrons. For example, if the rate of corrosion is limited by the presence of adherent, protective films on the corroding surface, these films may be damaged by the fast neutrons. Again, if the rate-determining step involves the formation of a complex molecule (like $\text{FeO} \cdot 2\text{Na}_2\text{O}$), fast neutrons could break up the molecule and accelerate corrosion. Another factor which might be expected to increase corrosion is knockout of atoms from the steel surface into the liquid-metal stream by elastic collision of fast neutrons. A fast flux of 10^{16} nv will, according to calculations, knock about 1 mil/year of iron atoms from a steel surface. Because of greater surface exposure and more intimate contact of these displaced atoms with the sodium, interactions could be accelerated. Also neutron exposure of 10^{24} nvt could produce sufficient transmutation to change alloy corrosion properties from normal.

These speculations are presented to indicate possibilities of accelerated corrosion of steels by sodium in the presence of large fluxes of high-energy neutrons. The only sure way of finding out whether this is really important is to subject the steels to massive doses of fast neutrons at realistic fast breeder power reactor operating temperatures.

4. 8. 4 Irradiation Testing in the FFTF

In view of the problems described in the preceeding paragraphs, it is evident that an irradiation testing program is required for materials development for fast breeder reactors. The irradiation testing program should have four main objectives:

- a) Conduct irradiation screening studies to establish broad service limits of candidate materials.
- b) Perform basic studies on the effects of irradiation on flow and fracture behavior and structural stability of alloy systems.
- c) Determine the effect of flux intensity, flux spectrum, and integrated exposure on the nature of irradiation damage, gas generation, and transmutation reactions.
- d) Conduct long-term surveillance irradiations of materials applied to fast reactor components, to include in-service surveillance of components and postservice component testing.

The irradiation screening studies of candidate materials would, by means of discriminating tests, establish the resistance of materials to a broad range of irradiation temperatures and exposures.

Since the effect of (n, α) and (n, p) reactions on the properties of materials in fast neutron environments will be substantial and potentially limiting to component service life, measurements of gas generation and permeation in metal would be made both pre- and postirradiation and, when possible, by in-reactor measurement. Similar investigation on the physical properties of instrumentation elements would also be conducted.

These studies would involve large numbers of specimens and would require a fast irradiation facility. The bulk of the irradiations could be conducted in open loops of the FFTF. A smaller quantity of specimens representing alloys having potential application at high temperatures would be irradiated closed loops in the FFTF.

Basic studies of irradiation effects on deformation and fracture behavior and on metallurgical stability would involve a wide variety of specimens and irradiation parameters and would require in some cases special instrumentation and temperature control. Specimens would be located in a number of core and periphery positions in the FFTF.

Experience at Hanford with many reactors has shown the need to incorporate corrosion evaluation test assemblies into selected portions of the reactor. These test assemblies monitor the corrosion performance of the reactor components. In the FFTF, these test assemblies would also become part of the radiation damage studies and help in the evaluation of the corrosion performance of prototype fuel elements. Where practical, test assemblies would be located in-reactor and out-of-reactor in portions of the primary loop as well as in heat exchangers, secondary and tertiary coolant loops, and in other locations indicated by the testing requirements of the fuels or materials evaluation program.

The FFTF will be the only facility which provides the flexibility in test parameters and instrumentation required for these studies. The high flux rates will make possible accumulation of high total neutron exposure in short periods of time. Thus it would be possible in the FFTF to survey economically a broad selection of candidate materials and develop property data under realistic fast reactor conditions. An understanding of the processes affecting material performance could make possible a new group of alloys specifically designed for fast breeder service. There is every reason to believe that today's alloys which were developed for the chemical industry or steam plant service are not optimum for fast flux use in sodium.

4. 9 Hanford Reactor Technology Experience

Hanford personnel qualifications for reactor programs have been demonstrated by design and/or operation of 8 plutonium production reactors, one Pu-and-power dual purpose reactor, one 70 Mwt test reactor, and a number of small critical facilities or test reactors. In fuel development, Hanford has achieved a leading position in many areas, including the development, application, and reprocessing of plutonium-bearing fuels, which are particularly important to fast reactor systems.

Hanford competence in reactor design analysis is recognized, and one of the normal functions of the reactor engineering group is the review of reactor proposals for the Atomic Energy Commission.

Because of the overwhelming importance of nuclear safety at Hanford, safeguards analysis and evaluation of all types of reactors is a continuing function for a group of Hanford people.

Hanford has some key personnel who have had experience in sodium technology. Comprehensive experience is available in the analysis and control of the helium cover gas for the sodium system; extremely high purity will be required in this cover gas system to minimize sodium degradation. For development and operation of the Fast Flux Test Facility, extensive requirements will also exist for analytical chemistry services for sodium purity analyses and quality control, analyses of metals, analyses of films on metals, and the like. Hanford is well equipped and staffed to perform these services.

Hanford personnel have had extensive experience in reactor corrosion monitoring and trouble-shooting, have a strong background in corrosion fundamentals and corrosion mechanism research, have performed a number of successful in-reactor capsule experiments, and have complete facilities for handling radioactive materials.

Hanford experience with decontamination Research and Development and actual decontamination would be valuable background in fast reactor decontamination problems.

Hanford is recognized for competence in reactor physics, and for project management firmness in demanding demonstrated practicability and operability in all plant designs.

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