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FAST TRACK
TEST FACILITY
PROJECT REPORT

FFTF DEVELOPMENT
MONTHLY PROGRESS REPORT

F. W. Woodfield

January 1968

AEC RESEARCH &
DEVELOPMENT REPORT

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PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

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FAST FLUX TEST FACILITY
DEVELOPMENT MONTHLY PROGRESS REPORT
January, 1968

F. W. Woodfield, Deputy Project Manager
for Development
February 10, 1968

Approved By:

J. W. Helm

PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON

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PREFACE

This report, issued monthly, covers technical details and status of development programs performed at PNL in support of FFTF. Other reports in this series are:

September, 1967	BNWL-582
October, 1967	BNWL-615
November, 1967	BNWL-639
December, 1967	BNWL-666

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FFTF DEVELOPMENT MONTHLY PROGRESS REPORT, JANUARY 1968SUMMARY

Measurements of the reaction of methane with sodium have been extended to 1400°F, where 100 ppm in helium completely reacted in 3-1/2 hours. It was also determined that the volume of sodium in the system has little effect on the reaction rate.

An important milestone was achieved in the development of electrical heaters for simulation of fuel pins for use in future heat transfer studies on FFTF fuel assembly models. In a demonstration test, a tantalum coaxial heater was operated continuously for six hours at a heat flux of 1,000,000 Btu/hr-ft² in water at 320°F outlet temperature. The heater then operated at 1,300,000 Btu/hr-ft² for an additional 4-1/2 hours before the test was terminated due to electrode failure.

The fabrication of the test facility required to demonstrate and establish the feasibility of gas cooling of FFTF fuel elements during inspection and disassembly was completed.

A preliminary study was made to determine the feasibility of utilizing facilities such as the Hydraulic Core Mockup, the Seven-Duct Mockup, the Reactor Cover Mockup, and the Sodium Facilities Building for Fuel Handling Machine and Radioactive Maintenance equipment interface testing. With modification, each or all of the above facilities could be applied in the program. Final PINL recommendations will be made in February.

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

A preliminary assessment, from computer simulations, of radiation damage processes yields the following ranking in relative damaging power for void production in fuel cladding: EBR-II > DFR > FTR. This ranking is based on the number of primary knock-on atoms (PKA) with energy greater than 3 KeV produced per interaction with neutrons. All neutrons with energy greater than 25 eV were considered. Results also showed no significant difference in PKA production between samples the size of tensile specimens and samples the thickness of FTR fuel clad.

COOLANT SYSTEMS DEVELOPMENT

B. M. Johnson

CORROSION AND CHEMISTRY

D. W. Shannon

Selection and Testing of Radioactive Sodium Analysis Methods

(F. A. Scott)

The objectives of this program are to evaluate sodium analysis requirements for radioactive sodium and to establish standard sodium analytical procedures at PNL.

High Level Analytical Facility Design - (F. A. Scott, J. F. Jarosch)

A preliminary study of the design of the FFTF chemical analyses laboratory has been completed. A program plan for the required remote mockup of the analytical hot cell and testing of remote analysis methods is being developed based on the needs revealed by the study.

Total Carbon Determination - (L. F. Lust)

Efforts have continued to shorten the time required for the carbon determination and thus permit the handling of a greater sample load. Investigation has centered on developing an adequate procedure for the quantitative trapping of carbon dioxide from a pure oxygen combustion gas stream. The approach involves absorption of CO_2 on a molecular sieve at room temperature followed by a "dump" of the trap contents into the chromatographic carrier gas stream by a flash heating of the trap.

Selection and Testing of In-Line Sodium Analysis Instruments

(T. F. Demmitt)

The selection, proof testing, procurement and pre-startup testing of in-line sodium analysis instruments are being coordinated.

A detailed Program Plan was written to identify the test requirements and schedules for FFTF in-line instrumentation following discussions with UNC, APDA, LASL, and AI (LMEC) personnel on development and demonstration testing of in-line instruments.

Sodium Cover Gas Interaction Studies

(H. P. Maffei)

Investigations on sodium cover gas interactions were continued during the past month 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.

It was previously reported that 100 ppm of methane in helium reacted very little at 800°F during a period of 21 hours, whereas at 1200°F, the reaction proceeded to virtual completion over a period of 24 hours. During the past month the reaction was studied at 1400°F with 100 ml of sodium as in previous runs, and at 1200°F with 250 ml and 500 ml of sodium.

The reaction of methane and sodium at 1400°F was considerably faster than the reaction measured in previous runs. Virtually all of the methane was consumed over a period of 3-1/2 hours. The methane decreased from an original concentration of 100 ppm to less than 2 ppm as measured by a gas chromatograph.

Increasing the sodium volume from 100 ml to 250 ml and 500 ml did not appear to increase the reaction rate of methane and sodium at 1200°F. The methane decreased from an original concentration of approximately 100 ppm to a concentration of less than 3 ppm over a period of 22 hours for both the 250 ml and 500 ml runs. This compares to a period of 24 hours for the 100 ml volume. The fact that the volume of sodium is unimportant emphasizes the reaction rate is probably affected by the surface area of sodium exposed, rather than controlled by volume. Plans are being made to measure the reaction rate constants using argon as an inert gas.

Evaluation of FFTF Sodium Cleaning Requirements and Decontamination Feasibility (T. F. Demmitt, G. R. Bloom)

The objectives of this study are to evaluate FFTF sodium removal processes and to develop decontamination techniques.

A detailed Program Plan was written to identify the test requirements and schedules for performing the sodium removal and chemical decontamination studies required in support of FFTF.

Sodium cleaning procedures are being evaluated for FFTF fuel assemblies, components and piping systems. Material specimens for these evaluation tests are now being prepared and conditioned in sodium. Specimens at 1000, 1200 and 1400°F have completed their exposure in sodium for 1000 hours. Specimens of 2000 hours at 1400°F are still in the sodium.

Specifications have been written for a Capsule Testing Stand to permit exposure of metallic specimens to sodium containing fission products and fuel materials at various levels of sodium temperature and purity. The facility will be used to prepare specimens for use in sodium removal, fission product, and (possibly) chemical decontamination experiments.

Evaluation of Radioisotope Transport and Deposition

The objectives of this task are to define the parameters which cause deposition of radioisotopes and predict deposition patterns in FFTF piping.

A review of the literature has revealed that the following topics pertinent to FFTF operation are incompletely covered by the present state-of-the art:

- Lack of knowledge of parameters affecting cold-leg deposition patterns is apparent.
- The selective leaching studies on stainless steel deal with Fe, Cr, Ni, and Mo, but the main active species produced in a fast flux are Co and Mn. (Mn is reported to be selectively leached by low-oxygen sodium.)
- The effect of irradiation on susceptibility of stainless steel to sodium corrosion is still unknown. Some PNL work has shown that grain boundaries of irradiated stainless steel responded to etching more strongly than unirradiated stainless.⁽¹⁾

The program plan is being rewritten to evaluate and possibly modify the experimental program in light of these considerations.

Radioisotope Transport Loop Operation - (E. A. Kovacevich)

The radioisotope transport loop was prepared to receive an irradiated Type 304 stainless steel specimen.

The radioisotope counting assembly containing the sodium iodide crystal was completed and installed within the loop chamber. Preparations for background counting of the loop are being initiated.

A radiation survey of the irradiated test specimen indicated 92 R/hr at the surface of the specimen and 160 mR/hr at a distance of 6 inches. A

(1) T. T. Claudson. Unpublished Data. PNL Metallurgy, January 1968. Private Communication.

survey utilizing the 4096 channel analyzer indicated the major activity from the specimen is due to Cr^{51} , with Co^{60} and Fe^{59} contributing to the total radiation. In order to obtain a minimum amount of radiation background when an irradiated specimen is in the loop, the test specimen area of the loop was shielded with approximately four inches of lead.

To date, the radioisotope transport loop has completed over 2000 hours circulating high temperature sodium. Loop operation with the irradiated specimen inserted is anticipated for the next reporting period.

The Baird-Atomic 100 channel analyzer, which will be used for attempts at in situ scanning of loops, has arrived. A successful method of positioning and shielding the irradiated specimens is still being developed.

Thermal Convection Sodium Loops - (W. F. Brehm)

One of the groups of the irradiated stainless steel samples that will be used for deposition studies was dissolved in acid and submitted for radiochemical analysis. Preliminary results indicate the principal constituents are Co^{60} , Cr^{51} , and Fe^{59} ; a detailed qualitative analysis is pending.

The first irradiated specimen will be inserted this month in one of the three thermal convection loops.

Radioisotope Transport Loop Expansion - (T. J. Owens, V. P. Kelly)

Design is continuing on the loop modifications. A modified version of a SRE cold trap is being redesigned to incorporate a larger settling area and a different thermowell arrangement. The hood design is complete.

Sodium Purification Loop - T. J. Owen, V. P. Kelly)

The oxygen probe has been relocated at a point downstream from the plugging valve, to allow for better temperature control. Startup preparations are underway with preheating and outgassing nearing completion.

Closed Loop Contamination Control System

The objectives of this task are to define the operating problems resulting from running fuel tests to failure and to develop methods to control radioactivity in FFTF closed loops. Bench-scale studies, tracer level runs in the radioisotope loop and eventually full-level engineering studies in the Radioactivity Control Test Loop (formerly called Engineering Scale Loop) will be utilized.

Fission Product Control Techniques - (P. L. Koehmstedt)

A review of progress and objectives of all AEC programs for fission product removal from sodium-stainless steel systems, as outlined in the LMFBR Program Plan for Sodium Technology (Section 5), was conducted. The purpose was to ascertain the extent to which FFTF needs in this area will be met by these programs.

Removal of gaseous Kr and Xe fission products from helium is well established by cryogenic cold traps. Removal of radioactive Kr and Xe from argon by cryogenic methods is difficult because argon condenses at normal liquid nitrogen temperatures. However if liquid nitrogen is pressurized to 100 psi, the temperature rises to -280°F , which is above the condensation temperature of argon (-302°F) but below that for Kr (-243°F) and Xe (-162°F). This means a pressurized liquid N_2 trap packed with a molecular sieve could possibly remove radioactive fission gas from argon. It is planned to evaluate this concept more thoroughly since an effective method to remove fission gases from argon could greatly increase the applicability of argon to FFTF cover gas systems.

THERMAL HYDRAULICS

G. M. Hesson

Heat Transfer Characteristics of FFTF Fuel Assemblies

(J. M. Yatabe, V. R. Hill, K. D. Hinkle)

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

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

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

During this report period, an important milestone was achieved when a tantalum coaxial heater (designated rod 650-34) was operated continuously for six hours at a heat flux of 1,000,000 Btu/hr-ft² in water with outlet temperature of 320°F (160°C). Power to the heater was increased, and it was operated at a heat flux of 1,300,000 Btu/hr-ft² (water outlet temperature 320°F) for an additional 4-1/2 hours before the test was terminated because of a break in the molybdenum electrode approximately 1/16 inch from the end of the heater assembly. The probable cause of this break was a combination of mechanical and thermal stresses caused by the method of end-connector cooling. To prevent reoccurrence of this problem, new end connectors were designed and fabricated for future testing.

The next heater tested (rod 650-32) was fabricated in the same production run as rod 650-34. This heater was brought up to a heat flux of 1,500,000 Btu/hr-ft² (water outlet temperature 320°F) and operated under these conditions for 1-1/4 hours before a failure occurred. Detailed post operation inspection and analysis of this heater, including complete X-radiography, will be performed during the next report period.

Detailed design of the electrical lead-in and auxiliary cooling system required for sodium testing of the FFTF heated pins has been completed. Fabrication of the system will begin in the next report period.

Gas Cooling of FFTF Fuel Assemblies During Inspection

(J. M. Yatabe, V. R. Hill, K. D. Hinkle)

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

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 using 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 CFM

Transverse 0-2300 CFM

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, 1/4-in. OD and 3-ft long, capable of producing 40 kW total.

EQUIPMENT DEVELOPMENT

J. R. Carrell

DEVELOPMENT FACILITIES

K. G. Toyoda

Hydraulic Core Mockup

(K. D. Hayden, D. L. Ballard)

The objective of the Hydraulic Core Mockup program is to provide development and testing support for the resolution of FTR core hydraulic and mechanical design problems.

HCM Instrumentation - (G. W. Riedeman)

Work has progressed on developing proper circuitry for converting changes in conductivity cell resistance into useful electrical signals, in a form suitable for recording or other processing. A basic circuit has been designed that can easily detect very small additions of salt to tap water, in a slowly flowing system. This circuit will be prototyped and made ready for tests.

Single Tube Hydraulic Balance Test - (P. M. Jackson)

The purpose of this full-scale single tube model is to evaluate the hydraulic balance feature being 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 this month 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 cause an increase of about 18 pounds in the downward force, upon the duct, at a flow of 350 gal/min.

Tubesheet Seal Configurations Test

The goal of this program is to evaluate alternate tube-to-tubesheet sealing and seating configurations. Piping for the test apparatus has been installed at 321 Building and testing will begin soon. The initial test will investigate a labyrinth-type seal arrangement.

Annular Inlet Feature Model - (L. R. Sweetin)

The purpose of this tenth size hydraulic mockup of the FFTF annular coolant supply arrangement is to assist in evaluating the mixing

characteristics of the inlet streams as they flow within the annulus formed by the main reactor vessel and the removable core vessel. Fabrication is essentially complete, and the installation of this model at the 314 Building is nearing completion.

Outlet Region Model - (L. R. Sweetin)

The purpose of this model is to assist in evaluating the characteristics of the coolant flow as it exits from the shrouds and moves through the complex geometry of the upper reactor internals to the outlet nozzles.

The proposed model will be geometrically similar to the reactor and will be made to a scale of one-third size. A scope drawing and test proposal for this model were completed and issued for comments.

Sodium Facilities

(K. D. Hayden, K. G. Toyoda)

The objective of the sodium development facilities program is to establish the technical basis and requirements of sodium core development facilities responsive to the long- and short-range needs of the FFTF program and to provide the necessary technical support for acquisition of facilities.

A FFTF Sodium Facilities Building to house several new and existing sodium facilities necessary to implement the FFTF development program is needed. The project proposal and design criteria for the building was completed and is being routed for PNL approval. The building as scoped is 70-ft wide x 100-ft long and with a 24-ft eave height. The entire area is serviced by a 5-ton crane.

Another facility in the FFTF reactor assembly development and testing program is the Seven-Duct Sodium Environment and Mechanism Mockup. This mockup will contain seven prototype ducts in a static sodium pot about 40 ft in length. Scope drawings of the facility and building layouts are being prepared.

Fuel Handling Machine Interface Testing

A study has been made to determine the feasibility of using the Hydraulic Core Mockup, the Seven-Duct Mockup, and the Reactor Cover Mockup for Fuel Handling Machine and Radioactive Maintenance interface testing. A preliminary study of alternate modifications to the original scope of these facilities was made and results were discussed with RDT personnel on January 8, 1968.

Review of the alternates available to combine FHM/RM testing features with the anticipated FTR design support facilities indicates that a decision should be made promptly so that work on the facilities involved with or without FHM modifications can proceed on schedule.

More detailed evaluations are in progress. It is planned that a PNL recommendation on the possible utilization of HCM for the Fuel Handling Machine testing will be made in February.

Sodium Studies Facility

(J. D. Berg, L. J. Defferding)

Sodium Studies Facility provides a centralized laboratory in the 324 Building basement for small experimental test loops and sodium chemistry study equipment.

Small Components Evaluation Loop - (M. O. Rau, J. D. Smith, M. E. Shockley)

The loop was operated at 1060°F and 18.5 gal/min for 744 hours. Total operating time at test conditions is 4944 hours.

Static Sodium Facility - (M. O. Rau, J. D. Smith)

The system helium leakage problem has been partially resolved with the installation of new silver-plated seal rings. It was determined that the previously installed rings were not properly finished by the manufacturers.

Sodium Heat Transfer Loop No. 1 - (M. E. Shockley)

The loop was received on site December 20, 1967. No apparent shipping damage was found during the preliminary inspection of the loop. However, it was noted that the cover gas blanket had leaked off.

The design for the installation of the loop in the 324 SSF complex has been completed. Loop installation is scheduled for completion by mid-February, at which time checkout of the facility will begin.

The design of a dc power supply for the FFTF single pin sodium heat transfer tests is complete. The capacity of this power supply is 144 kW at 96 volts dc and 1500 amp.

High Velocity Sodium Erosion

A program to evaluate the effects of sodium leakage between fuel duct and tubesheets is being formulated. The first phase of this program will establish the presence or absence of the "wire drawing effect" in the leakage

area as effected by clearances and seal designs when subjected to high velocity flowing sodium bypass leakage.

LMEC has been formally requested to review in detail the environmental test requirements submitted to them by letter on January 12, 1968 and advise us of their capability to carry out the tests. Earlier contacts with representatives of LMEC indicated that there may be some difficulty in their providing the needed environment, particularly with respect to pumping requirements. The test program would include continuous flow tests with sodium temperatures between 800° and 900°F, pressures between 75 and 140 psig, and flow rates from 5 to 60 gal/min. The test would be continuous for a six-month period and at that point, terminate or continue on a minimum 30-day cycle depending on the results.

Fin Tube Evaluation

(J. D. Berg)

An effort has been initiated to conduct a proof test of the presently planned fin tubing, for use in the FFTF heat dumps, to assure that the fin tubing can withstand the thermal shocks of the FFTF operating conditions.

A review of the Small Components Evaluation Loop, to determine the compatibility of the facility to perform thermal cycle tests on commercially available fin tubing, is complete. No major modifications to the loop would be required to perform the tests. However, an increase of heat input (~5 kW) may be required to offset the heat loss of the proposed test section.

The methods and facilities to be used for thermal shock tests are being evaluated.

EQUIPMENT PROTOTYPES

C. A. Munro

Reactor Nuclear Control Mechanisms

(J. C. Noakes)

The nuclear control mechanism development program is subdivided into individual component evaluation, prototype control mechanism life testing, and production mechanism pre-operational checkout. Component evaluation

test information will be incorporated into prototype and preliminary design to provide an efficient and reliable control mechanism for the FTR.

Component Evaluation - (R. O. Niethammer and J. R. McBride)

Decelerator Section - The deceleration test facility consists of a test section containing a stationary deceleration section into which a simulated safety rod is dropped. The test section is mounted on pivots to allow an orientation of up to 20° from vertical. Compressed springs, mounted in the upper portion of the test section, impart an initial acceleration to the simulated rod. The test system is connected to a sanitary water supply, as well as to the EDEL-I loop, permitting water at various temperatures and flow rates to be directed through the test section to simulate liquid sodium flow. The test section is instrumented to measure velocity of the piston and fluid pressure at the bottom of the decelerator.

The following modifications to the test facility have been completed:

1. New ram for reduced dash pot clearances.
2. Packing type seal for upper sleeve guide.
3. Dash pot alignment shoulders to speed reassembly.

The decelerator has been reassembled and is now ready for test runs with the new lower clearance ram. The packing type seal will allow high temperature (250°F) runs to be made when necessary.

Latch Component - 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.

Design work is proceeding on the latch test stand. Actuation and loading of the latch will be accomplished with hydraulic cylinders powered by a dual pressure system incorporating an air over oil low pressure reservoir and an air pressurized hydraulic intensifier for high pressure. The actuation system will be designed for automatic operation. Actuation rate, load, and holding time will be adjustable.

The stand will be mounted in a manner similar to the decelerator test stand to allow orientation from the vertical as required.

Drive Assembly - This effort is 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.

The stepping motor and the ball screw, with preloaded ball nut, are scheduled for delivery by the first of February. The test stand should also be installed in the 314 Building by that time. Design of the drive assembly is underway. 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.

Conceptual Studies - (J. C. Hoakes, J. W. Kolb, B. G. Smith) A conceptual design of a remotely operated holddown assembly for the Marvel-Schebler type drive, safety and control rods was completed and is shown on SK-3-14001. This allows a decrease in height of about two feet above the reactor cover, with the possibility of lowering the mechanism further into the upper vessel area.

Reactor Hardware

(J. W. Kolb, B. G. Smith, A. C. Richards)

The reactor hardware development plan calls for preparation of prototype designs for all expendable core components except fuel, coolant duct and irradiation testing facilities.

A conceptual design of a creep compensating holddown connector for driver fuel was completed and is shown on SK-3-14000, sheet 1. A simple hydraulic actuator to operate it is shown on sheet 2. This type quick disconnect has few moving parts, compensates for thermal expansion and creep, and a greater pressure on the mechanical seal can be applied without the galling problems associated with threaded devices.

Work is underway detailing remote connectors for both external and internal latching concepts, for model fabrication.

A conceptual arrangement of reactor cover hardware including closed loop coolant and instrument jumpers was completed during the month and is shown on SK-3-12799. This arrangement is based on the external latching connectors for driver nozzles, and remote welded closures for closed loop nozzles and interconnecting piping. A nominal 10-in. lattice was used as

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SCALE: _____		FAST FLUX TEST FACILITY	
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CONNECTOR LATCHED POSITION

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CHECKED		SCALE 300 GPM [REDACTED] 1500 GPM
BY	B. G. SMITH	
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CONCEPTUAL

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a basis. Electrical and instrument jumpers are remotely removable with a crane and impact wrench. Individual connectors are unlatched and hung to one side to provide clearance for refueling operations.

Closed Loop Nozzle Spool Piece

(J. W. Kolb, R. Kolowith, B. G. Smith)

A nozzle spool piece is being developed which will provide satisfactory internal and external seals.

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 has been modified for this evaluation, and modification of a 200 amp transformer rectifier power supply is underway. Arrangements are being made for investigating the contamination and cleaning problems.

INSTRUMENT DEVELOPMENT

M. R. Wood

Pressure, DP Flow, and Level Sensors for Sodium Service

(T. J. Crocker)

The objective of this task is to develop pressure, flow (differential pressure type), and level instrumentation for FFTF in-vessel and piping applications.

Current activities include: (1) Final checkout of the Static Sodium Facility; (2) Installation of procured transducers and initiation of long-term tests in the facility; (3) Updating of the state-of-the-art report to reflect latest available information; and (4) Preparation of a request for a data acquisition system.

As a final checkout of the Static Sodium Facility, a high temperature test was conducted for a period of three days. Sodium was charged into the four pots, and the temperature and pressure increased to a maximum of 1100°F and 180 psia.

Four transducers have been installed in the facility. These units include three electromechanical transducers and a NaK capillary system.

Initial calibration curves were obtained prior to installation, using helium as the process media. The temperature was limited to 600°F for the electromechanical transducers and 1000°F for the volumetric. The data obtained indicated the short-term operating capabilities for three of the transducers met the manufacturer's specifications, but one transducer showed a 12% nonrecurring zero shift due to aging of the strain gage wire.

In-Reactor Coolant Temperature Sensors

(D. C. Thompson, J. L. Jackson, C. E. Huck)

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

The 54 thermocouples (304 SS sheathed, MgO insulation, chromel-alumel, grounded and ungrounded) in the second set of capsules, were placed into capsules for irradiation at EBR-II and have been shipped.

Reactor Environmental Effects on Signal Cables

(G. J. Dau, R. R. Bourassa, J. L. Stringer)

The object of this task is to develop sufficient knowledge of the processes by which radiation induces currents into a signal cable so that a cable with minimum radiation induced current can be specified for in-core FFTF use. A secondary objective is to determine the instantaneous radiation or dose rate effects on thermocouple calibration.

A capsule has been designed to operate in-core at the freeze point of molten aluminum (approximately 1220°F). The design is similar to that previously used in lead and tin capsules with the exception that better thermal insulation has been provided between the inner cell, containing the aluminum, and the outer container. This improvement will increase the length of the freeze plateau so that any changes in thermocouple calibration occurring during a startup, at full power, or during a scram can be determined continuously over longer periods of time than previously possible.

Experience gained during the previous lower temperature experiments in the TRIGA has led to changes in the measuring technique. The steady state run will continue for several hours in order to fully investigate effects reported by Leonard.⁽¹⁾ Radiation effects on several thermocouples can now be measured simultaneously so that the statistics will be improved.

Low Level Neutron Flux Instrumentation and Delayed Neutron Failed Fuel Monitor

(D. P. Brown)

The objective of this study is to test and evaluate nuclear components of a counting system in an adverse gamma to neutron 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, having less than 20 nsec rise and decay times for pulses from a fission or proportional counter. Although the preamplifier noise requires a high discriminator setting, a 1-in. diameter 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

(1) J. H. Leonard and D. B. Hunkar. Nuclear Applications, vol. 3, p. 718. 1967.

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. It was returned to see if the noise level could be decreased. Two each of three types of fission chambers of a larger size (2-in. diam by approximately 12-in. length), two 1-in. diam boron-lined proportional counters and two additional 0.25-in. diam boron-lined proportional counters have been ordered for further testing. In addition, 25-ft lengths of three types of high temperature (1000°F) cable were ordered.

The detectors will be heated to their rated temperatures to test the effects of both higher temperatures and intense gamma levels. An experimental heating element provided actual tests up to 482°F, with additional heating capacity available. This heating element, designed to heat from one side of the detector, places no neutron absorbing material between the neutron source and the detector.

METALS AND MATERIALS DEVELOPMENT

J. C. Tobin

METALS IRRADIATION AND PROPERTIES

J. E. Irvin

The purpose of this study is to provide mechanical physical properties information for metallic materials which will be used in the FFTF. Operating limits for cladding and structural alloys will be established as a function of temperature, irradiation exposure and reaction with a sodium environment.

Structural Materials and Fuel Cladding Studies

(R. W. Barker, H. R. Brager, J. J. Holmes, R. E. Robbins, A. J. Lovell)

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 applications and at providing a description of material behavior in fast reactor service.

Pre- and post-irradiation biaxial stress-to-rupture properties are being determined for thin wall clad tubing (0.208-in. OD x 0.008-in. wall thickness) of Types 304 and 316 stainless steels. The samples are being irradiated in the EBR-II; discharge date for a fluence of 1×10^{22} n/cm² is estimated to be April 1968.

A limited quantity of unirradiated stress-to-rupture data, for Type 304 stainless steel, has been developed for 1400°F (760°C) test temperature in a helium atmosphere. The material of primary interest is the as-received containing a slight degree of cold work from the final fabrication step. The as-received material has been tested at stresses sufficient to cause rupture in the range of 0.1 hours to 139 hours, as shown in Figure 1. Identical material, annealed at 1950°F (1066°C) for 30 minutes 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.

Figure 1. Stress to Rupture Properties of As-Received (Slightly Cold Worked) and Annealed Type 304 Stainless Steel Prototype FFTF Fuel Cladding

It is observed in Figure 2 that the rupture strain for the annealed material is significantly less than for the as-received material. The failure mode was intergranular for all specimens except for two short time tests. Similarly, the room temperature tensile values shown in Table I show that the as-received material (slightly cold worked) possesses 50% greater ductility than the annealed material.

TABLE I

Uniaxial Tensile Properties of As-Received and Annealed Type 304 Stainless Steel Tubular Material (0.208-in. OD x 0.008-in. Wall)

Material	0.2 Offset Yield Strength, psi	Ultimate Tensile Strength, psi	Uniform Elongation, %	Total Elongation, %
304 As-Received	40,000	100,000	65	76
304 As-Received	41,000	99,400	66	73
304 Annealed*	35,800	89,000	43	52
304 Annealed*	34,800	88,400	37	45

* Annealed at 1950°F (1066°C) for 30 minutes in vacuum.

The inferior ductility of the annealed material has been attributed to the large grain size which was produced during annealing of the slightly cold worked material. A comparison of the grain size of the two conditions is shown in Figure 3. Note, in the annealed material, bands of fine grains adjacent to grain whose diameters are approximately equal to wall thickness.

Ultimately, stress-to-rupture testing will be performed to establish base curves for Types 304 and 316 stainless steels at temperatures of 1400, 1200, and 900°F (760, 649, and 482°C).

The effects of swelling in austenitic stainless steels are being studied, after irradiation to high neutron fluences. Density measurements have been made on sections of two annealed Type 304 stainless steel capsule tubes which accumulated nominal fluence levels of 2.0 and 2.8×10^{22} n/cm² at about 800°F (EBR-II estimate). The results for the 2.0×10^{22} capsule were reported earlier for the as-irradiated condition. Subsequently, those

Figure 2. Rupture Strain of As-Received (Slightly Cold Worked) and Annealed Type 304 Stainless Steel Prototype FFTF Fuel Cladding

Figure 3. A Comparison of the Grain Size of the As-Received (Top) and Annealed (Bottom) Type 304 Stainless Steel Fuel Cladding

specimens were annealed for one hour at 1000°F and a density average of 7.917 g/cm² was measured. This compares to 7.913 ± .004 prior to annealing and 7.928 ± .003 prior to irradiation.

Results from the 2.8×10^{22} capsule indicate a maximum decrease in density of 0.37%, the actual measured density being 7.899 g/cm². In this latter capsule, the change in density corresponds to the fluence and temperature along the capsule. The maximum swelling occurs somewhat above the midplane.

Density measurements have also been taken on 304 and 348 tensile specimens which were irradiated to a fluence of about 1.7×10^{22} at 1100°F but not tested. Specimens in both the annealed and 25% cold worked state were studied. While no control specimen is available, it is possible to compare densities between the cold worked and annealed conditions. The cold worked specimens were found to be somewhat more dense than the annealed. The difference in densities was about 0.15%.

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 is presented in Figure 4, which 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 voids having diameter less than about 50 Å.

A critical need exists for specimens of annealed 304 stainless steel which have a fluence level of 5×10^{22} n/cm² or above. Without data at these exposures, it will be impossible to accurately predict swelling at LMFBR service exposures which will, in general, exceed 10^{23} n/cm². Figure 4 predicts about a 10% volume change at 9.2×10^{22} n/cm² and about 60% at 2×10^{23} n/cm². The magnitude of these volume changes indicates the importance of developing an accurate and reliable swelling relationship.

Figure 4. Irradiation Swelling of Austenitic Stainless Steel

The electron microscopy examination was continued on AISI Type 304 stainless steel irradiated in the ETR to 6×10^{21} n/cm² (>1.0 MeV) at 554°F (290°C). This material was examined in order to better understand the high temperature irradiation hardening which persisted to the highest test temperature, 1382°F (750°C). The photomicrographs, presented last month, of the gage section of the material as tested at 1382°F (750°C) and of the same material having an additional one hour vacuum anneal at 1600°F (870°C) were analyzed, Table II. In addition, sections of the material tested at 1382°F (750°C) were vacuum annealed one hour at 1800°F (980°C) or 2000°F (1090°C), and examined by electron microscopy (Figures 5 and 6 respectively, and Table II). The significant observations were:

1. The cavities are more than sufficient to cause high temperature hardening.
2. The cavities were present at annealing temperatures up to 2000°F (1090°C).
3. Cavities were located on grain boundaries and the grain boundaries did not have a denuded zone.
4. No strain fields were observed around the cavities for material vacuum annealed within the temperature range 1380-2000°F (750-1090°C) and furnace cooled.
5. The amount of helium gas required to be in equilibrium with the cavities surface tension (assuming $\gamma = 1500$ ergs/cm²) was more than 25 times the amount of helium calculated to be produced by the reactor irradiation (~ 12 ppm helium).
6. The total surface area of the cavities was constant over the investigated temperature range of 1382°F (750°C) to 2000°F (1090°C) while the total volume of the cavities increased with increased annealing temperature. The consistency of macroscopic cavity surface area indicates that the cavities are in equilibrium with the surface tension.

Mass spectrometry analysis of gases evolved from vacuum melting of the highly radioactive samples are being initiated. This technique should provide information on the magnitude and species of gases in the ETR irradiated Type 304 stainless steel.

TABLE II

ELECTRON MICROSCOPY OBSERVATIONS OF TYPE 304 STAINLESS STEEL
IRRADIATED IN THE ETR TO 6×10^{21} n/cm² (>1.0 MeV) AT 554°F (290°C)

Material Tensile Tested at 1382°F (750°C)
 Plus a Vacuum Anneal One Hour at the
 Indicated Temperature

	As Tested	Anneal Temperature °F		
		1600	1800	2000
Cavity density (cc ⁻¹)	2×10^{15}	3×10^{14}	2×10^{14}	1.5×10^{14}
$\Delta V/V$ (%)	0.17	0.25	0.50	1.2
Helium in equilibrium with surface tension of cavities (ppm)	300	275	400	600
Average cavity diameter (Å) (first moment/third moment)	110/115	210/250	270/350	460/575
Total cavity surface area (10^{19} Å ² /cc) (Calculated using first moment average cavity diameter/third moment average cavity diameter)	2.5/3.0	1.5/2.0	2.2/2.7	3.0/4.0
Total cavity volume (10^{21} Å ³ /cc) (Calculated using first moment average cavity diameter/third moment average cavity diameter)	3.0/3.5	2.5/4.5	4.5/9.2	13.5/21

Figure 5. Microstructure of AISI Type 304 Stainless Steel after Irradiation at 554°F (290°C) to a Fast Fluence of 6×10^{21} n/cm² and Vacuum Annealed for One Hour at 1800°F (980°C) after Testing at 1382°F (750°C).

Figure 6. Microstructure of AISI Type 304 Stainless Steel after Irradiation at 554°F (290°C) to a Fast Fluence of 6×10^{26} n/cm² and Vacuum Annealed for One Hour at 2000°F (1090°C) after Testing at 1383°F (750°C)

Type 410 stainless steel was irradiated to 6×10^{21} n/cm² (>1.0 MeV) in the ETR at 554°F (290°C). This ferritic stainless steel has been previously tensile tested over a temperature range of 68-1112°F (20-600°C). At temperatures below 932°F (500°C), the material had the usual displacement damage strengthening and embrittlement behavior. However, at the 1112°F (600°C) test temperature, the strength and ductility of the material was consistent with control sample values.

A piece of the grip section of the Type 410 stainless steel, which was tensile tested at 1112°F (600°C) was examined by transmission electron microscopy. Due to the material being ferritic, the sample caused significant distortion and shifting of the transmitted electron beam, greatly reducing the resolution of the bright field image. Under these adverse observational conditions, no cavities were observed.

The effect of irradiation on Type 316 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 III.

The tensile specimens from GEH-22-3 were tested at room temperature, 572, 752, 932, 1112, 1292 and 1382°F. Their strength data are shown as a function of test temperature in Figure 7. These data show that irradiation at approximately 1112°F and 1004°F results 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. The elongation data for these various test temperatures are presented in Figures 8 and 9. The uniform and total elongation are lower at all test temperatures than the as-fabricated controls.

The tensile data for AISI Type 304 stainless steel also irradiated in the flowing sodium capsule GEH-22-3 and tested at 572, 752, 932, 1112, 1292, and 1382°F are presented in Figures 10, 11, and 12. Type 304, like the

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

TABLE III
IRRADIATION CONDITIONS IN THE FLOWING SODIUM CAPSULES*

<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	$\sim 1.4 \times 10^{21}$	~ 100 days
GEH-22-3	3 & 4	1004 ± 90°F	$\sim 1.4 \times 10^{21}$	~ 100 days
GEH-22-4	1	1400 ± 45°F	$\sim 8.0 \times 10^{20}$	~ 55 days
GEH-22-4	2	1382 ± 45°F	$\sim 8.0 \times 10^{20}$	~ 55 days
GEH-22-4	3	1346 ± 45°F	$\sim 8.0 \times 10^{20}$	~ 55 days
GEH-22-4	4	1274 ± 45°F	$\sim 8.0 \times 10^{20}$	~ 55 days
GEH-22-4	5	1202 ± 45°F	$\sim 8.0 \times 10^{20}$	~ 55 days

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

Type 316, shows an increase in strength and a decrease in ductility at all test 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 III. 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 at test temperatures from room temperature to 1382°F. The specimens tested from this capsule showed increased yield strengths (see Figure 13), 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 hardening that is effective up to and including the 1382°F test temperature. The ductility of the irradiated Type 316 specimens as shown in Figures 14 and 15 are lower than as-fabricated controls for all test temperatures.

Figure 7. The Effect of Test Temperature on the Strength of AISI 316 Stainless Steel Irradiated in a Flowing Sodium Environment (GEH-22-3)

Figure 8. The Effect of Test Temperature on the Ductility of AISI 316 Stainless Steel Irradiated in a Flowing Sodium Environment (GEH-22-3)

Figure 9. The Effect of Test Temperature on the Ductility of AISI 316 Stainless Steel Irradiated in a Flowing Sodium Environment (GEH-22-3)

Figure 10. The Effect of Test Temperature on the Strength of AISI 304 Stainless Steel Irradiated in a Flowing Sodium Environment (GEH-22-3)

Figure 11. The Effect of Test Temperature on the Ductility of AISI 304 Stainless Steel Irradiated in a Flowing Sodium Environment (GEH-22-3)

Figure 12. The Effect of Test Temperature on the Ductility of AISI 304 Stainless Steel Irradiated in a Flowing Sodium Environment (GEH-22-3)

Figure 13. The Effect of Test Temperature on the Strength of AISI 316 Stainless Steel Irradiated in a Flowing Sodium Environment (GEH-22-4)

Figure 14. The Effect of Test Temperature on the Ductility of AISI 316 Stainless Steel Irradiated in a Flowing Sodium Environment (GEH-22-4)

Figure 15. The Effect of Test Temperature on the Ductility of AISI 316 Stainless Steel Irradiated in a Flowing Sodium Environment (GEH-22-4)

Fuel Cladding Characterization

(M. M. Paxton)

The characterization of two lots of as-received AISI Type 304 stainless steel, .250-in. diam x .016-in. wall, is continuing.

Room temperature uniaxial tensile tests have been run on the "E" and "F" lots of tubing. The resulting data are being analyzed.

Room temperature hydrostatic burst tests on "E" tubing have been completed. Due to equipment design, the load time was one to two minutes, as compared to 14 minute load times on the elevated temperature burst tests.

Burst tests on defective and nondefective "E" lot tubing have been completed at 1800°F. The loading time was 13 to 14 minutes.

Ten defects were identified by NDT methods in a random sampling of tubing. Six of the ten defects were identified as cracks about 1/8-in. long by 1 mil deep. Some dents and pits were also identified. Only one defect per tube was included in the sections tested. The defect region was carefully identified by the distance from either end of the tube. In all but a few tubes, it was impossible to identify the defect with a naked eye.

The average burst pressure for defect free tubing is 1400 psi or 9530 psi hoop stress. The average burst pressure for the defective tubing is 1404 psi or 9560 psi hoop stress. Three of the specimens burst in the defective zone. However, no decision has been made as to whether or not the failure was initiated at the defect. Three specimens ruptured entirely outside the defective region. The data are summarized in Table IV.

Test Facilities

(R. H. Todd, J. A. Yount)

The comprehensive investigation of candidate structural and cladding materials for the FFTF has created a need for a facility to provide the necessary test environments. The facility is under construction and is approximately 72% 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.

TABLE IV
BURST DATA FOR DEFECTIVE AND NONDEFECTIVE "F" LOT TUBING

<u>Test Temperature</u>	<u>Burst Pressure, psi</u>	<u>Hoop Stress, psi</u>	<u>Measurable Post OD in.</u>
Room Temperature	10,700	72,870	.300
	10,700	72,870	.305
	10,700	72,870	.298
	10,800	73,550	.304
	10,800	73,550	.305
1800°F	1,365	9,296	--
	1,390	9,465	.316
	1,395	9,500	.303
	1,410	9,620	.305
	1,450	9,865	.294
1800°F*	1,375	9,365	.327
	1,380	9,400	.321
	1,385	9,435	.325
	1,420	9,670	.305
	1,425	9,705	.315
	1,440	9,805	.303

* Defective tubing.

Two uniaxial stress-to-rupture tests, conducted at 1202°F (650°C) in air, have been completed during the last month. Type 304 stainless steel specimens were tested to 87.5 and 440 hours. The solution treated type 304 steel was cold worked 20%, at room temperature or at liquid nitrogen temperature. The specimen cold worked at liquid nitrogen temperatures had only an 87.5 hour life compared to 440 hours for the specimen cold worked at room temperature. The specimen structure is being further analyzed to determine the reason for this difference.

Two additional stress-to-rupture machines are awaiting tests for accuracy of alignment. Parts and equipment have arrived for these alignment tests, which will soon be completed. Piping for building helium service has been installed and a high pressure helium compressor has been received. Testing in a helium atmosphere will soon be possible.

FABRICATION

S. M. Gill

Fuel Duct and Closed Loop Development

(S. M. Gill)

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

On-Site Fuel Duct Fabrication Development - (F. B. Quinlan)

A hardened steel mandrel of hexagonal cross-section was designed and fabricated. The design permits attachment of additional sections so that straight hexagonal and tapered hexagonal shapes may be formed. Current use for the mandrel is in forming straight hexagonal shapes for Core Components Test Loop (CCTL) tubes, but it can be adapted for forming the more complex shapes required for flow guides.

The mandrel is currently being used in forming an additional CCTL tube, BW-3, utilizing a rolled and welded tube blank.

The first draw pass on BW-3 was made without the mandrel. The mandrel had to be forced into the tube, indicating that the guide tube ID, in the cylindrical portion, is too small. An increase in the dimensions of the cylindrical tube to $5.130 + 0.020$ -in. OD with a $0.150 \pm .005$ -in. wall from $5.120 + 0.020$ -in. OD would improve this condition. The third pass, which forms the tube tightly onto the mandrel, required a drawing force of 43,000 pounds. This was about 25% more than the force required when a sand core was used. The high force is caused by reduction of the tube wall due to nonuniform and excessive thickness. This wall thinning and resulting high deformation forces also indicate the need for more precise control of the original tube dimensions. A detrimental effect of these high deformation forces is the shrinking of the tube onto the mandrel so tightly that a tube expanding operation is required to remove the mandrel.

Six seamless Type 304 stainless steel tube blanks are being fabricated. These tubes will be formed by rocking in a "Pilger" mill starting with fully machined and honed billets. This process is being followed in attempts to obtain seamless tube with uniform wall thickness.

Cleaning of Tubes and Components - (R. V. Bowersock)

Procedures for cleaning the components of fuel ducts and closed loops are under investigation. All components that are exposed to the sodium coolant system must be clean so as not to contaminate the system, either with nuclear poisons or with materials that might have adverse effects on 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 that could possibly 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 that will do the job and will meet the product criteria established in Provisional Product Specification for FFTF Fuel Cladding, No. PNL 70.0, Specification for Cleaning FFTF Core Components, has been requested from several manufacturers that specialize 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 is considerable difference in removing 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.

Type 304 stainless steel CCTL tube BW-1 (weld orientated in corner of hex) was wiped with zylene, then alkaline cleaned. This tube was used in setting up a nondestructive testing facility in the 314 Building. Although the tube in general looked bright and clean, the cleaning did not remove lead pencil markings nor the light oxide film present on one end. Close examination by NDT disclosed a few small drops of drawing lubricant still remaining on the internal surface.

It should be noted that the cleaner used does not meet all the requirements of Specification PNL 70.0 in that it contains silicates. This probably is not detrimental when cleaning small components that can be transferred from the hot bath to the rinse rapidly. However, for heavy wall sections or for large components that cannot be transferred to the rinse without the initiation of surface drying, this type cleaner should not be used because of the rinsing difficulties associated with the potential silicate "burn-on".

Screening of cleaning procedures will continue with the scope of the search extended to include cleaning of Type 316 stainless steel and Incoloy alloy 800.

Wear Resistant Coatings for Tube Components - (J. H. Hoage)

Standard test specimens, for testing material combinations in liquid sodium at the Liquid Metals Engineering Center, are composed of wear plates and pins. A number of the specimens are being prepared from 304 stainless steel for subsequent application of wear resistant coatings and testing. Discussions are continuing regarding preparation of experimental coatings on the standard test specimens.

Closed Loop Tube Insulation Studies - (G. R. Sawtelle)

The closed loop insulation studies to date have shown the following:

- A double-walled tube with an insulation gap between them is the desired basic construction.
- Selection of an insulation gap filler will require additional information from analysis and experimental verification.
- Argon, neon or some other low thermal conductivity gas would provide better insulation than helium.

- Solid insulation materials like Min-K (a felted alumina fiber blanket) and radiation barrier foils may provide advantages as insulation gap fillers.
- A recirculating gas should be considered as a means of heat transfer control and/or backup heat removal system.

Table V indicates the experimental evaluation required on this program.

NONMETALLIC MATERIALS

D. E. Mahagin

Irradiation Behavior of Shield Materials

(D. E. Baker, D. E. Mahagin)

The irradiation behavior of candidate shield materials is being investigated to determine the one most suitable for use in the FTR.

The first of three serpentine irradiation capsules was transferred to the Radiometallurgy Laboratory for disassembly and examination. This capsule contained five sample subassemblies which were irradiated to an estimated 2×10^{20} nvt, $E > 0.18$ MeV, at five temperatures ranging from 500 to 900°F. Iron, nickel and aluminum-cobalt flux monitors were recovered from the capsule and are being counted. After disassembly, the 0.016-in. thick type 321 stainless steel fluted cans containing the serpentine appeared to be in good condition. There was no evidence of swelling or gross dimensional changes. Detailed examination of the specimens is in progress. Irradiation of the two remaining capsules is continuing.

The initial loading of the 36 carbon and graphite irradiation capsules has been shipped to ETR. Fabrication of the second series (45 capsules) is in progress.

Radiation Effects in Absorber Materials

(A. L. Pitner)

Final capsule design has been established for the irradiation of B_4C in an EBR-II seven-pin subassembly. Three forms of B_4C will be irradiated - pellets (99% TD and 70% TD) and powder. The as-received pellets were dimensionally out of specification and will require considerable grinding prior to pin assembly. Characterization, which has begun on the material,

TABLE V

INSULATION EVALUATION FOR CLOSED LOOPS

<u>Insulation Methods</u>	<u>Experimental Evaluation</u>	
A. Stagnant Gas Cap	A. Determine Effective Thermal Conductivity	F. Determine Differential Wall Expansion Effect
1. Helium Neon Argon	1. Gases with spacers	1. Spacers
2. Polish radiating surfaces to reduce emissivity	2. Solids with gas	2. Solid materials
		3. Bellows
B. Solid Insulants with Interstitial Gas	B. Develop Polishing Technique to Obtain Optimum Emissivity	G. Determine Effect of Flooding Gap with Sodium
1. Min-K, Foils	1. Determine gas purity required to maintain emissivity	H. Determine Effect of Tube Bowing on Integrity of Insulation
C. Method for Control of Thermal Conductivity	C. Determine Optimum Spacer Design	
1. Flood gap with sodium	1. Considerations:	
2. Circulate gas	a. Thermal Conductivity	
	b. Local Thermal Gradients	
	c. Attachment Methods	
	2. Determine the need for spacers with solid insulant materials	
	D. Determine Effect of Reduced Gap Width (less than 1/8-inch)	
	E. Develop Assembly Procedure with:	
	1. Spacers	
	2. Solid materials	

includes chemical analyses of B/C ratio and impurity levels and metallographic examination to determine grain size, phases present, and porosity size and location. Following material characterization, the samples will be encapsulated in capsules 0.8-in. diam x 58-in. long and shipped to EBR-II for irradiation. Capsule design was adjusted to provide irradiation temperatures near 1100 and 1300°F for each sample type. Target burnup levels of the B¹⁰ are 3, 5, 7, 9, and 12%. These burnups will be achieved by discharging and replacing selected capsules at staggered intervals. The capsules are identical in sample loading, and discharging them in pairs will provide a checkpoint at each exposure interval.

Chemical Compatibility of Shield and Control Rod Materials

Tests are in progress to determine the compatibility of shielding and container materials which have potential application in the FTR shield. Representative container materials chosen are Types 304 and 316 stainless steel and AISI 1015 steel.

These tests are conducted by placing five tensile specimens of the container material in a stainless-steel tube and packing the tube with the shield material. The tube is then evacuated, to remove air, and backfilled with helium. One end of the tube containing the serpentine and metal samples is positioned in a furnace and heated to 932°F (500°C). The other end is permitted to extend out of the furnace and maintained at a low enough temperature to permit gage measurements of the pressure.

In order to determine the extent of reaction, three methods are used: (1) metallography, (2) microhardness tests, and (3) short-time tensile tests.

During this report period, 330 hour-runs have been conducted at 932°F (590°C) with serpentine and both stainless steel (Type 304) and mild steel.

As the test progressed, it was noted that water was released from the serpentine until a partial pressure equal to the vapor pressure of water in the cool end of the tube was built up. Excess water vapor condensed in the cool end of the tube.

Evaluation of the metal samples is not complete. However, they were oxidized by the treatment. No significant weight changes were recorded.

The yield strength of both steels was increased, and this is indicative of solution hardening (probably by hydrogen). The remaining portions of the examination will define the extent of the reaction and its nature.

MATERIALS, CODES AND STANDARDS

R. A. Moen

FFTF Materials Design Data

(R. A. Moen)

The intent of the FFTF Materials Design Data Manual is to provide design curves for material identified as having potential use 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 VI is an example of 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 are currently in progress for graphite (electrode grade), carbon, serpentine, and tantalum. These will be issued by the end of the month. Little or no irradiation effects data will be presented in these summaries because very little is recorded in the literature. There is, however, a relatively large amount of irradiation data on nuclear-grade graphite which will be presented in a later draft summary on that subject and which will be applicable to the electrode-grade graphites under investigation at PNL (Task RS-5, Irradiation Behavior of Shield Materials).

The literature is continually being examined to insure the availability of up-to-date radiation effects data for austenitic stainless steels, 304 and 316.

Codes, Standards and Specifications Analysis

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.

TABLE VI
DATA SUMMARY ON POTENTIALLY USEFUL PLASTICS

<u>Useful Materials for Range of Temperature 450-550°F</u>	<u>Resistance to Alkalies</u>	<u>Radiation Effects</u>
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 effect 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 molecules per 100 ev absorbed

PROCESS AND MATERIALS SPECIFICATIONS

J. W. Helm

DEVELOPMENT PROGRAMS

J. W. Helm

Surveillance and In-Service Testing

(J. H. Cox)

The objective of this program is threefold:

- Surveillance and in-service inspection of critical items which have a direct influence on the reliable and safe operation of FTR and which must be analyzed in depth for inclusion in the Safeguards Analyses Reports.
- Surveillance and in-service inspection of critical items for early detection of abnormal or unexpected behavior.
- Obtain information on materials and component behavior in a high fast flux, sodium-cooled, reactor system to extend knowledge of benefit to the LMFBF Program.

The surveillance work plan is completed through preliminary design and all interfaces have been identified. Input to preliminary design is scheduled to be complete, for the whole system, by January 1969. Major milestones are:

Initiate Component Development	4-68
Establish System Requirements	5-68
Preliminary Damage Calculations	5-68
Preliminary Dosimetry System	8-68
Preliminary Sample Program	10-68
Preliminary Method of Data Interpretation	10-68
Preliminary Nondestructive Testing System	11-68
Final Report for Preliminary Design	1-69

Three sites were visited, during this report period, to inquire about surveillance requirements and availability of applicable systems. The three sites were Atomic Power Development Associates, Inc., (APDA), Detroit, Michigan; LMFBF Program Office, ANL; and Southwest Research Institute in San Antonio, Texas.

The APDA surveillance program consists primarily of the irradiation of one subassembly containing metallurgical and corrosion samples. The subassembly can be discharged, any sample removed and replaced, and recharged during an outage. There is extensive boroscope development underway which, when completed, should prove valuable for the FFTF. There were no non-destructive testing techniques that they considered ready for application to a vessel or primary piping.

The FFTF approach to development of a surveillance system was discussed at the LMFBR Program Office and the general approach was found consistent with their planning. At the present time it appears that the project deadlines are too near to allow incorporation of surveillance requirements generated through the Program Office. However, there should be areas of mutual benefit as each program develops.

Southwest Research appeared to be experienced in some surveillance areas, especially in identification of requirements and application of ultrasonic inspection techniques for the vessel and primary piping.

Materials Damage Analyses - (G. E. Russcher)

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 the thickness of 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 16. 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 that primarily the PKA population above 3 KeV contributes to the stable vacancy cluster production and hence, we assume, void density. Applying this criterion and its implied assumptions to the PKA spectra leads to the following conclusions:

(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.

Fig. 16. Probable Energy of Primary Knock-on Atoms (PKA)
Produced in Iron

1. Specimens irradiated in DFR and EBR-II incur larger populations of high energy PKA's than do specimens irradiated in the FTR fuel region (see Fig. 16). This indicates that a FTR material design based on EBR-II and DFR data would be conservative. The ranking of 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.
2. 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.
3. The major elements of stainless steel produce different PKA spectra; however, these calculations show that iron and chromium PKA spectra are nearly identical. 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; they are:

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

Analysis of Radiation Damage at Elevated Temperatures - (D. G. Doran)

An essential element in applying damage calculations to problems of interest to the FFTF Project is to introduce finite temperature effects, i.e., to account for the thermally produced changes in the initial damage state that takes place during irradiation at elevated temperatures.

A computer code called ANNEAL has been obtained, which was developed by J. R. Beeler and D. Besco at General Electric Company, to simulate annealing behavior. This code takes a primary (0°K) damage state, consisting

of a particular spatial distribution of single and clustered defects (vacancies and interstitials) on a three-dimensional lattice, and allows them to interact in accordance with predetermined activation energies. These activation energies determine the relative probabilities that a mobile defect will move to one of several possible neighboring sites -- the motion will not be random if another defect is sufficiently close, but will be correlated to the position of the neighboring defect. In the operation of the code, each time step consists of examining for possible motion (in a random order), each element (vacancy or interstitial) in a table of defects, moving the element if appropriate, revising the tabulation of defects, taking into account the formation of new clusters and the recombination of vacancy-interstitial pairs. As many time steps can be run as desired and the final configuration can be stored on punched cards to be used as input for further runs.

The ANNEAL code has run satisfactorily on the Univac 1108 computer, but in its present form is seriously limited in scope. It is set up to simulate the annealing of pure α -iron (body-centered cubic) at essentially room temperature, at which temperature only single and di-interstitials are mobile. Dissociation of clusters is not permitted, so that interstitial clusters can only grow, while vacancy clusters can only suffer attrition through recombination with interstitials migrating to them.

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 is being made to extend ANNEAL to cover this condition. This major modification to the code involves several steps:

1. A method for handling the multitude of vacancy interactions must be found that incorporates the essential physics, yet is tractable. The study of α -iron will be continued, not only because of the start made in ANNEAL, but also because there are more pertinent data in the literature for α -iron than for any other metal. This study of published results has led to the tentative formulation of a method for handling vacancy migration. It entails identifying a few key configurations involving correlated motions of vacancies.

2. This identification of key configurations and the quantitative analysis of these configurations must be corroborated. The published computer calculations on α -iron have been primarily concerned with determining stable configurations and migration energies of isolated defects, rather than activation energies for correlated motion of defects in the vicinity of other defects. Only for the interaction of two single vacancies are extensive results available. It will be necessary, therefore, to determine a number of activation energies using the DEFECT computer code. This code, also developed by Beeler, has been operational here for several months. It is used to calculate the energies of particular defect configurations in a crystal lattice, the principal required input being an appropriate interatomic potential. A suitable potential is available for α -iron; however, it is believed that the ultimate results from ANNEAL will not be very sensitive to the choice of a potential for DEFECT.
3. Correlated jump probabilities must be determined from the results of Steps 1 and 2.
4. ANNEAL must be modified to incorporate the new physics obtained in Steps 1-3.

Some aspects of the steps outlined above are being carried out concurrently. For example, DEFECT runs (Step 2) are being set up, and identification of those parts of ANNEAL that need revision or amending has begun (Step 4).

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