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BNWL-639

FAST FLUX TEST FACILITY DEVELOPMENT
MONTHLY PROGRESS REPORT

F. W. Woodfield

November 1967

AEC RESEARCH &
DEVELOPMENT REPORT

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PACIFIC NORTHWEST LABORATORY

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FAST FLUX TEST FACILITY
DEVELOPMENT MONTHLY PROGRESS REPORT
November, 1967

F. W. Woodfield, Deputy Project Manager
for Development
December 10, 1967

Approved by:

J. W. Helm

PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON

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PREFACE

This is the third issue of a report issued monthly covering technical details and status of development programs performed at PNL in support of FFTF.

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FFTF DEVELOPMENT MONTHLY PROGRESS REPORT, NOVEMBER, 1967SUMMARY

The FFTF needs and the current status of development programs being performed at various laboratories throughout the USA on in-line instrumentation for sodium purity measurements have been described.⁽¹⁾

Type 304 stainless steel capsules containing mixed fission products and sodium are being run at 1200°F to measure the depth of diffusion for the fission products. The first capsule was opened after 1052 hours at 1190 ± 20°F. Preliminary results indicate 99% of the activity had diffused less than 1 mil.

Data are reported on the control rod decelerator tests. A maximum velocity of 10 fps has been attained. Concept drawings for a regulating non-scam rod have been completed.

Reactor hardware concept drawings have been completed for an open loop nozzle cap with a 10-inch lattice. A remote welded nozzle mockup for closed loops is being fabricated.

Post-irradiation testing of AISI-304 stainless steel which had been irradiated in the EBR-II to 1.7×10^{22} nvt at 1000 ± 100°F revealed a marked increase in 0.2% offset yield strength and a decrease in ductility. The data demonstrated that irradiation hardening can occur in metals irradiated at one-half the absolute melting temperature to high fluences.⁽²⁾

Annealed and 25% cold worked AISI-304 stainless steel irradiated at 550°F (290°C) to neutron fluences ranging from 0.15 to 6×10^{21} nvt exhibited increased yield and ultimate tensile strengths over a test temperature range of 73 to 1382°F (23-750°C).

(1) T. F. Demmitt, In-Line Instrumentation for Sodium Purity Measurements in FFTF: Status Report, BINWL-562. Pacific Northwest Laboratory, Richland, Washington, November, 1967.

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

(3) R. E. Robbins, J. J. Holmes and J. E. Irvin, "Post-irradiation Tensile Properties of Annealed and Cold-Worked AISI-304 Stainless Steel," Transactions, Vol. 10, No. 2, p. 488-9, American Nuclear Society, 1967.

The decomposition of serpentine, being investigated as a shield candidate material, was found to follow an Arrhenius rate equation at temperatures greater than 900°F.

Comparison of the relative spectral shapes reveal the increase in the fraction of lower energy neutrons in the FTR as compared to both EBR-II and DFR, and emphasize the necessity of detailed analyses of material damage effects in the three reactors.

COOLANT SYSTEMS DEVELOPMENT

B. M. Johnson

CORROSION AND CHEMISTRY

D. W. Shannon

Selection and Testing of Radioactive Sodium Analysis Methods

(F. A. Scott)

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

Oxygen Analysis - (T. F. Demmitt)

This procedure utilizes mercury amalgamation to separate the insoluble residue from the sodium sample. The residue is then hydrolyzed and analyzed by titrimetric and spectrophotometric methods.

Setup and calibration of the mercury amalgamation method is complete and the procedure is being used routinely to perform service analyses for various PNL components. A total of five samples were analyzed during the past month. The equipment has been dismantled for the move to the 324 Building Sodium Studies Facility and will be out of service four to six weeks.

Hydrogen Analysis - (T. F. Demmitt)

The procedure involves amalgamation of the sodium sample with mercury followed by refluxing of the amalgam to release hydrogen.

Calibration has been delayed pending receipt of the chromatograph needed to perform hydrogen analyses. The chromatograph is scheduled to arrive during December. The equipment has been dismantled for the move to the 324 Building Sodium Studies Facility.

Nitrogen Analysis - (T. F. Demmitt)

Evaluation of the silver nitrate polarographic titration method for measuring cyanide was discontinued. This action was taken after additional tests confirmed the preliminary conclusion that the procedure does not have enough sensitivity to detect cyanide levels as low as 25 micrograms, the minimum requirement for the method.

An alternative method for cyanide analysis involving potentiometric measurement with a glass membrane electrode (similar to pH measurement) will be evaluated next. Procurement of the electrode assembly required

for this work is now in progress. Data supplied by the manufacturer indicates an ultimate sensitivity of 10 micrograms of cyanide may be possible.

Carbon Analysis - (F. A. Scott)

The PNL carbon analysis method utilizes a total combustion gas chromatograph method. We have been consistently recovering only 70% to 80% of the carbon added to a sodium sample as a spike. In October data were obtained which indicated that the low recoveries were related to insertion of the spike samples through the glove box air lock. A series of recovery tests in which this variable was controlled was completed in November. Recoveries continued to be low. A continuing effort is underway to define the cause of the problem.

Planning for the dismantling of the equipment for the move to the 324 Building Sodium Studies Facility has begun. No development work on carbon analysis will be done during the four to eight weeks needed for this move.

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

Work has continued on the design study of the high level analytical facility to support the FFTF. The hot cell floor plan is complete and the floor plan of the unshielded laboratory facilities is nearing completion.

Selection of FFTF Sodium Sampling Methods

(F. A. Scott)

The objective of this study is the development of remote sampling methods for radioactive sodium. It has been concluded⁽¹⁾ that a flow-through sampler tube appeared best with a distillation sampler possibly desirable for some applications. Remote hardware development work on the flow-through sampler has been initiated at PNL. Recommendations are being prepared for redirection of distillation sampler work at LASL to support the FFTF distillation sampler concept.

Selection and Testing of In-Line Sodium Analysis Instruments

(T. F. Demmitt)

The FFTF needs and the current status of instrument development programs,

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

being performed at various laboratories throughout the USA, on in-line instrumentation for sodium purity measurements have been described.⁽¹⁾

Information obtained from two companies regarding development program status and plans for in-line oxygen, carbon, and hydrogen meters was evaluated in terms of FFTF needs. It was concluded that the content, schedules, and specific goals of these programs are not positively coordinated with FFTF requirements. Redirection of these programs to ensure provision of information needed on FFTF decision dates appears to be mandatory. Specific recommendations for modification of the programs to meet FFTF needs are being prepared.

Sodium Cover-Gas Interactions

(H. P. Maffei)

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

The reaction rate of 100 ppm methane, in helium, with molten sodium at 800 to 1200°F was measured. Very little reaction occurred between the methane and the molten sodium at 800°F. During a period of 21 hours the methane concentration decreased from 100 ppm to 70 ppm and appeared to have reached an equilibrium with the sodium at this point. An evolution of hydrogen was noticed during the first few minutes of this run. After an initial concentration of 48 ppm, the hydrogen diminished to 29 ppm during the duration of the run. At 1200°F, the reaction rate of the methane with the molten sodium was much faster. Virtually all of the methane was consumed by the sodium over a period of 24 hours. The methane decreased from an original concentration of 95 ppm to a concentration of approximately 1 ppm, as measured by a gas chromatograph. A large evolution of hydrogen was seen during this run. The initial hydrogen con-

(1) T. F. Dermitt, In-Line Instrumentation for Sodium Purity Measurements in FFTF: Status Report, BNWL-562. Pacific Northwest Laboratory, Richland, Washington, November, 1967.

centration was 120 ppm which decreased to 39 ppm during the course of the run. A steady increase of nitrogen in the cover gas was measured during the course of the run, building up to a concentration of 700 ppm after 24 hours. It is not yet known if this nitrogen buildup was due to a leak in the system or diffusion of nitrogen through the wall of the 304 stainless steel capsule.

Preliminary analysis of the data indicate the reaction is second order with respect to methane concentration. The low reaction rate at 800°F indicates appreciable methane concentrations may be tolerable during FFTF shutdown periods.

Liaison on Corrosion and Mass Transfer and Sodium Technology

(D. W. Shannon)

Visits were made to General Electric Company, San Jose, California, to discuss the plan for operation of Loop 9 as a corrosion model of an FFTF closed loop, evaluating high heat flux, high velocity and sodium temperatures above 1300°F. Tentative startup date for this test is February 1968.

A visit was made to Aerojet-General Nucleonics in San Ramon, California, to review their capabilities for research and development in support of FFTF.

A request for D. W. Shannon to serve as a consultant to the Sodium Technology Task Group of the LMFBR Task Force was received from E. C. Kovacic of APDA. A trip was made October 18 and 19, to attend the working meeting of the Task Group in Detroit. The draft of the Task Group report was reviewed and commented upon. Subsequent review indicates a sizable number of the suggestions made by PNL were adopted and the report generally supports FFTF development needs.

Evaluation of FFTF Sodium Cleaning Requirements and Decontamination Feasibility

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

Decontamination Process Screening - (T. F. Demmitt)

A goal of this effort is to establish the decontamination program requirements for the PNL Radioactivity Control Test Loop (previously called Engineering Scale Loop).

Planning efforts to define program objectives and milestones continued. A PERT network was prepared to provide a guideline for scope definition and progress appraisal.

A prerequisite for definitive decontamination screening tests is availability of a supply of test specimens that have been exposed in sodium systems containing fission products and/or activated corrosion products. Capability for preparing such specimens at PNL are limited at the present time. A capsule test system is being designed to provide facilities for exposing coupons in radioactive sodium. Concurrently, efforts are in progress to obtain specimens (preferably piping or components from forced circulation systems) from other AEC contractors. LASL personnel are currently preparing an inventory of materials they have available from LAMPRE. Similar inventories will be requested from other sites, especially FERMI and EBR-II.

Evaluation of Sodium Cleaning and Decontamination Feasibility - (G. R. Bloom)

Sodium cleaning procedures are being evaluated for FFTF fuel assemblies, components and piping systems. Present efforts are to prepare 304 stainless steel specimens which have been exposed in 1000 to 1400°F high purity sodium. Four capsules were charged with specimens, filled with sodium and welded inside a high purity inert-gas glove box. The four sodium capsules have been held at constant temperatures of 1000, 1200, and 1400°F during the past week. The specimens will be discharged after a one month sodium exposure period. The corrosive effects of removing sodium with alcohols, steam, and ammonia will be determined.

Contamination Control Development

(P. L. Koehmstedt, W. F. Brehm, E. A. Kovacevich, T. F. Demmitt, W. K. Winegardner)

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

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

The modifications to the Radioisotope Transport Loop, which were found necessary in the acceptance test, were completed. These modifications included reworking the plugging loop cooling system, installation of new heater bus bars, and providing additional cooling to the cold trap. The loop was refilled with sodium and flow was established. Adequate cooling of the plugging meter allowed successful plugging runs to be achieved in

the low sodium oxide range, below 10 ppm oxygen (equivalent). With additional cooling to the cold trap, proper temperature control of the cold trap was realized.

The loop was operated under the following conditions:

Electrical Power

From primary	28.98 kw
Into heater	28.15 kw

Temperatures

Test section	1385°F
ΔT (system)	453°F
Pump	932°F
Heat exchanger	990°F

Flow	1.5 gpm
------	---------

During operation, the power requirements to operate on these conditions increased. It was found that the mechanical connection of the primary heater (nickel) bus bar to the ground (copper) bus bar had oxidized due to excessive temperature. This increased the electrical contact resistance thereby increasing the power requirements and the joint temperature. The loop was shutdown and the ground bar attachment was cleaned of oxide. Platinum gauze was placed in the joint to allow better electrical contact.

After startup, the flow was noted to be unstable. This condition was noted slightly in the previous runs, however, it was intermittent. On November 13 a sodium leak developed in the primary pump flow duct. The cause of the duct failure is not known at present. Replacement of the pump duct will take 2-4 weeks.

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

1. Geometry of the primary loop will be revised and rescaled from a 1/144 scale loop to a 1/250 scale loop, reflecting changes in the nominal maximum flow specified for FFTF closed loops.

2. The loop piping will be rearranged so that piping and components are, to the greatest extent possible, on a common plane, to facilitate scanning.
3. The secondary system piping will be rearranged to facilitate equipment access, drainage, and decontamination.

Application of loop scaling factors to calculation of the primary loop piping geometry is currently being carried out.

Irradiated Stainless Steel Corrosion Tab - (E. A. Kovacevich, W. F. Brehm)

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

Thermal Convection Loop Radioisotope Transport - (W. F. Brehm)

The thermal convection loop test, without radioactivity, was completed after a successful 500-hour run at 1350 F T_{max} and 350 F ΔT .

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

- A ferritic layer was probably formed in the grain boundary regions of the hot leg, for a distance of about 1 mil from the sodium-stainless steel interface.
- No gross intergranular penetration was observed; no intergranular effects were observed in specimens decontaminated in alcohol, then pickled.
- Carbide precipitation was observed in hot-leg specimens but not in cold (800° to 1000°F) leg specimens.
- No evidence of mass transfer crystals were seen in cold-leg or dead-leg specimens.

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

Samples of irradiated 304 stainless steel fuel cladding are to be run in thermal convection loops where many of the parameters affecting deposition can be studied. The irradiated stainless steel specimens will be ready soon. The irradiated thermal convection loop experiments are being designed and built.

Diffusion of Fission Products into Stainless Steel - (W. K. Winegardner)

Type 304 stainless steel capsules containing mixed fission products and sodium are being run at 1200°F to measure the depth of diffusion of the fission products. The first capsule was opened after 1052 hours at $1190 \pm 20^\circ\text{F}$. After sodium removal it was found that there were about 43,500 counts/min. of activity tightly bound to the surface. The samples were electropolished to remove increments of surface and recounted. Preliminary results indicate 99% of the activity had diffused less than 1 mil.

Closed Loop Contamination Control System - (P. L. Koehmstedt, W. F. Brehm, E. A. Kovacevich)

The results of this study will affect the FFTF closed loop design, particularly in the ability to recover from tests involving defect, failed, or vented fuels. Recovery is dependent upon control and decontamination techniques; an intolerable portion of the fission products are expected to "plate out" upon piping surfaces. Requirements for the Radioactivity Control Test Loop (formerly called Engineering Scale Loop) are being developed based on the requirements for operation of FFTF closed loops under hot cell remote conditions. The fission product control program plan requirements are presently being formulated and are scheduled for completion by the end of the month.

A small sodium loop needed for fission product screening studies is being designed. This loop will provide a means of introducing fission products, screening the various techniques of trapping, equilibrium control, etc., with prescribed impurities and controlled test parameters.

Calculations have been made to scale down the present design of the FFTF closed loop to an engineering scale model loop. Velocities, stainless steel surface areas, sodium flow to stainless steel area ratios, and Reynolds numbers have been calculated for the various sections of the FFTF closed loop. From these calculations, it is possible to model a scaled-down loop.

THERMAL HYDRAULICS

G. M. Hesson

Heat Transfer Characteristics of FFTF Fuel Assemblies

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

The program to determine the heat transfer characteristics of FFTF fuel assemblies requires heated models of FFTF fuel assemblies. In the three principal phases, fuel element models of different maximum capabilities will be required. These are:

	<u>Q/A (Btu/hr-sq ft)</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.

In addition to the tantalum and graphite heaters being developed under this program, heaters with central elements having a zero or very low temperature coefficient of electrical resistivity are being developed for the local heat generation and clad temperature hot spot tests. For this reason, heaters with central elements of Hastalloy C and Kanthal Alloy A-1 were fabricated and tested.

The Hastalloy heater failed prematurely due to an electrical fault, near the heater rod to electrode connection, at a very low heat flux. The Kanthal heater, on the other hand, attained a heat flux of 1,100,000 Btu/hr-sq ft (water temperature 300°F) and held it for almost three hours. At this point, a cooling line in the power supply ruptured and safety circuits immediately shut off all power. The unusual thermal shock caused by this immediate cessation of power caused the test section components to part.

Fabrication of six tantalum coaxial heaters has been started at PNL under the quality control program. Specific improvements in the details of fabrication include:

- Procurement of a larger size range of boron nitride particles to provide mixes which should increase the packing densities in the ceramic layer during the vibrational compaction step.
- Procurement of higher purity arc cast tantalum and molybdenum (electrode) in place of the powder metallurgy rods previously used.
- Procurement of tantalum rods with tighter limits on straightness and diameter variation. The new rods should give more uniform axial heat generation and better assembly concentricity.
- Nondestructive testing of the stainless steel sheath material to assure failures will not be caused by defective sheaths.
- Baking of the final heater assemblies in a vacuum furnace at 450°F for 15 hours to remove gas or water vapor from the ceramic layer.

Two graphite coaxial heaters are also being fabricated. These heaters are similar to the test section which attained a heat flux of 1,400,000 Btu/hr-sq ft in water tests. The only major change is in the electrode-heater-rod connection. The previous heater used a male-female mechanical joint while these heaters will employ threaded connectors.

Design studies on several alternate heater assemblies with a single ended electrical design and a tubular heater element (to reduce electrode heating) have been started. Also, design work has been started on the flow housing and electrical lead-in assembly required to test single heated pins in the Small Heat Transfer Facility.

In addition to the heated pin development work, visits were made to three sites, General Electric Co. (APED and APO), Atomics International, and Eitel McCullough, a division of Varian Associates, to review heater development programs at each of these sites from the standpoint of present and future FFTF thermal hydraulic testing requirements.

Gas Cooling of FFTF Fuel Assemblies During Inspection

(V. R. Hill, J. M. Yatabe)

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

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

Design of the heat transfer facility in the 185D Building is complete. Design of the electrically heated, simulated fuel assembly is complete and fabrication is 50 percent complete. The heated section of the fuel assembly will consist of 169 Inconel pins 1/4 inch OD by 0.055 inch wall by 36 inches long, capable of producing 40 kw total.

EQUIPMENT DEVELOPMENT

J. R. Carrell

DEVELOPMENT FACILITIES

K. G. Toyoda

Hydraulic Core Mockup

(K. D. Hayden, C. H. Henager)

The objective of this effort is to provide an adequate core mockup. The hydraulic core mockup will aid in determination of the system hydraulic and mechanical characteristics of the FTR. In addition, it will provide support for the development and performance verification of many reactor components.

Site preparation activities, authorized to date, are approximately 90% complete. All tanks and equipment have been removed from the tank farm areas, and the demolition forces are scheduled to immediately move into the canyon area for removal of remaining cell walls and excess piping.

The initial authorized building work, site preparation and decontamination, has progressed to a point where additional design work is required before this phase can be completed. The present activities, that can be completed without design, are scheduled for completion before the end of November.

Hydraulic Core Mockup Test Program

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

This study will quantitatively define the hydraulic core mockup program. The next objective is operation of the mockups to provide information in support of the FFTF core and vessel design and fabrication.

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

The purpose of the annular feature model is to assist in evaluating the mixing characteristics of the inlet streams within the annulus formed by the main reactor vessel and the removable core vessel. This is a tenth size hydraulic feature mockup of the FFTF "Concept IV" coolant supply arrangement.

The test plan for this experiment has been written and the model design is complete. Fabrication has begun and is scheduled for completion by mid-December. The model will be located in the 314 Building and will utilize the existing four hundred gpm hydraulic system.

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

The purpose of the inlet plenum model is to provide information required for an initial refinement of the configuration of the inlet arrangement. This is a one-third size hydraulic feature mockup of the FFTF "Concept III" coolant supply arrangement including downcomers, torus, high pressure supply plenum, and the inlet portions of the reactor tubes.

During the past several weeks, work on the mockup has temporarily slackened to permit maximum effort to be placed on the annular inlet feature model. Detail design has now been resumed and completion of the presently scoped design work is scheduled for the end of November.

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

Application of the hydraulic balance feature to the FFTF fuel shroud ducts is being investigated. This feature consists of a tube arrangement whereby the coolant flow enters and leaves the tube through side entry and exit holes. The lower end of the fuel tube is closed. This arrangement results in essentially balanced axial forces on the fuel duct, and thereby reduces the mechanical holddown device requirements. A single-tube full-scale model has been fabricated to evaluate the hydraulic balance concept and to measure the leakage rates between the duct and the upper and lower tubesheets.

This month additional tests have been conducted in which tube to tube-sheet clearance has been altered. To date, testing has been completed with diametral clearances of 0.005, 0.011, 0.017, 0.025, and 0.040 inches. Results show that the tubesheet leakage is proportional to the differential pressure across the tubesheets and that the axial force is a downward force proportional to the flow. Leakage rates for the upper and lower tubesheet are shown in Figures 1 and 2. These data were obtained with a four-inch thick upper tubesheet and a five-inch thick lower tubesheet.

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

FIGURE 1. Single Tube Hydraulic Balance

FIGURE 2. Single Tube Hydraulic Balance

Computer Program for Tubesheet Drilling - (C. L. Wheeler)

During the design of the tubesheet for the inlet plenum feature model, the coordinates and angles of the tubesheet penetrations were required. The FFTF Core Design Unit uses a drawing procedure to locate these penetrations, which consists of swinging a series of arcs to establish the triangular pitch of the control rod positions and then a second series of arcs to locate the centerlines of the shroud positions around the control rods. This graphical procedure is adequate for drafting purposes but is not well suited for actual shop layout work, which generally employs cartesian coordinates. In addition, this system does not define the angle of inclination of each hole. Hand calculation of the coordinates and angle of each of the $150 \pm$ holes is a formidable task. Accordingly, a program was written and processed through the onsite computer which gave the abscissa, ordinate, and angle of each hole in the tubesheet. It is applicable to the prototype and models of any scale.

Sodium Facilities

(K. D. Hayden)

Test facilities are being prepared to provide a centralized laboratory for liquid sodium studies.

Scope engineering for preparation of design criteria, cost estimate, and project proposal for the proposed FFTF Sodium Facilities Building was initiated. The building plan presently developed is for a steel pre-fabrication construction approximately 100 ft. x 70 ft. x 24 ft. high.

Sodium Studies Facility

(J. D. Berg, L. J. Defferding, M. O. Rau, M. E. Shockley, J. D. Smith)

This facility provides the capability to conduct the initial experimental programs in support of sodium and cover gas chemistry, mass transport, radioisotope deposition, and component evaluation.

Radioisotope Loop

The Radioisotope Loop was preheated, filled and flow established through primary and secondary circuits. Initial observations of system operating characteristics verified a need for additional modifications to correct deficiencies on plugging meter cooler, coldtrap blower, main heater and miscellaneous instrumentation. These repairs were completed and the system

flow and temperature were increased to required test conditions (1.5 gpm, 1300°F, 400°F ΔT). A sodium leak was observed at the stem of the inlet valve to the plugging meter loop. Approximately two grams of sodium dripped from the valve before it was closed and the system frozen off. The valve was replaced.

The loop was again shut down November 13 because of a sodium leak in the primary pump throat. It was necessary to remove the secondary pump to get at the primary pump. It was noted at this time that the secondary pump throat also had a small leak. Investigation of the cause of failure is continuing.

Small Component Test Loop (314 Sodium Loop)

The Small Components Loop was shut down and drained after 3600 hours of operation at maximum test conditions (1060°F, 18 gpm). The fuel test specimen was extracted, cleaned, and delivered to test sponsors for disassembly and examination.

PNL Heat Transfer Loop #1

This loop is being fabricated by MSA Research Corporation. The loop is designed for 15 gpm with 650,000 Btu/hr, heat dump capacity. The piping fabrication and trace heater installation have been completed.

324 Facilities

In the event of a major incident (such as a broken water main or a large fire) requiring the evacuation of the Sodium Studies Facility in the 324 Building Basement, it is necessary to deactivate all sodium loops in the facility. To meet this requirement, the design of a master scram circuit with scram buttons at each exit and a high water level trip have been completed. The circuit is designed to place the sodium equipment in the most safe condition.

A test was conducted to determine the compatibility of molten sodium with high temperature polyimide film. No chemical reaction was observed up to, and including, burning of sodium. This material will be used for personnel protection if additional testing is successful.

Design of the Sodium Studies Facility liquid nitrogen and low pressure nitrogen system has been completed.

EQUIPMENT PROTOTYPES

C. A. Munro

Reactor Nuclear Control Mechanisms

(C. A. Munro)

The nuclear control mechanism development program is subdivided into 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.

Decelerator Components Tests - (R. O. Diethammer, J. R. McBride, R. G. Sullivan)

The deceleration test facility consists of a test section containing a stationary deceleration section (dash pot) into which a simulated safety rod is dropped. The test section is mounted on a framework which permits vertical and non-vertical orientations. Springs installed in the upper portion of the test section impart an initial accelerating force to the simulated rod. The test system is connected to sanitary water service, as well as the EDED-1 loop, permitting water at various temperatures and pressures to be directed through the test section, to simulate liquid sodium flow.

Previously, several modifications were made to the decelerator piston and dummy rod assembly to increase the assembly's maximum velocity. Three additional modifications were completed and tested during the month.

These were:

1. The piston diameter, except the guide sections, was reduced from 3.55 inches to 3.15 inches, increasing the maximum velocity from 7 fps to 8.75 fps. Total weight of the ram, piston, and ballast is now 162 pounds.
2. Acceleration springs, with a spring constant of 167 lb/in., were installed, replacing the 88 lb/in. springs. Maximum velocity increased from 8.75 fps to >10 fps.
3. Clearances in the release mechanism were increased to improve its operation.

With the achievement of a maximum velocity of 10 fps, emphasis was directed toward optimization of the decelerator assembly. To increase the deceleration capability of the assembly, the clearances between the step type ram and dash pot were decreased. Annular clearances tested are as follows:

<u>Step</u>	<u>Clearance Ram #2 (in.)</u>	<u>Clearance Ram #3 (in.)</u>
1	0.070	0.060
2	0.050	0.030
3	0.030	0.010

The velocity profiles for these two rams are shown in Figure 3. It may be noted that the ram velocity is significantly lower when the Ram #3 "bottoms out." The effective length of the dash pot is 6 inches.

Another ram will be built to further reduce clearance for increased damping capacity. Clearances for Ram #4 will be:

<u>Step</u>	<u>Clearance (in.)</u>
1	0.040
2	0.030
3	0.010

Several modifications are to be made to the test stand in preparation for high temperature water test runs. These include:

1. Add a packing type seal to the upper sleeve guide (Item 5, SK-3-13140) to stop leakage which could be dangerous when making high temperature water tests.
2. Add an alignment ring to the dash pot to facilitate alignment of the dash pot and tube at reassembly.

Conceptual Studies - (J. C. Noakes, W. H. Davenport)

The conceptual design of a regulating (non-scramble) rod drive unit was completed and is shown in SK-3-12435. The control rod drive mechanism extends no higher than the adjacent driver fuel subassembly connector, thereby providing an open area for the refueling machine operations. This design

FIGURE 3. Ram Velocity Versus Distance from Bottom of Dashpot

features a stepping drive motor which also provides both clutch and static brake functions. The rotary to linear translator mechanism is located below the sodium level; therefore, a small, simple rotary sodium vapor seal (similar to a free surface sodium pump seal) is located near the drive unit. A disconnect is provided below the linear translator to facilitate replacement of the spent control rod and radioactive extension rod. A Hanford type remote connector was incorporated to seal the nozzle opening.

Reactor Hardware

(J. W. Kolb)

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 drawing of the open loop nozzle cap and quick disconnect 10-inch lattice spacing is shown on SK-3-12955, Sheet 2. This drawing details spacing requirements for latching the duct closures, the instrument leadout hardware, and the sodium vapor seal expansion, in the duct closure. Additional studies are in progress to relocate the subassembly expansion point lower in the duct, to give more room at the top face. Layout drawings are being prepared for driver fuel, sodium sample, rupture monitor lines.

Closed Loop Nozzle Spool Piece

(R. Kolowith, J. W. Kolb)

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

A vendor survey of seals is being prepared. Two drawings, "Seal Test Set," and "Clamp Detail Assembly," showing FFTF requirements for sodium sealing, and potential clamping methods, have been completed for the vendor survey. One method of mechanically sealing closed loops is shown in "Closed Loop Top Face Hardware and Disconnect Arrangement," SK-3-12960.

"Welded closures" development on closed loops has continued and a test mockup is being fabricated to test the design shown in SK-3-12956. A tooling concept has been developed for cutting out the weld to allow remote replacement of test samples or loop duct flanges. Both external cutter and internal cutter concepts appear feasible for welded positive sealing of closed loops. Work is now underway on interconnect piping for influent and effluent sodium coolant.

SECTION "A-A"
FULL SCALE

SECTION "B-B"
FULL SCALE

NOTE: ELECTRICAL LEADOUT NOT SHOWN—SIMILAR TO DRIVER SUBASSEMBLY SK-3-12577

CONCEPTUAL

1ST ISSUE 11-7-67

U. S. ATOMIC ENERGY COMMISSION	
RICHLAND OPERATIONS OFFICE	
PACIFIC NORTHWEST LABORATORY	
OPERATED BY BATTTELLE MEMORIAL INSTITUTE	
REGULATING ROD DRIVE	
IN SODIUM BALL NUT	
REMOTE DISCONNECT	
FAST FLUX TEST FACILITY	
300 GEN	1901.01
SK-3-12435	

REV	DATE	DESCRIPTION
1	11-7-67	CONCEPTUAL

CONCEPTUAL 11-7-67

U. S. ATOMIC ENERGY COMMISSION
RICHLAND OPERATIONS OFFICE
PACIFIC NORTHWEST LABORATORY
OPERATED BY BATTTELLE MEMORIAL INSTITUTE

OPEN LOOP NOZZLE
CAP, QUICK DISCONNECT
TYPE, 10" ARRANGEMENT

FAST FLUX TEST FACILITY
PLANT NO. 300 GEN. 1500.01

SK-3-12955 2 2

REV.	DATE	DESCRIPTION
1	11-7-67	CONCEPTUAL

REV.	DATE	DESCRIPTION
1	11-7-67	CONCEPTUAL

REV.	DATE	DESCRIPTION
1	11-7-67	CONCEPTUAL

Closed Loop System Testing

(R. Kolowith)

A study of the need for systems proof testing and design information on closed loop tubes is being conducted. A cost estimate for fabrication of a complete prototype mockup closed loop system was prepared by Hanford Engineering Services. The estimate includes the funds for the required auxiliary services, engineering, and development to support the prototype system.

INSTRUMENT DEVELOPMENT

M. R. Wood

Reactor Environmental Effects on Signal Cables

(R. R. Bourassa, G. J. Dau, J. L. Stringer, D. C. Thompson)

This study 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 instantaneous radiation or dose rate effects on thermocouple calibration.

Six coaxial cables received October 10, were X-rayed at each end and in the middle to confirm dimensional uniformity. Two of these cables were selected for an experiment in the ^{60}Co gamma-ray facility at PNL (2.35×10^7 R/hr maximum). This experiment was carried out to confirm the feasibility of the measurement technique for induced current per unit cable length. The two cables are designated 4D and 4E. Both of the cables have stainless steel sheaths and center conductors, and use powdered MgO as the electrical insulator. Cable 4E has a 250 mil O.D. with a 19 mil sheath thickness and a 25 mil diameter center conductor. Cable 4D has a 187 mil O.D. and the same sheath thickness and center conductor diameter as cable 4E. This, the cables differ only in O.D. and thickness of the insulator.

The sealed end of each cable was lowered fully into the radiation field, and then was withdrawn in increments of three inches. After each movement of the cable, the total induced current was measured. A graph of the total induced current versus the distance from the bottom of the gamma facility is shown in Figure 4 for both cables. The gamma ray intensity profile, previously measured with an ion chamber, is shown for reference.

The induced current per unit length is shown in Figure 5. The current has been normalized at the maximum field intensity of the facility. The ion chamber profile is again shown for reference.

The data show that the measuring technique gives the desired information about the cable since the induced current per unit length is proportional to the profile of the gamma-ray intensity.

FIGURE 4. Cable Current as a Function of Position

FIGURE 5. Normalized Incremental Cable Current as a Function of Position

The bulk conductivity of the powdered MgO insulator was measured at selected radiation intensities in the facility.

The bulk conductivity shown in Figure 6, as a function of distance from the bottom of the facility, follows the gamma-ray intensity profile curve which is included for reference.

A second partial shipment of ten cables was received November 2. This leaves 14 cables outstanding on the initial order of 30 cables.

Fabrication of the apparatus for the experiment in the pulsed TRIGA reactor at Washington State University has been completed, and the experiment is scheduled for November 13 through 15. The radiation induced currents and insulator conductivity for cable 4E will be measured. In addition the instantaneous radiation or dose rate effects for three chromel-alumel and one iron-constantan thermocouples will be measured.

In-Reactor Coolant Temperature Sensors

(D. C. Thompson, J. L. Jackson and 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 environments.

Electrical Tests

Thermal response time constants were measured for the remainder of the thermocouples. Time constants to the 63% point averaged 0.76 second for the 25 insulated-junction thermocouples and 0.17 second for the 29 remaining grounded-junction thermocouples. The range of time constants for the insulated junction thermocouples was from 0.48 to 1.2 seconds and the grounded junction thermocouples had a range of 0.11 to 0.24 second. Time constants for both increasing and decreasing temperature steps were averaged to obtain these figures. The tests consisted of applying temperature step increases of from 800°F to 1000°F and temperature step decreases of from 1200°F to 1000°F and recording the thermocouple transient responses to these steps.

One grounded-junction thermocouple failed by open circuiting during the response time tests. Radiographs failed to locate the break point in the wires. This break point will be difficult to locate because the open-circuit only occurs at temperatures of about 900°F.

FIGURE 6. Insulator Induced Conductivity as a Function of Position

Fifty of the candidate thermocouples are being encapsulated in preparation for the EBR-II irradiations. Four thermocouples, two with grounded junctions and two with insulated junctions, were kept for control units.

Fast-Reactor Irradiation Testing

The pin design for the irradiation of this second group of FTR candidate thermocouples in EBR-II is completed. The core gamma heat estimates are still uncertain and heat-transfer calculations have been made over the available range of estimates to ensure the safe operation of the assemblies. Melt wires will be incorporated into each assembly to provide an independent indication of irradiation temperature. These two pins are still scheduled to be charged late in calendar year 1967.

Pressure, DP Flow and Level Sensors for Sodium Service

(T. J. Crocker, J. R. McBride, K. O. Creek)

The objective of the present effort is to develop pressure and differential pressure flow measuring 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 modifications and improvement, and final proof testing.

Static Sodium Facility

Installation of the facility in the 324 Building was completed. The sodium-containing portions of the facility have been cleaned and sodium was charged into the storage tank. The system has been operated at low temperature (300°F), and sodium has been successfully transferred from the storage tank into a pot and back to the storage tank. A sample was obtained during the charging operation. Analysis verified the oxygen impurity content to be less than 10 ppm. Plans for a preliminary high temperature test are being prepared.

Three electromechanical transducers and one volumetric transducer are being prepared for screening tests in the facility. The readout equipment required for each transducer is being modified and calibrated. Specifications are being prepared for a digital type data acquisition system, which will provide more accurate and less time consuming transducer monitoring than the equipment currently available.

Delayed Neutron Failed Fuel Monitor

(D. P. Brown)

The objective of this effort is to test and evaluate nuclear components of a counting system for use in the delayed neutron failed fuel monitor.

Three 0.25 inch diameter ^{10}B proportional neutron counters, two clusters of these 0.25 inch counters connected in parallel, and three one inch diameter fission counters are being subjected to operating tests in high gamma fields. Initially, tests are being conducted at ambient temperatures with similar testing planned at elevated temperatures.

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

Chromatograph Fuel Failure Detector

(G. B. Barton)

A fuel failure monitoring system based on gas chromatography is being developed for the scanning of many sample points in the FFTF sodium system. Data collected on the state-of-the-art of fuel failure detection by gas chromatographic methods indicate the chromatograph should function very well in this service. The high fission product inventory in the FTR fuel will permit burst fuel detection by measurement of the activity in the cover gas.

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. Properties for specific materials are to be thoroughly characterized in support of the development of a high performance long life driver fuel assembly. In addition 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

(T. K. Bierlein, T. T. Claudson, J. E. Irvin, F. R. Shober)

The objective of this effort is to determine the combined effects of environment and fast reactor irradiation 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.

Studies have been performed to determine if previously reported uniform elongation values for AISI Type 304 stainless steel are correct. It appears that the procedure of identifying the uniform elongation as the strain at maximum load, from the stress-strain curve, is not entirely correct. Metallographic evidence supports the fact that it is possible to have uniform strain beyond the point of maximum load; however, in contrast to what was believed earlier, it is felt that the error involved in determining the uniform elongation is not large.

Transmission electron microscopy is being performed on 304 stainless steel tensile specimens that were irradiated in the ETR at 290°C to a fast fluence of 6.0×10^{21} n/cm² (E > 1 MeV). The microstructure of the specimen tested at 750°C consisted of cavities situated on dislocations and in the matrix. The estimated average cavity size was 50 Å and the density was 9×10^{15} cavities per cm². Assuming equilibrium, the amount of helium

required to completely fill all the cavities is about 500 ppm. The amount of helium from the ^{10}B (n, α) He and the (n, p) reactions was estimated to be about 13 ppm. Therefore, it appears that the cavities are not helium filled but are similar in nature to those observed in 304 stainless steel after irradiation in the EBR-II. The significance of this observation is that it may be possible to relate fast reactor irradiations to thermal reactor irradiations in terms of the damage state present at high test temperatures. No cavities were observed in the specimen tested at room temperature. The as-irradiated damage state consisted of a high density of vacancy-interstitial clusters normally present after irradiation at low temperature in a thermal reactor.

Materials examined in support of FFTF and LMFBR Programs are "bulk" samples cut from mechanical test specimens. The specimen holder in the Philips 200 Electron Microscope allows the specimens to tilt around one axis only. By providing another axis of rotation in the specimen holder, the sample can be tilted into many positions to provide better electron diffraction and dark field analysis. A double-tilt specimen holder which will accommodate "bulk" samples in the Philips 200 Electron Microscope, therefore, has been designed and is being built.

Two papers were presented at the winter meeting of the American Nuclear Society. In summary the first paper⁽¹⁾ states:

The tensile properties of AISI-304 stainless steel were determined after irradiation in the EBR-II to 1.7×10^{22} n/cm² total neutron fluence at $1000^\circ\text{F} \pm 100^\circ\text{F}$. Tensile tests were performed in vacuum at a strain rate of 0.002/min at temperatures from 70 to 1600°F. The steel composition was 18.4% Cr, 9.7% Ni, 0.050% C, 0.88% Mn, 0.018% P, 0.010% S, 0.18% Mo, and 14 ppm B.

Post-irradiation testing revealed a marked increase in 0.2% offset yield strength at all test temperatures below 1500°F. Below 1300°F, the increase was a factor of 2 to 3; at higher

(1) J. J. Holmes and J. E. Irvin. "Effect of Fast-Reactor Irradiation on the Tensile Properties of Type 304 Stainless Steel," Transactions, Vol. 10, No. 2, P. 487-8, American Nuclear Society, 1967.

temperatures, the change in strength decreases with increasing test temperature and approaches unirradiated values of 1600°F. No significant changes in ultimate tensile strength was observed, except at 70°F where an increase occurred.

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

The present data are of special importance as they demonstrate that irradiation hardening can occur in metals irradiated to high exposures at one-half the absolute melting temperature ($T_m/2$) and the hardening is stable to $0.67 T_m$. This is in marked contrast to low-exposure (10^{21} n/cm²) results from thermal reactors, in which hardening is observed only at test or irradiation temperatures below $T_m/2$. These results also indicate that displacement damage, which leads to radiation hardening, must be considered as a source of high-temperature ductility loss in addition to the helium embrittlement problem.

The second paper ⁽¹⁾ may be summarized as follows:

The effect of neutron irradiation on the tensile properties of annealed and 25% cold-worked AISI Type -304 stainless steel has been determined over the test temperature range 23 to 750°C. Stainless steel tensile specimens in the two conditions were irradiated at 290°C in the Engineering Test Reactor (ETR) hot-water loop to fluence levels of 1.5×10^{20} , 1.45×10^{21} , and 6.05×10^{21} n/cm² ($E > 1$ MeV), and tested in vacuum at a strain rate of 0.0024/min.

(1) R. E. Robbins, J. J. Holmes and J. E. Irvin, "Post-Irradiation Tensile Properties of Annealed and Cold-Worked AISI-304 Stainless Steel," Transactions, Vol. 10, No. 2, P. 488-9, American Nuclear Society, 1967.

Irradiation increased both the yield and ultimate tensile strengths of the annealed and cold-worked 304 stainless steel, the increase being more pronounced for the annealed material. After exposure to 1.5×10^{20} n/cm², the annealed material has more uniform elongation than the cold-worked steel at all test temperatures investigated. However, the ductility of the annealed material is better only at test temperatures above 400°C after the intermediate exposure of 1.45×10^{21} n/cm². This partial recovery of high-temperature ductility of the annealed 304 does not occur at the highest exposure. Although the cold-worked ductility is greater at temperatures below 500°C, the ductility above 500°C is identical for the two material conditions.

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

In-Reactor Flowing Sodium Experiments

This program is designed to provide high temperature irradiation of candidate FFTF cladding materials in a flowing sodium environment. The capsules will provide information on neutron damage and flowing sodium-metal interaction, important to corrosion and mass transfer studies. Liquid sodium temperatures monitored and controllable are normally regulated to a nominal 1000°F (538°C) at the capsule bottom and 1400°F (760°C) at the capsule top, giving a range of irradiation temperatures for the specimens.

Previous capsules have irradiated bulk specimens. Last month's progress report⁽¹⁾ discussed the need to study a high surface to volume ratio specimen such as a thin walled tube. Mention was made of two future capsules, GEH 22-6 and 22-7. An appraisal of the current design for an in-reactor,

(1) F. W. Woodfield, "Fast Flux Test Facility Development Monthly Progress Report," BNWL-615. Pacific Northwest Laboratory, Richland, Washington, October, 1967.

liquid-sodium biaxial stress-rupture capsule resulted in a design modification which combines the two experiments. GEH 22-6 will simultaneously irradiate both stressed and unstressed 304 stainless steel tubular samples of FFTF cladding geometry. This irradiation will not only provide in-reactor biaxial stress-rupture data for 304 stainless steel but will also furnish, for post-irradiation tests, stress-rupture specimens irradiated under essentially the same conditions. The effects of stress, during irradiation, on rupture life can be studied by this double irradiation.

Realizing the importance of relating sodium chemistry to weight-gain data, Radiometallurgy Operation has been requested to explore ways of obtaining representative sodium samples from these capsules for analysis during disassembly. Pre- and post-irradiation analysis of the sodium could then be provided.

Test Facilities

A facility with capabilities for testing irradiated specimens in sodium and controlled gas atmosphere is under construction.

During the past month, five creep tests in air environment were completed. Four of these were material acceptance tests. One test in static sodium was completed. One test capsule was filled with high purity sodium from the Sodium Metering Station.

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.

Vendor Development - (R. C. Aungst)

Five potential tube fabricators were visited to assess their capabilities for producing the fuel ducts and closed loop tubes for FFTF. Three were large producers of steel tubing and were approached on the basis of producing

round tube blanks to close tolerances from which the fuel ducts and closed loops could be fabricated. The three companies were: Allegheny Ludlum Steel Corporation, United States Steel Corporation, and Babcock and Wilcox. All agreed that meeting the required dimensional tolerances after annealing would be extremely difficult, particularly in lengths required for a weld-free structure. In spite of the difficulties, all are interested in making an effort.

Nuclear Components, Inc., (formerly Dresser Products, Inc.) in Great Barrington, Mass., has well established skills and experience in fabricating components for nuclear and aerospace applications. Their capabilities lie mainly in the areas of machining, forming of sheet and plate, and welding. They could do the entire fabrication and assembly job for both fuel ducts and closed loops if welding were freely allowed. They were particularly interested in the assembly of the closed loop components. Parsons Corporation in Traverse City, Michigan, is a producer of precision tubular parts, primarily for the aerospace industry. Their basic forming method is a proprietary process called "Flow-Turn", which involves rolling a tubular part down on a precision mandrel. They work to diameter and wall thickness tolerances equivalent to about one-half the ± 0.005 in. asked for on the FFTF tubes. They are also certain that they could tool up to form the straight and tapered hexagonal sections required for the fuel ducts.

It is anticipated that a vendor information meeting will be held at PNL early in 1968 to present firm design and specification information to all potential fabricators of fuel ducts, closed loops, or sub-components thereof.

Offsite Fuel Duct Procurement - (R. C. Aungst)

Carpenter Steel Company shipped one of the Core Components Test Loop (CCTL) tubes with an out of dimension transition region on November 15. The balance of the order produced fully to specification was shipped a week later.

Onsite Fuel Duct Fabrication Development - (R. S. Kemper)

Five welded cylindrical tubes have been received from the Carpenter Steel Company as starting blanks for CCTL tube fabrication. One of these blanks has been drawn using hexagonal forming dies.

The first tube was 10' 3" long and its outer diameter varied from 5.120" to 5.139" which fell within the -0 to +20 mil allowable variation. The wall thickness was also within the ± 5 mil tolerance.

At the start of the forming operation a drawbar was welded onto one end of the tube and it was filled with a fine dry sand (vapor blasting grit-- 200 mesh). A 3/4 in. thick steel disc was welded over the end to seal off the tube.

The tube was then drawn, for a distance of 72 inches, through a series of three hexagonal forming dies of the following flat to flat dimensions: 4.870 inch, 4.620 inch, and 4.520 inch. The final pass produced a flat to flat dimension that fell within the $\pm$ mil tolerance on the outer surface of the tube. The entrance angle of these drawing dies is 16° included angle. This angle produces an abrupt transition zone (approximately 1 inch long). A die with an 8° 30' approach angle was fabricated and used to lengthen the transition. This die produced a zone slightly greater than five inches in length.

The sand core has been removed from the tube and extensive measurements are being made. Measurements at the open end of the hexagonal section show that the wall has thickened in the corners. This thickening has reduced the corner to corner dimension below the allowable limit of 4.815 inches. Methods of correcting this condition are being studied. An additional tube has been provided with end fittings and sand filled in preparation for forming.

The use of a hardened steel mandrel instead of the sand core should correct this corner condition and insure a proper radius. A mandrel has been designed so that a tapered hexagonal mandrel may be attached to one end and a cylindrical mandrel, incorporating a transition zone, attached to the other end. Procurement of this mandrel is in process.

One seamless steel tube blank has been received from another vendor. This tube does not meet specifications. The wall thickness is 0.150 ± 0.019 in. rather than the requested 0.150 ± 0.005 in. The OD, however, is within specifications and measures 5.120 ± 0.010 in. This tube will be used in some of those tests that do not require rigid dimensional control. Discussion concerning the remaining four tubes of the purchase order is continuing with the vendor.

Water Resistant Coatings for Fuel Ducts - (J. H. Hoage)

Initial contact has been made with Liquid Metal Engineering Center, operated by Atomics International, to test candidate material combinations in liquid sodium environment. Standard test specimens will be coated by selected methods such as flame or plasma spraying and sent to AI for evaluation.

Also to be evaluated are the thickness limits for a satisfactory coating and the effect of application method on the dimensional stability of a full size shroud tube section.

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

Insulation studies on closed loop tubes have involved the review of insulation requirements, basic heat transfer calculations using typical insulation concepts, and the collection and review of information defining available insulation materials. Some information has also been gathered on insulated tube concepts studied by other investigators.

The basic insulation requirements can be found in BNWL-470⁽¹⁾ with supplementary requirements in BNWL-571⁽²⁾. The more applicable information from the literature survey has provided the following:

- Tubes consisting of Zircaloy-2 pressure tubes, Min-K insulation, and a liner tube of Zircaloy of S.A.P.-895 have been subjected to thermal cycling. When the temperature of

(1) E. R. Astley, Progress Report Fast Flux Test Facility Reference Concept, BNWL-470. Pacific Northwest Laboratory, Richland, Washington, August, 1967.

(2) E. R. Astley, Fast Flux Test Facility Monthly Informal Technical Progress Report, BNWL-571, September, 1967.

the stressed tube was varied from 200°F to 700°F, while the liner tube was maintained at 50°-60°F, rippling was observed in each case, in as few as 60 thermal cycles.^(1,2)

- A tube consisting of a stainless steel pressure tube, woven and loose granular silica insulation and a stainless steel liner tube, was subjected to thermal cycling. After a few thermal cycles from room temperature to 189°F, rippling of the tube was observed.⁽³⁾
- A tube with a graphite insulant, which is cooled by the flowing coolant, has been patented.⁽⁴⁾
- A power tube end connection has been patented for use where separate fluids are used as coolant, moderator and insulant.⁽⁵⁾

The closed loop tubes must provide a large test section volume, consequently the wall thicknesses must be kept to a minimum. One method of achieving this would be to coat or attach some insulation like an oxide or elastomer to the wall, however, the known available materials of this type are not suitable. The most feasible insulation concept appears to be a double-walled tube with an insulation gap.

Calculations have shown that a 1/8 in. helium filled gap will provide sufficient insulation to maintain acceptable thermal stresses. These calculations also indicate that thermal radiation is the major

- (1) P. Z. Rosta and G. R. Cambell, Solid Insulant Coolant Tube Development Test on Prototype No. 4 and No. 6., No. R6CAP53, C.A.P.D., Canadian General Electric Company, Limited.
- (2) P. Z. Rosta and G. R. Cambell, Solid Insulant Coolant Tube Development Test on Prototype No. 5, No. R64CAP54, C. A. P.D., Canadian General Electric Company, Limited.
- (3) J. Teurissen and D. Caple, Test Report - Pressure Tube Model with External Lines and Insulation, Hawker Siddeley, Canada.
- (4) Nuclear Power Plant, French Patent No. 1,32,767 (1963).
- (5) Double-Walled Pressure Tube for a Nuclear Reactor, German Patent No. 1,201,927.

mode of heat transfer, which suggests a thermal-radiation insulation material like metallic foil.

NONMETALLIC MATERIALS

D. E. Mahagin

Irradiation Behavior of Shield Material

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

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

Irradiation testing of commercially-available carbon and graphite will be conducted in the ETR. Samples will be irradiated in hot capsules which are scheduled for charging during January and February. The capsule design and heat-transfer calculations have been completed. The materials will be irradiated at temperatures from 200 to 800°F and fluences from 5×10^{19} to 5×10^{20} nvt ($E > 0.18$ MeV). Each capsule will contain flux and temperature monitors. Four of the eight candidate materials, ⁽¹⁾ chosen on the basis of low cost and ready availability, were received and sample fabrication was initiated.

Radiation Effects on Absorbing Materials for FTR Control and Axial Shields

(D. E. Mahagin)

A literature search on irradiation effects in boron carbide has been completed. This material is the prime candidate for use in FTR control rods, and as such, its behavior under irradiation is extremely important. Based upon the results of the literature search, design of an irradiation program, which will cover the expected FTR control rod operating conditions, is presently in progress.

Chemical Compatibility of Shield and Control Rod Materials

(P. G. Pallmer)

The thermal stability of serpentine is currently under investigation in order to establish the material's usefulness as an FTR shield material.

Tests on the long-term dehydration of serpentine continue to demonstrate that the water retention is a function of both temperature and time. The Quebec serpentine being heated at 940°F continues to lose water after a total of 1829 hours. Based on an initial water content of 13%, the measured water losses indicate water retention after this period is 5.8 and 2.0% for the coarse and fine grades, respectively, or 4.2% for a 60/40 mixture.

(1) J. W. Helm, Trip Report Discussions on FTR Shield and Control Rod Materials, BNWL-641, Pacific Northwest Laboratory, Richland, Washington. In Publication.

Because a more detailed description of the kinetics would aid interpretation of the irradiation tests of serpentine, the water loss rate was simulated by numerical integration. The decomposition of the grades of Quebec serpentine were found to follow the Arrhenius rate equation, $-dW/dt = [W] A \exp (-\Delta E/RT)$, where W is the weight percent water retention. Values of the frequency factor, A, and activation energy ΔE , which agreed with both constant temperature and increasing temperature weight-loss measurements, are $A = 1.7 \times 10^{11} \text{ sec}^{-1}$ and $\Delta E = 60 \text{ to } 65 \text{ kcal/mole}$. It should be noted that these constants are valid only for the water given off at temperatures greater than 900°F. In order to obtain them, it was necessary to heat the sample for 100 hr. at 900°F to deplete the water which is given off with lower activation energy. This modification of the experimental procedure eliminated much of the complication of multiple sources of water.

Development of Plastics and Elastomers for FFTF

(J. H. Kleinpeter)

The literature search for useful information on plastics and elastomers is continuing. During this report period, fifty additional references dealing with irradiation effects were reviewed, and summaries of their pertinent data were added to the FFTF Materials Design Data Manual.

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.

Draft summaries of data collections on properties of Types 304 and 316 stainless steel were prepared. The section on 304 contains information on the mechanical properties, some on physical properties and a small amount on fabrication and welding. The section on 316 contains information on mechanical properties. Included are some summary data of irradiation effects on mechanical properties.

The data presented in the draft summaries generally show the property value as a function of test temperature. A smooth curve is drawn through the minimum points with the range at a given temperature indicated. Creep, stress-rupture and tensile data are presented in this manner.

The method of data presentation described above appears even more advantageous when handling short-term tensile data from post-irradiation tests. The scatter is so great that only the point of catastrophic property change can be determined, if it does indeed exist.

The data show the following trends for 304 stainless steel:

1. The minimum creep rate at 1200°F, for irradiated material, falls in the middle of the range of values for unirradiated material, indicating little or no effect.
2. The time for rupture to occur at a given stress level and temperature is decreased by irradiation.
3. Yield strength increases due to irradiation persist up to about 950°F, above 950°F they fall slightly below unirradiated values.
4. Ultimate strength of irradiated material falls below that for unirradiated.
5. Ductility, as measured by total elongation, is less for irradiated material and reaches a minimum value of about 3% between 600 and 1400°F.

The 304 was irradiated over a range of fluences from 10^{18} to 10^{22} nvt and tested at temperatures equal to or greater than the irradiation temperature.

The data collection on plastics and elastomers will be available by the end of December. Preparation of a draft on control rod materials (boron carbide and tantalum) has been initiated and should be issued about January 30, 1968.

Codes, Standards, Design Data

(R. A. Moen)

Design stress curves for unirradiated 304 and 316 stainless steel, based on ASME Section I criteria, were reviewed upon receipt from Babcock

and Wilcox.^(1,2) The following qualifications accompanied the curves:

"The major difference between the Code criteria for 304 and 316 is that the "60% of the average stress to produce rupture in 100,000 hours" criterion for 304 is well above the "80% of the minimum criterion." This difference between the two criteria seems to be the result of the increasing strength of 304 due to increase in residual elements, such as nitrogen due to scrap remelting practices, and the higher rupture strengths now attainable only result in an increase in the average criterion. The ASME Code Committees are sufficiently confident in the reproducibility of the higher strength 304 and so they have recently increased the allowable stresses for 304 at elevated temperatures (compare 1965 Section I Code and Summer 1965 Addenda)."

This information supports a recently reached conclusion that 304 is actually stronger than 316, from 1350°F to 1500°F. This was based on the same data used by B&W. If we further eliminate the "80% of minimum" criterion, which is now believed justifiable, 304 then appears stronger than 316 for applications above 1250°F.

(1) Letter, T. P. Farrell to W. B. McDonald, "Stress vs. Temperature Curve, Type 316 SS" dated October 11, 1967.

(2) Letter, T. P. Farrell to W. B. McDonald, "Design Criteria Curves," dated October 13, 1967.

PROCESS AND MATERIALS SPECIFICATIONS

J. W. Helm

DEVELOPMENT PROGRAMS

J. W. Helm

Development Planning

(J. W. Helm)

PERT Programming for Development - (J. D. Schaffer)

Planning and scheduling of development efforts for FFTF are essential to insure the overall Project needs are met on schedule, the development work is integrated into the PERT network, and that development is proceeding on a timely basis. FFTF development efforts have now been integrated into the Project PERT System via issuance of approved fragnets by each Development section. These fragnets will be included in the package forwarded to RDT for comments.

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 the 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 LMFBP Program.

A work plan and schedule have been made for the period through preliminary design. Interfaces have been identified and are being verified. The program will provide four major contributions to the Project:

- Develop an accurate, energy related, neutron fluence measurement system.
- Calculation of neutron damage in the FTR vessel and critical components, using the most accurate damage models.

- Develop an effective operational surveillance system to monitor the FTR for irradiation damage, corrosion damage, component movements or vibration, any abnormal behavior and general information of benefit to the LMFBF Program. The system development will consider non-destructive testing techniques as well as conventional techniques such as probes, coupons, and visual methods.
- Develop a method of data interpretation, integrating all phases of the program, to extract the maximum benefits toward meeting the objectives.

The major effort at the present is aimed at establishing the requirements of the system i.e., what is measured or monitored and at which locations. Major companies in the nuclear industry will be contacted in an effort to benefit from their present state-of-the-art in surveillance.

Dosimetry for the FTR - (J. L. Jackson)

Neutron dosimetry systems and methods are required for the FFTF to provide detailed information on flux levels, flux distributions throughout the systems, and fluences received by materials and components. A state-of-the-art report recently issued,⁽¹⁾ presents the techniques being developed for accurate measurement of flux and fluence. These techniques include use of activation monitors carefully chosen for each particular application, accurate differential neutron cross-sections for the monitor materials and detailed neutron spectra calculations.

Radiation Damage for FTR Components - (J. L. Jackson)

Calculations of radiation damage to FTR components will be based on calculated neutron spectra. The relative spectral shapes for three fast reactor cores (EBR-II,⁽²⁾ FTR,⁽²⁾ DFR,⁽³⁾), a thermal reactor core spectrum

(1) J. L. Jackson, R. E. Dahl, Measurement of Neutron Flux in the FFTF Using Activation Monitors, BNWL-577, Pacific Northwest Laboratory, Richland, Washington, October, 1967.

(2) Personal communication from W. Little, FFTF Reactor Physics Section, to J. L. Jackson, 1967.

(3) A. J. Perks, J. W. H. Simmons, Preliminary Results of Graphite Irradiations in the DFR, AERE-R-4780.

(ETR) above 100 ev and a Watt fission spectrum are compared in Figure 7. The curves have not been normalized and do not indicate the relative magnitude of the fluxes. The spectral "softening" from the fission spectrum or even the fast part of a thermal reactor spectrum is apparent in all three fast reactor spectra. Comparison of the relative spectral shapes reveal the increase in the fraction of lower energy neutrons in the FTR, as compared to both EBR-II and DFR, emphasizing the necessity of detailed analysis of material damage effects in the three reactors. More defined multigroup, multiregion, multidimensional spectral calculations will be required for these analyses.

Materials Damage Analyses - (G. E. Russcher)

Materials damage analyses are being made so that effects of irradiation on the materials and components in the FFTF can be determined. Revised calculations of the neutron spectrum in the FTR core are now being evaluated and incorporated into the damage analysis and dosimetry calculations. Additional spectrum calculations necessary to develop adequate dosimetry and surveillance programs have been identified. The requirements are:

1. Neutron energy spectra (to within 5%) with a group structure having at least 18 groups above 1 KeV must be determined for the following physical locations.
 - In-core fuel element region.
 - Structural lower support grid.
 - Upper support grid.
 - Pressure-vessel and component surveillance sites.
2. These spectra must be examined to determine if they are sensitive to reactor temperature.
3. The influence of the predicted fuel burnup upon the neutron spectrum for the in-core fuel element region must be determined.

TECHNICAL SPECIFICATIONS

J. W. Helm

FFTF specifications will be reviewed prior to issuance to insure the latest information from the development efforts has been used in their preparation.

FTR - SPLIT CORE CONCEPT
DRIVER FUEL ZONE SPECTRUM

COMPARISON OF CORE SPECTRA IN A LIGHT WATER MODERATED REACTOR AND IN EBR-II
TO THE WATT FISSION SPECTRUM

Sodium Specifications

(J. W. Helm)

A FFTF sodium specification will be prepared to assure that fuel development and testing is conducted with a prototype material. Initial efforts on the preliminary specification are underway. The tentative draft of the LMFBR Sodium Quality Standard (RDT-M-700) is being reviewed to determine how best to incorporate the information into the preliminary specification.

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