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FAST FLUX TEST FACILITY
PERIODIC TECHNICAL REPORT
JULY-DECEMBER 1967

AEC RESEARCH &
DEVELOPMENT REPORT

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Technology

FAST FLUX TEST FACILITY
PERIODIC TECHNICAL REPORT
JULY-DECEMBER 1967

By
E. R. Astley
Project Manager

Compiled by C. P. Cabell

March 1969

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FAST FLUX TEST FACILITY
PERIODIC TECHNICAL REPORT
JULY-DECEMBER 1967

E. R. Astley

ABSTRACT

Work performed by Battelle-Northwest and its supporting contractors on the AEC-sponsored Fast Flux Test Facility Project is summarized and interpreted for the reporting period. Project activities are reported herein under the following major headings: Plant Design, Construction, and Operation; Components Development; Instrument and Controls Development; Sodium Technology, Core Development; Reactor Materials Development; Fuels Development; Physics; and Reactor Safeguards.

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FAST FLUX TEST FACILITY
CONSOLIDATED TECHNICAL REPORT
JULY-DECEMBER 1967

INTRODUCTION

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 and interpret progress in the Fast Flux Test Facility Program during the subject reporting period.

The report consists of a summary section and nine definitive sections. These sections are arranged under the following major headings; Plant Design, Construction, and Operation; Components Development; Instruments and Controls Development; Sodium Technology; Core Development; Reactor Materials Development; Fuels Development; Physics; and Reactor Safeguards.

This report was compiled from "Official Use Only" progress reports submitted by BNW to the AEC. A large number of BNW scientists and technical investigators participated in the work and were identified in the original submissions. However, a number of changes in personnel work assignments have been made since this information was originally reported.

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SECTION 1.0SUMMARY1.1 OVERALL PROGRESS

During the reporting period, evaluation and refinement of the split-core reference concept culminated in a determination that this split-core concept would not provide adequate flux to meet FFTF design goals.

Phase A critical experiments in ZPR-III were completed. Results disclosed need to modify reactivity worth computation methods.

Progress was accelerated in the Sodium Technology Program, and the Radioisotope Transport Loop was brought on line.

In the Reactor Materials Program, initial data appeared to confirm the phenomenon of stainless steel swelling under irradiation.

1.2 FFTF DESIGN, CONSTRUCTION, OPERATION

Design work continued on the reference split-core concept. The eighth conceptual variation was defined and designated as a "Baseline Concept" representing the optimum approach to the split-core design.

Despite extensive analysis, no practical method of achieving the desired flux of 7×10^{15} n/cm²-sec was found.

The major impediment to achieving the desired design conditions was the requirement for approximately 10% more fuel in the split-core conical array than for a compact core of the same effective volume.

1.3 COMPONENT DEVELOPMENT

The control rod decelerator component evaluation test stand was installed in the 314 Building. After some modifications in the initial setup, a design velocity of >10 ft/sec was attained.

A test was conducted to determine the compatibility of molten sodium with high temperature polyimide film. No chemical reaction was observed up to, and including, burning of sodium.

1.4 INSTRUMENTS AND CONTROL DEVELOPMENT

No significant dose rate radiation effects on thermocouples, other than γ -heating at temperatures up to 230 °C (the melting point of tin), and in fluxes up to 10^{16} nv fast, 10^{16} nv thermal, and 10^{10} R/hr, were found.

A new liquid level probe for use with a time domain reflectometer readout was fabricated and tested.

1.5 SODIUM TECHNOLOGY

Setup and calibration of a mercury amalgamation method for oxygen analysis was completed, and the procedure was used routinely to perform service analyses for various PNL components.

A conceptual design for the FFTF sodium sampling system was devised. The design was based on the use of a bypass loop for the initial sampling phase.

The reaction rate of 100 ppm methane in helium, with molten sodium at 800 to 1200 °F, was measured. Very little reaction occurred between the methane and the molten sodium at 800 °F. At 1000 °F, the sodium consumed all the methane in 29 hr. At 1200 °F, the reaction was complete in 24 hr.

The modifications to the Radioisotope Transport Loop found necessary as a result of the acceptance test were completed. The loop was operated at 1.5 gal/min with a test section temperature of 1385 °F.

A thermal convection loop test without radioactivity was completed through a successful 500-hr run at 1350 °F maximum and 350 °F differential temperature.

1.6 CORE DEVELOPMENT

Heated models of FFTF fuel pins are being developed for heat transfer tests. A tantalum heater attained and held a heat flux of 970,000 Btu/hr-ft² without failure, and a new type heater representing a graphite alternative design operated at a heat flux of 1,400,000 Btu/hr-ft² before failing at the electrode connection. Conceptual design was completed, and detailed design started on the electrical lead-in and auxiliary electrode cooling system required for sodium testing of the FFTF heated pins.

A preliminary stress analysis of FTR conceptual driver fuel cladding considering elastic strain, plastic strain, and stress relaxation (creep) resulting from thermal stress and fission gas pressure (but not including thermal bowing or fuel-cladding mechanical interaction) showed the dominant factor to be thermal stress at the midplane resulting from an over-power transient. Based upon all known experimental evidence, the design is safe by a factor of 12.

With the use of digital simulation, loss of BPA power transients within the primary loops and the reactor were investigated. Typical results are presented.

During this time, two digital computer codes for analyses of heat transfer and turbulent mixing in pin bundles--MANTA and COBRA--were operational. Application of these codes to the

FFTF fuel rod bundle was investigated. It was found that MANTA predicts substantially less turbulent mixing than COBRA.

Application of the hydraulic balance feature to the FFTF fuel shroud ducts was investigated. This feature consisted of a tube arrangement whereby the coolant flow enters and leaves the tube through side entry and exit holes. The lower end of the fuel tube was closed.

1.7 REACTOR MATERIALS DEVELOPMENT

The effects of swelling in austenitic stainless steels after irradiation to high neutron fluence were studied. Density measurements were taken on sections cut from an annealed 304 SS EBR-11 irradiation capsule tube and compared with a nonirradiated control. A density decrease was observed in the irradiated specimens. The data are being evaluated and verified.

Tensile samples of 304 SS were examined by transmission electron microscopy. These stainless steel specimens were irradiated to fluences up to 6×10^{21} n/cm², ($E > 1.0$ MeV) in the ETR at 554 °F and tested at temperatures ranging from 68 to 1382 °F. At a fluence of 6×10^{21} n/cm², the irradiation hardening of the material persisted at all test temperatures up to 1382 °F. At lower fluences (1.5×10^{21} and lower), the samples tested at 1382 °F did not show hardening. Based on these observations, hardening of austenitic stainless steel appears dependent upon the fluence of fast neutrons and independent of the reactor neutron energy spectrum.

In-reactor creep tests conducted on 304 SS have been at strain rates greater than 1×10^{-6} hr⁻¹. Tests were conducted on 304 SS at temperatures of 400 and 450 °C and at low stress levels to determine whether greater effects of irradiation on creep rates at low strain rates could occur.

1.8 FUELS DEVELOPMENT

Plutonium dioxide of acceptable purity was made from plutonium buttons vacuum melted in a tantalum (Ta) crucible at 1075 °C. This temperature was held for 5 min before cooling to pouring temperature (950 °C).

Analysis of the performance of two 19 capsule sub-assemblies (PNL-1 and PNL-2) in accordance with the "Guide for Irradiation Experiments in EBR-II" did not reveal any hazards to the reactor.

A review of existing UK and US fast flux mixed oxide fuel irradiation data and irradiations both in progress and planned, was completed. It was found that none of the existing data approach the FTR conditions.

Irradiation tests were run under molten fuel conditions to compare the use of metal and ceramic restrictors. In all cases, the metal restrictors appear to have inhibited axial fuel motion regardless of geometry.

Four short-term (120 hr) Na-bonded, 304 SS clad compatibility capsules containing UO_2 - 25% PuO_2 were heated at 650 and 1000 °C. No evidence of reactions or incompatibility between the oxide-Na bond or the sodium bond-stainless steel interfaces was observed.

Preparation for the first tests to demonstrate the feasibility of decay heat removal from FFTF fuel assemblies with inert recirculated gas systems is proceeding. All individual components, including hexagonal liner tubes, have been fabricated.

1.9 PHYSICS

Comparisons were made between computed reactivity worths ("PERT IV" Program) and Phase A critical experiments performed

in ZPR-III. These comparisons illustrated an unresolved difficulty in obtaining agreement between measured and calculated worths of small fissile samples. Calculated Pu sample worths are higher than the experimental. Calculated depleted uranium worths also are higher than the experimental worths.

The Phase B experimental program at the ZPR-III was started. Modifications to the matrix and preparations for loading of Assembly I of the Phase B Criticals, designated Assembly 51 of the ZPR-III, began in October, 1967.

A number of activation traverses were made through sodium to provide a basis for evaluating calculational techniques and cross section data. Relatively good agreement between calculations and experiments was achieved for the threshold distributions, while fair agreement was achieved for the distribution of resonance activities.

A computer program was written to plot the spatial flux distributions from multigroup, two-dimensional transport codes.

A multizone, one-dimensional maximum accident code employing a point kinetics model with fixed power and worth profiles was developed. Linear and Doppler feedback are included in the code. The hydrodynamics model consists of numerous compressible and movable mesh points.

1.10 REACTOR SAFEGUARDS

The reactor response to a wide range of reactivity ramp inputs and initial conditions was computed to obtain a preliminary estimate of a maximum nuclear accident. All calculations were performed using the MAX code. It was found that the cold core excursions are quite sensitive to the Doppler coefficient, whereas the hot core excursions are rather insensitive. This difference is caused, for the most part, by the inverse dependence of the Doppler coefficient on temperature.

Also, the zero Doppler excursions are more severe in a cold core, whereas moderate Doppler excursions are more severe in a hot core.

The FFTF Design Basis Accident is a postulated nuclear excursion terminated by explosive disassembly of the reactor core. The magnitude of the DBA has not been finally defined, but preliminary calculations indicate a magnitude in the range of 1000 to 3000 MW-sec energy release. Further analyses continued with the goal of defining the DBA early in preliminary design of the facility.

To provide basic data for the evaluation of the relative suitability of various Hanford sites for the FFTF, the potential radiological consequences of a Design Basis Accident (DBA) were calculated. The assumptions made lead to the conclusion that the bone dose is the most important factor in siting the FFTF.

SECTION 2.0PLANT DESIGN, CONSTRUCTION, OPERATION2.1 PHYSICS STATICS CALCULATIONS FOR THE FTR SPLIT CORE

A model of the FTR split core in x-y geometry was developed for use in physics statics calculations (Figure 2.1). The model is a representation of a transverse cut at the core midplane. The 2DF transport theory computer code employing this x-y model was used to calculate critical mass, control rod worths, fluxes, and fission rates in the FTR. A four energy group cross section set derived from the PNL 26-group fast reactor cross section library was used for all calculations. The cross sections include an axial leakage correction.

Table 2.I lists the characteristics of the FTR for a core with a uniform driver fuel and for a core with two equal-volume driver fuel zones. It is to be noted that the calculated flux levels are significantly below the goal of 0.7×10^{15} n/cm²-sec.

2.2 CORE AND FUEL DESIGN FOR PNL SPLIT CONICAL ARRAY

Several sets of design data were generated for the fuel and core for the split core reactor concept. The design effort, based on considerations of flux, testing volume, and safety, was directed toward an optimized split conical array.

Table 2.II lists key parameters used in this study and the target or constraint values of each.

FIGURE 2.1. Schematic Drawing of FTR Split Core in X-Y Geometry

TABLE 2.I. Comparison of One and Two-Zone FTR Split Cores

	Unzoned	Zoned	
		Zone 1	Zone 2
Vol. Frac. PuO ₂ ($\rho = 11.46$) (90/10, ²³⁹ Pu/ ²⁴⁰ Pu)	0.0743	0.0695	0.0834
²³⁸ UO ₂ ($\rho = 10.96$)	0.2687	0.2645	0.2686
Void	0.0610	0.0590	0.0620
Stainless Steel	0.2510	0.2560	0.2460
Sodium	0.3450	0.3510	0.3400
Split Volume, liters (includes test volume)	154.0	154.0	
Test Volume, liters	65.0	65.0	
Core Volume, liters	1000.0	500.0	500.0
Driver + 4 Open Loops	912.8	447.7	465.1
Control Rods	87.2	52.3	34.9
Core Length, cm	85.0	85.0	
Total Reflector Savings, cm	22.0	22.0	
k_{eff}	1.04	1.04	
²³⁹ Pu Critical Mass, kg	616.7	282.9	352.7
²³⁹ Pu Test Mass, kg	15.9	15.9	
Total ²³⁹ Pu Mass, kg	632.6	651.5	
²³⁸ U/ ²³⁹ Pu Ratio	3.859	4.061	3.436
Center Point Flux, $\frac{n}{cm^2\text{-sec}}$	5.920 x 10 ¹⁵	5.461 x 10 ¹⁵	

TABLE 2.II. Key Parameters

Centerpoint flux	$\geq 7 \times 10^{15}$
Fertile-to-fissile ratio	≥ 4.0
Core volume, including central test channels	~ 1000 liters
Maximum power	400 MW
Core ΔP	60 to 90 psi
Core axial ΔT	300 to 400 °F
Pin spacing	$1.16 \frac{\text{c/c spacing}}{\text{Pin OD}}$

The major impediment to achieving the desired design conditions is the requirement for approximately 10% more fuel in the split core conical array than for a compact core of the same effective volume. The peak flux in this 1000 liter core is substantially less than 7×10^{15} at 410 MW.⁽¹⁾ The increased fuel loading requires tight fuel packing (c/c spacing/pin OD ~ 1.12) within the subassembly in order to achieve a fertile-to-fissile ratio near 4:1.⁽²⁾

2.3 CONCEPT VARIATIONS FOR THE SPLIT CORE

Seven reactor concept variations have been previously reported (in BNWL-541). An eighth concept was defined and subsequently adopted with minor revisions as the "Baseline Concept" to serve as the basis for formal evaluation. Table 2.III lists the major features and Figure 2.2 shows the baseline core geometry. Flux attained in this concept is less than 0.7×10^{16} n/cm²-sec.

-
1. Including 10 MW from the closed loops.
 2. The 4:1 fertile-to-fissile ratio may be required for an adequate Doppler coefficient, $\frac{Tdk}{dT} = -0.004$.

TABLE 2.III. *Baseline Concept*

Subassembly Size	4.52 in. hexagon across flats
Pins Per Subassembly	217
Pin Diameter	0.250 in.
Minimum Pin Spacing	0.030
Pitch/Diameter	1.12
PuO ₂ Fraction	0.225
Fissile/Fertile Ratio	3.9:1
Core ΔT	350 °F
Core Inlet	550 °F
Pin Bundle ΔP	60 psi

2.4 CONTAINMENT CONCEPT

The containment concept is intimately related to the reactor concept. It was found that the 200 ft diam containment sphere shown in the Building Arrangement Plan XVI (BNWL-470) can be reduced to 190 ft diam by a general rearrangement of the structures within the containment vessel (Figure 2.3).

2.5 NUCLEAR PROOF TEST FACILITY

2.5.1 NPTF Conceptual Design

The Nuclear Proof Test Facility Reactor is a nuclear mockup of the FFTF reactor. Figure 2.4 shows the NPTF driver fuel module and Figure 2.5 shows a safety control rod. Figure 2.6 shows a general arrangement of the NPTF.

2.5.2 NPTF Instrumentation

In order to develop specifications for operational and experimental instrumentation, flux levels expected in the NPTR and its environs were estimated. For detectors located

SK-3-12877

BNWL-567

2.7

BNWL-567

Neg 0673354

FIGURE 2.3. Section Through Containment Sphere

REFERENCE CONCEPT

SK-3-12823

Neg 0673727-2

FIGURE 2.4. Nuclear Proof Test Facility Driver Fuel Module

QTY	PT	NO	DESCRIPTION	MAT'L	REF
1	1	1	ASSEMBLY		
1	2	2	SHROUD	SST	
1	3	3	HANGER TUBE	SST	
1	4	4	UPPER TUBE SHEET	ALUMINUM	
1	5	5	LOWER TUBE SHEET	ALUMINUM	
1	6	6	REFLECTOR	NICKEL	
56	7	7	FUEL PIN TUBE	ALUMINUM	
6	8	8	RETAINING RING	SST	
14	9	9	FUEL PIN 240 OD		DP 3-27635
54	10	10	FUEL PIN 60 SPACING WITH CLIPPED		SK 3-27635
139	11	11	FUEL PIN 60 SPACING WITH REMOVED		SK 3-27635
2	12	12	PLUG 3/8 NPT FLUSH FACE	SST	
6	13	13	CAPSCREW 1/8 DIA 1 1/2 LG 304 SS	SST	
6	14	14	CAPSCREW 1/8 DIA 1 1/2 LG 304 SS	SST	
AR	15	15	FILL	SODIUM	

2.8

BNWL-567

2.9

BNWL-567

Neg 0673727-1

SK-3-12824

FIGURE 2.5. Nuclear Proof Test Facility Safety Control Rod

REFERENCE CONCEPT

0 5 10
SCALE 1/4" = 1'-0"

2.10

BNWL-567

Neg 0673727-3 SK-3-12822

FIGURE 2.6. General Arrangement of Nuclear Proof Test Facility

20 to 30 cm from the core just inside the reflector, the ranges were as follows: shutdown, $10^{-1} - 10^2$ n/cm²-sec; and at ~2 kW, $10^8 - 10^9$ n/cm²-sec. Of this flux, approximately 40% was in the energy region above 0.1 MeV. The shutdown values stem from the spontaneous fission source in the fuel ($\sim 6 \times 10^7$ n/sec) and those for operating power levels result from scaling down values obtained in FTR shielding calculations. The anticipated flux levels in the NPTR core at 2 kW are of the order of $10^{11} - 10^{12}$ n/cm²-sec, with the same spectral distribution as the FTR.

SECTION 3.0COMPONENT DEVELOPMENT3.1 REACTOR REFUELING REQUIREMENTS

The system functions and design requirements for the Reactor Refueling System were defined. The two broad concepts identified for evaluation are the multiple special-purpose fuel handling machine concept, and the single multiple purpose machine concept. Conceptual features of these concepts identified for evaluation are the type of cooling (wet or gas) and the type of access during normal fuel handling operations.

Concept studies were performed at Aerojet-General and Atomics International to develop and evaluate these basic concepts and features.

3.2 AEROJET-GENERAL STUDIES ON REFUELING SYSTEMS

Aerojet-General studied a number of potential refueling systems and provided concepts and evaluations for a wet turret type system, a combination wet/dry turret type system, and a wet single tube system. Each concept employs a single positioning and alignment system capable of using interchangeable casks for the various components to be installed and removed from the reactor.

All of the concepts were based on remote control and operation in a remote hot cell with an inert helium atmosphere and with maximum visual monitoring from controlled areas.

3.2.1 Wet Turret System

The wet turret system utilizes liquid sodium from the reactor as the coolant media during subassembly removal and transfer operations. Two subassembly handling tubes mounted on a rotating turret mechanism permit complete subassembly exchanges with a minimum of positioning, alignment, and engagement operations.

The study indicated the following to be high risk development items:

- Grapple, drive system (racks, chains, screws, etc.)
- Grapple actuation motors
- Interface system (gate valve, interface seal)
- Seals in bottom of cask
- Instrumentation and cables.

This system offers the following advantages and disadvantages.

Advantages

- Fuel element continuously submerged in coolant
- Cooling achieved by natural convection
- Short refueling cycle time
- Double valving prevents loss of coolant
- NaK cooling system also accomplishes preheat.

Disadvantages

- Differential pressure control required for Na flooding
- Internal surfaces exposed to Na vapors
- Difficult to clean and decontaminate.

3.2.2 Combination Wet/Dry Turret System

The combination wet/dry turret system utilizes liquid sodium from the reactor as the coolant media during subassembly removal operations, and helium gas as the coolant media during transfer operations.

The helium cooling system is then made ready, the sodium is pushed back into the reactor, and the reverse flow helium cooling is initiated.

High risk development areas are as follows:

- Interface subsystem seals and valve
- Mechanical design of coolant connector and seal near upper cooling holes on subassembly
- Effects of residual Na on gas pump design
- Sliding face seal at lower end of tubes.

The advantages and disadvantages for this system are as follows:

Advantages

- Cooling gas utilized to preheat replacement subassembly
- Decreased sealing problems
- Decreased equipment exposure to Na
- No Na transported within hot cell
- Cooling ensured during subassembly removal.

Disadvantages

- Coolant pumping station required
- Reliability and safety of cooling system decreased
- Difficult to design and fabricate
- Problem involved in completely removing Na vapor from helium gas
- Difficult to clean and decontaminate.

3.2.3 Wet Single Tube System

The wet single tube system utilizes single tube casks flooded with liquid sodium from the reactor. Primary subassembly cooling is accomplished by natural convection of the liquid sodium inside the cask tube, and secondary cooling is accomplished by natural convection of NaK in an annulus provided around the inner cask tube. Two casks would be required in this concept for the most efficient operation. High risk development areas are as follows:

- Valves
- Interface seals
- Cask mounting and tilting mechanism.

The advantages and disadvantages for this system are as follows:

Advantages

- Simplicity of single tube cask
- Increase in alignment stability
- Decontamination and maintenance facilitated
- Smaller, lighter cask simplifies handling
- Multiple cask can be provided at less cost.

Disadvantages

- Two remote joints
- Cask valve wet during transfer
- Increased cask length and hot cell height
- Increased grapple length
- Decrease in visual access to reactor top
- Positions and alignment system are more complicated.

3.3 ATOMICS INTERNATIONAL FUEL HANDLING MACHINE STUDIES

Studies by Atomics International were directed primarily to an evaluation of types of fuel handling machines. Of particular interest were studies on Sodium Cooled Fuel Handling Machines and Helium Cooled Fuel Handling Machines.

3.3.1 Sodium Cooled Fuel Handling Machines

The Sodium-Cooled Fuel-Handling Machine System is a shielded system capable of inserting and removing reactor core components under controlled environmental conditions.

The freeze seat valve and the sodium level control system are high risk components requiring development. Other components, such as the snorkel and the freezing system, the grapple and drive (sodium wetted chains), and the sodium cooling system, require proof-of-principle testing. As the wet fuel handling

machine is a first-of-a-kind item, the machine should be proof tested prior to installation in the operating reactor. Advantages and disadvantages of this concept (assuming a multiple purpose machine) are as follows:

Advantages

- The element surfaces are not contaminated since the coolant in contact in the FHM is the reactor coolant.
- The magnitude and direction of the temperature gradient are the same in the FHM as in the reactor.
- The upper limit of the secondary cooling system, is well above the 30 kW driver fuel cooling requirement, and provides a wide design margin.
- The large heat capacity of the primary liquid system results in slow temperature transients.
- The WFHM can handle the reference driver fuel.
- Interface with reactor is readily accomplished.
- Radioactive material is contained inside main shielding.

Disadvantages

- The machine cannot readily handle coolants other than sodium.
- The differential pressure control system used for sodium level control in the machine is not a proven design.
- Some of the mechanisms must be capable of operation in sodium liquid and/or vapor.
- The transfer cell interface poses a cell design problem by requiring a sodium filled pot to accept fuel from the wet fuel handling machine.
- The significant quantity of radioactive sodium, used as the primary coolant in the machine, poses a potential safety hazard and a cross contamination problem.
- Use of the machine as a dry handling machine poses a potential safety hazard for irradiated fuel handling. The interlock system and/or administrative control requirements are a disadvantage.

3.3.2 Helium-Cooled FHM

The gas-cooled fuel-handling machine removes fission product decay heat during the transfer of an irradiated subassembly by the forced convection of helium. Transitional cooling from the reactor coolant (sodium) to the gaseous coolant (and vice versa) occurs when the fuel zone of the subassembly is just covered by the liquid metal coolant.

When the subassembly is totally within the cask, the tank system is secured, and the cask recirculating gas system is brought into operation. This system is a complete heat transfer system with a gas circulator used in conjunction with a NaK heat exchanger to dump decay heat directly to the reactor hot cell. For a decay power of 30 kW, a helium flow rate of 150 lb/hr is required to limit the fuel pin cladding to a maximum temperature of 900 °F. Advantages and disadvantages of the system are listed as follows:

Advantages

- Small barrel and guide tube provide good guidance during the vertical transfer of a subassembly.
- Sodium carryover on fuel elements is reduced so that substantial amounts of sodium are not involved during the subsequent cleaning required before examination.
- Cask preheating with gas is gentle and hot spots are avoided.
- Viewing of the subassembly during vertical transfer is possible with periscope or closed circuit TV.
- Cask internals are easily removable for inspection and maintenance.
- Indexing and alignment tolerances for cask interfaces are not unusually restrictive.

Disadvantages

- The high pressure gas system (15 psig) makes gas containment difficult.

- Flow of gas from the top to bottom of the subassembly reverses the axial thermal gradient.
- Mass flow of gas is sensitive to flow restrictions in the subassembly.
- Transitional cooling from sodium to gas may thermally shock subassembly to some extent.
- Operator reaction time to cooling system malfunction must be short to prevent meltdown because of the rapid temperature increase that occurs upon loss of gas flow.
- No appreciable heat sink is available in event of fuel subassembly hang-up and/or loss of coolant.
- The potential for gas carryover to the primary sodium pump loop will impose design restrictions on the reactor or process system designers to prevent the passage of gas bubbles through the core.

SECTION 4.0INSTRUMENT AND CONTROL DEVELOPMENT4.1 REACTOR ENVIRONMENTAL EFFECTS ON SIGNAL CABLES

This study was performed to develop knowledge of the processes by which radiation induces currents into a signal cable so that cables most resistant to this type of induced current can be specified for in-core FFTF use. A secondary objective was to determine instantaneous radiation or dose rate effects on thermocouple calibration.

Tests performed in a Hanford reactor and in the Washington State University Pulsed TRIGA Reactor indicated no significant dose rate radiation effects on cables and thermocouples, other than γ -heating at temperatures up to 230 °C (the melting point of tin), and in fluxes up to 10^{16} nv fast, 10^{16} nv thermal, and 10^{10} R/hr.

4.1.1 Hanford Reactor Tests

To determine the magnitude of signals induced in thermocouples by the presence of a radiation field, experiments on three chromel-alumel thermocouples were performed in a Hanford reactor. No significant effect was found.

The melting point of lead was used as a constant temperature reference for both in-core and out-of-core measurements. Any recorded difference in the in-pile and out-of-pile readings is due to three effects. These are the effect of the radiation field on the Seebeck Coefficient, the induced currents produced by the radiation field, and the lack of perfect reproducibility of the lead melting point because of inherent measurement error. The last of these previously investigated effects leads to an uncertainty of ± 25 microvolts (μ V) or approximately $\pm 1/2$ °C.

Voltages were recorded for each couple during several traversals of the melting-point plateau. For each thermocouple, the average voltage measured outside the radiation field was subtracted from the average voltage recorded in-core to form ΔV , the voltage induced by the radiation field. These are listed below:

<u>Thermocouple</u>	<u>$\Delta V(\mu V)$</u>	<u>Characteristics (Chromel-Alumel)</u>
1	-10	1/16 in. OD grounded junction
2	-71	1/16 in. OD grounded junction
3	+1	1/8 in. OD grounded junction

Number 1 was a vendor-supplied thermocouple, while the junctions for 2 and 3 were fabricated in the Battelle-Northwest Standards Laboratory. These data indicate no significant effect on chromel-alumel thermocouples due to the radiation field encountered.

4.1.2 TRIGA Experiments

Experiments in the Washington State University Pulsed TRIGA Reactor included the measurement of both induced currents and insulator conductivity in cable and instantaneous or dose rate effects on chromel-alumel and iron constantan thermocouples. The peak pulse induced current per centimeter of cable length and the bulk insulator conductivity was measured for a 0.250 in. OD cable for a combined high intensity neutron and gamma-ray flux.

Figure 4.1 shows the peak induced current per centimeter versus the distance from the bottom of the TRIGA core. The curve shows both negative and positive induced currents over the length of the core. The negative component dominates outside the core region, while the positive component dominates near the center of the fuel elements, i.e., in the active core region. The cable current data imply that the prompt component of the reactor induced current can be minimized.

FIGURE 4.1. Induced Cable Current as a Function of Reactor Position

Figure 4.2 shows the bulk conductivity of the MgO cable insulation versus the distance from the bottom of the TRIGA core. The change in insulator conductivity does not appear to depend upon whether the net induced current per centimeter was positive or negative, but only upon the total radiation intensity.

Although the reactor core gamma photon profile has not been measured, the WSU reactor staff estimated neutron flux to be $\sim 10^{16}$ nv ($E > \text{keV}$), $\sim 10^{16}$ nv ($E < 10 \text{ keV}$), and the gamma dose rate to exceed 10^{10} R/hr in this region at the peak power of the pulse.

The instantaneous radiation or dose rate effects on three chromel-alumel and one iron constantan thermocouple were measured in the TRIGA reactor simultaneously with the cables. The results in every case indicated an increase in temperature of less than 3°C during the pulse.

In addition to the pulsed experiments, one test was carried out at steady state reactor power. The reactor was brought to full steady state power of 1 MW in 7 min. No voltage change was observed on the thermocouples during this time. A temperature change as small as 0.2°C would have been easily detectable. Full power was maintained for 30 min during which no change in voltage of the thermocouples was detected. During this time, however, the instrumentation noise level increased. After 30 min of irradiation, the reactor was scrammed. No voltage change was detected during the scram. Data were not taken after the scram due to the high instrument noise level. The flux level during this steady state run was greater than 10^{13} nv fast, 10^{13} nv thermal, and 10^7 R/hr.

FIGURE 4.2. Induced Insulator Conductivity as a Function of Reactor Position

4.2 LIQUID LEVEL MEASUREMENT

A new liquid level probe for use with a time domain reflectometer readout was fabricated and tested. A ceramic insulator completely surrounded the center electrode (Figure 4.3). Tests in water indicated that, because of the conductivity of water, the ratio of the diameter of the center conductor to the diameter of the outer conductor would change at the liquid-air interface. This change would produce an impedance mismatch reflecting a portion of the time domain pulse. Because the conductivity of sodium is better than water, a probe of similar design used in conjunction with a ceramic insulator compatible with 1200 °F sodium should be feasible for liquid level measurements in the FFTF. Potentially, this concept offers two features which, taken together, should give it an advantage over other liquid level measuring techniques. These are:

- No mechanical actuation is required.
- Continuous signal display over the length of the probe enables the operator to "see" the condition of the surface of the probe and be able to compensate for temperature effects where calibration steps are built into the probe.

4.3 IN-REACTOR COOLANT TEMPERATURE SENSORS

The objective of this work was to develop and provide reliable, commercially-available thermocouples for measuring coolant temperatures in the FTR thermal and nuclear environment.

Subassembly X030 containing the first two thermocouple irradiation capsules (BT-1 and BT-2) remained in position 6E1 in EBR-II and had accumulated an estimated exposure of approximately 2×10^{21} n/cm² ($E > 0.18$ MeV).

PROBE

PROBE IN AIR

Liquid-Air Interface

PROBE WITH TIP
IN WATER

FIGURE 4.3. *Liquid Level Probe (Reflectometry Type)*

Six additional, 32-inch long bare-wire, chromel-alumel thermocouples were fabricated, and calibrated from 200 °F to 1400 °F. They will be inserted in the second set of capsules being prepared for EBR-II.

4.4 DELAYED NEUTRON FAILED FUEL MONITOR

The objective of this effort was to test and evaluate nuclear components of a counting system to be used in the delayed neutron failed fuel monitor.

Three 0.25 in. diam ^{10}B proportional neutron counters, two clusters of these 0.25 in. counters connected in parallel, and three one-in. diam fission counters underwent operating tests in high gamma fields. Initially, tests were being conducted at ambient temperatures. Similar testing was subsequently planned at elevated temperatures.

Each of these counters was exposed to increasing levels of gamma flux, and pulse height distribution curves were obtained by the use of a multichannel analyzer. Two 0.25 in. proportional counters operated up to about 10^5 R/hr. Others in the multi-element assemblies broke down at dose rates as low as 10^3 R/hr. The analytical equipment used limited testing of fission counters to 6×10^5 R/hr. Simpler electronics must be used for counting at higher dose rates.

4.5 CONTROL SYSTEM ANALYSIS

The purpose of this activity was to provide a control system simulator facility for evaluating FFTF control methods, developing data acquisition and control program structure, and performing overall control process analysis and simulation.

Nine additional digital-to-analog converter channels were installed and interfaced to the PNL hybrid system. Eight additional multiplexer channels were installed and interfaced. The hybrid now has a total of 24 multiplexer channels and 17 digital-to-analog converters. This expansion in digital-analog communication channels provided the capability for higher computing speed and greater scope of process simulations for the FFTF.

Performance tests of the drum system to be used for intermediate-speed storage of data and control programs were completed. An extensive evaluation of the total array of read-write heads has led to modifications providing reliability improvements.

SECTION 5.0
SODIUM TECHNOLOGY

5.1 SELECTION AND TESTING OF RADIOACTIVE SODIUM ANALYSIS METHODS

Standard procedures for chemical analysis of impurities in sodium are being established, and methods for these procedures determined.

5.1.1 Oxygen Analysis

Setup and calibration of a mercury amalgamation method is complete and the procedure is being used routinely to perform service analyses for various PNL components.

5.1.2 Nitrogen Analysis

Evaluation of a silver nitrate polarographic titration method for measuring cyanide was discontinued because the procedure is not sufficiently sensitive to detect required cyanide levels as low as 25 μ g.

5.1.3 Carbon Analysis

The PNL procedure for analysis of carbon in sodium utilizes total combustion followed by gas chromatographic analysis of the evolved CO_2 .

A new ignition procedure was developed which prevented damaging creep of the oxide during ignition on samples as large as 1 g. The primary modifications of the procedure involve the use of a small amount of silicon dioxide powder as a "sponge" and the use of a longer (2 hr) burn at lower (400 °F) temperature.

5.1.4 Sodium Sampling

A conceptual design for the FFTF sodium sampling system was developed, and was described in document BNWL-491.⁽¹⁾ This design is based on the use of a bypass loop for the initial sampling phase. The system can be versatile enough to permit the addition of the more complex dipping bucket-distillation sampler in the event that future development should prove the capability of remotely operating this device with a high degree of reliability. Secondary sampling will be accomplished by removal of a portion of the loop sodium and subsequent sectioning. Backup initial and secondary sampling techniques are under consideration.

Additional information will be required to select the final design of the FFTF sampling system. The major areas requiring attention are impurity segregation, sampler preconditioning, general sampler requirements, blind leg problems, the integrity of high temperature mechanical seals, and the effects of atmospheric exposure on sample ends.

The experimental program required to develop and test prototypical samplers has been outlined. This program entails the construction and remote operation of prototype samplers in nonradioactive environments, followed by the design, construction, and proof testing of prototypes on radioactive streams.

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1. F. A. Scott. Sampling FFTF Sodium Coolant for Analytical Purposes, BNWL-491. Battelle-Northwest, Richland, Washington, September 1967.

5.2 SELECTION AND TESTING OF IN-LINE SODIUM ANALYSIS INSTRUMENTS

Current FFTF requirements and the status of instrument development programs performed at various laboratories throughout the USA on in-line instrumentation for sodium purity measurements were described.⁽¹⁾

5.3 SODIUM COVER-GAS INTERACTIONS

Gas sampling and instrument selection for FFTF cover gas applications are being evaluated. Investigations on sodium cover-gas interactions were started. The goal of the experimental work will be to establish allowable impurity levels for FFTF cover gases.

The reaction rate of 100 ppm methane in helium, with molten sodium at 800 to 1200 °F, was measured. Very little reaction occurred between the methane and the molten sodium at 800 °F. The methane concentration decreased during a 21-hr period from 100 ppm to 70 ppm and appeared to have reached an equilibrium with the sodium at this point. An evolution of hydrogen was noticed during the first few minutes of this run. After an initial concentration of 48 ppm, the hydrogen diminished to 29 ppm during the duration of the run. At 1200 °F, the reaction rate of the methane with the molten sodium was much

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1. T. F. Demmitt. In-Line Instrumentation for Sodium Purity Measurements in FFTF: Status Report, BNWL-562. Battelle-Northwest, Richland, Washington, November 1967.

faster. Virtually all of the methane was consumed by the sodium over a period of 24 hours. The methane decreased from an original concentration of 95 ppm to a concentration of approximately 1 ppm, as measured by a gas chromatograph. A large evolution of hydrogen was seen during this run. The initial hydrogen concentration of 120 ppm decreased to 39 ppm during the course of the run. The reaction of methane and sodium at 1000 °F was found to be similar to the 1200 °F test. The sodium consumed virtually all of the methane in 29 hr.

5.4 EVALUATION OF FFTF SODIUM CLEANING REQUIREMENTS AND DECONTAMINATION FEASIBILITY

The objectives of this task was to evaluate FFTF sodium removal processes and to develop decontamination techniques.

5.4.1 Decontamination Process Screening

Preparation of the test specimens required to assess effects of decontaminants on structural materials was started. Arrangements were made to obtain contaminated specimens from PNL and LASL systems for decontamination effectiveness studies. The laboratory facilities needed to begin beaker-scale determination studies were set up and installation of the decontamination hood was completed.

5.4.2 Evaluation of Sodium Cleaning

Sodium cleaning procedures are being evaluated for FFTF fuel assemblies, components and piping systems. Material specimens on which the sodium cleaning procedures will be tested are now being prepared. The specimens have been held 600 hr in sodium at constant temperatures ranging from 1000 to 1400 °F. The specimens will be discharged and cleaning techniques investigated after a 1000-hr exposure period is complete.

5.5 EVALUATION OF RADIOISOTOPE TRANSPORT AND DEPOSITION

The objectives of this task were to define the parameters causing deposition of radioisotopes and to predict deposition patterns in FFTF piping.

5.5.1 Radioisotope Transport Loop Operation

The modifications to the Radioisotope Transport Loop found necessary in the acceptance test were completed. These modifications included reworking the plugging loop cooling system, installation of new heater bus bars, and providing additional cooling to the cold trap. The loop was refilled with sodium and flow was established. Adequate cooling of the plugging meter allowed successful plugging runs to be achieved in the low sodium range below 10 ppm oxygen (equivalent). With additional cooling to the cold trap, proper temperature control of the cold trap was realized.

The Radioisotope Transport Loop was operated under the following conditions:

Electrical Power	
From primary	28.98 kW
Into heater	28.15 kW
Temperatures	
Test section	1385 °F
ΔT (system)	453 °F
Pump	932 °F
Heat Exchanger	990 °F
Flow	1.5 gal/min

Design is proceeding on expansion of the loop into a pit under the loop. Design criteria for the loop pit extension have been modified as follows:

- Geometry of the primary loop will be revised and rescaled from a 1/144 scale loop to a 1/250 scale loop, reflecting changes in the nominal maximum flow specified for FFTF closed loops.
- To facilitate scanning, the loop piping will be rearranged so that piping and components are, to the greatest extent possible, on a common plane.
- The secondary system piping will be rearranged to facilitate equipment access, drainage, and decontamination.

Application of loop scaling factors to calculation of the primary loop piping geometry is currently in process.

Dummy Type 304 SS fuel clad samples were irradiated in a Hanford reactor. The samples are now cooling prior to loading into the radioisotope transport loop for thermal convection parametric studies.

5.5.2 Thermal Convection Loop Radioisotope Transport

The thermal convection loop test, without radioactivity, was completed after a successful 500-hr run at 1350 °F maximum and 350 °F differential temperature.

The thermal convection loops were sectioned and examined metallographically. The examination led to these conclusions:

- A ferritic layer may have been formed in the grain boundary regions of the hot leg for a distance of about 1 mil from the sodium-stainless steel interface.
- No gross intergranular penetration nor intergranular effects were observed in specimens decontaminated in alcohol and then pickled.
- Carbide precipitation was observed in hot-leg specimens but not in cold (800 to 1000 °F) leg specimens.
- No evidence of mass transfer crystals were seen in cold-leg or dead-leg specimens.

These results gave a baseline for the metallurgical condition of the steel exposed to sodium.

5.5.3 Fission Product Transport

A comparative study of sodium transport of fission products and corrosion product materials, following cladding failure in a FFTF closed loop, has concluded that the gamma radiation from the former probably would be at least 10 times as great as from the latter. This conclusion was the result of the typically high fission product inventory in a fuel pin or subassembly due to the high power density (related to the high neutron flux) that is anticipated. One fact brought out in the study was that more ideal deposition conditions (of material transported by the sodium) may exist in the pump region than in the intermediate heat exchanger immediately upstream from the pump.

5.6 CLOSED LOOP CONTAMINATION CONTROL SYSTEM

The methods of controlling fuel failure debris resulting from a closed loop fuel test are being analyzed, developed, and proof tested.

The development program for FFTF contamination control involved a series of laboratory-scale studies of sodium-fission product-metal surface interactions and a testing program in an engineering-scale model of an FFTF closed loop contained in a shielded cell capable of handling fully irradiated fuel material.

The diffusion penetration of fission products into the stainless steel metal surface is being studied by exposing test coupons to sodium tagged with fission products. A static

isothermal diffusion test was terminated after 1052 hr at 1190 ± 20 °F. The entire alcohol-acetone cleaning reduced the amount of activity by a factor of two to three. Adequate "non-smearable" activity for analysis remained. The activity on the coupon side adjacent to the fission product release area is obviously higher than the opposite side. Identifiable activity includes ^{144}Ce , ^{141}Ce , ^{103}Ru , ^{95}Zr , and ^{95}Nb .

The Radioisotope Transport Loop, used primarily to study parametric effects of operating variables on material transport and deposition, was to be used for preliminary tests of decontamination methods.

5.7 DIFFUSION OF FISSION PRODUCTS INTO STAINLESS STEEL

Type 304 SS capsules containing mixed fission products and sodium are being run at 1200 °F to measure the depth of diffusion of the fission products. The first capsule was opened after 1052 hr at 1190 ± 20 °F. After sodium removal, about 43,500 counts/min of activity tightly bound to the surface was disclosed. The samples were electropolished to remove increments of surface and recounted. Preliminary results indicated that 99% of the activity had diffused less than 1 mil.

SECTION 6.0
CORE DEVELOPMENT

6.1 HEAT TRANSFER TESTING OF FFTF FUEL ASSEMBLIES

The program to determine heat transfer characteristics of FFTF fuel assemblies required heated models of FFTF fuel assemblies. In the three principal test phases, fuel element models of different maximum capabilities were required.

These were:

	<u>Q/A, Btu/hr-ft²</u>	<u>Max. Coolant Temp., °F</u>
● Phase I (Local cladding temperature)	500,000-800,000	800
● Phase II (local boiling and loss-of-coolant tests)	1,000,000	1000-1200
● Phase III (critical heat flux)	2,000,000-2,500,000	1500

The purpose of the present work is to develop the technology required for the design and construction of these fuel assembly models and to demonstrate their performance in water and sodium.

6.1.1 Pin Fabrication Quality Control

The results of performance tests on heated pins have indicated need for the highest standards of quality control in the fabrication of the heated pins. To attain this goal, each phase of the fabrication process was carefully reviewed to produce:

- Better mechanical contact between central heater rod and the ceramic layer and the sheath and ceramic layer
- Higher densification of the ceramic layer
- Improved centering of the central heater rod and electrode with respect to the sheath
- Better cross-sectional uniformity of the central heater rod.

6.1.2 Results of Quality Control

Several of the improved techniques involving modification in the vibrational compaction and swaging processes have already been implemented.

The first heater produced under the new program phase consisted of a tantalum central heater with resistance upset welded molybdenum electrodes. This assembly was surrounded by a layer of boron nitride and an outer sheath of stainless steel. X-radiographs of the heater showed excellent centering and straightness of the central tantalum rod. In the first performance tests of this heater, heat fluxes of $950,000 \text{ Btu/hr-ft}^2$ were attained and held for 30 min in water at an exit temperature of 342°F (172°C). In the next phase of testing, this heater pin will be operated at heat fluxes in excess of the $2,000,000 \text{ Btu/hr-ft}^2$ design figure.

A second heater produced in this new program phase used beads of boron nitride in place of the vibrationally compacted and swaged powder used in the first heater. In tests of this heater pin, it attained heat fluxes of $(700,000) \text{ Btu/hr-ft}^2$ at an exit water temperature of 349°F (176°C).

A third heater, of graphite alternative design, was also tested. This heater pin attained a heat flux of $1,400,000 \text{ Btu/hr-ft}^2$ at water temperatures of 145°F (63°C) before failing at the mechanical joint between the tungsten electrode and the central graphite heater. The graphite heater is in a very preliminary stage of development and the high heat flux attained was considered an excellent proof of the basic heater design principle.

A tantalum heater has attained and held a heat flux of $970,000 \text{ Btu/hr-ft}^2$ without failure, and a preliminary heater representing the graphite alternative design operated at a heat flux of $1,400,000 \text{ Btu/hr-ft}^2$ before failing at the electrode connection.

Three tantalum coaxial heaters have been fabricated under the new program of exacting quality control. Radiographic examination of the assemblies show excellent centering of the electrode-heater rods within the sheath for two of the three pins. Sheath-central conductor electrical resistance was on the order of 10 megohms at 75 °F. The heaters are presently being baked in a vacuum furnace at 450 °F (233 °C) to remove residual traces of water from the boron nitride layer. These heaters will be tested in water during the next report period.

Three graphite heaters were tested to destruction in examining failure modes of this type. Two were made from grade FBGF graphite and the third from ATJ. All previously constructed graphite heaters used grade ATJ exclusively. Post test examination of these three heaters is not complete. However, FBGF graphite apparently is an unsuitable material for this application due to failure at very low heat fluxes.

6.1.3 Auxiliary Systems

Conceptual design has been completed, and detailed design started on the electrical lead-in and auxiliary electrode cooling system required for sodium testing of the FFTF heated pins. The basic system consists of a double chamber. The first is directly cooled with Dowtherm A, and the second chamber will contain a pressurized gas system to permit both electrical connection and isolation of the power lead-in components. Two separate self-contained Dowtherm coolant systems will provide cooling for critical components in the second chamber. The advantages of this arrangement is that adequate cooling will be provided to critical components without excessive axial thermal stresses. In addition, the difficult problem of electrically isolating the heater sheath from the power lead-in and electrode while simultaneously providing adequate cooling in this region is circumvented. The principal disadvantage of this system is its mechanical complexity.

6.2 PRESSURE DROP FOR 37-PIN SUBASSEMBLIES

Hydraulic analyses employing de Stordeur's method were made of the 0.030 and 0.040 in. diam wire wrap 37-pin subassemblies for EBR-II. Figure 6.1 illustrates the calculated pressure drop across the rod bundles as a function of 800 °F sodium flow. Data from water flow/ ΔP measurements (converted to 800 °F sodium flow) on a 0.0915 in. wire wrapped, 37-pin subassembly are also shown for comparison purposes. These data indicate that the maximum flow obtainable in EBR-II with a plenum-to-plenum ΔP of ~ 40 psi will be about 35 and 50 gal/min for the 0.030 and the 0.040 in. diam wire wrap subassemblies, respectively, when subassembly entrance and exit losses are also considered. These flow rates appear satisfactory for meeting EBR-II thermal hydraulic requirements.

6.3 SODIUM TESTING OF MULTIPLE PIN ASSEMBLY

To make an inspection, the seven-rod flow test cluster was removed from sodium flowing at ~ 1060 °F and at a velocity of ~ 25 ft/sec after a total of 3430 hr of test. Fuel rods (Figure 6.2) were generally bright.

No evidence of erosion, corrosion dimensional change, or wire wrap looseness was evidenced. Tests are continuing.

6.4 COOLANT TEMPERATURE AND FLOW DISTRIBUTION IN THE SUBASSEMBLY

Analysis of the coolant temperature and flow distribution within the 217-pin bundle, including the transition from hexagonal to circular shape of the subassembly duct, was made. COBRA, the computer code which analyzes heat transfer and turbulent mixing in rod bundles, was used.

FIGURE 6.1. Pressure Drop Versus Flow for EBR-II 37-Pin Subassemblies

6.6

FIGURE 6.2. *Seven Rod Cluster After 3430 hr Sodium Flow*

BNWL-567

The present design (Figure 6.3) shows a transition from a hexagonal to circular cross section which is completed at 46 in. above the bottom of the active core. This change in subassembly shape reduces the pressure drop in the fission gas plenum region.

The parameters used in this analysis are:

Pin OD, in.	0.250
Wire wrap OD, in.	0.030
Active fuel length, in.	32
Upper axial reflector length, in.	6
Fission gas plenum length, in.	24
Axial power peaking factor	1.24
Subassembly average pin linear heating rate, kW/ft	9.7
Subassembly average coolant ΔT , °F	398

Figure 6.4 shows the temperature distribution with a range of assumed values for the turbulent mixing parameter.

6.5 FUEL CLADDING STRESSES

A preliminary stress analysis of FTR conceptual driver fuel cladding considering elastic strain, plastic strain, and stress relaxation (creep) resulting from thermal stress and fission gas pressure (but not including thermal bowing or fuel-cladding mechanical interaction) showed the dominant factor to be thermal stress at the midplane resulting from an over power transient. This factor causes a plastic strain of 0.0011 in./in. (Figure 6.5). Based upon known experimental evidence, the design is safe by a factor of 12.

The conditions at the core midplane (maximum temperature gradient), top of fuel pin (maximum temperature, least gradient), and at a position three-fourths along fuel pin (intermediate temperature and gradient) were calculated. The calculations included the following sequence of events:

- (1) Initial startup in 450 °F sodium
- (2) Irradiation to end-of-life (6600 hr)

FIGURE 6.4. Coolant Temperature Distribution in Subassembly

FIGURE 6.5. Midplane Temperature, Pressure, Stress, and Strain Relationship

- (3) Shutdown (to 450 °F sodium)
- (4) Overpower transient (6\$/sec reactivity insertion)
- (5) Scram (to 450 °F sodium).

Calculations were performed for both 0.012 and 0.016 in. Type 304 cladding thickness. The cladding temperature, T , was assumed to be a linear function of time, t , during startup, shutdown, reactivity insertion, and scram. To evaluate the internal pressure, p , the contained gas was assumed to behave as a perfect gas. At steady-state conditions, the cladding temperature was assumed to be independent of time and the pressure was assumed to increase linearly from 200 to 1000 psi in 6600 hr. The computer program (PECT) was used for the analyses.

The viscous flow factor η was defined by

$$\eta = \bar{\sigma}^{1-B}/3C$$

where the effective stress $\bar{\sigma}$ is defined in terms of the radial stress, σ_r , the tangential stress, σ_θ , and the axial stress, σ_z , comprising the principal stress components for an axially symmetric plane strain problem according to

$$\bar{\sigma} = \sqrt{(\sigma_r - \sigma_\theta)^2 + (\sigma_\theta - \sigma_z)^2 + (\sigma_z - \sigma_r)^2} / \sqrt{2}$$

On the assumption that the material properties are independent of fluence, PECT showed the maximum nonelastic effective strain, $\bar{\epsilon}$,⁽¹⁾ to occur during the reactivity insertion at end of life. The analysis assumes viscous flow only during steady-state operation, and that time parameters during startup, shutdown, reactivity insertion, and scram can be any value provided that the steady-state heat conduction equation is satisfied.

(1) Defined in terms of the nonelastic radial strain, $\bar{\epsilon}_r$, tangential strain, $\bar{\epsilon}_\theta$, and axial strain, $\bar{\epsilon}_z$, by

$$\bar{\epsilon} = \sqrt{2} \sqrt{(\bar{\epsilon}_r - \bar{\epsilon}_\theta)^2 + (\bar{\epsilon}_\theta - \bar{\epsilon}_z)^2 + (\bar{\epsilon}_z - \bar{\epsilon}_r)^2} / 3$$

PECT also showed that shutdown and startup after steady-state operation (but before reactivity insertion) was possible without changing $\bar{\epsilon}$.

Figure 6.5 shows calculated relationships between temperature, pressure, stress, and plastic strain at the middle of a fuel pin. This is the location of maximum thermal stress. The abscissa is a variable time scale. The upper figure indicates the pressure and temperatures used in the calculations. The lower figure shows effective stress and plastic strain at the outer surface of the cladding (thickness, 0.012 in.).

6.6 HYDRAULIC TESTING OF FFTF FUEL ASSEMBLIES AND COMPONENTS

The objective of this program is to determine experimentally the overall and local hydraulic conditions for an FFTF fuel subassembly. Specifically, these experiments will:

- Determine the pressure drop-flow behavior of presently proposed FFTF fuel subassemblies
- Extend the range of these data to possible future and alternative fuel subassembly designs
- Determine local coolant flow distribution in the fuel subassembly flow channels during both normal and distorted conditions.

Detailed preliminary planning of the test program required to accomplish the objectives of this work was performed. Conceptual designs of test sections were formulated, and desired ranges of flow variables to be investigated were established. The majority of tests are expected to be performed on 37-rod bundle test sections. However, limited experimentation is also planned on 19 and 217-rod test sections to assure proper accounting for the bundle size effect.

One of the problems associated with the design of FFTF fuel subassemblies is radial temperature disparities across the

assembly. Radial temperature disparities could arise because the coolant flowing between the pin bundle and flow housing wall receives less heat per unit mass than does that flowing within the bundle. Cross-channel mixing of the coolant tends to reduce this temperature disparity. A computer program, COBRA,⁽¹⁾ is available to analyze the effects of mixing on the temperature distribution in a pin bundle. However, the program requires values of the cross-channel mixing for input. These values are not known, in general, for rod bundles with wire wraps or other spacing devices. Therefore, calculations were made using this code with values of cross-channel mixing as a parameter.

These calculations were made for a 169-pin assembly of 0.250-in. pins, spaced 0.030-in. apart and with a heated length of 36 inches. The spacing between the bundle and flow housing was assumed to be 0.030 inches. A chopped-cosine axial heat flux distribution (1.24 peak to average) and a uniform radial heat flux distribution were assumed. Calculation results show the following three potential problems for this case:

- The radial coolant temperature differences across the outside row of pins will be between 50 and 100 °F for expected values of radial mixing. Temperature differences of this magnitude may cause severe pin bowing.
- The coolant temperature increase at the center of the fuel assembly, about 15% greater than the bulk average increase, will impose a 1.15 hot channel factor on the assembly.

6.7 ANALOG MODEL DEVELOPMENT

For the analog dynamic simulation, the regions within the reactor vessel were modeled as single nodes in parallel and in

(1) D. S. Row. Cross-Flow Mixing Between Parallel Flow Channels During Boiling, Part I, COBRA-Computer Program for Coolant Boiling in Rod Arrays, BNWL-371 Pt. 1, Battelle-Northwest, Richland, Washington, March 1967.

series. As an example, the coolant in the subassembly shroud tubes from core inlet to core outlet is modeled as a single node. A more accurate representation could be obtained by dividing the same volume into several nodes and using a finite difference type of solution to obtain the coolant temperatures at the various node points. This method, however, would require additional analog computing equipment not now available because of the present size of the total FFTF system simulation.

As shown in Figure 6.6, results of computations employing the single node model are comparable to those based on multiple nodes.

FIGURE 6.6. Comparison of 3-Node and Single Node Coolant Outlet Temperature Calculations for Inlet Temperature Step of $200\text{ }^{\circ}\text{F}$

6.8 DIGITAL MODEL DEVELOPMENT

Various methods of simulating the fuel, cladding, and core coolant (in shroud tubes) have been compared against a detailed fuel pin-coolant model solved by means of the transient heat transfer code, TIGER. The TIGER model, including fuel, cladding, and coolant, has 17 axial nodes and 7 radial fuel nodes. The TIGER model, assumed to be the most accurate because of its detail, is therefore used as the reference.

The same fuel, cladding, and coolant volumes, as well as the same density, specific heat, and thermal conductivity data were used in all models.

Two basic methods of solution used in the coolant region are the single node type and the finite difference type. In the single node type, the total coolant volume is divided into several axial nodes in series. Each individual node is treated by itself, and a linear temperature profile through each node is assumed. In the finite difference type of solution, the total volume is divided into several axial nodes and the axial temperature profile $\left[\frac{\partial T}{\partial X}\right]$ at each location of temperature calculation is approximated by finite difference techniques.

Figure 6.7 shows the models that were investigated. One additional model, the same as Model 4A with three radial fuel nodes will be tried. Models 2, 4, 4A, and the TIGER model all use the single node type of simulation in each axial coolant node. The other three models use a finite difference type of solution. Wherever single radial fuel nodes are used, the fuel resistance to heat transfer is calculated as

$$R_{\text{fuel}} = \frac{\ln (r_{\text{fuel}}/0.707r_{\text{fuel}})}{2 \pi k_{\text{fuel}}}$$

The average (radial) fuel temperature in this procedure is always assumed to be located at $0.707r_{fuel}$. All temperatures in the central radial fuel nodes in multinode (radial) models are also assumed to be located at $0.707r_{1F}$, where r_{1F} is the radius of the central fuel nodes (see Figure 6.7).

A chopped cosine axial power level distribution for obtaining the heat generation in each fuel node was assumed. The total heat generation at each axial location was proportioned on a volume basis between the radial nodes if there are more than one. This profile was modified on a steady-state basis to better fit the finite difference approximations in models 1, 3, and 3A.

The following two transients were studied to compare the models:

- (1) Loss of BPA power to all pumps with immediate scram.
Flow decay for primary pump clutch time constant,
 $\tau = 1$ sec. Power decay for 1 G rod acceleration
scram with total reactivity = 15\$.

- (2) 1 g rod acceleration scram with full flow.

These two transients cover a fairly wide range and are assumed to be adequate for a comparison. The models are available and other transients can be investigated if necessary.

Figure 6.8 shows typical results, with the heavy line representing the reference. Core model-1 provides the most accurate results. The differences between the model responses to the second transient are not as great. Core model-1 provides the best results in the first 3-4 sec of the second transient. Good agreement is obtained by all models after 10 sec.

Based on these findings, core model-1 was incorporated in the total reactor model. The reactor model was debugged and checked out at steady-state conditions.

FIGURE 6.7. Core Models Investigated

FIGURE 6.8. Bulk Core Coolant Outlet Temperature for Loss of BPA Power to all Pump Motors with Immediate Scram

6.9. TYPICAL RESULTS OF DIGITAL SIMULATION

With the use of digital simulation, transients within the primary loops and the reactor resulting from BPA power were investigated by delaying reactor scram by 3/4 sec, by use of a primary pump clutch time constant of 1 sec, and with a 1 G rod acceleration scram (total δk - 15\$).

The primary system model, PRIM, and reactor vessel model, REMOD, including the core model-1 described under model development, were used for these analyses. The latest FFTF concept information was included in both models.

Primary coolant flow, pump and motor speed, and temperature data are included in Figures 6.9, 6.10, and 6.11 for a typical transient. The following temperatures were plotted:

- Core Outlet Coolant Temperature: Coolant in fuel duct at the top of the fuel zone
- Shroud Tube Coolant Outlet Temperature: Coolant in fuel duct at the duct outlet (inlet to the outlet mixing plenum)
- Reactor Vessel Coolant Outlet Temperature
- Volume Average Fuel Temperature.

The core outlet and tube outlet temperatures were for an average driver fuel channel in the core. The volume average fuel temperature was for an average fuel pin in an average driver fuel channel in the core.

The flow decay during the transient was too rapid to uniformly remove all of the stored heat within the driver fuel elements. Thus a substantial rise was obtained initially in the core outlet and tube outlet temperatures, with the latter found to be somewhat damped by the axial reflector and fission gas plenums located between the top of the core and the tube outlet. The vessel outlet temperatures in the transient are not affected much because of the large sodium volume in the outlet plenum (1000 ft³ assumed).

FIGURE 6.9. Primary Coolant Weight Flow Rates for Loss of BPA Power with Scram Delayed 3/4 sec. Emergency Power Drives Pony Motors. 1 sec Clutch Time Constant

FIGURE 6.10. Primary Pump and Motor Speeds for Loss of BPA Power with Scram Delayed 3/4 sec. Emergency Power Drives Pony Motors. 1 sec Clutch Time Constant

FIGURE 6.11. Reactor Temperature Responses to Loss of BPA Power Followed by a Scram Delayed 3/4 sec to 1 sec, Clutch Time Constant, 1 G Rod Acceleration Scram (Total Reactivity = 15\$)

6.10 HYBRID MODEL DEVELOPMENT

Work has been performed using an independent all-digital approach as a check on the analog-hybrid model. Figures 6.12 and 6.13 show the results for two types of flow transients as (1) change in excitation current, and (2) loss of motor power. The two approaches compare extremely well, thus strengthening confidence in the analog-hybrid pump simulation.

FIGURE 6.12. Primary Loop Flow Transient for Step Change in Excitation Current from 50 to 10%; Comparison of Hybrid Model with All-Digital Model

FIGURE 6.13. Primary Loop Flow Transient for Loss of Motor Power; Comparison of Hybrid Model with All-Digital Model

6.11 COMPARISON OF MANTA AND COBRA CODES FOR CROSS CHANNEL MIXING

At the present time, there are two operational digital computer codes for analyses of heat transfer and turbulent mixing in pin bundles--MANTA⁽¹⁾ and COBRA.⁽²⁾ Application of these codes to the FFTF fuel rod bundle was investigated.

1. S. F. Armour and D. L. Smith. MANTA - Mixing Analyzed Nodal Thermal-Hydraulic Analyses, GEAP-4805. February 1965.
2. D. S. Rowe. Cross-Flow Mixing Between Parallel Flow Channels During Boiling, Part 1. Cobra - Computer Program for Coolant Boiling in Rod Arrays, BNWL-371, PT1. Pacific Northwest Laboratory, Richland, Washington. March 1967.

MANTA is not directly applicable to large rod bundle problems because of a limited number of connections for coolant nodes.

In order to obtain a model for comparison of MANTA and COBRA, coolant Subchannels 1 through 14 of a 30 degree segment were approximated by the narrow width shown in Figure 6.14. Each coolant node in this approximate model has only four connections, thus circumventing the nodal connection limitations of MANTA. The parameters used for the analysis were:

Pin OD, in.	0.250
Wire Wrap OD, in.	0.030
Gas Plenum Length, in.	36
Active Fuel Length, in.	36
Axial Power Peaking Factor	1.24
Radial Power Peaking Factor	1.0
Average Pin Linear Heating Rate, kW/ft	13.6
Coolant Mass Velocity lb/hr-ft ²	7.62 x 10 ⁶

The geometry of the fission gas plenum is the same as that of the core and does not allow for any increase in coolant flow area by changing the shape of the shroud tube.

FIGURE 6.14. Approximate Model for 30° Segment of 169-Pin Subassembly for Comparison of COBRA and MANTA

Since there are no experimental mixing data for prototypical FFTF subassemblies, the value of the turbulent intensity, β , for use in COBRA must be estimated. It was assumed that a value of 0.01 would be applicable to rod bundles without spacers and that wire wrap spacers could increase β to 0.04. For the wire-wrapped pin bundle under discussion, a value of 0.01 is considered to be conservative and a value of 0.04 is considered to be optimistic.

Computations with COBRA verified the approximate model to be a suitable reference case.

Figure 6.15 shows radial coolant temperature profiles for the approximate model using COBRA for turbulent intensities of 0.01 and 0.04.

Figure 6.16 shows the corresponding MANTA results with and without coolant conduction. MANTA predicts substantially less turbulent mixing than COBRA, even for the intended conservative value of 0.01 for the turbulent intensity.

6.12 COMPUTER CODE DEVELOPMENT

The FORE-II code, a computational program for the analysis of steady-state and transient reactor performance was converted. Previously, the program required two separate runs to obtain a long-edit or a short-edit. The program is now set up to be executed as one program.

6.13 SINGLE TUBE HYDRAULIC BALANCE FEATURE TEST

Application of the hydraulic balance feature to the FFTF fuel shroud ducts is being investigated. This feature consists of a tube arrangement whereby the coolant flow enters and leaves the tube through side entry and exit holes. The lower end of the fuel

FIGURE 6.15. *Mixing Results for Approximate Model Using Cobra Computer Program*

FIGURE 6.16. *Mixing Results for Approximate Model Using Manta Computer Program*

tube is closed. This arrangement results in essentially balanced axial forces on the fuel duct, and thereby reduces the mechanical holddown device requirements. A single-tube full-scale model has been fabricated to evaluate the hydraulic balance concept and to measure the leakage rates between the duct and the upper and lower tubesheets.

To date, testing has been completed with tube-to-tube sheet diametral clearances of 0.005, 0.011, 0.017, 0.025, and 0.040 inches. Results show the tubesheet leakage to be proportional to the differential pressure across the tubesheets, and that the axial force is a downward force proportional to the flow. Leakage rates for the upper and lower tubesheet are shown in Figures 6.17 and 6.18. These data were obtained with a 4-in. thick upper tubesheet and a 5-in. thick lower tubesheet.

The next series of tests planned on this model will evaluate the axial forces on the fuel duct as a function of coolant outlet flow area versus inlet flow area.

FIGURE 6.17. Single Tube Hydraulic Balance

FIGURE 6.18. Single Tube Hydraulic Balance

SECTION 7.0REACTOR MATERIALS DEVELOPMENT7.1 MATERIALS PROPERTIES STUDIES

The objective of this program is to determine the combined effects of environment and fast reactor irradiations on the mechanical properties of candidate fast reactor cladding and structural alloys. The program is specifically directed at providing a basis for the selection and optimum use of engineering materials for FFTF applications and obtaining a description of the material behavior in fast reactor service. Operating limits for cladding and structural alloys will be established as a function of temperature, irradiation exposure and reaction with a sodium environment.

7.1.1 Swelling Tests

The effects of swelling in austenitic stainless steels after irradiation to high neutron fluence are being studied. Density measurements were taken on sections cut from an annealed Type 304 SS EBR-11 irradiation capsule tube. The tube was exposed to 1.3×10^{22} n/cm² total fluence at temperatures from 700 °F to about 800 °F. Density was measured at various distances along the tube and compared with densities taken on 10 sections of a similar tube in the unirradiated condition.

The mean density of the unirradiated section was 7.927 ± 0.003 g/cm³ where the limits given refer to the 95% confidence limit. After irradiation, the density was found to be 7.913 ± 0.004 g/cm³. The difference between these two mean values is 0.014 ± 0.007 . The density decrease may be due to irradiation damage or surface effects on the irradiated samples not present on the control. The specimens were irradiated in flowing sodium for 156 days, while the control has not been exposed to

sodium. The sodium exposure may have changed the surface chemistry sufficiently to cause the observed density change. In addition, the control tube may have slightly different chemistry than the irradiated material.

In order to determine whether decreases in the observed density are real, the irradiated specimens will be annealed to remove the irradiation damage contribution to density damage. The densities before and after annealing will be compared to isolate the irradiation damage component.

Several other irradiation experiments are being conducted in EBR-II. Of these experiments, two appear to be of primary importance with respect to the recent results on void formation and subsequent swelling in structural materials. These experiments are contained in Subassembly X-010 and Subassembly X-018, described in Table 7.I.

TABLE 7.I. *A Description of Experiments Available for Investigation of Swelling in Cladding Materials*

<u>Subassembly Number</u>	<u>Quantity and Pin Type</u>	<u>Reactor Location</u>	<u>Number and Type of Specimen Per Material</u>	<u>Approximate Fluence^(a) 12-8-67</u>	<u>Target Fluence^(a)</u>
X-010	4 Mk A	7F3	18 304 SS Tensile 10 304 SS Creep	2×10^{22}	--
X-018	1 Mk B7	2B1	75 304 SS Tensile ^(b) 69 316 SS Tensile ^(b) 27 321 SS Tensile ^(b) 27 348 SS Tensile ^(b)	1×10^{22}	1×10^{23}

(a) Neutron Energies >0.18 MeV.

(b) Can be used for Tensile or Creep Rupture Tests.

Subassembly X-010, located in Row 7, currently has the highest exposure, $\sim 2 \times 10^{22}$ ($E > 0.18$ MeV) peak fluence.

Subassembly X-018, located in Row 2, has accumulated a fluence of $\sim 1 \times 10^{22}$. However, being in a higher flux position, X-018 accumulates fluence at a greater rate.

7.1.2 Tensile Tests

Tensile samples of 304 SS were examined by transmission electron microscopy. These stainless steel specimens were irradiated to fluences up to 6×10^{21} n/cm², ($E > 1.0$ MeV) in the ETR at 554 °F and tested at temperatures ranging from 68 to 1382 °F. At a fluence of 6×10^{21} n/cm², the irradiation hardening of the material persisted at all test temperatures up to 1382 °F. At lower fluences (1.5×10^{21} and lower), the samples tested at 1382 °F did not show hardening.

The thin foil specimens were obtained from the grip section of the tensile samples tested at 68 °F and 1382 °F. The electron microscopy examination of the grip of the sample tested at 68 °F showed the as-irradiated condition of the material (Figure 7.1). The microstructure was indicative of the low temperature irradiation displacement damage commonly observed and is presently thought to consist of a high concentration of small vacancy and interstitial clusters.

The grip section of the specimen tested at 1382 °F (Figure 7.2) contained a high concentration of bubbles (about 3×10^{15} cm³). Stereo microscopy showed that apparently all of the bubbles in the matrix were attached to dislocations. In a few instances, an isolated bubble could be found for a given foil tilt. In these cases, however, a dislocation could be brought into contrast and would appear to be attached to

1000 Å

Neg 5094-E

FIGURE 7.1. High Concentration of Small Defect Clusters in the As-irradiated Condition of Type 304 SS. 6×10^{21} n/cm² (E > 1 MeV) at 290 °C

1000 Å
|-----|

Neg 5138-F

FIGURE 7.2. Bubbles Located on Dislocations in Matrix of Type 304 SS Irradiated to 6×10^{21} n/cm² at 290 °C and Annealed 1 hr at 750 °C

the bubble. The density of the bubbles could more than account for the high temperature hardening of the material tested at 1382 °F.

An additional sample of the grip section tested at 1382 °F was annealed 1 hr at 1600 °F. The purpose of annealing the sample to higher temperatures was to provide microstructure information useful in understanding the mechanisms of vacancy and interstitial cluster growth and annihilation. The bubble density of the material had decreased to $3 \times 10^{14}/\text{cm}^3$ and the average bubble diameter had increased (Figure 7.3). The bubbles in the matrix were still attached to dislocations.

The high and low angle boundaries of the irradiated 304 SS annealed 1 hr at 1600 °F were examined (Figure 7.4). Stereo microscopy showed over 90% of the bubbles on the high angle boundary to be located in the plane of the boundary. The boundaries do not appear to be sinks for the bubbles. This was indicated because the zone next to the boundaries were not depleted of bubbles. Also the size of the bubbles on the boundaries were not significantly different than the bubbles on dislocations in the matrix.

7.1.3 Irradiation Hardening

High temperature irradiation hardening was also shown by 304 SS, irradiated in the EBR-II to $1.4 \times 10^{22} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) at 1022 °F. The $5 \times 10^{14} \text{ cavities/cm}^3$ were present in the material irradiated at 1022 °F. After annealing, the number of the cavities decreased at temperatures above 1112 °F, and completely disappeared after annealing 1 hr at 1600 °F. The yield stress of the irradiated material tested at temperatures above 1500 °F was similar to control data.⁽¹⁾

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1. Quarterly Progress Report, April, May, June, 1967. Reactor Fuels and Materials Development Programs for Fuels and Materials Branch of USAEC Division of Reactor Development and Technology, BNWL-473. Battelle-Northwest, Richland, Washington, September 1967.

1000 Å

Neg 5222-E

FIGURE 7.3. A Type 304 SS Sample (Shown in Figure 7.2)
After an Additional Anneal of 1 hr at 870 °C

1000 Å
|-----|

Neg 5224-A

FIGURE 7.4. *Typical Grain Boundary of Type 304 SS
Sample (Shown in Figure 7.3)*

7.1.4 Papers Presented

Two papers were presented at the winter meeting of the American Nuclear Society. In summary, the first paper⁽¹⁾ states that tensile properties of AISI-304 stainless steel were determined after irradiation in the EBR-II to 1.7×10^{22} n/cm² total neutron fluence at 1000 °F $\pm$ 100 °F. Tensile tests were performed in vacuum at a strain rate of 0.002/min at temperatures from 70 to 1600 °F. The steel composition was 18.4% Cr, 9.7% Ni, 0.050% C, 0.88% Mn, 0.018% P, 0.010% S, 0.18% Mo, and 14 ppm B.

Postirradiation testing revealed a marked increase in 0.2% offset yield strength at all test temperatures below 1500 °F. Below 1300 °F, the increase was a factor of 2 to 3. At higher temperatures, the change in strength decreases with increasing test temperature and approaches unirradiated values of 1600 °F. No significant changes in ultimate tensile strength, except for an increase at 70 °F, were observed.

The postirradiation ductility was lower than the unirradiated control values for all test temperatures. Below 900 °F, the minimum uniform ductility was about 20%. Above 900 °F, the uniform elongation decreased with temperature to a minimum of 0.5% at 1500 °F. Total elongation follows the same trends but goes through a minimum of 5% at 1300 °F. The decrease in uniform elongation between 900 and 1500 °F corresponds to the temperature range in which irradiation hardening causes the yield strength to approach the ultimate strength.

1. J. J. Holmes and J. E. Irvin. "Effect of Fast-Reactor Irradiation on the Tensile Properties of Type 304 Stainless Steel," *Transactions*, vol. 10, no. 2. p. 487-8. American Nuclear Society, 1967.

The present data are of special importance as they demonstrate irradiation hardening in metals irradiated to high exposures at one-half the absolute melting temperature ($T_m/2$), and stable hardening to $0.67 T_m$. This observation is in marked contrast to low-exposure (10^{21} n/cm^2) results from thermal reactors in which hardening is observed only at test or irradiation temperatures below $T_m/2$. These results also indicate that displacement damage, which leads to radiation hardening, must be considered as a source of high-temperature ductility loss in addition to the helium embrittlement problem.

The second paper,⁽¹⁾ in summary, states that the effect of neutron irradiation on the tensile properties of annealed and 25% cold-worked AISI Type 304 stainless steel has been determined over the test temperature range 23 to 750 °C. Stainless steel tensile specimens in the two conditions were irradiated at 290 °C in the Engineering Test Reactor (ETR) hot-water loop to fluence levels of 1.5×10^{20} , 1.45×10^{21} , and $6.05 \times 10^{21} \text{ n/cm}^2$ ($E > 1 \text{ MeV}$), and tested in vacuum at a strain rate of 0.0024/min.

Irradiation increased both the yield and ultimate tensile strengths of the annealed and cold-worked 304 stainless steel, the increase being more pronounced for the annealed material. After exposure to $1.5 \times 10^{20} \text{ n/cm}^2$, the annealed material has more uniform elongation than the cold-worked steel at all test temperatures investigated. However, the ductility of the annealed material is better only at test temperatures above 400 °C after the intermediate exposure of $1.45 \times 10^{21} \text{ n/cm}^2$. This partial recovery of high-temperature ductility of the

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1. R. E. Robbins, J. J. Holmes and J. E. Irvin, "Post-Irradiation Tensile Properties of Annealed and Cold-Worked AISI-304 Stainless Steel," *Transactions*, vol. 10, no. 2, p. 488-9. American Nuclear Society, 1967.

annealed 304 does not occur at the highest exposure. Although the cold-worked ductility is greater at temperatures below 500 °C, the ductility above 500 °C is identical for the two material conditions.

The ductility advantage of annealed stainless steel over cold-worked material decreases with increasing fluence at 290 °C.

7.1.5 In-Reactor Measurements of Mechanical Properties

The purpose of the in-reactor measurements program is to determine dynamic irradiation effects on the mechanical properties of reactor structural materials. Current effort involves measuring creep in AISI type 304 SS during neutron irradiation.

There is evidence that, in some of the 300 series austenitic SS, a substructure developed at a high stress will result in improved creep resistance at lower stresses and higher test temperatures. Neutron irradiation during the development of the substructure might provide further stabilization of the substructure and be an effective method of strengthening. The in-reactor creep capsule is well suited for this type of study because the substructure during irradiation can be developed under a closely controlled temperature and stress, and the strain can be closely monitored. In a preliminary study, an unirradiated specimen of annealed type 304 SS was pre-crept to 18% under a stress of 50,000 psi at 450 °C, held at 760 °C for 137 hr without stress and subsequently tested at 650 °C under a stress of 20,000 psi. The resulting strain rate was $1.0 \times 10^{-3} \text{ hr}^{-1}$. The corresponding strain rate in a specimen given 18% strain at 650 °C and 20,000 psi was $9.9 \times 10^{-4} \text{ hr}^{-1}$. No significant difference in the creep resistance representing the two pretreatments is apparent.

In-reactor creep tests conducted on type 304 SS have been at strain rates greater than $1 \times 10^{-6} \text{ hr}^{-1}$. Tests are being conducted on type 304 SS at temperatures of 400 and 450 °C and at low stress levels to determine whether greater effects of irradiation on creep rates at low strain rates may possibly occur.

7.1.6 Design Curves for 304 and 316 SS

Design stress curves for unirradiated types 304 and 316 SS, based on ASME Section I criteria, were reviewed upon receipt from Babcock and Wilcox. The major difference between the Code criteria for 304 and 316 SS is that the "60% of the average stress to produce rupture in 100,000 hours" criterion for 304 SS is well above the "80% of the minimum criterion." This difference between the two criteria seems to be the result of the increasing strength of 304 SS from an increase in residual elements such as nitrogen because of scrap remelting practices. Consequently, the minimum rupture data were obtained many years ago and the higher rupture strengths now attainable only result in an increase in the average criterion. The ASME Code Committees are sufficiently confident in the reproducibility of the higher strength 304 SS and so they have recently increased the allowable stresses for 304 SS at elevated temperatures (compare 1965 Section I Code and Summer 1965 Addenda).

This information fortifies a recently reached conclusion that type 304 SS is actually stronger than type 316 SS from 1350 to 1500 °F. This conclusion was based on the same material used by Babcock and Wilcox. Elimination of the "80% of minimum" criterion, now believed justifiable, would enable type 304 SS to appear stronger than type 316 SS for applications above 1250 °F.

7.2 SURVEILLANCE AND IN-SERVICE TESTING

The objective of this program is threefold:

- Surveillance and in-service inspection of critical items directly influencing the reliable and safe operation of FTR and requiring analysis in depth for inclusion in the Safeguards Analyses Reports
- Surveillance and in-service inspection of critical items for early detection of abnormal or unexpected behavior
- Obtaining information on materials and component behavior in a high fast flux, sodium-cooled, reactor system to extend knowledge of benefit to the LMFBR Program.

7.2.1 Monitoring Requirements

A state-of-the-art report recently issued⁽¹⁾ presents the techniques being developed for accurate measurement of flux and fluence. These techniques include use of activation monitors carefully chosen for each particular application, accurate differential neutron cross-sections for the monitor materials, and detailed neutron spectra calculations.

A preliminary analysis of the basic requirements for flux, fluence, and spectral monitoring for the split conical core has been completed. This neutron monitoring will be required in addition to that generally performed by an experimenter. The locations of interest and the types of monitoring are summarized in the following tabulation.

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1. J. L. Jackson and R. E. Dahl. Unpublished Data. Measurement of Neutron Flux in the FFTF Using Activation Monitors. Pacific Northwest Laboratory, Richland, Washington, October 1967. (Preliminary Report)

NEUTRON MONITORING LOCATIONS AND
TYPE OF MONITORING REQUIRED FOR THE FTR

<u>Location</u>	<u>Type of Monitoring*</u>		
	<u>Flux</u>	<u>Fluence</u>	<u>Spectral</u>
Central Loop Area	I	LT	I
Driver Fuel Area	I-ST	LT	I
Reflector	ST-LT	LT	
Removable Core Vessel	I	LT	I
Reactor Vessel	I	LT	I
Guard Vessel & Blast Shield	LT	LT	ST
Bottom Tube Sheet	LT	LT	LT

*LT - Long-Term - monitors in place during all operating cycles
 ST - Short-Term - monitors in place for selected cycles of one cycle duration
 I - Intermittent - monitors in place for part of cycle.

The monitoring information obtained from these locations will provide the basic flux information needed by experimenters to correlate the NPTR with FTR, to test and refine theoretical calculations, and to estimate lifetimes of reactor components.

7.2.2 Radiation Damage for FTR Components

Calculations of radiation damage to FTR components will be based on calculated neutron spectra. The relative spectral shapes for a fast reactor core (EBR-II), a thermal reactor core spectrum (ETR) above 100 ev, and a Watt fission spectrum are compared in Figure 7.5. The curves have not been normalized and do not indicate the relative magnitude of the fluxes.

7.3 FABRICATION DEVELOPMENT

A reliable technique for fabrication of closed loops and driver fuel shroud tubes is being developed and demonstrated.

FIGURE 7.5. Comparison of Core Spectra in a Light Water Moderated Reactor and in EBR-II to the Watt Fission Spectrum

7.3.1 Onsite Fuel Duct Fabrication Development

Five welded cylindrical tubes received from the Carpenter Steel Company have been designated to BW-1 to 5 inclusive.

Hexagonal forming of BW-1 resulted in a tube with the correct flat-to-flat dimensions. However, use of a sand core (mandrel) resulted in excessively large inner corner radii. The corner radii varied from 0.230 in. to 0.250 in., while the required radii were 0.170 in. To correct this condition, a series of three tube-expanding plugs, all having the same flat-to-flat dimension (4.220 in.) but with diminishing corner radii (0.200 in., 0.185 in., and 0.170 in.), were made. After two expansion operations, and with an intermediate vacuum anneal at 1740-1830 °F, the tube met dimensional specifications.

7.3.2 Offsite Fuel Duct Procurement

Carpenter Steel Company is working to produce experimental tube sections of fuel duct configuration to place in the Core Components Test Loop (CCTL) at ANL and in other test facilities. These are called CCTL tubes and the first specimens are now undergoing measurements in the laboratory.

7.3.3 Radiation Effects on Boron Carbide

Boron carbide is the prime candidate for use as an absorber in FTR control rods. However, a recent literature review failed to satisfactorily document its behavior under anticipated FTR operating conditions. As a result of this survey, an irradiation testing program designed to characterize the material's behavior under these conditions was formulated. These irradiations are planned to be performed in the EBR-II.

An initial order of B_4C pellets and powder has been procured, and characterization studies of these materials have been started. Heat transfer calculations have been performed, and irradiation capsule fabrication is in progress.

7.4 NONDESTRUCTIVE TESTING

Inspection of the F-lot fuel cladding tubes for defects was completed. Of the 1051 7-ft long tubes tested, 218 were found with one or more defects of rejectable size. The tubes were rejected solely on the basis of ultrasonic test results. The recently developed ultrasonic test performed better than anticipated, and the 1-mil deep $\times$ 30-mil long standard notch was readily detected. All tubes emitting signals equal or greater than those received from the standard notch were rejected. Destructive analyses of 8 rejected tubes verified the ultrasonic results.

An initial investigation of potential ultrasonic techniques for use in the end closure weld test was conducted.

One technique briefly investigated was the conventional pulse-echo method. Although various types of transducers were used, the required resolution was not obtained. However, this method is not ruled out altogether and will be investigated more fully at a later date.

A test using a send-receive transducer system similar to the one now used in the ultrasonic cladding test was also set up. Preliminary runs on some sample welds showed several indications possibly attributable to cracks or unwelded areas in the weld volume. Subsequent cross-sectioning and inspection revealed nothing of this nature. Although the origin of these indications has not been explained, additional attempts to clarify the situation will be made.

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SECTION 8.0
FUELS DEVELOPMENT

8.1 CHARACTERIZATION OF PuO_2

8.1.1 Preparation of Plutonium Metal for Oxidation Under
Controlled Conditions

The surface of a plutonium button may have a relatively high level of impurities caused by the remnants of slag left from the reduction process. Vacuum melting and recasting will remove the slag and also decrease metallic impurities that may have originally alloyed with the plutonium.

Plutonium dioxide of acceptable purity was made from plutonium buttons vacuum melted in a tantalum (Ta) crucible at 1075 °C. This temperature was held for 5 min before cooling to pouring temperature (950 °C). The molten metal was poured into a clean graphite mold to produce slabs approximately 2 in. × 8 in. × 0.1 in. in size.

Table 8.I shows typical analyses of oxides obtained from recast metal as compared with those prepared by oxidation of the untreated plutonium button.

8.1.2 PuO_2 Obtained from Metal Oxidized Under Controlled
Conditions

Plutonium dioxide prepared by controlled oxidation of vacuum melted, recast plutonium is characterized by small grain size and by relatively high purity. Subsequent calcination of the oxide at 450 to 500 °C produces "sinterable grade" material, while calcination at 900 to 950 °C yields an oxide suitable for blending with UO_2 for densification by pneumatic impaction. Table 8.II shows analyses of the two types.

TABLE 8.I. *Analyses of PuO₂ Obtained from Recast*

Element	Vacuum Cast Metal			Untreated Metal
	"Saxton" PuO ₂ Lot #41-76M	CPD PuO ₂ #25-6-5	CPD PuO ₂ #25-7-1	PuO ₂ #22-9-141 from Untreated Button
F	<2	6	<5	680
Cl	<10	8	<2	1176
Al	2	2	<5	100
Ca	1	5	20	50
Cr	<25	<1	<1	500
Fe	20	<25	<25	530
Mg	<5	<2	<2	2500
Mn	5	<5	<5	25
Na	20	<1	<1	100
Ni	<10	10	1	250
Si	<5	<15	<15	25

TABLE 8.II. *Analyses of Two Types of PuO₂ Powders*

Element	450-500 °C	900-950 °C	
F	<5	<2	<2
Cl	<10	<10	<10
H ₂ O(to 200 °C)	0.55%	(100-200 ppm)	(100-200 ppm)
C	230 ppm	(50-100)	(40-100)
Surface Area (M ² /gram)	12(10)	4	-ND-

The oxidation procedure uses a small stainless steel oven whose walls are warmed by strip heaters (Figure 8.1). Within the oven a 2-compartment basket made of 10-mesh stainless steel screen supports two plutonium strips (approximately 2 × 8 × 0.1 in.) for oxidation. A perforated stainless steel tube,

FIGURE 8.1. *Plutonium Oxidation Furnace*

placed slightly below but parallel to the plutonium strips, serves to introduce an oxidizing atmosphere, H_2O and argon, from separately controllable sources of air.

8.2 THERMAL FLUX IRRADIATIONS OF HYPOSTOICHIOMETRIC UO_2 - PuO_2 MATERIALS

Postirradiation examination of MTR capsule BNW-1-7, prematurely discharged when the lead tube failed, was completed. The thermocouple data indicates this capsule operated at a peak power generation of 12.6 kW/ft. The pellets in this capsule were annular with an O/M of 1.93. A transverse section of the fuel pin was taken at the midpoint of the pin (Figure 8.2). This section indicates that the original 0.006 in. cold diametral gap had decreased to ~ 0.003 in. after 30 hr of operation. The fuel pellet was eccentrically located in the cladding at the time of examination and the slightly eccentric area of columnar grains indicated that it operated this way in-reactor.

8.3 EBR-II CAPSULE TUBE EVALUATION

Metallography was completed on the tubing from two experimental capsules from EBR-II tubing subassembly X011. The metallurgical condition of the tubing was good. No variation, either with temperature or from inner to outer surface on the capsule wall, was indicated. The wall thickness was within the tolerance of the original tubing, thus indicating little or no corrosion took place. The welds appeared sound and no accelerated corrosion in the weld zones was indicated.

FIGURE 8.2. *Macroscopic Cross Section from BNW-1-7. The Cladding Diameter of This Fuel Pin is 0.250 in.*

8.4 EBR-II IRRADIATION TESTS PNL-1 AND PNL-2

Analysis of the performance of two 19 capsule subassemblies (PNL-1 and PNL-2) in accordance with the "Guide for Irradiation Experiments in EBR-II" did not reveal any hazards to the reactor. Table 8.III gives the nominal fuel specimen parameters for these experiments and the calculated specimen heat generation rates and temperatures.

TABLE 8.III. Fuel Specimen Data (Design Values). Specimens in Subassembly PNL-2 are Identical to those in PNL-1

Capsule Designation	Fuel Column Length, in.	Fuel Pellet OD, in.	Fuel Pellet ID, in.	Fuel Pellet Density, %TD	Fuel Smeared Density, %TD	Fuel Weight, g	Fuel-Cladding Diam Gap, in.	Depleted UO ₂ Pellet Diam, in.	Depleted UO ₂ Pellet Length, in. (total)	Depleted UO ₂ Pellet Weight, g	Cladding OD,* in.	Cladding Thickness, in.	Axial Motion Restrictor
PNL 1 - 1	13.5	0.212	-	93	88	80.62	0.006	0.214	1.0	5.8	0.250	0.016	no
- 2	13.5	0.212	-	93	88	80.62	0.006	0.214	1.0	5.8	0.250	0.016	no
- 3	13.5	0.212	-	93	88	80.62	0.006	0.214	1.0	5.8	0.250	0.016	no
- 4	13.5	0.212	-	93	88	79.12	0.006	0.214	1.0	5.8	0.250	0.016	yes
- 5	13.5	0.212	-	93	88	79.12	0.006	0.214	1.0	5.8	0.250	0.016	yes
- 6	13.5	0.212	-	93	88	79.12	0.006	0.214	1.0	5.8	0.250	0.016	yes
- 7	13.5	0.212	0.054	96	85	77.87	0.006	0.214	1.0	5.8	0.250	0.016	no
- 8	13.5	0.212	0.054	96	85	77.87	0.006	0.214	1.0	5.8	0.250	0.016	no
- 9	13.5	0.212	0.054	96	85	77.87	0.006	0.214	1.0	5.8	0.250	0.016	no
-10	13.5	0.212	0.054	96	85	76.43	0.006	0.214	1.0	5.8	0.250	0.016	yes
-11	13.5	0.212	0.054	96	85	76.43	0.006	0.214	1.0	5.8	0.250	0.016	yes
-12	13.5	0.212	0.054	96	85	76.43	0.006	0.214	1.0	5.8	0.250	0.016	yes
-13	13.5	0.212	-	93	88	80.62	0.006	0.214	1.0	5.8	0.238	0.010	no
-14	13.5	0.212	-	93	88	80.62	0.006	0.214	1.0	5.8	0.238	0.010	no
-15	13.5	0.212	0.054	96	85	77.87	0.006	0.214	1.0	5.8	0.238	0.010	no
-16	13.5	0.212	0.054	96	85	77.87	0.006	0.214	1.0	5.8	0.238	0.010	no
-17	13.5	0.212	-	89.9	85	77.87	0.006	0.214	1.0	5.8	0.238	0.010	no
-18	13.5	0.212	-	89.9	85	77.87	0.006	0.214	1.0	5.8	0.238	0.010	no
-19	13.5	0.210	-	91.6	85	77.87	0.008	0.214	1.0	5.8	0.238	0.010	no

* Cladding material is type 304 SS

Theoretical density of 25% PuO₂ - 75% UO₂ is 11.0925 g/cm³

8.5 FAST FLUX IRRADIATION DATA REVIEW

A review of existing UK and US fast flux mixed oxide fuel irradiation data and irradiations in progress and planned was completed. Table 8.IV summarizes existing data and the anticipated results from the planned PNL unencapsulated specimen irradiations in EBR-II (PNL-3, -4, -5). Of particular note are the fuel specimen fuel burnup/cladding fluence ratios as compared to those for the FTR driver fuel. It is obvious that none of the existing data approach the FTR conditions. Even in the planned unenriched fuel (PNL-3) test, a difference factor of about two exists between the ratios for the anticipated EBR-II test results and the average FTR driver fuel. Moreover, a difference factor of three was noted between the ratios for the highest cladding fluence UK data and the average FTR fuel pin. From the foregoing, it is apparent that the existing data and, to only a smaller extent, the data yet to be obtained are not directly applicable to the FTR driver fuel design.

8.6 RESTRICTOR SCREENING CAPSULES IRRADIATIONS

Irradiation tests were run under molten fuel conditions to compare the use of metal and ceramic restrictors. In all, cases, the metal restrictors appear to have inhibited axial fuel motion regardless of restrictor material or geometry. Gross fuel motion appears to be restricted in different ways by the two types of restrictors, annular and grooved. With the annular restrictor, the molten fuel appears to run into the central hole where it is quenched rapidly and forms a solid plug to prevent further downward motion of the fuel. The grooved restrictors quench the fuel above and below the restrictor and in no case has evidence of fuel motion into the grooves been observed. Figure 8.3 shows the two types of restrictors.

TABLE 8.IV. *Effect of Fuel Composition and Enrichment
on Fuel Burnup/Cladding Fluence Ratio*

Subassembly Designation	Reactor	Peak Flux, n/cm ² sec x 10 ⁻¹⁵	Fuel Composition	Peak Fuel Specific Power, W/g	Peak Burnup MWd/tonne	Peak Cladding Fluence n/cm ² x 10 ⁻²²	Fuel Burnup/ Cladding Fluence Ratio x 10 ⁻¹⁹	Cladding Fluence @ 82,500 MWd/tonne Fuel Burnup n/cm ² x 10 ⁻²²
XG06	EBR-II	1.95	(0.2 Pu ^{0.93E} 0.8U)O ₂	226	56,000	3.34	14.98	5.51
XO11	EBR-II	2.17	[(0.2 Pu ^{0.93E} 0.8U)O ₂ (0.3 Pu ^{0.93E} 0.8U)O ₂]	241	33,900	2.51	13.5	6.1
MK I	DFR	2.5	(0.15 Pu ^{0.6E} 0.85U)O ₂	178	47,000	5.32	8.85	9.34
MK IB	DFR	2.5	(0.15 Pu ^{0.6E} 0.85U)O ₂	177	58,700	7.1	8.27	10.05
MK II	DFR	2.5	(0.25 Pu ^{0.75E} 0.75U)O ₂	217	59,750	5.95	10.02	8.22
PNL-3	EBR-II	~2.07	(0.25 Pu ^N 0.75U)O ₂	84	Goal is 56,300	10.65	5.29	17.6
PNL-4	EBR-II	~1.93	(0.25 Pu ^{0.45E} 0.75U)O ₂	140	Goal is 56,300	5.95	9.46	9.83
PNL-5	EBR-II	~1.93	(0.25 Pu 0.75 ^{0.93E} U)O ₂	224	Goal is 56,300	3.72	15.12	6.14
FTR Driver	FTR	~6	~(0.25P ^N 0.75 U)O ₂	202	82,500	21.3	3.87	21.3
FTR Average	FTR	4.61 Average	~(0.25P ^N 0.75 U)O ₂	109.5 Average	45,000 Average	16.4 Average	2.76 Average	---

8.8

BNWL-567

FIGURE 8.3. *Macrophotographs of Typical Capsules Containing Annular and Grooved Metallic Restrictors*

The ceramic restrictors failed, in virtually all instances, to prevent the axial movement of the fuel. Figure 8.4 shows longitudinal sections of capsules BNW-1-33, -34, and -35 containing ZrO_2 , ThO_2 , and $^{\text{D}}\text{UO}_2$ restrictors, respectively.

Both the ZrO_2 restrictors reacted with the fuel and allowed it to slump. The $^{\text{D}}\text{UO}_2$ restrictor did not allow fuel motion, but considerable reaction occurred on both the top and bottom of the UO_2 pellet.

8.7 FFTF FUEL ANALYSIS PROBLEMS

Pu and U fluorescence intensities were measured from a 20 wt% PuO_2 - UO_2 fuel pellet through a 1/16 in. slit. The aperture exposed an area of the pellet approximately 1/16 $\times$ 1/8 in. A statistical analysis of the intensity measurements is summarized in the following tabulation. Peak to background ratio was 20.

	Counts/sec	2 σ Precision		Total Counts	Time, Minutes
		Relative %	Absolute %		
20 wt% PuO_2	77	± 1	± 0.2	45,000	9.7
		± 2	± 0.4	11,200	2.4
		± 4	± 0.8	2,800	0.6
80 wt% UO_2	290	± 0.67	± 0.53	98,000	5.6
		± 1	± 0.8	45,000	2.6

This information established what could be done by this X-ray method to nondestructively determine homogeneity of these very small fuel pellets. Homogeneity data probably will be desired on a more nearly micro scale. Consequently, autoradiography or electrical resistance measurements with a microprobe appear to be the most promising of the analytical techniques.

BNW-1-32
ZrO₂ Restrictors

BNW-1-34
ThO₂ Restrictors

BNW-1-35
^DUO₂ Restrictors

FIGURE 8.4. Macrographs of Longitudinal Sections from Capsules Which Contained ZrO₂, ThO₂, and ^DUO₂ Restrictors

8.8 RELATED STUDIES⁽¹⁾

8.8.1 Measurement of Melting Heat Ratings for UO_2 - PuO_2

Analyses of microstructures and fission fragment and plutonium distributions in irradiated UO_2 -25 wt% PuO_2 fuel pins with compositions of O/M = 2.00, 1.99, 1.97, and 1.94 showed all hypostoichiometric specimens to have slightly lower melting heat ratings than did the stoichiometric standards. The heat rating required for melting decreased 3.8 and 5.5%, respectively, for O/M values of 1.99 and 1.90 relative to 2.00. Results, based on measurements of a decrease in thermal conductivity and an increase in melting points as O/M decreases below 2.00, agreed with expectations.

8.8.2 Oxygen Redistribution at Heat Ratings Slightly Below Melting

Redistribution of oxygen or uranium in nonstoichiometric UO_2 fuels during irradiation with center temperatures near but below melting is under study. Previously irradiated specimens with O:U's of 1.90, 2.00, 2.02, 2.05, and 2.15 were cross-sectioned, and macrographs and γ - β autoradiographs were made available for detailed study. Sections were cut from each specimen for radial core drilling and subsequent O:U analyses. Attempts to remove cores by using diamond drills and techniques developed previously for molten-center fuels were not successful. The drillings were finely divided and the entire core was lost in the flowing drill coolant. Apparently, oxides exposed to high temperatures become stronger and

(1) *Although not financially supported by FFTF Project, these studies by Materials Department provide information needed for design and performance analyses.*

less friable. Pressure impregnation of the specimens with epoxy resin may solve the problem.

8.8.3 Fuel-Cladding Compatibility

Four short-term (120 hr) sodium-bonded, Type 304 SS cladding compatibility capsules containing UO_2 - 25% PuO_2 were heated at 650 and 1000 °C. No evidence of reactions or incompatibility between the oxide-sodium bond or the sodium bond-stainless steel interfaces was observed.

8.9 PURIFIED PLUTONIUM FOR EMISSION SPECTROSCOPY STANDARDS

Twenty-two grams of purified plutonium dioxide was recently prepared for use in preparing standards to calibrate the emission spectrographic analysis of impurities in UO_2 - PuO_2 and PuO_2 . Of 30 elements checked for in the purified oxide, only five contaminants were detected. They were copper (0.1 ppm), magnesium (0.5 ppm), calcium (5 ppm), sodium (5 ppm), and aluminum (1 ppm). Calibration curves now used are made from purified uranium (no plutonium). When plutonium is present, the matrix change may cause errors in the analyses. Now, with the purified plutonium dioxide, the reliability of the emission spectrographic analysis of UO_2 - PuO_2 can be evaluated.

8.10 CLADDING SWELLING

The UKAEA recently reported observations of volume increases in M316L cladding caused by DFR fast fluence,⁽¹⁾ apparently as the result of vacancy clustering. To understand the relevance of cladding swelling to fuel pin diameter increase in DFR, appropriate UK documents are under review. In addition, PNL's

1. Nature, vol. 216, pp. 575-6, November 11, 1967.

Metallurgy Research Section is attempting to observe similar phenomena. Plans are underway to obtain a control rod thimble from EBR-II which has an appropriate fluence and temperature history. This material has the highest known fast fluence in the US. PNL's Radiation Damage Unit has begun work to obtain flux and fluence correlation between DFR, EBR-II, and FTR.

8.11 GAS COOLING OF FFTF FUEL ASSEMBLIES DURING INSPECTION

The objectives of this program are to perform laboratory tests which will:

- Establish the feasibility of decay heat removal from FFTF fuel assemblies by the forced circulation of an inert gas stream during fuel inspection and disassembly.
- Establish thermal hydraulic criteria for the design and operation of fuel assembly decay heat removal systems.
- Obtain data necessary to permit the establishment of operating limits for the specific heat removal system developed for the FFTF.

Preparation for the first tests to demonstrate the feasibility of decay heat removal from FFTF fuel assemblies with inert recirculated gas systems is proceeding. All individual components, including hexagonal liner tubes, have been fabricated.

Installation of the Gas Heat Transfer Facility, to be used in all phases of the test program, is 40 percent complete.

SECTION 9.0PHYSICS9.1 CRITICAL EXPERIMENTS9.1.1 Rod Worth

The Phase A experiments, conducted as part of the central rod worth studies, included measurements of the worth of an enriched, distributed rod versus length of insertion axially through the center of the ZPR-III Assembly 48. The rod worth was determined at six steps of partial insertion and at full insertion.

The stepped-insertion values were calculated with the 2-D transport code 2DF, in R-Z geometry. The enriched B₄C rod was cylindricized and homogenized. Specifically in the case of the last four steps of insertion, the degree of agreement with experiment is consistent with all other homogenized central rod calculations previously made in the analysis of the Phase A experiments. Table 9.I presents a comparison of experimental and calculated values.

TABLE 9.I. *Worth Versus Length of Insertion of
Central Enriched Control Rod*

<u>Rod Insertion, in.</u>		<u>Exp. % $\Delta k/k$</u>	<u>Calc. % $\Delta k/k$</u>	<u>E/C</u>
Blanket	8	0.003	0.002	1.5
	12	0.027	0.027	1.0
Core	19	0.287	0.225	1.28
	27	0.892	1.056	0.84
	35	1.483	1.777	0.83
	42	1.719	2.042	0.84
Blanket	54	1.738	2.061	0.84

9.1.2 Reactivity Worth

Comparisons of calculated and experimental reactivity worths and reaction rates provide integral tests of cross section sets and theoretical models used in reactor design. Initial comparisons of this nature for ^{239}Pu , enriched uranium, depleted uranium, ^{10}B , and stainless steel measurements were made in the FTR Phase A Critical Experiments performed in ZPR-III Assemblies 48 and 48A.

The calculated worths and reaction rates were computed using the small sample perturbation code, "PERT IV".⁽¹⁾ Flux and adjoint inputs for "PERT IV" were calculated using a "2DF" quarter core, x-y geometry model.

These comparisons illustrated an unresolved difficulty in obtaining agreement between measured and calculated worths of small fissile samples. Further, more detailed spatial dependence of cross section sets is required to describe spatial worths and reaction rates near core-reflector boundaries and in reflector media.

Calculated Pu sample worths are higher than experimental worths. Calculated depleted uranium worths, also higher than the experimental worths, is thought to result from substantial uncertainties in the fission and absorption cross sections for ^{283}U . These cross sections are being revised. Worths or reaction rates having large thermal components tend to exhibit large discrepancies between calculation and experiments. It appears that neither core nor Inconel average cross sections are appropriate for these reflector calculations.

1. R. W. Hardie and W. W. Little, Jr. PERT-IV, A Two-Dimensional Perturbation Code in FORTRAN-IV. BNWL-409.

Low energy components of the neutron spectrum, nearly negligible in the core and blanket materials, are quite significant in the reflector and are apparently not well reproduced by calculation.

9.1.3 Reflector Control Rod Worth Calculations

A half core and a full core 2DF Model were employed in calculating the worths of the asymmetric configurations, including possible interaction across the core. Figure 9.1 illustrates the actual position of the reflector control rods and the in-core ^{239}Pu enriched safety and control rods. It will be noted that the safety rods are not symmetric with respect to the x-axis.

Table 9.II summarizes all the reflector control rod worth experiments and calculations involving rods enriched in ^{10}B . The calculated numbers do not include any correction for axial leakage nor for ^{10}B cross section revisions. These are nearly compensating effects; each is expected to be worth about $5\% \frac{\Delta k}{k}$. These corrections are being investigated.

Table 9.II. Experiment-Theory Comparisons

Case	Rod Location	Exp (E) % $\Delta k/k$	Calc (C) % $\Delta k/k$	E/C	Core Size
1	L10	0.392	0.411	0.954	Full
2	T10	0.382	0.414	0.923	Full
3	N9	0.350	0.394	0.888	Full
4	R9	0.350	0.394	0.888	Full
5	P9 or P23	0.417	0.448	0.931	Full
6	P9 or P23	0.412	0.427	0.965	1/2
7	P9 + P23	0.821	0.831	0.988	1/4
8	N9 + R9	0.674	0.703	0.959	1/2
9	N9 + P23 + R9	1.092	1.134	0.963	1/2
10	N9 + P9 + R9	0.939	0.966	0.972	1/2
11	L10 + T10	0.754	0.754	1.000	1/2
12	L10 + P23 + T10	1.188	1.183	1.004	1/2
13	L10 + P9 + T10	1.097	1.101	0.996	1/2

FIGURE 9.1. Assembly 48A

9.1.4 Critical Experiments Program: Phase B

The Phase B experimental program at the ZPR-III, ANL Idaho, was started. Modifications to the matrix and preparations for loading of Assembly I of the Phase B Criticals, designated Assembly 51 of the ZPR-III, began in October. A description of Assembly I Phase B experiments are listed as follows:

- Core - right circular cylinder core $H \approx 86$ cm, $R \approx 35$ cm
- Side and end reflectors $\Delta t = 30$ cm
- Core volume ≈ 322 liters
- Core composition, volume percent:
 - Fuel ≈ 36
 - Stainless Steel ≈ 25
 - Sodium ≈ 37
 - Void ≈ 3
- Critical Mass $^{239}\text{Pu} \approx 220$ kg
- Radial Reflector Composition, volume percent:
 - Nickel ≈ 80
 - Na ≈ 20

The final core drawer loadings for Assembly 51, shown in Figure 9.2, were chosen according to the following guidelines:

- (1) The platelets should be arranged to give an even distribution of fuel.
- (2) Plutonium and depleted uranium should be adjacent to each other.
- (3) Fuel is not to be at a drawer boundary.

9.1.5 Effect of "Group Collapsing" Schemes on Neutronic Calculations

In order to conserve computer time, multigroup calculations are often done with a small number of energy groups. The effect of using few-group cross sections in calculating the effective multiplication factor, k_{eff} , and the central

FIGURE 9.2. Core Drawer Loading for Assembly 51-FPR-3

worth of ^{239}Pu for ZPR-III Assembly 48 was investigated by means of DTF-IV (spherical geometry). Cross section input consisted of a modified version of the Russian 26 group data compilation punched by the FCC-IV fundamental mode code. Calculations were performed using 1, 2, 5, 10, 15, 20, and 26 groups. The few group cross sections were obtained by averaging the 26 group data.

Figure 9.3 indicates the dependence of k_{eff} on the number of groups. The discrepancy from the "correct" value (the 26 group calculation) decreases quite sharply up to 5 groups, and then levels off.

9.2 ANALYSIS OF SODIUM SHIELD ATTENUATION

The purpose of these studies is to establish a valid method for evaluating FTR shield concepts, and for establishing the FTR shield design.

A number of activation traverses were made through sodium to provide a basis for evaluating calculational techniques and cross section data.⁽¹⁾

The computer program ANISN was used to calculate the spectral and spatial distribution of the neutron flux from a point fission source using spherical geometry. A homogeneous sodium-aluminum shield mixture was used in the calculations to provide the same atomic composition as that employed in the experiment. A computer program was written to integrate the point source results over a disc source distribution to approximate the experimental geometry. Results of the calculations are shown in Figure 9.4, 9.5, and 9.6 and are compared with the experimental results. Relatively

1. A. W. Thiele. Neutron and Gamma Ray Attenuation in Sodium Shields, NAA-SR-MEMO-12467, June 30, 1967.

FIGURE 9.3. Dependence of k_{eff} on Number of Energy Groups for Assembly 48

FIGURE 9.4. Threshold Activation Versus Sodium Thickness

FIGURE 9.5. Resonance Activation Versus Sodium Thickness

FIGURE 9.6. Resonance Activation Versus Sodium Thickness

good agreement between theory and experiments was achieved for the threshold distributions while fair agreement was achieved for the distribution of resonance activities.

9.3 FEEDBACK EFFECTS UPON NEUTRON NOISE

Before neutron noise measurements can be used to predict anomalies during at-power operation, the behavior of at-power neutron noise as it routinely exists in the normally operating reactor must be described and understood.

A new two-temperature coefficient model for feedback neutron noise was derived. This model has now been programmed and calculations were made for comparison to an available experiment.

Differing mostly in complex detail rather than form, the equations for the two-coefficient model are generally quite similar to those of the one-coefficient model. A key point obtained from this study was that the resulting neutron noise spectral density function could be viewed as the sum of several contributions.

Figure 9.7 shows two of these contributions for the two-coefficient model. The lower curve represents the contribution to the spectral density produced by the natural statistical fluctuations in the neutron population in a reactor system having feedback effects, and it differs considerably from the no-feedback result. In the "conventional" theory, one normalizes upward a curve of this form in an attempt to fit theory to experiment. The second curve also includes the contribution to the neutron fluctuations produced by moderator temperature fluctuations. A third contribution will come from fuel temperature fluctuation, which is not shown. The basic theoretical form for the gain constant appropriate for these contributions is still not developed and the normalization shown is arbitrary.

FIGURE 9.7. *Neutron Fluctuation Spectral Density Function Including Feedback*

The basic shape of the upper curve, however, is much closer to the shape observed in the experiment than the "conventional" approach.

9.4 DECAY HEATING

9.4.1 Driver Fuel Decay Heat

Decay heat from irradiated FTR driver fuel was estimated by using the computer program RIBD. Input parameters for three RIBD calculations were defined as follows:

- (1) "Best estimate": 5.13 MW/channel for 332 operating days
- (2) Upper limit: 5.72 MW/channel for 600 operating days
- (3) Lower limit: 4.55 MW/channel for 158 operating days

Typical results are given in Figure 9.8.

9.4.2 Decay Heat from Test Assembly

Decay heat from plutonium fuel test assemblies irradiated in the FTR was estimated by using the computer code RIBD. Figure 9.9 presents results of these calculations for a number of assumed conditions. The present concept for a closed loop system provides a heat removal capacity of 6 MW. Therefore, curve A in Figure 9.9 constitutes a realistic upper limit for decay heat. The other curves indicate relative importance of the test assembly operating power and the irradiation time on the decay heat.

9.4.3 Decay Heat from Irradiated FTR Fuel

Calculations were made to estimate the amount of decay heat deposited within the assembly of fuel pins contained in a driver assembly as a function of time after shutdown. Figure 9.10 presents the results for a typical "average" driver fuel assembly irradiated to a goal exposure of 45 MWd/kg. The

FIGURE 9.8. Decay Heat from FTR Fuel Element as a Function of Time -
"Best Estimate"

FIGURE 9.9. Decay Heat from Closed Loop Test Elements

FIGURE 9.10. Decay Heat from FTR Fuel (5 MW Operating Power, 333 Day Irradiation)

results can be scaled directly by a factor of 1.22 to estimate heat generation from the highest power assembly in the FTR, and by 0.82 to estimate heat generation from the lowest power assembly. The curves are applicable directly over a wide range of exposures; it is estimated that a factor of 0.9 needs to be applied for an exposure of 15 MWd/kg, whereas a factor of 1.1 approximates an exposure of 75 MWd/kg.

9.5 ENERGY DEPOSITION IN FTR

A preliminary analysis was made to estimate the distribution of the energy deposited in the FTR environs as a result of the fission process to provide a basis for the conceptual design of fuel. The 1000 liter core was assumed to consist of mixed oxide fuel encased in stainless steel with sodium as the coolant. The actual deposition distribution will be dependent on the fuel pin geometry. Therefore, this preliminary estimate can define only the gross features of the deposition distribution, with later refinements to be made as the fuel geometry becomes established. Table 9.III summarizes the results.

TABLE 9.III. *Energy Deposition Distribution in the FTR*

<u>Source of Energy</u>	<u>Fuel</u>	<u>Steel</u>	<u>Sodium</u>	<u>Total</u>	<u>Outside Core</u>	<u>System Total</u>
Fission Fragments	174.0	0	0	174.0	0	174.0
Decay Betas	6.3	0.7	0	7.0	0	7.0
Neutron Kinetic Energy	2.4	1.05	1.05	4.5	1.0	5.5
Gammas (Prompt, Decay)	7.8	5.0	0.7	13.5	1.5	15.0
Neutron Capture Energy	<u>3.6</u>	<u>2.4</u>	<u>0.3</u>	<u>6.3</u>	<u>7.7</u>	<u>14.0</u>
MeV/Fission	194.1	9.15	2.05	205.3	10.2	215.5
Percent of Energy	90.1	4.25	0.95	95.3	4.7	100.0
Total MW	360.4	17.0	3.8	381.2	18.8	400.0

9.6 RADIOACTIVITY PROBLEMS ASSOCIATED WITH PLUTONIUM FUEL

Radioactivity can influence the design of facilities in which unirradiated fuel for the FTR and NPTF will be fabricated, handled, and stored. In addition to radiation exposure received by operators, heat generation is also an important consideration, especially in the design of the NPTF. Half life and decay energy data sufficiently accurate to estimate heat generation as a function of isotopic composition are known. However, data for estimating the radiation environment are not so well established.

Table 9.IV shows heat generation for each of the isotopes possibly present in the FTR fuel. All these values are associated with the decay process.

TABLE 9.IV. *Heat Generation of Fuel Isotopes*

<u>Isotope</u>	<u>w/g</u>
$^{237}\text{U}^*$	$1.4 \cdot 10^{-5}$
^{238}U	$8.2 \cdot 10^{-10}$
^{238}Pu	$5.5 \cdot 10^{-1}$
^{239}Pu	$1.8 \cdot 10^{-3}$
^{240}Pu	$6.9 \cdot 10^{-3}$
^{241}Pu	$2.7 \cdot 10^{-2}$
^{242}Pu	$1.1 \cdot 10^{-4}$
$^{241}\text{Am}^{(a)}$	$1.06 \cdot 10^{-1} \text{f(t)}$

(a) Heat generation per gram of ^{241}Pu .

Heat generation from decay will be about 2.2 kW for a core containing about 2100 kg of uranium, 600 kg of ^{239}Pu and ^{241}Pu , and 100 kg of other plutonium isotopes. Heat generation for any quantity and isotopic composition of fuel can be estimated from the values in Table 9.IV.

The neutron source strength associated with any assemblage of material can be obtained by summing the neutron production rates from spontaneous fission and from the (α, n) reaction in oxide fuel. Table 9.V presents the contribution from each of the isotopes possibly present in FTR fuel.

TABLE 9.V. *Neutron Production in Oxide Fuel*

<u>Isotope</u>	<u>n/g-sec</u>
$^{237}\text{U}^*$	0
^{238}U	$1.4 \cdot 10^{-2}$
^{238}Pu	$1.64 \cdot 10^4$
^{239}Pu	$4.6 \cdot 10^1$
^{240}Pu	$1.17 \cdot 10^3$
^{241}Pu	3.4
^{242}Pu	$1.8 \cdot 10^3$
$^{241}\text{Am (a)}$	$2.5 \cdot 10^3 f(t)$

(a) Neutron generation rate per gram of ^{241}Pu , where $f(t) = [1.0 - \exp(-0.053t)]$ for $0 \leq t \leq 3$ years.

The neutron exposure rate from a particular assemblage of material can be estimated from the source strength and geometric considerations. For large arrays of material, attenuation and multiplication also must be considered.

Gamma radiation is emitted from fuel isotopes as a result of both decay and spontaneous fission. A series of ISOSHL D calculations were made to estimate the gamma intensity in the vicinity of a reference fuel pin 85 cm long, 0.267 cm in radius, clad in SS 0.041 cm thick, and with a continued fuel density of

10/0 g/cm³. Table 9.VI gives gamma intensity at the surface and 2 ft from a reference pin for the isotopes of interest based on ISOSHL D calculations and assumed decay spectra.

TABLE 9.VI. *Gamma Intensity Near Reference Pin*

<u>Isotope</u>	mR/hr per g of isotope	
	<u>Surface</u>	<u>@ 2 ft</u>
²³⁷ U (a)	67.7	1.5 10 ⁻¹
²³⁸ U	1.1 10 ⁻⁶	2.2 10 ⁻⁹
²³⁸ Pu	26.2	6.2 10 ⁻²
²³⁹ Pu	0.268	6.1 10 ⁻⁴
²⁴⁰ Pu	0.12	2.4 10 ⁻⁴
²⁴¹ Pu	0.0002	5.3 10 ⁻⁷
²⁴² Pu	0.016	3.2 10 ⁻⁵
²⁴¹ Am (a)	2430. f(t)	5.5 f(t)

(a) Per gram of ²⁴¹Pu; f(t) as above.

9.7 INVENTORY OF FISSION PRODUCTS IN FTR

The computer code RIBD was used to make a parametric study of the fission product inventory in FTR fuel as a function of irradiation and decay time. RIBD is a general purpose, multiple-path, multiple-chain, grid processor. With the present library, this grid processor will calculate the concentration of each of the 450 isotopes or isomeric states that can be formed directly from the fission process or as the result of possible combinations of neutron captures and/or decay processes from other fission products. Accuracy of the calculation is limited primarily by the availability of nuclear data for the many isotopes of interest.

For this study, continuous irradiation times of 10⁶, 2 × 10⁶, 10⁷ and 10⁸ sec were used, while decay times were selected for up to three years. Table 9.VII gives a brief

TABLE 9.VII. *Curies of Gaseous Fission Products in FTR as a Function of Irradiation and Decay Time*

10^6 sec - Irradiation Time							
	0	1 min	1 hr	1 day	10 days	100 days	1 year
Noble Gas	$1.9 \cdot 10^8$	$1.1 \cdot 10^8$	$5.6 \cdot 10^7$	$2.6 \cdot 10^7$	$5.7 \cdot 10^6$	$1.9 \cdot 10^3$	$1.3 \cdot 10^3$
Halogens	$1.5 \cdot 10^8$	$1.1 \cdot 10^8$	$8.4 \cdot 10^7$	$3.0 \cdot 10^7$	$5.6 \cdot 10^6$	$1.5 \cdot 10^3$	$9.2 \cdot 10^{-3}$
Volatile Solids	$2.0 \cdot 10^8$	$1.4 \cdot 10^8$	$5.6 \cdot 10^7$	$1.8 \cdot 10^7$	$3.3 \cdot 10^6$	$1.9 \cdot 10^5$	$2.5 \cdot 10^4$
All Remaining Fission Products	$1.1 \cdot 10^9$	$8.3 \cdot 10^8$	$3.6 \cdot 10^8$	$1.4 \cdot 10^8$	$3.8 \cdot 10^7$	$5.6 \cdot 10^6$	$8.9 \cdot 10^5$
2×10^6 sec							
	0	1 min	1 hr	1 day	10 days	100 days	1 year
Noble Gas	$2.0 \cdot 10^8$	$1.2 \cdot 10^8$	$6.4 \cdot 10^7$	$3.1 \cdot 10^7$	$7.2 \cdot 10^6$	$3.5 \cdot 10^3$	$2.6 \cdot 10^3$
Halogens	$1.6 \cdot 10^8$	$1.1 \cdot 10^8$	$8.9 \cdot 10^7$	$3.5 \cdot 10^7$	$7.2 \cdot 10^6$	$2.1 \cdot 10^3$	$1.9 \cdot 10^{-2}$
Volatile Solids	$2.1 \cdot 10^8$	$1.4 \cdot 10^8$	$6.0 \cdot 10^7$	$2.0 \cdot 10^7$	$4.4 \cdot 10^6$	$3.5 \cdot 10^5$	$4.9 \cdot 10^4$
All Remaining Fission Products	$1.1 \cdot 10^9$	$8.8 \cdot 10^8$	$3.9 \cdot 10^8$	$1.7 \cdot 10^8$	$6.0 \cdot 10^7$	$1.0 \cdot 10^7$	$1.7 \cdot 10^6$
10^7 sec							
	0	1 min	1 hr	1 day	10 days	100 days	1 year
Noble Gas	$2.0 \cdot 10^8$	$1.2 \cdot 10^8$	$6.6 \cdot 10^7$	$3.4 \cdot 10^7$	$7.6 \cdot 10^6$	$1.5 \cdot 10^4$	$1.3 \cdot 10^4$
Halogens	$1.6 \cdot 10^8$	$1.1 \cdot 10^8$	$9.2 \cdot 10^7$	$3.7 \cdot 10^7$	$7.6 \cdot 10^6$	$2.4 \cdot 10^3$	$9.1 \cdot 10^{-2}$
Volatile Solids	$2.1 \cdot 10^8$	$1.4 \cdot 10^8$	$6.2 \cdot 10^7$	$2.4 \cdot 10^7$	$7.6 \cdot 10^6$	$1.1 \cdot 10^6$	$2.6 \cdot 10^5$
All Remaining Fission Products	$1.2 \cdot 10^9$	$9.6 \cdot 10^8$	$4.9 \cdot 10^8$	$2.6 \cdot 10^8$	$1.3 \cdot 10^8$	$3.5 \cdot 10^7$	$7.4 \cdot 10^6$
10^8 sec							
	0	1 min	1 hr	1 day	10 days	100 days	1 year
Noble Gas	$2.4 \cdot 10^8$	$1.6 \cdot 10^8$	$9.1 \cdot 10^7$	$4.8 \cdot 10^7$	$1.2 \cdot 10^7$	$1.5 \cdot 10^5$	$1.4 \cdot 10^5$
Halogens	$1.6 \cdot 10^8$	$1.2 \cdot 10^8$	$9.6 \cdot 10^7$	$4.0 \cdot 10^7$	$8.8 \cdot 10^6$	$2.9 \cdot 10^3$	$8.0 \cdot 10^{-1}$
Volatile Solids	$2.5 \cdot 10^8$	$1.7 \cdot 10^8$	$8.5 \cdot 10^7$	$4.2 \cdot 10^7$	$2.0 \cdot 10^7$	$6.0 \cdot 10^6$	$4.2 \cdot 10^6$
All Remaining Fission Products	$1.4 \cdot 10^9$	$1.2 \cdot 10^9$	$7.2 \cdot 10^8$	$4.3 \cdot 10^8$	$2.4 \cdot 10^8$	$8.4 \cdot 10^7$	$2.9 \cdot 10^7$

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summary of the results. The complete output listing includes concentration of each isotope, the activity of each isotope, the heat generation rate of each isotope, and various summations by element.

9.8 COVER NOZZLE SHIELDING

The reference core requires penetration of the cover shield by about 150 nozzles to provide access for driver fuel, control elements, reflector material, and irradiation test space. Sufficient radiation attenuation characteristics must be provided in the design of these entryways to ensure that the effectiveness and value of the cover shield are not compromised. Preliminary calculations to determine the effect of a simple annular clearance between the nozzle and tube, and to estimate the effect of neutron streaming through sodium coolant in closed loop tubes were completed. This study was based on the conceptual arrangement shown in Figure 9.11.

In the conceptual arrangement, the inner nozzle radius of about 9.2 cm and the outer pressure tube radius of about 8.25 cm, provides an annular clearance of about 0.375 in. From geometric considerations, the neutron flux through an annular gap is given by

$$\phi = \frac{10 \phi_0}{L^2} \left[(2R^2 - r^2) \cos^{-1}(r/R) - r \sqrt{R^2 - r^2} \right]$$

where R is the outer radius, r the inner radius, and L the annular gap length (shielding thickness).

Table 9.VIII presents results of calculations for various clearances, where the leakage flux was averaged over the cell area. Simple annular clearance will not provide adequate attenuation for cover shielding required to reduce

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FIGURE 9.11. Closed Loop Re-entrant Tube (Arrangement III)

the neutron flux intensity by more than three orders of magnitude. A stepped or conical arrangement would be required to eliminate streaming through the annular clearance.

TABLE 9.VIII. Neutron Leakage as a Function of Annular Clearance

Nozzle Inner Radius, cm	Tube Outer Radius, cm	Clearance, in.	Cell Averaged Leakage Flux, n/cm ² sec
9.2	8.5	0.275	$0.62 \times 10^{-3} \phi_o$
9.2	8.25	0.374	$1.33 \times 10^{-3} \phi_o$
9.2	8.0	0.473	$4.31 \times 10^{-3} \phi_o$

9.9 PHYSICS COMPUTER PROGRAMS

This section summarizes the more important physics computer programs written during the reporting period.

9.9.1 Cross Section Codes

Computer code AVECX-II was written to calculate group constants for fast reactor calculations from currently available cross section data. Output from this code includes punched cards in the Russian (Bondarenko) format usable as input to the code FCC-IV.

9.9.2 Computer Plotting of Flux

A computer program was written to plot the spatial flux distributions from multigroup, two-dimensional transport codes. Options are available to plot the total flux, the flux by energy group, or the difference between a case with a perturbation and the unperturbed base case. Figure 9.12 shows a typical plot made by the computer.

FIGURE 9.12. Computer Half Core Plot of the Low Energy Group ($1 \text{ keV} > E_n > 0$) for ZPR-III 48-A with No Control Rods

9.9.3 Fission Product Capture Cross Sections

Computer code FPAV was written to calculate one-group (FTR-spectrum-weighted) average capture cross sections for the 450 fission product isotopes used as input in program RIBD.

9.9.4 Burnup Code Development

A multigroup, 2-dimensional burnup code, 2DB, suitable for calculating k_{eff} , α , material concentration, and zone thickness, was developed.

9.9.5 Maximum Hypothetical Accidents

Briefly, MAX is a multizone, 1-dimensional maximum accident code employing a point kinetics model with fixed power and worth profiles. Linear and Doppler feedback are included in the code. The hydrodynamics model consists of numerous compressible and movable mesh point.

9.9.6 Meltdown Patterns in the NPTF

A computer code (currently designated as MELT-I) was developed to analyze possible meltdown patterns in the NPTF. Because core nodal temperatures are determined by a projection of temperatures accurately computed in only a very small region of the reactor, some comparisons of the MELT-I code with selected NUTIGER results were made.

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SECTION 10.0REACTOR SAFEGUARDS10.1 PARAMETRIC ANALYSES OF MAXIMUM HYPOTHETICAL ACCIDENTS

The maximum possible nuclear accident in a fast reactor is extremely difficult to estimate. Perhaps the greatest uncertainty lies in predicting the failure mechanisms and initial conditions.

To get a preliminary estimate of the maximum nuclear accident, the reactor response was computed for a wide range of reactivity ramp inputs and initial conditions. All calculations were performed using the MAX⁽¹⁾ code.

All calculations were performed in spherical geometry for a one-zone reactor similar in size and composition to the FTR. Table 10.I gives the principal input parameters.

Figure 10.1 shows the total energy as a function of the Doppler coefficient for various reactivity ramps in a cold reactor. The total energy, available work, and maximum pressure are plotted in Figures 10.2, 10.3, and 10.4 as a function of the Doppler coefficient in a hot core.

The cold-core excursions are quite sensitive to the Doppler coefficient, whereas the hot-core excursions are rather insensitive. This difference is caused, for the most part, by the inverse dependence of the Doppler coefficient on temperature. Also, the zero-Doppler excursions are more severe in a cold core, whereas moderate Doppler excursions are more severe in a hot core.

1. W. W. Little, Jr. BNWL-612. A One-Dimensional Maximum Hypothetical Accident Code in FORTRAN IV. Battelle-Northwest, Richland, Washington. November 1967.

TABLE 10.1. Input Parameters for Maximum Accident Calculations

Core Radius	65 cm
Core Volume	1150 liters
Delay Fraction	0.0030
Neutron Generation Time	3.5×10^{-7} sec
Number of Precursor Groups	6
Inertial Density	5.4 g/cm^3
Fuel Density	3.4 g/cm^3
Melting Point	3000 °K
Heat of Fusion	280 j/g
Eq. of State	Taken from Menzies ^(a)
Power and Worth Profiles	$1 - c(r/r_0)^2$, with a peak-to-average of 2
Hot Core:	
Initial Power	540 MW
Initial Core Temp.	3000 °K (solid)
Cold Core:	
Initial Power	0.54 MW
Initial Core Temp.	300 °K

-
- a. D. C. Menzies. The Equation of State of Uranium Dioxide at High Temperatures and Pressures, TRG Report 1119 (1966).

10.2 DESIGN BASIS ACCIDENT

10.2.1 Definition

Results of work to date in defining the Design Basis Accident are summarized in the following:

- The FFTF Design Basis Accident may be a nuclear excursion terminated by explosive disassembly of the reactor core.

FIGURE 10.1. Total Energy as a Function of Doppler Coefficient

FIGURE 10.2. Total Energy as a Function of Doppler Coefficient

FIGURE 10.3. Available Work to 20 Atmospheres

FIGURE 10.4. Maximum Pressure Versus Doppler

The final definition of the magnitude of the DBA has not been made, but preliminary calculations indicate a magnitude in the range of 1000 to 3000 MW-sec energy release. Further analyses are continuing with the goal of defining the DBA early in preliminary design of the facility.

- The DBA may be expected to result in a shock wave, blast force, sodium-hammer, missile generation, and quasistatic pressure and temperature buildup, with secondary damage potentially associated with all of these. The distribution of energy among these phenomena cannot be accurately predicted. Therefore, parametric analyses to define "worst" conditions will be necessary.
- Integrity of the reactor containment must be maintained following the DBA. Thus, the containment must be protected from the damaging potential of the accident. The DBA also results in the vaporization of significant quantities of fuel and the corresponding release of plutonium and other relatively nonvolatile solids, together with the more volatile fission products.
- The design of the containment must consider the potential for a sodium spray fire as a result of the DBA. If such a fire is a possibility, the structure must be able to sustain the resulting pressure and temperature buildup as well as the direct consequences of the DBA itself.
- The theoretical values for maximum pressure and temperature resulting from a sodium spray fire are about 120 psig and 3600 °F, respectively, as cited in (ANL-7230, Reactor Development Program Progress Report, June 1966). Test data reproduced in the FARET Preliminary Safety Analysis Summary (ANL-7168, vol, II, p. 166) show that values up to about 80 psig and 3600 °F can be attained in practice, at least under experimental conditions. It may be possible

to develop valid arguments for lower values in a specific engineering application. Even then, however, values of several tens of psi and thousands of °F would seem likely.

10.2.2 Fault Tree Analysis

The fundamental assumption in the FFTF Design Basis Accident analysis is that fuel meltdown and core compaction occurs. In order to assess all possible failure mechanisms leading to fuel meltdown, a fault tree analysis of the entire FFTF system is currently in process. Initial work in this area was carried out by Atomics International, and the present effort will result in an expanded and updated version of the fault tree. Figure 10.5 depicts a typical fault tree analysis sequence.

10.3 DESIGN BASIS ACCIDENT (DBA) ANALYSIS FOR THE NPTF

A program is in progress to obtain a preliminary estimate of the energy release in the DBA for the NPTF. Assumed in this program is the capability of dividing an accident into the following three distinct phases:

- A failure, malfunction, or operator error will initiate a power transient severe enough to damage the core.
- The core will lose its structural integrity and the fuel will begin to slump into a more reactive configuration, thus producing a reactivity addition which could aggravate the accident.
- Rising of the energy content of the core to a sufficiently high value will generate pressures which, in turn, will disperse the core and terminate the accident.

Recent analysis has shown that the computer codes planned for use in these calculations do not adequately describe the transition between accident phases. Work during the present reporting period has been concerned with modifying these codes.

FIGURE 10.5. Typical Fault Combinations and Sequences Required for Release of Radioactive Material

Thus far, it has been assumed that the cladding must fail before any gross fuel movement can occur. Presently, the code assumes cladding failure whenever any one of the following four mechanisms occur:

- The cladding reaches its normal melting point
- Fuel in the outer node adjacent to the cladding melts (and, hence, comes into direct contact with the cladding)
- Pressure in the center of the fuel pin (caused by either fission gases or fuel vapor pressure) rises sufficiently to rupture the cladding
- The coolant on the outer side of the cladding reaches its vaporization point.

Once the cladding has failed, all fuel above that point will slump under the force of gravity (a viscous term is also contained in the equation of motion). Before that point, however, local cladding must fail before the fuel is allowed to slump. Whenever fuel motion takes place, the reactivity feedback caused by collapse is determined by the total displacement down a worth curve. Since the worth curves (input) are stationary in space throughout the transient, the model is no longer valid after a sufficiently large change in the core configuration develops. Early results indicate, however, that fuel temperature characteristics of an explosive core are generated before much fuel movement has occurred.

This code should provide a smooth transition between Phases 1 and 2 and lead logically into Phase 3. For the NPTF analysis, the code may even cover Phases 1 and 2 completely. However, because the temperature calculations assume complete separability in space and time and, furthermore, are based on a constant conductivity, comparative runs on NUTIGER or FORE-II up to the melting point must be made before a reasonable degree of confidence can be attained.

The computer code WEAK (ASTRA 417-6.0) will be used to calculate the energy release in the DBA. In the present DBA

program, this code must pick up the accident calculation in the fuel slumping phase and carry it through the dispersion of the core. In its original form, the WEAK code was not versatile enough to allow for a smooth transition from the modified K-III code (Phases 1 and 2). The code has subsequently been modified so that the initial power, energy content, reactivity, and delayed neutron precursor densities may be supplied as input. In addition, the input reactivity (due to collapse and the initiating mechanisms) as a function of time now has the form,

$$\Delta k = \alpha t + \alpha_1 t^2 \quad .$$

These modifications should allow WEAK to duplicate the modified K-III calculation over a reasonable time interval and thus render the final energy release calculation less sensitive to the choice of when the accident calculation is transferred to WEAK.

10.4 DBA RADIOLOGICAL CONSEQUENCES

To provide basic data for the evaluation of the relative suitability of various Hanford sites for the FFTF, the potential radiological consequences of a Design Basis Accident (DBA) were calculated. After a review of the FTR inventory of fission products as related to the DBA assumptions, the following items were evaluated:

- Whole body penetrating radiation dose from passing cloud
- Direct gamma dose from the container
- Thyroid dose
- Bone dose from mixed fission products
- Lung dose from mixed fission products
- Bone dose from plutonium
- Lung dose from plutonium

The assumptions used probably result in a substantial overestimate of the lung, thyroid, and bone dose. However, the work thus far completed does permit some conclusions regarding the relative importance of these seven items. The bone dose potential is concluded to be the most important factor in siting the FFTF.

Figure 10.6 presents the results of the calculations. Using 10 CFR 100 siting criteria as limits for the whole body and thyroid dose, and an allowable containment vessel leak rate of 1%, the exclusion area boundary was calculated to be 1/2 mile and the low population zone boundary 1 mile for the FFTF. Reduction of containment vessel leak rate does not appreciably change the required exclusion boundary distance (Curves A and B of Figure 10.6). A reduction of the containment vessel leak rate from 1% to 1/10% per day would reduce the low population boundary from 1 mile to about 1/2 mile.

10.5 FFTF SITE EVALUATION CRITERIA

The FFTF Site Evaluation Criteria have been developed. Salient portions are quoted in the following.

10.5.1 Isolation

The site shall be capable of meeting the isolation requirements imposed by 10 CFR 100.

10.5.2 Physical Characteristics of the Site

- Seismology

No facility should be located closer than 1/4 mile from the surface location of a known active earthquake fault.

- Geology

The geology of the site must provide formations that will give sound footings for the plant structures.

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FIGURE 10.6. Allowable Containment Vessel Leak Rate as a Function of Boundary Distance

- Hydrology

The site must be at an elevation providing protection from the highest floods on record. The site should afford protection from accidental flow of significant amounts of radioactive effluent into nearby rivers or the underground water table.

- Meteorology

Meteorology of the site should ensure good dilution and transportation of gases released from the facility.

10.5.3 Utilities

- A dependable electrical supply must be available. The system supply of electricity must have a history of excellent continuity of service.
- The site must afford an adequate and dependable supply of cooling water.

10.5.4 Test Support Features

The site should offer the following:

- Convenient access to laboratory, examination facilities, and technical services
- Good physical accessibility for testers and tests
- Minimum restrictions on personnel access
- Flexibility for extending the test program.

10.6 Reliability Goals

It has become evident that studies of systems effectiveness must consider separately each of two or three major criteria categories such as safety and testing effectiveness. The latter category might be broken down to testing irradiation effectiveness. The selection of such specific overall criteria permits the selection and specification of specific criteria for each system and

for each major mode of operation for each system so that goals for reliability, availability, and performance based on the overall criteria can be established.

A starting point for development of a testing irradiation effectiveness criterion is the unavailability allocation to the various systems. These allocations are shown in Figure 10.7.

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