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FAST FLUX TEST FACILITY

MONTHLY INFORMAL

TECHNICAL PROGRESS REPORT

MAY 1969

AEC RESEARCH &
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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT
MAY 1969

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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT

May 1969

ABSTRACT

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 May 1969.

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FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT

May 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 and, beginning with this issue, includes a "Design Definition Summary" which has been compiled by the editors from the report contents. This section summarizes significant design decisions and recommendations made, alternative designs proposed, requirements established, and technical design problems identified.

Chapter II, FFTF Design Data, presents a tabulation of principal concept characteristics.

Chapters III, IV, and V follow previous practice and describe the work of the various sections of the Engineering, Reactor, and Plant Technology and Fuels Departments respectively. The possibility of reporting this work by 189's was being discussed with AEC at month end.

I. OVERALL FFTF PROGRESS

A. DESIGN DEFINITION SUMMARIZATION

This section has been compiled from submissions to this report to present on the basis of systems and fuel, a summary of significant design decisions and/or recommendations made during the reporting period, alternative designs proposed, requirements established, and technical design problems identified.

10. Electrical

11. Primary Electrical Power System

Based on cost estimates by BNW, Bechtel, and the Bonneville Power Administration (BPA), the use of existing 100-H 230 kV steel transmission line towers for the new lines to FFTF has been rejected. The design will be based on the use of wood poles.

20. Structures, Utilities and Containment

27. Reactor Containment System

Design recommendations for the containment system are:

- 1) A machinery dome,
- 2) Six closed loop cells located within the containment vessel,
- 3) Spherical steel containment vessel.

Location of the six closed loop cells within the containment vessel is under DRDT review. This recommendation was made for safety and economic reasons; i.e., to locate primary sodium systems inside the containment vessel and near the reactor, and to minimize the length of piping and shielding runs.

A steel containment vessel was selected over prestressed concrete because of economics.

30. Reactor

31. Reactor Core

A Westinghouse proposal for a nonrotating plug design has been incorporated in the current draft of CSDD 31.

32. Reactor Vessel and Shield Component

A final post-fabrication stress relief at or slightly above the vessel operating temperature of 1000 °F is recommended as the most favorable means for achieving dimensional control.

A significant upward shift of the neutron energy spectra at the FTR vessel wall is expected from irradiated fuel assemblies stored within the reactor vessel. Increased damage to the reactor vessel material is expected in the vicinity of the stored fuel because of the hardened spectrum.

33. Reactor Nuclear Control Components

Primary safety system worth was established at 3.20% $\delta k/k$ exclusive of a stuck rod allowance (system design will consider the highest worth rod stuck), and the secondary safety/shim control system worth was set at 8.0% $\delta k/k$.

35. First Core Fuel Assembly Components

Based on a relatively larger Doppler coefficient and the reduced potential for cladding ridging, flat end pellets are recommended for use in the FTR fuel pins.

40. Fuel Handling and Radioactive Maintenance

43. Irradiated Fuel Handling System

The reference concept for axial gas cooling during fuel transfer has been modified to utilize downward suction of gas.

It has been found that irradiated test assemblies employing beryllium-loaded fuel will emit a sufficient number of

photoneutrons to merit special consideration in the design of in-vessel storage and out-of-vessel handling machines. The calculated neutron source strength of 10^{10} n/sec one day after shutdown from such a test assembly is almost four orders of magnitude greater than that of a driver fuel element.

Although such test assemblies could be accommodated in the FTR on a nonroutine basis, special design provisions must be included if significant through-put is contemplated.

50. Heat Transport System

51. Reactor Heat Transport System

Intermediate Heat Exchangers. Babcock and Wilcox reports that specified design pressure (200 psig primary side and 250 psig secondary side), coupled with the 1200 °F structural design temperature, is considered to be a major problem. These conditions are dictating unusually thick tube sheets (9 to 10 in.) and heavy-wall shell forgings. Consequently, problems with thermal stresses in heavy wall sections are anticipated.

Dump Heat Exchanger. The dump heat exchanger (DHX) simulation work using the thermal analyzer program (TAP) has shown that a log mean temperature difference correction factor is not needed for the proposed 10-pass, cross-flow DHX. Elimination of the corrector has resulted in an 8% reduction in the required DHX surface.

60. Irradiated Test Facilities

61. Closed Loop System

The range of testing capabilities of the reference conceptual design closed loop were analyzed.

Steady-state results from the systems study show that for test ΔT 's of 400 °F, the admissible testing powers range from 1.2 to 5.7 MW. For test ΔT 's of 200 and 300 °F, test powers range from 0.5 MW (minimum power investigated) to 2.8 MW and from 0.5 to 4.3 MW, respectively. Allowable coolant

temperatures leaving the test fuel, for test ΔT 's of 400 °F, range between 900 and 1200 °F. For test ΔT 's of 200 and 300 °F allowable coolant temperatures leaving the test fuel range from 800 to 1200 °F.

80. Sodium and Inert Gas Receiving and Processing System

82. Inert Gas Receiving and Processing System

It has been concluded that the gas chromatographic approach is suited for the in-line analysis of FFTF cover gas.

83. Auxiliary Heating and Cooling System

A function for pipe and vessel heating will be included in System 83. The method of heating has not been established, but will be evaluated in the process of establishing the bases and preparing the CSDD.

90. Instrumentation and Control System

General

Evidence of possible incipient failure occurred during exploratory tests at 900 °F for both of the two available RSN-10A sensors.

In-vessel storage of fuel samples containing beryllium may compromise the effectiveness of sub-critical monitoring sensors because of high decay neutron flux.

91. Central Control and Data Handling System

Bechtel has indicated preference for a 2-computer control system over the 3-computer system proposed in the current draft of CSDD 91. This comment is being evaluated.

93. Plant Instrumentation System

An initial evaluation of the use of a venturi in the primary system to measure flow (in addition to the EM flowmeter has revealed many disadvantages.

94. Fuel Failure Monitoring System (FEDAL)

Recent calculations show that the 1.65 MeV gamma from the decay of Neon-23 which is created by the $^{23}\text{Na} (n,p) ^{23}\text{Ne}$ reaction will be generated in sufficient quantities to affect the FEDAL location system design. The decay of the ^{23}Ne in the location system gas detector loop provides a beta-gamma background which will reduce the sensitivity of the location detection system if adequate sample delay time is not provided.

A concept of an integrated FEDAL sample module that may be combined with other modules in a compact shielded array has been devised. This module array has 34 positions in a single cylindrical tank approximately 4 1/2 ft in diameter. The tank and its modules provide FEDAL sample monitoring for one-third of the FTR core; hence, three tanks are required to service the complete core. The monitoring tanks and their connecting piping services are located adjacent to and outside of the machinery dome at the level of the reactor head.

B. SUPPORTING TECHNOLOGY

This section summarizes major supporting technology submissions which drew specific conclusions, or reported tentative decisions or events of wide interest.

1. Materials Technology

Data from uniaxial creep-rupture tests on irradiated Type 304 and 316 SS show reductions in rupture life of up to a factor of 10 after irradiation to the relatively low fluence of $1 \times 10^{22} \text{ n/cm}^2$ (calculated total). This reduction appears to be increasing with fluence and is primarily the result of reduced ductility.

A review of strain-age cracking in welded austenitic stainless steels has been completed. The possibility of strain-age cracking in the heat affected zone of welds can be precluded by:

- 1) Use of Types 304 or 316 SS only,
- 2) Limiting titanium and niobium content to 0.02% maximum each,
- 3) Checking each heat of steel by appropriate weld and stress relief tests before acceptance.

2. Critical Assembly Experiments

The experiments on Assembly 56b on ZPR III have been completed.

Planning and calculations are being done in support of the FTR II and III critical assemblies. In particular, the possibility of measuring the sodium void worth using prototypical FTR fuel pins is being investigated.

3. Fuel Pin Design

The phenomenon of ridging of fuel rod cladding was investigated and related to dishing of the ends of fuel pellets. (See System 35.)

4. Computation Methods

A computer program capable of predicting axial and radial temperature distributions in the in-reactor closed loop tube, for both steady state and transient conditions, has been completed.

An extremely flexible fuel shuffling option has been added to the 2-dimensional diffusion-burnup code, 2DB. The option has been debugged and is in use.

A 3-dimensional synthesis code, RSYN, has been debugged and used for sample cases to compare with 2DB.

C. FUELS DEVELOPMENT

1. Radiation-Induced Swelling

Equations describing the effect of fluence and temperature on swelling in solution treated Types 304 and 316 SS and in 20% cold worked Type 316 SS were refined and the results agreed upon by Westinghouse and BNW.

Specifically the equations as delineated are as follows:

Solution Treated Types 304 and 316 SS

$$\Delta V/V = 5.0 \times 10^{-36} (\phi t)^{1.66} [e^{-6800/RT} - (1.87 \times 10^4) e^{-27,000/RT}] \quad (1)$$

20% Cold Worked Type 316 SS

$$\Delta V/V = 9.4 \times 10^{-33} (\phi t)^{1.48} [e^{-5600/RT} - (1.78 \times 10^5) e^{-30,000/RT}] \quad (2)$$

Where $\Delta V/V$ = percent volume change

ϕt = neutron fluence, $E = 0.1$ MeV

Q = activation energy in cal/mole-°K.

In view of the few data available for developing equations for swelling in cold worked Type 316 SS it appears that, for the present our best design guidance is to use Equation (2) for situations in which the temperature is 500 °C or less and predicted swelling is less than 1%, and assume that the swelling will be between the limits of Equation (1) and Equation (2) at higher fluences and temperatures.

2. Offsite Fuel Manufacturing Support

The training of offsite representatives was started. Plutonium safe handling and radiation protection orientations, discussion of handling techniques and fabrication methods and a general familiarizing with the product specifications and pin design was completed. General lab observation and glove box work is being performed. The training in other areas such as NDT testing, analytical chemistry testing, process equipment and capabilities, and the like is continuing.

3. Analysis of Fuel Pin Performance

A concerted effort is in progress to assemble and analyze available data from irradiation experiments with stainless steel clad oxide fuel pins. Data on the physical and mechanical properties of the fuel pin system, the reactor environment, and the pre- and post-irradiation measurements are being combined with the best mathematical models available.

II. FFTF DESIGN DATA

This Chapter presents, as Table 2-I, a current summary of principal FFTF concept characteristics and data.

TABLE 2-I. *Summary of Principal Concept Characteristics and Data*

(Compiled by C. W. Higby - May 1969)

	<u>Units</u>	<u>Values</u>	<u>Status^(a)</u>
<u>A. General Plant Data</u>			
1. Core arrangement		vertical	1
2. Design life	years	20	1
3. Peak flux	n/cm ² /sec	7.2×10^{15}	2
4. Total power	MWt	400	1
5. Reactor coolant		sodium	1
6. Core volume	liters	1033	2
7. Core flow	lb/hr	1.5×10^7	2
8. Core pressure drop			
design maximum	psi	120	2
9. Reactor bulk inlet temp.			
initial	°F	600	2
design maximum	°F	900	1
10. Reactor bulk outlet temp.			
initial core	°F	900	2
design maximum	°F	1200	1
11. Core temperature rise avg.			
initial	°F	300	2
design maximum	°F	400	1
12. Reactor cover gas		argon	1
<u>B. Reactor Vessel</u>			
1. Diameter	feet	17	2
2. Height	feet	50	2
3. Wall thickness	inches	2	2
4. Wall fluence, total	nvt	1.0×10^{21}	2
5. Material		304 SS	1
<u>C. Core Design</u>			
1. Number of core lattice positions		91	1
2. Number of driver fuel subassemblies		76	2

(a) 1 - Firm requirement or definite concept characteristic.
 2 - Current reference design value.

TABLE 2-I. (contd)

	Units	Values	Status ^(a)
3. Number of closed loops		6	2
4. Number of contact instrumented in-core open test positions		2	2
5. Number of proximity instrumented open test positions		1	2
6. Number of in-core safety rods		3	2
7. Number of in-core control rods		3	2
8. Number of peripheral control rods		15	2
9. Equivalent core diameter	inches	~47	2
10. Active core height	inches	36	2
11. Reflector material		nickel	1
12. Reflector thickness			
axial	inches	6	1
radial	rows	3	2
13. Fuel pin heat transfer area	sq ft	2980	2
14. Core coolant velocity, maximum	ft/sec	35	2
15. Direction of coolant flow		upward	1
D. <u>Driver Fuel</u>			
1. Fuel composition		18-25 vol% PuO ₂ 75-82 vol% UO ₂	2
2. Cladding material		316 SS	1
3. Linear heat generation rate, average	kW/ft	6.7	2
4. Overpower factor		1.25	2
5. Engineering hot channel factor		1.13 to 1.25	2
6. Peak linear heat generation at overpower	kW/ft	18.0	2
7. Target burnup, average	MWd/tonne (metal)	45,000	2
8. Cladding heat transfer coefficient	Btu/hr-°F-ft ²	40,000	2
9. Maximum fuel temperature	°F	4,200	2
10. Fuel assembly length	feet	14	2
11. Fuel geometry		Hexagonal pin cluster	1
12. Pin diameter	inches	0.230	2
13. Spacer wire diameter	inches	0.056	2
14. Number of pins per assembly		217	1
15. Subassembly cross-section outside dimension (across flats)	inches	4.615	2

TABLE 2-I. (contd)

	Units	Values	Status ^(a)
16. Lattice spacing	inches	4.715	2
17. Duct wall thickness	inches	0.140	2
E. <u>Physics Data</u>			
1. Delayed neutron fraction		0.003	2
2. Neutron lifetime	seconds	3.5×10^{-7}	2
3. Doppler	$T(\frac{dk}{dt})$	-0.004	2
4. Power density	MW/liter	0.39	2
5. Power distribution (peak/average)			
radial		1.40	2
axial		1.24	2
total		1.74	2
F. <u>Test Facilities</u>			
1. Closed loops			
- number		6	2
- location		2 adjacent to core center	2
		1 @ mid-radius	2
		3 @ core peri- phery	2
- power handling capability	MW	3-5	2
- test flow rate	gal/min	200-350	2
- test section outlet temp.	°F	1,400 (bypass flow permitted)	1
- test section length	inches	36	2
- test section diameter	inches	2.5-3.0	1
- pumping head - primary	lb/in ²	250	2
- test section pressure drop, maximum	lb/in ²	90	2
- material (in-core tube)		316 SS	1
2. In-core open test positions			
- number			
with contact instrumentation		2	2
with proximity instrumen- tation*		1	2
- power			
- coolant flow rate			
- test assembly length		Same as driver	
- test assembly cross section		fuel subassembly	
- coolant			

* One open test position will have superior accessibility. Driver fuel positions can be used as open test positions with limited accessibility.

TABLE 2-I. (contd)

	Units	Values	Status ^(a)
3. Short term irradiation facility			
- type		trail cable	1
- number		1	2
- location		core periphery	2
- minimum irradiation time at constant flux	minutes	1	2
- sample length	inches	24	2
- sample cross section, maximum	inches	2.5	2
G. <u>Heat Transport System</u>			
1. Primary loops			
- number		3	1
- primary loop material		304 SS	1
- primary loop flow (per loop)	lb/hr	5.0×10^6	2
2. Primary pumps			
- number		3	1
- design pump head	feet	500	2
- net positive suction head	feet	43	2
- design temperature	°F	1200	1
- speed control		wound rotor motor with liquid rheostat	1
- motor power			
brake	HP	1,665	2
rated	HP	3,000	2
3. Intermediate heat exchangers			
- number		3	1
- type		vertical shell and tube	1
- LMTD			
initial	°F	75	2
ultimate	°F	100	2
- capacity			
initial	MW	133	2
ultimate	MW	177	2
H. <u>Shielding</u>			
1. Within reactor vessel			
- material		stainless steel	2
- configuration		hexagonal assemblies	2
- thickness	inches	21	2
2. Cover shield			
- material		low alloy steel	2
- thickness	feet	4	2
- coolant		argon, helium or Na	2

III. GENERAL TECHNOLOGY

P. L. Hofmann

A. REACTOR AND PLANT ENGINEERING

L. M. Finch

1. Reactor Engineering

D. Marinos

a. Further Concept Studies

G. R. Waymire and A. J. Anthony

Conceptual definition of core components was essentially completed and reported in last month's progress report.⁽¹⁾ One component not complete conceptually was the in-vessel portion of the FEDAL system. Further definition was accomplished for this component during May.

Figure 3.A.1 shows the FEDAL probe that is an integral part of the fuel instrumentation. Present length increments as shown indicate a length of 54 in. is required for the overall instrument probe length. Although this length is approximately 12 in. longer than previous designs, the added length does not appear to impact significantly on other components.

Figure 3.A.2 shows a concept for a connector section between the FEDAL tube bundle in the instrument tree and the fixed riser in the vessel. Further detailing in this area indicates that in order to route the FEDAL below the top of the fuel assemblies without interfering with the restraint mechanism, an additional intermediate joint between the straight run and the riser leg is required. The joint shown utilizes a bellows joint for each individual FEDAL tube similar to that shown in Figure 3.A.1 with a single seal weld connection for the bundle housing. Removal of either the riser or horizontal section requires cutting the seal weld.

1. FFTF Monthly Informal Technical Progress Report, BNWL-1062, Battelle-Northwest, Richland, Washington, April 1969.

FIGURE 3.A.1. Instrument Probe and FEDAL Sampler

SK-3-14435

FIGURE 3.A.2. FEDAL System In-Vessel Arrangement

SK-3-14436

To provide the basis for continued conceptual design activity by Westinghouse, a FEDAL monitoring concept was defined by four drawings (Figures 3.A.3 through 3.A.6) which were transmitted to the RPD during May. The significant feature of the concept of an integrated FEDAL sample monitoring module that may be combined with other modules in a compact shielded array has been devised. The module array has 34 positions in a single cylindrical tank approximately 4 1/2 ft in diameter. The tank and its modules provide FEDAL sample monitoring for one-third of the FTR core; hence, three tanks are required to service the complete core. The monitoring tanks and their connecting piping services are located adjacent to and outside of the machinery dome at the level of the reactor head. This location allows the monitoring dome at the level of the reactor head. This location allows the monitoring modules to be at the high point of their individual sodium sampling lines thereby avoiding entrapment pockets for the entrained fission gas in the FEDAL samples. If the concept is developed to its initial promise the reference FEDAL system may be designed without excessive space allotment or restrictive location requirements for the monitoring apparatus of the FEDAL system.

An invention report was submitted for the FEDAL monitoring module by A. J. Anthony, R. R. Cone, and L. M. Finch.

b. Core CSDD Preparation

G. R. Waymire

Following approval of the System 31 CSDD by the BNW Technical Evaluation Board, informal comments were obtained from Westinghouse and RDT. All Westinghouse comments were incorporated in this draft. Most comments dealt with Section I requirements covering the rotating plug. On the basis of present studies, Westinghouse proposes a nonrotating plug design. Most of the RDT comments have also been incorporated. Several areas, however, require further discussion before resolution can be obtained.

FIGURE 3.A.3. FEDAL Monitoring Module

SK-3-14640

FIGURE 3.A.4. FEDAL Monitoring Equipment Arrangement Plan

SK-3-14641

FIGURE 3.A.5. FEDAL Monitoring Equipment Arrangement Section

FIGURE 3.A.6. FEDAL Monitoring Vessel

SK-3-14642

c. Wire Wrap Vibration Study

M. A. Fischer

A preliminary wire wrap vibration study was completed. Both lateral and torsional vibration of segments of the wire were studied. Objectively, the study was accomplished to produce data related to cladding wear effects and wrap fatigue life. In the case of lateral vibration, kinetic (impact) energies were treated as the determinant of cladding wear. In the case of torsional vibration, surface velocities (rubbing rates) were considered to be the pertinent factor. For fatigue life considerations, bending and shear stresses were the respective parameters obtained. The parametric data developed in this study will be applied in subsequent investigations of the response of wire and pin systems to mechanical and hydraulic forces (as derived from analysis and experiment). The study results also will be compared with flow test data of the CCTL to establish that the range of movements selected for the study were valid.

d. Reactor Nuclear Control Component

D. Marinos and W. M. Noble

Interim approval of CCDD 33 - Reactor Nuclear Components was received from the Technical Evaluation Board prior to a series of review meetings with Westinghouse and RDT. Comments received as a result of those meetings were incorporated in the CCDD draft when they improved or clarified the design description. Technical Evaluation Board concurrence has been obtained on the modified CCDD. The description is being submitted to the Configuration Control Board for approval.

e. FTR - Oscillator Conceptual Design Considerations

H. C. F. Ripfel

The work on the FTR oscillator was concentrated on those aspects of conceptual design where the testing requirements

seem to exceed the state-of-the-art or where their attainment has caused difficulties with other oscillators.

The data collecting and processing system received special attention. It was concluded that digital data collection (or data converted from analog to digital) is highly desirable inasmuch as a computer system can be used to improve the accuracy and to reduce the total amount of data by on-line processing to selectively omit those data which are not within given error limits. Information on the reactivity status (phase angle) of the oscillator preferably can be obtained in digital form, e.g., using a photoelectric pick-up device. Multiple (up to 24) pulses per revolution will allow a check on second order rotary oscillations (which may result from vibrations and a rod which sticks intermittently). This technique will give the required phase angle information for rod calibration purposes as well.

Two methods to reduce angular position error caused by backlash between the sensor and the rod are suggested:

- (1) A torsion-free extension from the active rod to the pick-up sensor attachment.
- (2) A duplication of the pick-up sensor on the part of the drive shaft which is considered representative for the total backlash.

Either of these methods provide additional error information that can be used in the data reduction process.

From other oscillator performance characteristics, e.g., EBR-II, Mark II oscillator, it was concluded that thermal bowing and vibration may become quite troublesome. As a consequence of these effects, the design frequency range may not be reached or data may be invalidated because the rod reactivity is frequency-dependent and cannot be determined through an out-of-reactor experiment. Therefore, thermal bowing and vibration must be considered in the design of the oscillator rod. Several

corrective options are available such as: mixing and preferential distribution of the coolant, local rotation of the (poison) rod itself relative to the oscillator, rotation, and location of the oscillator component in the core.

Important reactor parameters pertinent to oscillator design are the power density gradient, the reactivity gradient and, to some extent, the absolute power density. These parameters have been studied for the FTR.⁽¹⁻³⁾ Figure 3.A.7 shows the normalized power density gradient at the core and axial midplane such as derived from (2). Figure 3.A.7 shows the radii of core positions that are possible oscillator positions (open loops and reactivity control positions). The power densities of typical oscillator materials, (poison and stainless steel) such as shown in Table 3.A.I are subject to appreciable uncertainty as they were originally determined for a smaller core (3). Power density correlations for stainless steel with other data (1) showed reasonable agreement.

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FIGURE 3.A.7. Normalized Power Density Gradient at the Axial Midplane

TABLE 3.A.I. Power Densities in Oscillator Materials

Materials	W/g	
	In-Core	Peripheral
Stainless Steel	12.6	6.2
B ₄ C natural	69	36.3
B ₄ C containing 40 at.% ¹⁰ B	118	64
B ₄ C containing 60 at.% ¹⁰ B	167	91
B ₄ C containing 80 at.% ¹⁰ B	216	118
B ₄ C containing 100 at.% ¹⁰ B	265	145

f. Invention Report

H. C. F. Ripfel

An invention report, "A Very High-Speed Chopper Rotor for Efficient Pulsed Beam Experiments," was submitted during May.

g. Scram Rate for Control Rods

F. W. Fox

Using the control rod conceptual design shown on SK-14558, Sk-14559, and SK-14560,⁽¹⁾ control rod insertion rates were determined by use of the computer code SCRAM for the following drive configurations:

- (1) Gravity fall
- (2) Spring load with 1 gravity assist at start of stroke.
(Spring reaches free length after 14 in. travel)
- (3) Same as 2 except with 2 gravity assist at start of stroke.
- (4) 1000 lb constant assist load
- (5) 1500 lb constant assist load
- (6) 2000 lb constant assist load.

1. Not shown in this report.

The analytical results are as follows:

<u>Case</u>	<u>Time for 30-in. Stroke</u>	<u>Terminal Velocity</u>	<u>Terminal Velocity Reached At</u>
1	0.8383 sec	41 in./sec	0.5 sec (16.23-in. travel)
2	0.7246 sec	41 in./sec	0.4365 sec (18.34-in. travel)
3	0.6585 sec	41 in./sec	0.43 sec (20.78-in. travel)
4	0.5000 sec	69.5 in./sec	0.42 sec (24.43-in. travel)
5	0.4300 sec	81.17 in./sec	0.39 sec (26.75-in. travel)
6	0.3800 sec	91.5 in./sec	0.38 sec (30.00-in. travel)

Control Rod Expulsion Rate

An investigation has been started to determine the control rod expulsion rate induced by sodium boiling as a result of flow blockage.

2. Plant Engineering

D. P. Schively

a. System 51 - Heat Transport System

L. G. Henke

A computer program (MEC-21/7094) piping flexibility analysis is presently underway for the conceptual piping arrangement of the new, RDT approved, Integral Primary and Emergency Core Coolant System. This arrangement, as developed to suit Building Plan XXII, is delineated on PNL sketches SK-3-14653, SK-3-14564 and SK-3-14565.⁽¹⁾

This analysis is a necessary check in the determination of the adequacy of the limited installation space available for the System 51 Heat Transport System within the 200-ft containment sphere of Building XXII. This analysis will also provide preliminary data concerning the nozzle loadings for the

1. *Not shown in this report.*

reactor vessel, the hot-leg primary pump, and the intermediate heat exchanger. Should these nozzle loadings be considered unacceptable, corrective measures will require a rearrangement of the piping and possibly building changes.

Pipe line configuration diagrams have been prepared which indicate all anchor points and the thermal growth of all major equipment attached to the building structure. Also indicated are those pipe line guides which are necessary for the reduction of the bending moments on the various nozzles.

In this initial Building XXII piping arrangement, each of three primary heat transport circuits utilizes a 30-in. pipe line between the reactor vessel outlet nozzle and the hot-leg pump for a desirable low ΔP and resulting minimized sodium drawdown in the pump tank.

The single 20-in. primary pump discharge line to the IHX, and from the IHX to the reactor vessel inlet nozzle, will have an acceptable velocity of approximately 16-ft/sec. A short 24-in. section at the IHX inlet nozzle will reduce this velocity to about 11-ft/sec.

It has been suggested that a total of six 14-in. inlet nozzles would have less structural problems than the three 20-in. presently shown. Therefore, for a comparison between the six 14-in. and three 20-in. inlet nozzles, an additional piping flexibility analysis is being run wherein the downstream half of the single 20-in. IHX discharge line will be replaced with two 14-in. lines.

Also under study is the possible use of 26-in. reactor vessel outlet nozzles with the 30-in. vessel to hot-leg piping indicated. The transition between these two sizes would be via a 30 x 26 in. reducer located as close to the vessel outlet nozzle as possible.

It should be mentioned that the primary piping arrangement indicated for Building XXII is located in such a manner that the pipe line anchor points (indicated on SK-3-14565)⁽¹⁾ include a rigid, optional, static gas seal between reactor vessel cavity and the pump/IHX vaults, the nitrogen environments of which will be at different operating temperatures.

b. Reactor Flow and Pressure Drop Study

R. E. Keyes

A report, Reactor Flow and Pressure Drop Study, BNWL-1070, has been prepared in draft form. A comment distribution has been made, with the first issue scheduled for June 1969.

c. Reactor Vessel and Shield CCDD

D. P. Schively

The Reactor Vessel and Shield CCDD has been approved, with comment, by RDT. Consideration of these comments has been completed and appropriate portions thereof incorporated as revisions. BNW approvals are now pending after which CCDD No. 32 will be published as a formal document.

3. Radiation and Shield Analysis

W. L. Bunch

a. Photoneutrons From Test Assemblies

W. L. Bunch and C. A. Mansius

Irradiated Test assemblies employing beryllium-loaded fuel will emit a sufficient number of photoneutrons to merit special consideration in the design of in-vessel storage and out-of-vessel handling machines. The calculated neutron source strength of 10^{10} n/sec one day after shutdown from such a test assembly is almost four orders of magnitude greater than that of a driver fuel element. After a 100-day decay period, such

1. *Not shown in this report.*

a test assembly would still emit about 10^8 n/sec, giving a calculated exposure rate of 250 mrem/hr through a 35 cm thick lead cask. When stored within the vessel, the neutron flux from the test assembly to the out-of-vessel instrument system sensors would reduce the dynamic range of the system to the extent that subcritical monitoring using these sensors would be compromised. Although such test assemblies could be accommodated in the FTR on a nonroutine basis, special design provisions should be included if significant through-put is contemplated.

b. Reactor Components Cask

W. L. Bunch and C. A. Mansius

Reflector assemblies, control rod followers, and other reactor hardware will be routinely replaced because of radiation damage. It is calculated that a cask for transporting these components to a burial ground will need to contain about 8.5 ± 1 in. of lead shielding or equivalent. Cobalt is calculated to be the dominant gamma source in these components with iron, the major constituent of stainless steel, being of secondary importance. Shield requirements could be reduced slightly if the cobalt content is rigorously limited and controlled.

c. Evaluation of Radial Shield Experiment

D. R. Marr

Radiation damage considerations require that adequate attenuation be provided in the region between the FTR core and the reactor vessel. Attenuation characteristics of interleaved sodium and stainless steel slabs, whose homogenized nuclear composition approximates that of the radial region between the core and reactor vessel, were measured in the Shield Test and Irradiation Reactor facility operated by Atomics International. Calculated intensity and spectral distribution of the neutron flux at the simulated vessel are

shown in Figure 3.A.8 together with the experimental spectrometer and foil activation data. Although the experimental data sample a limited portion of the spectrum, the spectrometer data include an important region for evaluation of irradiation damage. The agreement indicates the confidence that can be placed in conceptual design calculations employing the same techniques and nuclear data.

d. Radioactive Neon in FTR Sodium

C. A. Mansius

Failed fuel elements may be located in the FTR by analyzing gas disengaged from the sodium coolant from individual fuel subassemblies. The presence of ^{23}Ne in the coolant may influence the basic design of the system. The 38 sec half-life ^{23}Ne is formed by the (n,p) reaction with ^{23}Na . Preliminary calculations indicate about 1.0 mCi of ^{23}Ne will be present per cm^3 of coolant as it leaves the fuel zone. The FEDAL system design will need to incorporate a sample holdup time to allow the ^{23}Ne to decay so that fission product gas activity is not masked by the neon.

3. ZPPR Shield Experiments

E. T. Boulette

A series of shield experiments is being planned for the ZPPR facility to provide an evaluation basis for the nuclear data and numerical techniques being used for FTR shielding calculations. The experimental shield configuration is representative of the FTR shield, and will include a stored fuel zone within the shield. In support of the design of this experiment, two-dimensional diffusion theory calculations were performed on a variety of configurations. From these calculations, the following conclusions may be reached:

FIGURE 3.A.8. Incident Spectrum at Simulated Vessel

- (1) The presence of a control ring around the ZPPR core hardens the neutron spectrum significantly, and this in turn results in less attenuation through the shield than if the control ring were absent.
- (2) The stored fuel does not significantly (<10%) increase the total flux at the outer face of the shield; however, the spectrum at that location is greatly hardened.

f. In-Vessel Stored Fuel Calculations

E. T. Boulette

In-vessel storage of irradiated fuel subassemblies must be carefully analyzed for its possible effects on radiation damage to the vessel wall and on out-of-vessel flux monitoring during power buildup. To investigate these effects, calculations were made on vertical core Concept V-A using 2-dimensional diffusion theory. The geometry selected was R- θ , and a 120 degree section with the reflective boundary conditions at 0 and 120 degrees was assumed to approximate the entire FTR. The configuration is depicted in Figure 3.A.9. A graphite zone was placed around the vessel to simulate the out-of-vessel monitoring environment.

Total neutron flux and spectra were calculated throughout the configuration. The total neutron flux at positions A-E (see Figure 3.A.9) are listed in Table 3.A.II. As can be seen, in-vessel storage does not greatly affect the total flux at the vessel. However, the spectra vary greatly from position to position. To illustrate this point, spectra at A and D are illustrated in Figure 3.A.10. It is apparent that the spectrum at the position nearest the stored fuel is considerably harder. Spectral shifts of this kind will definitely be of significance in the shield design for vessel protection.

FIGURE 3.A.9. Zone Configuration for Concept V-A

FIGURE 3.A.10. Neutron Spectra at the Vessel Wall

In considering out-of-vessel monitoring, the primary concern is the flux intensity and not the spectrum, because the monitor will be surrounded by moderating material. At full power, stored fuel will not present a major problem insofar as out-of-vessel monitoring is concerned.

TABLE 3.A.II. *Total Flux at Vessel*

<u>Position</u>	<u>Total Flux, n/cm²-sec</u>
A	9.3×10^{11}
B	1.1×10^{12}
C	1.3×10^{12}
D	1.1×10^{12}
E	9.5×10^{11}

B. NUCLEAR AND PROCESS TECHNOLOGY

R. E. Heineman

1. Nuclear Analysis Section

W. W. Little, Jr.

a. Code Development

R. W. Hardie

An extremely flexible fuel shuffling option has been added to the 2-dimensional diffusion-burnup code, 2DB. Between any two burnup cycles, material in any number of regions can be removed, replaced with new fuel, or replaced with fuel from any other position in the reactor. Material can also be burned, removed from the reactor, and then reinstated at a later time.

This option has been debugged, and is currently in production use.

b. Sodium Void Reactivity Versus Rod Configuration and Core Temperature

Z. E. Waltar

Since the reactivity consequences arising from a fast reactor sodium voiding incident can have a significant bearing upon the subsequent course of the transient, it is important to ascertain key core parameters which affect the magnitude of the sodium void worth. Two such parameters recently investigated were the control rod configuration and the core temperature at the time of voiding.

As shown in Figure 3.B.1, the removal of the reflector control rod bank results in a more positive void coefficient than if the rods are in. This is mainly the result of enhanced leakage in the case for rods in.

The voiding reactivity becomes more positive as the fuel temperature increases from the broadening of the ^{238}U resonance cross section. Hence, the reactivity consequences

FIGURE 3.B.1. Total Core Voiding Reactivity Effects as a Function of Fuel Temperature and Rod Configuration.

in a loss of coolant accident may differ significantly from that of a power transient, even for identical voiding patterns.

Also shown in Figure 3.B.1 is a rods-out case having the same shape as the uppermost curve but normalized at 1500 °K by an R-Z calculation.

c. VENUS

A. E. Waltar

An early version of the 2-dimensional neutronics-hydrodynamics code, VENUS, has been acquired from Argonne. Initial computer conversion problems (mainly dealing with core storage requirements) have been overcome essentially and a sample case has been executed.

d. RSYN

K. B. Stewart and H. S. Zwibel

Debugging of the computer program RSYN was completed and the program is considered to be ready for use.

During the past months the latest version of 2DB was also added to RSYN and a procedure for running multiple cases was developed. Because of the vast amount of data that RSYN can generate, (up to 6.25 million data points) the output consists of the eigenvalue and the mixing functions.

e. PuO₂-UO₂ Fuel Pins in Air and Sodium

L. C. Davenport

Critical parameters are being completed for 31 wt% PuO₂-U(Nat.)O₂ fuel pins containing no ²⁴⁰Pu. Monte Carlo calculations are being used to determine the critical numbers of pins in air in a hexagonal array plus error limits for three cases: (1) pins close-spaced touching cladding-to-cladding, (2) pins close-spaced touching wirewrap-to-wirewrap, and (3) pins spaced 0.150-in. between nearest neighbors (no wirewrap). Critical numbers of pins in sodium in a

hexagonal array plus error limits are being determined for two cases: (1) pins close-spaced touching cladding-to-cladding, and (2) pins close-spaced touching wirewrap-to-wirewrap.

f. FFTF Fuel Fabrication

D. R. Oden

Three criticality Safety Memoranda relative to FFTF fuel fabrication operations were issued: NCS Memo No. 69-24 - Permits closer spacing limits between loaded shipping containers; NCS Memo No. 69-26 - Provides technical limits for blending 12 kg $U(93)O_2$ powder in the 325 Building Basement; NCS Memo No. 69-27 - Provides technical limits for operations with ≤ 31 wt% PuO_2 - $U(\leq 100)O_2$ at Stations 7 and 8 in FFTF Line 2 (308 Building).

Temporary Deviation No. 308-1 to Criticality Safety Specification 308-2.5 was approved to allow pellet fabrication operations with PuO_2 in fully enriched UO_2 in Line 2 (Stations 1-6) instead of Line 1 (Stations 1-6).

Technical limits are being generated to (1) allow ≤ 31 wt% PuO_2 - $U(\leq 100)O_2$ pellets in plastic tubes at Station 1, Line 2, in the 308 Building; and (2) permit storage of up to 1.5 kg Pu as PuO_2 in plastic vials in Room 215, 308 Building.

2. Core Analysis Section

P. D. Cohn

a. Partial Blockage of Subassembly Inlet

P. J. Gross

An analysis of the coolant flow rate in a driver subassembly following a partial blockage of the inlet section has been completed. The results of this study are presented in Figure 3.B.2, which shows the percentage of initial flow rate versus time for varying percentages of blockage. The blockage is assumed to be instantaneous.

FIGURE 3.B.2. Coolant Flow Rate Following Partial Blockage of Inlet, Core $\Delta P = 100$ psi.

The model is built around the inlet flow area, which is taken as an orifice. The blockage is manifest as a reduction in the flow area. When the blockage occurs the local flow velocities are initially increased and correspondingly the pressure drop increases. With larger and larger blockages, the possibility of cavitation must be considered. Conditions can exist where the vapor bubbles from cavitation could be carried up to the fuel pin region. As a consequence voids to the core would be introduced.

b. Power Versus Time Curve for Scram from Full Power

P. J. Gross

A FORE II run was made to determine the reactor power as a function of time following a scram from full power. The results of this run are shown in Figure 3.B.3.

The reactor and safety system parameters used are summarized in Table 3.B.I. The scram reactivity versus time values used were developed from the control rod scram code. These data are for a 61 poison rod bundle. However, these data have not been fully verified. A plot of scram reactivity versus time is shown in Figure 3.B.4.

c. Ridging and Pellet End Configuration Evaluation

E. G. Stevens

Circumferential cladding deformation ("ridging") has been observed to occur at pellet interface locations, during both in-reactor^(1,2,3) and out-of-reactor tests.^(4,5) Calculations have shown that finite elastic cylinders in a radial thermal gradient have a greater diametral expansion at the ends than in the center;^(6,7) also, the effective axial expansion has been experimentally shown to be a function of the fuel central temperature in flat-end pellets up to the point at which plastic flow begins to occur (750 to 1000 °C),⁽⁶⁾ i.e., the ends "mushroom." However, cladding ridging has not been reported

1. *References are listed at the end of Item C.*

FIGURE 3.B.3. Scram from Full Power at 0.0001 Sec

TABLE 3.B.I. *Safety System Parameters*

75 msec time delay	
1-g assist rod acceleration	
Rod starts from 10 cm above the core	
61 rod bundle with P/D of 1.2	
Worth for two rods is 0.028 (not enriched)	
Total reactor power	= 400 MW
Power deposited in fuel	= 360 MW (90% of 400 MW)
Radial peak to average power	= 1.35
Peak linear power density	= 12.4 kW/ft
Core inlet temperature	= 600 °F
Core outlet temperature	= 900 °F
Coolant flow rate	= 1470 $\frac{\text{lb}}{\text{sec/ft}^2}$

FIGURE 3.B.4. Scram Reactivity from Time of Initiation

in recent high burnup, stainless steel cladding, fast reactor irradiations⁽⁸⁾ (flat pellet ends),⁽⁹⁾ (annular),⁽¹⁰⁾ (unknown pellet and configuration) which produced measurable diametral increases in cladding. Therefore, ridging was not a significant problem. The fine structure observed in the profilometer traces of pins from recent EBR-II irradiations to 58,000 MWd/tonne⁽¹¹⁾ could possibly be interpreted as the beginning of the ridging phenomenon.

The out-of-reactor thermal gradient experiments have shown that ridging can result from temperature effects alone. "Ridges" probably existed at the start of irradiation of the fast reactor fuels cited above. But since the stainless steel cladding thermal expansion is 3X that of Zircaloy, they did not deform the cladding as reported in the high pressure coolant, Zircaloy systems.^(1,2) However, with time and temperature, microstructural changes occur in the fuel which tend to close the fuel-SS cladding gap in a reasonably uniform manner. A contributing factor to explain the conflicting data might be that the Zircaloy-high pressure coolant systems cause the cladding to grip the fuel and thus impede the initial axial expansion, which is then manifested by deformation of the ends of the pellets. Ridal investigated this hypothesis⁽¹⁾ with one deliberately defected specimen to equalize pressure across the cladding; and concluded that axial loading did not play a significant part in ridging in his experiment, i.e., ridge heights were independent of the external coolant pressure.

Dishing the pellet ends has been observed to increase in-reactor ridging.^(2,3,11) The use of bevelled or chamfered pellet edges would also reduce the extent of ridging as shown in out-of-reactor tests.⁽⁵⁾ A short time in-reactor investigation with aluminum-clad specimens irradiated in a hydraulic rabbit system did not show any reduction of ridging with bevelled edge pellets. However, Notley has questioned the

usefulness of these high-intensity, short-duration irradiations for obtaining information on the in-reactor thermal expansion behavior.⁽⁶⁾

The relationship of the observed reactivity effects during the initial low power operation of RAPSODIE to the flat-ended pellet configuration was investigated. The lack of specific analysis data on this phenomenon precludes a direct correlation. However, Stachura's⁽¹³⁾ general observations indicated that axial expansion of the fuel column probably caused the reactivity changes. As the power level was raised, fuel central temperatures increased with resultant pellet cracking, central fuel plasticity and central void formation due to sintering. After operations at full power, the initial reactivity effects apparently were no longer observed. The conclusion which might be drawn from the experience in the realm of pellet end configuration is that the degree of the observed reactivity effect might possibly have been reduced by the use of dished-end pellets (since the initial fuel column expansion would have been less).

Comparing the manufacturing differences of flat and dished end pellets leads to the conclusion that there is no significant cost difference for large scale pellet production.

Comparing the two end configurations from the overall core Doppler coefficient produces some potential differences which must be further investigated in preliminary design. To accommodate dishing requires either removal of some fuel material or the production of denser pellets. The former approach would decrease the fertile-to-fissile ratio (since the constant core volume requires approximately a constant fissile loading and only fertile material may be removed) and therefore decrease the Doppler coefficient. This provides less inherent negative reactivity feedback for shutdown during transients. Additionally, the lesser degree of axial fuel expansion, when compared

to flat end pellets, also provides less negative reactivity feedback. The flat-end pellets can be qualitatively classed as possessing a higher degree of self shutdown capability. The higher density pellet affects burnup potential as well as other design areas which have strong requirements to maintain the density within certain bounds.⁽¹⁴⁾ More investigation in this area is required before the tradeoffs can be determined.

Many pressurized water, thermal reactors have been built with dished end pellets (built by W, B & W and CE) while many boiling water reactors have flat end pellets (GE). No specific reasoning for the choice was given in the literature. The principal consideration appears to have been the relative axial expansion characteristics between the two types of pellets for long fuel columns.

Other considerations pertinent to the pellet end configuration have been presented in a BNW monthly progress report.⁽¹⁵⁾ This concluded that the minimal advantage of dishing [initially reduced fuel column expansion as presented in Section D.1.1⁽¹⁶⁾] is outweighed by the increase in potential for ridging.

Based on a relatively larger Doppler coefficient and the reduced potential for cladding ridging, flat end pellets are recommended for use in the FTR fuel pins.

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d. Wear Rate on Wire Wrap for Fuel Pin Bundles

F. W. Fox

Fabrication tolerances plus assembly tolerances on a pin bundle cause a deformation in the bundle before it is assembled into the fuel duct. Assembly of this deformed bundle and the fuel duct will result in a normal compressive load between the wire wrap and the duct wall. As burnup progresses, the duct will tend to expand to a greater degree than does the wire wrap thus changing this normal compressive load.

Under these conditions any relative motion between the wire wrap and duct will result in metal wear on the wire wrap and duct.

The principal contributor to relative motion would be changes in relative temperature gradients in the bundle and duct induced by refueling cycles, scrams, and power changes.

Considering 10 refueling cycles, 25 scrams, and 1000 power changes from full power to one-half power (and return) during 80,000 MWd/T burnup period, the metal wear rate (primarily on the wire wrap) will be 0.0025 mils (depth) per 1000 MWd/T burnup or about 0.2 mils for 80,000 MWd/T burnup.

3. Experimental Physics

R. A. Bennett

a. FTR Critical Experiments in ZPPR

R. A. Bennett

Critical assembly FTR-1 has been built in ZPPR and is to be referred to as ZPPR Assembly 1. The compositions of the drawers in the core and reflectors are approximately the same as those of ZPR III Assembly 56b. Exact compositions and drawer configurations are expected to appear in the May 1969, Argonne National Laboratory, FFTF Informal Technical Progress Report. The following measurements are essentially complete:

- (1) Reactivity worth versus separation of halves
- (2) Reactivity reproducibility of half closure
- (3) Temperature coefficient
- (4) ZPPR rod reactivity worth calibration
- (5) Source calibration
- (6) Edge worth measurements
- (7) Critical mass.

Upon completion of these measurements the core size is to be reduced to the matrix profile of ZPR III 56b to assess the reactivity difference between the systems. This work will be followed by the FTR-2 series of experiments.

b. Heterogeneity in ZPR III Assembly 51

W. R. Young

An estimate of the increase in the eigenvalue caused by the plate heterogeneities has been calculated for the core of ZPR-III Assembly 51. Eigenvalues of two similar reflected spherical reactors were compared. Flux-volume weighted cross sections were used for the core material in one calculation. In the other calculation it was assumed that the core was homogeneous and the core cross sections were obtained directly from FCC-IV.⁽¹⁾ Flux-volume weighted cross sections were obtained from a 1D DTF-V⁽²⁾ S-12 cell calculation on the A-A* core drawers of Assembly 51. Both calculations were in 26 energy groups. The calculated increase in $\Delta k/k$ induced by heterogeneities was $0.9\% \pm 0.03\%$. Of this increase about

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1. R. W. Hardie and W. W. Little, FCC-IV, A Revised Version of the FCC Fundamental Mode Fast Reactor Code, BNWL-450, August 1967, Battelle-Northwest Laboratory, Richland, Washington.
 2. DTF-V is a BNWL-modified version of DRF-IV, K. D. Lathrop, DTF-IV, A Fortran IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering, LASL, LA-3373, July 15, 1965.

0.8% was caused by an increase in the infinite multiplication of the core and 0.1% due to a decrease in reactor leakage. It is expected that the correction will be greater for a cylindrical reactor calculation than it was in this comparison of spherical reactors.

An accurate account of the heterogeneity effects in these critical areas will significantly decrease the uncertainties in applying critical experiment results to the FTR core design and safety. Further study to improve and prove these techniques is continuing.

c. Analysis of ZPR III Assembly 56b

S. L. DeMyer

Models for the final analysis of ZPR III Assembly 56b reactivity worths and reaction rate distributions, central fission ratios and reactivity worths, and control rod worths, have been prepared. The analysis is incomplete.

d. FTR-2 and -3 Planning - Effects of Flux Perturbations on Subcritical Reactivity Monitoring

V. O. Uotinen

Movements of control rods or fuel in the FTR are expected to perturb local fluxes beyond that expected because of the resulting changes in reactivity. Such perturbations can seriously affect subcritical reactivity measurements. Calculations have been performed, using 2-dimensional diffusion theory in $R\text{-}\phi$ geometry, to predict approximate magnitudes of typical perturbations.

The results of these calculations indicate that flux distortions in the FTR, caused by movements of control rods or additions of fuel to a peripheral fuel storage tank, can be large and can have long-range effects. It is evident that the accuracy of a subcritical measurement of the worth of some

change depends on the position of the detector relative to the change, as well as on the magnitude of the change. For example, the worth of a peripheral control rod, as deduced from the calculated ratio of ^{235}U fission rates, is overestimated by an amount ranging from 25% to less than 1%, depending on where in the radial shield the detector chamber is located.

The results of this study emphasize the need for an experimental investigation of the problem. Experimental information is needed on the magnitude and extent of flux distortions, before subcritical reactivity measurements can be interpreted meaningfully. It seems that ultimately, a logistic procedure should be developed experimentally for the FTR, to determine which FTR flux monitors are to be used, and which ones are not to be used, to monitor each specific reactivity change.

Results from initial experiments performed in ZPR III Assembly 56b have just been received and are being analyzed.

e. FTR-1, 2 and 3 Preanalysis

J. V. Nelson

Precalculations of the neutronics characteristics of FTR-1, FTR-2 and FTR-3 criticals to be built in ZPPR are underway. The estimated critical mass of FTR-1, using an R-Z, 13 group, diffusion model, is 367 kg of fissile material. The core compositions are, however, only approximate.

f. FTR-2 and -3 Preplanning - Pin-Fuel - Sodium Void Experiments

Q. L. Baird

The interpretation of sodium voiding experiments in the FTR criticals is complicated by the atypical platelet fuel used to simulate the pin system. The sodium void coefficient is of prime interest in the analysis of accidental power

transients. The reduction of the uncertainty in experiment-theory correlation by the use of pin fueled zones in the platelet critical is, therefore, a very timely consideration.

Prototype FTR pins, in small quantity, are presently being fabricated. These pins have the composition of the outer zone of the proposed 2-zone core. The $\text{PuO}_2\text{-UO}_2$ is 25 wt% PuO_2 and the UO_2 is from natural uranium. The pins are being fabricated in nominal 28-in. lengths. It appears feasible to use these pins (about 300) in the ZPPR experiments. They could be used either in sodium filled calandria or potted in sodium filled boxes. It is proposed to load a modified platelet zone in the center of either FTR-II or III that will be matched in composition by the pin-filled calandria or box and yet not be so different as to drastically modify the spectrum in the zone. Preliminary efforts to achieve matching compositions have been quite successful for both of these cores. Having achieved a matching zone, the central sodium voiding experiment can be performed in both a platelet and pin-fueled zone of near equal composition. This sequence will, thus, allow the effects of platelet heterogeneity on the coefficient to be determined.

The degree to which the pin composition can be matched to a platelet loading is illustrated by considering the key nuclides involved for the FTR-II case. The ^{239}Pu can be matched to within 2-3%, ^{238}U to within about 5% and oxygen to within 9 to 12%, depending upon the modification plan chosen. The following table gives an indication of the approximate atom densities for the FTR-II loading in ZPPR, a 39-pin calandria, and two modifications of the FTR-II loading. The FTR-II, Mod.-B loading has the advantage that no Na_2CO_3 is used in achieving it and, thus, a 100% sodium voiding experiment can be performed while leaving other components unchanged except for a slight variation in the stainless steel fraction. All compositions given in Table 3.B.II are approximate and the FTR-II values do not exactly represent the as-built core.

TABLE 3.B.II. *Approximate Compositions for Modified FTR-II
Drawer Loadings with a "Matched" Pin Load
($\times 10^{-2}$ atoms/b-cm)*

<u>Nuclide</u>	<u>FTR-II</u>	<u>FTR-II, Mod. A</u>	<u>FTR-II, Mod. B</u>	<u>39-pin Calandria</u>
Na	0.835	0.939	0.907	1.011
Fe	1.32	1.459	1.474	1.595
Pu-239 (and 241)	0.130	0.130	0.130	0.125
Pu-240	0.017	0.017	0.017	0.017
U-238	0.595	0.438	0.361	0.420
O	1.460	1.179	1.027	1.148
C	0.100	0.100	-	-
Al	-	0.072	0.163	-
U-238/ Pu-239	4.576	3.366	2.779	3.364
O/Heavy Atom	2.117	2.032	2.016	2.034

C. SAFETY AND SYSTEMS ANALYSIS

R. E. Peterson

1. Safety Analysis

D. D. Stepnewski

a. Reliability Analysis of the Plant Protection System

O. B. Monteith

An analysis is in progress to determine the long-term availability of the Plant Protection System. Results of this analysis are based on the PPS as described in CSDD 99 which includes the instrument sensors and electronics, the various logic assemblies, and the rod release circuit. Since established failure rates for the components are nonexistent, a method of analysis outlined in ORNL 6922, Demonstration of Reliability Analysis Techniques Applicable to RDT Standards Program Using a Containment Cooling System as an Illustration, is being used. The analytical method employs a range of failure rates to obtain a range of results from the reliability expressions. The range used in an example given here is shown in Table 3.C.I.

TABLE 3.C.I. *Rod Scram Logic Circuit Failure Rates
(failures per million hours)*

	<u>Worst</u>	<u>Nominal</u>	<u>Optimistic</u>
R_5, R_6, R_{12}	10	1	0.01
Contacts R_5, R_6, R_{12}	100	10	0.1
R_9, R_{10}, R_{13}	100	10	0.1
Contacts R_9, R_{10}, R_{13}	100	10	0.1

The rod scram logic circuit for the particular system analyzed is made up of the three safety circuits that operate the three safety relays (R_9, R_{10} , and R_{13}), and the rod scram release circuit that contains the four contacts of each safety relay. This arrangement of components provides for the safety and reliability requirements established by the single failure

criterion. In addition it reduces the number of spurious trips that are caused by "fail-safe" failures of these components. The reactor scram logic that is implemented in this design is a two-out-of-three coincidence of the protective channel input. There are additional safety and reliability provisions embodied in the arrangement of the safety relay contacts. These include: (1) A coexistence of six relay contacts must fail in the shorted mode to cause an unsafe failure, (2) Two coexistent inadvertent electrical groundings are needed to cause an unsafe failure, and, (3) A coexistence of two relay contacts failed in the open mode are needed to cause a spurious scram.

The reliability assessment was made by approximating the logic of the actual scram logic circuit. The availability that is determined from this approximation is believed to be pessimistic because some of the success paths of relay contacts are not included. The error involved in this approximation is very small because other components have a greater effect on the availability.

Results are shown in Table 3.C.II. From those results it was concluded that a testing interval of less than 1500 hours will be required and, in fact, testing may have to be more frequent than weekly.

TABLE 3.C.II. *Rod Scram Logic Circuit Unavailability*

	Hours Between Test		
	8 hr	168 hr	1500 hr
Worst	4×10^{-6}	2×10^{-3}	≈ 0.16
Nominal	4×10^{-8}	2×10^{-5}	2×10^{-3}
Optimistic	4×10^{-12}	2×10^{-9}	2×10^{-6}

b. Sodium Fires in Low Oxygen Concentration

P. R. Shire

When a 1200 °F sodium spray expulsion occurs in inerted reactor equipment cells at low oxygen concentrations ($\leq 5\%$) pressure may reach 33 psig quite rapidly ($\sim$ seconds) with reference design conditions. Assuming the principal product of combustion is sodium oxide ($4\text{Na} + \text{O}_2 \rightarrow 2\text{Na}_2\text{O}$), heat balance calculations⁽¹⁾ were performed for various oxygen concentrations. Equation (B-9) in Reference 1 is applicable for excess sodium; however, as a limiting case the present calculations were performed for the 4:1 stoichiometric sodium-oxygen ratio indicated by the chemical balance equation.

Equilibrium temperatures and pressures were computed for oxygen concentration 0-5%, (see Figure 3.C.1). The initial inert gas temperatures were 100 and 500 °F. Relatively small amounts of sodium spray are required for the stoichiometric ratio; in 2% oxygen 4.5 lb Na for 100 °F gas and 2.6 lb Na for 500 °F gas per 1000 ft³ inert gas. The curves show the reduction in cell accident temperatures and pressures by controlling oxygen concentration and initial operating temperature.

c. Reactor Vessel Repsonse to the DBA

G. L. Fox

Design Basis Accident analysis has required development of analytical models which estimate forces and response for the reactor containment system. A computer code has been developed which predicts the vessel and vessel cover loadings from work energy production that occurs over many milliseconds. The code is called ASPRIN (A Simulated Propellant Reactor Incident). A complete description of the code including

1. J. Hopenfled, "FFTF Sodium Fires and Blast Analysis Study," *Atomics International*, (AI-68-MEMO-127) February 13, 1967, p. 67-8.

FIGURE 3.C.1. Sodium Spray Fire in Low Oxygen Concentration

computer print out, flow diagrams, and a program listing has been written as a BNWL document titled ASPRIN - A Tool for Predicting Vessel Response for Design Basis Accidents, (to be published).

d. Containment Systems Analysis

G. L. Fox, R. D. Peak and P. R. Shire

An integrated analysis of the design basis accident containment requirements is in progress. The objective of these studies is to determine the time dependence of various accident conditions over the long term. Transient heat transfer analysis will be performed using the TAP-3F computer code for simulated times of from 10 to 50 hours or longer following the accident. The length of time depends upon factors tending to increase containment pressures and temperatures over long time periods.

Equalization of pressures among the cells comprising the first containment boundary by intra-cell vents is being investigated. The one-dimensional equations for venting of compressible gases have been formulated and a simplified version for quick answers has been programmed using an analog simulating language (MIMIC).

Work on conceptual feasibility of a molten core catcher and decay heat removal system has been started. As a primary analytical tool for this work, conversion of CONTEMPT, a computer program for predicting the containment pressure - temperature response to a loss-of-coolant accident is in progress. This program, described in IDO-17220, was prepared for water reactors in air containments. The main conversion is to sodium properties for the FTR sodium reactor in a nitrogen containment. This code will be used to determine equilibrium temperature and pressure in the reactor cavity for the post-DBA condition of total core meltdown.

2. Systems Analysis

H. G. Johnson

a. Heat Transport System Analysis

Dump Heat Exchanger Analysis (R. N. Madsen). The mathematical simulation of the DHX using the Thermal Analyzer Program (TAP) computer code has been re-evaluated in considerable detail. Specifically, the steady-state temperature distribution in the nodal representation of the fins was compared with the distribution obtained from the exact analytical solution. This evaluation led to an adjustment in the location and size of the fin nodes, and resulted in almost exact agreement with the theoretical temperature distribution.

In addition, agreement with the calculations of the design code ADOPT was also excellent and led to a more accurate determination of DHX size and performance than heretofore. In determining the DHX size the ADOPT code uses a correction factor for the logarithmic mean temperature difference (LMTD) to adjust the size to account for the fact that the DHX is not a counter-flow but a cross-flow heat exchanger for which the LMTD is not exactly applicable. However, the LMTD correction factor which is used in the ADOPT code is for a 2-pass cross-flow exchanger while the DHX which is simulated is a 10-pass heat exchanger. Previous experience and reasoning led to the belief that a 10-pass exchanger so closely approximated a true counter-flow arrangement that the correction could be eliminated, but since no analytical justification could be provided, the correction factor was never removed. This resulted in some conservatism in the DHX size. The DHX analysis using TAP has now verified the assumption that the LMTD corrector is not required for a ten-pass exchanger. Using exactly the same size DHX which the ADOPT code calculated for a duty of 133.3 MW_t , and using the same flows and coolant entering conditions, the

TAP simulation calculated a DHX duty of 133.66 MW_t, with less than a 2-degree error in the coolant outlet temperatures. Elimination of the correction factor resulted in a reduction in the calculated DHX size by about 8%.

The revised TAP model has also been checked for transient computations and for cases of unbalanced flows. In all these cases the performance has been excellent. A good reference DHX simulation is thus available.

A simplified DHX model has been developed using six nodes each for the sodium metal and air portions of the DHX. The model is currently being programmed and will then be evaluated by comparison with the TAP reference simulation.

b. Sodium Hammer Analysis

C. D. Flowers and J. E. Krajicek

The check valve model which represents the transient closing characteristics of a tilting disk check valve has been incorporated into a model which simulates the FFTF Primary Heat Transport System. Checkout of these combined programs has been completed.

Parametric studies to determine the effect of dash pot stiffness and torsional springs on the Heat Transport System coastdown have been completed for a 0.3 ft² break located at the reactor inlet. A parametric study to determine the effect of the check valve on the system coastdown caused by a pump power failure is partially completed.

c. Hybrid Simulation of the Reactor and Heat Transport System

R. D. Benham and A. L. Gunby

Simulation Development -- The analog patching has been reworked, resulting in a savings of an amount representing the cost of 25 amplifiers. Several controller simulators were

connected to the new hybrid simulator, Hybrid-2 (AD-4, PDP-9). The controller simulators checked out satisfactorily; others will be added in the next reporting period. Models for the IHX were modified to account for bypass flow on both the primary and secondary sides. IHX finite differencing techniques were reviewed and a model change was made to provide for better unbalanced flow conditions. The model alternates central differencing with three-point-backward differencing, to reduce temperature distribution irregularities for unbalanced flows. The changes were programmed and checked out, and are operating well.

3. Control Analysis

R. A. Harvey

a. Driver Channel Instrumentation Analysis

J. J. Regimbal and G. H. Dickerson

The TIGER V code was modified to facilitate the computation of sensor effectiveness in loss-of-coolant flow problems. The change includes an improved restart capability and the automatic inauguration of scram conditions upon reaching selected criteria for protective action.

Heat transfer simulations of 1/8-in. insulated junction thermocouples indicated that response times longer than 0.5 sec result from impaired thermal conduction through the low-density ceramic material which is repacked at the thermocouple tip after forming the junction. Arbitrary variation of the thermal resistance values for the insulation around the hot junction is the means chosen for simulating response time variations in the core safety studies.

The overall analysis of sodium boiling prevention by deploying thermocouples and flowmeters in the duct flow is continuing, with emphasis on detecting over-temperature or limiting temperature rate of rise during the occurrence of sudden or slowly-gathering flow loss.

b. Assessment of FFTF Control Adequacy

R. A. Harvey

Planning was started for a study to describe bases for assessing the safety and operational adequacy of FFTF control systems. The study is expected to provide guidance to both BNW and the contractor in evaluating control system performance.

A high-speed simplified hybrid computer model of the Heat Transport System which includes the major transport delays and time lags will be used to identify trends for controller settings in various overall control schemes. The trends will be verified on the more detailed model that now exists.

IV. ENGINEERING DEPARTMENT

D. L. Condotta

A. HEAT REMOVAL SYSTEMS

B. M. Johnson

1. Sodium Components

T. W. Withers

a. Sodium Pumps

W. Babcock

A combined equipment specification and work statement has been completed for the MHTS and closed loop primary prototype pumps. It was written for the purpose of initiating hardware procurement contracts with a supplier.

The Westinghouse pump design study is nearing completion. Several draft reports have been completed and comments and approvals have been given. It appears that the study will be completed on schedule and within currently approved funding.

The revised pump State-of-Technology report and Program Plan have been completed.

b. Sodium Valves

P. F. Shaw

Functional criteria and special requirements for the FFTF Main Heat Transport System Valves were provided to LMEC for utilization in the LMFBR Valve Technology Development Program. These criteria were derived from the conceptual system design effort, modified somewhat through BNW reviews. The criteria and requirements were reviewed by BNW and LMEC jointly. The review resulted in clarification and improvement of the information presented.

c. Closed Loop Components

P. F. Shaw

Functional criteria for closed loop components were reviewed by BNW, W, and LMEC with respect to requirements for the vertical core concept. A preliminary identification of components which may require development was made. Refinement in these areas is a part of the AI review of the Closed Loop CSDD, currently in progress.

d. Intermediate Heat Exchangers

L. A. Whinery

Work on the Babcock and Wilcox Company IHX design study contract was reviewed during a progress review meeting held April 25. The Babcock and Wilcox effort to date has been focused on development of alternative conceptual arrangements. Four preliminary concept sketches have been prepared including two designs featuring removable tube bundles and two designs using fixed tube bundles with provision for in-place tube plugging. These basic concepts are presently being evaluated by the Babcock and Wilcox engineering staff. Babcock and Wilcox advised that an interim concept selection report will be issued in the near future in which two concepts would be recommended for follow-on design analysis.

Babcock and Wilcox reports that the specified design pressure (200 psig primary side and 250 psig secondary side), coupled with the 1200 °F structural design temperature, is considered to be a major problem. These conditions are dictating unusually thick tubesheets (9 to 10 in.) and heavy-wall shell forgings. Consequently, problems with thermal stresses in heavy wall sections are anticipated.

The LMEC has received proposals from Westinghouse, Foster-Wheeler and Combustion Engineering for performing supporting IHX design study work under on-going LMFBR Program contracts.

The three proposals have been evaluated by the LMEC and a recommendation has been submitted to BNW for selection of two contractors.

B. PLANT SYSTEMS

E. M. Johnston

1. Structures and Utilities

F. H. Shadel

a. Reactor Containment System, System No. 27

C. C. Steele

CSDD No. 27 for the Reactor Containment System was approved by the Configuration Control Board on April 14, 1969 and transmitted to RDT on May 6, 1969. Firm design choices for the system are:

- (1) A machinery dome.
- (2) Six closed loop cells located within the containment vessel.
- (3) Spherical steel containment vessel.

Location of the six closed loop cells within the containment vessel is under review. This choice was made for safety and economic reasons; i.e., to locate primary sodium systems inside the containment vessel and near the reactor, to minimize the length of piping and shielding runs.

The spherical steel containment vessel was chosen because it provides a higher degree of confidence in which to proceed into Preliminary Design. Transfer of long test assemblies in the vertical position was one of the requirements which indicated a spherical shape was more suitable than the right cylindrical shape. The spherical shape also has a greater pressure capability for the same plate thickness. A steel containment vessel was selected over prestressed concrete because of economics.

A rationale for containment pressures and temperatures in the spaces of the vessel was prepared. Pressures and temperatures selected as tentative design values are based upon the work of BNW, AI and Westinghouse.

b. Service Piping System, System No. 23

E. R. Rybarski

CSDD 23, Service Piping System, issued on November 22, 1967, has been revised. The revised document incorporates resolutions to the RDT comments on the original document. CSDD 23 was approved by the Configuration Control Board on April 23, with all board members concurring and it is now pending transmittal to RDT.

2. Electrical Systems

G. H. Strong

a. Primary Electrical Power System (System No. 11)

M. F. Wiitala

Based on cost estimates by BNW, Bechtel, and the Bonneville Power Administration (BPA), the use of existing 100-H 230 kV steel transmission line towers for the new lines to FFTF has been rejected. The design will be based on the use of wood poles.

BPA prepared and transmitted for BNW comment and consideration a protective relaying scheme for the FFTF primary power supply. The scheme shows the relays required by BPA for the protection of their Midway-Benton tie-line.

A proposed policy for maintenance, operation, control methods, and location of control equipment was prepared by BNW and reviewed and tentatively agreed to in a joint meeting with ITT/FSS and AEC-RL.

C. AUXILIARY SYSTEMS

B. M. Johnson

1. Sodium Receiving and Processing Systems

R. V. Dulin

Technical discussions were held with C. Biery and R. McPheeters of LASL on their work in sodium cold trap analysis. They have developed a computer program which correlates with experimental results on deposition and dissolution of sodium oxide in cold trap systems, and which may be adaptable to analysis of FFTF cold trap system designs.

2. Auxiliary Heating and Cooling

R. V. Dulin

In addition to the function of cooling cold traps and fuel decay storage, a function for pipe and vessel heating will be included in System 83. The method of heating has not been established, but will be evaluated in the process of establishing the bases and preparing the CSDD. A preliminary issue of Section One of the CSDD was completed by Bechtel and forwarded for BNW review.

D. CORE DEVELOPMENT AND ANALYSIS

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

S. O. Arneson and J. M. Yatabe

a. Coolant Mixing and Flow Distribution

The objective of the FY-1969 task is to provide flow distribution and coolant mixing data for the thermal analysis of the fuel design. The activities during this reporting period include:

- Coolant mixing and flow distribution studies were completed on a full scale 217-pin prototype fuel assembly. This was an 0.250 in. OD pin with 0.030 in. spacing using wire wraps on 12 in. pitch. These data were collected using direct sampling techniques. Preliminary analysis of the data indicates that the degree of mixing is considerably greater than expected. The subchannel velocity measurements showed large variations between apparently identical channels. This effect was probably created by damaged impact tubes, and further evaluation of the data must await a disassembly inspection of the test section.
- Conductivity probe development for mixing determinations has shown the possibility of a temperature effect on the probe signal. Since this effect would affect the accuracy of the mixing measurement, a modification of the flow system to permit a constant conductivity fluid circulation at various velocities has been made to evaluate these effects.

b. Sodium Flow Testing

The purpose of the sodium flow tests is to evaluate the effect of potential long-term failure modes on FFTF fuel assemblies. The Mark I assembly, a full sized fabrication

prototype, has now been tested to approximately 3600 hr at full flow and temperature conditions (400 gal/min, 1060 °F) in the Core Component Test Loop at Argonne National Laboratory. The fuel assembly has been removed from the facility, cleaned and returned to BNW for detailed inspection. The first results of this inspection are from visual observations of the outer pins only and indicate that the assembly appears to be in excellent condition. The only unusual behavior observed to date is a slight loosening of the wire wrap spacers on the outer pins.

c. Loose Wire Wrap Effects

It has been postulated that wire wraps may loosen during reactor operation due to differential swelling. To identify potential wear problems, a 7-pin fuel assembly model has been fabricated (3 pins with wire stretched 0.020 in. and 3 pins with wire stretched 0.030 in.) and tested in the Small Components Test Loop. The conditions for the life test are steady sodium flow velocities of 30 ft/sec and temperature of 1150 °F. Sodium purity has been maintained at less than 10 ppm oxygen and 20 ppm carbon as measured by analytical chemistry performed on samples. The assembly has been exposed to these conditions thus far for approximately 650 hr.

2. Heat Transfer Characteristics of FFTF Fuel Assemblies

S. O. Arneson and J. M. Yatabe

The FY-1969 objective of this program is to develop bundles of electrically heated pins of actual size, and perform sodium, thermal-hydraulic testing in support of FFTF design and for general use in the LMFBR Program. The current effort is concentrated on developing improved quality assurance procedures for each pin and developing the capability and quality assurance.

Effort was continued toward developing improved nondestructive tests of the final pins. A new thermal transient technique

was attempted using a rapid scanning infrared pyrometer. Preliminary testing demonstrated that this technique appears promising for quantitatively determining the surface heat flux uniformity of the completed pins.

These preliminary tests were performed on a pin with known regions of reduced boron nitride compactness. The center conductor was rapidly heated by an electrical current for three seconds with the pin resting horizontally in the air. The surface temperature was measured during and shortly after the resulting thermal transient with the rapid scanning pyrometer. During the transient, the less compacted regions had lower surface temperatures resulting from the poor thermal conductance. A quantitative record of this test was obtained by placing the output of the pyrometer on an oscilloscope and photographing the oscilloscope.

The results of the preliminary infrared test agreed well with results of other nondestructive tests (neutron radiography and the phosphorescent coating technique) on this defective pin. Future effort will be directed toward optimizing these nondestructive tests and establishing the combination of tests that will best ensure pin quality.

Tubing and rod material to fabricate the first major heated pin order was received, nondestructively tested, and returned to the vendor (Watlow Electric Manufacturing Co.). Watlow presently does not have the capability to test this tubing or rod by ultrasonic methods. However, they are attempting to procure this service directly from source vendors to reduce their dependence on BNW in this area.

The failure of the electron beam welding machine which was to perform the welds on the 7-pin bundle assembly has forced the development of a second and third source of EB welding. Samples have been welded and are currently undergoing non-destructive and destructive tests.

The modification of the Small Heat Transfer Facility required to permit it to perform 7-pin bundle testing has been completed.

3. Development Facilities Management

J. R. Carrell and K. G. Toyoda

a. Hydraulic Core Mockup Test Program

H. Leigh, J. Muraoka, D. L. Ballard, P. M. Jackson,
L. R. Sweetin, J. Spalek, G. W. Riedeman

The objective of the Hydraulic Core Mockup Test Program is to provide development and testing support for resolution of the FTR vessel and core hydraulic and mechanical design problems. Meetings were held with Westinghouse to further identify Westinghouse development requirements for thermal/hydraulic design and to provide Westinghouse with current test program results.

Final testing of the Inlet Plenum Feature Model with the torus type inlet chamber was completed. This is a third-scale hydraulic model of the inlet plenum arrangement. The purpose is to provide information to aid in the evaluation of inlet plenum configurations for the reactor vessel system. The data collected during these tests showed that the flow distribution through the individual tubes (simulated fuel ducts) was quite even and virtually unaffected by downcomer position or the number of downcomers carrying the flow. Flow mixing within the plenum was observed by injecting dye into the individual downcomer streams. Completion of testing (up to 100% of model design flows) was terminated in preparation for conversion of the model to an elliptical plenum with a single tubesheet and new inlet plenum basket. An electrolyte injection system is essentially complete and will be ready for operation with the modified assembly to permit quantitative mixing tests.

b. Sodium Core Development Facilities - CM-3

K. D. Hayden

Construction is proceeding on the FFTF Sodium Facilities Building (Bldg. 335). The rough earthwork preparation is completed and the underground utilities are being installed. Design work by Hanford Engineering Services has started on the LMFBR Core Segment Development Facility. RDT has concurred with the Fast Reactor Thermal Engineering Facility Project Proposal, but several areas require further attention (schedule, quality assurance, transient operation, location, etc.). The project proposal for the LMFBR Cover Development and Sodium Engineering Building was submitted to RL-AEC on April 28. The project provides for relocating the 16,000 ft² 190-FA Bldg. to 300 Area. In addition to space for the Reactor Cover Development activities, the building has sufficient room for the Fast Reactor Thermal Engineering Facility, the Sodium Chemistry Technology Facility and a multipurpose mockup area.

c. Sodium Facilities Operation - TF-3

J. D. Berg and L. J. Defferding

The objectives of this section and description of equipment were given in BNWL-915.⁽¹⁾ The status of the sodium facilities is given in the Table 4.D.I.

TABLE 4.D.I. *Sodium Facilities Status - May 1969*

Component	Status
Small Component Evaluation Loop	Tests of an FFTF fuel subassembly for 600 hr, initiated last month, was completed. Loop availability was about 80% with two unscheduled outages occurring.
Small Heat Transfer Loop	The modification to increase flow and thermal capabilities required for simulated FFTF 7-pin fuel bundles was completed.
Static Sodium Pots	Tests of the pressure transducer in Pot No. 3 continued. Pots 1, 2 and 4 were shut down this month.

1. *FFTF Monthly Informal Technical Progress Report, BNWL-915. Battelle-Northwest, Richland, Washington, October 1968.*

TABLE 4.D.I. (contd)

Component	Status
Sodium Purification Loop	<p>Tests of the direct reading of the UNC carbon meter continued. Impurities were placed in the loop to observe the characteristics of both probes with intermittent cold trap operation. A trace heater on the cold trap inlet line failed on 4-28-69.</p> <p>The loop was cooled down and repair initiated.</p>
Fission Product Screening Loop	<p>The new primary pump buss (failure reported last month) was installed 4-4-69. The loop was started up and the pump checked out at loop temperatures up to 1200 °F. Thermocouples mounted on the pump buss revealed no excessive temperatures. Following this checkout, calibration of the flowmeters was resumed and data obtained at 500, 600 and 800 °F. While increasing the temperature to 1000 °F, a heater near the primary pump failed and the loop was shut down. Removal of the insulation revealed a sodium leak (~20 g) in the primary pump throat. Analysis of this failure is in progress.</p>
Mechanical Properties Loop	Idle this period.
Cover Gas Evaluation Loop	<p>Fabrication and installation continued. A checkout of the electrical wiring and heater revealed minor discrepancies and repair is in progress.</p>

E. REACTOR SYSTEMS

1. Reactor Design

S. O. Arneson and R. C. Walker

a. Reactor Core

J. F. Wett

BNW-W discussions during a meeting conducted at Richland on April 25, resulted in agreement on minimum control worth requirements as a basis for proceeding with design activities. Primary safety system worth was established at 3.20% $\delta k/k$ exclusive of a stuck rod allowance (system design will consider the highest worth rod stuck), and the secondary safety/shim control system worth was set at 8.0% $\delta k/k$.

Technical Evaluation Board concurrence has been given to the following:

- CCDD-35 - First Core Fuel Assembly Component
- CSDD-31 - Reactor Core System
- CCDD-33 - Reactor Nuclear Control Component

b. Reactor Vessel and Shield

R. C. Walker and O. W. Priebe

The Request for Proposal (RFP) for the Reactor and Guard Vessels was submitted for AEC approval on 4/23/69. The RFP was approved by AEC on 5/16/69 subject to modification prior to formal submittal to interested qualified vendors. Also, RDT requested clarification of specific items in the RFP prior to contract negotiation and start of detail design.

A senior group of BNW Westinghouse and AEC personnel visited management of qualified vendors during the week of 5/5/69 to ascertain their interest in bidding on the RFP. Response appears favorable within the brief inquiry period.

The Conceptual Component Design Description for System No. 32, Reactor Vessel and Shield is being revised in accordance with comments received from RDT and W.

Babcock and Wilcox issued report No. BAW-1280-59, Topical Report Fast Flux Test Facility Reactor Vessel Conceptual Design Studies. This report covers the work performed by Babcock and Wilcox during the period of September 1968 through February 1969. The primary objectives of this effort were to assist in studies used in establishing a basic vessel configuration as well as feature studies required in the establishment of conceptual design. Additional conceptual feature studies by Babcock and Wilcox have been proposed by BNW but have not been approved by AEC-RL.

2. Reactor Equipment

S. O. Arneson and C. A. Munro

a. Reactor Hardware

L. R. Besel, J. W. Kolb, and B. G. Smith

(1) Core Structural Feature Test (L. R. Besel, W. C. Kinsel, and R. Hill). The Work Statement for the experimental program to provide the RPD with data for use in the FTR core and radial restraint design was prepared and discussed with Westinghouse and RDT representatives. Concurrence was obtained on single heated duct testing which consists of experimentally confirming the analytical models for thermal bowing and thermally induced creep of a flow duct. Another Work Statement will be issued for evaluation of the core restraint mechanism when agreement is reached on the test method to be used for investigating the effects of radiation swelling and thermal and radiation creep on the restraint mechanism

(2) Reactor Cover Model and Studies (R. Duncan and J. Darby). Conceptual prints have been completed to present possible methods of fabricating a full scale wooden or steel reactor

cover model and associated support stand. This model would be a design aid in locating and routing the reactor cover components.

In addition to the above effort, preliminary calculations and design for the Reactor Cover Mockup cover support structure and heating requirements were prepared. A conceptual print was completed to illustrate one method of providing hot gas heating to the underside of the simulated reactor cover in the Reactor Cover Mockup.

b. Closed Loop Nozzle Closure Program

(1) Mechanical Seals (R. Kolowith, M. R. Kreiter, and E. R. Antal). The objectives of this task are to: (a), provide experimental test data verifying the principle of operation of a creep compensating connector assembly; (b), provide experimental test data for each of six specified seal concepts being considered for FFTF closed loop nozzle closure application; and (c), establish a reference clamping device design and seal(s) for subsequent reliability testing.

- Seal Development. It was intended that the NDA 4-point and the nickel seals would be the first seals to undergo evaluation. However, quality assurance requirements imposed upon purchase of the materials have led to rejection of the material in one case and the inability to locate a material supplier in another. This has resulted in a total program delay of approximately three weeks thus far.

A seal module has been designed and is in BNW Tech Shops for fabrication for the NDA 4-point Armco ingot iron seal. An order has been placed for the iron for the seal but difficulty was experienced in meeting Quality Assurance requirements since material suppliers do not presently

have the capability to furnish a check chemical analysis or to perform ultrasonic testing. Armco Corporation has subsequently advised that material from the suppliers can be analyzed and ultrasonically tested in their research laboratory.

Preliminary design has been completed on a nickel seal test module and a seal loading force determination apparatus. A preliminary sealing face prototype was made to study hard-face welding techniques and hard-face machining procedures. The prototype was sectioned and welds were found to be satisfactory with adequate penetration, but the groove profile was ragged. The hard-face material appears to tear when machined. Improved machining techniques will be studied.

Requisitions to purchase seal test modules and seals from the Marman Division of Aeroquip Corporation and Haskell Engineering and Supply Company were issued and approved. The requisition for purchase of seals from Gray Tool Company has been issued and is now awaiting approval.

- Connector Development. Engineering and design on the creep compensating connector nozzles has been completed and the unit is now being fabricated in the BNW Tech Shops. All the material has been ordered and most of it has been delivered. Delivery has been held up on some forged items after the vendor's forging press broke down.

The design of the Belleville spring stack for seal loading and creep compensation has been completed. The design is based on a two-stage spring stack, a changeable spring stack for seal loading forces, and a spring stack to withstand DBA pressure before flattening.

A prototype of a changeable spring was fabricated from Type 4340 steel and hardened to the equivalent properties of 17-4 PH condition H-900, that will be used in actual

springs. A force-deflection diagram was obtained on the prototype spring and the results compared with the calculated data. The force required to flatten the spring was 8000 lb versus 8500 lb calculated. This difference is associated with the variation between calculated and fabricated dimensions.

All the manual actuator components have been received and fabrication is complete. The unit is now ready for use.

The remote actuator preliminary design has been completed and the design is now ready for review. Preliminary design schematic drawings and arrangement drawings of the sodium system to be used in the actuation test have been completed.

(2) Welding and Cutting (J. W. Koib, R. F. Gilmore, L. J. Rousseau, and F. C. Smith). Primary objectives of this task are: (a) demonstrate the feasibility of employing remote controlled, mechanically-driven, tungsten arc welding as a means of making a final reliable closure seal between the closed loop nozzle plug and reactor top face spool piece; and (b) evaluate mechanical cutting techniques on the simulated weld test pieces to facilitate repeated removal and replacement of the closed loop test specimens.

During this report period, the first reweld was made using the No. 1 test specimen. Temperature of the test specimen was maintained at 500 °F for a period of four hours, prior to the introduction of sodium at 500 °F. The weld was made with the sodium within 1 in. of the bottom of the weld insert filler ring. Runout of the weld torch was encountered during the test run. It was necessary to terminate the weld and reposition the weld torch before the sequence could be re-initiated and the weld completed. The appearance of the

completed weld was satisfactory and the sodium did not appear to affect the weld. The wild runout was caused primarily by the clamping mechanism which is being used on the preliminary weld head. The design of the prototype closed loop weld head is incorporating changes to eliminate this problem.

The detailed drawings of the modifications to the environmental chamber are being revised to minimize complexities and to incorporate proven mechanisms into the design. Revisions are also required to accommodate the Vitro prototype cutter.

Machine cuts performed on welded test plates to determine the cutter tool configuration for separation of the welded closed loop nozzle closures has produced a desired chip for a portion of the cutting process. Design modifications of the cutter bit and improved tooling material is expected to produce consistent chips that can be extracted away from the cutting area.

A trip was made to Industrial Magnetics, Inc. to investigate the hydrodynamic welding process. This concept consists of heating a tube joint area with an induction coil and then squeezing dynamically with an induced magnetic pressure for a fraction of a second. This process is being reviewed for possible remote maintenance applications in the sodium piping systems.

3. Test Engineering and Coordination

J. R. Carrell and R. L. Roske

a. General Electric Contract BDR-341

R. L. Roske

The BDR-341 contract has been redirected from the "Backup Core" effort and now consists of four technical tasks and a project management task. Activities on these tasks are as follows:

(1) Molten Fuel-Sodium Duct Interaction - The objective of this task is to evaluate the effect of transient temperature and pressures (induced by mixing molten fuel and sodium) on the subassembly duct in a severe loss-of-coolant accident.

The thermal transient upon loss-of-coolant flow was programmed with three simplified geometries to represent the core. The first divides a single fuel pin annular flow model into ten axial sections to study the axial and radial temperature profiles. The second model treats the edge fuel pin and the channel wall. The third model treats the circumferential variation around a fuel pin, including the wire wrap spacer. First results show that coolant boiling will occur between 0.5 and 1 sec. after loss of flow for superheats 0 and 400 °F, respectively. The duct temperature may increase about 100 °F before voiding occurs.

Studies of sodium voiding and re-entry were continued. Thermal transients of the core, reflector, plenum and sodium show that the reflector and plenum region acts as a heat exchanger for sodium being expelled and re-entering the core. The heat capacity of the reflector and plenum is much greater than that of the hot sodium expelled from the core, causing the sodium to be cooled to near the design exit temperature when expelled to the plenum. This results in a rapid reduction in voiding pressure and subsequent re-entry of the liquid slug. At an initial superheat of 400 °F, the sodium is expelled to about 15 in. above the top of the core and re-enters the core about 0.2 sec. after initial voiding. Voiding and re-entry 'chugging' will be followed until significant core melting occurs.

A parameter study of molten fuel-sodium interaction was completed using the heat transfer and thermodynamic models developed. Parameters studied were fuel, cladding and coolant

temperatures, and sodium superheat before vapor blanketing. Acoustic energies available from the interaction of molten fuel or cladding and sodium are in the range of 6 to 60 in.-lb/in.² of contact area. Available work from expanding sodium vapor is found to be about 1 to 5 in.-lb/in.² of contact area. The partition of this available energy to the duct will be investigated next.

Investigation of the duct response beyond the elastic limit has begun. Deflections and strain energies absorbed have been calculated when plastic hinges are formed at the hex corners and the middle of the flats.

Conditions which could lead to voiding and fuel melting of adjacent assemblies have been investigated. Distortion of the fuel pin, gas or vapor injection and flow blockages have been studied to characterize the resulting sodium and fuel temperatures and the time-to-failure. While small deflections of the fuel pin may cause cladding breach and release of fission gases, gross overheating induced by flow reduction should not occur until significant fuel pin distortion occurs. Sodium boiling would not occur at pin pitch-to-diameter ratios above 1.10 if uniform spacing were to be maintained.

(2) Dynamic Control Drive Seal. The objective of this task is to provide a dynamic sealing concept that is applicable to the FFTF control rod drive mechanism. This seal is to maintain separation of the sodium laden cover gas and the atmosphere within the control drives. Both environments operate at nearly atmospheric pressure but the cover gas will be at the design sodium outlet temperature of 1200 °F. The seal must accommodate a control rod stroke of 38 in. with a shim rate in both directions of 9 in./min and an initial scram acceleration of 3 g.

The literature survey has been continued during the reporting period. Additional contacts with vendors on seal designs,

materials and operating experiences have been made. Prime emphasis was given to the sliding seal designs with components that use dry lubricants or materials compatible for sliding duty.

A heat transfer model that describes the convection heat transfer from the shaft to the adapter plug wall in a time dependent fashion has been programmed. This is intended to provide more precise operating temperature distributions for the sealing parts which are in contact with the shaft and to estimate the amount of sodium vapor deposition.

Sketches of more than a dozen sliding seal design concepts have been made. Each of them has been drawn to scale, possible materials and/or commercial components have been selected, and a tentative ranking for FFTF application has been conducted.

(3) Axial Fuel Cooling. The objective of this study is to provide a conceptual design for an axial flow gas cooling device to remove decay heat from a fuel assembly during transfer within a fuel handling cell. Transfer is to take place between stations for cleaning, examination, disassembly, etc., and the fuel assembly is to be first withdrawn from a sodium-filled pot. The reference concept for axial gas cooling during fuel transfer has been modified to utilize downward suction of gas. This arrangement offers the following significant advantages over the previous methods utilizing upward suction during transfer reported last month:

- The flexible suction duct connection can be made with significantly less allowance for vertical motion, since the suction manifold is maintained at a low level.
- Valve sequencing is not required for transfer from gas cooling at fixed stations to gas cooling during transfer, since the direction of gas flow does not change.

- Less difficulty is encountered in handling fuel assemblies with long vertical extensions.
- The downward flow of gas improves the removal of residual liquid sodium from the fuel assembly.

The interfacing of this transfer device with fixed stations represents no significant change, except in the case of the disassembly station which is now required to have an independent upward suction cooling system for use during disassembly of the driver fuel assemblies. The need for an independent downward suction cooling system at the disassembly station for disassembly of the test fuel assemblies is now eliminated.

A comparison has been made between a flexible hose and a flexibly jointed tube for the suction connection. The reference design may be changed to utilize a two-piece tube with flexible joints at the ends and center. This arrangement had not been considered previously since the former arrangement required extensive vertical travel. Provisions are being made to permit interchangeability between the jointed tube and the flexible hose so that either option will be available.

(4) Design Provisions for Operation with Fuel Gas Emission.

The objectives of this task are to:

- Summarize the state-of-the-art for vented reactor fuel.
- Estimate the fission product release from future vented fuels in the FFTF.
- Recommend design features which should be incorporated in the FFTF design to cope with fission products from both failed and vented driver fuel.

Preliminary selections were made of the vented fuel characteristics (which directly affect the activity in the cover gas) as follows:

- Two percent of the fission gas is released instantaneously from the fuel matrix into the vent chamber.

- Ninety-eight percent of the fission gas is held up for five days before being released from the fuel matrix into the vent chamber.
- The holdup time of fission gas in the vent chamber will depend upon the vent design. Values from 0 to 40 days will be considered.

The cover gas activity was calculated using these characteristics. The worst case occurs when the vent holdup time is zero and the cover gas purge rate is zero. For this worst case the total cover gas activity is 10 million curies. The main contributor to the activity is ^{133}Xe .

b. Fin Tube Testing

D. J. Lagrou

The commercial fin tube vendors have not been successful in attaching Type 410 SS fin material of 3/4 in. fin height to a 1 1/2 in., Schedule 10S, Type 304 SS pipe with the high frequency resistance weld process. Kentube Company made several attempts to achieve the required continuous contact bond between the fin and tube, but could not produce a continuous bond because of difficulties encountered in wrapping the 3/4 in. fin height as a result of hardness and lack of ductility of Type 410 SS material. Contact with other fin tube vendors has provided the same fabrication problem data. Efforts will be continued in an attempt to attach a 5/8 in. fin height of Type 410 SS to the 1 1/2 in., Schedule 10S, Type 304 SS pipe with a continuous contact bond by the high frequency resistance weld process.

F. REFUELING AND RADIOACTIVE MAINTENANCE SYSTEMS

S. O. Arneson and W. E. Cawley

1. Conceptual System Design Descriptions (CSDD)

E. J. Ruane

System 41 - Reactor Refueling System

The System 41 CSDD has been approved by the BNW Configuration Control Board and transmitted to RDT.

System 42 - Nonirradiated Fuel Handling

E. D. Grazzini

The BNW Technical Evaluation Board has reviewed the System 42 CSDD and recommended it for approval.

System 43 - Irradiated Fuel Handling System

E. D. Grazzini

The BNW Technical Evaluation Board reviewed the System 43 CSDD and reported that some of the functions and interfaces with System 71 are overlapping and in some cases may be redundant. BNW redefined interfaces and equipment requirements between Systems 43 and 71 with the following results:

- a. System 71 will provide the facility and facility equipment for the fuel handling cell.
- b. System 71 will provide temporary in-cell storage for irradiated and/or contaminated core components.
- c. System 71 will provide the steam/argon cleaning station located in the fuel handling cell and share it with System 43. (Note: System 71 will use the cleaning station for irradiated and/or contaminated core components that are going to be examined. System 43 will use the cleaning facility for irradiated and/or contaminated core components that are going to waste disposal or reprocessing.)

- d. System 43 will provide the transfer port in the transfer tunnel.
- e. System 43 will provide the forced gas cooling/heating transfer device located in the fuel handling cell.

System 44 - Centralized Maintenance Facility

E. E. Garrett

The Configuration Control Board directed that System 44 should consist of centralized radioactive and nonradioactive maintenance facilities. The CFDD will be prepared under the direction of BNW with inputs from contractors.

G. FUEL EXAMINATION SYSTEM

E. M. Johnston and C. L. BOYD

1. Examination Equipment

W. S. Chenault and E. B. Ramsey

a. Neutron Radiography Equipment

The FFTF Radiography Facility Conceptual Design Study, GEIC-1114, IPO, General Electric Company, November 26, 1968, was distributed by BNW. The Neutron Radiography Facility concept recommended by the study utilizes a small, special-purpose reactor as the neutron source. The reactor is located in a reactor cell adjacent to the Interim Examination and Disassembly Cell. Transfer imaging of highly radioactive fuel, materials and components irradiated in the FTR is conducted in the Interim Examination and Disassembly Cell by directing a high-intensity, high-quality neutron beam into this cell through a sealed aluminum neutron beam window in the cell wall. Complete assemblies up to 15 ft in length or the lower 15 ft of longer assemblies up to about 45 ft in length may be neutron radiographed in 3-ft increments. A small shielded area is located next to the reactor cell to allow utilization of a second neutron beam for the direct imaging of nonirradiated fuels, materials and components.

b. Disassembly-Reassembly Equipment

BNW has contracted with ORNL for a conceptual design of a machine to disassemble and reassemble irradiated fuel assemblies remotely.

The initial contract work included a literature and technology survey for information applicable to the conceptual design. The draft of this survey entitled Survey and Analysis of Remote Technology Applicable to Disassembly and Reassembly of FTR Core Components was reviewed⁽¹⁾ and approved with comments by BNW for final issuance by ORNL.

1. FFTF Monthly Informal Progress Report, April, 1969, BNWL-1062, Battelle-Northwest, Richland, Washington, May 7, 1969.

c. Sodium Removal Equipment

A meeting was held at General Electric/BRDO, Sunnyvale, California, on April 30 where agreement was reached between BNW and GE on a work package for conceptual design of sodium removal equipment. The agreed-to scope of work will provide separate conceptual designs of sodium removal equipment utilizing (1) the argon moist air demineralized water rinse process for all FFTF core components except failed, vented, liquid metal bonded or carbide driver and test assemblies, and (2) the distillation process for failed, vented, liquid metal bonded or carbide driver and test fuel assemblies. A secondary objective is to identify development programs necessary to support follow-on design or development of the equipment or processes.

The work package and justification for selection of the two cleaning processes was forwarded to RDT for approval prior to placement of the contract.

2. Gas Cooling of the FFTF Fuel Assemblies During Inspection J. M. Yatabe

Before removal of the metal fuel assembly liner during disassembly, cooling will be accomplished by passing an axial flow of argon through the liner. If the liner cannot be slipped off easily it may be necessary to slit it, thereby allowing some of the coolant to escape. The purpose of the work reported this month was to determine the additional axial flow necessary to provide adequate cooling with a slit liner.

Experiments were performed using the electrically heated model of a fuel assembly which had been used in previously reported tests. This model contained 169 pins with a 36-in. heated length, 0.250 in. diameter, wrapped and spaced with a 0.030 in. wire on a 12 in. pitch. The pin bundle was surrounded by a close-fitting hexagonal liner. Two longitudinal slits were cut in the liner near two opposite vertices over

the 36-in. heated length. Measurements were also made with slit widths at 0.027 and 0.060 in.; a third test with a 1/8 in. slit will be completed soon. Air was passed from a blower, over the unheated (gas plenum) end of the heated bundle, over the heated portion and discharged into the atmosphere. At each slit width, runs were made over a range of power inputs with the air flow adjusted to give maximum pin temperatures near 600 °F.

The inlet flow required to maintain all pin temperatures below 600 °F was about 1.4 and 2.0 times as great for the 0.027 and 0.060 in. slits, respectively, as that required with an unslit liner. This agrees fairly well with flow increases by factors of 1.6 and 2.1 predicted using a previously developed mathematical model. This suggests that previously reported calculated flow increases required for slit liners are valid.

H. IRRADIATION FACILITIES

B. M. Johnson

1. Prototype Closed Loop System Test Plan

R. Kolowith, L. J. Defferding, and J. D. Smith

The purpose of the test plan is to provide a planned program in a logical sequence that will result in a design verification of the closed loop prototype system, including the nozzle, re-entrant tube, piping, pumps, and related hardware. The effort will include testing for stress analysis, vibration analysis, and verification of operational features.

Efforts in support of Task IT-8 consisted primarily of the coordination of efforts in preparation of the preliminary test plan for the prototype closed loop system. Included in this effort was the preparation and execution of the final direction for preparation of this test plan. The test plan is approximately 44% complete.

2. Closed Loop Systems Study

D. L. Koreis

The purpose of this study is to determine the testing capability of the reference conceptual design closed loop for off-design testing conditions. The testing capability of the loop is presented in terms of testing envelopes which define a range of admissible testing parameters.⁽¹⁾

The design point, used to size the heat exchanger components, is listed in Table 4.H.I.

The closed loop in-reactor tube is a re-entrant design with a 3.05-in. diameter test section. The IHX (intermediate heat exchanger) is a counter flow shell and tube arrangement with 146 ft² of heat transfer area. The DHX (dump heat exchanger) is a cross counter flow design with 2150 ft² of surface area on the air side.

1. FFTF Monthly Informal Technical Progress Report, BNWL-915. Battelle-Northwest, Richland, Washington, October 1968.

TABLE 4.H.I. *Closed Loop Design Point*

Test Power	5 MW
Test Fuel ΔT	400 °F
Coolant temperature leaving test fuel	1200 °F
$\dot{m}_p/\dot{m}_s$ (LBm/HR)/(LBm/HR)	1.0
IHX LMTD	100 °F
DHX Air Inlet Temperature	100 °F
DHX Air Outlet Temperature	500 °F

Steady-state results from the systems study show that for test ΔT 's of 400 °F, the admissible testing powers range from 1.2 to 5.7 MW. For test ΔT 's of 200 and 300 °F, test powers range from 0.5 MW (minimum power investigated) to 2.8 MW and from 0.5 to 4.3 MW, respectively. Allowable coolant temperatures leaving the test fuel, for test ΔT 's of 400 °F, range between 900 and 1200 °F. For test ΔT 's of 200 and 300 °F, allowable coolant temperatures leaving the test fuel range from 800 to 1200 °F.

A computer program capable of predicting axial and radial temperature distributions in the in-reactor tube for steady-state and transient conditions has been completed.

I. INSTRUMENTATION AND CONTROL

C. D. Swanson

The completion of the Conceptual System Design Description for the Plant Protection System (System 99) has been delayed to include changes recommended as a result of meetings between RDT and BNW between April 28 and May 2. It was agreed that basic design requirements should be rewritten to be more definitive. Plant Protection System general safety criteria which are applicable to many instrumentation and control systems will be placed in CSDD 99 and referenced by the applicable Instrumentation and Control Systems. Conceptual protective functions will be tabulated in Section 2 of CSDD 99.

1. Reactor Monitoring and Control

R. R. Cone

a. Reactor and Vessel Instrumentation - System 92

W. Dalos

Submission of the CSDD for System 92 to RDT will be delayed until the method for incorporating Design Safety Criteria has been resolved. The estimated submission date is July 7, 1969.

b. Fuel Failure Detection and Location (FEDAL) - System 94

R. R. Cone and W. Dalos

BNW provided assistance to Westinghouse in the FEDAL System CSDD preparation areas involving system interfaces and containment space allocations.

Recent calculations show that the 1.65 MeV gamma from the decay of Neon-23 which is created by the $^{23}\text{Na} (n,p) ^{23}\text{Ne}$ reaction will be generated in sufficient quantities to affect the FEDAL location system design. The decay of the ^{23}Ne in the location system gas detector loop provides a beta-gamma

background which will reduce the sensitivity of the location detection system if adequate sample delay time is not provided. Additional details on the investigation of ^{23}Ne are provided in the Shielding Analysis Section of this report.

2. Plant Instrumentation

J. W. Mitchell

a. Central Control and Data Handling - System 91

G. L. Waldkoetter

Draft copies of CSDD 91 have been reviewed by Bechtel and Westinghouse, and their comments have been received. These comments do not include any major disagreements with the concept, and many of them are directed at a clarification of intent.

Comments relating to the computer configuration have been made along with stated preference for a two computer system (by Bechtel) over a three computer configuration which is described in CSDD 91. The reasons given for this preference include cost and efficiency.

BNW is currently reviewing estimates of the required input scanning load, display load, and the calculational load. Using these values and the software system concept described in CSDD 91, estimates of program execution times are being made for typical 16-bit, 24-bit, and 32-bit computers. Early results indicate that a two computer configuration is feasible with a 32-bit and possibly with a 24-bit computer. However, changes to the concept cannot be justified until the review is complete and any changes in concept are contingent upon a favorable loading of each computer which allows future expansion for the experiments and/or increased calculational aids.

b. Plant Instrumentation - System 93

M. O. Rankin

The CSDD for Plant Instrumentation (System 93) was reviewed by RDT. Significant agreements after BNW/RDT discussions of these comments (April 28 to May 2) were as follows:

- (1) General Design Safety Criteria will be deleted from CSDD 93, but included in CSDD 99 (Plant Protection System). However, reference to these criteria will be made in CSDD 93.
- (2) The measurement lists requested by RDT will be limited to variables which would require development of instrumentation.
- (3) The Reactor Containment Instrumentation section will be rewritten to include all radiation monitoring equipment pertinent to containment operation.
- (4) BNW will investigate the possibility of using a venturi in the primary system to measure flow in addition to the EM flowmeter. A preliminary review of this item indicates that venturi's have the following disadvantages:
 - A nonrecoverable pressure loss of approximately 5% of the total pressure drop can be expected at full flow.
 - There is a requirement for a long straight section of piping before and after a venturi meter.
 - The cost of fabricating a venturi meter is high.
 - A venturi requires measurement of small differential pressures. The pressure tap lines are subject to plugging. Differential pressure transducers for high temperature sodium service are not readily available.
 - A venturi throat becomes a point of high velocity and is subject to erosion. Erosion of the throat changes the calibration.

- Very limited field experience related to the application of venturi's in sodium systems is available. Field experience with venturi's in sodium systems is limited to a few applications of small diameter units.

c. Process Data Display - Task IC-51

M. D. Erickson, B. E. Dozer, and B. R. Smith

Programs have been written to permit the 2-way transfer of data between the PDP/9 computer and the D-30 display unit. A program to plot graphs has been completed but not tested. Light pen routines (using the D-30 display unit) are being considered, but lack of information on required display formats prevents firm decisions on pen operations.

To date, the circuit modules and video camera have been received. The camera has been used with a 35 mm transparency to project a background for the display unit. While these tests were favorable, additional work is necessary to produce an image with more contrast.

A scan converter has been ordered. This device converts conventional x-y analog voltages into signals compatible with the raster scan TV system used on the D-30 display unit. This device will be used to study trend displays.

d. Control and Data Handling System Development - Task IC-50

Current work on debugging the program for the FFTF Monitor System model on the SEL 840A computer has been completed. All parts of the basic system have been made operational and the total system has been operated successfully. Minor parts of the operating system are being revised to permit more efficient operation. These include such features as interrupt handling and character formatting. Additional operation codes

are being written which will further test the real-time capabilities of the system, and additional I/O routines are being developed to permit simultaneous operation of a number of I/O devices.

A test program is being implemented to demonstrate some of the real-time features of the operating system. This program will currently be limited to devices related to nonprocess I/O functions, such as a Teletype. This program is presently in the debugging stage.

A compiler program is being written to permit the assembling of application programs (tasks) directly from the Operation Codes. This program is still in the development stage.

Staff and Training Estimates - Instrumentation

G. G. Thieme

Estimates for FFTF instrumentation staffing requirements have been defined on the basis of presently foreseen needs. These estimates show that 21 Craft people are needed to support instrument maintenance and repair, including the digital computers. In addition, it is estimated that five technical people are needed for instrumentation engineering for the process and for tests, and two programmers are needed for software maintenance and changes. NPTF would require two additional people in support of that facility.

f. Plant Operation and Control Document

G. G. Thieme

A report, Plant Operation and Control, BNWL-1023, was officially transmitted to RDT on May 5. This document describes an approach to plant operation of the FFTF and outlines control of the reactor and main heat transport systems and the closed test loops. In addition, this document provides support data for the Central Control and Data Handling System (CSDD 91).

3. Instrumentation Development

M. R. Wood

a. FFTF Low Level Neutron Flux Instrumentation (IC-4)

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

Under combined high temperature and intense gamma environments, the RSN-10A fission counter experienced some permanent though relatively minor changes at temperatures well above specification limits. Evidence of possible incipient failure occurred during exploratory tests at 900 °F for both of the two available RSN-10A sensors.

As revealed in Figure 4.I.1 temperatures of 820 and 900 °F caused an upward shift in the bias curves, particularly in the region corresponding to 4 to 10 V bias values. After cooling to 100 °F, the bias curve remained at a higher value corresponding with the recorded curve for the 820 °F combined environment. Exposure to 600 °F, slightly above the specification limit of 570 °F, caused no permanent or temporary change in the detector signal characteristics. The precise superpositioning of the curves for 100, 330, and 600 °F indicates stable operation during the exposure.

Two supplemental tests provided additional detailed information about operation of the system with specific configurations, as noted in Figure 4.I.2. Measurements with a timer-scaler revealed the shape of the integral bias curve at high count rates

Neg 0691979-1

FIGURE 4.I.1. Combined Gamma and Temperature Effect on Fission Counter (Preliminary Data)

FIGURE 4.I.2. *Effect of Shaping Time Constants on Neutron Sensitivity for RSN-10A Fission Chamber at 5×10^5 R/hr, γ -Background, Chamber Collection Time = 250 nsec*

corresponding to 0.1 to 1.3 V bias values. The system provides a capability for count rate measurements well beyond the usual instrumental cut off that occurs at 300,000 counts/sec in the log count rate meter. The second test yielded a steep bias curve located near the ordinate that indicates the background noise generation of the instrumentation components with cable and sensor not connected to the system. The small amplitude of this noise response indicates a favorable signal-to-noise ratio and subsequent measurements show this system noise amounts to only a fraction of the background signal from alpha pile-up inherent in the fission counter.

Throughout this investigation, improved gamma pile-up tolerance was found to occur for the faster time constants. Consistently fast pulse shaping and consequently short system

revolving times tend to improve gamma pile-up rejection. In particular, time constants much shorter than the detector collection time provide significant improvement, as shown in Figure 4.1.2. Best results occur with the 20 nsec pulse shaping, the upper limit of available instrumentation. However, as the time constants reduce, loss occurs in signal pulse amplitude, thus requiring the use of greater system gain. At present, 50 nsec pulse shaping seems to offer a reasonable compromise between gamma rejection and gain loss.

The experience of Goodings⁽¹⁾ with a 1-in. diameter, argon filled fission chamber with a 250 nsec collection time, provides further evidence that faster shaping time constants yield greater gamma tolerance. A chamber with faster collection time (factor of 4 or more) promises to improve system performance. With a "faster" chamber, the collection time could match the pulse shaping time constants of 50 nsec. The BNW tests used a fission counter with 200 nsec collection time.

A short piece of high temperature cable was given to ANL to test in their facilities as a first step in a cooperative effort in developing neutron detecting systems.

b. FFTF In-Reactor Coolant Temperature Sensors, (IC-13)
N. C. Hoitink, J. L. Jackson, and D. C. Thompson

Fabrication of the new equalizing block has been started. The 6-in. long by 2 5/8-in. diameter aluminum-bronze assembly will accommodate 34 test thermocouples and a standard. Based on good performance data obtained last year on a similar, but shorter (3-in. long by 2 1/2-in. diameter) block, the new unit should prove to be fully satisfactory. Upon completion of the block, tests will be initiated to ensure that thermal gradients

1. A. Goodings. A Note on the Irradiation Testing of Pulse Fission Chamber Design, AEEW-M-493 (1964).

in the block meet acceptable limits and to determine whether or not the thermocouples can be locked in position with the present design in a hot cell.

c. Pressure, ΔP Flow and Level Sensors for Sodium Service (IC-11)

R. D. Crosier and K. O. Creek

Pressure testing (at constant temperature) of a Consolidated Electrodynamics Corporation unbonded strain gage transducer, Type 4-361-0001 SN 2260 (rated at 600 °F and 250 psi) has been completed at 625 °F in the sodium pots. The test is being rerun at 650 °F. A second CEC transducer of the same type is on hand and is ready for testing.

Consolidated Controls Corporation LVDT Type 412M44-30 transducer rated at 900 °F and 30 psi is being prepared for testing. Failure analysis of the CCC unit 412M44-200 S/N1439, which had previously failed, has been completed by the vendor and a failure report is being written by the vendor.

The Kaman-Nuclear transducer IC-1709 SN892, rated at 900 °F and 30 psia, was destroyed at Kaman during failure analysis and will be replaced by a new transducer. Cause of failure was a faulty internal wire-to-coil connection. The new transducer has passed a 1000-hr factory test at 1000 °F and has been received at BNW for testing in the sodium pots.

Two HRC Electronics transducers, Type PT-1000, rated at 850 °F and 250 psia, are on order and are in test at the vendor's plant.

Both Barton-Statham hybrid NaK systems are undergoing post failure analysis. Unit Pa-822 S/N 2983 is being returned to the Barton Company for examination.

d. Driver and Test Position Connector Concepts (IC-17)

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

Thermocouples and Belleville springs for use in the first closed loop connector have been received. A load versus deflection test was performed on the springs and found to be satisfactory.

Insulation resistance tests were performed on Physical Sciences Corporation Silfex II cables. Insulation resistance was found to be 334 M Ω at 500 V dc and 1200 °F. This is well above the acceptable level of one M Ω at 1200 °F specified for the instrument connector.

A leakage current transient found in high temperature lead wire at 1200 °F, will require that the insulation resistance tests be modified to allow a two-minute delay from the application of the voltage to the insulation resistance measurement.

A comment draft of a Quality Assurance Plan for Task IC-17 has been completed.

J. FEDAL SYSTEM

B. M. Johnson

1. Fuel-Failure Sample Lines

D. W. Shannon and P. L. Koehmstedt

Authorization to proceed with design of the SG-11 FEDAL test loop was received. Scope design of the FEDAL test loop to incorporate testing capability for the alternative configurations has been initiated.

2. Channel Instrumentation Development (IC-23)

J. M. Yatabe

The present FEDAL system gas disengagement 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. Fabrication of the first vortex generator is approximately 50% completed. Delays occurred as a result of fabrication difficulties; however, these problems have been overcome and progress has resumed on fabrication of the first test generators.

K. SODIUM AND GAS TECHNOLOGY

B. M. Johnson

1. Corrosion and Chemistry

D. W. Shannon

a. Evaluation of Radioisotope Transport and Deposition

W. F. Brehm, E. A. Kovacevich, and J. J. McCown

The metallographic examination of Radioisotope Transport Loop components has been completed and the specimens submitted for microprobe examination. Evaluation has continued on the radioisotope deposition patterns. Sections of pipe near flow transitions were cut up into 0.25-in. increments, dissolved in aqua regia, and the solutions counted in the well of a NaI crystal. The deposition of radioactivity in general shows an increased deposition in the velocity transition as illustrated schematically in Figure 4.K.1, but the method is not completely satisfactory because of the difficulty in obtaining uniform specimens. Use of an in-place technique with the diode detector will be attempted next.

FIGURE 4.K.1. *Schematic Representation of Deposition Near Flow Perturbation*

b. Selection and Testing of Radioactive Sodium Analysis Methods

(1) High Level Analytical Facility Design (J. F. Jarosch and L. F. Lust). Construction of the air atmosphere portion of the shielded mock-up facility has been completed and temporary installation in EDL-102 of Building 324 has begun. The initial design of the inert atmosphere mock-up analytical cell is nearly complete. The entire mock-up is to be located in Building 335 upon its completion and will be used to test remotely-operated analytical equipment and procedures, using nonradioactive sodium metal samples. Consideration is being given to the feasibility of designing the inert portion as an evacuable chamber to permit rapid restoration of the pure inert atmosphere after contamination with air.

A necessary step in the amalgamation oxygen procedure is the removal of oxide from the outside of the sample tube. This is presently accomplished by mechanical buffing but a buffing operation will be difficult to perform with a master-slave manipulator. A program has therefore been initiated to evaluate chemical cleaning methods which lend themselves to remote handling.

c. Selection and Testing of FFTF Sodium Sampling Methods
K. C. Knoll, T. J. Owen, and J. Gibson

Installation of the bypass loop sampler on the Building 314 loop is complete except for the final welds on the sodium supply lines. Helium leak testing of the manually operated portion of the loop has been completed and the few minor leaks discovered have been corrected. Consistent leak-free replacement of removable sample tube sections has been demonstrated using new Conoseal gaskets during each replacement. As expected, difficulty was encountered when an attempt was made to reuse gaskets.

The initial test with master-slave manipulators have revealed the need for several significant modifications to the sampler, and plans have been made to complete them prior to the start of sodium flow tests of the sampler.

Progress has been made toward the development of a crimp sealing technique for sodium sampler tubes. Seals which will not pass air under pressure have consistently been made on sodium samples contained in annealed nickel tubing. Less than half of these seals passed the more sensitive helium leak test however. The investigation is continuing.

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

G. B. Barton and H. P. Maffei

(1) Selection of FFTF Gas Analysis Instruments. A report entitled Instrumentation for Analysis of the Inert Gases Used in FFTF by G. B. Barton has been issued as BNWL-957. It can be summarized as follows:

Inert gas monitoring needs of the FFTF are reviewed and the selection criteria for the required instrumentation are defined. Current technology and commercially available on-line gas analytical instrumentation are reviewed and compared to FFTF requirements. Results of this study conclude that FFTF cell blanket gas analytical needs can be met in almost all instances by commercially available instrumentation. Current technology is inadequate to meet sodium cover gas monitoring needs and a development effort is required. Gas chromatography and possibly mass spectrometry appear to hold the greatest promise of meeting sodium cover gas monitoring needs.

(2) Mass Spectrometry. A laboratory study of the use of heated sample lines during the analyses of sodium cover gas has been completed. In the study, argon gas containing known concentrations of hydrogen and methane was analyzed by the

time-of-flight (TOF) mass spectrometer after passing through molten sodium and both heated and cooled sample lines. Little difference was observed between measured impurity concentrations using heated and unheated lines. Blanks were significantly higher on the samples passed through heated lines induced by the outgassing of the lines.

Based on this observation, previously reported data, and considerable experience with the TOF instrument, it has been concluded that the gas chromatographic approach is better suited for the in-line analysis of FFTF cover gas. The evaluation of the mass spectrometric approach will be concluded by documenting the laboratory investigation and conducting a series of tests in which the instrument will be operated in parallel with a gas chromatograph on the cover gas of an operating sodium system.

(3) Gas Sampling System Development. Measurements have been completed on three more cores designed for the reference removable core sodium vapor trap. One of the new cores was packed with stainless steel Raschig rings.

The goal set for trap performance is the attainment of equilibrium conditions with liquid sodium at 180 °C which is equivalent to a sodium vapor concentration of about 0.5 ppm. Six trap designs have been evaluated to date. Two of the six hold promise of attaining the performance goal at low flow rates. Efforts are continuing to develop a trap design of reasonable size which will attain the performance goal at high flow rates.

(4) Carbon Dioxide - Sodium Reaction Rates. Initial investigations to determine the reaction rates of sodium and carbon dioxide were initiated during the past month. It has been demonstrated, however, that small amounts of sodium hydroxide in the sodium vapor traps and gas sampling lines completely interfere with the measurement of carbon dioxide in

the effluent gas, thus far preventing reaction rate measurements. With sodium hydroxide present, all of the CO_2 was completely absorbed even at room temperature. This probably means CO_2 analysis will be questionable if sodium vapor collects in the sodium vapor traps and gas sampling lines. The sodium in these systems would be converted to hydroxide whenever the system was opened for maintenance and exposed to atmospheric moisture resulting in complete absorption of CO_2 on restart-up of the system.

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

J. J. McCown

(1) Oxygen Meter. A Na_2O addition to the sodium purification loop followed by cold trap cleanup has demonstrated that the oxygen meter is still operating satisfactorily. No evidence of a "buffer effect" or drift to lower oxide values following an addition has been experienced on the BNW loop.

Day to night variations in ambient temperature of 90 to 115 °F are being experienced which result in meter output signal variations of 1155 to 1185 mV. United Nuclear has recently changed their method of assembling oxygen probes and state that they are no longer sensitive to ambient fluctuations. One of their recently manufactured probes has been ordered and will be installed on the sodium purification loop for evaluation.

(2) Carbon Meter. Two additions of carbon compounds to the purification loop were attempted in order to study the carbon meter reaction. The meter did not respond to either NaCN or Na_2CO_3 ; subsequent analysis of the addition tubes indicated very little if any material was dissolved. The next test will be done using a carbon rod in the addition tube.

f. Evaluation of FFTF Sodium Cleaning Requirements

G. R. Bloom

Aqueous sodium processes are being evaluated for sodium reactor fuel elements and components. Tests in the evaluation include evaluation of corrosion of sensitized Type 304 SS during storage if NaOH is left on the metal. The initial tests are to establish the most severe damage that would be caused to compare with later, less aggressive test conditions.

Type 304 SS tubing samples from the Radioisotope Transport Loop were exposed 600 hr at 1300 °F with flowing sodium on the inside and 98% nitrogen and 2% oxygen on the outside. The storage test environments were:

- (1) Stagnant tap water for one week
- (2) Caustic on the metal and stored in air for 24 hr at 600 °F
- (3) Caustic on the metal and stored in air for 24 hr at 1400 °F.

The caustic was formed by heating the samples in air and allowing the unremoved sodium to form sodium hydroxide. The test of the samples has been completed and they have been examined metallographically for intergranular corrosion. Samples heated to 600 and 1400 °F were oxidized severely. The oxide formed at 1400 °F was black in color and flaked off when the sample was cooled. The sample heated to 600 °F was covered with a bright green coating. The samples heated to 600 °F were reddish brown in color after the green crystals were removed. The sample exposed to stagnant tap water for one week showed no brown stains or iron oxide deposits. However, there were small light-colored spots which were spread uniformly over the sample.

Metallographic examination showed intergranular penetrations in samples heated in air with caustic. The tube sample heated to 1400 °F had general oxidation up to 1 mil deep both inside and outside with intergranular penetrations. The sample heated to 600 °F had no general measurable oxide. Evidently the observed brown iron oxide coating was quite thin. There

were intergranular penetrations up to 0.25 mils deep inside where the original sodium exposure took place and up to 0.5 mils deep on the outside where the original nitrogen and oxygen exposure had taken place. Metallographically the sample exposed to stagnant tap water appeared to be no different on the inside from the sample examined after only a sodium exposure and alcohol cleaning. Future tests will determine the weight change data for the water test.

Conclusions are that caustic corrosion will give intergranular penetrations in 24 hours at temperatures as low as 600 °F in air. Future tests will include temperatures below 600 °F and longer times.

g. Decontamination

G. R. Bloom

Chemical cleaning procedures used for decontaminating water reactors are being tested for decontamination of sodium reactors. One proposed procedure would (1) oxidize a layer of stainless steel about 0.1 mils deep and (2) remove the oxidized layer with a standard water reactor decontamination process. The activation products which have diffused into the stainless steel would be oxidized along with the stainless steel. They would then be removed in the second step of the decontamination process along with the oxide film.

The contaminated samples used were Type 304 SS tubing sections which were exposed to flowing 1300 °F sodium for 6000 hours in the Radioisotope Transport Loop. The radioactive species were diffused into the samples in the Radioisotope Transport Loop under prototypical FFTF closed loop operating conditions as described in previous progress reports. Initial tests have demonstrated that without initial oxidation a 0.4M ammonium citrate solution at 203 °F for four hours caused no detectable intergranular attack or decontamination on Type 304 SS samples.

A similar sample was oxidized with 1023 °F steam at 300 psi for 96 hours. This step was followed by a permanganate-ammonium citrate process to remove the adherent black oxide film. Counting techniques demonstrated that the decontamination factors for the radioactive species present were 2.4 for ^{60}Co , 1.1 for ^{54}Mn and 1.6 for ^{51}Cr . Calculated uniform penetration from weight loss data was 0.3 mils. The oxide film was not completely removed so higher decontamination factors and a higher uniform penetration for one cleaning cycle should be possible.

L. METALS, MATERIALS AND CODES

J. C. Tobin

1. Materials Applications

J. E. Irvin

a. Uniaxial Creep

A. J. Lovell

The uniaxial creep-rupture testing of Type 304 and 316 SS specimens aged in sodium for time and temperatures which simulate the thermal history of MK-B-X022 EBR-II subassembly, has been initiated. Preliminary data indicate that there are some significant effects on the creep-rupture properties induced by the aging treatment. The general trend indicates increased creep rates and lowered rupture lives after thermal soaking. The changes in such values as structure and chemistry, that could effect the changes in creep-rupture properties are being investigated.

The difference in creep flow characteristics between a Type 304 SS, solution-treated control specimen and a Type 304 SS specimen aged for 3045 hr at 800 °F is illustrated in a longitudinal view of the fractures (see Figures 4.L.1 and 4.L.2) for a 1000 °F, 45,000 psi creep-rupture test. The aged specimen grains are elongated, but the solution-treated control shows little evidence of any grain deformation.

b. Subcritical Crack Growth

E. B. Schwenk and L. A. James

The objective of this work is to determine the fatigue crack growth behavior of Type 304 SS at temperatures and environments that are expected to be characteristic of FTR piping system operation. During operation many of the piping system components may be subjected to cyclic loading that can cause the initiation and growth of fatigue cracks. Such

Neg 469175A

50X

FIGURE 4.L.1. Fracture of Type 304 SS Solution-Treated Control
Creep Tested at 1000 °F, 45,000 psi, in Helium.

Neg 469177A

50X

FIGURE 4.L.2. Fracture of Type 304 SS Solution-Treated Plus
Aging for 3045 hr at 800 °F in Static Sodium,
Then Creep Tested at 1000 °F, 45,000 psi, in Helium.

cracks may initiate at strain concentrators such as geometric discontinuities or flaws and propagate due to variations in system pressure, temperature and external loadings. Analysis of a failure survey covering general primary pressure piping systems indicates that crack growth accounted for about 55% of the failures and that fatigue was a dominant failure mode.⁽¹⁾ Knowledge of the fatigue crack growth behavior in conjunction with the application of fracture mechanics analysis techniques can provide a measure of the remaining service life of flawed or cracked system components.

Basic fatigue or stress versus cycles to failure (S-N) data on metals provide a measure of the number of load cycles to initiate cracks. However, little information can be gained from such tests on the number of load cycles required to grow a fatigue crack a given distance from a pre-existent crack or flaw of a given size. Fatigue crack growth rates can be determined using appropriate specimens, and can be related to the stress intensity factor K used in linear elastic fracture mechanics. The equation expressing this behavior, is known as the power law of crack propagation,

$$\frac{da}{dn} = c(\Delta K)^n \quad \text{where}$$

$$\frac{da}{dn} = \text{Crack growth rate (inches/cycle)}$$

$$\Delta K = \text{range of stress intensity factor (ksi } \sqrt{\text{in.}})$$

$$c, n = \text{constants for a given material, metallurgical condition and test state (for example, high temperature and sodium environment).}$$

-
1. E. B. Schwenk and L. A. James. Estimate of the Fatigue Cracking Behavior of the Fast Test Reactor Primary Piping System, BNWL-1035. Battelle-Northwest, Richland, Washington. (In Publication)

Twelve fatigue crack growth specimens have been machined from 1/2-in. Type 304 SS plate that was characterized under the Irradiation Effects to Reactor Structural Materials Program. Initial tests will be conducted at room temperature to determine crack growth rates as a function of several load ratios and crack orientations.

c. High Strain Rate Effects

J. M. Steichen and R. G. Hoagland

The objective of the high strain rate studies is to evaluate the effects of high strain rates on the mechanical properties of materials which are expected to be used in the FTR vessel and other critical components. The results from this program are needed to provide information on the behavior of the vessel during impact from within and on the behavior of core components during thermal transients or other conditions of rapid loading.

The parameters to be investigated are based on FFTF requirements and on the results of a literature survey of the effects of strain rates on austenitic stainless steels. Initially, Type 304 SS will be evaluated at temperature and strain rate ranges of 500 °F to 1400 °F and 0.01 sec^{-1} to impulse, respectively. Since fast neutron irradiation will likely alter the high strain rate behavior of these materials, a significant portion of the program will be performed on irradiated specimens.

Most of the initial testing will be done with a Draft-Hahn type of dynamic loader. This loader is a pneumatically operated loading machine which is capable of providing strain rates of about 30 sec^{-1} . Strain rates in the impulse range will be obtained by using a Hopkinson bar-block machine. The new equipment will provide strain rates from about 20 sec^{-1} to about 150 sec^{-1} .

The dynamic loader was calibrated and a small furnace to fit the confined test space was built. The Type 304 SS specimens (miniature button heat type) for the high strain rate effects testing have been acquired. The metal has been well characterized under the Irradiation Effects on Reactor Structural Materials Program. Temperature control along the length of the specimen has been demonstrated and testing was initiated.

d. Radiation Effects on Absorbing Materials for Control Rods

D. E. Baker and D. E. Mahagin

The purpose of this task is to determine the extent of irradiation damage to boron carbide and tantalum, candidate materials for use as neutron absorbers in the FTR control rods.

The B_4C capsules irradiated to moderate fluence levels in EBR-II have been transferred to BNW, and a detailed examination of the specimens has been initiated.

The first of two assemblies for thermal-reactor irradiation of B_4C was charged into a Hanford test facility. Five different forms of B_4C are being investigated in the test: 60 and 80% TD powder, and 65, 80, and 99% TD pellets. Sample temperatures range from 500 to 1000 °F as measured with sheathed chromel-alumel thermocouples. The indicated temperatures are lower than the design temperatures, and the source of the discrepancy is under investigation. Fabrication of the second test assembly was completed and it will be charged during the next reactor shutdown.

e. FTR Vessel Heat Treatment Considerations

R. A. Moen

Planning for the fabrication, delivery and erection of the FTR vessel involves detailed consideration of the various possible heat treatments that may be given to this component.

Several requirements which to some extent are satisfied by heat treatment can be listed:

- The FTR vessel should be put into service in a dimensionally stable condition.
- The vessel material should be fully annealed to minimize or eliminate possibilities of stress corrosion cracking or intergranular corrosion.
- The vessel material should be free of residual stress to minimize the possibility of stress corrosion cracking.

Possible heat treatments available range from none to full solution anneal. In case of no final post-fabrication heat treatment, the assembled vessel will contain at least two circumferential weld seams that will be sensitized and contain residual stress.

If the vessel is given a partial stress relief anneal at a temperature just above the operating temperature (above 1000 °F), the vessel should be dimensionally stable in service, but will be sensitized, as well as contain some residual stresses.

Full solution anneal of the vessel would require heating the vessel to above 1800 °F followed by rapid cooling. Rapid cooling subjected to an assembled vessel of this type would reintroduce some residual stress. However, there are few furnaces available big enough to house a component the size of the FTR vessel so full-solution annealing is virtually impossible.

Of the possible heat treatments, a final post-fabrication stress relief at or slightly above the vessel operating temperature of 1000 °F is recommended as the most favorable means for achieving dimensional control.⁽¹⁾ From the standpoint of stress corrosion cracking possibilities, the most effective action to

1. R. A. Moen. *Unpublished Communication. FTR Vessel Dimensional Control*, Battelle-Northwest, Richland, Washington, May 2, 1969.

prevent its occurrence is to establish procedures to reduce the exposure to chlorides during fabrication, shipping and construction.⁽¹⁾ This must be done regardless of which heat treatment is selected. Accomplishing the chloride control then permits one to ship and erect a vessel containing some residual stress and sensitized material, with a minimized chance for stress corrosion cracking prior to reactor startup.

2. Fabrication Development

R. N. Johnson

a. Duct Welding Development

W. F. Brown and T. B. Correy

The automatic tube welding machine to be used in the joining of duct components was scheduled for inspection at the Vendor's plant May 16, 1969.

Inconel 718 and Stellite 6B sheet materials have been ordered for hardfacing attachment studies, and the Inconel 718 has been received. Forming of Inconel 718 pads to fit the hexagonal tubes is proceeding.

A survey was made of ten manufacturers engaged in different types of welding or metallizing, with the objective of obtaining suggestions for methods of attaching wear resistant pads to the fuel duct surfaces. Several interesting proposals were received and involved such processes as explosive welding, electron-beam welding, co-extrusion bonding, tungsten-arc welding, plasma spraying, hydrodynamic welding, and resistance welding. The merits of each process or combination of processes are being evaluated for possible incorporation into the welding development program.

1. D. P. O'Keefe. *Unpublished Communication. Stress Corrosion Cracking, Battelle-Northwest, Ricland, Washington, May 2, 1969.*

b. Duct Gaging and Characterization

R. H. Todd and R. N. Johnson

A purchase specification and work package for a gaging system capable of characterizing the FTR fuel duct has received final approval of the Technical Evaluation Board. Duct drawings have been completed, checked, and approved for release with this specification. Replies from advertisements in the Commerce Business Daily for interested vendors have been received and the list of potential vendors has been completed. The Request for Proposal was completed and scheduled to be sent out during the week of May 12.

c. Duct Procurement Activities

R. N. Johnson

Specifications and drawings for the fuel duct have completed the necessary review and comment circuit, and appropriate modifications have been made. Final Technical Evaluation Board approval was scheduled for May 16, and the Request for Proposal scheduled for release to prospective vendors the following week.

d. Closed Loops - Vendor Survey

S. M. Gill

More than a year ago, a vendor survey⁽¹⁾ was performed to evaluate potential vendors for fuel ducts and closed loop tubes. At that time, the closed loop concept was not well defined. Subsequently, to coordinate development schedules, funding was purposely deleted and work stopped for more than six months.

1. R. C. Aungst. Fuel Duct and Closed Loop Vendor Qualification Program, Report of Visits to Potential Vendor Plants, BNWL-695. Battelle-Northwest, Richland, Washington, March 1968.

Present fabrication plans stipulate off-site construction of two prototypical closed loop tube structures with delivery to BNW by 7/71. To this end, additional vendor evaluation is deemed necessary to ensure receiving adequate and competitive proposals. Actions and results to date are as follows:

- (1) An advertisement was placed in the Commerce Business Daily. Of five responses received, only one is deemed qualified to construct the loop tube structures.
- (2) Letters requesting an indication of interest were sent to six aerospace/aircraft/nuclear companies prejudged to have the required qualifications. Of these, two responded "no interest," and the other four indicated interest. Visits have confirmed that these four are probably qualified.
- (3) An additional potential fabricator was visited to confirm his interest.

As a result of the above actions, a list of six probable qualified fabricators is available at this time for the closed loop tube prototypes.

e. Strain-Age Cracking Study

S. M. Gill

The FFTF reactor vessel and connecting piping will be made from one of the 300 series stainless steels. Type 304 SS has been recommended for the vessel. This structure will be put together from plate, nozzle forgings, rolled rings and similar structures, with thicknesses in the range of two inches. The vessel will contain numerous circumferential and vertical welds. The results of the study of one of these mechanisms, cracking in the heat affected zone (HAZ) of welds, are reported below.

When heavy sections of austenitic stainless steels are reheated after welding, either for stress relieving or during

high temperature service, there is a danger that cracking may occur in the HAZ if the restraint is sufficiently high. It has been shown that cracking is the result of the formation of a fine strain-induced precipitate of columbium or titanium carbide within the grains of the HAZ. This precipitate stiffens the grains so that during the creep deformation which results in the relief of stress at high temperature, high stress concentrations may occur at certain positions at grain boundaries, leading to intergranular cracking.

The austenitic stainless steels are increasingly susceptible to strain age cracking in the weld heat affected zone in the order of increasing niobium or titanium concentrations, with Type 347 SS being the worst, Type 321 SS next, and Types 316 and 304 SS least and about equal. Trace amounts, such as 0.05% Cb or Ti can cause susceptibility in the 316 or 304.

The cracking occurs at temperatures in the 1112 to 1562 °F range. The greater the niobium (or titanium) content, the lower the cracking temperature. The implication of any cracks in the FTR vessel and/or piping are obviously serious. It should be recognized that the FTR will operate at the low end of the temperature range in which this phenomenon has been observed, and Type 347 SS, the most susceptible grade, will not be used. The possibility of strain age cracking in the HAZ of welds can be precluded by:

- (1) Using only Types 304 and 316 SS with a 0.02% maximum (each) of niobium and titanium content, and
- (2) Performing appropriate weld and stress relief tests on each heat of steel used, including metallographic examination of the HAZ for precipitates.

3. Component Surveillance

J. W. Helm

a. Calculated Damage Functions

G. E. Russcher and J. W. Helm

The damage functions to be used in determining the relative effectiveness of different reactor neutron spectra in producing radiation damage have been calculated.⁽¹⁾ These calculated damage functions are presented as a function of energy for the HOMSET, COMPRASH, and Russian neutron energy group structure in Tables 4.L.I, II and III. For determining the relative effectiveness of various designs for the production of radiation damage the clustered vacancy ($N > 10$) damage function should be used, as it is the more realistic representation of damage at temperatures from 500 to 1200 °F. This function will be used for updating the present fluence limits and EBR-II core Row 2 midplane will be used as the reference spectrum. The calculated damage per unit fluence for EBR-II Row 2 midplane ($D_{\text{EBR-II}}/\phi_{\text{EBR-II}}$) using the clustered vacancy damage function is 2.2.

The calculated damage functions are for stainless steel and the calculated irradiation has no fluence limit since the effectiveness is based on fluences that produce the same damage. The damage functions relate irradiation effectiveness only in terms of differences in neutron spectra and other parameters such as material, purity, and temperature must be held constant to obtain meaningful correlations.

1. W. B. McDonald. Unpublished Data. Calculated Damage Functions for Determining Irradiation Effectiveness, MMR-15. Battelle-Northwest, Richland, Washington. (Letter to D. C. Spencer, dated April 23, 1969.)

TABLE 4.L.I. *Neutron Damage Functions - Weighting Factors for HOMSET Group Structure*

HOMSET Group Number	Neutron Lower Limit Energy (eV)	Weighting Factors	
		Gross Displacements F_{GD}	Clustered Vacancies $N \geq 10$ F_{CV10}
0	1.0×10^7	100.0	4.0
1	6.065×10^6	109.8	4.2881
2	3.679×10^6	113.5	4.3885
3	2.231	119.3	4.619
4	1.353	101.8	4.180
5	8.209×10^5	72.14	2.936
6	3.877	51.61	2.222
7	1.832	24.43	1.076
8	6.738×10^4	15.25	0.6884
9	2.479	10.70	0.1393
10	1.930	5.603	0
11	9.119×10^3	3.329	0
12	3.355	0.9642	0
13	1.234	0.689	0
14	4.54×10^2	0.09558	0
15	2.754	0.02219	0
16	2.145	0.007966	0
17	1.013	0.01323	0
18	6.144×10^1	0.01296	0
19	2.902	0.01751	0
20	1.371	0.02540	0
21	1.068	0.03335	0
22	3.928	0.04278	0
23	1.445	0.06977	0
24	0.9764	0.1015	0
25	0.6826	0.1198	0
26	0.4140	0.1466	0
27	0.2775	0.1829	0
28	0.1860	0.2282	0
29	0.1247	0.2826	0
30	0.07595	0.3448	0
31	0	0.7631	0

TABLE 4.L.II. *Neutron Damage Functions - Weighting Factors for COMPRASH Group Structure*

COMPRASH Group Number	Neutron Lower Limit Energy (eV)	Weighting Factors	
		Gross Displacements F_{GD}	Clustered Vacancies $N \geq 10$ F_{CV10}
0	1.0×10^7	100.0	3.8
1	8.5×10^6	100.1	3.931
2	7.5	111.3	4.345
3	6.5	114.1	4.445
4	5.5	114.7	4.452
5	4.5×10^6	117.1	4.519
6	3.5	120.4	4.631
7	2.5	121.5	4.689
8	1.5	104.5	4.291
9	8.0×10^5	74.87	3.054
10	4.0	51.66	2.225
11	1.0	21.72	0.9509
12	5.0×10^4	11.98	0.4498
13	1.5	7.170	0
14	5.0×10^3	2.604	0
15	1.5	1.779	0
16	5.0×10^2	0.5775	0
17	1.5	0.01607	0
18	5.0×10	0.01325	0
19	1.5	0.02103	0
20	5.0	0.03723	0
21	1.5	0.06406	0
22	0.55	0.1089	0
23	0.14	0.1920	0
24	0	0.5635	0

TABLE 4.L.III. *Neutron Damage Functions - Weighting Factors for Russian(a) Group Structure*

Russian Group Number	Neutron Lower Limit Energy (eV)	Weighting Factors	
		Gross Displacements F_{Gd}	Clustered Vacancies $N \geq 10$ F_{CV10}
0	10.5×10^6	100.0	4.000
1	6.5×10^6	108.87	4.255
2	4.0×10^6	116.81	4.510
3	2.5×10^6	121.27	4.678
4	1.4×10^6	103.28	4.241
5	8.0×10^5	72.29	2.947
6	4.0×10^5	51.24	2.206
7	2.0×10^5	26.69	1.1715
8	1.0×10^5	17.268	0.7702
9	4.65×10^4	11.841	0.4165
10	2.15×10^4	8.110	0.03714
11	1.0×10^4	3.728	0
12	4.65×10^3	1.6066	
13	2.15×10^3	1.6740	
14	1.0×10^3	0.6346	
15	4.65×10^2	0.07838	
16	2.15×10^2	0.01385	
17	1.0×10^2	0.01322	
18	4.65×10^1	0.01440	
19	2.15×10^1	0.02017	
20	1.0×10^1	0.03105	
21	4.65×10^1	0.04230	
22	2.15×10^1	0.06206	
23	1.0×10^1	0.09287	
24	0.465×10^1	0.1308	
25	0.215×10^1	0.1872	
26	0	0.5313	

a. Group Constants for Nuclear Reactor Calculations,
 I. I. Bondarenko, Ed., translated from the Russian by
 Consultants Bureau, New York, 1964.

b. Inservice Inspection

J. H. Cox

(1) Surveillance and Inservice Inspection Requirements.

The BNW effort to establish firm requirements has been initiated. Input from the Westinghouse and Southwest Research Institute (SWRI) reports (following) are being evaluated as part of this effort. The Westinghouse and SWRI results are not to be used as firm requirements until this evaluation is complete.

(a) Westinghouse Task 12. The final report on the Westinghouse Task 12 study on surveillance and inservice inspection recommendations has been issued.⁽¹⁾ Surveillance and inspection requirements of the FTR reactor internals and reactor system components have been concurred by Westinghouse and BNW to assist in preventing failures that could affect the preservation of public health and safety, and to guard against failures that could have significant effects on plant availability, repair cost, and time. For the reactor internals, the recommended inspections include: continuous capability for diagnostic monitoring using noise analysis; frequent in-vessel general visual inspection under sodium using an acoustic viewing system; a materials program consisting of continuous irradiation of specimens in the FTR with periodic out-of-reactor testing; infrequent in-vessel detailed visual inspection under sodium using an acoustic viewing system; and infrequent, and only as required depending upon reactor operation and other inspection results, in-vessel and out-of-vessel dimensional measurements, visual inspection, and nondestructive testing.

1. J. J. Buggy, D. L. Dill and E. Paxson. Surveillance and Inservice Inspection Recommendations for the FTR: Techniques, Program, Development, WARD-3977-1, February 1969.

For the reactor system components, the recommended inspections include: Periodic visual inspection of the reactor vessel exterior and interior surfaces and head, primary piping system hangers and supports, valves, primary sodium pumps, and primary equipment cells; periodic ultrasonic nondestructive testing of selected welds in the primary piping system, and of the reactor outlet nozzle to shell welds; continuous vibration monitoring of the primary sodium pumps and valves; continuous temperature measurement of primary sodium pump bearings; continuous leak detection of valve seals; and periodic disassembly of selected valves. These recommendations are based on the FTR concept applicable at the time of the study.

The final report also includes a state-of-the-art survey dealing with inspection experience and techniques, descriptions of recommended inspection equipment and techniques, recommendations pertaining to access for and handling of inspection equipment, and a description of a recommended inservice inspection and surveillance program. Finally, four areas requiring research and development are identified, and recommended programs are given. These are: diagnostic monitoring using noise analysis for detecting reactor internals malfunction; an acoustic viewing system for viewing and inspecting under sodium; an ultrasonic testing system for inspecting reactor systems components; and accelerometers for use with reactor systems components in high radiation and high temperature environments. A suggested schedule to produce results in time for the FTR and an approximate program support estimate is given for each program.

(b) *Southwest Research Institute Contract BDR-570.* Work has been initiated by SWRI on the conceptual design of the inservice inspection equipment. The effort includes the areas of:

- Structural Analysis
- Systems Engineering
- Flow and Vibration Effects

- Mechanism Design
- Transducer Evaluation and Specification
- Electrical Circuit and Signal Analysis
- Materials Serviceability Evaluation
- Health Physics.

The initial (Phase A) report on criteria has been completed.⁽¹⁾

The areas selected for inspection are at geometric, metallurgical, environmental, or loading discontinuities which include a sufficient sample of general overall structural behavior as well as those locations where operating condition service factors are more severe; hence to monitor failure initiation and propagation sites.

The experience acquired in the development of inspection programs for light water nuclear reactors provides a useful background for establishing inspection criteria for liquid-metal-cooled fast reactors; but care must be taken in applying this background to establish the differences in process factors. In the light water nuclear reactor pressure vessel, for example, shell stresses are in the elastic regime and stress analysis can be used to pinpoint the highest stressed locations and predict the most probable site and orientation of low cycle fatigue cracks reasonably accurately. This is not the case in the elevated temperature regime. Here stresses are redistributed through inelastic deformation, and the materials' properties change with temperature. The available analytical procedures are deficient, for example, in providing such a straightforward and simple basis for siting and orienting crack detection sensors as that of monitoring

1. R. D. Wylie, et al. "Criteria for Surveillance and Inspection of the Fast Flux Test Facility," 17-2552-01-A, May 1, 1969.

material at, and normal to, highest elastic tensile stresses induced by operating loads.

Another major difference between the light water nuclear reactor and the liquid metal nuclear reactor is in loadings, or the sources of significant stresses and strains. Since internal pressure is the most significant loading of the light water nuclear reactor, extensive experience and research effort has developed both intuitive and analytical methods of predicting its effects. By comparison, the most significant loadings of the liquid-metal-cooled nuclear reactors are thermal and there is far less experience background for the handling of this problem. Many factors must be kept in mind that are not usually significant. For example, a strain difference value is not meaningful without a corresponding temperature difference value; time is an important factor, and metallurgical changes in material may occur in service. These, and other factors, make the prediction of behavior from simple analysis and specimen test data far more difficult in the case of FFTF than in conventional nuclear reactors. More inspection and surveillance data must be acquired for the FFTF systems to ensure safety, than is considered essential for light water nuclear systems.

V. FFTF FUELS DEPARTMENT

E. A. Evans

A. FUEL ELEMENT DEVELOPMENT

C. A. Burgess

1. Process Demonstration and Development Section

R. E. Bardsley

a. Process Development

M. J. Barr

Binder-Lubricant Evaluation. The effect of various lubricants and/or binders on sintered pellet properties is being evaluated in process experiment MJB-17. The percent of theoretical densities of green and sintered pellets are presented in Table 5.A.I below. All test groups were from the same basic powder batch (nitrate PuO_2 and United Nuclear UO_2) and were sintered in the same furnace run. Test groups in Table 5.A.I were arranged with respect to the green density obtained with 20,000 psi press pressure.

TABLE 5.A.I. *Percent of TD of Green and Sintered Pellets*

Binder Lubricant (wt% addition)	% of Theoretical Density	
	<u>Green</u>	<u>Sintered</u>
1/2% PVA - 5% butyl-stearate	65.22	93.49
3% Carbowax	64.77	93.90
1/2% PVA - 3% butyl-stearate	64.22	93.70
5% Carbowax	64.06	93.56
1/2% PVA - 1% butyl-stearate	63.60	94.22
1% Carbowax	63.44	93.62
No binder	62.35	93.72
1% PVA	58.74	90.67
3% PVA	57.77	85.03
5% PVA	56.73	79.83

As indicated by the increase in green density, the lubricating effect of Carbowax and butyl-stearate is readily apparent. Also obvious is the effect of PVA (polyvinyl alcohol) on both green and sintered densities. A significant point is that less than one percent in sintered theoretical density separates the first seven test groups. Data on pressing behavior, diameter taper, and density spread is being evaluated currently to more thoroughly understand the attributes of each system investigated. Microstructure photographs are not yet available.

Impurity Addition Through Milling. Wet milling is significantly more efficient than dry milling with respect to both size reduction and intimacy of blending per unit time. Criticality and batch size considerations have all but eliminated this technique from use in large scale production. However, wet milling is still very useful for development purposes or small scale production batches. When using a wet mill, contamination becomes an important consideration. To determine the extent of impurity pickup from grinding media, test batches were milled with alumina (Al_2O_3) and tungsten carbide (WC) grinding media for varied times. Table 5.A.II below indicates the amount of impurity acquired with respect to milling time. Impurity levels for tungsten have been converted to W from the measured WO_3 .

TABLE 5.A.II. *Amounts of Impurity Required
in Relation to Milling Time*

Milling Time (hr)	Impurity Pickup	
	Al	W
0	20 ppm	--
1	50	Not Determined
2	50	0.24% (2400 ppm)
4	100	2.1%
8	200	3.6%
16	500	5.3%

The excessive tungsten pickup may be partially explained by the high grinding media to powder charge ratio used. Standard practice is to fill jars from 40 to 50% of their volume with grinding media and add enough powder to cover the media. This "rule-of-thumb" produced a media to powder charge weight ratio of from 5/1 to about 12/1 at the 16 hour level (a portion of powder was taken out at various times with the media quantity remaining the same). The media-to-powder-charge ratios in the Al_2O_3 study were modified to produce ratios of from 1/1 to a maximum of 5/1. Additional work with WC media has indicated that impurity pickup can be considered reduced when low media to powder charge weight ratios are used and the wet slurry is relatively thick (no viscosity values are available).

b. Specifications and Manuals

C. A. Strand

Analytical Chemistry Methods. The FFTF Analytical Chemistry Methods in Support of Driver Fuel Fabrication, BNWL-1024, (formerly identified as A-0089) was issued. Methods remaining to be completed for later incorporation into the manual include: (1) analysis of stainless steel alloying elements and impurities, (2) analysis of residual halogens, (3) alpha-autoradiography to measure fuel pellet homogeneity, and (4) fuel pellet ceramography.

Plutonium Dioxide and Uranium Dioxide Material Specifications. Material specifications for plutonium dioxide powder and uranium dioxide powder were revised and updated. Both FFTF Material Specification for Ceramic Grade Plutonium Dioxide and FFTF Material Specification for Ceramic Grade Uranium Dioxide have been reviewed internally and are ready to be submitted for formal FFTF Project review.

c. Processing and Equipment

C. A. Strand

(1) Fabrication of 300 Fuel Pins for Criticality Studies (C. A. Strand, J. E. Sammis, and H. T. Blair). Fabrication of a total of 300 prototypical FFTF type fuel pins was started in the FFTF fuel pilot line. As requested, these pins have been designed for both criticality testing and PCTR experiments. With the exception of isotopic content, fuel column length, fuel pin length, and configuration and internal components, these pins will be fabricated to meet the requirements of the FFTF End Item Detail Specification for the Driver Fuel Pin, A-0080-R, dated November 6, 1968. Detail requirements for these test fuel pins are shown in Drawing No. H-3-29448, "Mixed Oxide Fuel Pin for Criticality Safety Experiments Assembly and Details."

The fuel pins will be fabricated according to the rough draft of the FFTF Manufacturing Process Specifications for the Driver Fuel Pin, A-0097. Processing according to these specifications will serve two purposes: (1) establishment of a uniform fuel fabrication process, and (2) testing and evaluating the process specifications.

The following list indicates the current status:

- 2 kg of calcined PuO_2 is ready for cross-blending with additional calcined PuO_2
- 3.5 kg of calcined PuO_2 has been cross-blended and is presently being blended with UO_2
- 5 kg of mixed oxide is ready for binder addition and granulation
- 4 kg of mixed oxide has been granulated and is ready for pressing
- 2 kg of pellets are in the binder removal stage
- 1.4 kg of mixed oxide pellets have been sintered.

(2) Binder Removal (H. T. Blair). A temporary binder removal furnace was tested and installed in the sintering glove box in Line One of the Process Demonstration and Development Facility. Mixed oxide pellets being fabricated for the Physics Test Pins were successfully debonded in the furnace. The debonding cycle requires one working day. Present indications are that the furnace capacity is equivalent to a day's output from the press and equivalent to half the sintering furnace capacity. Thus, a very satisfactory continuous batch process can be maintained through these three operations.

A second binder removal furnace is being prepared for installation in the sintering glove box in Line Two.

(3) Analytical Standards Program (H. T. Blair). The results of those studies completed on samples from the sintered mixed oxide pellets produced for the second phase of the Analytical Standards Program are summarized in the following paragraphs. The average density of the pressed green pellets was 48.61% of theoretical density. Some pellet lamination was encountered during pressing when the feed powder was not completely dry.

A statistical review of the dimensions and geometrically determined densities of the sintered pellets produced the following results:

Average Shrinkage:	Diameter 20%
	Length 19%
Average Taper:	0.0012 in. with a standard deviation of 0.006 in.
Average Density:	91.34% of TD with a standard deviation of 0.71% TD

There was no exhibited tendency for the pellets to assume either an hourglass or bowed shape. Both shapes were equally present with the average hourglass or bow being 0.0005 inches.

The vertical location of the pellets in the sintering furnace only influenced the dimensions and densities obtained in the top tray. Pellet samples from the top tray averaged 0.0023 in. longer, 0.0005 in. larger in diameter, and 0.34% lower TD than the overall average dimensions and density for the sintering run. All of the analytical chemistry results with the exception of F and Cl content have been received and are included in Tables 5.A.III and 5.A.IV.

TABLE 5.A.III. *Impurity Levels in Analytical Standards Pellets, Phase II*

<u>Impurity</u>	<u>Analytical Standards Pellets ppm by Weight</u>	<u>Product Specifications (A-0080) ppm by Weight</u>
Ag	0.1	(a)
Al	10	500
C	45-60	150
Ca	100	250
Cl	Not Received	25
Cr	25	250
Cu	1	(b)
F	Not Received	25
Fe	20	500
Mg	5	25
Mn	1	(a)
Na	2-5	500
Ni	10	500
Pb	1	(a)
Sn	5	(a)
V	20	500
Zn	50	(b)

- a. The total amount of Cu, Zn, Si, and Ti was 800 ppm by weight
 b. The total amount of Ag, Mn, Mo, Pb, and Sn was 200 ppm by weight.

TABLE 5.A.IV. *Analytical Chemistry Results on Analytical Standards Pellets, Phase II*

<u>Analysis</u>	<u>Analytical Standards</u>	<u>Product Specifications (A-0080)</u>
wt% Pu	22.0 - 22.2	22.10 ^(a)
wt% U	66.0 - 66.8	66.05 ^(a)
Gas Content	0.02 cm ³ /g	0.05 cm ³ /g
Water Content	6 ppm by weight	30 ppm by weight

a. 75% ^NUO₂ - 25% PuO₂ as required for the Analytical Standards Program.

The oxygen-to-metal ratio of the as-sintered pellets was 1.99. This was the first large mass of mixed oxide pellets to be sintered in the new facility, and the relationship between O/M, load mass, and gas-flow had not been well established for the new furnace. Oxygen-to-metal ratios of 1.94 had been achieved with test batches of 100 to 200 grams. Since a ratio between 1.95 and 1.97 was desired, a series of firings using Ar-8% H₂ gas made in the demonstration facility sintering furnace to reduce the oxygen-to-metal ratio of 200 pellets. The results of the firings are given in Table 5.A.V.

TABLE 5.A.V. *Oxygen-to-Metal Ratio Reduction of Analytical Standards Pellets, Phase II*

<u>Load Mass (g)</u>	<u>Gas Flow (SCFH)</u>	<u>Temperature (°C)</u>	<u>Time at Temperature hr</u>	<u>Average O/M</u>
260	8	1200	3	1.989
254	8	1500	3	1.972
260	8	1650	3	1.962

Centerless grinding of 100 of the sintered pellets did not alter the average density determined by geometric methods. However, the spread of densities determined for the ground pellets

ranged from 89.70 to 92.39% TD while the spread obtained measuring as-sintered pellets was from 89.44 to 92.91% TD.

(4) Firing Shrinkage Versus Green Density Relationships (J. E. Sammis). During the fabrication of the mixed oxide pellets for the second phase of the analytical standards program, green density versus firing shrinkage information was generated which may be compared and contrasted to process development studies being performed as cited in BNWL-881. Figure 5.A.1 gives the important relationships in graphical form. The coding PL-2 on the graph represents the first 25 wt% PuO_2 mixed oxide pellets to be fabricated in the pilot line. Each point of this designation represents one pellet. The coding MJB-7 represents pellets produced in the other process development studies. Each point of MJB-7 represents the average of seven pellets. It is apparent from the graph that the slope of the percent shrinkage versus green density relationship is more negative for damp powder. The same material, after redrying, exhibited nearly identical results to those of MJB-7. The average difference between the shrinkages reported for PL-2 and MJB-7 for thoroughly dried powder was 0.75%.

2. Subassembly Development Section

J. W. Thornton

a. FFTF Fuel Fabrication Demonstration Facility

M. D. Jackson

A modified Hoskins furnace has been installed in Pellet Line No. 1 where it will be used to remove binder from mixed oxide pellets. The amount of mixed oxide pellets processed through this furnace as of May 14, 1969 was 3.3 kg. The pellets appear to be completely satisfactory. There is no sign of cracking, lamination or other defects.

FIGURE 5.A.1. Percent Shrinkage in Diameter Versus Green Density for 25 wt% $\text{PuO}_2\text{-UO}_2$

Fuel Subassembly Fabrication Demonstration Facility (Project BCP-034) (M. D. Jackson). The design work of this facility is 92% complete. Demolition work in Room 138 was 10% complete as of May 1. The total project is 2 to 3% complete. Demolition work is scheduled to be complete by July 1. Construction is scheduled to start on July 1, and is scheduled to be complete December 1, 1969.

b. Subassembly Development and Testing

D. E. Blahnik and R. L. Myers

Work was started on the CCTL MARK II subassembly. Plenum springs, plenum tubes, and caps and reflector rods are being fabricated in the technical shops. A total of 11,000 pellets had been pressed for the subassembly as of May 9, 1969.

Wire Wrapping of Fuel Pins Using 20% Cold-Worked Type 316 SST Wire. Results of wire wrap development thus far have shown that wrapping pins with 20% cold-worked, Type 316 SST, 0.056 diameter wire can be done. Further development and testing is needed to remedy some fabrication and assembly problems.

From an ease of fabrication standpoint, the end cap-wire wrap weld configuration most desired at the present time is one with either a 90 degree hole or welding directly along the side of the end cap.

3. Special Product Fabrication Section

E. T. Weber

a. Fuel Fabrication for Test Pins for Irradiation in EBR-II

W. E. Warden

Fuel pellet fabrication for EBR-II subassembly PNL-8 was completed. This fuel consisted of 93% enriched (^{235}U) in the standard EBR-II test composition 75% UO_2 -25% PuO_2 . To obtain

fuel pellets in the acceptable density range ($93 \pm 2\%$ TD) it was necessary to modify process operations to include preslugging and reslurrying of the feed material prior to pressing fuel pellets.

Fuel fabrication for EBR-II subassembly PNL-6 is in process. This fuel consists of 30% enriched (^{235}U) in the standard EBR-II test composition of 75% UO_2 -25% PuO_2 . Eight kilograms of PNL-6 mixed oxide were processed through the powder preparation operations.

b. Fabrication of Fuel and Special Test Pins for Irradiation Tests Under Task FP-3

W. E. Warden

Four special vipack fuel pins were fabricated to meet requirements for irradiation tests on density effects (capsules FP-38 and FP-39) to be conducted under Task FP-3. A fuel pin density of $83.0 \pm 5\%$ was obtained on three of the pins and a density of $80.0 \pm 5\%$ on one pin using crushed pellets of $94.0 \pm 0.5\%$ TD. This fuel is 75% UO_2 -25% PuO_2 .

Fuel for gap closure capsules FP-33 and FP-34 was fully characterized and pins are being loaded and welded. Fuel for six of these pins was recovered from pins with cracked cladding initially intended for calibration capsule FP-32. All the fuel pellets for capsule FP-33 and for two pins from capsule FP-34 were centerless ground from 0.320-in. diameter.

c. Assembly of Test Pins for Irradiation in EBR-II

R. M. Crawford

All of the fuel pellets for PNL-7 were characterized and made up into columns. The cladding tubes required for fabricating PNL-7 were air gaged and the bottom end-caps welded in.

Eighty percent of these welds were radiographed and found acceptable. Fuel pellets were loaded in 20 of the tubes and are ready for xenon tagging and final closure welding.

The BNW procedure for xenon tagging was formally submitted to and approved by Argonne National Laboratory. A representative from ANL will witness the xenon tagging of PNL-7 pins.

Each of the bottom end cap welds was monitored during welding by an acoustic emission technique currently being developed for nondestructive testing and monitoring fuel pin end closure welds. An effort is now being made to correlate the acoustic emission traces with metallography samples taken from selected welds.

B. FUELS QUALITY ASSURANCE

H. G. Powers

1. Development and Control of Inspection and Testing

a. Eddy Current Tubing Tester

J. Ryden, Jr., E. F. Perrizo, and A. C. Callen

Significant advancements were made in the development of a new eddy current technique for inspecting tubing for defects. This method utilizes two encircling coils as drivers, and multiple pancake coils as pickups. The new coil configuration consists of two encircling coils wound in 30-mil wide slots spaced 0.050 in. apart. Eighteen, 0.040 in. diameter pancake coils are positioned between these slots around the coil form circumference. This number of pancake coils provide 100% inspection coverage of the 0.250 in. OD tubing in one pass.

The prototype coil system has been used to inspect about 500 ft of tubing with excellent results. The technique for fabricating both the coils and the coil forms is being continually improved. A coil form and coils for use in inspecting 0.230 in. OD tubing has been fabricated.

b. High Temperature Water Analyzer (Urie)

W. L. Delvin

The analyzer was checked out using a hydrate of sodium tartrate. An average recovery of 96.3% with a standard deviation of 0.84 was obtained. The instrument shop is scheduled to check the instrument before installation in a glove box.

2. Quality Assurance Systems and Procedures

a. Calibration Program

W. F. Stevenson

The calibration program to ensure accuracy, and traceability to NBS of all measuring and test equipment and standards

used in BNW fabrication development of FFTF fuels, has been effected. All such equipment and standards certified initially by BNW standards Laboratory will be entered on a computer recall program to ensure periodic recall and recertification at pre-scribed intervals of time. The procedures document covering this activity is ready for management concurrence and signature.

b. In-Process Inspection Plans

W. F. Stevenson

Acceptance sampling plans and accompanying acceptance criteria were developed for all specified irradiation test fuel pellet characteristics. The Operating Characteristic curves for each of these plans were calculated and are being reviewed with design and fabrication personnel to obtain agreement on producer and consumer risks.

3. Offsite Fuel Manufacturing Program Support

a. Prequalification, Quality Assurance Representatives Training

D. A. Stranik

The training for these representatives was started. Plutonium safe handling and radiation protection orientations, discussion of handling techniques and fabrication methods and a general familiarizing with the product specifications and pin design was completed. General lab observation and glove box work is being performed. Continuation of the training in other areas such as NDT testing, analytical chemistry testing, discussions of process equipment and capabilities, and other training will continue.

b. Plans for Evaluation of Prequalification Suppliers

W. F. Stevenson

Plans were made, and included in the Statement of Work, for the acquisition of sufficient inspection data from each

participating supplier to conduct process capability studies. Such studies and evaluations will indicate industry's current capability and future potential to provide FTR fuels of the necessary quality.

C. CLADDING DEVELOPMENT

T. T. Claudson

1. Cladding Development and Processing

J. C. Tverberg

a. Procurement Activities

R. C. Aungst, R. J. Lobsinger, and R. R. Studer

A purchase requisition, product description and work statement were prepared for procurement of 25,000 ft of Type 316 SS tubing, to be used in the FFTF fuel pilot line. A product description and statement of work for this order were adapted from the documents drafted for use with the vendor development contracts. BNW will supply the material for this order in the form of 1.1875-in. diameter solid bar.

An order of Inconel X-750 plenum springs for prequalification fuel pins and subassemblies PNL-9, 10, and 11 was rejected because of extensive warping and relaxation suffered during heat treatment. Discussion with the vendor revealed no obvious reasons why this occurred. Negotiations are in progress with the next lowest bidder in an endeavor to procure these springs from an alternative source.

Fabrication of Inconel-600 reflectors for bare-pin subassemblies PNL-9, 10, 11, 14, 15 and 16 was completed. This was accomplished at BNW by swaging, annealing, and centerless grinding characterized 0.625-in. diameter bar.

The 0.250-in. diameter, double vacuum-melted Types 316 SS bar received from Crucible Steel was shipped to their Trent Tube Division for bright annealing to satisfy requirements of the specification.

Extensive evaluation of the 20% cold-worked and the annealed Type 316 SS tubing received from Carpenter Technology

Corporation is in progress. Preliminary data from ultrasonic and eddy-current tests show metal particles impressed in both exterior and interior surfaces. These particles are generally stainless steel; however, iron particles have also been found.

Samples of Superior tubing given the warm 10% HNO₃ treatment for brown spot removal show no intergranular attack upon visual examination at 500X.

b. Vendor Development

J. C. Tverberg

The status of the vendor development contracts is as follows:

- Sandvik Steel, Inc. Signed and in place.
- Carpenter Technology Signed and in place.
- Wolverine Tube Approved by Wolverine, returned by RL for minor contract wording changes, resubmitted to Wolverine, returned to RL and awaiting final approval.
- Universal-Cyclops Submitted to both contractor and RL for approval.
- Superior Tube Co. Currently in BNW signoff before submission to RL and contractor for approval.

Work is underway at Sandvik Steel. Carpenter Technology indicates they have begun preparation of the vacuum induction melt of steel in anticipation of contract approval.

c. Vendor Qualification

J. C. Tverberg

Vendor qualification efforts will commence in the second quarter of FY 1970. However, it is anticipated that several vendors will be qualified through the vendor development program and through special procurements being made for large quantities of irradiation quality tubing. Plans are underway to establish a procedure whereby this qualification may be accomplished.

d. Parameter Studies

R. C. Aungst and J. C. Tverberg

Considerable progress was made in the preparation of ingots for the Type 316 SS cladding chemistry variables program. General Electric Company - Nuclear System Programs has begun preparation of the 400 lb master heats from which the 14 experimental heats will be made.

2. Cladding Evaluation

J. J. Holmes

a. Acceptance Testing and Characterization

M. M. Paxton

The objective of this program is to ensure that acceptable cladding material is obtained and utilized throughout the FFTF fuel development and fuel pin irradiation testing program. Under this program, lot-to-lot variations in selected mechanical properties of AISI Type 304 and 316 SS tubing are being determined and evaluated.

Burst test results, reported previously⁽¹⁾ depend on the rate of loading. In order to establish this loading rate dependency, a series of short term (0.01 to 1.0 hr) stress rupture tests were performed on lots G and H of Type 316 SS tubing. When the results of short term stress rupture tests were extrapolated back to 0.001 hr and burst test data were treated as a very short stress rupture test, reasonable agreement between the stresses to cause failure was observed except for the 1800 °F tests.

1. FFTF Monthly Informal Technical Progress Report, BNWL-1062. Battelle-Northwest, Richland, Washington, April 1969.

Four biaxial creep tests, on lots E and F of Type 304 SS specimens were completed in an effort to resolve their variation in rupture strengths. Observed values for minimum creep rate do not completely resolve the difference in rupture strength.

b. Biaxial Stress-Rupture Studies

R. W. Barker

The objective of this program is to determine the combined effects of environment and fast reactor irradiation on the biaxial stress-to-rupture properties of candidate fast reactor cladding alloys. The current program is specifically directed at providing a description of material behavior during fast reactor service.

Biaxial stress-rupture tests were initiated during April on AISI Type 304 SS irradiated to $\sim 0.8 \times 10^{22}$ n/cm² (E > 0.18 MeV) in EBR-II. Four tests were completed at 1200 °F in argon and data reported except for the rupture strain of one test of 0.06 hr duration. The rupture life of the specimen in question did not appear to be characteristic of other tests at similar stress and temperature. The rupture strain was measured. Its value, 2.4%, likewise does not appear to be characteristic for tests at this stress and temperature. The conclusion reported previously that the data from this test be accorded very low weight appears to be substantiated. A completed summary of all data from these tests is given in Table 5.C.I. Additional tests to extend the data for this material to longer rupture times are planned.

Biaxial stress test specimens of AISI Type 316 SS irradiated in EBR-II to $\sim 1 \times 10^{22}$ n/cm² (E > 0.1 MeV) in an environment of pure sodium have been sectioned, cleaned, and new fittings attached by welding with the technique previously established for Type 304 SS cladding. Biaxial stress-rupture tests on this (Type 316 SS) material have been initiated.

TABLE 5.C.I. *Biaxial Stress-Rupture Tests on AISI Type 304 SS Irradiated to $\sim 0.8 \times 10^{22}$ n/cm² (E > 0.1 MeV) in EBR-II. Test temperature was 1200 °F. The irradiation temperature was 1000 °F $\pm$ 100.*

Unirradiated Data ^(a)			Irradiated Data			
Hoop Stress, psi	Rupture Time, hr	Rupture Strain, %	Irradiated Test No.	Hoop Stress, psi	Rupture Time, hr	Rupture Strain, %
34,000	0.1	16.5	11	34,000	0.06	2.4
29,000	1.0	14.5	7	29,000	1.55	14.8
19,000	100	11.8	5	19,000	30.4	0.96
15,800	1000	10.5	4	15,800	110	2.66

(a) *Stress and strain values were taken from unirradiated test curves at rupture times shown.*

c. In-Reactor Measurements

N. E. Harding and E. R. Gilbert

EBR-II Creep Capsule Development. Testing of a Type 347 SS bellows at room temperature and 900 °F through a pressure range of 0 to 300 psi was completed. This type of bellows will be used to apply the load to the uniaxial specimen in the EBR-II creep capsule. The bellows was set up in an Instron tensile frame, and a load versus bellows pressure curve run at room temperature and 900 °F. There was very little difference between the two curves, but enough to warrant elevated temperature calibration of each bellows before use in an in-reactor capsule.

A mockup of the EBR-II creep capsule is being assembled to check out the capsule design and to point out assembly problems anticipated because of the small capsule diameter (1.7 in.) and the complexity of the design. Assembly work to date has confirmed some of the anticipated problems. Techniques for assembly will be developed as work progresses.

Procurement of capsule components and materials is in progress. Long delivery dates on some off-site purchases may make it difficult to meet the present schedule for capsule completion.

An approval-in-principle letter for the EBR-II creep capsule experiment was submitted to DRDT.

d. Damage Analysis

J. L. Straalsund and H. R. Brager

The objectives of this effort are to establish the radiation-induced swelling characteristics of FFTF alloys, to relate fast reactor induced substructural changes in microstructure to corresponding changes in mechanical properties, and to interpolate the effects of reactor conditions on properties using both microscopic modeling and empirical approaches.

Equations describing the effect of fluence and temperature on swelling in solution treated Type 304 and 316 SS and in 20% cold-worked Type 316 SS were refined and the results agreed upon by Westinghouse and BNW. The agreed upon equations have the general form:

$$\frac{\Delta V}{V} = A(\phi t)^n [e^{-Q_1/RT} - \alpha e^{-Q_2/RT}]$$

Specifically the equations as delineated are as follows:

Solution Treated Type 304 and 316 SS

$$\frac{\Delta V}{V} = 5.0 \times 10^{-36} (\phi t)^{1.66} [e^{-6800/RT} - (1.87 \times 10^4) e^{-27,000/RT}] \quad (1)$$

20% Cold-Worked Type 316 SS

$$\frac{\Delta V}{V} = 9.4 \times 10^{-33} (\phi t)^{1.48} [e^{-5600/RT} - (1.78 \times 10^5) e^{-30,000/RT}] \quad (2)$$

where $\frac{\Delta V}{V}$ = percent volume change

ϕt = neutron fluence, $E > 0.1$ MeV

Q = activation energy in cal/mole - °K.

Equation (1) was developed by using regression analysis to fit an equation of the form:

$$\frac{\Delta V}{V} = A(\phi t)^n e^{-Q/RT} \quad (3)$$

Thirty-seven points were selected from available bulk density measurements from Type 304 SS components and specimens taken from the EBR-II reactor to make the analysis. In the selection of data, only points corresponding to locations within 10 in. of the midplane were used. The analysis resulted in the following equation.

$$\frac{\Delta V}{V} = 5.0 \times 10^{-36} (\phi t)^{1.66} e^{-6800/RT}$$

The second portion of the temperature dependence term in Equation (1) ($\propto e^{-Q_2/RT}$) was empirically fit to force the swelling to a maximum at 625 °C and to zero at 750 °C. The peak temperature of 625 °C was chosen primarily because there are indications of such a peak near 625 °C in the U.K. Type 316 SS pin data.

An equation was developed for the solution treated Type 316 SS by assuming the same fluence dependence as obtained for Type 304 SS and by using the U.K. fuel pin diameter changes to calculate the temperature dependence and the pre-exponential factor. It was assumed in this analysis that the pin diameter increase was induced by cladding swelling. For temperatures above 450 °C, the results of this equation agreed with Equation (1) within 10% of the predicted swelling values over all

ranges of fluence. It was, therefore, decided that Equation (1) could be used for both Type 304 SS and Type 316 SS in the solution treated condition.

Equation (2) for cold-worked Type 316 SS was agreed upon up to a total volume change of 1% and at temperatures less than 500 °C. In the development of Equation (2), U.K. data on fuel pin diameter increases and void volume based on electron microscopy were used. In a preliminary analysis, all of the data available were fit to an equation of the form given by Equation (3). However, inspection of the results indicated that at high temperatures and fluences (density changes in excess of about 1%) the swelling behavior departed from that predicted from lower temperature (lower swelling). A change in mechanism apparently occurred at the higher temperatures, resulting in much greater increase in pin diameters. It should be noted that all of the data points indicating this increase were associated with the upper portions of DFR Mark II subassembly interior pins.

The data at high temperatures (alternatively at volume changes in excess of 1%) are too fragmentary to allow meaningful analysis. Therefore, Equation (2) was developed only from data at temperatures below 530 °C.

If the increase is real, it may result from one of the following effects:

- (1) A large portion of the pin diameter increase at high temperatures may be due to fuel swelling.

This appears doubtful as the microscopy measurements of void volume appear to agree with the $\Delta D/D$ measurements. There is some uncertainty, however, as to whether the microscopy data are representative of the pins in question. Thus, fuel swelling in this region must be considered as a possibility, in which case Equation (2) represents the best estimate for swelling in cold-worked Type 316 SS at the high, as well as low ranges of temperature and fluence.

- (2) The actual temperature and flux conditions may be different than the data indicate. The large departures from low temperature swelling behavior must presume significant errors in the data.
- (3) Recovery of cold work may occur in the pins during long exposure at high temperatures (above 500 to 550 °C). If this is the case the results may be asymptotic to the curve described by Equation (1). Both pin diameter data and microscopy data for temperatures above about 530 °C and fluences greater than $3.5 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$) indicate swelling close to that given by Equation (1).

In view of the few data available for developing equations for swelling in cold-worked Type 316 SS, it appears that for the present our best design guidance is to use Equation (2) for situations in which the temperature is 500 °C or less and predicted swelling is less than 1%, and assume that the swelling will be between the limits of Equation (1) and Equation (2) at higher fluences and temperatures.

Transmission microscopy is continuing on irradiated Type 316 SS. In the previous monthly report,⁽¹⁾ it was stated that solution treated Type 316 SS, irradiated to $0.8 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$) at 900°F contained small polyhedral and rod shaped M_{23}C_6 precipitates. It was also reported that the carbide morphology was similar to that reported in the literature for unirradiated stainless steels annealed at higher temperatures. In order to determine if a similar carbide distribution could be created in our control material, an unirradiated Type 316 SS control sample, annealed 3050 hr at 1400°F was examined. This specimen contained only a small concentration ($\sim 10^{10}$ to $10^{11}/\text{cm}^3$) of polyhedral precipitates within the matrix. There were essentially no precipitates within the grain boundaries. The low amount of carbides, as compared to the irradiated specimen and results reported in the literature for nonirradiated specimens after similar anneals, indicates that the unirradiated thermal control sample may have been decarbonized during the annealing process. Chemical analyses are presently being made for the carbon content in irradiated and unirradiated specimens with various thermal treatments to determine if this is the case.

An examination of the EBR-II safety rod thimble (Type 304 SS) indicates that the carbide morphology varies along the length of the thimble. Furthermore, the peak fluence region exhibited a higher concentration of precipitates, thought to be

1. *FFTF Monthly Informal Technical Progress Report, BNWL-1062. Battelle-Northwest, Richland, Washington, April 1969.*

$M_{23}C_6$, than was observed in the same general area in the EBR-II control rod thimble. The control rod thimble is made from the same heat of material as is the safety rod thimble. Thus, the microscopy results indicate carbon transfer during service. Sections of the two thimbles taken from the inlet, outlet and peak flux regions are presently being chemically analyzed for carbon.

Seven samples of irradiated Types 316 and 304 SS have been sent to Atomics International for quantitative helium analysis.

e. EBR-II Irradiations

P. K. McDaniel

The EBR-II irradiations program is to provide coordination and direction for the timely accomplishment of fast reactor irradiations on structural materials. Structural materials irradiation experiments are in various stages of development; some in design, some under construction and others under irradiation in the EBR-II. The status of these experiments is described.

A number of subassemblies, X-009, X-101, X-014, X-018, X-021 (including X-021A), X-022, and X-030 containing BNW structural material experiments have been discharged from EBR-II. The suffix "A" designates pin interchange. A summary of these irradiation experiments is given in Table 5.C.II. All flux monitors from these experiments, with the exception of X-010 and X-018, have been counted. Mechanical properties tests for specimens irradiated in X-009 and X-014 have been completed. Uniaxial and biaxial tests on specimens irradiated in X-022 are in progress.

Experiments presently in EBR-II include subassemblies X-019, X-020, X-027, X-067 and full subassemblies X-021, (now X-021B) X-041, and X-057. The details of these subassemblies are given in Table 5.C.III.

TABLE 5.C.II. Summary of Experiments Discharged from EBR-II

Subassembly Designation	Reactor Position	Date Charged	Date Discharged	Type of Specimen	Material	Accumulative ^(d) Exposure n/cm ²
X-009 ^(a)	4A2	3-24-66	11-15-66	18 Tensile 10 Uniaxial Creep	304 SS	2.0×10^{22}
X-010 ^(a)	7F3	3-24-66	4-24-69	18 Tensile 10 Uniaxial Creep	304 SS	6.6×10^{22}
X-014 ^(a)	2D1	7-17-66	12-31-66	50 Tensile	304 SS	1.7×10^{22}
X-018	2B1	12-6-66	2-25-69	198 Tensile or Uniaxial Creep	304 SS, 316 SS 348 SS, 321 SS	4.4×10^{22}
X-021 ^(b)	2D1	2-27-67	9-16-68	127 Tensile or Uniaxial Creep	304 SS, 316 SS, ^(c) Ni Base Alloys	3.2×10^{22}
X-021A ^(b)	2D1	9-27-68	11-16-68	86 Density 16 Biaxial Creep 14 Tensile 10 Uniaxial Creep	316 SS ^(c) Ni Base Alloys	5.0×10^{21}
X-022	7C4	2-27-67	6-15-68	595 Tensile or Uniaxial Creep 65 Biaxial Creep	304 SS, 316 SS, 348 SS, 321 SS	1.1×10^{22}
X-030 ^(a)	6E1	10-10-67	3-6-68	5 Tensile 3 Uniaxial Creep	304 SS	5.2×10^{21}

(a) Mark A pins irradiated on space available basis, all other pins are B-7 type.

(b) Single pin interchange.

(c) Variety of other alloys also included.

(d) $E > 0.1$ MeV.

TABLE 5.C.III. Summary of Experiments in EBR-II

Subassembly Designation	Reactor Position	Date Charged	Estimated Discharged Date	Type of Specimens	Material Contained	Accumulative ^(b) Exposure n/cm ²	Goal Exposure n/cm ²
X-019	6D2	1-13-67	July 1969	64 Tensile 24 Uniaxial Creep	304 SS 316 SS	2.5×10^{22}	1.9×10^{22}
X-020	6B5	1-13-67	July 1969	28 Tensile 8 Uniaxial Creep	304 SS 316 SS	2.7×10^{22}	1.9×10^{22}
X-021 ^(a)	2D1	2-27-67	Aug. 1970	501 Tensile or Uniaxial Creep 65 Biaxial Creep	304 SS, 316 SS, 348 SS, 321 SS Ni Base Alloys ^(c)	4.1×10^{22}	8.8×10^{22}
X-027	4B3	11-21-67	Aug. 1970	5 Tensile 3 Uniaxial Creep	304 SS	2.7×10^{22}	5×10^{22}
X-041	7C4	7-24-68	May 1970	713 Tensile or Uniaxial Creep 74 Biaxial Creep	304 SS, 316 SS, 321 SS, 348 SS	1.0×10^{22}	3×10^{22}
X-057	2B1	2-24-69	March 1971	577 Tensile or Uniaxial Creep 164 Biaxial Creep 58 Density	304 SS, 316 SS, ^(c) 321 SS, 348 SS, Ni Base Alloys	4.8×10^{21}	6.0×10^{22}
X-067	4FZ	1-5-69	Sept. 1969	204 Tensile or Uniaxial Creep 20 Fatigue	304 SS, 316 SS, Weldment	None	1×10^{22}

a. Original space available with two single pin interchanges

b. Accumulative to end of run 33, ($E > 0.1$ MeV)

c. Variety of other alloys.

Subassembly X-067 consists of four pins from Battelle-Northwest, two from Argonne National Laboratory, one from General Electric.

The four Battelle-Northwest pins contain specimens for a study of the effect of fast reactor irradiation on Type 304 and 316 SS weldments in support of the FFTF pressure vessel and fuel ducts. The goal exposure is 1×10^{22} n/cm² (E > 0.1 MeV).

This report section will provide for the periodic updating of the status of irradiation experiments in the EBR-II.

3. LMFBR Fuel and Cladding Information Center

F. R. Shober

An information center has been established to document and to record all qualitative and quantitative information relating to the irradiation performance of LMFBR fuel pins. Those areas of materials information included in the system are melting processes, fabrication techniques, heat treatments, mechanical properties, physical properties, quality inspection results, and performance records of individual lots of material. The system is a computerized information storage, retrieval, and data analysis system consisting of six sections; cladding fabrication; fuel fabrication; fuel pin fabrication; fuel subassembly fabrication; irradiating history; and postirradiation examination. The sectional nature of the system permits analysis and evaluation of data from each section independent of others or, if preferred a multisection analysis.

It appears that the major problem confronting the staff is to identify on and offsite sources of data and to interface with these sources.

D. FUELS EVALUATION

J. E. Hanson

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

G. E. Culley and E. O. Ballard

a. Irradiations in GETR

Capsules PNL-59-7 and 8 completed the twelfth GETR cycle of irradiation which was calculated to achieve an average burnup of 50,000 MWd/tonneM. The capsules were removed from the reactor on April 14, and neutron radiographed on April 30.

Capsules PNL-59-9 and 10 continue their irradiation with two more cycles required to reach goal exposure, 50,000 MWd/tonneM. Capsules PNL-59-11 and 12 were inserted in the RAFT's vacated by capsules 7 and 8 for an additional two cycles of irradiation during FY-1969. The performance of these two capsules is being monitored closely with alternative control thermocouples because the normal control thermocouple in each capsule is not functioning.

b. Postirradiation Examinations

The postirradiation examination of capsule PNL-59-5 was continued. Sectioning of the fuel pin was completed and three longitudinal and two transverse samples mounted and polished. Macrographs of the as-polished sections as well as beta-gamma and alpha autoradiographs were obtained. The sections are now being etched and photomicrographs taken.

Capsule PNL-59-7 is now available for destructive examination; however, the T-2 shipping cask to transport the capsule to BNW will not be available until mid June.

c. Postirradiation Examination Analysis

Results of the nearly completed nondestructive and destructive examinations of PNL-59-5 are being used to develop

performance analyses for mixed-oxide fuel pins irradiated in GETR to a further degree. Calculations to determine the radial and axial fuel temperature profile were made based on assumed operating characteristics of PNL-59-5. A comparison of the calculation results with the radial and axial temperature profiles determined from microstructure features of fuel will be made to provide an indication of which parameters (e.g., gap conductance, sintering temperature, fission gas release, and radial fission profile) need to be altered for better agreement between analysis and observations.

The axial power profile deduced from the gamma scan compares well with the profile predicted from the integrated cycle average rod bank insertion of 25.8 in. during the irradiation of PNL-59-5.

d. Transient Irradiations

The TREAT calibration test proposal was approved for duplication. The problems of shipping this experiment between BNW and ANL-Idaho appear to be resolved.

Data packages for the zero burnup GETR/TREAT transient tests, the irradiated GETR/TREAT, and the EBR-II/TREAT transient tests are being prepared.

2. Thermal Flux Irradiation of Hypostoichiometric UO_2 - PuO_2 R. D. Leggett and L. A. Pember

The status of the BNW-1 series capsules is summarized in Table 5.D.I. Capsules BNW-1-10 and 12 were recharged into the ETR after interim neutron radiography at 55,000 MWd/tonneM burnup showed no abnormal features. Their goal exposure is 100,000 MWd/tonneM.

Ceramographic examination of sections of pins BNW-1-9 and 11 was completed in the LASL hot cells. The negatives were sent to BNW for printing, laying-up of mosaics and detailed review. Microprobe examination is planned on certain of these specimens when the LASL shielded electron beam microprobe becomes operational.

TABLE 5.D.I. Summary Status of Irradiation Experiments
Conducted Under Task FP-6

Test Designation	Description ^(a)		Facility	Status			Disposition
	Fuel Form	O/M		Goal Exposure MWd/tonneM	Actual Exposure MWd/tonneM	Discharge Date	
BNW 1-1	Solid	1.93	MTR	100,000	(Est)120,000	2-69	Stored at the reactor.
BNW 1-2	Solid	1.93	ETR	50,000	(Est) 70,000	2-69	Stored at the reactor.
BNW 1-3	Solid	1.96	MTR	100,000	(Est)150,000	2-69	Stored at the reactor.
BNW 1-4	Solid	1.96	MTR	50,000	71,000	6-68	Exam. completed at BNW.
BNW 1-5	Solid	2.00	MTR	100,000	(Est)160,000	2-69	Stored at the reactor.
BNW 1-6	Solid	2.00	MTR	50,000	85,200	10-68	Exam. completed at BNW.
BNW 1-7	Annular	1.93	MTR	50,000	<1,000		Lead tube failed. Exam. at BNW.
BNW 1-8	Annular	1.93	MTR	100,000	29,000	1-68	Failed. Exam. at BNW.
BNW 1-9	Annular	1.96	MTR	50,000	81,800	6-68	Exam. completed at LASL.
BNW 1-10	Annular	1.96	ETR	100,000	(Est) 55,000	(Est)2-70	Discharged 2-69 for interim exam. Recharged 5/69.
BNW 1-11	Annular	2.00	MTR	50,000	87,400	8-68	Exam. completed at LASL.
BNW 1-12	Annular	2.00	ETR	100,000	(Est) 55,000	(Est)2-70	Discharged 2-69 for interim exam. Recharged 5/69.

(a) Fuel
 25% PuO₂ - 75% UO₂
 Pellets 0.212 in. OD (0.052 in. ID for annular pellets)
 Smeared density - 87.5% TD
 Pellet density - solid pellet 93% TD, annular pellet 95% TD
 Fuel column length 13.5 inch

Cladding
 OD - 0.250 in.
 Wall thickness - 0.016 in.
 304 SS

A specimen from BNW-1-7 was examined with the BNW shielded electron microprobe for redistribution of Pu or U or fission product segregation. Since this is a low burnup specimen (<1000 MWd/tonneM), it will serve as a base case for the higher burnup pins BNW-1-8, 4, and 6. No gross redistribution was found in BNW-1-7, but homogenizing on a micro scale was observed in the sintered regions of the fuel. Residual concentrations of UO₂ and PuO₂ were encountered in nonrestructured areas, as expected.

3. Electron Microprobe Examination of Irradiated Fuels -

E. D. Jenson

Shielded electron microprobe examination of irradiated fuel specimens was directed toward determining the effect of a radiation field on probe operation, changes in the Pu/U ratio along the fuel pin radius, and the occurrence of fission products and iron in the oxide matrix. Characterization and history of two samples examined are given in Table 5.D.II.

TABLE 5.D.II. *Characterization and History of Microprobe Samples*

	BNW 1-7	PNL 1-3
Composition	$(U_{0.75}Pu_{0.25})_{0.93}$	$(U_{0.75}Pu_{0.25})_{0.97}$
Exposure	300 MWd/t	8140 MWd/t
Power Rating	14.6 kW/ft	8.8 kW/ft
Reactor	MTR	EBR-II
Surface Temperature	~850-900 °C	~850-900 °C
Center Temperature	~2100 °C	~2100 °C
Discharge Date	July 17, 1967	June 15, 1968
Cooling Time	21 months	10 months
Radiation Level @ 1 ft (now)	1.8 R/hr	~4 R/hr

The Pu/U ratio for four radial scans on sample BNW 1-7 showed much scatter. No distinct trend was obvious in two radial scans. Two other scans showed a slight increase with increasing radius. Sample PNL 1-3 gave one radial scan with a nearly constant Pu/U and one with a general increase toward the central void.

Ingots of fission products up to seven microns diameter were found near the central void of PNL 1-3. Molybdenum, ruthenium, and technetium were identified. Palladium and rhodium were not observed, possibly because of the high background. Zirconium was indicated in the matrix but not in the ingots. Iron-nickel particles up to 30 microns diameter were observed throughout the fuel pin. Low concentrations of cesium were found in two areas in the fuel just inside the cladding. These areas were about 10 microns diameter.

Sample PNL 1-6 was loaded into the microprobe for examination.

Sample BNW 1-21, which had previously been reported to show a region of increased PuO_2 (to about 30%) near the outer edge of the large grain region, was further examined to see if this region extends around the fuel pin. Three additional radial scans were made, two perpendicular to the original scan, and one directly opposite from the original scan. None of these scans showed an increase in PuO_2 concentration other than the residual concentration regions such as were seen in the first scan. Thus, the increase in PuO_2 originally reported was possibly induced by nonhomogeneity caused by a PuO_2 -rich particle.

4. Fuel Pin Irradiations in EBR-II

L. A. Pember and E. O. Ballard

Five PNL-X pins are undergoing postirradiation examination in the LASL hot cells. They were irradiated in sodium filled capsules in EBR-II: three to estimated exposures of 5000 MWd/tonneM at 6 kW/ft and two to 8000 MWd/tonneM at 10 kW/ft. All pins were successfully recovered from their respective capsules, cleaned in Dowanol and water, and photographed at three axial orientations with and without the wire wrap. Each capsule was gross gamma scanned, radiographed with the betatron and punctured for gas sampling. As expected, all pins were intact and appeared to be in excellent condition. The pins are presently being measured with a profilometer after which they will be punctured to collect the internal gas for analyses.

Examination of sections from PNL-1-15, 17 and 19 was completed at LASL. The negatives were sent to BNW for printing, laying-up of mosaics and detailed review.

Subassembly PNL-4 (X059) was charged into the EBR-II at the beginning of run 34. BNW experimental subassemblies in EBR-II had attained the following burnup at the end of EBR-II run 33:

<u>Subassembly</u>	<u>Burnup, MWd/tonneM</u>
PNL-2 (X032)	30,000
PNL-3 (X051)	3,130
PNL-4 (X059)	0
PNL-5 (X054)	2,736

The final draft of the Engineering Test Plan for EBR-II subassemblies PNL-6, 7 and 8 was completed.

Preparation of the Experiment Description and Hazards Analysis for subassemblies PNL-9, 10 and 11 is progressing. All of the hazard analyses were completed with the exception of: (1), loss of coolant calculations, and (2), fission product activity as a function of cooling time.

With the EBR-II operating at 50 MWt, PNL-9, 10 and 11 will be in reactor grid positions 4N1, 4N1, and 3N1, respectively. Respective ^{235}U enrichments are 25 wt%, 65 wt% and 93 wt% of total uranium. However, the EBR-II is expected to go to full power (62.5 MWt) sometime after August, 1969. At that power level, PNL-9, 10 and 11 will need to be in positions 3N1, 4N1 and 3N1 with enrichments of approximately 6, 43, and 65 wt% ^{235}U in total U, respectively. These positions and enrichments are based on a fuel smeared density of 88% TD and peak linear heating rates of 6.5, 10.2 and 13.9 kW/ft. Subassemblies PNL-9, 10 and 11 are being designed on the basis of a reactor power of 62.5 MWt, although they may operate at 80% of normal power until the EBR-II attains 62.5 MWt.

5. EBR-II Instrumented Fueled Subassembly

M. K. Millhollen, S. H. Christensen

An approval-in-principle request letter was submitted to AEC for the fueled EBR-II instrumented subassembly test (PNL-17). The test subassembly will comprise 36 fueled pins and one unfueled pin which will serve as an instrument lead conduit. Pins will be irradiated at a maximum heat generation rate of approximately 12 kW/ft, but the UO_2 enrichment will be lowered from the previously published figure to compensate for the impending increase in EBR-II power level.

An analytical comparison of the thermal hydraulic characteristics of 19- and 37-pin subassemblies with a prototypical FTR 61-pin subassembly was completed. The purpose of the study was to determine which subassembly more closely approximates the coolant flow characteristics of the prototypical FTR subassembly.

A principal purpose for the proposed instrumented subassembly test is the determination of the magnitude of the forced mixing of the coolant. Forced mixing is caused by the wire wrap and an exact estimate is not available.

The flow characteristics were determined using the COBRA computer code. Results from this were used in the HAZAN computer code to obtain cladding temperatures.

The coolant inlet temperature was assumed to be uniformly 700 °F across the bottom of the subassemblies. The inlet flow rates were divided among the channels to give equal pressure gradients over the length of the bundles.

A comparison of the velocity and flow distributions is given in Table 5.D.III. It is seen that the 37-pin bundle

TABLE 5.D.III. Velocity and Flow Distribution

<u>Type</u>	<u>Velocity, % of Average</u>		
	<u>Central Channel</u>	<u>Peripheral Channel</u>	<u>Corner Channel</u>
61-pin	90.6	116.0	106.4
37-pin	97.6	115.8	105.6
19-pin	89.2	112.1	86.3

<u>Type</u>	<u>Flow, % of Average</u>		
	<u>Central Element</u>	<u>Peripheral Element</u>	<u>Corner Element</u>
61-pin	69.4	140.0	166.3
37-pin	70.6	124.0	144.0
19-pin	63.1	116.7	127.0

better approximates these characteristics of the 61-pin sub-assembly than does the 19-pin subassembly. This result is also true when considering turbulent interchannel mixing (forced mixing).

The calculated flow characteristics are largely dependent upon the geometry of the subassemblies and not upon their power generation. It was concluded that the 37-pin bundle will yield the most representative data.

6. Mechanical Testing of Fuel Pin Cladding

R. L. Fish

Development of the end-sealing technique for irradiated fueled pin specimen segments is continuing. Possibilities being explored include both direct fuel support of the Swagelok end fittings and the use of mandrels for support of these fittings. If these techniques do not produce an effective seal, remote welding of an end plug will be investigated.

Results from the first two burst tests indicate a strain ($\Delta d/d$) of 19.2% for the unirradiated specimen and a strain of 18.4% for the plenum region specimen.

7. FERMI Reactor Irradiation Test Subassembly

M. K. Millhollen and T. B. Burley

In anticipation of the return of the FERMI reactor to operation within the next year, conceptual design of a 37-pin cluster was completed and thermal-hydraulic calculations started.

8. Status of Remote Fuel Pin Gamma Scanning and Profilometer Equipment

J. W. Weber

PNL-1 fuel pin 1-10 was transferred to the 324 Building East Cell for use in aligning the mechanical and electronic equipment of the gamma scanner. The collimator slit alignment was completed and a series of scans run over the length of the fuel. The scans produced a gamma ray spectrum similar to that obtained in the initial scanning conducted in late 1968 with equipment in the PRTR basin.

Data on the fission product gamma ray spectrum obtained during tests run on the PNL 1-12 fuel pin in the PRTR basin are being analyzed by the Reactor Experiments Section to provide quantitative information on the distribution of several fission product isotopes. Three additional single channel analyzers, numerical type readout, and a Tally paper punch are being added to increase the capabilities of the gamma scanning equipment in the 324 Building.

Fabrication of parts for modifying the 324 East Cell profilometer was completed. The new transducers are being attached to the measuring head in an out-of-cell mock up to test the electronic system. Design of the electrical controls for the fuel pin translator was completed.

9. Fuel Performance Analysis

R. D. Leggett and R. L. Fish

The SINTER program (a fuel restructuring code) was successfully placed on the hybrid computer and several preliminary

structural analyses made. The program can provide both axial and radial fuel temperature profiles, central void size and restructured fuel diameter as a function of certain input parameters. The initial cases considered were the expected structures in typical FTR, 0.230 in. diameter, fuel pins operating at 6 kW/ft (low), 10 kW/ft (average), 14.5 kW/ft (peak) and 18.2 kW/ft (25% overpower). The output from the hybrid computer is displayed on a color image CRT to produce cross-section simulations that are color coded to correspond to the expected structure at a given fuel radius. A cross-section was generated and photographed in color at each of some 1600 axial segments per power level. These were then put together with appropriate titles into a short continuous visual display using animated filming techniques. This provided a simulation for the change in fuel structure from one end of a pin to the other at the four power levels considered.

A concerted effort was continued to assemble available data from irradiation experiments with stainless steel cladding oxide fuel pins in order to critically analyze the fuel pin performance. Our best estimates of the physical and mechanical properties of the fuel pin system, the reactor environment, and the pre- and postirradiation measurements are being combined with the best mathematical models available to develop a rationale of the performance of fuel pins. In addition to the data available from BNW experiments, data is also being collected from ANL, GE/BRDO, NUMEC, and DFR programs. We are emphasizing the detailed analysis of existing data in order to calibrate and refine the various existing mathematical models using the expertise available in the Materials, Stress-analysis, Physics, and Reactor Engineering disciplines.

E. OFFSITE FUEL PROGRAMS

G. A. Last

1. LMFBR Fuel Development

B. R. Hayward and F. M. Smith

United Nuclear has fabricated three batches of mixed oxide pellets in the process development program having a goal to eliminate the need for pellet grinding. Between 90 and 100% of these pellets met the present FFTF reference pellet dimensional specifications in the as-sintered condition. The main variables influencing pellet properties were found to be the characteristics of the original powder and press. Other sites performing related work have observed similar results. New work was start at NUMEC and UNC on characterization of column density (weight per unit length) of both pellets and fuel columns.

NUMEC is designing two subassemblies for irradiation in EBR-II. These tests are based on FFTF driver fuel pin conceptional design and program requirements. An objective is to test the performance of fuel pins prepared by a typical fuel supplier. The Approval in Principle letter was submitted to RDT and was recommended for approval by BNW. An engineering test plan was completed by NUMEC and sent to BNW and RDT for review and approval. The BNW comments have been forwarded. Two major design problems are being discussed, (a) fuel smeared density in the pin and (b) effect of 62.5 MW EBR-II operation versus the current 50 MW operation.

Six sites (AI, GE, B&W, UNC, NUMEC and BNW) have started work on Phase II of the Analytical Chemistry Program. Aside from the small difficulties that arise when starting a new program, no large problems have been encountered. A third shipment of special primary spectrographic standards for use in Phase II was sent to the laboratories from LASL during the week of May 12.

Kerr-McGee, Oklahoma City, expressed an interest in participating in the Analytical Chemistry Program as soon as their plutonium laboratory facilities are ready and contractual negotiations permit.

General Electric submitted a proposal to extend Task F-2 work to improve mixed oxide pellet homogeneity detection and control. They propose extending previous work on a more quantitative basis by implementing additional features which were not explored earlier, specifically:

- a. Refine the alpha autoradiography technique so that interpretation and data analysis can be accomplished by fast machine scanning.
- b. Prepare additional standards for both alpha autoradiography and neutron radiography to aid in calibration of machine scanning techniques.
- c. Examine available machine scanning equipment for reliable readout of alpha autoradiographs and neutron radiographs.
- d. Evaluate FFTF reference process pellets for homogeneity by correlating alpha autoradiography, microprobe and neutron radiography, to arrive at specific and practical methods for homogeneity measurement during pellet fabrication.

A contract with General Electric to cover this work is being negotiated.

2. FTTF Fuel Procurement

J. F. Williams

Negotiations are nearing completion with recommended contractors to participate in the prequalification of LMFBR fuel suppliers.

The statement of work was revised to include reporting data on variables for quality assurance purposes and some relaxation of product requirements where final design is not firm enough to justify the fabrication cost of tight tolerances.

All materials to be supplied by BNW are on hand except the end plug stock and plenum springs. The end plug stock was received but was found to have been improperly annealed. It was returned to the supplier to be annealed according to specification of the original order. The plenum springs were to be made from Inconel 750X according to the specifications, but the supplier has been unable to comply. An emergency order is being placed for 300 series SS springs for prequalification. Cladding tubes are being cut to length and cleaned. Other components are being inspected and packaged.

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