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**FAST FLUX TEST FACILITY
MONTHLY INFORMAL
TECHNICAL PROGRESS REPORT**

FEBRUARY 1969

**AEC RESEARCH &
DEVELOPMENT REPORT**

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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT

FEBRUARY 1969

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March 7, 1969

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FFTF MONTHLY INFORMAL TECHNICAL PROGRESS REPORTFEBRUARY 1969ABSTRACT

This report was prepared by Battelle-Northwest under Contract No. AT (45-1)-1830 for the Atomic Energy Commission, Division of Reactor Development and Technology, to summarize technical progress made in the Fast Flux Test Facility Program during February 1969.

INTRODUCTION

This Informal Technical Progress Report is divided into five chapters. Chapter I presents a brief summarization of FFTF overall program progress during the reporting period. Chapter II ("Systems") describes the conceptual features of the FFTF design and summarizes design management progress. Chapter III (General Technology) and Chapter IV (Development) describe work done by the Reactor and Plant Technology Department and the Development Department, respectively, in support of FFTF design. Chapter V describes progress in the FFTF Fuels Program.

The various chapters of the report reflect, as appropriate, results of progress made by offsite supporting organizations.

I. OVERALL FFTF PROGRESS

A. DECISIONS ON FFTF CONCEPTUAL DESIGN

1. NPTF

Work on NPTF will be deferred during FY-1970, with no funds allocated to the NPTF 189a.

Future planning will be based on the installation of the NPTF in its own building, completely separated from the FFTF.

2. Modified Reference Concept Reactor and Cover Plan Arrangement

The reactor plan and cover plan shown in Figure 3.A.1 (SK-3-14251), in Chapter III has been adopted by BNW as a modification and improvement of the conceptual reactor arrangement. The improvements of the modified arrangement include:

- Elimination of lazy susan in-vessel storage by use of fixed positions.
- Elimination of the lazy susan access penetrations in the cover.
- A reduction in vessel and cover diameter by approximately one foot.
- An increase in the allowable instrument plate support shaft diameter from approximately 10 to 12 in.
- Elimination of the necessity to locate in-line control rod drives on the rotating plugs - control rod locations incore are unchanged.
- Capability for fuel transfer to and from adjacent core sectors without fuel removal from the vessel.
- Reduced interference between the instrument shaft lower guide bearing and the core restraint mechanism.

As a result of the modification in arrangement, the rotating plugs have increased in diameter by approximately 10 in. with corresponding increases in the maximum instrument plate and refueling arm dimensions.

This modification has been informally reviewed with W, and further steps are being taken to ensure agreement.

3. Vessel

The piped bottom inlet vessel was selected as the reference design, with Westinghouse agreement and DRDT acceptance.

4. FEDAL System

Fission gas detection was selected as the reference Failed Element Detection and Location (FEDAL) System concept in a review meeting between BNW/W/RDT.

B. DESIGN PROGRESS OF GENERAL INTEREST

1. Core

Design areas undergoing further definition include (a) number of fuel subassemblies, and (b) instrument tree design concept.

Inconel 718 has been selected as the prime candidate for FTR safety rod springs. LMEC has been requested to perform stress relaxation tests on Inconel 718 in elevated-temperature sodium in order to confirm the adequacy of the material's selection for the spring application.

The use of stainless steel within FTR control rods has been computed to reduce significantly the reactivity worth of the control rod. For example, a 15 vol% decrease in stainless steel content is calculated to produce approximately a 15% increase in worth.

Type 316 stainless steel has been tentatively selected as the material for the structure of the in-reactor tube of the FFTF closed loop system.

The area on the fuel duct perimeter has been examined for flow suppression using options including an edge liner and half-size wire wrap on the perimeter pins. Use of half-size wire in the pins reduces the hot channel flow factor from a maximum of 20% for the nominally tight bundle to between 2 and 4%. The use of half-size wire on these pins is being considered seriously as a design alternative.

New correlations have been obtained for calculating pin bundle pressure drops. This has resulted in increased confidence in the predicted FFTF pin bundle pressure drop.

A new method of sizing the fission gas plenum is presented which differentiates between primary and secondary stage creep. This results in a firmer basis for creep and safety limits in the plenum design.

The PECT II and INEQ codes are operational for nonlinear elasticity for creep and for plasticity. This will enable more detailed computation of fuel pin performance and operational behavior.

2. Containment

Sodium slug impact loadings of a 300-ton vessel head have been estimated for thin-wall and heavy-wall vessels. If the head is unrestrained, and aluminum honeycomb is used as an energy absorber, the required missile barrier thickness to decelerate the head is estimated at ~2 ft and ~6 ft for thin-wall and heavy-wall vessels, respectively.

C. PROGRESS ITEMS OF GENERAL INTEREST IN FUNDAMENTAL TECHNOLOGY

Critical experiments, being performed in ZPR III, are on schedule with completion date estimated to be March 7.

Transfer of the programs to a larger facility for the prime portion of the program will be delayed if authorization for operation of ZPPR or ZPR 9 is not received by ANL by March 7, 1969.

Creep-rupture testing of Type 304 stainless steel irradiated to a fluence of $\sim 1 \times 10^{22}$ nvt (total) continues to demonstrate that the irradiation caused marked reductions in rupture life, creep ductility, and creep rate.

Neutron radiography of vibratory compacted B_4C powder irradiated in EBR-II to a B^{10} burnup of $\sim 1.5\%$ indicates that the poison column expanded during irradiation. One column increased in length by approximately 0.8% whereas the length of another column increased by about 1.8%.

D. FFTF FUELS

Powder preparation for the first mixed oxide pellets to be made in the Process Demonstration Facility commenced on January 29. All stations in Line No. 1 except the sintering furnace, debinding and 2-binder addition gloveboxes have been committed to Pu. The furnace glovebox will receive Pu by mid-month. The granulation - debinding glovebox will remain uncontaminated until all additional equipment has been installed. Two sintering runs were made in Line No. 1 to determine the cause of pellet cracking reported last month. It was found that cracking could be prevented by protecting the pellets from air exposure after the binder was removed.

Fabrication of the 217-pin coolant mixing test subassembly was completed on February 11.

Rebids from vendors on the revised specification for fuel pin plenum springs were not responsive to requirements for surface cleanliness which stipulated 0.0005 mg/in.^2 maximum halogen contamination. Vendors were unable to meet this requirement and it will probably have to be waived to obtain

springs. Springs obtained under this waiver will be cleaned and qualified at BNW, with respect to halogen contamination prior to use.

Density measurements were begun on an EBR safety rod thimble. This thimble has accumulated approximately 9.6×10^{22} n/cm² total neutron fluence. Initial results indicate peak swelling values of about 2.8% at an irradiation temperature of approximately 800 °F.

The presence of precipitates in irradiated Type 316 SS and their absence in the unirradiated thermal control material of the same heat, indicates that fast neutron irradiation induces precipitation in this alloy.

Irradiation of four capsules of 50,000 MWd/tonneM exposure continues in GETR. The approximate burnup for capsules PNL-59-7 and 8 is 40,000 MWd/tonneM. For capsules PNL-59a and 10 it is 31,000 MWd/tonneM.

Fuel-cladding interactions have been observed in Fuel Pin BNW-1-6 (~50,000 MWd/t). Evidence of severe attack was observed at part of the cladding boundaries which in some instances extended as far as 0.006 in. into the cladding; this attack effectively reduced cladding strength by one-third.

II. SYSTEMS

W. B. McDonald

A. REACTOR PLANT DESIGN

S. O. Arneson, Acting Asst. Department Manager

1. Reactor

R. C. Walker

a. Reactor Core - Fuel Pin Spacing Method

J. F. Wett

A BNW-Westinghouse meeting was held in February to discuss the status of the comparative study of spacing the fuel pins of the FTR by use of wire wrap or grids. The following conclusion was reached:

- The stainless steel spacer grid (PFR or hard point type) is preferred for the FTR and effort will be concentrated on this type grid during the remainder of the comparative study.

Knowledge of fuel swelling is vital to an adequate assessment of the spacer systems. The following swelling equations were agreed upon by BNW and W for use on an interim basis so that both groups will be using consistent values.

a. For Type 304 SS

$$\% \frac{\Delta v}{v} = 1.3 \times 10^{-34} (\phi t)^{1.6} e^{(-7000/RT)}$$

b. For Type 316 SS (annealed)

Prime: $\% \frac{\Delta v}{v} = 6 \times 10^{-34} (\phi t)^{1.6} e^{(-10200/RT)}$

Alternative: $\% \frac{\Delta v}{v} = 1.3 \times 10^{-34} (\phi t)^{1.6} e^{(-7000/RT)}$

c. For Type 316 SS (20% C.W.)

$$\% \frac{\Delta v}{v} = 1.4 \times 10^{-35} (\phi t)^{1.6} e^{(-5700/RT)}$$

Work is continuing on this evaluation and another status meeting will be held in March.

BNW and W have agreed that a fixed position in-vessel decay storage will be used as the baseline concept.

W has concluded from their evaluation of layout drawings provided by BNW that there is space to increase the instrument shaft diameter from 10 to 12 inches to give more space for instrumentation lines inside the shaft.

b. Redirected Backup Core Effort

The November 1968 Monthly Technical Progress Report (BNWL-937) reported termination of the backup core design activity being performed by General Electric Company, and indicated that other tasks for continuing GE support of the FFTF Project were under active consideration. The redirected GE work is strongly oriented toward FFTF development program support. The FFTF Development Department has assumed responsibility for future reports on this subject.

c. Reactor Vessel and Shield

R. C. Walker and O. W. Priebe

The results of the Reactor Vessel studies by BNW, W and B&W were compiled in Document No. BNWL-1001, FFTF Reactor Vessel Concept Evaluation Summary. This report evaluates the annular inlet and the piped bottom inlet reactor vessel concepts proposed for the FTR. The concepts differ mainly in the location of the inlet nozzles, the method of pressure vessel wall cooling, and the containment of system inlet pressure. The Annular Inlet Concept locates the inlet nozzles above the core, cools the pressure vessel by the main coolant stream in series with the core, and contains the inlet pressure within the entire pressure vessel. The Piped Bottom Inlet Concept places the inlet nozzles below the core leading into a

relatively small high pressure plenum while most of the cylindrical vessel is at outlet plenum pressure (low pressure) with vessel wall cooling in parallel with the core.

The detailed evaluation of the concepts presented in the document clearly presents an overall advantage for the Piped Bottom Inlet Concept.

A meeting held on February 12 and 13, 1969 at RDT reviewed the results of the study work and selected the piped bottom inlet design for the reactor vessel configuration.

Joint meetings have been held to resolve the use of RDT Standards. A review of the available Standards has been made for comment and applicability as they pertain to the FFTF Project but an agreed summary list and extent of applicability is yet to be developed.

A summary listing of action items of engineering effort required to accelerate the acquisition of the reactor vessel was issued, and an updated status report is planned on a semi-monthly basis.

During the February 12 and 13, 1969 meeting at RDT, it was agreed that B&W could assist in the design engineering effort of the reactor vessel. A scope of work was developed and a means of funding this work through FY-1969 is being considered.

2. Heat Transport and Irradiation Testing

J. H. Westsik

a. Heat Transport System

C. M. Cantrell

(1) Isolation Valve Functional Requirements for the Primary Loop of the Heat Transport System. Present information on functional requirements for the isolation valves in the heat transport system based on the latest configuration is as follows:

(a)	<u>Number of Isolation Valves/Loop</u>	<u>Size, in.</u>	<u>Max. ΔP, psi</u>	<u>Wt Flow, gpm</u>
	1	26	0.08	13,000
	2	12	0.6	6,500
	↑ or			
	1	18	0.4	13,000
	<u>Number of Check Valves/Loop</u>			
	2	12	0.6	6,500

- (b) Internal working parts must be remotely replaceable from operating floor without the need of draining the primary loop.
- (c) Isolation valves must fail in open position (fail open).
- (d) Double seal barrier is required to prevent radioactive sodium vapor from penetrating to the operating floor.
- (e) Valve must be capable of being installed in horizontal pipe runs at the heat transport piped system.
- (f) Isolation valve rate of closing time must be less than 0.2 in./sec based on water reactor power plant experience, to avoid sodium hammer effects.
- (g) Total check valve closure time is three seconds based on the Fermi modified check valve. The rate of closures

should be such that the reverse weight flow is reduced to a minimum as rapidly as possible while liquid sodium hammer is avoided.

A review of the FTR system flow characteristics was made. A firm set of system flow requirements was established for the primary pump as 13,000 gpm and 500 ft TDH.

b. Irradiation Test Facilities

M. A. Vogel

(1) System 61, 61-P - Closed Loop System

(a) Project Management - (M. K. Mahaffey, R. R. Wyer). CSDD 61 for the closed loop system has been revised in accordance with RDT and WARD comments. Review of the CSDD by AI as approved by RDT, has been initiated.

The start of prototype closed loop design has been deferred until July 1. Work packages for Westinghouse and AI have been completed and will be submitted with the project proposal for RDT approval.

(2) Short-Term Irradiation Facility (STIF) -

(R. R. Derusseau, R. E. Keyes, M. K. Mahaffey). Calculations have been performed to investigate the feasibility of testing a 1 MW fuel test in the STIF reflector position. Pressure drops, capsule sizes, specimen weights and flow velocities were calculated using a 19-pin bundle with a test diameter of 2.5 inches and a pitch diameter ratio (p/d) of 1.3. The results of the calculations indicate that the test described could operate at 1 MW with ΔT 's ranging from about 250 to 400 °F. Figure 2.A.1 shows operating characteristics of the tests described previously. Gas cooling of this typical test was also investigated. The gas heat removal system is designed to remove 20 kW of decay heat with a closed argon loop. The gas-circulating blower will be sized to provide a

FIGURE 2.A.1. Power Versus Flow Versus Pressure Drop

head of 4.18 psi plus line losses at a flow of 106 ft³/min (50 psi gas pressure). Calculations of the operating capability of the system show that movement of the test carriage by gas propulsion is feasible for the specimen described.

3. Fuel Handling and Radioactive Maintenance

W. E. Cawley

Plant Arrangement Study

W. E. Cawley, F. H. Shadel

A study of various FFTF Fuel Handling and Radioactive Maintenance System/Plant Arrangements has been completed by a joint study group of BNW and subcontractor personnel. The objective of this study was to establish the reference Plant Arrangement to be used in the Fuel Handling and Radioactive Maintenance CSDD's.

The reference Plant Arrangement concept recommended for CSDD's features a spherical containment vessel with individual vertical fuel transfer and radioactive maintenance ports located adjacent to each other between the FHM gantry rails. Separate transfer tunnels connect these containment transfer ports to Fuel Handling and Radioactive Maintenance cells located outside containment. The Fuel Handling and Radioactive Maintenance cells are positioned above grade with the below-grade transfer tunnels connecting at the cell floor. The fuel handling cell is combined with the examination cell into a common inline cell. Fuel is transferred from containment in sodium filled pots carried in a top loading transfer machine which operates in the transfer tunnel. Radioactive components are transferred from containment using a transport carriage which operates in the radioactive maintenance transfer tunnel. A fuel decay storage is located adjacent to the Fuel Handling and Exam cell.

4. Reactor Plant Instrumentation

E. M. Johnston, C. D. Swanson

a. Reactor Instrumentation and Control

R. R. Cone

(1) Reactor and Vessel Instrumentation System - CSDD 92 (W. Dalos, L. E. Fort). Proposed locations for coupon irradiations within the vessel in critical areas (such as the grid plate and inlet plenum) have been identified through development task RM-5. Although it is agreed such measurements are necessary to determine fluence and neutron energy levels at these locations, it will be nearly impossible to provide extra holes in the reactor top specifically to insert and withdraw coupons. Therefore, it appears that such coupons must be placed in other structures (such as subassembly nose pieces or at the bottom of flux chamber thimbles) to utilize existing vessel-top openings for these surveillance purposes.

In the revision of CSDD 92, this change in accessibility will be indicated. Also, the reactor surveillance panel will be located outside the containment so that in the event of containment evacuation, the vessel conditions can be observed.

A heat transfer analysis performed by the Thermal Hydraulics group indicated that the driver fuel instrument connectors will be required to operate in a temperature environment of approximately 400 °F. Task IC-17, Signal Lead Nozzle Connectors, will be modified to reflect this change in requirements.

The increase in the diameter of the instrument tree support post from 10 in. to 12 in. permits an increase in the maximum diameter of the eddy current flow sensor from 7/8 in. to 1 inch.

(2) Flux Monitoring and Control Instrumentation - CSDD 95 (L. W. McClellan). Safety analysis has required that instrumentation be provided which will initiate a reactor scram upon detection of excessive decreases in reactor power when this power is between 1% and 100% of full power. Protection is provided in case of a single element fuel meltdown and/or sodium voiding in the outer radial fuel elements. A negative rate-of-change circuit based on the high level flux was added to the concept. An adjustable setpoint will determine the rate-of-change level which would initiate a primary trip to the Protective Logic part of the PPS.

Reactor Physics has required that instrumentation be provided for making subcritical reactivity measurements. The indirect reactivity measurement they propose will use the low-level neutron sensors located in-vessel.

The basic instrument requirement states that the value of the neutron reaction rate must be monitored to an accuracy of $\pm 2.5\%$ when the reactor is subcritical 30\$. From these data, it has been determined that at 30\$ subcritical any change in reactivity can be resolved within $\pm 1\%$. Absolute reactivity determinations which must use additional information such as rod calibrations and source strength are expected to be within $\pm 20\%$ at 30\$ subcritical. It appears possible to meet these requirements with present technology provided a satisfactory means is found to get a sensor into the reactor.

(3) Fuel Failure Detection and Location (FEDAL) - System 94 (R. R. Cone, W. Dalos).

(a) Functional requirements.

- The FEDAL system will be designed so that the reactor need not be operated with failed fuel in order to locate the failed fuel. It is understood that during the early stages of FTR operation, the reactor will not be operated

intentionally with failed fuel. "Early stages" of FTR operation was defined as anytime during the first operation of the first core, probably six months to a year. "Will not be operated" was understood to allow reactor cycling to induce fission product release if necessary to find or confirm the failure location prior to reactor shutdown for subassembly replacement.

- Fuel will be considered to be failed if the cladding integrity has been violated at any location, including the gas plenum region. Cladding integrity violation occurs when there is a hole or rent in the cladding which will release fission products. A swelling condition in the cladding which has not released gas does not represent a cladding integrity violation.
- The fuel failure must be detected rapidly; the detection time has not been established. "Rapidly" is defined as the value which can be obtained by a well-engineered system using state-of-the-art principles. Present estimates for this value are approximately 3 minutes for a low sensitivity gross monitor and 20 minutes for a high sensitivity chromatographic monitor.
- The failed fuel location system should be designed to locate the failures or failure zone positively without contributing to reactor unavailability.

It will be necessary to locate the failure or failure zone before stripping the top face for refueling access. In special cases, it is understood that reactor cycling may be required to induce additional fission gas releases to determine or confirm the failure location.

(b) Concept for Location System. The basic fuel location system will operate on the principle of detecting a fission gas release.

However, a study will be performed during preliminary design to determine the feasibility of providing a location sampling system which can accommodate the flows required by a DN monitor without extensive design changes in in-vessel hardware.

(c) Concept for Detection Systems. It was agreed that the reactor failed fuel detection concept will include a reactor cover gas sampling system to detect abnormal gas activity, and a delayed neutron monitoring system which monitors the coolant in the primary loops, either from the reactor vessel directly, or from the location system sample line grouped flow.

The closed loop failed fuel detection concept will include a delayed neutron monitor and a gas activity monitor for each closed loop.

(d) Vented Fuel Considerations. The following conclusions were established regarding the prospect of operating with vented fuel or with leaky driver fuel:

- The basic systems proposed for both detection and location contain the basic elements expected to be valuable during vented fuel operation; namely, that a means be provided to obtain a sodium sample from each S/A for detection of fission gas, DN precursors, or other fission products.
- In-cell space for an alternative fuel failure location system must be considered in the system design.
- The FEDAL system should be designed for a nonvented element, but consideration should be given to the use of vented-to-coolant driver fuel elements at some future time.

B. GENERAL PLANT DESIGN

E. M. Johnston

1. Sodium and Gas System

R. V. Dulin

The Inert Gas Receiving and Processing System (System 82) CSDD was revised incorporating RDT's comments and the design safety criteria were incorporated into the document.

2. Structures, Containment, and Utilities

F. H. Shadel

Containment Space and Arrangement Layout

F. H. Shadel, C. C. Steele

A plant arrangement study team comprising AGC, AI, and BNW personnel has been working on fuel handling and radioactive maintenance equipment handling problems associated with various containment building arrangements. Details are given in the Fuel Handling and Radioactive Maintenance section of this report.

A meeting was held with Bechtel and BNW to exchange containment space and arrangement ideas. Fuel handling and radioactive maintenance equipment handling models were shown to Bechtel and discussed with them. In turn, Bechtel presented their layout drawings and the reasoning behind them. Present Bechtel drawings show a 200 ft diameter cylindrical containment vessel. Additional work is planned to determine the optimum size vessel and cell arrangement within the vessel.

3. General Plant Instrumentation and Electrical

C. D. Swanson

Central Control and Data Handling System Programming

D. W. Hartmann, G. W. Main

Conceptual design has been completed on an analog signal conditioning program which is compatible with the real-time monitor program being developed for the BNW SEL-840A computer. The signal conditioning program provides high or low limit checking, filtering, conversion to engineering units, and verification of transducer status.

Plant Protection System

J. P. Thomas

The relationship of Plant Protection System documents IEEE 279 and AEC 10CFR50 to FFTF design requirements was discussed during a meeting between RDT, BNW, Westinghouse and Bechtel. It was concluded that all the requirements of these two documents should be included in the FFTF instrumentation CSDD's unless exceptions are clearly stated. Where the CSDD wording must differ from that in the IEEE 279 or 10CFR50, the intent of the different wording will be stated.

III. GENERAL TECHNOLOGY

P. L. Hofmann

A. DESIGN EVALUATION

L. M. Finch

1. Reactor Evaluation

D. Marinos

a. Core Concept Studies

G. R. Waymire

Work initiated last month⁽¹⁾ to resolve problems in core design which will affect preparation of CCDD 31 was continued during February. Several problem areas noted previously have been resolved, while other major features have been identified as problems.

Problem areas noted last month which have been partially resolved include:

- Below Core Length Increment.

Additional irradiation test data and further analysis of existing data by Metals and Materials Development have indicated that the shift in neutron spectra to lower energy levels at the tubesheet location will allow damage weighting factors to be utilized with the total spectra criteria utilized previously. Preliminary indications are that the addition of this weighting factor will allow the below core shielding (in the fuel assembly) to be removed, thus reducing the fuel assembly pressure drop significantly.

With the change in damage criteria, the controlling requirement for below the core changes from shielding to

1. *FFTF Monthly Informal Technical Progress Report, BNWL-990. Battelle-Northwest, Richland, Washington, January 1969.*

the length required below the core for the control rod follower. The length from the assembly seat to bottom of the core thus becomes 38 to 42 inches.

- In-Vessel Storage Configuration.

The fixed position in-vessel storage concept investigation was completed and the fixed storage design was selected as the reference approach. Figure 3.A.1 (SK-3-14251) depicts the reference configuration. Minor modifications may be required to accept a 14-in. OD instrument tree "trunk." Further design definition by Instrumentation and Control indicates that this diameter may be required to accommodate instrumentation imposed by the control rod safety criteria.

Further definition has resulted in the following feature changes to the table shown last month for the fixed storage vessel design:

● Vessel OD, ft	17
● Refueling Plug OD, in.	66
● Radial Distance to Plug	
Centerline, in.	61
IVHM Arm Length, in.	40
IVHM Maximum Reach, in.	61
Available Lead-Out Inst.	
Post, in.	10
In-Vessel Storage	
Positions	48 + 3 common transfer
Other Cover Penetrations	3 finned pot locations
	with ~12 in. exit ports.

In addition to resolving the storage arrangement, the storage design effort resulted in further cover definition which incorporates a head penetration for a short term

FIGURE 3.A.1. Fixed Storage 3 Rotating Plug Arrangement Plan Study

SK-3-14251

irradiation facility (STIF) and two additional optional test positions located in the first reflector row (Figure 3.A.2, SK-3-12897).

The resulting core map is shown in Figure 3.A.3. Areas requiring further resolution include:

- Determination of the Number of Required Fuel Assemblies. This problem remains unresolved pending agreement between Battelle and Westinghouse on the fuel temperature peaking factors and total reactor power.

Additional areas requiring action prior to completion of CCDD 31 include:

- Design definition of all major components within the reactor. The instrument tree is one major component which requires further definition. Effort during February on this component was limited to defining functions and requirements. The instrument tree is a complex multi-function component which must operate reliably in the most severe region of the reactor environment. To date the instrument tree has been assigned 15 separate functions and requirements in addition to its basic function of providing mounting and mobility to the fuel and control instrumentation.

b. Fuel Assembly Duct Design

G. R. Waymire

Review of the fuel assembly design by fabrication development personnel has resulted in further configuration changes which have insignificant effect on function but improve ease of fabrication. The overall assembly is shown on Figure 3.A.4 (SK-3-14581).

SK-3-12897

CORE MAP CONCEPT 5-A

FIGURE 3.A.3. Core Map Concept

FIGURE 3.A.4. Driver Fuel Assembly

SK-3-14581

c. Core CCDD Preparation

G. R. Waymire

A first draft of the requirements section of the CCDD is complete. Review of this section with RDT and Westinghouse personnel is anticipated early in March.

Further definition of the interface between the core (System 31) and the reactor instrumentation (System 92) was completed. The instrumentation system will include that portion of the instrument tree which can be maintained without removal of the instrument tree from the vessel. The one exception is the vortex generator which is integral with the instrument structure but will be included in System 92. The remaining structural components, e.g., the probe can, the instrument tubes and the FEDAL lines will be included in System 31.

d. Reactor Vessel Conceptual Design

D. P. Schively

The basic configuration of a piped bottom inlet vessel with external guard vessel was accepted by RDT at a meeting in Germantown, Md. on February 12 and 13. The timeliness of this acceptance permits pursuance of the corresponding Design Description for the Vessel and Shield Component preparation according to existing schedule.

e. Thermal Stress Analysis - Reactor Vessel

R. L. Myers

A letter report was issued containing results of a study of thermal stresses in the upper portion of the reactor vessel caused by an axial temperature gradient. This report also contains the results of a study of thermal stresses in the vessel wall induced by a sudden injection of cold argon gas

into the space above the liquid sodium level. The maximum stresses in both cases were approximately 12,000 and 16,000 psi respectively, well within the maximum allowable for secondary stresses.

A computer code was used to check previous calculations of the radial temperature profiles and thermal stresses for different vessel wall thicknesses subjected to two specific temperature gradients. Results by both methods are comparable, with the computer giving slightly lower stresses. Also, the temperature profile in the baffled wall was determined, using one gradient on the inside of the baffle and a different gradient in the annulus between the baffle and wall. The baffle was shown to give adequate protection from excessive thermal stresses on the vessel wall. Without a baffle, the thermal stresses exceed the allowable stress with the worst condition existing in the thicker walls.

f. Nuclear Control CCDD (System 33)

D. Marinos and W. M. Noble

The initial draft of the requirements section of the CCDD for nuclear control is complete with few exceptions. Undefined areas are concerned with some safety design criteria requirements. Work is continuing on remaining sections of the CCDD concurrently in order to meet the scheduled issue dates. Concept improvement studies are continuing towards enhancement of the drive and extension shaft disassembly time for refueling operations. Figures 3.A.5, 3.A.6, and 3.A.7 (SK-3-14558, SK-3-14559, and SK-3-14560) show a 3-piece extension shaft from the drive to the poison subassembly which permits the instrument tree to carry the main shaft in the vessel and thus eliminate the need for cask operations to remove the sodium wetted shaft during refueling. Evaluation of the trade-offs

FIGURE 3.A.5. Control Safety Rod Drive Mechanism

SK-3-14558

SK-3-14559

FIGURE 3.A.6. Control/Safety Rod Drive - Main Hanger Shaft Assembly

FIGURE 3.A.7. *Poison Tip Section Control/Safety Rods*

SK-3-14560

of this system as compared to the 2-piece shaft in the original concept will determine the most optimum design. The draft of Section 2 (design description) of the CCDD can then be completed.

B. REACTOR PHYSICS AND OPERATIONS

R. E. Heineman

1. Nuclear Analysis

W. W. Little

a. Comparison of MELT-II with FORE II and NUTIGER

A. E. Waltar and A. Padilla, Jr.

A comparison of MELT-II with two other coupled neutronics - heat transfer digital computer codes, FORE II and NUTIGER, was carried out to assess its accuracy both under steady-state and transient conditions. The region of investigation of this study was limited to below substantial fuel melting.

The case chosen for comparison was a 1040-liter core divided into three equal-volume radial channels and seven equal-volume axial nodes.

The steady-state axial temperature distributions of the fuel, cladding and coolant for the hottest (inner) radial channel was compared with the exact solution. The coolant temperatures calculated by MELT-II, FORE II, and NUTIGER, agree well with the exact solution.

The steady-state radial temperature distributions at the axial midpoint of the hottest channel was also calculated. The coolant and cladding temperatures are identical for MELT-II and FORE II and slightly higher for NUTIGER because of the connection between the cladding and the coolant. The fuel temperatures for both MELT-II and FORE II tend to be higher than the exact solution near the outer surface of the fuel pellet and then lower than the exact solution near the center.

The effect of a 6\$/sec reactivity ramp, without scram, on cladding and fuel temperatures was obtained. Although the cladding temperatures for MELT-II and FORE II are initially

identical, the slightly higher fuel temperatures in MELT-II causes its cladding temperature to eventually rise faster than that of FORE II. The transient fuel temperatures for MELT-II, FORE II, and NUTIGER have essentially the same shape and compare well with the results from the finite-difference analysis. Therefore, the difference in the time it takes to reach the melting point of the fuel is the result only of the different initial steady-state temperatures. For MELT-II, there is a slight (10 °K) rise in temperature as the fuel passes through melting.

b. Flux Level in the Subcritical FTR

J. R. Sheff

Preliminary estimates have been made of the subcritical flux levels in the FTR. The calculations were performed for a 2-zone, 2-dimensional (R-Z) model of the vertical FTR core. The core was made subcritical by use of B₄C rods in the first ring of the reflector. Shown below are the axial peak total fluxes at various radial locations for the case where $k = 0.938$, along with an estimate of the fluxes at $k = 0.9$ or full shutdown.

<u>k</u>	<u>Core Center</u>	<u>Fuel Zone Interface</u>	<u>Core Refl Interface</u>	<u>Center Third Refl Ring</u>	<u>Center First Shield Ring</u>
0.938	4.34 (10 ⁵)	2.95 (10 ⁵)	1.07 (10 ⁵)	2.21 (10 ⁴)	1.04 (10 ⁴)
0.9	2.70 (10 ⁵)	1.84 (10 ⁵)	6.68 (10 ⁴)	1.38 (10 ⁴)	6.49 (10 ³)

These results are based upon a total source strength in the core of $1.24 (10^8)$ neutrons per second for an assumed fuel composition of ²³⁸Pu = 0.15%, ²³⁹Pu = 85.5%, ²⁴⁰Pu = 11.5%, ²⁴¹Pu = 2.5% and ²⁴²Pu = 0.35%. A change in the ²³⁸Pu concentration by 0.1% or a change in ²⁴⁰Pu by 1% will change the result by about 6%.

The overall accuracy of the results in the core is felt to be about 20%. The errors are perhaps greater for the fluxes in the reflectors, but this magnitude is uncertain at this time.

c. GAND/GAFGAR

K. B. Stewart

The computer program GAND was converted for use at BNW. GAND is used to process the ENDF/B evaluated neutron cross section library into the form used by GAFGAR. The only difficulties encountered were allocation errors associated with a memory overlay. Once the proper overlay for CSC's UNIVAC 1108 system was created, the GAR versions of ^{238}U were updated. The 300 °K ^{238}U was known to contain an error and the 1100 °K ^{238}U was also updated for a comparison test of the temperature effects on reactivity for the Zebra 6A critical experiment. The calculation yielded a result within the range calculated by the codes ETOX and MC².

d. ETOX

J. L. Baker

Revisions were made on the code ETOX, a code which generates group average cross-section constants from the ENDF/B Library to be used as input into Fast Reactor Code. The revisions included modifying and improving the resolved resonance region calculation and adding error-messages to insure proper input data.

Appropriate isotopes have been processed into a 26-group half-lethargy-width library to be used as input for 1DX.

e. Critical Parameters for $\text{PuO}_2\text{-UO}_2$ Systems

C. L. Brown, L. C. Davenport, and C. A. Rogers

Studies were continued to establish minimum critical parameters for various forms of FFTF fuel for use in criticality safety analyses. Calculations completed include homogeneous $\text{PuO}_2\text{-UO}_2$ -water systems, $\text{PuO}_2\text{-UO}_2$ fuel pins in air, $\text{PuO}_2\text{-UO}_2$ pins in water, and dry and water-moderated subassemblies in concrete.

2. Experimental Physics

R. A. Bennett

a. Phase B Critical Experiments

R. A. Bennett

Critical experiments being conducted with ZPR III in FTR I are on schedule. It appears that the scheduled program will be completed on March 7, 1969, and hence, the second series of experiments could begin on a large facility at that time. However, it is anticipated that the authorization to operate either of the larger facilities, ZPR-9 or ZPPR, will not be given on the required schedule. Plans are, therefore, being developed for some additional experiments that could be performed in ZPR III pending the operating approvals.

b. Detailed Plans for FTR II Experiments

R. A. Bennett, S. L. DeMyer, and W. R. Young

A major portion of the calculations have been completed in support of the detailed planning for the second series of experiments in this resumed Phase B program. These experiments are to be performed in FTR II, which is a critical assembly to be built in either ZPR-9 or ZPPR.

Previous detailed plans have been revised based upon these calculations, reviews by ANL personnel, and discussions with Westinghouse personnel. The plans have been transmitted⁽¹⁾ to ANL personnel for concurrence and to RDT for information.

c. Reflector Control Strength - Influence of Stainless Steel Content

S. L. DeMyer and R. A. Bennett

The relatively large scattering cross sections of the prime constituents of the stainless steel within FTR reflector control elements will result in partial reflection of incident neutrons. This will reduce their reactivity worth. The magnitude of this effect has been estimated with diffusion theory to be of the order of 10%. Hence, one should consider carefully the use of stainless steel within or even near FTR control devices.

Specific calculations were made on homogeneous rods placed at the core-reflector interface of a homogeneous core whose size and composition approximated that of the reference FTR. A volume fraction change in the stainless steel content from 15% to 33% for a homogeneous rod whose initial composition was 18 vol% void, 22 vol% B₄C (natural), 45 vol% sodium, and 15 vol% stainless steel, resulted in a calculated reduction in reactivity worth of approximately 15%. It can be expected that stainless steel surrounding a control rod would produce similar effects of similar magnitude. This effect per unit of stainless steel should be assessed in heterogeneous calculations also.

1. R. A. Bennett. *Unpublished Data. Battelle-Northwest, Richland, Washington, BNW-FFTF-69378. (Letter to D. Meneghetti.)*

d. DITCH-1, A Diffusion-Transport Chain Program in One Dimension

J. V. Nelson and W. R. Young

The one dimensional diffusion code, 1DX,⁽¹⁾ and the 1-dimensional SN transport theory code, DTF-IV,⁽²⁾ have been chained together. This chain program, DITCH-1, was developed to reduce the time required to run a DTF transport calculation.

Resonance-shielded cross sections are computed and a diffusion calculation is made in the 1DX link of the program. These cross sections are then used in the DTF link with the fluxes from 1DX, serving as an initial flux guess. A modification to DTF allows the cross sections to be flux-volume-weighted and energy-averaged. The running time of a typical problem is cut to one-half. In addition, the setup time of the using physicist may be reduced approximately 33% if DITCH-1 is used, because preparation of redundant sets of input data which are required for FCC-IV and DTF-IV are eliminated.

e. NPTF

R. A. Bennett

In compliance with direction received from RDT, all work on NPTF is being terminated. Completed work in support of NPTF conceptual design has been scheduled for documentation.

The NPTF work will be resumed at some future time and it will be assumed to exist sometime after FTR startup. The

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1. R. W. Hardie and W. W. Little, Jr. Unpublished Data. 1DX, A One Dimensional Diffusion Code for Generating Effective Nuclear Cross Sections. Battelle-Northwest, Richland, Washington. (Preliminary report.)
 2. K. D. Lathrop. DTF-IV, A FORTRAN IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering, LA-3373. Los Alamos Scientific Laboratory, July 15, 1965.

impact of this delay on the initial startup of the FTR, proof-testing of FTR nuclear core components, initial fuel management, and subsequent fuel and test loading, startups, and general fuel management problems will be assessed.

f. NPTF Conceptual Design

H. C. F. Ripfel and Q. L. Baird

A wrap-up of the results of conceptual design work is in progress (estimated date of completion, March 1). Documentation will include a complete set of SK-drawings and other information originally prepared for incorporation into the revised NPTF-CFDD. This minimum effort is required for an orderly termination of NPTF work and will allow for resumption of the work at a later date.

3. Radiation and Shield Analysis

W. L. Bunch

a. Radiation Intensity from Vaporized Fuel

C. A. Mansius

Accidental vaporization of a discharged FTR fuel element must be safely accommodated in the design of the FFTF. Estimates of the radiation intensity from the fission products associated with a fuel assembly at goal exposure were made assuming the vaporized products to be contained either within an 18-ft diam machinery dome or within a 160-ft diam containment sphere. Radiation intensities were calculated for decay times from zero to 10 days, distances from five to 1000 ft, and through concrete shields from zero to four feet thick. The results provide a basis for conceptual arrangement of shield walls to protect selected occupied positions.

b. Neutron Spectra at the Grid Plate

E. T. Boulette

Radiation damage to the FTR grid plate is a design criterion which must be carefully defined. To help define this criterion, neutron spectra were generated at the grid plate for various below-core shields. Grid plate locations which correspond to a fixed nvt exposure in each shield were established. Because of spectral differences associated with the shields, the energy fluence to the grid plate was calculated to vary by a factor of 40. This indicates that the neutron spectrum needs to be considered in defining radiation damage criteria as well as the total neutron fluence.

c. In-Vessel Fuel Storage

E. T. Boulette

The in-vessel storage of irradiated fuel elements will require detailed consideration, particularly from the standpoints of radiation damage to the vessel, neutron monitoring outside the vessel, and decay heat production in the stored fuel. Calculations were made to provide a basis for preliminary design of in-vessel fuel storage experiments to be performed in zero-power reactor facilities in conjunction with related shielding experiments. These calculations indicate significant hardening of the neutron spectrum in the vicinity of the stored fuel.

d. Cross Section Evaluation

D. R. Marr and M. G. Zimmerman

It is required that adequate group cross section sets be selected for use in the design of FTR shields. For economic reasons, as few groups as possible should be utilized. Neutron transmission experiments,⁽¹⁾ through various thicknesses of

1. *II. Bondarenki, et al, Atomnaya Energiya, vol. 18, No. 6, June 1965, pp. 593-601.*

iron, stainless steel, and nickel, were used as a basis for evaluating the 31-group HOMSET and the 100-group ENDF/DLC-2 cross section sets. The spectral distribution of neutrons transmitted through the shields was calculated with the transport code, ANISN. Typical results are shown in Figure 3.B.1. Since absolute fluxes were not available from the experiment, the calculations were arbitrarily normalized so the ENDF set agreed at low energies. Preliminary studies, using the two dimensional diffusion code 2DBS, which accounts for transverse leakage from the experimental sample, yield slightly harder spectra and indicate better agreement with the experiment. Studies are underway to determine if a few group set can be derived from the 100-group ENDF set that will give better agreement than HOMSET.

FIGURE 3.B.1. *Calculated and Experimental Transmission Spectra Through 30 cm of Stainless Steel*

C. REACTOR AND SYSTEMS ANALYSIS

R. E. Peterson

1. Reactor Core Analysis Section

P. D. Cohn

a. Bundle Friction Factor

D. C. Kolesar

The reference subassembly friction factor was scaled from experimental data obtained from a 217-pin bundle to correct for differences in pitch-to-diameter and lead-to-diameter ratios. The "XYZ" method of Sangster⁽¹⁾ was used in conjunction with experimental data obtained recently at BNW.⁽²⁾ Results lend increased confidence to predicted FTR fuel subassembly pressure drops.

b. Transient Fuel Pin Analysis

F. E. Bard and D. E. Dutt

Code development for transient analysis of stresses in the FFTF fuel pellet and cladding is nearly complete. Two codes are being developed. The first, PECT-II, is for the elastic-plastic creep analysis of a single axisymmetric body (such as the cladding) under thermal and mechanical loading. This code is operational for nonlinear elasticity, for creep, and for plasticity.

The second code, INEQ, is an extension of PECT-II, to two concentric axisymmetric cylinders for the simulation of the fuel pellet and cladding, respectively.

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1. W. A. Sangster, "Calculation of Rod Bundle Pressure Loss," ASME Paper 68-WA/HT-35, ASME Winter Annual Meeting, December 1-5, 1968.
 2. T. C. Reihman, Personal Communication.

INEQ is now operational for nonlinear elasticity, for creep in cladding and for plasticity in cladding. Further work is underway to define the INEQ input fuel material properties in function form.

c. Fuel Pin Plenum Design

R. J. Jackson

The fission gas plenum is sized in such a manner that it can safely contain 100% of the fission gas generated in a pin with an average burnup of 64,500 MWd/tonne (80,000 MWd/tonne peak). The critical parameter is the limited cladding ductility for mechanical strain at end-of-life fluence. The allowable stress must be compatible with this ductility.

Figure 3.C.1 shows the resulting allowable stress values at temperatures of interest.

A well-established technique for calculating the fission gas generation⁽¹⁾ was used.

The resulting gas pressures are shown in Figure 3.C.2, and allowable pressures determined by the allowable stress curves and the effective cladding thickness are shown in Figure 3.C.3.

d. Flow Suppression at Duct Perimeter

C. L. Wheeler

Figure 3.C.4 summarizes the effectiveness of four methods in reducing subassembly peak enthalpy rise for a tightly packed model of the fuel pin bundle and an alternative model characterized by the natural expansion⁽²⁾ of the bundle.

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1. *Liquid Metal Fast Breeder Reactor Design Study, Vol. II Appendix 6.5, January 1964, GEAP 4418.*
 2. *Wire Spacer Briefing, Participants: RDT, BNW, AI, Westinghouse, Jan. 21 and 22, 1969.*

FIGURE 3.C.1. Allowable Stress Values for Type 316 SS.

FIGURE 3.C.2. Fission Gas Pressure at 64,500 MWd/tonne
Average Burnup in 36 in. Long Fuel Column.

FIGURE 3.C.3. Plenum Design Curves for 10 Mil and 11 Mil Effective Cladding Thickness Based upon Allowable Secondary Creep Strain of 0.05%.

FIGURE 3.C.4. Comparison of Peripheral Flow Suppressors

2. Systems and Instrument Analysis

H. G. Johnson

a. Emergency Core Cooling Analysis

C. D. Flowers and H. G. Johnson

(1) ECCS Concept Simplification. A simplification of the ECCS concept, which was described in the November 1968 Progress Report,⁽¹⁾ is to run a single 18 in. pipe line from the discharge of each primary pump to a header just upstream of the check valves, in each loop, but continuing with two parallel 12 in. pipe runs from each header to the reactor inlet. This arrangement permits one flow controller per primary loop instead of two controllers per loop.

A digital simulation model of the new arrangement was developed and used for analysis of double-ended breaks at the check valve header, with and without simultaneous check valve failures. The maximum fuel cladding temperatures were shown to be safely below the allowable temperature limit, which has been established in the safety criteria as 1600 °F for major accidents, for the check valves closing upon reverse flow, and for either one or both valves failing to close.

(2) Natural Circulation Using the HTS. Studies were completed to establish the effect of the following variables on primary loop natural circulation cooling of the shutdown reactor:

- Elevation difference between reactor core and IHX. The range studied was from 12 to 21 ft.
- Check valve pressure loss characteristic.
- Free sodium volume in the primary pump barrels.

1. BNWL-937, FFTF Monthly Informal Technical Progress Report, Battelle-Northwest, November 1968.

All three variables have an important influence on the primary coolant flow and reactor coolant temperature responses. These and other variables must be given careful consideration during design. A change in a variable in one direction may be beneficial to natural circulation following the unlikely occurrence of loss of all electrical power, but the change may also be detrimental to other emergency conditions.

b. Analog-Hybrid Simulation of the Reactor and Heat Transport System

R. D. Benham and A. L. Gunby

(1) Hybrid-2 Simulation. The newly-installed hybrid computer (AD-4 analog machine, PDP-9 digital machine) has been programmed for the FFTF system simulation and is fully operational with multiple coolant circuits.

3. Plant and Facilities Analysis

R. J. Hennig

a. Blast Containment Analysis

G. L. Fox

Analysis of possible vessel cover loadings from a DBA is continuing. A new model of the ASPRIN, computer code which defines an effective impact pressure on the cover and upper vessel wall, has been completed.

A comparison of DBA pressure histories within vessels of 2 and 7 in. wall thickness is shown in Figure 3.C.5. The gas bubble pressure occurs within the core. The cover impact pressure occurs primarily on the cover and the upper vessel region. Vessel 1 expands more rapidly causing the gas bubble pressure to decrease the fastest. Because of upper vessel deformation, Vessel 1 transmitted only 6% of the DBA energy to the cover, whereas Vessel 2 transmitted 17%. Thus a thicker vessel wall presents a greater hold-down problem.

FIGURE 3.C.5. Pressure History for Reactor During DBA with 8 ft Inert Gas Gap over Core.

(1) Design Pressures for FFTF Vessel Cover. A comparison of reactor vessel concepts includes analysis of cover integrity under Design Basis Accident conditions. It is assumed that the cover should be able to withstand pressures which build up rapidly from DBA and yet are not sufficient to rupture the vessel. Definition of an effective steady-state pressure is complicated by lack of characterization of the pressure history of fuel-coolant vapor explosion. However, by using appropriate assumptions, it appears that preliminary effective pressures can be established as follows:

	<u>Maximum Cover Pressure</u>
Vessel No. 1	
Piped Bottom Inlet	
2 in. wall 18 ft diameter	1150 psi
Vessel No. 2	
Annular Inlet	
7 in. effective wall	
19 1/2 ft diameter	3150 psi*

* *Reduced because of 1500 MW-sec energy limitation in vessel wall.*

b. Machinery Dome Missile Protection

A. B. Webb

The missile potential of an unrestrained reactor head resulting from a Design Basis Accident was evaluated. Figure 3.C.6 shows the impulse-momentum relation for a typical reactor head. As stated in the preceding article, loading the reactor head during a Design Basis Accident is dependent on the reactor vessel thickness. Using Case 1 (2 in. vessel thickness) an allowable 40 psi/sec impulse is transmitted to the head, while Case 2 (7 in. wall thickness) would allow a 70 psi/sec impulse. From Figure 3.C.6, the head velocities are found to be: 76 ft/sec and 133 ft/sec for Cases 1 and 2, respectively.

FIGURE 3.C.6. Reactor Head Velocity from DBA Impulse Loading.

One concept to arrest the head/missile is an energy-absorbing barricade which converts the kinetic energy of the head into mechanical work. Figure 3.C.7 shows the required depth of aluminum honeycomb absorbing material as a function of head impact velocity. Figure 3.C.7 calculations are based on a 25 ft diameter barricade cross-section. A heavy steel plate on the face of the barricade distributes the load from the smaller diameter reactor head over the barricade cross section. The required depth of the barricade is inversely proportional to the effective cross-sectional area. For the 2-impulse Cases 1 and 2, Figure 3.C.7 shows that the required thicknesses of energy absorber are, respectively, 1.75 ft and 5.6 ft. The head is arrested by dissipating its kinetic energy through deformation work as it crushes the barricade. The aluminum honeycomb energy-absorbing material can be crushed to 25% of its original depth. However, to provide a margin of safety, Figure 3.C.7 is based on a crush depth of 50% of original depth. In designing the energy-absorbing plug arrester for the machinery dome for the Fermi Power Plant, APDA assumed a crush depth of about 40% of original depth.⁽¹⁾

Although it is technically feasible to arrest the reactor head/missile with an energy-absorbing barricade, restraining the head to prevent it from becoming a missile is also being studied.

1. N. H. Wagner, *Evolution and Design of the Machinery Dome for the Enrico Fermi Atomic Power Plant*, APDA-303.
June 10, 1968.

FIGURE 3.C.7. Required Energy Absorber Thickness for Various Reactor Head Velocities.

D. SAFETY AND RELIABILITY ANALYSIS

D. E. Simpson

1. Reliability Analysis

R. A. Harvey

a. Two-Loop Operation of the Heat Transport System

A prediction was made of the amount of time that two-loop operation of the Heat Transport System might be required. The prediction was based on the assumption that two-loop operation would result only from half of the pump failures. The total failure rate for each pump was considered as 0.014 failures per thousand hours.⁽¹⁾ Counting both the primary and the secondary pumps, the total failure rate for failures that might result in running the reactor with one loop shut down is 0.014 failures per thousand hours per loop, or 0.042 failures per thousand hours for three heat transport loops. Since the FFTF run time is anticipated at 1500 hours (nine weeks) the expected number of failures per run that results in two-loop operation is 0.063. This gives a failure probability of approximately 0.06 or 6%; therefore one might expect that one out of about sixteen runs would require operation on two loops instead of three as a result of pump failures.

It was assumed that spare parts were available and on site for rapid replacement.

b. Effectiveness Goals

H. C. Martin

A brief study indicated that the emergency power source would have negligible effect upon reactor availability provided

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1. R. A. Harvey, et al., Technical Basis for FFTF Main Heat Transport System Concept and Concept Configuration Selection, BNWL-554, Battelle-Northwest, Richland, Washington, September 1967.

that reactor operation would continue if only one of the four emergency sources were unable to start.

A simulation study using rough data indicated that the failure rate for protective system loss caused by logic circuit failure is in the order of 10^{-15} per year.

2. Safety Analysis

D. D. Stepnewski

a. Analysis of Cold Sodium Incident

C. L. Fies and A. L. Gunby

The reactivity effect resulting from the injection of cold sodium into the FTR core has been analyzed at low power with one heat transport loop operating at low flow to maintain full core ΔT (800 °F - 1200 °F). This incident can be considered as representative of the accidental opening of the control valves in two primary loops. The variation of flow and temperature as a function of time was investigated with the FFTF hybrid simulation of the primary pumps, hydraulics, and vessel inlet plenum.

The results of the transient indicate that the inlet plenum temperature drops 400 °F in 20 seconds based on a 270 ft³ inlet plenum volume.

The flow and temperature versus time (270 ft³ inlet plenum volume) data from the hybrid simulation were used as input to the MELT-II computer code.

The resultant reactivity ramps were computed for Doppler coefficients of -0.003 and -0.004. In both cases the reactivity ramp during the first three seconds is approximately 30¢/sec. This is well within the capability of the scram system. The computations did not include a reactor scram, however. After three seconds the Doppler effect turns the incident around as

a result of the fuel temperature increase. The power rise as a result of the cold sodium incident was shown to be approximately 200 MWt in four seconds. The power increase does not vary appreciably with the Doppler coefficient.

The hot spot fuel temperatures for the two cases are shown in Figure 3.D.1. For a Doppler coefficient of -0.003 the fuel begins to melt 20 seconds after the incident begins. For a Doppler of -0.004 the fuel temperature is 200 °K below melting at 20 sec.

b. Argon Circulation in Reactor Heat Transport System

R. D. Peak

Entrapped argon bubbles will circulate along with sodium in the primary sodium system during normal operation. Such bubbles in passage through the reactor core may cause reactivity perturbations, thermal fatigue of fuel cladding, and fuel pin vibrations.

A study was made of the potential for dissolution or growth of an argon bubble while circulating in the Reactor Heat Transport System. Because of saturation potentials, an argon bubble would have a tendency to dissolve in passage to the intermediate heat exchanger, and thence, if it survived, to grow in passage back to the reactor. This latter tendency is important because it provides potential for argon bubbles to collect in high-pressure pockets, such as in valve bonnets, along the piping leading to the reactor.

Pockets of high-pressure argon gas in primary piping leading to the reactor pose hazards each time one or more primary pumps slow down. The pressure reduction that accompanies pump speed reduction will permit a gas pocket to expand, possibly to the extent of venting gas into the sodium flowing into the reactor. The consequences of this vented gas in these

FIGURE 3.D.1. Cold Sodium Incident Hot Spot Fuel Temperature.

circumstances will have to be evaluated during preliminary design when the conditions of this hazard are better defined.

Apart from providing potential for creating high-pressure gas pockets, the actual amount of argon dissolved in sodium is too small to pose any apparent hazard. Estimates were made of the total amount of argon released and the reactivity effect⁽¹⁾ when this total amount of gas is introduced into the reactor as one bubble. These estimates are listed below.

Reactor Operating Temperatures		Total Argon Released from Solution in One Standby HTS Loop	Reactivity Effect When Argon Passes Through Core in One Bubble
Inlet, °F	Outlet, °F		
500	800	240 cm ³ at 1 atm, 500 °F	0.004\$ worth
600	900	830 cm ³ at 1 atm, 600 °F	0.013\$
800	1200	12. liters at 1 atm, 800 °F	0.18\$

The configuration of each HTS primary loop guarantees that the argon released from solution will not collect in one bubble and, thus, the actual reactivity effect will be much less than that estimated for one bubble.

c. Design Safety Criteria

K. L. Berrett, C. L. Fies, J. W. Hagan, and R. D. Peak

Configuration Control Board approval has been obtained on ten additional Design Safety Criteria. These are Site Facilities-22, Service Piping-23, Radioactive Waste-24, Heating and Ventilation-25, Fuel Protection-26, Reactor Core-31, Reactor Vessel-32, Nuclear Control-33, Closed Loop-68, and Inert Gas Receiving and Processing-82. The Introduction and Severity Levels, Section I of BNWL-760, Design Safety Criteria for the Fast Flux Test Facility was also approved.

1. Fast Flux Test Facility Monthly Technical Progress Report, BNWL-964, Figure 3.18, p 3.66 (C. L. Fies). Battelle-Northwest, Richland, Washington, December 1968.

IV. DEVELOPMENT

D. L. Condotta

A. ENGINEERING DEVELOPMENT

J. R. Carrell

1. Sodium Components Development

T. W. Withers

a. Sodium Pumps

W. Babcock

Technical concepts which were recommended by Westinghouse for the primary FFTF pump under contract AT(30-1)-3762 in the January 16 technical meeting are as follows:

- (1) A wound rotor motor with liquid rheostat control capable of a 2.8 to 1 speed control without electric brake.
- (2) A Fermi pump design using a double suction impeller with a design speed of 1110 rpm. The pump is to have a rated capacity of 13,000 gal/min at a design head of 500 ft TDH from 22 ft NPSH. The theoretical maximum capability of the pump, using affinity law extrapolation, is 13,000 gal/min at 570 ft TDH at 1185 rpm full-load motor speed.
- (3) A pump shaft design consisting of a stepped shaft with one sodium bearing at the impeller end and one oil lubricated bearing above the shaft seal.

In addition to the above recommendations, Westinghouse has confirmed that the design temperature for the pump will be 1200 °F without jeopardizing lower temperature operation, except for the sodium bearing which is limited to 1050 °F until proof tested. Westinghouse will use potential 1200 °F materials for the sodium bearing. However, until the bearing is proof tested for 1200 °F, the operating limit of the pump

will be 1050 °F. It is anticipated that the only changeout required for pump operation from 1050 to 1200 °F is a changeout of the bearing cartridge and shaft journal. The pump will be designed to facilitate this changeout when required.

b. Intermediate Heat Exchangers

L. A. Whinery

A BNW-recommended work scope establishing technical bases for performing supporting IHX design studies by LMFBR Program contractors has been prepared and submitted to RDT for concurrence and implementation. The work scope is generally structured for mutual compatibility between both near-term LMFBR and FFTF needs. RDT/LMEC/W/BNW concurrence on technical aspects of the work was established during meetings held on February 6 and 7 at RDT Headquarters offices. It was agreed that the work will be incorporated within two of the three LMFBR contracts with Westinghouse, Foster-Wheeler and Combustion Engineering which are currently in force. RDT, through the AEC Field Offices, will request cost and schedule estimates from the above contractors as a first step in initiating the work.

2. Equipment Development

C. A. Munro

a. Control Rod Drive Mechanism Development

J. C. Noakes, P. K. Telford, J. R. McBride, B. C. Fryer, D. E. Rasmussen, and R. Kolowith

(1) Roller Nut Drive Initial Sodium Tests. A meeting was held in Galveston, Texas, to discuss the use of the replacement N.S. Savannah Roller Nut Control Rod Drive Mechanism. It was concluded that this drive would meet the initial sodium test requirements pending additional material compatibility review. Two areas of the drive may require modification prior

to sodium testing. These are the decelerator ram and the lead screw anti-rotation spline. Changes to the test program and inspections were recommended to obtain the greatest amount of data from the initial sodium test. The drive unit will be shipped to LMEC along with the control console for this test series.

(2) Modified Fermi Drive Latch Tests. A stand for testing the modified Fermi control rod latch has been assembled and preliminary operation is underway to evaluate the latch assembly. The test stand (shown in Figure 4.A.1) is equipped to cycle the latch automatically through its test sequence of latching, loading, and unlatching. The stand can also be used to determine the holding force, magnetic force, and other latch characteristics as required.

The latch assembly is shown in Figure 4.A.2. In operation the cam opens and closes the latch fingers over the control rod spool. The cam is guided by the cocking tube and actuated by the latch rod. The drive extension carries the entire assembly. The latch is cycled at one cycle per 30 seconds with a load of 2350 pounds. The test assembly has been disassembled after 632 and 1651 cycles. Photographs were taken to record the wear patterns on the mating parts. Static tests to determine the required latch rod holding force versus latch load were performed initially and after completion of the above cycles (see Figure 4.A.3). These latch rod holding forces, which are less than the initial values, indicate an increase in the coefficient of friction between the latch fingers and the locking cam.

(3) FTR Oscillator Development (P. K. Telford). A statement of work for FTR oscillator development has been drafted for use by potential oscillator development contractors. A supporting document entitled "Design Survey and Conceptual Definition of a Reactivity Oscillator for the FTR" is also in

Neg 0684709-2

FIGURE 4.A.1. *Modified Fermi Control Rod Latch Test Stand*

4.5

Neg 0684334-1

FIGURE 4.A.2. Modified Fermi Control Rod Latch Test Assembly

BNWL-1006

FIGURE 4.A.3. Latch Rod Holding Force Versus Latch Load for Modified Fermi Control Rod Latch Test Assembly

preparation which includes drawings depicting the basic features of the oscillator. As a result of discussions with FFTF Reactor Physics and Operations, provisions for variable amplitude have been added to the basic requirements of the oscillator.

b. Closed Loop Tube and Hardware Development

J. W. Kolb, R. Kolowith, and B. G. Smith

(1) Reference Concept Improvements (J. W. Kolb and B. G. Smith). Further study of the Reference Concept closed loop and open loop top works and cross-arm piping layout and arrangement indicates that nine full-size closed loops or nine open loops (or any mixed combination) can possibly be provided with complete interchangeability of positions, including sodium sampling capability for open loops in any position. The arrangement has three interchangeable modules consisting of three test positions each with double-walled interconnecting piping for each test, sodium sampling for each test, and confinement cans encasing the piping, shielding, and insulation on each module. When closed loop internals are in place in the spool, the sodium sampling line is blanked off by the internals, and connection to the inlet and outlet coolant lines is made. When open loop internals are in place, the coolant lines are blanked off and connection to the sampling line is made. Figures 4.A.4 and 4.A.5 show the arrangement of loops within the module. Also shown are the confinement cans which are provided to prevent or reduce spread of contamination in the event of a sodium spill during refueling or rupture of the double-walled interconnecting piping.

The advantages of complete interchangeability of open and closed tests within the module and of module interchangeability are clear. The advantage of having nine possible in-core closed loop positions with only six closed loop cells

SK-3-14461

FIGURE 4.A.4. Test Loop Block Assembly (Sheet 1 of 2.)

SK-3-14461

FIGURE 4.A.5. Test Loop Block Assembly (contd) (Sheet 2 of 2.)

may not be as clear. In order to profit from this capability, it would be necessary to provide a jumper arrangement in the primary loop cells which would permit remote connection of any one of three in-reactor tests to a given primary cell heat transport system. This does not appear to be too difficult. With this provision, a total of three closed loop cross-arm piping inner wall failures, or in-reactor tube failures, could occur without the need for emergency shutdown for removal and replacement of the modules. Operation of six closed loops and three open loops could possibly continue until a planned shutdown for replacement or maintenance of other equipment is scheduled. This should be beneficial from the plant availability viewpoint.

(2) Closed Loop Nozzle Closure Program - Mechanical Seals (J. W. Kolb and R. Kolowith). Conceptual design of the creep-compensated connector assembly is being modified to accommodate the "engine block" concept employed in the Reference Core Concept. Restriction of the seal to a 1/8 in. square cross section has necessitated extensive conceptual design changes.

Preliminary conceptual designs of the manual actuator, remote actuators, and actuation test setup are being prepared.

The interconnection diagram of the instrumentation connector into the seal test facility is being prepared. This diagram is being made to show the test capabilities of the seal test facility and the compatibility of the necessary recording and test equipment.

(3) Closed Loop Nozzle Closure Program - Welding (J. W. Kolb and R. Kolowith). The primary objectives of this subtask are to demonstrate the feasibility of employing TIG welding as a means for making the final reliable closure on the closed loop nozzle spool piece, and to demonstrate the

techniques required for remotely cutting the weld in order to facilitate repeated removal and replacement of the closed loop test specimens.

Remote closure welds have been performed on two simulated closed loop test pieces during this report period. Test weld No. 1 was performed while temperature of the test specimen was maintained at 250 °F. Sodium was held at a level of one inch below the simulated nozzle plug at the above temperature for approximately two hours and then drained back to the storage reservoir prior to weld initiation. Test weld No. 2 was performed at a temperature of 300 °F, while the sodium level was maintained in the test piece at approximately two inches below the weld area. General weld bead appearance of both test specimens was good.

3. Development Facilities

K. G. Toyoda

a. Hydraulic Core Mockup

H. Leigh, D. L. Ballard, J. L. Poe, C. H. Henager,
L. M. Polentz, L. R. Sweetin, P. M. Jackson, and
G. W. Riedeman

The objective of the Hydraulic Core Mockup Program is to provide development and testing support for the resolution of the FTR vessel and core hydraulic and mechanical design problems. Preliminary test data from the one-third scale inlet model (toroid design) on the effect of rotating the inlet pipe discharge nozzles from 0 (directed radially to model axis) to 90 deg (tangentially) show the following conclusions.

- Flow distribution to the "core" is not affected.
- Fluid mixing is increased.
- Vibration in the model is reduced.

Experimental evaluations of electrolyte injection techniques

for flow mixing studies show that a 10 ppm salt concentration may provide ample conductivity variations for transit time measurements between electrodes more than four feet apart.

b. Sodium Core Development Facilities

K. D. Hayden

Changes to the design of the Sodium Facilities Building by HES were completed and invitations to bid were sent out by the AEC to prospective bidders.

The project proposal and design criteria for the Fast Reactor Thermal Engineering Facility were completed and are now being circulated for BNW approval signatures. The completed project proposal has a sodium loop with 600 gal/min, 250 lb/in.² head, 1200 °F temperature rating, and 1-1/2 MW electric heating capacity to the test section.

c. Sodium Studies Facility

J. D. Berg and L. J. Defferding

The objectives of this section and description of equipment were given in BNWL-915.⁽¹⁾ Status of facilities is given in Table 4.A.I.

4. Engineering Testing Services

R. L. Roske

Table 4.A.II lists tests administered or processed by Engineering Testing Services in which the status changed during the reporting period. Objectives were given in BNWL-915.⁽¹⁾

1. Fast Flux Test Facility Monthly Informal Technical Progress Report, BNWL-915. Battelle-Northwest, Richland, Washington, October 1968.

TABLE 4.A.I. *Sodium Facilities Status - February 1969*

Small Component Evaluation Loop	The loop was charged on 1-24-69 and flow established in all circuits. The bulk sodium temperature was increased to 800 °F and system oxide content reduced to <10 ppm by cold trapping. A startup checkout has established the reliability of the safety and control circuits and the loop is now operating unattended.
Radioisotope Transport Loop	This loop remained in operation at 1.5 gal/min, 1300 °F with 450 F° ΔT from late in December until 1-28-69. The system was shut down on this date after a sodium leak developed at the base of the sodium to air heater changer.
Small Heat Transfer Loop	This loop was operated approximately 48 hours during this report period for additional operator training and qualification. Since this brief period of operation the loop has remained shut down for general systems checkout.
Static Sodium Pots	Testing of pressure transducers on Sodium Pots 3 and 4 has continued throughout this report period. Pot No. 2 started up on 1-16-69. After a period of check-out, testing was resumed on the transducer in this facility. This test continued until completion on 1-27-69. A new pressure transducer was installed in Pot No. 1 on 1-24-69.
Sodium Purification Loop	This loop has operated continually throughout this report period at 0.5 gal/min $\cong$ 400 °F bulk temperature, while maintaining the United Nuclear Corporation (UNC) oxygen probe at 600 °F. The cold trap was shut down on 1-24-69 and a known quantity of sodium oxide injected into the loop through the sampling tube. The effects of this oxide addition are presently being observed on the UNC probe.

TABLE 4.A.I. (contd)

Fission Product Screening Loop	This new loop was charged with 40 pounds of sodium on 1-17-69 for checkout of the primary pump and all instruments and controls. Flow was established through the primary circuit with few minor difficulties. The primary circuit is presently operating at 800 °F and low flow.
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TABLE 4.A.II. Engineering Testing Services - FFTF Tests

Test No.	<u>Title</u>	<u>Status</u>
T-1A	<u>Water Flow Rate Test -</u> Reactor Core - BNW Thermal Hydraulics Laboratory	Water flow rate tests completed. Preliminary report prepared.
T-1B	<u>Flow Duct/Tubesheet Sodium</u> <u>Erosion Test - Reactor</u> Core - LMEC (LCTL)	Test apparatus to be shipped to LMEC in February. Testing to start in March.
T-19	<u>Spring Tests - Reactor</u> Core - LMEC	Revision 1 to test request sent to LMEC for approval

B. COOLANT AND PROCESS DEVELOPMENT

B. M. Johnson

1. Corrosion and Chemistry

D. W. Shannon

a. Evaluation of Radioisotope Transport and Deposition

W. F. Brehm and E. A. Kovacevich

The radioisotope transport loop has been operating to establish the behavior of activated corrosion products formed in the core and transported through the heat transport system. The rate and patterns of deposition of radioactivity during operation of the primary heat transport and closed loop systems are of particular concern because of the potentially great effect on the operation and maintenance of the system.

Revised estimates were prepared of the dose rates in the FTR primary piping and components from deposition of the three principal activated corrosion products, based on preliminary observations of their corrosion rates and deposition behavior. These estimates were based on the following assumptions:

- Reactor operation of 1200 °F for one year with sodium containing 10 ppm oxygen under flux conditions to produce 5410 Ci of ^{54}Mn , 4260 Ci of ^{58}Co , and 68 Ci of ^{60}Co .
- Cobalt will be released at one-half the bulk corrosion rate and manganese at twice the bulk corrosion rate.
- Major deposition site for cobalt will be in the circulation pump (hot leg) .
- Major deposition sites for manganese will be in the IHX, cold leg piping, and cold trap.

The resulting estimate ranges of dose rates are shown in Table 4.B.I.

TABLE 4.B.I. Range of Dose Rates, R/hr 2 ft from Component

	<u>^{60}Co</u>	<u>^{58}Co</u>	<u>^{54}Mn</u>	<u>Total</u>
IHX	0-0.7	0-46.0	16.6	16.6-63.3
Pump	1.75	38.8		40.5
Cold Trap	0	0	16.6	16.6
Hot Leg Piping	0.35	7.45	--	7.8
Cold Leg Piping	0-0.3	0-0.7	19.8	19.8-20.8

These radiation levels are believed to be unacceptable for contact maintenance. Extensive further work is needed to (1) substantiate the rate of leaching (corrosion) of each species under high flux heat conditions, (2) determine the rate at which radioactive species move through the system as the result of continuous exchange reactions, (3) define more quantitatively the mechanism of transport and deposition to allow extrapolation of results from the test loop to the reactor systems, and (4) explore methods to induce deposition in selected locations.

High resolution gamma counting of the entire loop with a 40 cm³ lithium-drifted germanium diode has shown that cobalt deposition varies with boundary layer thickness. At each transition region (between sections of different diameter where the boundary layer must be re-established) a peak in the activity is noted. Beyond the transition regions the activity is higher in the smaller diameter pipe segments where the velocity is higher.

The high resolution counting, which necessitated removing sections of the loop because of the high background, followed the termination of a run with an irradiated specimen of a special heat of Type 316 SS containing 5% cobalt. The run was terminated after 754 hr of a 900 hr run because of a sodium

leak which developed in the bottom header of the primary loop heat exchanger. Operating conditions during the run were as follows:

Maximum Test Specimen Temperature, °F	1300
System Thermal Gradient, °F	400
Primary Sodium Flow, gal/min	1.5
Oxygen Content, ppm	<10

Other preliminary results include the observations that residual sodium contributes essentially nothing to the activity of the loops, i.e., the activity is deposited on the walls, and that cobalt activity generally decreases from the specimen area to the heat exchanger.

b. Sodium Cover Gas Interactions

H. P. Maffei

Experimental studies on the reactions of cover gases and their impurities with molten sodium are being conducted to establish allowable impurity levels for FFTF cover gases. This is being accomplished by determining the equilibrium values and kinetic reaction rates between the cover gas impurities and molten sodium.

The investigation of kinetic reaction rates between carbon monoxide and sodium was initiated during the past month. Kinetic reactions were determined by flowing argon cover gas containing 900 ppm of CO over molten sodium and measuring the effluent cover gas with an infrared analysis for CO content. The reaction was studied at temperatures of 500, 650, and 800 °F. It was found that the reaction proceeded as a homogeneous gas phase second order reaction and was sufficiently temperature-dependent that the specifications for CO in the cover gas will very possibly be a function of the reactor operating temperature and markedly less stringent for the shutdown condition.

c. Selection and Testing of In-Line Sodium Analysis Instruments

J. J. McCown

In-line oxygen and carbon meters currently under development by various contractors are being considered for FFTF applications.

(1) Oxygen Meter. The oxygen meter installed in the 324 sodium purification loop has been functioning well over the past two months and has closely monitored oxide concentration changes during two additions of Na_2O and subsequent cold trapping operations. The meter readings agreed with theoretical considerations and were in good agreement with results obtained by wet chemical oxide determinations. In all cases it was necessary to keep the sodium flow rate, sodium temperature, and ambient temperature at the oxygen probe constant in order to get reliable meter readings.

d. Evaluation of Techniques for the Controlled Addition of Impurities to Sodium

J. J. McCown

Initial results of studies on the use of Na_2O additions to control the oxide content of sodium have been encouraging. Two consecutive Na_2O additions were made to the sodium purification loop and the effects monitored both by wet chemical analyses and by an in-line oxygen meter. The oxide content of the loop sodium was changed from about the 3 ppm level to ~24 ppm and then to ~44 ppm. The oxygen meter data fell on the previously determined calibration line and the analytical results gave 24 ppm and 41 ± 5 ppm. After cold trapping the loop for several days the oxygen meter and chemical analysis measurements both returned to the values obtained prior to the start of additions (2 to 5 ppm). Chemical analysis of the bypass tubes indicated >99% of the Na_2O had been dissolved into

the sodium system. These results did not show the large "oxygen buffering" effect of the loop itself that has been noted by others. These preliminary studies were carried on at the relatively low temperature 600 °F, which may contribute to the absence of this buffering effect.

e. Selection and Testing of Radioactive Sodium Analysis Methods

(1) Sodium Studies Facility Analytical Laboratory

(R. F. Keough, K. C. Knoll, L. F. Lust, S. E. Nichols, and E. L. Osborne).

(a) Hydrogen Determination. Consistent recoveries of 95+% of the hydrogen from hydroxide standards have now been obtained with the mercury reflux technique using aluminum as the release agent. The need for aluminum addition has been clearly demonstrated.

(b) Trace Metal Determinations. Preliminary work has been done on a colorimetric method for chromium in sodium based on the use of s-diphenyl carbazide reagent. A detection limit of less than 0.1 ppm has been obtained in the aqueous solution. If a suitable extraction system can be found, it may be possible to improve this sensitivity by as much as a factor of 10. Several promising reagents have been ordered for some of the other transition metals and work on the development of procedures for them is expected to start during the next report period.

f. Selection and Testing of FFTF Sodium Sampling Methods

K. C. Knoll and T. J. Owen

Cutting and crimp sealing one-half inch diameter stainless steel tubing is under investigation. A 4-ton Manco guillotine crimper is being used. Some difficulty has been encountered in obtaining hermetic seals with room temperature crimps.

Success has been achieved using hot crimps in the absence of sodium. Nickel tubing has just been received for evaluation in the sampling program. It is hoped that hermetically sealed crimps can more readily be obtained with this material.

g. Selection and Testing of Cover Gas Instruments and Sampling Systems

G. B. Barton

(1) Gas Sampling System Development. Tests have been completed on a second vapor trap design and results show it to be more effective than the traps described in last month's report. The data indicate that the new design will adequately remove sodium vapor at flow rates up to $150 \text{ cm}^3/\text{min}$. This flow is sufficient to supply the analytical instruments now contemplated but offers no margin of safety. Additional traps of alternative design are being constructed.

2. Thermal Hydraulics

J. M. Yatabe

a. Heat Transfer Characteristics of FFTF Fuel Assemblies, Task CTH-2

The major FY-1969 objective of this program is to develop bundles of electrically heated fuel pins exactly modeling FFTF driver fuel assemblies for thermal-hydraulics tests in support of FFTF fuel design. The current effort is concentrated on developing single instrumented pins which can be subsequently assembled into 7-pin bundles. Procurement of a major order of 30 instrumented pins from the Watlow Electric Company for 7-pin bundle fabrication is underway.

Continuing implementation of the quality assurance program is underway to validate the use of methods of determining boron nitride (BN) density as a function of heat transfer performance of the pins. Three methods are under consideration:

neutron radiography, ultrasonic ring decay, and phosphorescent coatings. Results to date indicate that the neutron radiographs and the phosphorescent coating methods give good agreement, qualitatively, as to location of lower density areas of BN. The ultrasonic ring method appears to need further development and will not be pursued further at this time. Efforts are now focused on improving the phosphorescent and the neutron radiograph methods so that a quantitative measure of BN density can be obtained.

b. Sodium Flow, Hydraulic and Mechanical Testing of FFTF Fuel Assemblies

(1) Coolant Mixing and Flow Distribution. Equipment preparations for the study of coolant mixing and flow distribution in full scale bundles using the direct sample technique is completed, and the prototype fuel assembly will begin on February 14.

(2) Sodium Flow Testing. The purpose of the sodium flow tests is to evaluate the effect of long-term failure modes on FFTF fuel assemblies. As of February, the CCTL Loop (Core Component Test Loop located at Argonne, Illinois) has exposed the Mark I subassembly to 3614 hours in flowing, heated sodium. 2250 hours were at full flow and temperature conditions (400 gal/min, 1060 °F). Chemical analyses performed on samples of CCTL sodium have shown both carbon and oxygen impurities to be ≤ 10 ppm.

General agreement has been reached with ANL on the schedule and required facility modifications for the life testing of the new reference fuel assembly.

(3) Mechanical Testing. It has been postulated that wire wraps may loosen during reactor operation, mainly as a result of differential swelling. If movement of this slack

wire should occur relative to the bundle, the coolant channel configuration could change and give rise to hot spots. In addition, excessive vibration problems could result from the change in geometry.

As a first step to evaluate this problem, a demonstration 19-pin bundle with wires of varying degrees of tension is being built. The bundle will be used to qualitatively examine the forces needed to move the slackened wire.

In addition, plans for water and sodium flow tests to assess this problem more quantitatively in FY-1970 are proceeding.

c. Gas Cooling of Fuel Assemblies During Inspection Process

Experiments to evaluate the gas cooling of FFTF fuel assemblies with transverse flow of coolant are now in progress. Results have been obtained regarding the efficiency of cooling and pressure loss across the bundle for transverse flow with various baffle spacings.

During operation under maximum flow conditions, mechanical vibrations were encountered which were sufficient to crack a glass viewing window in the test apparatus. The exact cause of these vibrations and their impact on fuel cooling is not known. This phenomenon will be studied further as the task progresses to determine if a similar problem could possibly exist in the proposed reactor facility.

d. FFTF Channel Instrumentation - Task IC-23

The major FY-1969 goal of the task is to develop and evaluate individual flow-channel fission product gas disengagement devices to be used in the FEDAL system to locate failed fuel assemblies. The present concept calls for the use of a vortex generator to concentrate the released gas bubbles to

the sampling core located in the center of the duct. The gas enriched sodium sample is then removed from the reactor through the sampling tube for disengagement of the gas and detection of the released fission products.

Initial design evaluations of a range of potential vortex generator disengagement devices has provided the following results. Table 4.B.II gives the average tangential velocity in the vortex, the axial pressure drop with and without recovery vanes, the pressure at the sampling point relative to that downstream of the vortex, and the minimum bubble diameter which can be removed from the outside of the vortex. The length shown in this table is the axial length of the swirl chamber only. About two inches should be added on each end for the vanes. In the calculations of axial pressure drop, an efficiency of 85% was assumed for the swirler vanes and 70% for the recovery vanes.

The following data are, of course, only approximate estimates and experimental data will be required for a more exact answer. However, these data should provide useful input to preliminary trade-off studies to determine what length and tangential velocity (or length and pressure differentials) are most desirable, or at least to specify a range of these parameters within which future investigation can be concentrated.

3. Instrument Development

M. R. Wood

a. Low Level Neutron Flux Instrumentation

C. N. Jackson, N. C. Hoitink, and D. C. Thompson

Accomplishments during this reporting period included gamma irradiation testing and combined gamma irradiation plus temperature evaluation of ^{10}B lined and ^{235}U counters. Conducted at the BNW 3×10^5 Ci ^{60}Co gamma facility, these

TABLE 4.B.II. *Initial Performance Data for a Range of Vortex Generator Designs*

Tangential Velocity ft/sec	Axial ΔP , psi		Sample Tube Pressure Radiation, psi	Minimum Bubble Diameter, microns			
	With Recovery	Without Recovery		*L = 24 in.	L = 18 in.	L = 12 in.	L = 6 in.
For generator core diam = 0.5 in.							
10	0.2	0.7	-3.0	120	155	210	370
15	0.5	1.5	-6.6	75	95	130	230
20	0.9	2.7	-12	55	65	95	160
For generator core diam = 2 in.							
15	0.5	1.5	-3.0	75	95	130	230
20	0.9	2.7	-5.4	55	65	95	160
25	1.4	4.2	-8.4	40	50	70	125

*L *Swirl chamber length*

investigations simulated anticipated FFTF conditions of up to 10^6 R/hr at temperatures to 900 °F. Other laboratory tests investigated analyzer circuit problems and identified signal amplitude and collection time characteristics of a fission counter employing a faster P-10 collection gas.

A relatively small change in the integral bias curve for temperatures up to 600 °F occurred in a 0.25 in. diameter proportional counter, as shown in Figure 4.B.1. At 700 °F, the detector exhibited a persistent high-level noise signal. Subsequent testing at room temperature confirmed the destructive nature of this failure. As shown in Figure 4.B.2, similar testing of a 700 °F rated fission counter caused temporary failure caused by erratic high-level noise at 900 °F. The number 700 at the 10,000 count per second level indicates incipient erratic operation at 700 °F. Environmental testing of a second fission counter (a Type RSN-10A manufactured by Reuter Stokes Electronic Components, Inc.) evidenced quite similar operational performance under combined gamma irradiation and temperature conditions.

A Westinghouse fission chamber filled by the vendor with a faster P-10 collection gas (argon and 10% methane) exhibited an improved collection time of about 125 nsec as compared to the 200 nsec for the usual design. The pulse amplitude from the P-10 filled chamber exceeds that from the older unit by almost a factor of two. These superior characteristics of the P-10 filled fission counter could lead to improved detector performance with regard to gamma tolerance, if the new gas does not alter the basic neutron and gamma sensitivities.

b. Pressure, DP Flow, and Level Sensors for Sodium Service

R. D. Crosier, V. F. Sedlacek, and K. O. Creek

The objective of this task is to develop pressure instrumentation for FFTF in-vessel and piping applications. Required

Neg 0690548-2

FIGURE 4.B.1. Integral Bias Curves Show Combined Effect of 10^5 R/hr Gamma Background and Temperature Ambients of 110, 300, 350, 400, 500, and 600 °F (Preliminary test data)

Neg 0690548-1

FIGURE 4.B.2. *Integral Bias Curves Show Combined Effect of 10^5 R/hr Gamma Background and Temperature Ambients of 110, 600, 700, and 800 °F (Preliminary data)*

activities include evaluation of present technology, procurement and testing of commercially available products, component modification, and improvement and final proof testing.

The Barton-Statham hybrid NaK capillary system, a pressure sensing unit rated at 0 to 200 psia with the transducer rated at 250 °F, continues to operate successfully after almost 900 hours with the bellows operating in sodium at 1200 °F. The drift in zero shift seems to have stabilized. The Consolidated Electrodynamics Corp. unbonded strain gage transducer, rated at 0-250 psia and 600 °F, has operated successfully for two months at temperatures to 575 °F. A Kaman nuclear variable impedance transducer rated at 900 °F failed after 14 hours at 800 °F. The welded seal of the unit failed and it was returned to the vendor for resealing.

c. Signal Lead Connectors

R. D. Crosier, S. K. Meyers, and K. O. Creek

The object of this task is to provide the basis for the development of an instrumentation connector capable of surviving in the FFTF environment. The instrument connector will have tubing and electrical service capabilities and will be remotely operable.

The design of a closed loop instrument connector for fabrication and testing is essentially complete, and a test procedure prepared. This connector concept will also be applied to the open test position, though the temperature requirements of the latter will be lower. Test procedures and facilities are being integrated with tests of the nozzle seal performed under Task IT-6 in order to make optimum use of the same test facility.

C. METALS AND MATERIALS DEVELOPMENT

J. C. Tobin

1. Materials Properties

J. E. Irvin

a. Uniaxial Creep

A. J. Lovell

The objectives of this program are to evaluate the effects of fast neutron irradiation and environment on the pre- and postirradiation uniaxial creep-rupture properties of candidate vessel and core structural materials for the FFTF and to establish the usable materials limits to ensure their safe and reliable performance.

The emphasis during the month has been placed on obtaining control tests on program heat of Type 304 SS in the as-solution treated condition and tests on the Type 304 SS specimens irradiated in MK-B-X022 at 900 °F to a calculated fluence of $\sim 1 \times 10^{22}$ n/cm² (total). The 1000 and 1100 °F test results on as-solution treated controls show the same low creep-rupture characteristics as were observed earlier for the testing at 1200 °F. That is, the creep rates are high and the rupture lives are short compared to the CANEL heat Type 316 SS and the nationally published averages for AISI Type 304. The fact that the Type 304 SS creep-rupture properties are not typical of the average properties of commercially produced heat points out the recognized need to compare all irradiation effects data with the proper set of controls rather than using the data at face value.

The results of 1000 and 1100 °F creep-rupture tests on the program heat Type 304 SS in the solution treated condition irradiated in the EBR-II at 900 °F to approximately 1×10^{22} n/cm² (total) are shown in Figures 4.C.1 through 4.C.3. The

Neg 0690705-3

FIGURE 4.C.1. Effect of Irradiation on the Postirradiation Creep-Rupture Properties of AISI Type 304 Stainless Steel

Neg 0690705-1

FIGURE 4.C.2. Effect of Irradiation on the Postirradiation Creep Properties of AISI Type 304 Stainless Steel

Neg 0690705-2

FIGURE 4.C.3. Effect of Irradiation on the Postirradiation Creep Properties of AISI Type 304 Stainless Steel

six completed tests represent 60% of the tests to be performed on this group of specimens. One additional specimen at each test temperature will be tested when the two tests that are running fail. The stress versus rupture life plots for controls and irradiated specimens tested at 1000 and 1100 °F are shown in Figure 4.C.1. The reduction in rupture life due to irradiation is substantial. The comparison of creep curves for tests at 1100 °F for controls and irradiated tests at 30,000 and 35,000 psi are shown in Figures 4.C.2 and 4.C.3, respectively. These curves show the effect of irradiation on both the rupture life and the first and second stage creep rates. The final measurements of failed specimens are not yet available, but the creep elongations as indicated by the tests shown in Figures 4.C.2 and 4.C.3 are rather low.

b. Compatibility of Boron Carbide with Sodium
and Stainless Steel

L. R. Bunnell and D. E. Mahagin

The objective of this task is an evaluation of the degree of interaction between B_4C and its environment resulting either from direct reaction or from mass transport between the poison and surrounding structural components.

Hot-pressed pellets of boron carbide having nominal densities of 65, 85, and 99% of theoretical were received and characterized this month. Five pellets of each density were measured, weighed and their apparent densities calculated. The open porosity was measured by vacuum-impregnation with water and comparison of wet and dry weight. The results are summarized in Table 4.C.I.

Metallography was performed on longitudinal sections of a pellet of each density. A region of low density appeared at the geometrical center of each pellet type, decreasing in degree as density increased. Etching revealed recognizable

TABLE 4.C.I. Densities and Pore Volumes of Hot-Pressed Pellets

Nominal Density, % Theoretical	Apparent Density, % Theoretical	% Closed Porosity (% of Apparent Volume)	% Open Porosity (% of Apparent Volume)
65	73	3	21
85	85	2	13
99	96	1	2

grains in all of the material, but grains were most prominent in the highest-density pellet. The measured grain size, by random line intercepts, was about 12 μ . The grains were heavily twinned in the most-dense pellet as noted previously on arc-melted material.

Compatibility-transport capsules containing these three pellet types are being heated at present and 100-hour test results should be available next month.

c. Radiation Effects on Absorbing Materials for Control Rods

A. L. Pitner and D. E. Mahagin

The objective of this task is a determination of the irradiation behavior of neutron absorber materials under conditions which are applicable to FTR control rod operating conditions. Only two candidate materials, boron carbide and tantalum, are being studied.

Subassembly X-042 was removed from the EBR-II at the end of Run 32 and two B₄C capsules and two tantalum capsules removed for disassembly. Estimated B¹⁰ burnup is 1.5%. The B₄C capsules were neutron radiographed, and it is apparent that the six-inch boron carbide sample columns had grown in both capsules. The length of the sample that was contained in 0.5 in. OD x 0.065 in. wall cladding expanded ~0.78%, and the sample contained in 0.5 in. OD x 0.035 in. wall cladding

elongated $\sim 1.82\%$. The sample cladding in both capsules appears to have been deformed during the irradiation. The apparent deformation will be studied in more detail during postirradiation examination, now scheduled for early March. The replacement capsules for those removed have been fabricated and shipped to EBR-II.

Sample measurements for the Hanford B_4C irradiation assembly that will operate at 1100 and 1300 °F have been completed. It is expected that the assembly will be ready for charging during the next report period.

d. Fluence Limits

R. A. Moen

Fluence limits were previously established for permanent FTR components such as the vessel and core support structure.⁽¹⁾ The limit was established as the point at which the material in question would still be capable of elongating 10% in a tensile test. To arrive at this limiting fluence, it was assumed that the damage which was produced in the material at the vessel wall or core support structure, for a given total fluence, was the same as the damage produced in the core of a fast reactor.

Russcher recently reported a technique for considering damage as a function of the neutron spectrum utilizing the clustered-10 vacancy and gross displacement models.⁽²⁾ Use of this technique requires detailed knowledge of the spectrum for each datum point used in establishing the fluence limit. A preliminary estimate was made of the fluence limit which might

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1. FFTF Monthly Informal Technical Progress Report, December 1968, BNWL-964. Battelle-Northwest, Richland, Washington.
 2. Fast Flux Test Facility Quarterly Technical Report, July, August, September 1968, BNWL-917. Battelle-Northwest, Richland, Washington.

be specified for the FTR grid structure. The resulting limit appears to be at least an order of magnitude above what was previously specified. In other words, a total fluence of $\sim 2 \times 10^{23}$ nvt might be permitted for the FTR core support structure at 900 °F, with the material still remaining ductile.

2. Fabrication Development

R. N. Johnson

a. Automatic Fuel Duct Welding Equipment

W. F. Brown

Welding equipment capabilities have been reviewed based upon duct concepts compatible with the vertical core. Based upon this review, equipment specifications were changed and new bids were requested. Bids were scheduled for review about February 15. The initial purchase request (for which bids have been received) for equipment would cover a range of tube diameters of 3-1/2 to 8-5/8 in. OD. The latter request will cover a range of tube diameters of 1-9/16 to 5-9/16 in. OD. Equipment delivery is estimated to be approximately six weeks.

An investigation of the problems involved in automatically welding hexagonal duct sections was started recently. In order to maintain the conditions required for depositing uniform high quality welds, the addition of more complex motion mechanisms to the basic welding equipment would be required.

b. FTR Duct Gaging

J. P. Pilger and R. H. Todd

A purchase specification for a gaging system that will characterize the FTR duct is now in rough draft form. Current planning is for the straight hexagonal section of the duct to be accepted at the vendor's facility as a result of go-no go gaging. Upon arrival on site the straight hexagonal section

will be completely characterized. This characterization will include longitudinal and cross sectional features both inside and outside the duct.

A second complete characterization is planned for the outside of the duct following insertion of the fuel bundle, and attachment of the duct end fittings. This second characterization of the outside of the duct will give a profile that can be compared to the initial profile to determine the effects of assembly weldments. This second characterization will also yield the final preirradiation dimensional profile of the duct.

Close liaison is being maintained with the Fuels Department to ensure that this system will be acceptable to them. It is planned to have the gaging system record all dimensional data. This record is to be stored on tape or cards and will have a digitized print-out of the stored information.

Requests for proposals for the gaging system should go to the vendors during March.

c. Fuel Duct Procurement Activities

R. N. Johnson

A task force effort was organized to develop a set of specifications for the fuel duct, to review the materials requirements, and to write a Request for Proposal for the development of fabrication procedures for and manufacture of the prototype fuel ducts. The rough draft of this document has been completed and is being reviewed internally prior to requesting RDT approval. The duct specifications will remain incomplete until a prototype duct design is received.

d. Closed Loops Material Selection

S. M. Gill

A preliminary evaluation and selection of materials for use for construction of the in-reactor tube structure of the

FFTF closed loop system has been completed. Type 316 SS was chosen as the reference alloy for service to 1200 °F and fast fluence to about 1.5×10^{23} nvt for the following reasons:

- (1) In contrast with other possible candidate materials, the 300 series stainless steels are readily available, more is known about them, and more irradiation experience has been accrued. Use of 300 series stainless steels is in accord with the policy of using "state-of-the-art" where possible in the FFTF.
- (2) Type 316 SS meets the required strength requirements.
- (3) Of the 300 series stainless steels, it has the highest code allowable strength at 1200 °F which is based on stress rupture strength at 10,000 and 100,000 hours.
- (4) Type 316 SS has the greatest creep strength for any strain limit selected.

Two other points were made in the evaluation. First, fatigue may be a limiting design factor for the closed loops yet there are little data available on the effects of fast fluence on the fatigue properties of the 300 series stainless steels. Second, a ductility limit for design cannot be selected at this time except arbitrarily. Quantitative post-irradiation ductility values are available only to exposures of around 10^{22} nvt, and extrapolation to fluences in excess of 10^{23} nvt cannot be done with any reliability.

e. Evaluation of Inconel 718 for FTR Safety Rod Springs

S. M. Gill

There is a need in the FTR for a backup safety rod drive which will insert the safety rods into the reactor rapidly for an emergency scram. Such a device must be fail-safe and be able to respond reliably after long periods in a cocked or fully compressed state in elevated temperature molten sodium

or sodium vapor. The device being considered is a compressed coil spring. Because of its excellent stress relaxation properties Inconel 718 has been considered as a primary candidate. An evaluation of the use of 718 for this application has been prepared.⁽¹⁾ While the published values of stress relaxation seem to be adequate for the intended use, no data have been obtained during immersion in liquid sodium under FFTF conditions. Therefore, it was recommended that LMEC perform stress relaxation tests at 1000, 1100, and 1200 °F, using torsion bars as test coupons. A test request, Spring Material Tests, T-19, Rev. 1, has been submitted to LMEC.

3. Component Surveillance

J. W. Helm

The objective of this effort is to establish a program for surveillance and inservice inspection. Objectives include early detection of abnormal or unexpected behavior and verification of expected behavior of critical components in the primary coolant boundary and closed loop systems which have a direct influence on the reliable and safe operation of the FFTF.

a. FTR Dosimetry

J. L. Jackson and J. H. Cox

A revised list of FTR positions requiring neutron monitoring has been completed. The recommendations are based on Figures 4.C.4 and 4.C.5 for the Reference Concept. The positions to be monitored are listed in Table 4.C.II.

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1. R. A. Moen, K. R. Wheeler, and J. E. Irvin. Evolution of Inconel 718 for Application in the Fast Test Reactor, BNWL-930. Battelle-Northwest, Richland, Washington, December 1968.

SK-3-14540

FIGURE 4.C.4. Reactor Arrangement for Reference Core Concept

FIGURE 4.C.5. Core Map of Reference Core Concept

SK-3-14567

TABLE 4.C.II. FTR Monitor Locations

<u>Radial Monitor Locations</u>	<u>Axial Positions in Each Location</u>	<u>Type of Monitoring</u>	<u>Remarks</u>
1. Core vertical center line	Bottom of fuel assembly Core center Top core edge Top of fuel assembly	Flux, fluence, and spectrum	Will require cover penetration or special discharge schedule.
2. Core edge 2 circumferential positions.	Same as core center	Flux, fluence, and spectrum	Will require cover penetration or special discharge schedule.
3. Inner reflector	Bottom Midplane Top	Fluence and spectrum Fluence and spectrum Fluence and spectrum	Will require cover penetration or special discharge schedule.
4. Inner shield edge	Bottom Midplane Top	Fluence and spectrum Fluence and spectrum Fluence and spectrum	Will require cover penetration or special discharge schedule.
5. Outer shield edge - Inside core support barrel	Bottom Midplane Top	Fluence and spectrum Fluence and spectrum Fluence and spectrum	Will require cover penetration or special discharge schedule

TABLE 4.C.II. (contd)

<u>Radial Monitor Locations</u>	<u>Axial Positions in Each Location</u>	<u>Type of Monitoring</u>	<u>Remarks</u>
6. Inside thermal liner	Bottom - support structure	Fluence and spectrum	Will require reactor cover penetration
	Midplane	Fluence and spectrum	
	Outlet nozzle area	Fluence and spectrum	
	Top of thermal liner	Fluence and spectrum	
7. Outside reactor vessel	Junction of vessel with high pressure plenum and others in 6 above	Fluence and spectrum	Will require reactor cover penetration
8. Grid plate 2 radial positions	Center of plate	Fluence and spectrum	Remove monitors with thimble assembly
9. Near core side - Storage carrousel (each carrousel)	Bottom	Fluence and spectrum	Will require cover penetration
	Midplane	Fluence and spectrum	
	Top	Fluence and spectrum	
10. Opposite side Storage carrousel (each carrousel)	Bottom	Fluence and spectrum	Will require cover penetration
	Midplane	Fluence and spectrum	
	Top	Fluence and spectrum	

In order to perform dosimetry, space must be provided through the reactor core, the pool, and in or near the components themselves for installation of a charge-discharge system for the monitor containers. This space must be reserved during preliminary design to ensure access to those reactor areas determined to be critical components and structures, and to those areas where flux, fluence, and spectral data are needed. The locations in Table 4.C.II are the minimum number of locations for flux, fluence, and spectral monitoring.

The in-core and reflector measurements (Items 1 through 4) will be used primarily to test the quality of physics calculations of flux, spectra, and reflector effectiveness, and to determine flux and spectral shifts. Such shifts may result from fuel burnup, rod pattern change, experimental assembly changes and different levels of reactor power. Core component fluences will also be obtained for surveillance use. The other locations (5-10) will be used primarily to obtain data for the estimate of the safe operating life of the major reactor components and structures.

b. Relative Inspectibility of Vessel Concepts

J. W. Helm, J. H. Cox, and J. J. Buggy (W)

As part of the study to select the FFTF reactor vessel configuration⁽¹⁾ an inspection and surveillance evaluation was made of the piped downcomer⁽²⁾ and the annular downcomer⁽³⁾ concepts. Two changes based on discussions with the BNW and Westinghouse design groups were made to the two reference

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1. O. W. Priebe. *Unpublished Data. "FFTF Reactor Vessel Configuration."* Battelle-Northwest, Richland, Washington, December 18, 1968. (Memo to D. P. Schively et al.)
 2. *Print SK-3-14540, BNW Reactor Arrangement Piped Downcomer, December 14, 1968.*
 3. *Print SK-3-45592J, W FFTF Concept SA Annular Top Inlet Vessel Concept, January 9, 1969.*

concepts. First, the piped concept outlet nozzle⁽¹⁾ was lowered to an elevation at the top of the core, the thermal sleeve was assumed to be the same as that in the annular concept, and the siphon breaker was removed from inside the vessel. Second, the guard vessel⁽²⁾ was opened at the top allowing access to the external surface of the vessel. The major components of each concept were rated on a relative scale of potential necessity for inspection and surveillance and of the availability of access and space. Based on this relative rating the piped downcomer arrangement⁽¹⁾ is the preference. For inspection the piped concept has a low relative rating in the inlet nozzle area, while the annular concept has a low rating in the critical core barrel area in relation to the piped concept support structure area. For surveillance the annular concept provides inadequate access and space in the reactor vessel and vessel liner (core barrel of footnote 2) areas.

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1. *Print SK-3-14540, BNW Reactor Arrangement Piped Downcomer, December 14, 1968.*
 2. *Print SK-3-45592J, W FFTF Concept SA Annular Top Inlet Vessel Concept, January 9, 1969.*

V. FFTF FUELS DEPARTMENT

E. A. Evans

A. FUEL ELEMENT DEVELOPMENT

C. A. Burgess

1. Process Demonstration and Development Section

R. E. Bardsley

a. Processing and Equipment - Fuel Preparation and Pellet Pressing

Powder preparation for the first mixed oxide pellets to be made in the Process Demonstration Facility commenced on January 29. Four thousand pellets (UO_2 -25 w/o PuO_2 ; 0.250-in. diam) are to be produced for the offsite Analytical Standards Program. A process flow sheet, sampling and testing plan, cost estimate and fabrication schedule were prepared. Completion of fabrication is tentatively planned for March 1.

All stations in Line No. 1 except the sintering furnace, debinding and two-binder addition gloveboxes have been committed to Pu. The furnace glovebox will receive Pu by mid-month. The granulation - debinding glovebox will remain uncontaminated until all additional equipment has been installed.

The 12,500 insulator pellets for irradiation test fuel pins PNL-6, 7 and 8 have been sintered. Analytical reports indicated that 2/3 of the pellets were acceptable. The remaining 1/3 of the pellets exhibited a high iron content, but this analysis is being rechecked.

b. Processing and Equipment - Preliminary Sintering Tests

D. C. Carlson

Two sintering runs were made (in Line No. 1) to determine the cause of pellet cracking reported last month. The pellets were protected from our exposure after the binder was removed. The dewaxed pellets were loaded into cans and maintained in an argon atmosphere.

These tests showed that pellet cracking was not caused during the sintering step but during the transfer time between binder removal (325 Bldg.) and sintering.

c. Analytical Chemistry Methods

C. A. Strand

Five additional analytical chemistry methods for the analysis of nuclear materials were approved and incorporated into FFTF Analytical Chemistry Methods in Support of FFTF Driver Fuel Fabrication, A-0089. The latest methods approved include: (1) oxygen-to-metal ratio measurements, (2) total plutonium and uranium analysis by X-ray fluorescence, (3) nitrogen analysis, (4) surface area determinations and (5) particle size determination. Methods for measuring homogeneity of mixed oxide pellets, residual halogen contamination of fuel pin component surfaces, stainless steel composition and stainless steel impurity levels remain to be defined and written.

d. Process Development

M. J. Barr

Block Experiments - (MJB-10, Density Control Through Addition of Pore Forming Binders). The addition of small amounts of specific pore-forming binders is considered to be an effective technique for density control. To determine the

effect of a pore-forming binder on density and microstructure, methyl cellulose (a known pore former) was added to a control batch at three percentage levels. The following effects were noted:

- (1) The percent shrinkage is still a function of press pressure and was not significantly affected by the increase in overall binder percent.
- (2) A slight pressure effect on sintered density was observed as the percent binder was increased.
- (3) A slight reaction between the molybdenum boat and pellets with higher binder percentages was observed. The reaction appears to be the result of poor gas circulation in the small boats used for test groups.

2. Subassembly Development Section

J. W. Thornton

a. FFTF Fuel Fabrication Demonstration Facility (Project BCE-023)

Final acceptance of the facility was made during this month. The last major piece of equipment (RCA automatic gaging equipment) was completed this month. M. D. Jackson witnessed a performance test of the equipment and it was accepted as meeting specifications. The equipment was shipped to arrive at BNW early in March.

b. Test Subassemblies

J. W. Thornton and D. E. Blahnik

Fabrication of the 217-pin coolant mixing test subassembly was completed on February 11.

This subassembly was designed to inject a salt sample into the flow stream near the bottom of the subassembly. The coolant is then sampled at various points along the length of

the subassembly and from various channels (positions between pins) in the subassembly. The concentration of the salt samples are then measured by chemical analysis. Coolant mixing characteristics can then be determined from the salt concentration information.

3. Special Products Fabrication Section

E. T. Weber

a. Irradiation Test Assembly Fabrication

R. M. Crawford, R. J. Evans and E. T. Weber

Pins for the EBR-II subassembly PNL-4 were completed. These include the 37 pins for the subassembly plus eight spare pins. The following summarizes the major variables in the fabricated pins.

<u>Group Number</u>	<u>Pellet TD %</u>	<u>Pellet Diam, in.</u>	<u>Local Smeared Density %</u>	<u>Number Required</u>	<u>Number Completed</u>
1	89.9 ± 1	0.2135 ± 0.005	85 ± 1.4	8	8
2	91.6 ± 1	0.2115 ± 0.005	85 ± 1.4	8	8
3	93.0 ± 1	0.2135 ± 0.005	88 ± 1.4	13	13
4	94.0 ± 1	0.2135 ± 0.005	89 ± 1.4	8	8
5	96.0 ± 1	0.2135 ± 0.005	85 ± 1.4	8	8
				45	45

b. Fuel Fabrication for Irradiation Test Pins for Snout Capsules, Task FP-3

W. E. Warden and R. J. Evans

Fuel pellets for Xenon tagging experiments were made during this reporting period. These pellets [75% UO_2 (20% ^{235}U enriched 25% PuO_2)] were sintered in the NRC Furnace to keep from interfering with the Pereny Furnace sintering of PNL-4 pellets. Fabrication of these pellets marks the fourth different

enrichment and the seventh different blend (including sources of materials) from which fuel pellets have been made during the fiscal year. Each of these blends has reacted differently to processing and sintering.

B. QUALITY ASSURANCE - FUELS

H. G. Powers

1. Development and Control of Inspection and Testing

a. Fuel Cladding Testing

J. Ryden and A. C. Callen

Measurement of artificial notches in test standards is being studied. Study of four Type 316 SS standards indicates that significant measurement error exists in the replication technique, at least in measurement of the replica, per se. The replication technique is, of course, necessary for non-destructive measurement of the ID notches.

b. Eddy Current Cladding Tester Development

J. Ryden and E. F. Perrizo

A coil form was designed and fabricated to be used to evaluate further the potential of using two encircling coils as drivers and two or more search coils as pickups. The effective field of just two search coils was determined. The results indicated that it may be feasible to use multiple search coils, mounted around the circumference of the coil form, to provide 100% circumferential inspection. Coils are currently being wound for a test of this hypothesis.

c. Ultrasonic End Closure Test

J. L. Thompson, R. W. Steffens and J. Ryden

Evaluation of the feasibility of an ultrasonic test for the FFTF fuel pin closure weld was continued. Very encouraging results were obtained on the normal weld configuration through an unusual test technique. The method employs a focused transducer with the ultrasound incident to the tube at a very high angle; quite near the critical shear angle.

2. Special Studies

a. Fuel Pellet Homogeneity Studies

D. A. Stranik

Microprobe analyses were performed on a number of sintered, mixed oxide pellets. These analyses are being reviewed to correlate the frequency of PuO_2 enriched areas with the increase or decrease in PuO_2 concentration from the nominal of 25 wt% PuO_2 - 75 wt% UO_2 matrix. As the PuO_2 concentration increases above 25 wt% the PuO_2 enriched areas decrease. Preliminary analyses indicate that above 37 wt% PuO_2 or below 17%, no enriched or depleted PuO_2 areas were apparent.

C. CLADDING DEVELOPMENT

T. T. Claudson

1. Cladding Development and Processing

J. C. Tverberg

a. Procurement Activities

R. J. Lobsinger and R. C. Aungst

Attempts to obtain Inconel X-750 prequalification fuel pin plenum springs were unsuccessful because of vendor exceptions to development specification A-0080R. The specification was revised as follows:

<u>Spring Detail</u>	<u>Original Requirement</u>	<u>Revised Requirement</u>
Free Length	$7 \pm 1/64$ in.	$7 \pm 1/8$ in.
Coils	112	Optional
Load at Length (4.9 $\pm$ 0.2 in.)	1.65 ± 0.01 lb	3.0 ± 0.1 lb

Rebids from vendors on the revised specification for fuel pin plenum springs were not responsive to requirements for surface cleanliness which stipulate 0.0005 mg/in.^2 maximum halogen contamination. Vendors were unable to meet this requirement and it will probably have to be waived to obtain springs. Springs obtained with this waiver will be cleaned and qualified at BNW with respect to halogen contamination prior to use.

Approximately 1860 feet of 0.187 in. OD by 0.014 in. wall Type 316 SS tubing was shipped from Carpenter Technology Corporation for use as plenum tubes during fabrication of the prequalification fuel pins. All material meets requirements of the revised prototypical development specification HWS-1001, Appendix D.

2. Cladding Evaluation

J. J. Holmes

a. Biaxial Stress-to-Rupture Studies

R. W. Barker

Biaxial stress-to-rupture tests were performed on three irradiated Type 304 SS test samples from which the residual sodium had been removed. All tests were performed in argon at 1200 °F. One test piece failed in two locations, namely, in a postirradiation-induced defect and in the cladding-weld interface. A second test failed in the clad-weld surface. The third test ruptured violently upon loading at well below design test pressure. The rupture location for this latter test was generally near the lower end plug. Figure 5.C.1 illustrates the locations of all ruptures found in the weld areas. The rupture time for the two nonviolent failures was significantly reduced from times obtained earlier for as-received material, as can be seen in Figure 5.C.2. The percentage reduction in rupture time is similar to other creep data obtained from uniaxial creep test specimens irradiated in EBR-II.⁽¹⁾ The occurrence of failures in and near the weld areas, however, suggests that the data are invalid in terms of cladding rupture life.

b. Damage Analysis

J. L. Straalsund and H. R. Brager

Density measurements were begun on an EBR-II safety rod thimble. This thimble has accumulated approximately 9.6×10^{22} n/cm² total neutron fluence. Initial results give peak swelling values of about 2.8% at an irradiation

1. *Personal Communication, A. J. Lovell, Battelle-Northwest, Richland, Washington, January, 1969.*

Neg 0690647-1

FIGURE 5.C.1. Location of Ruptures on Type 304 SS Biaxial Stress Test Specimens Irradiated to $0.8 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$)

Neg 0690647-

Figure 5.C.2. The Effect of Stress at 1200 °F on the Rupture Life of Type 304 SS Irradiated to 0.8×10^{22} ($E > 0.1$ MeV) from Which Residual Sodium Has Been Removed

temperature of about 800 °F. This is within the scatter band of the predictions based on lower fluence data. The density measurements will be completed next month.

Examination of the effects of fast neutron irradiation on the microstructure of Type 316 SS is continuing. An unirradiated thermal control sample of Type 316 SS annealed 3045 hours at 900 °F (equivalent to time in reactor) was examined. Except for a few inclusions normally found in commercial steels, the unirradiated thermal control sample did not contain visible precipitates either at grain boundaries or in the matrix (Figure 5.C.3.a). Last month, micrographs were presented showing rod and nearly spherical precipitates in this Type 316 SS irradiated in EBR-II at 900 °F to 8×10^{21} n/cm² ($E > 0.1$ MeV), (X-022). The presence of precipitates in irradiated Type 316 SS and their absence in the unirradiated thermal control material of the same heat, indicates that fast neutron irradiation induces precipitation in this alloy.

The identification of the rod shaped precipitates is being investigated by X-ray and electron diffraction of extracted precipitates. The long axes of the rod shaped precipitates have been determined to be parallel to the four $\langle 111 \rangle$ directions in the matrix, Figure 5.C.3.b. The "nearly spherical" precipitates in the irradiated Type 316 SS yielded electron diffraction "d" spacings corresponding to $M_{23}C_6$ and appeared to be polyhedral in dark field. This polyhedral shape is the normally observed morphology of $M_{23}C_6$ in unirradiated austenitic stainless steels at high temperature. Therefore, fast neutron irradiation appears to enhance nucleation and growth of precipitates in Type 316 SS.

Neg 6668A

17,250X

FIGURE 5.C.3.a. Bright Field Electron Micrograph of Unirradiated Thermal Control Sample of Type 316 SS Annealed 3045 hr at 900 °F in Static Sodium

Neg 6487-B

17,250X

FIGURE 5.C.3.b. *Bright Field Electron Micrograph of Type 316 SS Irradiated in the EBR-II at 900 °F to 0.8×10^{22} n/cm² ($E > 0.1$ MeV). Zone axis of foil is $\langle 111 \rangle$.*

The potential of the scanning electron microscope and replica electron microscopy in the study of voids and denuded zones near grain and twin boundaries were explored.

It appears that, in the absence of an extensive specimen preparation development effort, the scanning and replication techniques will not give usable results. Therefore, no further work along these lines is anticipated.

3. LMFBR Fuel and Cladding Information Center

F. R. Shober

Data processing, has continued with the addition of data for 70 new experiments. This brings the total tests available on Master File No. 1, austenitic stainless steels, to 1971. A retrieval of all data on file for austenitic stainless steels was made. Graphic plots of yield strength, ultimate strength, uniform elongation, and total elongation for selected fluences versus test temperatures, as well as stress versus strain plots for representative tests, were provided to the Metals and Materials Development Group of the FFTF Development Department.

D. FUELS EVALUATION

J. E. Hanson

1. Thermal Irradiation of $\text{UO}_2 + \text{PuO}_2$ Fuel Pins

G. E. Culley, E. O. Ballard, S. H. Christensen,
and N. D. Mills

a. Irradiations in GETR

Irradiation of four capsules to 50,000 MWd/tonneM continues in GETR. The approximate burnup for capsules PNL-59-7 and 8 is 40,000 MWd/tonneM, and 31,000 MWd/tonneM for capsules PNL-59-9 and 10.

b. Transient Irradiations

Work continued on the fabrication of the calibration test capsule and it is approximately 65% complete.

2. Thermal Flux Irradiations of Hypostoichiometric $\text{UO}_2 - \text{PuO}_2$

R. D. Leggett, J. W. Weber, and L. A. Pember

a. Examination of BNW-1 Fuel Pins at BNW

BNW-1-6 and BNW-1-4 fuel pins ($\sim 50,000$ MWd/tonne) were sectioned and the sections were prepared for ceramography. Macros and β - γ and α autoradiographs were completed on ten sections, and optical microscopy was conducted on several sections after final polishing and etching. A few examples of the more significant observations are presented here.

Figure 5.D.1. shows forms of metallic fission products that exist in the fuel in BNW-1-6-G, and the small inclusions

250X

750X

FIGURE 5.D.1. *Inclusions and "Rivers" of Fission Product
in BNW-1-6-G*

and "rivers" that extend almost to the cladding. It is not known whether the composition of these is different from those on the surface of the central hole or not. Efforts will be made to identify these inclusions in both regions of the fuel.

Figure 5.D.2 depicts some of the fuel-cladding interaction that was seen in the BNW-1-6-G specimen. The fuel is obviously in intimate contact with the cladding. A reaction layer exists along much of the interface and this has not yet been identified. Evidence of severe attack was also observed at part of the cladding boundaries which in some instances extended as far as 0.006 in. into the cladding. In some regions the boundaries actually seem to have pulled apart. The cause of this reaction is not understood at the moment and efforts are continuing to learn more about it.

Figure 5.D.3 shows a longitudinal section prepared at the top of the fuel column of BNW-1-4, showing one of the cladding breaches that occurred there. In Figure 5.D.4 tiny "metallic fingers" can be seen extending into the fuel from the cladding. Figure 5.D.5 shows the "lacy layer" type of interaction along with grain boundary penetration.

It is fairly obvious that fuel-cladding interactions of the type observed here will severely reduce the strength of the cladding. Even if no notch effects of the grain boundary penetrations are considered, the penetrations extending as much as 1/3 of the way through the cladding effectively reduces the strength by 1/3. It is not known whether the interactions are mechanical or chemical or a combination of both. In some cases the "fingers" almost appear to be slivers of cladding that were somehow removed and then worked their way into the fuel. In other cases the reaction layers appear to be interdiffusional in nature.

563X

188X

FIGURE 5.D.2. *Fuel-Cladding Boundary Intersection*

FIGURE 5.D.3. Longitudinal Section from BNW-1-4 at the Top of the Fuel Column

188X

FIGURE 5.D.4. *Metallic "Fingers" in BNW-1-4 Extending from Cladding into Fuel*

238X

FIGURE 5.D.5. "Lacy Layer" Interaction Between Fuel and Cladding in BNW-1-4

3. Encapsulated Pin Irradiation in EBR-II

L. A. Pember

The five PNL-X series capsules which were irradiated in EBR-II subassembly X039 were discharged from EBR-II. Three of the pins operated at approximately 10.6 kW/ft and two operated at approximately 6 kW/ft. The peak burnup of the 10.6 kW/ft pins was 8500 MWd/tonneM. These capsules will be neutron radiographed at the TREAT Facility and shipped to LASL for destructive examination.

4. Remote Fuel Pin Diameter Measuring Equipment

J. W. Weber

Mechanical design for modifications of the 324 Building remote fuel pin diameter measuring machine is about 95% completed. These modifications will permit diameter measurements to be conducted on the irradiated BNW fuel pins in any of the following modes:

- A longitudinal diameter scan at any circumferential angle over the full pin length at speeds from 0.72 in./min to 108 in./min.
- At any longitudinal position diameters can be measured through a full 360 degree rotation at rotational speeds from 0.05 rpm to 10 rpm.
- A spiral diameter with pitch variations from less than 0.1 in. up to more than 2000 inches.

E. OFFSITE FUEL PROGRAMS

G. A. Last

1. LMFBR Fuel Development

B. R. Hayward and F. M. Smith

All contractors participating in the Phase-I FFTF analytical chemistry round robin were visited by a group from Battelle-Northwest and LASL. Each analytical method and results of the contractors' work were discussed in detail.

The contractor discussions resulted in agreements on standard analytical methods, analytical standards to be supplied by LASL, and standard sample preparation techniques. One analysis will be omitted from Phase-II (Pu isotopic) since Phase-I results for this analysis were satisfactory. Two other changes in Phase-II work are being evaluated: (a) the solid solution aspects of homogeneity may be omitted and (b) an analysis for americium may be added.

2. FFTF Fuel Procurement

J. F. Williams

All recipients of the RFP for the prequalification of LMFBR fuel suppliers were advised by TWX that the due date for proposals was delayed to an unspecified date pending review of the FFTF and NPTF fuel prequalification, qualification and procurement programs and schedule.

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