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FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
JANUARY, FEBRUARY, MARCH 1968

**AEC RESEARCH &
DEVELOPMENT REPORT**

REPORT	DATE	REVISION	REVISION DATE

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FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
JANUARY, FEBRUARY, MARCH 1968

By
E. R. Astley
Project Manager

Compiled by C. P. Cabell

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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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Under steady-state conditions, with a field of approximately 7×10^{12} n/cm²-sec (fast and thermal) and 7×10^6 R/hr, no calibration shift (± 1.0 °C) was observed during startup, steady state, shutdown, or for one hour after shutdown.

Under pulsed conditions, no indication of a calibration error in Cr-Al or IC thermocouples due to instantaneous effects of radiation fields up to 10^{16} n/cm²-sec fast and thermal, and 10^{10} R/hr at temperatures up to 600 °C was observed.

The 180 psi testing of transducers in the Static Sodium Facility continued. A NaK filled capillary unit was successfully tested with sodium at 1200 °F for 30 days. An unbonded strain gage transducer and a film deposited strain gage transformer failed in similar tests at 600 °F. Two type B-7 pins containing FFTF candidate thermocouples were completed and shipped to Idaho Falls, Idaho, for irradiation in the EBR-II.

1.5 SODIUM TECHNOLOGY

To obtain visual information on sodium vapor trapping, experiments were performed in glass equipment. Two traps were fabricated, one with a simple cold finger condenser section and the other with a section packed with graduated sizes of glass rings topped with glass wool. These qualitative experiments indicated the packed trap as more efficient, and that a major portion of the sodium vapor contained in a hot gas stream could be removed by cooling to approximately 100 °C followed by filtration.

Testing is presently in progress to determine the time required for sodium removal from deadlegs by various alcohols. Alcohols used are methanol, ethanol, butanol, and Dowanol EB (ethylene glycol m-butyl ether 2-butoxythanol).

With sufficient reagent present, a 95% Dowanol EB-5% water solution removed sodium as rapidly as any of the other solutions tested. The Dowanol EB flashpoint (165 °F), is considerably higher than the flashpoints of 55 °F or less for methanol, ethanol, and butanol and, hence, Dowanol EB offers a lower fire hazard.

An irradiated piece of 304 SS fuel clad was run in the Radioisotope Transport Loop for 460 hr at 1300 °F maximum and 500 °F differential temperature, with a sodium velocity of 40 ft/sec. Gamma scans of the loop piping showed that, as anticipated, ^{51}Cr was the major radioactive species leached from the specimen. The deposition of ^{51}Cr was predominantly in the hot leg adjacent to the sample and immediately downstream.

1.6 CORE DEVELOPMENT

A major milestone was achieved in the development of electrically heated models of FFTF fuel pins for heat transfer experimentation. A rotary-swaged pin consisting of a nickel sheath, molybdenum resistance element, and boron nitride insulation was demonstrated at heat fluxes in excess of 1,500,000 Btu/hr-ft² in 334 °F water for over seven hours.

1.7 REACTOR MATERIALS DEVELOPMENT

Based on an analysis of stainless steel volume expansion data at exposures to 8×10^{22} nvt, it was estimated that volume may increase as much as 10% at 10^{23} nvt and irradiation temperatures of 800 to 1100 °F.

An electron microscopy examination performed on irradiated Type 304 stainless steel indicated that cavities in irradiated stainless steel annealed at 1596 °F (870 °C) and 10,000 atmospheres behave like gas-filled bubbles rather than voids.

PKA spectra were calculated for two 304 stainless steel specimens, a tensile specimen, and a specimen equal in thickness to the FTR fuel clad. PKA spectra are presented herein for each of the three primary elements of 304 stainless steel in each specimen geometry and for irradiations in DFR, EBR-II, FTR, and ETR spectra.

The activation energies for motion of vacancies in the vicinity of vacancy clusters are being determined with the computer code DEFECT. The initial runs have yielded higher activation energies than were anticipated from published data.

Pressure stress calculations for the present design of the fuel duct indicate that an 85 psi pressure differential would induce significant tube expansion. Experimental tests have verified the calculations.

A summary of data collected on plastics was distributed to RDT and the FFTF Project. The tabulations listed a number of mechanical, physical, and chemical properties primarily at room temperature. Two of the most important properties in these data compilations are maximum useful temperature and irradiation damage characteristics.

1.8 FUELS DEVELOPMENT

The FFTF prototypical 217-pin flow test subassembly was shipped to Argonne National Laboratory for sodium flow testing in the CCTL.

Current studies show that, at FTR goal fuel exposures, diametral expansion of the fuel pins is unavoidable because of fission product-induced swelling and growth of the fuel and the density decrease of the cladding due to radiation damage. This conclusion is diametrically opposed to that derived from a previous study based on data available in April 1967.

Prototypical wire-wrapped cladding tubing was tested to simulate the effect of fuel pin swelling. There was no observable deformation of cladding by the wires.

Recently published BNW information on a revised approach to the density correction for the thermal conductivity of mixed oxide fuels is being incorporated in the basic FFTF fuel design curves. A significant effect of lower thermal conductivity, computed on the revised bases for the high temperature region, is that postulated accidents will raise peak fuel temperatures to a greater degree during transients in which large portions of the fuel are at elevated temperatures.

Existing experimental data on behavior of flat and dished end pellets have been reviewed for application to the operating conditions and requirements for the FTR driver fuel. It was concluded that flat-end pellets should be retained as the reference FTR design.

1.9 PHYSICS

ZPR-III Assembly 51 (Assembly I of Phase B) became critical with a core fissile inventory of 215 kg. Results to date show that central tantalum rods of volumes equal to those of B_4C rods possess about 2/3 the reactivity worth of B_4C rods. Substantial differences were noted in the multigroup scattering cross sections for iron in a portion of the resonance region. The effect of these discrepancies on specific shielding problems is being assessed.

Calculations were made to compare ANISN and MAC using consistent cross section data. The results indicate the simpler ANISN diffusion calculations to be adequate for estimating the attenuation characteristics of these metals.

1.10 REACTOR SAFEGUARDS

A mechanistic approach is being taken by the DBA Task Force. This approach is divided into the following four major phases of analysis:

- Initiating accidents
- Core meltdown
- Core disassembly
- Evaluation of the energy partitioning.

Calculations to establish minimum critical parameters for FFTF driver fuel are continuing. Using the Monte Carlo GEM III code, the minimum critical number of 30 wt% $\text{PuO}_2\text{-UO}_2$, unmoderated subassemblies was estimated to be 10.5.

SECTION 2.0

FFTF DESIGN, CONSTRUCTION, OPERATION

This Section presents selected items of general interest.

2.1 CORE CONCEPT STUDIES2.1.1 Bisected Conical Core Studies

Problems in obtaining flux levels of 0.7×10^{16} n/cm²-sec were previously reported.⁽¹⁾ Continuing studies performed during this reporting period concluded that all design parameters significantly affecting the achievable flux level for the first core were limited by constraints of practical technology, safety, testing capability requirements, or economics. Significant flux improvements in the basic bisected conical core geometries appear virtually impossible without utilizing odd-sized subassemblies to fill the split gaps. However, maintaining a single driver subassembly design to avoid additional complexities in the core and refueling procedures was considered preferable.

2.1.2 Trisected Core Arrangement Studies

Two trisected core cross section arrangements as a basis for an anticipated study of trisected geometries were completed. The two cross sections appear the most compact array for the two geometries considered. Each utilizes the present baseline driver subassembly design with 6 in. OD closed loops. Driver dispersion rate and, consequently, unit volumetric fractions are similar to the baseline bisected core.

The Trisected Core Study I design (Figure 2.1) is split at three corners of the basic hexagonal core cross section. Splitting at the corners allows three additional subassemblies to be

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1. Fast Flux Test Facility, Quarterly Technical Report, July-December, 1967, BNWL-567. Pacific Northwest Laboratory, Richland, Washington, February 1969.

FIGURE 2.1. Trisected Core Study I

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added in each split corridor outboard of the "B, C, and D" closed loop positions. The main portion of the split other-sodium volume is thereby concentrated in the central region of the core. An option also exists for adding one to three additional closed loop positions.

The Trisected Core Study II places the splitting planes at the center of the flats of the basic hexagonal core cross section. This design (Figure 2.2) places six driver subassemblies adjacent to the central closed loop position as compared to three subassemblies in Study I. The resulting central test flux may thus be enhanced. However, a larger gap then exists around the outboard closed loops, resulting in a larger overall split other-sodium fraction.

2.1.3 Nuclear Analysis of Trisected Core Designs

Results of nuclear analyses comparing the trisected core designs with the reference bisected core are summarized in the following tabulation. The trisected core design was found to provide lower flux than the reference bisected core.

<u>Total Flux, n-cm⁻²-sec⁻¹</u>	<u>Bisected Reference</u>	<u>Trisected Study I</u>	<u>Trisected Study II</u>
<u>Driver</u>			
Average	4.35×10^{15}	4.14×10^{15}	4.05×10^{15}
Peak	6.21×10^{15}	5.39×10^{15}	5.52×10^{15}

2.1.4 True Conical Core Studies

An additional configuration on which work was initiated during the reporting period was a true conical core design with four central positions removed for insertion of vertical closed loops shown in Figure 2.3. The central positions can be replaced by three closed loops with an across-the-flats dimension (or OD) equivalent to that of the driver assemblies.

FIGURE 2.2. *Trisected Core Study II*

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DATA
CONICAL ARRAY DISPERSION
RATE .0158"/%
LATTICE 4.55"
SUB ASSY. FLAT TO FLAT 4.52"
3 VERTICAL CLOSED LOOPS
(SHROUD SAME SIZE AS
SUB-ASSY. SEE ENLARGED
CORE PLAN BELOW)

Neg 0681168-4

FIGURE 2.3. True Conical Core with Vertical Tests (Study I)

SK-3-14137

Since the replaced positions are vertical, the converging geometry does not allow the closed loops to penetrate the lower tubesheet. The primary advantage of this geometry is substantially improved packing efficiency due to removal of the large interstitial sodium gaps characteristic of the bisected design. Evaluation of this concept is in progress.

2.2 DATA HANDLING SYSTEM DESIGN

A study was performed to estimate the general magnitude and arrangement of the total number of inputs to the data handling system. Figure 2.4 represents a summary of detailed listings describing each measurement and the associated requirements available for the current system concepts. This summary shows how the estimated inputs can be organized to provide a manageable and efficient data handling system.

As shown in Figure 2.4, computer A acquires data from the core, vessel, and shield by means of approximately 800 analog sensors at an average scan rate of 1200 points/sec. Computer B is assigned the general housekeeping duties and serves as a backup for the other computers. Computer C acquires data from the process and experimental systems by means of 975 analog inputs with an average scan rate of 250 points/sec.

2.3 NUCLEAR PROOF TEST FACILITY LOCATION

An evaluation of five potential Nuclear Proof Test Facility locations was completed in March 1968. Conceptual design drawings were prepared for each location to serve as the basis for evaluating construction costs and schedules and operating considerations. Locations considered included (1) within FFTF Building along building perimeter, (2) adjacent to FFTF with a common wall, (3) separate building in FFTF area,

Neg 0681205-1

SK-3-14141

FIGURE 2.4. Tentative Inputs to Digital Data Handling System

(4) within FFTF building, below grade, and (5) several miles from the FFTF area. It was tentatively concluded that location (1) would be most advantageous.

Some minor difficulties may arise from early startup of the NPTF while FFTF is still under construction. However, other plants have, with little difficulty, established the precedent of early startup and operation of a critical facility with reactor facility construction continuing in the same building.

SECTION 3.0

COMPONENT DEVELOPMENT3.1 SEALING MECHANISMS TESTS

Preparation of a proof-of-principle test procedure and the design of test apparatus for the experimental testing of sealing mechanisms for the FTR was begun. The test procedure includes (a) proof pressure testing to 375 psig, (b) functional testing to check helium leak rates as low as 10^{-7} atm cc/sec, (c) thermal cycle from room temperature to 1200 °F and return in 8 hr, and (d) thermal shock from 1400 to 400 °F in 2 seconds. Resistance heaters and air cooling are to be used for the thermal cycle, while the thermal shock test will use flowing sodium to achieve the required rapid thermal transients.

3.2 CONTROL ROD DRIVE LATCH COMPONENT TEST STAND

The current effort is to design, build, and evaluate a safety rod latch based on the FERMI design, but with increased holding capacity for the heavier FFTF safety rod.

The latch test stand will be mounted vertically and gimballed to allow orientation from the vertical as required.

Design of the latch assembly and automatic control system was started. The following events describe this test cycle.

The spool piece (simulating the top of the poison element) will be attached to the bottom of the test stand. The test cycle will start by lowering the latch assembly over the spool piece, the latch will be actuated by the latch magnet, and the assembly raised to load the latch by two hydraulic cylinders. The latch magnet will then be de-energized to release the latch fingers, and the cycle repeated. Actuation, load, and holding times will be adjustable.

3.3 DRIVE ASSEMBLY TEST STAND

This effort was established to fabricate and evaluate a drive arrangement using a stepping motor and ball lead screw combined with a position indicator and limit switches. The drive will be tested in air to determine drive rate, lifting capacity, positioning precision, and rod position reproducibility.

Design of the drive assembly is complete. The stepping motor and the ball screw, with preloaded ball nut, have been ordered. Fabrication of the test stand has started. The position indication system has not yet been selected; however, a type of shaft encoder that is compatible with computer operation of the reactor and that will give a digital, visual display of rod position, appears to be most favorable.

3.4 WELDING STUDIES

Six simulated top face closed loop hardware sections were fabricated for evaluation of weld joint design, remote weld tooling requirements, and sodium contamination effects. An existing tungsten arc welding head was modified for this evaluation.

3.5 NOISE ANALYSIS EVALUATION

Noise analysis, or correlation analysis is a technique for distilling time-of-flight and similar information from sensed, system generated, random variations. Initial experiments with probes have shown clearly that electrochemical reactions at the probes noticeably affect performance. Effort toward gaining a better understanding of these reactions is therefore underway.

3.6 SODIUM-COOLED FUEL HANDLING CASK DESIGN STUDIES

Atomics International is attempting to simplify the cooling system for the sodium-cooled fuel handling cask. For normal service, the desired heat removal capability is for not less than 30 kW to maintain the maximum temperature of the hottest driver fuel pin cladding below 1200 °F.

A concept (Figure 3.1) was proposed for further design study. In this concept, the cask carried the core component in liquid sodium. The sodium can transfer heat by conduction directly to the walls of the cask, and by natural convection.

The concept provides for the attachment of fins to the FHM barrel, and a housing located outside the fins for confining the cooling gas to the fin area to increase the heat removal efficiency of the system. Natural convection currents alone in the "chimney" can remove sufficient heat to keep the hottest fuel pin clad temperature below the specified 1200 °F. This cask is therefore safe even when the gas circulating system cannot function.

Preheating and temperature control features of this cask, and the method of applying radiation shielding materials, will be the subject of future studies.

3.7 CLOSED LOOP TOP FACE HARDWARE DEVELOPMENT

Figure 3.2a depicts a removable shield plug exposing the test specimen grapple area. A procedure drawing was prepared showing a way of removal and replacement of closed loop specimens (Figure 3.2b), with or without removing the closed loop in-reactor tube.

FIGURE 3.1. Chimney and Fin Cask

Figure 3.2a. Closed Loop Subassembly

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(Sheet 1 of 2)

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3.6

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Figure 3.2b. Closed Loop Subassembly Replacement

The conceptual closed loop nozzle backup design was completed as shown in Figure 3.3. This top closure design features creep compensating latching for mechanical seals operated by a hydraulic actuator. Sintered stainless insulation is used in place of gas in the double wall flow tube. The possibility of this material as a thermal barrier is being studied.

Figure 3.4 shows the conceptual design of the top face hardware instrument lead routing and closed loop coolant piping.

3.8 SHIELDED CASK FOR HIGH ACTIVITY SOLID WASTES

High activity solid wastes will be transported in shielded casks from the FFTF to burial grounds in the Hanford Area. Figure 3.5 shows the conceptual design of a shielded cask and a typical loading operation. Wastes will be removed from spaces of inert gas atmosphere, thereby requiring the cask design to incorporate features for maintaining cell atmosphere purity during the waste container loading operation. This objective is accomplished by purging the cask with the same type of inert gas as the space atmosphere, and making a sealed connection to the space transfer port. The cask loading operation consists of positioning and purging the cask, opening the cask and transfer port valves, lowering the grapple onto the waste container (loaded, sealed, and positioned by the waste contributor), hoisting the waste container into the cask, and closing the transfer port and cask valves.

3.9 DIGITAL MODELING

3.9.1 Dump Heat Exchanger Simulation

Two models were constructed for digital simulation of the FFTF's air dump heat exchangers used in the heat transport system. One of these is a complex, 18-point, cross-flow model

FIGURE 3.3. Closed Loop Top Face Hardware (Arrangement C)

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3.9

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FIGURE 3.4. Reactor Top Face Plan and Sections (Instrument and Cooling)

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PLAN- SOLID WASTE REMOVAL AREA

SECTION A-A

0 5 10 FT

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FIGURE 3.5. Typical Shielded Solid Waste Transfer Cask

in which air is unmixed between the six successive tube passes. Each pass, consisting of two nodes, is described by three points. The other model is a simple, four-node counter-current model with sodium on the tube side. Log-mean temperature difference (LMTD) correctors are used with the counter-current model to account for the actual cross-flow pattern. The heat storage capacity of the tube wall and fins is treated as equivalent secondary sodium in both models. This assumption appears reasonable inasmuch as magnitudes of the heat transfer coefficients on the inside and outside are relative.

The use of the simple counter-current model affords a significant savings in computer time and an important reduction in computer storage locations. These advantages apparently can be realized without a serious sacrifice in simulation accuracy. Hence, the heat transport system models will incorporate the simple counter-current representation for the air heat dump exchangers. Additional work is being done to determine if the LMTD correctors are needed.

3.9.2 Secondary Loop Simulation

Digital simulation models of the heat transport system's secondary loops were completed. The models include sodium flow control and emergency trip switches as well as complete thermal and hydraulics simulation. The dump heat exchanger is included as a part of the secondary loop model, whereas the IHX is included in the primary loop model.

SECTION 4.0

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

The object of this task is to develop sufficient knowledge of the processes of inducing current in signal cables by radiation so that cables of maximum resistance to this effect can be specified for in-core FFTF use. A secondary objective is to determine the instantaneous radiation or dose rate effects on thermocouple calibration.

Previously reported tests at the Washington State University TRIGA Reactor were continued at a temperature of 1200 °F.

A summary and conclusion covering all thermocouple data obtained to date are contained in the following paragraphs.

The instantaneous or dose rate effects of nuclear radiation on thermocouple calibration were measured. In a ^{60}Co gamma facility with a maximum dose rate of 2.3×10^7 R/hr, a voltage shift of 4 ± 2 μV was observed for bare or grounded junction sheathed thermocouples (Cr-Al and IC). A shift of 8 ± 2 μV was measured for ungrounded sheathed junctions. Other experiments indicate that these shifts are due to heat transfer, from the bead to the reference temperature, both caused by gamma heating. This problem was minimized but could not be eliminated. Experiments in a pulsed TRIGA Reactor indicated a shift of 2.3 ± 0.3 °C for both Cr-Al and IC at 232 °C. Results at 600 °C indicate a shift of 2 to 4 °C. A simple heat transfer calculation predicts a shift of this magnitude due to local gamma heating. No indication of a calibration error in Cr-Al or IC thermocouples due to instantaneous effects of radiation fields up to 10^{16} n/cm² sec fast and thermal, and 10^{10} R/hr at temperatures up to 660 °C was concluded.

Separate experiments were carried out using the steady-state capabilities of the TRIGA Reactor. No calibration shift (± 1.0 °C) was observed during startup, steady state, shutdown, or for 1 hr following shutdown. The radiation field during the steady-state part of these experiments was approximately 7×10^{12} n/cm² sec (fast and thermal) and 7×10^6 R/hr. In fields of this magnitude, no transient changes in thermocouple calibration (± 1.0 °C) were concluded to follow a change in radiation level.

In the high temperature TRIGA Tests, an attempt was made to reproduce the experimental results of Leonard and Hunkar¹ who detected a shift of 0.42 mV or about 10 °C following a reactor shutdown. The net neutron flux in that experiment was given as 6.3×10^{12} n/cm² sec fast and 4.5×10^{11} n/cm² sec thermal. The WSU TRIGA steady-state power level of 1 MW produces a neutron flux of about 7×10^{12} n/cm² sec fast and thermal. The power level was brought to 1 MW and maintained at that level for 20 minutes. The reactor was then scrammed and measurements were made for one hour after shutdown. During this 80 min period, the freeze point plateau of aluminum was reached 11 times. The reactor was brought to power and scrammed while the thermocouples were at the freeze point and no changes were observed. One Cr-Al and one IC thermocouple were monitored during this time. A random change of up to ± 1.0 °C was found. This change, within the ± 1.0 °C estimated error inherent in this experiment, was due to drifts in reference temperature and instruments.

4.2 LOW LEVEL NEUTRON FLUX INSTRUMENTATION AND DELAYED NEUTRON FAILED FUEL MONITOR

The objective of this study is to test and evaluate nuclear components of a counting system in an adverse gamma to neutron

1. J. H. Leonard, D. B. Hunkar. Nuclear Applications 3, 718 (67).

flux environment for the detection of delayed neutrons, for failed fuel monitoring, and for the reactor start-up instrumentation channels.

A current-mode preamplifier, investigated briefly to determine its potential usefulness, proved to be extremely fast. Rise and decay times were less than 20 n/sec for pulses from a fission or proportional counter. Although the preamplifier noise requires a high discriminator setting, a 1-in. diam fission counter can be exposed to gamma fields of approximately 10^6 F/hr with no apparent increase in count rate due to pulse pileup in the preamplifier. The noise level was such that the alpha pulses could not be seen. However, the noise was within the manufacturer's specifications. The preamplifier was returned to determine whether the noise level could be decreased.

Acquisition of six additional fission chambers and two boron-lined counters brings the total number of detectors available for testing to 13. The remaining five are undergoing vendor modifications to replace the organic materials in the connector with high temperature ceramic insulation.

4.3 PRESSURE, DP FLOW, AND LEVEL SENSORS FOR SODIUM SERVICE

The objective of this task is to develop pressure, flow (differential pressure type), and level instrumentation for FFTF in-vessel and piping applications. The activities required include evaluation of present technology, procurement and screen testing of commercially available products, component modification and improvement, and final proof testing.

The 180 psi testing of procured transducers in the Static Sodium Facility is continuing. A NaK filled capillary unit has been tested with sodium at 1200 °F for 30 days and another

tested at 600 °F for the same period. An unbonded strain gage transducer and a film deposited strain gage transformer failed in similar tests at 600 °F.

4.4 IN-REACTOR COOLANT TEMPERATURE SENSORS

The objective of this task is to develop and test thermocouples to provide reliable commercially available thermocouples for measuring coolant temperatures in the FTR thermal and nuclear environment.

Two type B-7 pins containing FFTF candidate thermocouples have been completed and shipped to Idaho Falls, Idaho, for irradiation in the EBR-II. The pins are designated BT-3 and BT-4. Pin BT-3 contains 26 sheathed thermocouples, 13 with grounded and 13 with ungrounded beads. In addition, BT-3 contains a bare chromel-alumel thermocouple, iron and scandium flux monitor wires, and alloy melt wires for temperature monitors. BT-4 contains 24 sheathed thermocouples, 14 with grounded and 10 with ungrounded beads. In addition, BT-4 contains iron, scandium, and titanium flux monitor wires, a bare chromel-alumel thermocouple, and alloy melt wires for temperature monitors. All the sheathed thermocouples contain chromel-alumel thermocouple wires, are insulated with compacted magnesium oxide, and are sheathed with 304 stainless steel.

A total of 24 thermocouples were removed from EBR-II after irradiation in capsules BT-1 and BT-2. The fluence received by the SC thermocouples and the temperature at which they were irradiated will be known as soon as the flux monitor wires can be counted and melt wires examined.

All but two of the 24 thermocouples were removed intact from the capsules. One unsheathed, stranded W-5% Re/W-25% Re thermocouple and one sheathed Cr/Al thermocouple were broken

during capsule disassembly. The remainder appear ready for postirradiation electrical testing. Apparent swelling of the aluminum oxide insulators froze the thermocouples into the thermal equalizer blocks. Unusual removal techniques, including the use of excessive force, was required to free them.

Beta and gamma radiation dose rates measured up to 5 R/hr at 3 ft for an Inconel-sheathed, vitrified alumina insulated Cr/Al thermocouple. The dose rates will not cause excessive operator exposure.

SECTION 5.0

SODIUM TECHNOLOGY

5.1 SELECTING AND TESTING OF RADIOACTIVE SODIUM ANALYSIS

The objectives of this activity are to:

1. Establish PNL capability for the chemical analysis of oxygen, hydrogen, total carbon, nitride nitrogen, corrosion products, fission products, and uranium and plutonium in sodium (both in the presence and absence of plutonium contamination).
2. Provide proven analytical techniques and equipment for the analyses of highly radioactive sodium in the FFTF.
3. Provide design criteria and requirements for the chemical analyses laboratory of the FFTF--notably the hot cell facility.

5.1.1 Oxygen Determination

The determination of oxygen in sodium metal is continuing as a service to customers. One NaK sample has also been analyzed. Instrumentation has been purchased and installed to monitor the oxygen and water contents of the glovebox inert atmosphere.

5.1.2 Hydrogen Determination

A program to establish capability for measuring the hydrogen content of sodium by amalgamation is underway. A literature review has been completed, a gas chromatograph has been set up, calibration begun, and the redesign of the glass amalgamation vessel and vacuum train started.

5.1.3 Nitrogen Determination

Construction of a simplified Kjeldahl still has been completed. Determinations of the nitrogen content of 314 loop sodium have been made to gain experience and to define optimum operating conditions.

5.1.4 Total Carbon Determination

Preliminary attempts to use a flash heated molecular sieve trap to remove the carbon dioxide from a pure oxygen combustion stream have been discouraging. As anticipated, difficulty has been encountered in "cleaning up" the trap sufficiently to permit handling of the low levels of carbon dioxide involved. Efforts to clean and use the trap to quantitatively recover the carbon dioxide will continue.

5.2 SELECTION AND TESTING OF FFTF SODIUM SAMPLING METHODS

The objectives of this activity are: to select methods of sampling radioactive sodium; to build and proof test prototype FFTF sampler designs; to provide the design criteria and specifications for the FFTF sampler; to demonstrate the adequacy of the sampler for taking repetitive samples of radioactive sodium; and to demonstrate remote maintenance and operation of the sampler. The design of a prototype removable section bypass loop sampler has been completed.

5.3 SODIUM COVER GAS TECHNOLOGY

5.3.1 Sodium Cover Gas Interaction Studies

Investigations on sodium cover gas interactions were continued with an ultimate goal of establishing allowable impurity levels for FFTF cover gases. The current investigations continued to study the reaction of methane with molten sodium.

An extremely large increase in sodium surface area contacting the gas, obtained by bubbling 100 ppm methane in helium gas through 100 ml of sodium, did not appear to increase significantly the methane-sodium reaction rate. The decrease in methane concentration, from approximately 100 ppm to 16 ppm over a period of 4 hr, compares to a decrease of 100 ppm to 17 ppm over a period of 5 1/2 hr for previous runs at 1200 °F using 100 ml of sodium with a surface area of 2.2 in.

Subsequent experimental reruns with large sodium volumes have shown that the reaction rate of methane with sodium was actually decreased by increasing the sodium volume. With a volume of 500 cm³ of sodium, the decrease in methane concentration from 100 ppm to 67 ppm over a period of 5 hr compares with a methane concentration decrease of 100 ppm to 41 ppm for 250 ml of sodium and 100 ppm to 17 ppm for 100 ml of sodium for the same period of time at a temperature of 1200 °F.

It now appears that the reaction rate of methane and sodium may be controlled by the volume or the metal surface area of the gas plenum in the sodium capsules. In all of the experimental runs, the size of the stainless steel capsule remained constant. The volume, and therefore the area of the gas plenum, did not change during variation of sodium surface areas by introduction of a section of tubing. When the sodium volume was increased from 100 ml to 250 ml, and then to 500 ml, however, the relative plenum size was reduced by a corresponding amount. There was very little change, if any, in the reaction rate of the methane with sodium when the gas plenum area remained constant during the former runs. However, in the latter runs when the gas plenum was decreased, the reaction rate decreased.

5.3.2 Selection and Testing of Cover Gas Instruments and Sampling Systems

The objectives of this activity are to select and proof test gas analysis instrument systems, define design criteria for associated equipment, select instrument type, identify important specifications and verify that the sampling and analysis system selected will work satisfactorily in FFTF application.

A laboratory comparison of three potentially useful oxygen meters, a solid electrolyte instrument, an aqueous silver-lead cell, and an aqueous silver-cadmium cell is in progress.

A Time of Flight mass spectrometer has been received and checked for operability. The resolution of the various masses in the range of concern for FFTF application has been found to be adequate. Final calibration of this instrument is expected shortly.

To obtain visual information on sodium vapor trapping, experiments were performed in glass equipment. Two traps were fabricated, one with a simple cold finger condenser section and the other with a section packed with graduated sizes of glass rings topped with glass wool. These qualitative experiments indicated the packed trap as more efficient, and that a major portion of the sodium vapor contained in a hot gas stream could be removed by cooling to approximately 100 °C followed by filtration.

5.4 EVALUATION OF SODIUM CLEANING

Sodium cleaning procedures are being evaluated for FFTF fuel assemblies, components, and piping systems. All sodium must be removed from sodium piping systems prior to decontamination with aqueous solutions, including sodium in crevices,

deadlegs, and on components in the pipe system. Residual sodium could cause caustic corrosion when the piping is reactivated. Candidate processes for sodium removal in the pipes are steam, alcohol, and ammonia.

Testing is presently in progress to determine the time required for sodium removal from deadlegs by various alcohols. Alcohols used are methanol, ethanol, butanol, and Dowanol EB (ethylene glycol m-butyl ether 2-butoxyethanol). Water was added to ethanol and Dowanol EB to speed up the sodium removal rate.

With sufficient reagent present, the 95% Dowanol EB-5% water solution removes sodium as rapidly as any of the other solutions tested. Preliminary results in 1/4 in. diam deadlegs indicate the need for a reagent-to-sodium volume ratio of 110 to 1.

The 165 °F Dowanol EB flashpoint is considerably higher than the flashpoints of 55 °F or less for methanol, ethanol, and butanol. This higher flashpoint would give the Dowanol EB a lower fire hazard. Other sodium cleaning studies using Dowanol EB have been reported by Chulos.⁽¹⁾

5.5 RADIOISOTOPE TRANSPORT

The objectives of this work are to define the parameters causing deposition of radioisotopes, and to predict deposition patterns in FFTF piping.

5.5.1 Transport Loop Studies

An irradiated piece of 304 SS fuel clad was run in the Radioisotope Transport Loop for 460 hr at 1300 °F maximum temperature and 500 °F differential temperature, and with a sodium

1. L. E. Chulos. Sodium Removal and Cleaning of Reusable Hardware, BNWL-637, Pacific Northwest Laboratory, Richland, Washington. December 1967.

velocity of 40 ft/sec. Gamma scans of the loop piping showed ^{51}Cr was the major radioactive species leached from the specimen. ^{51}Cr was the predominate species in the sample. The deposition of ^{51}Cr was predominantly in the hot leg adjacent to the sample and immediately downstream. The initial results indicate decreasing deposition with decreasing velocity, and the deposition mechanism appears related to the probability of a radioactive atom hitting a pipe wall where a radioisotope exchange process occurs. Most of the radioactive atoms were deposited before reaching the heat exchanger. Results from a thermal convection loop capsule test (see 5.5.2) confirmed the same distribution found in the pumped system. However, more radioactivity reached the cold leg of the low velocity thermal convection loop than reached the cold leg of the higher velocity pumped system.

5.5.2 Convection Loop Studies

Tentative conclusions, based on work to date, confirm prior observations at GE and AI that Cr is preferentially leached from stainless steel by low-oxygen sodium, and that iron is preferentially retained. Apparently Co is also retained. The apparent rate determining step in radioactive mass transport is collision with the pipe wall.

5.6 FISSION PRODUCT CONTROL TEST EQUIPMENT

A surplus automated cryosorptive device has been obtained and is being modified to test retention time of krypton and xenon on molecular sieves as a function of gas composition and temperature during separation from argon. Sorber bed temperatures will be maintained by liquid nitrogen-cooled Freon-13.

5.7 OPERATION OF SODIUM LOOPS AND SIMILAR EQUIPMENT

The PNL Sodium Studies Facility provides a centralized laboratory for small experimental test loops and sodium chemistry equipment. This section of the report covers the operational aspects of these test facilities and equipment. The programmatic results obtained from the various loops and equipment are covered by the appropriate FFTF program sponsors.

5.7.1 Sodium Purification Loop

The Sodium Purification Loop is a system designed to provide sodium of a known oxide content for recharging thermal convection loops, capsule studies, and static pots.

The Sodium Purification Loop operated continuously for 1000 hr at 0.5 gal/min, at 400 °F bulk temperature, and with the cold trap in service at 250 °F internal temperature. Weekly plugging runs indicate sodium oxide content to be less than 10 ppm.

5.7.2 Small Components Evaluation Loop

The Small Components Evaluation Loop is an isothermal sodium system designed to operate up to 1500 °F at 50 psi. The loop was operated at 1060 °F and 18 gal/min for 816 hr until overheating of the nickel bus and excessive noise from the EM pump forced draining of the system. The loop accumulated 5760 hr of operation under test conditions.

5.7.3 Small Heat Transfer Loop

This 1500 °F sodium facility was designed to facilitate the study of thermal hydraulics characteristics in an FFTF single pin at prototypical conditions.

The loop was installed and services connected. Checkout is 90% complete.

5.7.4 Static Sodium Pots

The Static Sodium Pots Facility was designed to assist in the study of FFTF in-core instrumentation at isothermal conditions up to 1200 °F and 250 psi.

Cycling of temperatures and pressures on transducers in Pots 1 and 3 was performed during this report period.

SECTION 6.0

CORE DEVELOPMENT6.1 HEAT TRANSFER CHARACTERISTICS OF FFTF FUEL ASSEMBLIES

The program to determine heat transfer characteristics of FFTF fuel assemblies requires heated models of FFTF fuel assemblies. Fuel element models of different maximum capabilities will be required for the three principal phases.

A major milestone was achieved in the development of electrically heated models of FFTF fuel pins for heat transfer experimentation. A rotary-swaged pin consisting of a nickel sheath, molybdenum resistance element, and boron nitride insulation was demonstrated at heat fluxes in excess of 1,500,000 Btu/hr-ft² in 334 °F water for over seven hours.

The failure point occurred 4 in. from the upstream end of the rod. Visual evidence indicated clad failure due to local overheating. Detailed destructive examination will be performed.

This fuel pin model was fabricated utilizing a variation of the basic fabrication technique employed earlier in this program. Instead of the swaging and vibrational compaction process, rotary swaging of boron nitride beads was used as the final process to produce dense ceramic layers between the sheath and heater element. Additional changes included use of a nickel sheath in place of 304 stainless steel for determining the effect of reducing the thermal coefficient of expansion, and the use of a solid 1/8 in. diam molybdenum-rod in place of the welded tantalum-molybdenum assembly.

6.2 CORE TESTS

6.2.1 Single Tube Hydraulic Balance Test

The purpose of this full-scale single tube model is to evaluate the hydraulic balance feature presently considered for the inlet plenum, and to measure the leakage rates through the annular clearance between the fuel duct and tubesheets. The main test series was completed with a test to evaluate the effect of a more prototypic multi-tube outlet arrangement. A tubular section was placed concentrically around the outlet zone of the single tube to simulate the flow interference zone between ducts. The addition of this 6-3/4 in. ID tube caused an increase of about 18 lb in the downward force upon the duct at a flow of 350 gal/min.

6.2.2 Annular Inlet Model

The purpose of this tenth size annular inlet model of the FFTF coolant supply arrangement is to assist in evaluating the mixing characteristics of the inlet streams as they flow within the annular zone formed by the main reactor vessel and the removable reactor core vessel. The annular inlet model was completed, orifice calibration conducted, and familiarization runs made. Modifications to the model, including a restrictor plate over the inlet plenum to provide back-pressure and improve flow simulation, are presently being performed in an effort to correct vortex formation problems in the lower inlet region.

6.3 REACTOR SIMULATION

6.3.1 Analog Simulation

Reactivity feedbacks due to core expansion and tube bowing were added to the analog simulation of the reactor.

Expansion is a negative reactivity effect of the form

$$\delta k_E = \alpha_E (T_{ACTR} - T_{ACTZ}),$$

where T_{ACTR} and T_{ACTZ} are average sodium temperature and a reference sodium temperature, respectively. The coefficient, α_E , equals $-4 \times 10^6 \delta k/^\circ F$. Bowing, although assumed to be positive, may eventually be eliminated by core radial restraint. The form is

$$\delta k_B = \alpha_B (T_{CTR} - T_{CIP}),$$

where T_{CTR} and T_{CIP} are the outlet and inlet temperatures of the average tube coolant, respectively. The coefficient, α_B , equals $+6 \times 10^{-7} \delta k/^\circ F$.

6.3.2 Model Development, Hybrid Simulation

The hybrid simulation of the primary, secondary, and tertiary hydraulics for two distinct coolant loops was completed. It was combined, along with a digital simulation of a four-node intermediate heat exchanger for the second coolant loop and eight transport time delays, with a completely repatched analog system for the FFTF simulation. The model has been tested and functions properly. Figures 6.1 and 6.2 show the flow decays resulting from a step change in clutch excitation current and from a loss in motor power for both primary and secondary sodium loops.

6.3.3 Transient Analysis and Simulation

A hybrid simulation of the primary motor, clutch, pump, and hydraulics compared favorably with both the analog model and the complete digital model now in use, and will provide several advantages over the present methods. The hybrid simulation will be capable of reaching three-loop design flow conditions not attainable with the present analog. This new system frees approximately 31 analog amplifiers for other

FIGURE 6.1. Flow Transients for Hybrid Simulation of FFTF Hydraulics, Primary Sodium Loops

FIGURE 6.2. Flow Transients for Hybrid Simulation of FFTF Hydraulics, Secondary Sodium Loops

uses. It also frees two a-d converters and one d-a converter. Current plans for expansion of the FFTF simulation call for two complete heat transport circuits, each with a primary and secondary loop. With four complete pump-motor-hydraulic systems, the new hybrid program will provide a savings of 124 amplifiers, eight a-d converters and four d-a converters, assuming an equivalent simulation for each loop.

6.4 STEADY-STATE LOAD CURVES FOR HYDRAULICS SYSTEM

Steady-state load curves were plotted for the hydraulics system. Figure 6.3 shows them on a set of head-flow curves for a single primary loop. The lower curve was developed by adjusting the hydraulics to equal three-loop operating conditions and then varying clutch excitation current. The upper curve is for four-loop operating conditions. Differences in the head-flow relationship for the two cases arise from the core Δp as the most responsible factor in the loop pressure drop. Hence, the change from four-loop design flow (1040 lb/sec) to three-loop design flow (1380 lb/sec) for the same total reactor flow only increases the pump head by the amount of increased loop friction losses, a total of about 70 ft.

6.5 EFFECT OF FABRICATION TOLERANCES AND SELECTIVE ASSEMBLY ON COOLANT TEMPERATURE DISTRIBUTION

The limiting character of the coolant temperature transverse profile at the end of the fuel region within the sub-assembly imposed by fabrication and assembly tolerances has been established using the COBRA⁽¹⁾ computer program. This

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

FIGURE 6.3. Head-Flow Characteristic for Primary Pump Simulation

profile includes symmetric arrangements as well as various asymmetric bundle packings and/or component arrangements.

Program results, based on thermal-hydraulic considerations, disclosed the need for adequate assembly tolerances in addition to the usual component tolerances.

6.6 EFFECT OF DUCT WALL ROUGHENING ON COOLANT CHANNELING

The presence of wire wrap (or other spacers) in the fuel subassembly results in larger coolant channels (120-170%) beyond the outer row of fuel pins than within. The consequence of this nonuniformity is a tendency for the coolant to follow a preferred path ("channeling") around the pin bundle rather than to distribute uniformly over all the channels. Less energy per unit mass will be transferred to channeled coolant than to the coolant passing up the interior of the pin bundle. Therefore, departure of the radial temperature profile from the radial power distribution, and severe temperature gradients across the outer row of pins in the hottest subassembly can be expected. The extent to which these temperature gradients can be reduced by simple roughening of the interior duct walls is explored here.

In order to simulate the thermal-hydraulic processes operating in the fuel subassembly, the COBRA digital computer code (modified version) was used.

It was found that severe roughening of the duct wall could reduce the coolant temperature drop about the hottest pin in the most tightly packed subassembly by about 50%. However, this reduction would drop to less than 30% for the loosest packing.

6.7 THERMAL HYDRAULIC ANALYSIS INSTANTANEOUS FLOW BLOCKAGE

The flow reduction event is of special interest, since an occurrence of this nature can result in pin, assembly, or even partial core meltdown. The loss-of-coolant or flow reduction accident can result from various phenomena (inlet pipe rupture, foreign objects lodged in the inlet plenum, severe pump failure, etc.). The model used for this study assumes coolant flow loss or reduction throughout an entire fuel assembly. Results of the flow blockage study are presented in the following tabulation:

<u>Percent Blockage</u>	<u>Sodium Boils, sec</u>	<u>Cladding Melts, sec</u>	<u>Fuel Center-line Melts, sec</u>	<u>Fuel Surface Melts, sec</u>
100	0.92	2.9	1.8	3.9
90	0.72	1.95	2.1	19.8
80	0.74	4.3	2.5	85
70	0.93	225	3.0	575
60	20	1800	4.5	1250
50	2500	9500	11.0	51500

6.8 REACTOR SCRAM TRANSIENTS

Digital simulation models were used to study typical reactor temperature transients after a wide variety of rapid shutdowns (scrams). Typical results for a typical type of scram (a so-called manual scram) are given in Figures 6.4, 6.5, and 6.6. The manual reactor scram transients consists of:

- 1 g Rod Acceleration Scram - 15\$ Total Reactivity
- Time Delay Between Scram and Flow Coastdown
- Flow Coastdown.

FIGURE 6.4. Reactor Temperature Responses for Manual Scram with Coastdown Delayed 1 sec; Coastdown Initiated by Clutch De-energization; 1 sec Clutch Time Constant

FIGURE 6.5. Reactor Temperature Responses for Manual Scram with Coastdown Delayed 1 sec; Coastdown Initiated by Clutch De-energization; 2 sec Clutch Time Constant

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FIGURE 6.6. Reactor Temperature Responses for Manual Scram with Flow Coastdown Delayed 1/2 sec; Coastdown Initiated by Clutch De-energization; 1/2 sec Clutch Time Constant

SECTION 7.0

REACTOR MATERIALS7.1 EFFECT OF IRRADIATION ON PROPERTIES OF STAINLESS STEELS

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 currently in progress is specifically directed at providing a basis for selection of optimum alloys for FFTF design and operation.

7.1.1 Irradiation Swelling of Stainless Steel

Based on a summary of stainless steel volume expansion data at exposures to 8×10^{22} nvt, it appears that volume may increase as much as 10% at 10^{23} nvt and irradiation temperatures of 800 °F to 1100 °F.

The specific relationship between neutron fluence, irradiation temperature, and swelling cannot presently be established since data are not available over a sufficiently large range of experimental conditions. If the data for 304 and 316 are plotted together, and the difference in chemistry between these alloys is ignored together with irradiation temperature variations among the data, a tentative description of swelling as a function of fluence results. This relationship, presented in Figure 7.1 includes data from ORNL (from transmission electron microscopy), PNL (both electron microscope and density), and UKAEA (DFR). The electron microscopy results do not correlate closely with the actual density measurements because of errors in estimates of foil thickness and poor resolution of void diameters less than about 50 Å.

7.1.2 Effect of Irradiation on Strength of Stainless Steel

The effects of fast reactor irradiation on the strength of Type 304 SS EBR-II capsule sections are being studied. The proportional limit of ring sections taken from a capsule accumulating 2×10^{22} n/cm² total neutron fluence at about 750 °F was determined as a function of axial position and as a function of annealing temperature for sections near the midplane. Strength increases along the length of the capsule pin described a smooth curve. Increases in strength of about 6 ksi were found near the upper end of the capsule (above core), while near the midplane a maximum increase of 12 ksi was measured.

Rings cut from the midplane region were annealed for one hour at temperatures from 200 °F to 1800 °F in 100 °F increments. After annealing, the rings were tested at room temperature. No change in proportional elastic limit was found at annealing temperatures less than 1000 °F. The strength, rapidly decreasing above this temperature, reached a minimum value of 1300 °F. This behavior is in contrast to previously reported results showing irradiation strengthening after EBR-II irradiation to temperatures as high as 1500 °F. The increase in strength up to 1500 °F was attributed to cavity formation during irradiation. The lack of elevated temperature hardening in the ring tests suggests the absence of cavities in the structure. However, it is now known that cavity size remains small at irradiation temperatures near 750 °F. Small cavities anneal out at much lower temperatures than do large cavities. Therefore, the absence of elevated temperature hardening may be due to cavities annealing out below 1300 °F rather than the absence of cavities *per se*.

7.1.3 Microstructure of Irradiated Stainless Steel

An electron microscopy examination was completed on Type 304 stainless steel having the following history:

FIGURE 7.1. *Irradiation Swelling of Austenitic Stainless Steel*

- (1) Irradiated in the ETR at 554 °F (290 °C) to 6×10^{21} n/cm² (>1 MeV)
- (2) Tensile tested at 932 °F (500 °C)
- (3) Vacuum annealed for one hour at 1796 °F (980 °C)
- (4) Annealed one hour at 1596 °F (870 °C) in NaK at a pressure of 10,000 atmospheres.

Hardening of the irradiated steel at 1382 °F (750 °C) could be accounted for by the cavities. The cavities in the irradiated material persisted after vacuum annealing at temperatures of 1596, 1796, and 1994 °F (870, 980, and 1090 °C). The total surface area of the cavities was constant over the temperature range investigated, while the total volume of the cavities increased. The results indicate balancing of the internal gas pressure by the surface tension of the cavities. However, the amount of helium gas required to balance the surface tension of the cavities was more than 10 times the helium produced by transmutation.

The microstructure of the irradiated steel subjected to the high pressure anneal provided positive information about the nature of the cavities. The preliminary examination of the electron micrographs (Figure 7.2) provided the following observations:

- Approximately 2×10^{13} cavities/cm³ with an average diameter of about 700 Å were still present after annealing the material one hour at 1596 °F (870 °C) and 10,000 atm.
- There were no observable strain fields around the cavities.
- About 10^{13} precipitates/cm³ with an average diameter of about 400 Å were observed. The majority of the polyhedral precipitates were attached to cavities.

FIGURE 7.2. *Microstructure of Irradiation
Hardened Type 304 Stainless Steel*

- Approximately 10^{14} small particles/cm³ with an average diameter of 100 Å were observed. These small, uniformly dispersed particles were not vacancy or interstitial clusters.

The preliminary estimate of the amount of helium gas required to balance the 10,000 atmosphere applied pressure, plus the cavity surface tension (assuming $\gamma = 1500$ ergs/cm²), was about 800 ppm. The results indicate that the cavities do not respond like voids but behave like gas-filled bubbles.

7.1.4 Effect of Irradiation on Tensile Strength

The effect of irradiation on stainless steel in a flowing sodium environment at elevated temperatures has been investigated. Major emphasis was placed on determining changes in tensile properties.

Type 316 stainless steel tensile specimens were irradiated in two flowing sodium capsules, GEH-22-3 and GEH-22-4. The capsule design and operation has been described previously.⁽¹⁾ The irradiation conditions for the specimens are reviewed in Table 7.I.

(1) T. T. Claudson. Design and Fabrication of a Liquid Metal Temperature Controlled Irradiation Capsule, CONF-660511. 1966.

Table 7.I. *Irradiation Conditions in the Flowing Sodium Capsules^(a)*

<u>Capsule</u>	<u>Level</u>	<u>Irradiation Temperature</u>	<u>Fast Neutron Fluence (E > 1 MeV)</u>	<u>Time at Irradiation Temperature</u>
GEH-22-3	1 & 2	1112 ± 90 °F	~1.4 x 10 ²¹	~100 days
GEH-22-3	3 & 4	1004 ± 90 °F	~1.4 x 10 ²¹	~100 days
GEH-22-4	1	1400 ± 45 °F	~8.0 x 10 ²⁰	~ 55 days
GEH-22-4	2	1382 ± 45 °F	~8.0 x 10 ²⁰	~ 55 days
GEH-22-4	3	1346 ± 45 °F	~8.0 x 10 ²⁰	~ 55 days
GEH-22-4	4	1274 ± 45 °F	~8.0 x 10 ²⁰	~ 55 days
GEH-22-4	5	1202 ± 45 °F	~8.0 x 10 ²⁰	~ 55 days

a. The irradiations were in the core of the Engineering Test Reactor.

The tensile specimens from GEH-22-3 were tested at room temperature, and at 572, 752, 932, 1112, 1292 and 1382 °F. Their strength data show that irradiations at approximately 1112 °F and 1004 °F result in increased strength (compared to as-fabricated controls) over the entire range of test temperatures, with the greater increase occurring for the 1004 °F irradiation. Typical results are shown in Figure 7.3. The uniform and total elongations are lower at all test temperatures than are the as-fabricated controls.

Tensile data were also obtained for AISI Type 304 SS, irradiated in the flowing sodium capsule GEH-22-3 and tested at 572, 752, 932, 1112, 1292, and 1382 °F. Typical results are given in Figure 7.4. Type 304, like the Type 316, shows an increase in strength and a decrease in ductility at all test

FIGURE 7.3. The Effect of Test Temperature on the Strength of AISI 316 Stainless Steel Irradiated in a Flowing Sodium Environment (GEH-22-3)

FIGURE 7.4. The Effect of Test Temperature on the Strength of AISI 304 Stainless Steel Irradiated in a Flowing Sodium Environment (GEH-22-3)

temperatures. The strength difference between 1112 °F and 1004 °F irradiation temperatures is not observed for the Type 304 stainless steel.

The irradiation conditions for a second flowing sodium capsule, GEH-22-4, are reviewed in Table 7.I. It can be noted that irradiation temperatures are slightly higher than for GEH-22-3. The Type 316 specimens from this capsule have been tested from room temperature to 1382 °F. The specimens tested from this capsule showed increased yield strengths when compared to as-fabricated controls, although the irradiation temperatures were higher and the fast fluence was moderate. The combined effect of 1274 °F or 1382 °F irradiation to a fluence of 8.0×10^{20} (E > 1 MeV) has produced some effective hardening up to and including the 1382 °F test temperature. The ductility of the irradiated Type 316 specimens are lower than as-fabricated controls for all test temperatures.

7.2 STRESS-TO-RUPTURE DATA FOR 304 STAINLESS STEEL

A limited quantity of unirradiated stress-to-rupture data for Type 304 stainless steel has been developed for testing at 1400 °F (760 °C) in a helium atmosphere. Of primary interest was the "as-received" material containing a slight degree of cold work from the final fabrication step. The "as-received" material was tested at stresses sufficient to cause rupture in the range of 0.1 hr to 139 hr, as shown in Figure 7.5. Identical material, annealed at 1950 °F (1066 °C) for 30 min in vacuum, has been tested at stresses sufficient to cause rupture in the range of 0.2 to 78 hours. The data establish the tendency of the as-received and annealed curves to merge at rupture times of about 100 hours. The data from the annealed material agree with published values and appear to be representative of Type 304 stainless steel.

Post-test comparisons of grain size and microstructural characteristics with untested and tested 304 stainless steel cladding material can be made in Figure 7.6 (as-received condition) and in Figure 7.7 (annealed condition). The following general observations can be made for both material conditions.

- The growth of carbides is seen to increase with time-at-temperature. Cold work present in the as-received test material has had no discernible effect on the size or location of the carbide precipitate.
- The nominal grain size remains unchanged throughout the test, for the test times available.

FIGURE 7.5. *Stress-to-Rupture Properties of As-Received (Slightly Cold Worked) and Annealed Type 304 Stainless Steel Prototype FFTF Fuel Cladding*

FIGURE 7.6. Grain Size and Microstructure of As-Received Type 304 Stainless Steel Prototype Fuel Cladding in Untested Condition and After Rupture at 1400 °F (760 °C). (Magnification shown is before printing.)

FIGURE 7.7. Grain Size and Microstructure of Annealed Type 304 Stainless Steel Prototype Fuel Cladding in Untested Condition and After Rupture at 1400 °F (760 °C.) (Magnification shown is before printing.)

7.3 ANALYSIS OF RADIATION DAMAGE PROCESSES

7.3.1 Materials Damage Analyses

Materials damage analyses are being made to determine effects of neutron irradiation on materials and components in the FFTF. The SPECTRA program series has been used to develop analytical data pertinent to the analysis of the void generation damage mechanism. PKA spectra have been calculated for two 304 stainless steel specimens, a tensile specimen, and a specimen equal in thickness to the FTR fuel clad. PKA spectra are compiled for each of the three primary elements of 304 stainless steel in each specimen geometry and for irradiations in DFR, EBR-II, FTR, and ETR spectra.

A graphical comparison of the PKA spectra is given in Figure 7.8. Equal areas of the histogram represent equal probabilities or fractional PKA population. The cumulative products of all interval widths and the probability height is equal to unity. Beeler⁽¹⁾ has shown the PKA population above 3 KeV to contribute primarily to the stable vacancy cluster production and thus, it can be assumed, to void density. Applying this criterion and its implied assumptions to the PKA spectra yields the following conclusions:

- Specimens irradiated in DFR and EBR-II incur larger populations of high energy PKA's than do specimens irradiated in the FTR fuel region (Figure 7.8), thus indicating that a FTR material design based on EBR-II and DFR data would be conservative. The ranking of

(1) J. R. Beeler, Jr. "Vacancy and Interstitial Cluster Production in Neutron Irradiated α -Iron," Journal of Applied Physics, vol. 37, no. 8, p. 3000. 1966.

the spectra is EBR-II > DFR > FTR based on the number of high energy PKA per interaction. All neutrons above 25 eV are considered. The ranking of ETR is still being studied.

- PKA spectra calculations for fuel element cladding show the same relative damage rating as the tensile specimen calculations. A direct comparison of PKA spectra for the different specimens in a given environment shows that irradiation-produced void density data collected on tensile specimens is applicable to fuel element cladding design.
- The major elements of stainless steel produce different PKA spectra. However, these calculations show nearly identical PKA spectra for iron and chromium. Nickel PKA spectra differ slightly, but the deviations do not modify the relative damage rating among the irradiation environments.

These PKA spectral data will be used as the basis of a quantitative estimate of the vacancy clustering, or micro void density, produced in each of the four irradiation environments. The assumptions for that estimate have been chosen as follows:

- Vacancy cluster sizes will be for a crystal lattice at 0 °K.
- The crystal lattice is BCC, α -iron, not FCC, for stainless steel.
- The crystal lattice is perfect, undisturbed by previous or nearby PKA cascades.

7.3.2 Analysis of Radiation Damage at Elevated Temperatures

An essential element in applying damage calculations to problems of interest to the FFTF Project is the introduction of finite temperature effects in accounting for the thermally produced changes in the initial damage state that occur during irradiation at elevated temperatures.

FIGURE 7.8. Probable Energy of Primary Knock-On Atoms (PKA) Produced in Iron

A computer code identified as ANNEAL and developed by J. R. Beeler and D. Besco at General Electric Company to simulate annealing behavior was obtained.

The ANNEAL code, although satisfactorily run on the Univac 1108 computer is, in its present form, seriously limited in scope. It is set up to simulate the annealing of pure α -iron (body-centered cubic) at essentially room temperature. Only single and di-interstitials are mobile at this temperature.

Of primary interest to the FFTF Project is the study of radiation damage at higher temperatures (800 - 1200 °F) where vacancies and small vacancy clusters are also mobile. Therefore, a concentrated effort to cover this condition by extending ANNEAL is in progress.

The activation energies for motion of vacancies in the vicinity of vacancy clusters are being determined with the computer code DEFECT. The initial runs are yielding higher activation energies than were anticipated from published data.⁽¹⁾ The significance of this work will not be known until more runs are completed, since the purpose was to seek general modes of behavior for inclusion in the annealing simulation code ANNEAL.

7.4 PRESSURE STRESS IN FUEL DUCTS

Pressure stress calculations for the present design of the fuel duct indicate that an 85 psi pressure differential would induce significant tube expansion. Experimental tests have verified the calculations.

(1) J. R. Beeler and R. A. Johnson. "Vacancy Clusters in α -Iron," Phys. Rev., 156, 677. 1967.

7.4.1 Calculations

Pressure stress levels in flow ducts as now designed may lead to significant tube expansion. Differential pressures within a given flow duct are expected to be about 85 psi at the bottom of the active core, decreasing to about 45 psi at the top.⁽¹⁾ The hexagonal shape results in high stresses in the corners of the ducts. Based on actual dimensions of a recently fabricated 304 tube for CCTL testing, an 85 psi pressure will produce a maximum corner stress of about 17,000 psi.⁽²⁾ This stress would induce yielding of the corners at 400 °F based on minimum yield strength values for 304.⁽³⁾ The corresponding elastic wall deflection would be about .001 to .003 in. at 1000 °F and, depending on the degree of corner yielding, could be considerably larger. In addition, the effect of reactor pressure and temperature cycling, tube hot channel factors, increased pressures, and thermal bowing of the tube could also lead to wall deflection through creep deformation.

7.4.2 Experimental Program

A test evaluation program has been initiated to experimentally assess the effects of some of the variables identified analytically.⁽²⁾ A 30-in. long piece of a hexagonal section 304 CCTL tube was end-capped for an initial pressure-temperature evaluation. Flat-to-flat and corner-to-corner

(1) M. T. Jakub. *Pacific Northwest Laboratory, February 1968. Private Communication.*

(2) C. D. Flowers. Structural Analysis of a Hexagonal Tube or Ring Under Internal Pressure, BNWL-263. *Pacific Northwest Laboratory, Richland, Washington. June 1966.*

(3) W. F. Simmons and H. C. Cross. The Elevated-Temperature Properties of Stainless Steels, ASTM Stp. No. 124, p. 11. 1952.

measurements were taken at about the tube quarter points before, during, and after pressurizing to 120 psi. Essentially no change occurred in the corner-to-corner dimensions. The wall-to-wall dimensions, however, increased from 0.0015 to 0.0035 in. with an average increase of 0.0027 in. while pressurized. No permanent set was noted after depressurization. Based on an operating pressure of 85 psi, the 0.0027 in. room temperature displacement would be equivalent to about 0.003 in. at 1000 °F. Displacement calculations based on analytical equations⁽¹⁾ and using the same tube dimensions and pressure and temperature levels would be about 0.002 inches.

Determination of yielding and subsequent creep of the tube corners at possible operating temperature and pressure conditions has begun. A tube suspended within a furnace and statically pressurized to 85 psi at 1000 °F will be periodically depressurized, cooled, and measured at the same test pressure. The creep rate for this pressure, or a stress level of about 17,000 psi, is in the order of 0.0001%/hr, and thus a 1000-2000 hr test should be sufficient to detect wall displacement due to corner creep.

7.5 CHEMICAL COMPATIBILITY OF SHIELD AND CONTROL ROD MATERIALS

The interaction of potential shielding media with cladding material is under study to determine limitations of their application in the FFTF. During this report period, mild steel and austenitic stainless steel were held at 930 °F for 336 hr in contact with graphite and carbon. Thermal soaking was performed under one atmosphere of helium.

(1) C. D. Flowers. Structural Analysis of a Hexagonal Tube or Ring Under Internal Pressure, BNWL-263, Pacific Northwest Laboratory, Richland, Washington. June 1966.

The results showed both the carbon and graphite to react in a similar manner with all the metal samples. Metallographic analysis and microhardness traverses are presently in progress so that a correlation may be obtained between the amount of carburization observed experimentally and that predicted from theoretical considerations of carbon diffusion in the alloys.

7.6 FFTF MATERIALS DESIGN DATA

7.6.1 Design Data Manual

The intent of the FFTF Materials Design Data Manual is to provide design curves for material identified as potentially useful in the FFTF. The design curves (or values) will be adopted and used as standardized information in the FFTF Project.

A summary of data collected on plastics was distributed to RDT and the FFTF Project. The 69 draft summaries listed a number of mechanical, physical, and chemical properties, primarily at room temperature. Two of the most important properties in these data compilations are maximum useful temperature and irradiation damage characteristics. Table 7.II summarizes the type of material contained in the data collection and the manner in which it might be used in specifying plastics for FFTF applications.

Preparation of draft summaries or data collections was completed for graphite (electrode grade), carbon, and tantalum. Little or no irradiation effects data are presented in these summaries because very little is recorded in the literature. However, a relatively large amount of irradiation data on nuclear-grade graphite, to be presented on this subject in a later draft summary, will be applicable to the electrode-grade graphites under investigation at BNW.

An examination of the literature is continuing to ensure the availability of up-to-date radiation effects data for austenitic stainless steels, 304 and 316.

TABLE 7.II. *Data Summary on Potentially Useful Plastics*

Useful Materials for Range of Temperature 450 - 550 °F	Resistance to Alkalies	Radiation Effects
Polymide	Dissolves in 10% NaOH	Damage threshold 7×10^{11} ergs/g (C)
Polyphenylsulfone	Resistant	Unknown
Polytetrafluoroethylene	Resistant	Damage threshold 3.7×10^6 ergs/g (C)
(Following materials are promising but unproven)		
Alkyd, molding cpd- filled	Attacked	Gas evolved - 0.08 - 1.9 molecules/100 ev absorbed
Diallyl-Phthallate - glass filled	Slight attack	Little or no effects to 8.8×10^{12} ergs/g (C)
Epoxy resin - cast, unfilled	Resistant	Little or no effect to 9.5×10^{10} ergs/g (C)
Epoxy molding cpd- filled	Resistant	Little or no effect to 9.5×10^{10} ergs/g (C)
Epoxy resin - cast filled	Resistant	Little or no effect to 9.5×10^{10} ergs/g (C)
Phenol-formaldehyde molding cpd - inorg. filled	Attacked	Gas evolved < 0.08 mole- cules per 100 ev absorbed

7.6.2 Derivation of Data

A computer program is now in place to produce statistical materials design curves. These curves will show, with three-sigma limits, the average value of a particular property as a function of temperature. Also to be shown is the average value of a particular property as a function of temperature and, as a function of fluence level with 3-sigma limits.

A comprehensive literature review on B_4C was received. These data will subsequently be condensed into the standard draft summary format.

Work is currently in progress to assemble a "second edition" of the data collection on irradiation effects to 304 and 316 stainless steel.

A study aimed at determining the applicability of existing ASME design curves to the design of the FTR pressure vessel is in progress. The primary part of this study is involved in determining the effect of total neutron fluence on the tensile, stress-rupture, and creep properties of the materials under consideration (304 and 316 SS). The existing stress-rupture data indicate a decrease due to neutron exposure in the stress-to-rupture of about 25 - 35% at 1200 °F. However, these data were obtained over relatively short test periods (less than 1000 hr) and relatively high stress levels (at about the yield strength). Furthermore, the area on the vessel receiving even a modest exposure will operate below the lower creep limit (about 950 °F). Efforts to determine the applicability of the data are continuing.

The design curves for fatigue of 304 or 316 SS at temperatures from 800 to 1200 °F will be reviewed to insure an adequate safety factor. Recent work has indicated that the fatigue curves in Code Case 1331 may not be as conservative as originally believed because of increased hold times between cycles.

7.6.3 Codes and Standards

Consultation and assistance to the FFTF in matters pertaining to engineering codes is being provided through the consulting services of W. E. Cooper, Teledyne Materials Research Corporation. Dr. Cooper is a member of the ASME Subcommittee

on Nuclear Power and a recognized authority in the field of code interpretation and stress analysis. Initial efforts will be to insure adequate interpretation and application of existing codes (nuclear and nonnuclear) in the preliminary design of the FFTF.

7.7 IRRADIATION BEHAVIOR NON-METALLIC MATERIALS

7.7.1 Radiation Effects on Absorbing Materials for FTR Control Rods

Boron carbide is the prime candidate material for use in FTR control rods, with tantalum as its alternate. The irradiation behavior of both materials at FTR operating conditions must be determined in order to provide data for the selection, design, and procurement of the better absorber.

Final capsule design has been established for the irradiation of B_4C and Ta in an EBR-II seven-pin subassembly. As-received B_4C pellets were found to be dimensionally out of specification and will require considerable grinding prior to their use. Characterization of boron carbide, already started, includes chemical analyses of B/C ratio and impurity levels, and metallographic examination to determine grain size, phases present, porosity size, and porosity location.

7.7.2 Irradiation Behavior of FTR Shield Materials

The irradiation behavior of potential shield materials, other than nuclear graphite, is under investigation to determine the most economical (but adequate) shielding material for use in FTR.

Seventy-two capsules containing carbon and graphite samples are being irradiated in the ETR. Preirradiation measurements of thermal conductivity from 70 to 800 °F were completed. The thermal conductivity at 70 °F of three types of graphite ranged from 0.71 to 1.19 watts/cm-°C in the direction transverse to

the extrusion axis. The conductivity of five types of carbon at 70 °F ranged from 0.021 to 0.120 watts/cm-°C in the transverse direction. Stored energy measurements are in progress. The discharge of 36 capsules is scheduled for April, 1968.

The samples are quarter-round cylinders 2 in. in length and 0.5 in. in radius. Each capsule contains eight samples and monitors for temperature and flux. The samples are being irradiated at temperatures ranging from 200 to 800 °F and fluences from 5×10^{19} to 5×10^{20} nvt.

7.8 CLEANING OF TUBES AND COMPONENTS

Procedures for cleaning the components of fuel ducts and closed loops are under investigation. All components exposed to the sodium coolant system must be clean to avoid contaminating the system either with nuclear poisons or with materials that might adversely affect the coolant or components associated with the coolant system. In addition, special cleaning requirements arise during component fabrication and preassembly. For example, it is imperative that all surface soil possibly able to mask surface imperfections be removed prior to nondestructive testing when using dye penetrant techniques.

The specific problems associated with the cleaning of Type 304 stainless steel fuel ducts include the removal of asphalt-base drawing lubricant as well as removal of normal soils such as various marking materials, fingerprints, light oils, etc. An acceptable procedure for removal of surface oxides formed during annealing is also being sought. Recommendations of products adequate for the job and meeting the product criteria established in Provisional Product Specification for FFTF Fuel Cladding, No. PNL 70.0, (Specification for Cleaning FFTF Core Components) have been requested from several manufacturers specializing in chemical processing compounds. Recommended

products are being tested as they are received. Although this testing program has not been completed, the following observations have been made:

- The asphalt-base lubricant and several of the marking materials commonly used are not adequately removed by the heavy duty alkaline type soak cleaners tried to date.
- Neither alkaline cleaners nor sulfamic acid type descalers will remove surface oxides formed during annealing in an oxidizing atmosphere.
- There are considerable differences in the degree of removal of various marking materials.
- Manufacturing representatives have expressed concern about the low ppm restriction on Section 2.3 of Specification PNL 70.0 pertaining to the cleaning agents themselves, as opposed to such restrictions applying only to the as-formulated cleaning solution.

7.9 TEST FACILITIES

The demand for comprehensive investigation of FFTF candidate structural and cladding materials has created the need for a facility to provide the necessary test environments. The facility, presently under construction, is approximately 85% complete. This facility will be capable of testing irradiated and nonirradiated specimens in sodium and controlled gas atmospheres for uniaxial and biaxial creep and stress-to-rupture tests.

SECTION 8.0

FUELS DEVELOPMENT8.1 217-PIN FLOW TEST SUBASSEMBLY

The FFTF prototypic 217-pin flow test subassembly was shipped to Argonne National Laboratory for sodium flow testing in the CCTL. Figure 8.1 shows the completed subassembly (217 pins, 0.030 mil wire spacers, and end grids) held vertically in the assembly fixture. Figure 8.2 is a closeup of the top grid. The fuel duct is lowered over the subassembly and the subassembly positioned by three locking bars passing through the grid and welded to the duct. The assembled unit is then lowered to the horizontal position for examination (Figure 8.3).

Figure 8.4 shows a view of the top of the subassembly in position within the fuel duct. The exit holes for the sodium coolant can be seen in the wall of the flow duct.

The completed assembly is then sealed in plastic for dust protection and placed in a padded, fitted inner shipping box. Figure 8.5 shows the completed assembly, without the plastic dust protector, in the inner shipping box. This sealed inner box is placed within a larger protective crate for shipping.

In preparing the subassembly, the fuel pins are first pre-assembled into strip layers or rows (Figure 8.6). A technique developed for pinning and welding the fuel pin bottom end caps to the grid strips has worked satisfactorily (Figure 8.7). The strip layers, comprising varied numbers of pins as required by their subassembly positions, are individually welded in turn to a bottom grid plate (Figures 8.8 through 8.10) in building up the hexagonal array shown in Figure 8.11. No major assembly problems were encountered.

Neg 0680570-4

FIGURE 8.1. Completed 217-Pin Subassembly
Held in Assembly Fixture

Neg 0680570

FIGURE 8.2. *Closeup of Top Grid in Place
on the Subassembly*

Neg 0680570-48

FIGURE 8.3. Examination of Completed Fuel Subassembly-Fuel Duct Assembly

Neg 0680570-30

FIGURE 8.4. *View of the Top of the Subassembly
Within the Fuel Duct*

Neg 0680570-40

FIGURE 8.5. Completed Fuel Assembly in Inner Shipping Box

Neg 0680570-6

FIGURE 8.6. Assembling Strip Layers

Neg 0680570-8

FIGURE 8.7. *Welding of Strip Layer Retaining Pins to End Cap*

Neg 0680570-2

FIGURE 8.8. *Positioning Strip Layer in Bottom Grid Plate
(Hardback Tray Supports Grid Strip Layer)*

Neg 0680570-15

FIGURE 8.9. *Bottom Grid Plate with Five Strip Layers
Welded in Position*

Neg 0680570-1

FIGURE 8.10. *Welding Strip Layer to Bottom Grid Plate*

Neg 0680570-24

FIGURE 8.11. *Bottom Grid Plate and Partial View
of Completed Hexagonal Array*

Two metallographic sections taken transversely through a bottom end cap in the region of the pin and weld are shown in Figure 8.12. The photographs show the end cap (A) slotted to fit over the grid strip (B). The pin (C) is then placed through a hole in both the cap and the grid strip and is spot welded.

8.2 UPDATING OF THE FTR DRIVER FUEL PIN DESIGN BASES

An interim updating of the FTR driver fuel design bases and the reference FTR driver fuel was made. This review incorporated the latest available preliminary information on the behavior of stainless-steel-clad mixed oxide fuel irradiated in the DFR and EBR-II reactors.

The study shows that, at FTR goal fuel exposures, diametral expansion of the fuel pins is unavoidable because of fission product-induced swelling and growth of the fuel and the density decrease of the cladding due to radiation damage. This conclusion is diametrically opposed to that derived from a previous study based on data available in April 1967, and on the more recent data obtained from constant intensive surveillance of world wide fast reactor fuel development programs.

Detailed analyses of the bases for the fuel pin design show no substantial technical justification for departing from the previously established reference design, although these design bases have been modified as a consequence of the more recent data.

The recommended FTR driver fuel pin design parameters remain as previously established:

Fuel pellet density	$93 \pm 2\%$ TD
Cold fuel-cladding diametral gap	0.006 ± 0.0015 in.

Cold fuel smeared density	88 $\begin{smallmatrix} +2.8 \\ -2.6 \end{smallmatrix}$ % TD
Pellet geometry	solid, right circular cylinders with flat ends. L/D = 1.18

Some highlights of current design requirements for the First Core Fuel Assembly include:

- Mixed $\text{PuO}_2\text{-UO}_2$ fuel with a smeared density of 88% TD
- Fuel column length of 30 to 36 in.
- Cladding outside diameter in the range of 0.210 to 0.260 in.
- Pins spaced in a triangular array, with spacing in the range of 1.10 to 1.20 pitch-to-diameter
- Pins self-contained, gas bonded and hermetically sealed
- Fission gas plenum located at the top of the fuel pin
- Top reflector incorporated into each fuel pin, 6 to 12 in. long, made of nickel or high nickel alloy
- Design goal to achieve a maximum burnup of 80,000 MWd/tonne
- Design goal capability of operating at 1100 °F maximum cladding ID temperature for 12 months continuous operation
- Pin bundle region pressure drop of 60 to 100 psi at full flow
- Maximum allowable sodium velocity of 35 ft/sec
- Total first core power from 350 to 450 MW (thermal), exclusive of the closed loops
- Mean outlet of 800 to 900 °F with an inlet of 500 to 600 °F and a mean core subassembly ΔT of 250 to 350 °F.

Neg 568-5143C

Neg 568-5143A

FIGURE 8.12. Transverse Metallographic Sections Welded Through the Bottom End Cap Where it is Pinned to the Grid Strip. A: End Cap; B: Grid Strip; C: Pin

8.3 WELDING DEVELOPMENTS

A preliminary weld procedure for 304 SS end closures (Phase I of the weld development program) produced the section of a typical weld joint shown in Figure 8.13. This portion of the program will be repeated for 316 SS as soon as material is available. Although larger in diameter than FTR cladding, this material will provide experience in the welding of this alloy. The data obtained will be used in establishing the 316 SS preliminary weld procedure. FTR size tubing (316 SS) will be used to complete Phase I of the welding program.

8.4 GAS COOLING OF FFTF FUEL ASSEMBLIES DURING INSPECTION

The objective of this program is to establish the feasibility of removing decay heat from FFTF fuel assemblies preparatory to fuel inspection, interim inspection, and disassembly by the forced circulation of an inert gas stream.

The fabrication of the gas heat transfer facility was completed, and preliminary tests to verify the design characteristics are underway. This facility consists of two separate gas blowers providing an axial and transverse flow to an electrically-heated model of a proposed FFTF driver fuel assembly. These blowers may be operated simultaneously or independently. Air is used as the cooling gas. Test conditions will be established from accepted similarity principles to simulate the cooling capabilities of the proposed inert gas.

The operating capabilities of this facility are as follows:

Flow rates (70 °F air at 1 atm)

Axial 0-500 ft³/min

Transverse 0-2300 ft³/min

Available pressure differential at test section

Axial 15 psi maximum

Transverse 3 psi maximum

All components of the electrically-heated model of the FFTF driver fuel have been completed, and final assembly is in progress. This test section consists of 169 pins of 1/4-in. OD and 3 ft in length capable of producing 40 kW total.

8.5 WIRE WRAP BEHAVIOR DURING FUEL PIN SWELLING

Prototypic wire-wrapped cladding tubing was tested to simulate the effect of fuel pin swelling. There was no observable deformation of cladding by wires.

Empty, wire-wrapped cladding tubes were internally pressurized with gas. Figure 8.14 shows a typical result of this test. The failed tube burst at a pressure of 10,200 psig. Release of contained gas caused the failure zone to propagate as shown. The second tube was pressurized to 8000 psig for 1 min. In neither case did wire-wrap-produced deformation of the tubes occur.

8.6 REVISED λ kdT CURVES

Recently published information on a revised approach to the density correction for the thermal conductivity of mixed oxide fuels⁽¹⁾ is being incorporated in the basic FFTF fuel design curves. Figure 8.15 shows recommended plots of thermal conductivity and λ kdT versus T for the current fuel pin design. These supersede the curves previously used in the SINTER, TIGER, and MIDAS FTR Design Programs.

(1) W. E. Bailey et al. "Thermal Conductivity of U-Pu Oxide Fuels," AIME Nuclear Metallurgy Symposium, Phoenix, Arizona, Oct. 4-6, 1967.

Neg 467-1882A

Neg 467-1882B

FIGURE 8.13. *Typical FFTF Fuel Pin Closure Weld (304 SS)*

Neg 0680647-2

FIGURE 8.14. *Wire-Wrapped Burst Specimens*

FIGURE 8.15. Revised FTR $\int k dT$ Versus Temperature for Design

Figure 8.16 shows a comparison of the original and revised end-of-life fuel design curves. The thermal conductivity has been raised in the low temperature region but remains at a lower, constant value in the high temperature region. The net effect is that the $\int k dt$ end point at the melting temperature is increased and the shape of this curve is changed, thus resulting in a smaller operational linear power value to cause melting ($\sim 8\%$ if $T_{\text{surface}} = 1000^\circ\text{C}$). A significant aspect of lower thermal conductivity, computed on the revised bases for the high temperature region, is that postulated accidents will raise peak fuel temperatures to a greater degree during transients in which large portions of the fuel are at elevated temperatures.

8.7 PELLET END CONFIGURATION: FLAT VERSUS DISHED ENDS

Fuel swelling and growth behavior as manifested by radial deformation of the cladding is relatively insensitive to the initial fuel form. Consequently, selection of the pellet end configuration should be based on consideration of any possible advantages to be obtained from dishend and/or beveled-edge pellets with regard to fuel column stability and minimum fuel-cladding interactions at pellet-to-pellet interfaces. Existing experimental data on behavior of flat and dishend pellets have been reviewed for application to the operating conditions and requirements for the FTR driver fuel. It was concluded that flat-end pellets should be retained as the reference FTR design.

Advantages of the dishend pellets over the flat end type would appear to be:

- Less axial expansion by approximately 0.5%, on the average.

FIGURE 8.16. Comparison of Original and Revised $\int k dT$ Versus Temperature

- Slightly more reproducible axial expansion, although the flat end pellets merely reduce their $\Delta L/L$ to a greater degree than the dished end pellets with high temperature thermal cycling.

The disadvantage of using dished end pellets over flat end pellets appears to derive from the tendency of cladding to ridge when the pellet stack axial expansion is restricted by fuel-cladding interactions.

While pellet end configuration effects on fuel swelling with burnup have not been investigated, coalescence of the central regions of dished end pellets due to thermal effects has been observed in out-of-pile tests, as well as during in-pile tests with low burnups. This observation indicates that, for the conditions investigated, thermal expansion has effectively removed the difference between flat and dished end pellets before burnup swelling has begun.

8.8 VIBRATION TEST DEVELOPMENT

A method of obtaining fuel pin vibration displacements and frequencies is to measure the change in conductivity as electrodes move through an electrolyte. This method is being developed for application to a 19-pin bundle of prototypic FFTF fuel pins. A test section has been designed to permit conductivity probe instrumentation as well as to permit viewing of the pins throughout most of their length. Water, with 0.1% NaNO_3 to form the electrolyte, will flow past the pins to induce vibrations.

Extensive data were acquired by using a conductivity cell to determine the impedance characteristics. Figure 8.17 shows the test basin.

FIGURE 8.17. Test Basin

Two significant pieces of information were derived from the impedance studies. First, a cell capacitance between the electrodes becomes important at low frequencies (around 20 kHz). Frequencies near 100 kHz greatly reduce this effect. Higher frequencies, although reducing electrolyte capacitive reactance, would require compensation for cable capacitance in parallel with cell.

The second piece of information was that aging occurs on stainless steel surfaces immersed in electrolyte, i.e., the impedance of the system increases with time when starting with polished electrodes. The rate of increase seems proportional to time $^{1/2}$. Leaving the electrodes in the electrolyte for about 1 week should greatly reduce this problem.

8.9 SUBASSEMBLY

The seven-rod sodium flow test cluster was recently removed from test in flowing sodium when a sodium pump failure caused shutdown of the loop. At the time of shutdown, the cluster had endured 240 days (5760 hr) in flowing sodium at a flow velocity of ~ 25 ft/sec and a temperature of 1060 °F.

Measurements conducted on individual rods and wires disclosed no obvious decrease in diameter. Numbers grit blasted on the rods were still visible. Because these marks are normally 0.0001 in. deep, diametral wear on the fuel rod surfaces may be considered less than 0.0002 in.

The seven-rod cluster will be reinserted in the hexagonal flow basket, and the basket will be placed back in the loop after the new pump is installed. This pump will provide a higher velocity for sodium flow in the 27 - 30 ft/sec range.

8.10 MECHANICAL MODELING FOR FUEL PIN BEHAVIOR

A program⁽¹⁾ has been written to evaluate the effect of the parameters in the mechanical models for the fuel and cladding materials on the composite behavior of the FTR fuel pins. Parameters for this model have been evaluated with in-reactor creep data.⁽²⁾ The form of the radiation dependent creep is

$$\dot{\epsilon}_F = \left(\frac{F}{2 \times 10^{13}} \right) 10^{CF} \left(\frac{\sigma}{1000} \right)$$

where

$\dot{\epsilon}_F$ = additional secondary creep rate due to radiation

$\dot{F}$ = fission rate - (f/cm³)/sec

σ = applied stress

CF = coefficient.

The additional creep due to irradiation has been included in the computer program⁽¹⁾ and a set of calculations performed using CF as a parameter. The results indicate that predicted

(1) K. R. Merckx. A Program for Calculating the Steady State Deformation Rates for Cylindrical Fuel Elements. BNWL-663, January 1968.

(2) J. D. German. Analysis of In-Pile Creep Measurements of Bulk Oxide Fuels. WAPD-TM-687, May 1967.

cladding stresses, as well as the maximum stresses within the fuel material, are highly dependent upon this parameter. The creep resistance is so negligible in the high temperature ($T > 2500$ °F) region that this portion of the fuel material contributes very little to the buildup of stresses between the cladding and the fuel material. As the resistance to creep is decreased (CF increases), the predicted circumferential strain rates also decrease. Table 8.I contains values for the maximum equivalent stresses and circumferential strain rates in the cladding for a wide range of the value of CF.

TABLE 8.I. Dependence of Mechanical Variables on Creep Parameter CF

Cladding Temperature	820 to 896 °F
Fuel Temperature	1640 to 4040 °F
Burnup Rate	0.00137×10^{20} (f/cm ³)/hr

CF	Circumferential Strain Rate (hr) ⁻¹	Maximum Cladding Stress, psi	Maximum Fuel Stress, psi
-5	0.397×10^{-17}	1,040	330
-6	0.189×10^{-13}	3,320	3,240
-7	0.707×10^{-10}	9,632	25,600
-8	0.350×10^{-7}	21,500	82,150
-10	0.145×10^{-6}	25,800	104,600
-12	0.147×10^{-6}	25,870	104,800

The initial void density of the fuel pellet is one of the design variables requiring specification. As irradiation proceeds, these fabricated voids are altered in size and fuel swelling begins. Models for swelling or for thermal conductivity include as parameters the void distribution and size. Thus, these parameters should be evaluated for the reference fuel being used in the FTR. A program has been written for estimating the linear properties of the void distribution. This program can take into account the sampling and observational processes

used in preparing planar sections, and the derived estimation procedure can provide an unbiased estimate and an estimate of the variance of the linear property. The program has been written to estimate specific particle count, mean diameter, specific surface area, void fraction, and accumulated particle count of particles larger than a prescribed size.

SECTION 9.0

PHYSICS9.1 PHASE B (ASSEMBLY 5) OF CRITICAL EXPERIMENTS PROGRAM

Phase B of the FFTF Experimental Program is to provide support to the preparation of the PSAR for the FTR (and NPTR) and to the preliminary design of both reactors. The experiments are essentially idealized or clean in that the core compositions and physical arrays are simple and represent the reference/FTR characteristics including split-core, axial variation of core composition, and gross test loop heterogeneities in the driver zones.

9.1.1 Description of Assembly 51

ZPR-III Assembly 51 (Assembly I of Phase B) became critical with a core fissile inventory of 215 kg. Figure 9.1 shows an area profile of Assembly I.

The core height is ~86 cm and the axial and radial reflectors are 30 cm thick. Table 9.I lists the compositions of the driver zone and the reflectors. The composition is very close to the driver zone of the FTR, exclusive of the control and test loops. The safety drawers were spiked with fuel from neighboring drawers, i.e., the average composition of the safety drawers and their depleted neighbors approximates the core average composition. These drawers were spiked for a combined worth in excess of 1.5% $\Delta k/k$ as required for safe operation of the ZPR-III.

9.1.2 Rod Worth Tests

Measurements were made of the absolute and relative worths of natural boron carbide and tantalum rods. Samples of each material, having 1/2 in. by 2 in. cross sections and 36 in. lengths, were centered on the axis of the 34 in. core.

Driver Zone, 307 μ

Critical Mass, 215 kg Fissile Material

1-4 Spiked Safety Drawers

5 Control Drawer

☒ "Lean" Drawers

FIGURE 9.1. Phase B Assembly I, ZPR-III Assembly 51

TABLE 9.I. *Composition of FFTF Critical
Assembly I (ZPR-III Assembly 51)*

<u>Material</u>	<u>Core</u>	<u>Atom/l-cm</u>		
		<u>Axial Reflector</u>	<u>Radial Reflector I</u>	<u>Radial Reflector II</u>
$^{239}\text{Pu} + ^{241}$	0.001736			
$^{240}\text{Pu} + ^{242}$	0.0001724			
^{235}U	0.0000154			
^{238}U	0.007007			
C	0.003113			
O	0.01279			
Al	0.0000553			
Fe	0.01560	0.01046	0.007605	0.07485
Cr	0.003657	0.00259	0.001870	0.00115
Ni	0.001600	0.02893	0.05640	0.00049
Mn	0.0001527	0.0002	0.00023	0.00056
Si	0.0001792	0.0001	0.00009	0.00006
Mo	0.0003276			
Na	0.009256	0.01037	0.00416	

The reactor was approximately 740 inhours subcritical for the reference measurement. With the tantalum and natural B_4C samples inserted, the reactor was subcritical by 1313 inhours and 1646 inhours, respectively. The worths, therefore, were 573 inhours and 906 inhours, respectively, to produce a ratio of B_4C to tantalum worth of 1.58. The masses of tantalum and B_4C were 9432 g and 1262.2 g, respectively.

These experimental results show that central tantalum rods of volumes equal to those of B_4C rods possess about $2/3$ the reactivity worth of B_4C rods. One might expect this ratio to increase for peripheral rods, as illustrated in BNWL-428.⁽¹⁾

9.1.3 Summary of Programmed Tests

Table 9.II gives a summary of the Phase B Critical Experiments Program to this date.

TABLE 9.II. *Summary of Phase B Critical Experiments Program*

<u>Experiment</u>	<u>Date Completed</u>	<u>Results</u>
Critical Mass	Dec 1, 1967	215 kg fissile
Fuel Reactivity Worth at Core Edge	Dec 21, 1967	To be reported
Control Rod Calibration	Dec 27, 1967	Graph - see ANL FFTF Informal Monthly Report, Jan. 1968
Neutron Spectra by Proton Recoil Proportional Counting	Jan 30, 1968	IBID Measured at 3 in-core and one reflector position
Reactivity Worth of Fuel and Sodium in Central Column	Jan 31, 1968	To be reported
Axial Reactivity Worth of Core Material	Feb 2, 1968	To be reported
Fuel Compaction Experiment	Feb 6, 1968	To be reported

1. *R. A. Harris and L. L. Maas. Unpublished Data. FTR Control Rod Calculations. Pacific Northwest Laboratory, Richland, Washington. September 1967. (Preliminary Report)*

TABLE 9.II. (contd)

<u>Experiment</u>	<u>Date Completed</u>	<u>Results</u>
Large Sample Sodium Void	Feb 8, 1968	To be reported
Central Control Rod Worth	Feb 12, 1968	25 vol% Ta, 9.4 kg; $\Delta\rho = 573$ ih 25 vol% natural B_4C , 1.26 kg; $\Delta\rho = 905$ ih
Distributed Reactivity Worth of Sodium	Feb 16, 1968	To be reported
Doppler Measurements	In progress	
Small Sample Reactivity Traverses	To be completed	
Reaction Rate Traverses	To be completed	

9.1.4 Spectrum Measurements

The spectral measurements of the ZPR-III Assembly 51 were completed and the results received. Data were taken at the core center, midway between the core center and the core-reflector interface, at the core-reflector interface, and at the center of the inner radial reflector.

The neutron spectra, derived from the pulse height distributions of a proton recoil proportional counter, show a general softening of the spectrum radially out from the core center into the reflector.

A prominent feature of the spectra are the effects of the nickel and oxygen scattering resonances. These effects show up as neutron density depletions in the energy region of the resonances. The oxygen resonance shows up in the core spectrum because of the high oxygen concentration in the core. Likewise, the nickel resonance shows up in the reflector spectrum.

9.1.5 Fuel Inventory and Worth Distributions

Calculations of the cylindricized critical mass of ZPR-III Assembly 51 were made using the two-dimensional codes 2DB and 2DF described in the following tabulation:

<u>Code</u>	<u>Geometrical Model</u>	<u>Energy Groups</u>	<u>Sn</u>	<u>k</u>	<u>Critical Mass, kg of fissile</u>
2DF (Transport)	RZ	8	4	1.000	211
2DB (Diffusion)	RZ	8		1.000	215

The actual critical mass was 215 kg of fissile material. Since the driver profile of ZPR-III Assembly 51 is not a perfect circle, the critical mass of the cylindricized reactor is expected to be slightly less than 215 kg.

9.2 COMPARISON OF IRON CROSS SECTION SETS

Substantial differences were noted in the multigroup scattering cross sections for iron in a portion of the resonance region. In the continuing program of cross section evaluation, the scattering cross section was compared with the following seven different sources:

- ENDF/B
- Battelle Master Library⁽¹⁾
- MAC Library⁽²⁾
- Russian Set⁽³⁾

1. K. B. Stewart. BNW Master Library, BNWL-CC-325. Pacific Northwest Laboratory, Richland, Washington.
2. J. Greenborg. Two Cross Section Libraries for Use with MAC Shielding Code, HW-73381. General Electric Company, Richland, Washington, Sup 1.
3. I. I. Bondarenko. Group Constants for Nuclear Reactor Calculations, Consultants Bureau. 1964.

- HRG⁽¹⁾ using ORNL cross sections
- ANISN 27 group library
- KFK⁽²⁾

The group average values in the MAC structure for the first six of these listed sources are compared to the point-wise KFK values in Figure 9.2. As can be seen, considerable disagreement exists at these energies. The effect of these discrepancies on specific shielding problems is being assessed.

Calculations were made to compare ANISN and MAC using consistent cross section data. The results (Figures 9.3 and 9.4) show the two methods to be comparable for predicting the distribution of damage (>0.067 MeV) flux in iron, nickel, and sodium shields when comparable cross section data are employed. These results indicate the simpler ANISN diffusion calculations to be adequate for estimating the attenuation characteristics of these metals, and that transport (or removal approximation) calculations are not required.

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1. J. L. Carter, Jr. Reactor Physics Quarterly Report, July-Sept, BNWL-340. Pacific Northwest Laboratory, Richland, Washington. 1966.
 2. Neutron Cross Sections for Fast Reactor Materials, KFK-120 (EANDC-E035 U).

FIGURE 9.2. Comparison of Iron Scattering Cross Sections from Different Sets

FIGURE 9.3. Basis for Gas Volume Calculations

FIGURE 9.4. Nickel and Iron Comparison Above 0.07 MeV

9.3 FISSION PRODUCT INVENTORY IN FTR

A series of calculations were made to determine the FTR fission product inventory as a function of operating variables. Calculations were made using the computer program RIBD with a special nuclear data library. This library contains fission product yield factors for plutonium⁽¹⁾ and neutron transmutation cross sections for each isotope appropriately weighted for the FTR spectrum.⁽²⁾ As expected, saturation of most of the radioactive isotopes tends to be consistent with a consideration of the decay chain half-lives. However, the effect of neutron transmutations can be noted in several isotopes. An outstanding example is that of ^{134}Cs , produced entirely in FTR fuel by a subsequent neutron transmutation of a stable fission product.

Results of the calculations are shown in Figure 9.5 for a selected group of longer-lived isotopes. Curie values are based on the inventory contained in a quantity of FTR fuel that had generated 100 MW for a continuous operating period of 300 days at a flux level of 4×10^{15} n/cm²-sec. This level is equivalent to an exposure of about 40 MWd/kg.

Although the exposure and flux level are important parameters in establishing the inventory of several specific isotopes in irradiated fuel, the gross inventory (curies) and the total decay energy are found to be relatively insensitive to these parameters in the range of interest for FTR.

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1. C. A. Anderson, Jr. Fission Product Yields from Fast Neutron Fission of Pu-239, LASL-3383, December 1965.
 2. R. E. Schenter. Unpublished Data. "Fission Product Capture Cross Sections" Fast Flux Test Facility Monthly Informal Technical Progress Report. Pacific Northwest Laboratory, Richland, Washington, November 1967. (Preliminary Report)

9.4 DETERMINATION OF GROUP AVERAGE FISSION PRODUCT CAPTURE CROSS SECTIONS

Group average capture cross sections of 110 primary fission product isotopes were calculated for the top 17 groups ($E = 10.5 \text{ MeV} - 100 \text{ eV}$) in the Russian (Bondarenko) format. The results (Table 9.III) were used to construct Program FPPSE group average cross sections for a typical or "pseudo" fission product isotope representing all the fission product isotopes weighted according to cumulative yield curves associated with the fast fission of ^{239}Pu , ^{235}U , and ^{233}U .

TABLE 9.III. Group Average Capture Cross Sections for the "Pseudo" Fission Product from ^{239}Pu Fast Fission

<u>Group</u>	<u>$E_{\min}$</u>	<u>(σ_c)-FPPSE</u>	<u>(σ_c)-Bondarenko</u>
1	6.5 MeV	0.007	0.005
2	4.0	0.009	0.010
3	2.5	0.013	0.020
4	1.4	0.022	0.035
5	0.8	0.039	0.055
6	0.4	0.059	0.075
7	0.2	0.094	0.100
8	0.1	0.145	0.125
9	46.5 keV	0.229	0.200
10	21.5	0.375	0.275
11	10.0	0.569	0.450
12	4.65	0.889	0.750
13	2.15	1.45	1.10
14	1.0	2.42	1.60
15	0.465	4.07	2.75
16	0.215	6.67	6.00
17	0.100	11.43	6.00

9.5 NOBLE GAS RELEASE TO COVER GAS

The expected entry rate of fission products into the FTR cover gas was estimated to provide a basis for including this effect in the conceptual design of the system. It was assumed that only the noble gases (krypton and xenon) could penetrate the sodium and enter the cover gas. The production rate of these gases in the reactor depends on the effective yield factors for the particular fuel and neutron spectrum. The yield factors developed by Anderson for ^{239}Pu were used.⁽¹⁾

The rate at which radioactive gas nuclei reach the cover gas is:

$$N_i^* = f P_i \exp(-\lambda_i H) [1.0 - \exp(-\lambda_i T)]$$

where f is the fraction of generated nuclei escaping with an effective holdup time or path length, H , where P_i is the effective cumulative production rate for that isotope, T is the irradiation time, and λ_i is the decay constant.

For a simple, single pass cover gas flowing continuously at a rate of $F \text{ cm}^3/\text{sec}$, the specific activity of the cover gas becomes:

$$G_i = \frac{N_i^* \lambda_i}{F 3.7 \cdot 10^4} (\mu\text{Ci/ml})$$

If the cover gas is passed through a perfect filter that traps all of the noble gas, the number of atoms trapped on the filter is given by:

$$B_i^* = N_i^* / \lambda_i$$

where the transport time in the cover gas is assumed to be negligible compared to the holdup time.

1. C. A. Anderson, Jr. Fission Product Yields from Fast ($\sim 1 \text{ MeV}$) Neutron Fission of Pu-239, LA-3383, December 30, 1965.

Sample calculations were made for a reactor power of 400 MW to illustrate the use of the relations for the case of vented fuel and for an assumed cladding failure. Results of the calculations are presented in Tables 9.IV and 9.V, together with the input assumptions. Heat generation in a perfect filter was calculated from decay energies tabulated for RIBD.⁽¹⁾ It can be seen that the release fraction, the holdup time, and the flow rate will play important roles in establishing the radioactive level of the cover gas.⁽²⁾

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1. W. L. Bunch and D. R. Marr. Fast Reactor Library for use with RIBD, BNWL-CC-868. Pacific Northwest Laboratory, Richland, Washington, October 17, 1966.
 2. W. L. Bunch. Noble Gas Release to FTR Cover Gas, BNWL-744. Pacific Northwest Laboratory, Richland, Washington, March 1968.

TABLE 9.IV. Noble Gas Release to Cover Gas Vented Fuel Data

Assumptions: $t = 115$ days, $H = 5$ days, $f = 1.0$, $F = 10^4$ cm³/sec

	10^{17} atoms/sec to Cover Gas		10^{13} atoms/cm ³ in Cover Gas		Cover Gas Activity, $\mu\text{Ci/cm}^3$	Heat Generation in Filter, W		
	$\underline{N_i^*}$ $\underline{N_i}$	or $\underline{N_i}$	$\underline{Q_i^*}$ $\underline{Q_i}$	or $\underline{Q_i}$	$\underline{G_i}$	$\underline{H_\beta}$	$\underline{H_\gamma}$	$\underline{H_{Total}}$
<u>Krypton</u>								
83		0.44		0.44		12.00	0.19	12.19
84		0.60		0.60				
85	0.0030		0.0030		1.65^{-3}			
86		1.04		1.04				
<u>Xenon</u>								
131*	0.0225		0.0225		4.06	0	58.9	58.9
131		3.94		3.94				
132		5.28		5.28				
133*	0.0265		0.0265		2.49^1	0	99.1	99.1
133	3.44		3.44		1.40^3	5510	4510	10,020
134		7.48		7.48				
135	0.0009		0.0009		5.090	4.47	3.76	8.23
136		8.00		8.00				

9.16

BNWL-660

TABLE 9.V. Noble Gas Release to Cover Gas
Cladding Failure Data

Assumptions: $t = 115$ days, $H = 10$ sec, $f = 10^{-6}$, $F = 10^4$ cm³/sec

10 ¹¹ atoms/sec to Cover Gas		Cover Gas Activity, μCi/ml		Maximum Heat Generation in Filter, W	
<u>N_i</u> [*]	of <u>N_i</u>	<u>G_i</u>	<u>H_B</u>	<u>H_Y</u>	<u>H_T</u>
<u>Krypton</u>					
83*	0.44	0.0123	0	0.00029	0.00029
83			0	0	0
84	0.44		0	0	0
84	0.60		0	0	0
85*	0.80	0.00945	0.0037	0.0023	0.0060
85	0.003	(1.65 ⁻⁹)	0.000012	~0	0.000012
86			0	0	0
86	1.04		0	0	0
87	1.33	0.0503	0.0286	0.0232	0.0518
88	1.61	0.0299	0.0098	.0496	0.0594
89	1.78	1.74	0.0397	.0646	0.1043
90	1.38	7.82	0.0302	.0493	0.0795
91	0.62	11.6	0.0155	0.0198	0.0353
92	0.063	3.93	0.00275	0	0.00275
93	0.0067	0.627	0.00041	0	0.00041
94	0.00035	0.0468	0.00002	0	0.00002
95	0.00000	~0	~0	0	~0
<u>Xenon</u>					
131*	0.03	(50 42 ⁻⁶)	0	0.00008	0.00008
131			0	0	0
132	3.94		0	0	0
132	5.28		0	0	0
133*	0.12	(1.13 ⁻⁴)	0	.00045	0.00045
133	6.62	0.027	0.0106	.0087	0.0193
134			0	0	0
134	7.48		0	0	0
135*	2.14	0.0417	0	.0182	0.0182
135	7.94	0.0449	0.0394	0.0332	0.0726
136			0	0	0
136	8.00		0	0	0
137	7.31	5.85	0.173	0.117	0.290
138	5.71	1.27	0.0879	0.0382	0.1261
139	2.92	13.4	0.0934	0.0187	0.1121
140	1.08	12.6	0.0189	0	0.0189
141	0.0081	0.0891	0.00031	0	0.00031
142	0.0013	0.0162	<0.00004	0	<0.00004

SECTION 10.0

REACTOR SAFEGUARDS10.1 DESIGN BASIS ACCIDENT STUDY

The DBA Task Force is continuing work for definition of the DBA energy release for the reference conceptual design. Transmittal of a document describing a series of parametric accident studies was scheduled for June, 1968. A first draft of the report was scheduled for April, 1968. The results of this study will serve as a basis for the preliminary containment design. However, refinements in the analytical methods will continue throughout the preliminary design for incorporation in the Preliminary Safety Analysis Report.

A mechanistic approach is being taken by the DBA Task Force. This approach is divided into the following four major phases of analysis:

- Initiating accidents
- Core meltdown
- Core disassembly
- Evaluation of the energy partitioning.

The initiating accidents are dependent upon the Fault Tree Analysis in the mechanistic approach. Yet, throughout the studies, conservative assumptions are incorporated in areas of unknown or vague technology. One typical example is the phenomenon of sodium vaporization. Several accident paths, assumed both for shutdown and operating conditions, are under consideration. To date, a control rod ejection accident appears to establish the most severe ramp reactivity insertion rate. With an arbitrarily applied factor of safety on the ramp rate, an initial ramp rate of 100\$/sec is employed in the present analyses of the proposed operational DBA.

10.2 PRELIMINARY SAFETY STUDY OF THE LOSS-OF-COOLANT ACCIDENT

The loss-of-coolant accident involves a major leak of reactor primary coolant through a break in the coolant container (the reactor vessel), the piping, and other components of the primary coolant system. There is historical evidence that small leaks must be expected even in well designed and fabricated systems. However, leaks equivalent to those from double-end breaks of large pipes are quite unknown. Nevertheless, it is difficult to classify any smaller leak in the maximum possible size, and water reactor safety analyses commonly consider a double-ended break.

A preliminary study pertaining to the loss-of-coolant accident was conducted. The study concentrated on the piping between the primary coolant pumps and the reactor because a break in this region, owing to pumping action, would result in the fastest leak rate in the primary system. The results can be taken as indicative of a break occurring between the check valve and the reactor, or of a break between the primary pump and the reactor if the check valve in the broken pipe should happen to stick open. The analysis neglected the flow limiting effects of cavitation in the primary pumps because these hydraulic characteristics have not been established sufficiently to warrant inclusion at this time.

The study covered the early effects of coolant loss and core over-heating as a function of the pipe size involved in the break. For instance, the single 18 in. diam pipe between the primary pump and the reactor in each primary coolant loop⁽¹⁾ might be replaced by two 12 in. diam pipes. With judicious

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1. V. A. DeLiso. Unpublished Data. Conceptual System Design Description for Reactor Heat Transport System, System No. 51. Pacific Northwest Laboratory, Richland, Washington, January 7, 1968. (Preliminary Report)

design, the breakage of one of these smaller pipes would not immediately imply the sequential breakage of other pipes. Standard pipe sizes in the range from 18 in. down to 6 in. diam were included in this study, as noted in Table 10.I. Coolant loss from each break was evaluated from the standpoint of (1) the pump in the affected loop continuing to force coolant out the broken pipe, and (2) the two pumps in the other loops continuing to supply coolant to the reactor inlet plenum and, thence, either up through the core for recirculation or backward and out through the broken pipe.

TABLE 10.I. Core Conditions for Various Sizes of Primary Pipe Breaks

Number of Pipes in Each Loop	Size of Pipe Break, in.	Total Flow Through Two Pumps, gal/min	Flow Up Through Reactor, gal/min	Percent of Normal Reactor Flow	Average Core Outlet Temp ^(a) °F	Hot Channel Outlet Temp ^(a) °F
1	18	41,300	10,000	32	1850 ^(b)	-
2	12	39,300	16,700	49	1515	-
3	10	37,800	19,300	56	1432	1645 ^(b)
5	8	35,600	22,100	65	1364	1549
8	6	33,300	24,700	72	1315	1440
-	(400 MW, 2 Pumps)	29,100	29,100	85	1253	1394
-	(400 MW, 3 Pumps)	34,200	34,200	100	1200	1320

a. Inlet temperature of 900 °F

b. Calculated as if boiling does not occur at 1618 °F

The reactor was assumed to be operating normally at 400 MW and at core inlet and outlet temperatures of 900 and 1200 °F, respectively, before the onset of the accident. The time necessary to effect pipe breakage was taken as instantaneous, while the time necessary to overcome fluid inertia and change flow patterns in the primary system was also neglected. The normal conditions before the break and the pseudo-steady conditions after the break are illustrated in the Head-Flow diagram of pump and system curves for the

break sizes included in this study (Figure 10.1). For example, combined flow of the three pumps before the break would be 34,200 gal/min at the 155 psi head necessary to overcome losses through the core and primary system (point A of Figure 10.1). After the break, the affected loop pump would force 20,700 gal/min out of the break (point B), while the other two pumps would supply 41,300 gal/min to the reactor inlet plenum (point C). Only 10,800 gal/min of this supply would flow through the core (point D), while the rest (30,500 gal/min) would flow out the break. Thus, as quickly as the break occurs and the flow reversal in the broken pipe is completed, the flow through the core would reduce to about 32% of normal and soon thereafter the core outlet temperature would exceed the boiling point of sodium (1618 °F). Similar results for the other sizes of pipe breaks are summarized in Table 10.I.

The fast leakage from such pipe breaks terminates when the reservoir of sodium in the reactor upper plenum is drained below the reactor outlet pipes or when the pumps are shut down. The size of the sodium reservoir required in an emergency cooling core system to assure continued circulation was assessed with the assumption of some 2 sec needed for the instrumentation to detect the leak, to scram the reactor, and to de-energize the primary pumps. The time required for primary pumps to coast to a stop was assumed to be 20 sec. In the initial 2 sec, some 1710 gal of sodium would be lost through the 18 in. diam pipe break. The pumps coast down once they are de-energized, and the accompanying flow coast-down is similar to the curves for loss-of-power accidents. By scaling and integrating one flow coastdown curve through 20 sec of pump shutdown, some 5370 gal of additional sodium would be lost through the 18 in. diam pipe break, with total leakage amounting to about 7080 gal. Similar results for the other sizes of pipe breaks are listed in Table 10.II.

FIGURE 10.1. Head-Flow Characteristics of Primary System, Pumps, and Various Breaks

TABLE 10.II. *Coolant Leakage from Various Sizes of Primary Pipe Breaks*

Size of Pipe Break	Leakage from Reactor Inlet Plenum,(a) gal/min	Leakage from Pump of Broken Loop,(a) gal/min	Total Leakage Rate,(a) gal/min	Leakage in 2 sec(b) Before Pump Shutdown, gal	Leakage in 20 sec Pump Shutdowns, gal	Total Coolant Loss from Reactor Reservoir, gal
18	30,500	20,700	51,200	1,710	5,370	7,080
12	22,600	20,100	42,700	1,420	4,480	5,900
10	18,500	19,100	37,600	1,250	3,950	5,200
8	13,500	17,800	31,300	1,040	3,280	4,320
6	8,600	16,700	23,300	780	2,450	3,230

a. *Pseudo steady flow rates following flow reversal after pipe break.*

b. *Instrumentation lag to sense leak, scram reactor, and de-energize pump.*

The obvious conclusions following from the results listed in Tables 10.I and 10.II are that the potential for boiling sodium in the core and the quantity of sodium lost are reduced for the smaller pipe breaks. If widespread sodium boiling in the core is regarded as an intolerable condition, then the large 12 and 18 in. diam pipe breaks must be guarded with additional design safety features such as reverse flow restrictors at the reactor inlet plenum, double wall piping, etc. The smaller pipe breaks, 10 in. diam and smaller, do not immediately cause sodium boiling and, therefore, reservoir requirements can be more compatible with shorter fuel assemblies.

10.3 EFFECTS OF THE COLLAPSE DRIVING FORCE IN A MELTDOWN ACCIDENT

One of the uncertainties arising when constructing a core meltdown model is the driving force acting upon the

molten fuel. Studies reported in BNWL-611⁽¹⁾ assumed free fall, i.e., 1 g (981 cm/sec^2). Viscous effects included in the referenced model, however, are relatively minor. The major parameter is the accelerating force.

Figure 10.2 shows the variation of the ramp rate due to collapse (at the time of disassembly) as a function of the collapse driving force. The significant feature of the curve is that the ramp rate and, hence, the energy release is a nonlinear function. This behavior arises because the larger gravity values bring the core to a disassembly condition in considerably less time than do the lower gravity values, thereby somewhat offsetting the t^2 dependence.

Hence, even if driving potentials greater than 1 g could be postulated, the consequences are not altered in a one to one correspondence.

10.4 APDA SAFEGUARDS STUDIES

APDA personnel recently presented tentative FY-1968 results of a set of accident calculations for the FTR. The main variables considered in this study included the disassembly ramp rate, the Doppler coefficient, the fuel specific heat, core geometry (zoning and L/D ratio), and fuel type (oxide and metal). For all cases, the initial condition was assumed to be a totally molten core (without sodium). For each run (using the two-dimensional disassembly code MARS), both the total energy generation and the available work to 1 atm was obtained. A unique feature of the presentation was that of displaying the available work on a work per mole basis, i.e., specific available work. This approach allowed intercomparison with the calculation of others.

1. A. E. Waltar and R. A. Harris. Preliminary Safety Studies for the Nuclear Proof Test Facility, BNWL-611. Pacific Northwest Laboratory, Richland, Washington. November 1967.

FIGURE 10.2. Variation of the Ramp Induced by Collapse as a Function of the Collapse Driving Force

The study led APDA to rate the variables considered in the following order of importance:

- First Order: Fuel mass and reactivity insertion rate.
- Second Order: Doppler effect (for initially hot cores), for specific heat, initial zoning.
- Third Order: Additional zoning (i.e., test zone in an already zoned core), and L/D ratio.
- Unresolved: Core initial temperature, equation of state (volume expansion coefficient), and one-dimensional versus two-dimensional disassembly calculations (clean core).

The last point still concerns BNW because, even though APDA has calculated both one-dimensional and two-dimensional disassemblies, a clean comparison of the two calculational methods has not been made.

Of particular interest is the apparently narrowing difference between the energy release calculation of the metal versus an oxide core. This difference is somewhat contrary to earlier APDA results.

APDA issued a formal report, APDA-208, "Comparative Hypothetical Accident Analysis for Four Interim Core Designs."

10.5 CLOSED LOOP RELIABILITY ANALYSIS

Preliminary calculations indicated the availability for one closed loop to be 0.99408, and the reactor unavailability per 1000 hr as 5.92 hr per closed loop. These data relate only to the out-of-reactor portions of the loops.

These calculations were based on a reliability equation derived for FFTF conditions as follows:

$$R(t) = P(t)V^2(t)\{2 - P(t)\} + 2P(t)V(t)C(t)$$

where

$P(t)$ = pump reliability

$V(t)$ = valve reliability

$C(t)$ = valve closed state probability

If $\lambda_p = 25/10^6$ hr (the pump failure rate); $\lambda_{vo} = 2/10^6$ hr (the valve open failure rate); and $\lambda_{vc} = 2/10^6$ (the valve closed failure rate); then for $t = 1000$ hr, and using the exponential distribution,

$$P(1000 \text{ hr}) = e^{-\lambda_p \times 1000} = 0.9752$$

$$V(1000 \text{ hr}) = e^{-(\lambda_{vo} + \lambda_{vc})1000} = 0.99601$$

$$C(1000 \text{ hr}) = 1 - e^{-\lambda_{vo} \times 1000} = 0.002$$

Basic data employed are summarized in Table 10.III. This table shows values of failure rate (λ), mean time to restore (MTR), and probability of failure⁽¹⁾ for each component used in the overall calculation of system availability. These values are estimates based on values obtained from three documents^(1,2,3) and are considered neither optimistic nor pessimistic.

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1. W. R. Lahs, Jr. "FFTF System Effectiveness Evaluation FY 1966 Summary Report," NAA-SR-MEMO-12691.
 2. "FFTF Backup Design Study," Contract No. BDR-341 between PNL and the General Electric Co., May 22, 1967.
 3. "FFTF Availability Model," Contract No. 1 WR-54-RK-55034-B, April 4, 1966.

TABLE 10.III. *Values for Each Component Used in Calculations of System Availability*

<u>Component</u>	<u>Failures Per 10⁶ hr, λ</u>	<u>MTR (hr)</u>	<u>Probability of Failure Per 1000 hr, q</u>
Instrumentation	2	25	2×10^{-3}
Two Test Chambers @ 1/2 Each	1	500	1×10^{-3}
Intermediate Heat Exchanger	10	2000	10×10^{-3}
Secondary Loop	15	100	15×10^{-3}
Heater	1	500	1×10^{-3}
Piping	5	700	5×10^{-3}
Primary Pumps and Associated Check Valves	4.17	400	4.17×10^{-3}
Check Valves	2	200	2×10^{-3}
Strainer	1	200	1×10^{-3}

10.6 NUCLEAR CRITICALITY SAFETY

Calculations to establish minimum critical parameters for FFTF driver fuel are continuing. Using the Monte Carlo GEM III code, the minimum critical number of 30 wt% PuO₂-UO₂, unmoderated subassemblies were estimated to be 10.5. This value is about 20% lower than the value predicted from extrapolations of data obtained from diffusion theory calculations.

Also, by means of GEM III, the minimum critical number of unmoderated subassemblies in solid cylindrical geometry (simulating subassemblies melted together) were estimated

to be 8.0. The previously estimated value, based on transport theory, was 10.7 subassemblies. However, in the latter case, the fuel was of lower density. As this study continues, closer agreement will be expected between calculational methods.

Calculations to establish safe storage arrangements for driver fuel are in progress. Safe spacing must be established for storage in air, water, and concrete.

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