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FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
DECEMBER 1969, JANUARY, FEBRUARY 1970

AEC RESEARCH &
DEVELOPMENT REPORT

BNWL-1328

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PACIFIC NORTHWEST LABORATORY

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BATTELLE MEMORIAL INSTITUTE

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Technology

FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
DECEMBER 1969, JANUARY, FEBRUARY 1970

By

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April, 1970

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FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
DECEMBER 1969, JANUARY, FEBRUARY 1970

ABSTRACT

This report was prepared by Battelle-Northwest under Contract No. AT(45-1)-1830 for the Atomic Energy Commission, Division of Reactor Development and Technology, to summarize technical progress made in the Fast Flux Test Facility Program during December 1969 and January and February 1970.

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BNWL-325	FFTF Summary Technical Report	1-1-66 to 12-31-66
BNWL-501	FFTF Quarterly Technical Report	1-1-67 to 3-31-67
BNWL-541	FFTF Quarterly Technical Report	4-1-67 to 6-30-67
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BNWL-1174	FFTF Quarterly Technical Report	6-1-69 to 8-31-69
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INTRODUCTION

Battelle-Northwest, as FFTF Project Manager, is discharging dual responsibilities. One major responsibility is management of the end-product oriented design and construction program to ensure conformance to technical requirements and to established costs and schedules. Progress in this area is reported monthly through the FFTF Monthly Informal Technical Report.

The second major responsibility discharged by BNW is management of the discipline-oriented technological program required for ensuring technical adequacy of the FFTF design. Progress in this highly technical area is the subject of this and other FFTF Quarterly Technical Reports.

An overall summary of progress is given in Chapter I. Progress in discipline-oriented technological work is presented in Chapters II through X. These chapters generally correspond to the LMFB Program Plan Elements. Subdivisions of the chapters, however, are based upon specific FFTF components, systems, facilities, and programs.

Within Chapters II through X, reports on specific topics show in their titles the code of the responsible FFTF organization. These codes are listed in Appendix A.

CHAPTER I. PROGRESS HIGHLIGHTS*COMPONENTS

The development of an adequate, idealized digital simulation of the dump heat exchanger has been concluded with the adoption of a counter-current, concentric-tube model using 4-axial and 6-radial nodes.

INSTRUMENTATION AND CONTROL

Adequate low level FFTF neutron flux monitoring will be provided by the combination of a variable-position, pulse-type, in-vessel neutron monitor and a fixed-position, pulse-type, out-of-vessel neutron monitor. This arrangement will compensate for the effect on out-of-vessel monitor readings of neutron emission from irradiated driver fuel stored in the vessel.

It is now planned to use xenon gas tagging in the first FTR core to locate failed fuel. Gas disengagement tests at ANL have been redirected to evaluate the entrainment and solubility of xenon in sodium, and future ANL tests of FFTF in-vessel flowmeters will evaluate the effect of bubbles on the flowmeter signal.

CORE DESIGN

The development of solid-state diffusion methods for attaching Inconel 718 pads to stainless steel ducts is essentially complete. Distortion problems have been nearly eliminated through the combined use of a redesigned internal collet fixture and electron-beam seal welding of the pad edges. Development of methods to attach Stellite 68 wear pads to stainless steel ducts was unsuccessful.

* Compiled by C. P. Cabell (394)

Remote welding of the closed loop nozzle plug to the reactor top face spool piece will be required each time a closed loop tube is replaced in the reactor. A new BNW remote weld head being developed to perform this function has successfully completed a series of six remote seal welds over molten sodium inside an environmental test chamber.

Fabrication of the CCTL Mark II 217 pin full-scale fuel assembly has been completed. Shipment to Argonne National Laboratory for life-testing in the sodium test loop will take place in early March. Prototypic FTR environmental conditions of 1100 to 1150 °F at 525 gpm will be provided in a 9000-hr test to study pin bundle erosion, corrosion, fretting and vibration effects and instrumentation package performance.

ORNL has provided conceptual design for equipment to be used in remote disassembly of irradiated fuel subassemblies, subsequent nondestructive examination of pins, and reassembly of the fuel element.

FUELS AND MATERIALS

Creep-rupture results on specimens from Subassembly X018 show that large reductions (up to a factor of 200) in rupture life of Type 316 SS occur after irradiation to approximately 3×10^{22} n/cm² (total) in the temperature range 1000 to 1100 °F. These losses in rupture life imply that the allowable stresses for highly irradiated components may be significantly lower than for unirradiated components. For example, the stress to achieve a rupture life of 1000 hr in an unirradiated material at 1100 °F is about 32,000 psi, while the stress to reach a similar rupture life after irradiation to 3.5×10^{22} n/cm² (total) is only about 21,000 psi.

FUELS RECYCLE

Tests are in progress to evaluate fuel pins produced by contractors participating in the prequalification program. Evaluation of the initial shipment of 59 pins is nearly complete.

An order from Atomics International for tensile data is being filled by the LMFBF Fuel and Cladding Information Center; 180 stress-versus-strain curves have been prepared from 90 experiments in the file and forwarded to AI. The file for austenitic stainless steel mechanical properties now contains 2447 tests, including 30 tests added this month.

PHYSICS

Experimentation in the ZPR-9, FTR-3 assembly has resumed. Neutron spectrum measurements at the core center are complete. Doppler experiments are scheduled next.

Extensive parametric studies have proved the Fast Test Reactor (FTR) to be very stable. The negative Doppler effect is the dominant mechanism providing the prompt shutdown coefficient.

Comparisons of FTR-2 experimental data and calculated values indicate that the current neutronics design requirements of the FTR reflect approximately $\pm 10\%$ uncertainty in the calculated values of peripheral control rod strengths and approximately a 30% over-calculation of the reactivity worths of centrally located changes in fissile plutonium densities arising from fuel burnup or changes in the compositions of test loops.

The neutronic characteristics of the reference design of the FTR are to be verified in an engineering mockup of the FTR. This mockup is to be constructed in Argonne National Laboratories' ZPR-9 critical facility. Within the limitations

of the square matrix structure of ZPR-9 and the platelet materials inventory, the mockup will be as exact as possible. Detailed experimental planning is presently underway.

SAFETY

A consistent basis for establishing the reactivity ramp rate, effective Doppler coefficient, and appropriate equation of state during fast reactor excursions has been defined. Basically, the method consists of a coupling of the multichannel neutronics-heat transfer computer program, MELT-11, with the two-dimensional disassembly computer program, VENUS.

A more definitive code (CACEO) for calculation of the initial phase of containment transients after a hypothetical core disassembly accident is now operational. This code incorporates the effect of gaseous leakage from high to low pressure areas and thus provides more accurate computations of maximum pressure buildup.

CHAPTER II. PLANT DESIGNA. OVERALL PLANT DESIGN

Principal FFTF concept characteristics and data are under review. No official changes have been made during the reporting period.

TABLE 2.1. Baseline Design Data

<u>General Plant Data</u>	<u>Units</u>	<u>Values</u>
<u>Reactor Systems (31000)</u>		
Core arrangement		vertical
Design life	years	20
Total power, initial	MW _t	400
Reactor coolant		sodium
Reactor bulk inlet temperature, initial	°F	600
ultimate	°F	750
Reactor bulk outlet temperature initial core	°F	850
design maximum and ultimate	°F	1050
Core temperature rise average initial	°F	300
ultimate	°F	350
design maximum	°F	400
Reactor cover gas		argon
<u>Core</u>		
Number of core lattice positions		91
Direction of coolant flow		upward
<u>Driver Fuel</u>		
Cladding material		Type 316 SS
Fuel geometry		hexagonal pin cluster
<u>Reactor Vessel Material</u>		Type 304 SS
<u>Heat Transport System (32000)</u>		
Primary Loops		
- Number		

TABLE 2.1. (contd)

<u>General Plant Data</u>	<u>Units</u>	<u>Values</u>
- Primary Loop Material		Type 304 SS
Primary Pumps		
- Number		3
- Design flow rate	gpm	14,500
- Design pump head	feet	500
- Available net positive suction head	feet	51
- Design temperature	°F	1050
- Speed control		wound rotor motor with liquid rheostat
Intermediate heat exchangers		
- Number		3
- Type		Vertical shell and tube
- LMTD		
Initial	°F	85
- Capacity		
Initial	MW	133
Ultimate	MW	157
<u>Irradiation Testing System (33000)</u>		
<u>Test Facilities</u>		
Closed loops		
- Test section outlet temperature	°F	1400 (bypass flow permitted)
- Test section diameter	in.	2.5-3.0
- Material (in-core tube)		Type 316 SS

B. FFTF SITE, STRUCTURES, CONTAINMENT, AND UTILITIES

1. Electrical Insulation for Helical and Linear Induction Electromagnetic Pumps

R. G. Baumgartel (925)

EM induction-type pumps have been specified in the FFTF design for circulating radioactive sodium. Calculations indicate that these pumps will be exposed to a gamma radiation dose rate of 10^4 R/hr to 10^5 R/hr. Survey of the literature and conversations with EM pump designers and irradiation specialists indicate that a gross gamma dose of $\sim 2 \times 10^{10}$ * rads will only slightly deteriorate the electrical insulating properties of the pump windings.

Fabricators of the helical and linear induction-type pumps use a high temperature electrical insulation on the motor windings that consists of glass thread wrapped around a copper wire and impregnated with a silicone resin binder. The impregnated, insulated wire is wrapped around a metal core to form the stator coil of the EM pumps. Dipped in a silicone resin, the stator coil is baked at a high temperature to polymerize and harden the silicone resin. A flexible, laminated-glass mica-glass tape insulation is used to insulate the stator ground wire.

A survey of the published literature revealed that KAPL performed irradiation testing on the electrical insulating materials for the Submarine Intermediate Reactor Program.

C. Manna, KAPL, reported in Nucleonics June 1954 of irradiation test results on electrical insulation materials that were to be used in the SIR primary-coolant pump. The paper stated that "irradiations were conducted on four kinds

* *The 20-year accumulated radiation dose in a 1×10^5 R/hr gamma radiation field.*

of samples: a facsimile stator, voltage-breakdown samples, abrasive bars and resin films." Mannal summarized the results of these tests as follows: "Tests of voltage breakdown, mechanical properties, and gas evolution of irradiated samples indicate that a silicone-resin impregnated mica and glass tape insulation will be satisfactory up to 10^{10} R if it is not in a sealed container."

Glass and most inorganic materials are quite resistant to damage from gamma radiation. An accumulated gamma dose of $\sim 2 \times 10^{10}$ rads is not expected to change the physical or chemical properties of the glass insulation except to darken its color. The silicone resin binder, which is an organic material, will polymerize and become quite brittle and may break off into small particles or sand. Since the main function of the silicone resin is in the assembly of the motor windings, very little change is expected to occur in the electrical performance of the EM pump after the gamma irradiation to the above accumulated dose.

CHAPTER III. COMPONENTSA. REACTOR COMPONENTS1. Development of Methods for Attaching Wear Pads to Flow Ducts

W. F. Brown (AOO) and R. N. Johnson (962)

a. Summary

The development of wear pad attachment processes is essentially complete for the wear pad design. (Design is a 4 in. wide band of Inconel 718, 0.050 in. thick, wrapped around the duct.) The following conclusions summarize the results of the program:

- Resistance spot welding techniques were developed for attaching Inconel 718 to stainless steel. The techniques were not strictly welding, since no fusion occurred, but were solid-state diffusion bonding. These techniques allowed distortion-free, crack-free, high strength anchoring of the wear pad to the surface of the duct.
- Seal welding of the edges of the pad to the duct by TIG welding caused relatively severe distortions in the first attempts (up to 0.026 in. decrease in internal diameter). Redesign of the internal collet fixture and the use of electron-beam welding reduced this distortion to a maximum change in the diameter of 0.006 in. with the average change being much less.
- Since the duct and the wear pad materials have different thermal expansion coefficients, thermal cycling tests were conducted in an effort to determine the number of cycles to fail the attachment or seal welds. Thermal cycling under increasingly severe conditions for a total of 72 cycles at temperatures from 400 °F to as high as 1400 °F and back to 400 °F was unsuccessful in breaking, cracking, or disturbing in any detectable way any of the

seal welds or spot welds. Maximum heating and cooling rates were achieved by plunging the tube section into a molten salt bath at 1400 °F, stabilizing at temperature, then quenching in an air blast fixture to 400 °F, and repeating. Some slight tube distortions (average less than 0.005 in. change in diameter) accumulated during the thermal cycling, but were judged acceptable, especially considering the severity of the conditions and the fact that maximum wear pad temperature in the reactor should be less than 950 °F for the first core.

- Efforts to attach Stellite 6B wear pads were unsuccessful.

b. Resistance Spot Welding

Figures 3.1 and 3.2 show sections through the center of a conventional spot weld with fusion nugget between aged Inconel 718 and Type 304 SS. Although the tensile-shear strength ranges 5000 to 6000 lb, gross cracking occurred. Figures 3.3 and 3.4 show sections through the center of a solid-state bonded spot weld. Tensile-shear strengths for this sized weld ranges from 4000 to 5000 lb. Cracks were not observed in these type welds.

c. Thermal Cycling Tests

Thermal cycling of test assembly tube No. 61, was continued this quarter. The tube assembly has now been subjected to a total of 72 cycles. The conditions for these tests including last quarter's tests are shown in Table 3.1. After the fourth thermal cycling test, the Inconel 718 wear pad was reduced from 0.060 in. to 0.050 in. thick by grinding.

In order to increase the heating rate, the salt bath was held at 1470 °F during the sixth test run. When the part temperature approached 1200 °F, the part was removed manually and air quenched. The rapid heat buildup rate made it difficult to achieve a uniform maximum temperature. The average recorded

Neg 469-2536A

10X

Etchant: $\text{HCL-H}_2\text{O}_2$ and 10% Oxalic Acid

FIGURE 3.1. Section Through Center of a Conventional Spot Weld with Nugget Between Aged Inconel 718 (0.060 in. thick) and Type 304 SS (0.143 in. thick). Gross cracking consistency occurs with these welds.

Neg 469-2536C

250X

Etchant: $\text{HCL-H}_2\text{O}_2$ and 10% Oxalic Acid

FIGURE 3.2. Enlargement of One Area in Fusion Nugget from Figure 3.1, Showing Typical Cracks.

Neg 469-2540A

10X

FIGURE 3.3. Section Through Center of a Solid-State Bond Spot Weld Between Aged Inconel 718 (0.060 in. thick) and Type 304 SS (0.143 in. thick). Cracking was not observed in these type welds.

Neg 469-2540B

500X

Etchant: $\text{HCL-H}_2\text{O}_2$ and 10% Oxalic Acid

FIGURE 3.4. Photomicrograph of the Solid-State Spot Weld Shown in Figure 3.3. The aged Inconel 718 is shown at top and Type 304 SS is shown at the bottom.

maximum temperatures reached for the wear pad thermocouple was 1268 °F and for the tube thermocouple was 1364 °F. The range of the maximum temperatures recorded was 1130 to 1409 °F for the wear pad thermocouple and 1256 to 1418 °F for the tube thermocouple.

Visual and nondestructive tests were made on the welds and wear pads after each thermal cycling test. Ultrasonic inspection of the spot welds revealed no failures. Radiographic inspection showed no indication of cracking in spot welds or in the fusion welds. Liquid penetrant and visual inspection revealed a small crack 1/32 in. long in the center of the seal weld at a corner between sides 3 and 4 toward the unnumbered end of the tube. Very faint liquid penetrant indications were observed at four corners and in several locations on flat 6 at the toe of the seal weld adjacent to the stainless steel tube. Visual examination at 10X and 30X magnification revealed crack-like cavities. These crack-like cavities are caused by some of the tenacious scale formed during the thermal cycling tests in the salt bath furnace and which was not removed completely during the cleaning. After the final thermal cycling tests and before cleaning, these crack-like cavities were very pronounced and appeared around most of the circumference of the tube adjacent to the weld. Most of these crack-like cavities were removed during cleaning. Although some distortion did occur during the thermal cycling it was relatively small compared to that which occurred during the fusion welding of wear pads.

TABLE 3.1. Wear Pad Thermal Cycling Tests on Tube 61

<u>Test</u>	<u>Number of Cycles</u>	<u>Nominal Temperature Range, °F</u>	<u>Furnace Environment</u>	<u>Heating Furnace Maximum Temperature, °F</u>	<u>Cooling Medium</u>	<u>Heating Rate degrees/min</u>		<u>Cooling Rate degrees/min</u>	
						<u>Pad T/C</u>	<u>Tube T/C</u>	<u>Pad T/C</u>	<u>Tube T/C</u>
1	5	600-900	air	900	still air	38	----	55	----
2	10	500-1000	air	1000	still air	26	----	55	----
3	10	400-1100	air	1100	still air	47	----	47	----
4	10	400-1200	salt bath	1200	air blast	267	----	137	----
5	10	400-1200	salt bath	1200	air blast	300	406	140	156
6	27	400-1300	salt bath	1470	air blast	1730	2000	165	197

B. HEAT TRANSPORT PROCESS TECHNOLOGY

1. Dump Heat Exchanger Analysis

R. N. Madsen (822)

The development of an adequate, idealized digital simulation of the dump heat exchanger has been concluded with the adoption of a counter-current, concentric-tube model using 4-axial and 6-radial nodes. This node arrangement is presented in Figure 3.5. The determination that this is an adequate representation follows an extensive study in which a reference simulation was developed and a study was conducted of idealized simulations. These earlier developments have been reported in previous quarterly reports.

The effect of additional axial nodes for the 6-radial node model has been recently investigated. A comparison of the TAP reference simulation with 4- and 8-axial node, 6-radial node models is presented in Figure 3.6. There is insufficient variation in the results, particularly in the prediction of the more critical sodium exit temperature, to justify the additional complexity of the increased number of nodes. Therefore the 4-node model was adopted for inclusion in the heat transport simulation work.

For some transient investigations this model required an inordinate amount of computer time. The controlling time constant being the air node integrations, and this because of the very small heat storage associated with the air, a method of calculating air node temperatures by means of "pseudo-algebraic" equations instead of the differential equations was investigated. This method calculates the air node temperatures assuming that the air heat storage is zero, thus making this calculation algebraic. However, this arrangement creates algebraic feedback loops which are not allowed by the DYNASYAR code. To bypass this problem, a simple lag

FIGURE 3.5. Four Axial, Six Radial Node Model

FIGURE 3.6. Comparison of the TAP Reference Simulation with 4- and 8-Node Axial Models for a Simulated Loss of Power Incident at $t = 3$ sec. Extended surface fins are explicitly simulated.

differential equation is used with a time constant of 0.1. Hence the use of the term pseudo-algebraic. This method gives results identical to the differential equation in only about half the computer time.

This entire study is currently in preparation for issue as a topical report BNWL-1319.

2. Simulation of Fuel Region for Digital and Hybrid Pipe Rupture Models

C. D. Flowers (822)

The core zone for digital and hybrid pipe rupture reactor models is simulated with separate fuel, clad and coolant regions as shown in Figure 3.7. Fuel temperatures are calculated at four axial and three radial mesh points, while clad and coolant temperatures are calculated at four axial mesh points.

The axial mesh points are equally spaced between the bottom and top of the active fuel region. The radial mesh points in the fuel region, however, may be located in a variety of positions. Two schemes for dividing the fuel region radially have been investigated: (1) equal volume per unit length radial regions, and (2) equal A-radius radial regions. These models have also been compared with a much more complex model of the core, using the MELT-II computer code.*

MELT-II is a coupled neutronics heat transfer code which was designed primarily for investigating fast reactor core behavior under major accident conditions. However, the code can also be used to investigate less severe transients. For this study the code simulated six rows of subassemblies. The fourth row subassembly was simulated as the average core subassembly. The average subassembly fuel pin is divided into 18 axial nodes, with 10 radial nodes (seven radial fuel nodes

* *To be published*

FIGURE 3.7. Reactor Model for Digital and Hybrid Pipe Rupture Simulations

plus the bond, clad and coolant nodes). This compares with four axial mesh points and five radial fuel pin regions (three fuel plus clad and coolant) in the simplified reactor model used in the pipe rupture simulations. Both steady state and transient conditions were studied in the model comparison analysis.

The results of the analysis indicated that the equal A-radius fuel node geometry should be used in pipe rupture simulation models for the following reasons:

- It predicts a higher core average fuel temperature which will provide higher coolant and clad temperatures following the loss of coolant type of transient,
- It better represents nodal average fuel temperatures based on the fuel radial temperature profile,
- It predicts higher transient hot channel core outlet and clad temperatures which is conservative for safety analyses, and
- It provides good transient agreement with the more complex MELT-II core model.

3. Hybrid Simulation of the Reactor and Heat Transport System

R. D. Benham and A. L. Gunby (822)

The new simulation programs have been written and checked out: (1) BIAS, a program used to subtract steady-state or dc values in order to expand the scale on a transient variable; and (2) SIGEN, a routine which produces a predetermined step, ramp, or sinusoid function for input anywhere in the simulation.

The overall hybrid simulation (designated HYSIM-2A) of vessel and process systems (multiple loop) has been fully checked out. The reactor core model consists of 3 axial fuel nodes and 3 axial coolant nodes. System transients have been performed to provide base information for the simulation, and it is now being used to gather information on process control

design. HYSIM-2A now utilizes a shielded board with shielded patch cords. Model differences from the previous version of HYSIM-2A are:

- Three axial nodes in the core,
 - Wound-rotor-motor driven pumps, with level correction,
 - Four axial nodes for both IHX's and DHX's.
4. Vessel Thermal Transients for Scram with Full Flow;
Comparison of Oxide and Carbide Fuel
A. L. Gunby (822)

In order to obtain a more realistic estimate for primary hot leg transients at ultimate conditions, the hybrid simulation HYSIM-2A was used for both the current oxide core and a hypothetical carbide core (based on parameters reported in BNWL-914). A core scram with full flow was examined as a function of outlet plenum mixing.

Results are shown in Figures 3.8 and 3.9. Tube outlet temperature for a carbide core decreases significantly faster than for an oxide core at the same conditions. This effect is due to the relatively high conductivity of the carbide fuel, resulting in a much lower fuel temperature. Because of the lower fuel temperature, there is less stored heat available in the carbide fuel following core shutdown.

Figure 3.9 shows the effect at the vessel outlet. For a value of 700 ft³ mixing volume, the ratio of transient rates for carbide at ultimate conditions of oxide at initial conditions is 1.65. Since this extrapolation factor differs significantly from a previously assumed value of 1.25, we recommend the use of a hypothetical advanced core in simulation of vessel transients for ultimate conditions, as contrasted to the present 1.25 factor based on oxide fuel.

FIGURE 3.8. Tube Outlet Transient Core Scram, Full Flow

FIGURE 3.9. Vessel Outlet Transient Core Scram, Full Flow

5. Heat Transport System Control Study

R. A. Harvey and S. A. Hunt (823)

The simplified dynamic simulation of the heat transport system was compared with the more detailed hybrid simulation prepared by Gunby,* and was found to be suitable for high speed use in control system studies. Time scales of up to 1000 times faster than real time were used to define system dynamic characteristics and trends associated with controller adjustments. A method of determining the adequacy of controller adjustments and controller configurations was developed.

The heat transport system control configuration consisting of the following controllers is presented here to demonstrate the form of the results:

- Controller No. 1 - Controls the DHX sodium outlet temperature by manipulating the air flow.
- Controller No. 2 - Controls the IHX primary sodium outlet temperature by manipulating the set point of Controller No. 1.
- Controller No. 3 - Controls the reactor sodium outlet temperature by manipulating the primary loop sodium flow rate.

The process frequency responses for the three open loop systems are shown in Figure 3.10. The frequency response for the Controller No. 1 system did not include the air flow actuator characteristics.

The control modes for the controllers were selected to be:

- Controller No. 1 - Proportional plus derivative,
- Controller No. 2 - Proportional plus integral,
- a Controller No. 3 - Proportional plus integral.

An integral error-squared function was used as an index of performance (IP); an example of its use is shown in Figure 3.11.

* See Section III. B-3

NO. 1 - Response taken with all control loops open

NO. 2 - Response taken NO. 1 loop closed

NO. 3 - Response taken NO's. 1 and 2 loops closed

FIGURE 3.10. Open Loop Frequency Responses for the Controlled Processes in a Heat Transport Circuit.

FIGURE 3.11. Contours of Constant IP (Index of Performance: for Controller Number 2 Settings with (dashed contours) and Without (solid contours) Controller Number 3 Loop Closed. (The IP is the integral of the error squared for number 2 controller in response to a step increase in set point.)

The IP contours which enclose the optimum controller settings are shown for Controller No. 2 controller change when the No. 3 controller loop is closed. Other things that change optimum controller settings were found to be air temperature, sodium flow rates, power level, the settings of the other controllers, the type of disturbance introduced into the system, and the controller scheme used.

C. OTHER COMPONENT TECHNOLOGY

1. Reactor Deck Development Mockup

B. G. Smith (914), W. Trask, and W. S. Kelly (COO)

The objective of this mockup is to aid the Reactor Plant Designer in finalizing the conceptual design of the area above the reactor vessel cover by providing information about spatial relationships of components and related hardware, instrumentation, and piping.

Fabrication of a full scale three-dimensional mockup of the central cover hardware and shielding was completed as shown in Figure 3.12. This includes instrumentation, electrical and cooling lines to the control rod drives and test loops. Fabrication of a closed loop module and shielded pipe trench is being completed for installation.

The simulated components and shielding for this mockup were constructed of quarter-inch thick foamboard, cut and formed to the desired shape and secured with contact cement. All instrumentation, electrical, cooling lines, and piping is commercial grade plastic pipe and fittings, supported by steel angles.

Neg 700434-1

FIGURE 3.12.

Fabrication Progress of Reactor Deck
Development Mockup

CHAPTER IV. INSTRUMENTATION AND CONTROLSA. FFTF INSTRUMENTATION AND CONTROLS SYSTEMS1. FFTF Neutron Flux Monitoringa. Effect of Stored Fuel on Ex-Vessel Monitor System

E. T. Boulette and C. A. Mansius (813)

Monitoring of the FTR core neutron flux level from shutdown to full power (400 MW) is complicated by the presence of in-vessel stored fuel. At very low power levels (<1 W) a large fraction of the signal to the out-of-vessel monitor will be due to neutron flux from the stored fuel, and accurate core monitoring will be compromised. To circumvent this difficulty, the current design includes in-vessel monitors in the vicinity of the shield-reflector interface. Because of their proximity to the core, these in-vessel monitors saturate at a low reactor power level (10 to 100 W). The problem is to locate these detectors to assure adequate monitoring over the power range from shutdown to full power.

The monitoring capability of ^{235}U fission chambers is highly dependent upon the gamma/neutron flux ratio at the detector location. To determine the monitoring range of a detector of this type, one must know the variation in the gamma and neutron flux with changes in the power level. In addition, the effect of stored fuel must be taken into account.

Figure 4.1 shows the effect of stored fuel on the ex-vessel monitor signal as a function of the neutron source rate in the core for 25, 50, and 100 fuel subassemblies in storage, based on extrapolation of calculations made with 2DBS for concept V-A. In all cases, we assumed that a third of the stored fuel elements were placed in each 120° sector of the FTR. In all three cases (25, 50, and 100 stored fuel assemblies) we assumed that the fuel density in the stored fuel zones was the same. To check extrapolations involved

FIGURE 4.1. Percent of Out-of-Vessel Monitor Signal due to 25, 50, and 100 Stored Fuel Elements Versus Total Core Source Rate.

in these calculations, a two-dimensional diffusion theory calculation was made with 2DBS on concept V-A in which the neutron source rate was fixed at 1.1×10^9 n/sec (corresponding to a subcritical system with $k = 0.9$). Calculations were made with and without stored fuel in the sodium annulus. The results indicated that at full shutdown ($k = 0.9$) about 80 to 85% of the signal at the out-of-vessel monitor is due to stored fuel. This calculation assumed a steady state condition; i.e., the delayed neutron precursors generated at full power have decayed to the extent that the fixed source in the core (spontaneous fission and α, n reactions) predominates. The extrapolation shown in Figure 4.1 indicates that about 70% of the signal will be due to stored fuel. The agreement between these two numbers validates the extrapolation technique used to generate the curves in Figure 4.1.

In addition to these data, one must also know the time variation of the total core source rate following a large perturbation (e.g., a scram to a k of 0.9). In Figure 4.2 is plotted the core neutron source rate versus time after shutdown for instantaneous scrams to $k = 0.9$ and $k = 0.99$. To generate these curves, the following assumptions were made:

- The scrams are instantaneous.
- At full power, 90% of the fission is ^{239}Pu fission and 10% is ^{238}U fission.
- The neutron source rate in the core at any time following shutdown is simply $(1-k)^{-1}$ times the sum of the fixed source in the core (spontaneous fission and (α, n) reactions with oxygen) and the delayed neutron source rate.
- The total delayed neutron source rate, S_D , decays as the sum of exponentials

FIGURE 4.2. Total Core Source Rate Versus Time Following Instantaneous Scrams to $k = 0.9$ and $k = 0.99$

$$S_D(t) = S_o \left\{ 0.9 \left(\sum_i \beta_i^{239} e^{-\lambda_i^{239} t} \right) + 0.1 \left(\sum_i \beta_i^{238} e^{-\lambda_i^{238} t} \right) \right\} \quad (1)$$

where

$$S_o = 3.34 \times 10^{-19} \text{ n/sec (neutron source rate at 400 MW)}$$

$$\beta_i^{239} = i\text{-th delayed neutron fraction for } ^{239}\text{Pu fission}$$

$$\beta_i^{238} = i\text{-th delayed neutron fraction for } ^{238}\text{U fission}$$

$$\lambda_i^{239} = \text{decay constant associated with the } i\text{-th delayed neutron fraction for } ^{239}\text{Pu fission}$$

$$\lambda_i^{238} = \text{decay constant associated with the } i\text{-th delayed neutron fraction for } ^{238}\text{U fission}$$

$$t = \text{time in seconds}$$

Equation (1) does not account for secondary delayed neutrons (i.e., delayed neutrons generated by delayed-neutron-induced fission); however, this effect is believed to be negligible.

Figures 4.1 and 4.2 show at which time following a scram the out-of-vessel monitors become unreliable. The out-of-vessel monitors are considered unreliable when the percent of the out-of-vessel monitor signal due to stored fuel is so great as to mask-out changes in the signal from the core. Also, Figure 4.2, shows the ^{235}U fission rate at the in-vessel detector as a function of time following scram. At full power (core source strength = 3.34×10^{19} n/sec) the ^{235}U fission rate at the detector location is 1.4×10^{15} fissions/sec-g of ^{235}U . Since the ^{235}U fission rate at the detector location is essentially proportional to the core source strength, the ^{235}U fission rate may be obtained for any time following scram from Figure 4.2.

In addition to these data, one must know the gamma intensity at the detector location to assess the detector's

monitoring capability. The total gamma dose rate at the in-vessel detector location is plotted in Figure 4.3 versus time after shutdown. The major contributors to the total dose rate are $^{55}\text{Mn}(n,\gamma)$ and $^{59}\text{Co}(n,\gamma)$ reactions. The fission product decay gammas and the $^{23}\text{Na}(n,\gamma)$ gammas will yield dose rates of the order of 10^4 R/hr and 10^5 R/hr respectively, and therefore will not contribute significantly to the total dose rate for the first few hours following a scram. Note that a buildup time of 5 yr has been assumed for the $^{59}\text{Co}(n,\gamma)$ reaction. Thus, it is only at about 50% saturation.

Employing the data presented in Figures 4.1 to 4.3, one may determine at which time following a scram the ex-vessel monitors become unreliable. In addition, one can determine the required operating characteristics for the in-vessel detectors to assure continuous neutron monitoring. (See paragraph IV A 1 b)

b. Neutron Flux Monitor Range and Overlap Considerations

The individual neutron flux monitor ranges have been reinvestigated. We have determined that it is possible to provide an adequate flux monitoring system utilizing the following monitors:

- A variable-position, pulse-type, low-level, in-vessel neutron monitor.
- A fixed-position, pulse-type, low-level, out-of-vessel neutron monitor.
- A fixed-position, compensated, mean-current, logarithmic, intermediate range monitor.
- A fixed-position, uncompensated, mean-current, linear, high-level monitor.

The scaling calculations are summarized by Figures 4.4 and 4.5 in this report.

If the in-vessel low level channels are scaled as shown in Figures 4.4 and 4.5, three decades can be covered before a

FIGURE 4.3. The Total Gamma Dose Rate (R/hr) Versus Time After Shutdown at 96.0 cm from the Core Centerline on the Core Midplane.

FIGURE 4.4. Neutron Flux Monitoring Low Level Coverage

FIGURE 4.5. Flux Monitoring and Control Power Coverage

switchover to the out-of-vessel, low-level channels is required. The out-of-vessel, low-level channels are scaled to provide an adequate count rate on the low end and to provide proper overlap with the intermediate log channels on the upper end. The intermediate channels overlap the high-level channels for three decades. Figure 4.5 shows the complete power coverage.

The in-vessel channels are scaled such that the detector count rate at 30\$ shutdown is 50 cps. Calculations by Baird* and Uotinen* showed that when using a detector with 1.8 g of ^{235}U , 250 cps can be achieved at 30\$ shutdown. If the system is adjusted for operation at 1×10^6 R/hr (20% sensitivity), the resulting count rate is 50 cps. The power level covered by these channels is 10 mW to approximately 100 W.

The out-of-vessel channels are scaled to give 50 cps at the time the in-vessel channels are counting 10^5 cps (≈ 20 W). If the detector were located midcore,** the calculated count rate at 20 W would be 7×10^3 cps, indicating that an excess of neutrons would allow scaling the channel down to the 50 cps level by physical positioning of the detector. The power level covered by these channels is 1 W to about 500 W.

The worst case environment at which the in-vessel detector must operate is defined by analyzing the scram ($k = 0.9$) case. The out-of-vessel detector may be used down to a power level where 90% of the count rate is caused by core neutrons and 10% by stored fuel. This point is 17 min following a $k = 0.9$ scram with 50 stored fuel assemblies. Figure 4.4 shows that the in-vessel detector is counting below full-scale at about 10 min after a scram. At this time the total gamma at the in-vessel location would be 2.2×10^6 R/hr. The in-vessel detector would be used for the balance of the shutdown and

* *Experimental Physics Section.*

** *A position in the flux monitoring thimble at the midcore plane and 96 cm radially from the core centerline.*

through the refueling cycle. For startup, the in-vessel detector would be used up to a power level of 20 W when the out-of-vessel detector would be counting 50 cps. The out-of-vessel channels would be used to about 10 kW when the intermediate system would be used. The extreme case gamma at the out-of-vessel location is 5×10^3 R/hr due to ^{24}Na immediately following a shutdown. If a compensated ion chamber is used, this would mean that for a 10:1 neutron-to-gamma signal, a flux of 3.4×10^5 nv would be required by the intermediate channel for proper operation. The scaling is based on a neutron flux of 10^{10} nv (^{10}B equivalent) available at the out-of-vessel location at full power. This results in more than a decade overlap between the low-level and intermediate-level (Figures 4.4 and 4.5).

B. INSTRUMENTATION AND CONTROL DEVELOPMENT

1. Analysis of Single Versus Double Differentiation of Neutron Detector Signals

C. N. Jackson, N. C. Hoitink, N. S. Porter, R. C. Weddle, (MOO), and D. C. Thompson (931)

One of the questions consistently addressed to the detection of neutrons with fission counters concerns the relative advantages of double versus single differentiation of the detector signal with charge-sensitive preamplifiers. The experiment described below (designed to compare the effects of using both methods in our detector studies) answers this question. In this experiment we have shown that double differentiation is clearly superior when a low level neutron detector is exposed to gamma dose rates greater than 1×10^4 R/hr. This experiment required the measurement of integral bias curves at the PNL Gamma Facility using a single counting channel for various gamma levels and for different electronic differentiation conditions. Curves taken at each

gamma level (1×10^4 , 1×10^6 and 2×10^6 R/hr) showed the noise, gamma and alpha results and the neutron measurements for constant electronic gains and for constant counting rates at the 10 V bias operating points.

At gamma levels of 1×10^4 , use of double differentiation did not provide significant improvements in neutron sensitivity. At gamma levels of 1×10^6 and 2×10^6 R/hr double differentiation produced gain factors of approximately 3 and 4, respectively. These improvements were made both under conditions of constant electronic gains and constant counting rate conditions. These improvements shown by double differentiation result from two basic principles. With the double differentiation the narrower pulse pile-up becomes a problem. Furthermore, the use of a second differentiation circuit produces a filtering effect which reduces some of the lower frequency noise, hence producing a cleaner signal.

2. Low Level Neutron Flux Instrumentation

C. N. Jackson, N. C. Hoitink, N. S. Porter, N. C. Weddle, (MOO), and D. C. Thompson (931)

a. Thermal "Noise" Induced in Cables and Connectors

Low-level signals from high voltage neutron detectors are frequently obscured by "breakdown pulse noise" when coaxial cables supplying high voltage to the detectors are heated. Cable noise tests were made on a neutron detector wherein ~3 ft of two coaxial cables were heated to 370 to 390 °F for 1200 hr. (One coaxial cable is used for high voltage supply, while the other is used for signal). The two-cable system exhibited no detectable amounts of thermal noise. A single cable tested under similar conditions showed a severe noise problem.

b. Multiple Input Preamplifier

Evaluation tests on the developmental model of a commercial three-input preamplifier were completed. This unit comprises three separate channels of amplification, pulse shaping and discrimination prior to summation of the signals in an output stage. It also features discriminator analysis in each channel for rejection of gamma-caused pileup pulses.

The neutron count rate for three clustered detectors increased approximately three times over that of a single unit throughout a gamma background range from 10^4 to 5×10^5 R/hr. In principle, the multi-input instrument worked well and will probably find legitimate application where several detectors can be used simultaneously (following calibration and discriminator setting procedures) and where higher neutron sensitivity per channel may be needed.

c. BNW Gamma Test Facility Modification

Modifications to the BNW Gamma Test Facility will be complete by the end of March 1970. The fully modified facility will consist of an array of eight (4 in. OD $\times$ 0.065 in. wall) 304 SS thimbles fixed at a gamma flux level of about 1.5×10^6 R/hr and one movable (2.0×10^4 - 2.0×10^6 R/hr) 4 in. OD stainless steel thimble (previously installed). Figure 4.6 shows the modified assembly, where each thimble incorporates a retaining cup for precise positioning of the 80-g Pu/Be neutron source. The three concentric (broken) circles represent small, flat ^{60}Co elements added to increase the source strength to 6.1×10^5 Ci. For the eight fixed tubes, this configuration will place the heated detectors in a gamma flux of about 1.5×10^6 R/hr, and at a temperature of 100 °F to 1100 °F and in a neutron flux of 3.5×10^3 nv as desired.

Typical Test Thimble and
Neutron Source Holder at
Fixed 13-in. Radial Position

FIGURE 4.6. Modified Gamma Test Facility

Although the modification of this gamma test facility is not complete at this time, preliminary detector tests are already under way.

d. Preamplifier Evaluation

Performance measurements of preamplifiers allowed selection of two types for planned routine use in the testing program for low-level counting systems. Both a current-sensitive unit and a charge-sensitive unit were found to exhibit a rise time of about 30 nsec, thereby providing needed response capability for use with the various fission counters.

Table 4.1 provides specific characteristics of the two selected preamplifiers.

TABLE 4.1. Specific Characteristics
of Two Selected Preamplifiers

Model	Conversation Gain	Rise Time, nsec	Noise
(charge-sensitive)	0.25 V/pC	~30	2.6×10^4 equivalent ion pairs RMS with 3000 pF input capacitance
(current-sensitive)	0.02 V/ μ A	~30	0.08 μ A RMS maximum

The integral bias curve of Figure 4.7 illustrates the type of neutron and alpha characteristics relative to system background noise that can be obtained with the current-sensitive amplifier selected.

3. In-Reactor Coolant Temperature Sensors

N. C. Hoitink, N. S. Porter, R. C. Weddle, (MOO),
and D. C. Thompson (931)

This task focuses on the developmental testing and evaluating of thermocouples to provide reliable, commercially available sensors for measuring the coolant temperatures in the FTR thermal and nuclear environment. Pre- and post-irradiation

electrical characterization studies provide information about accuracy, reliability, response time, and other parameters of candidate sensors after exposure in a fast flux reactor (EBR-II) environment.

FIGURE 4.7. Integral Bias Characteristics Taken with Current Sensitive Preamplifier

Further comparisons were made of pre- and post-irradiation performance of thermocouples irradiated in EBR-II Capsules BT-3 and BT-4.

The preliminary conclusion (described in the previous quarterly report)⁽¹⁾ that irradiation did not produce a significant change in time-constant values was verified.

Analysis of the pre- and post-irradiation time-response recordings permitted redetermination of the 0.63% time constant for each sensor. To ensure uniformity for acquisition of data, the pre-irradiation values were redetermined by the same personnel and by the same methods involved in the post-irradiation examinations. Since only insulated-junction thermocouples will be purchased in the future (both for tests and reactor installation), the time-response tests reported here cover only such sensors.

Table 4.2 lists the pre- and post-irradiation time constants for the insulated-junction units. In each case, the listed values represent averages for three-time response measurements. Individual time constants for each recording were determined by a semi log plot of response versus time.

The values listed for the pre- and post-irradiation measurements indicate close agreement in most cases. No certain explanation exists at this time for the disagreement noted for a few of the sensors. However, possible conditions which could account for the discrepancies include limitations on the accuracy of the time-constant measurement (as imposed by a rather large amount of periodic noise on the pre-irradiation time response recordings) or variations in the stirring speed of the salt bath. Measurement accuracies should be about ± 0.10 sec for the pre-irradiation conditions.

1. *Fast Flux Test Facility Quarterly Technical Report, September, October, November 1969, BNWL-1275. Battelle-Northwest, Richland, Washington, January 1970.*

TABLE 4.2 Comparison of Pre- and Post-Irradiation Time Response

Thermocouple Identification	Pre-Irradiation Time Constant-sec		Post-Irradiation Time Constant-sec	
	800 °F to 1000 °F	1200 °F to 1000 °F	800 °F to 1000 °F	1200 °F to 1000 °F
G*	0.52	0.47	0.51	0.46
J	0.67	0.70	0.70	Leads Broke
L	0.46	0.49	0.44	0.41
M	0.68	0.77	0.64	0.66
O	0.47	0.45	0.45	0.40
P	0.67	0.70	0.66	0.53
R	0.55	0.48	0.48	0.44
S	0.97	0.95	0.91	0.93
T	0.46	0.50	0.51	0.43
U*	0.83	0.81	0.85	0.81
W	0.55	0.55	0.51	0.49
Y	0.67	0.60	0.63	0.50

* Denotes nonirradiated control

C. OTHER INSTRUMENT AND CONTROLS TECHNOLOGY

1. Review of Methods for Reactor Vessel Surveillance

K. O. Creek (932)

Four techniques were evaluated for viewing the outer surface of the reactor vessel under reference conditions. Of these techniques (fiber optics, borescope, telescope-mirror, and television), the television is considered the most practical. However, a surveillance space of 3 to 6 in. is needed to accommodate a coolable camera head.

a. Fiber Optics

Partial visual inspection of the vessel surface can be made via a glass fiber optic bundle. This bundle is capable of following a twisting access path into the guard vessel. The glass readily withstands 500 °F. However, 30% of the light striking the bundle is lost due to reflections at the entrance and exit to the fibers, and light is lost at the rate of about 10% per foot in the length of the bundle. A 3 ft long bundle will transmit about 50% of the incident light. Fiber optics are normally used in lengths up to 6 ft with a maximum feasible length of 25 to 30 ft.

The glass fibers are subject to browning in the radiation field requiring frequent replacement of the unit. Additional investigation of radiation tolerance is required.

b. Borescope

Borescopes are, in general, wide-angle, short-focus optical systems for close-up viewing of surfaces, such as the inside walls of piping, which cannot be inspected by line-of-sight techniques. Borescopes, 1 3/4 to 2 in. in diameter and 50 ft long, are available commercially. They can withstand the temperature and radiation. However, a nearly straight line access path is required.

The borescope is too inflexible to follow the curvature under the vessel. A borescope with a 10 ft section of fiber optics attached to the end could be considered to view the bottom of the vessel if access holes for the borescope were obtained.

c. Telescope-to-Mirror System

In this method an image of the surface of the vessel is transmitted to a telescope by a simple mirror attached to a long, movable rod. Access holes in the reactor cover must be provided for the telescope and rod.

This method is not recommended due to severe alignment and vibration problems of the moving mirror and inability to view under the vessel. If straight line access is obtained for the mirror on a rod, then a borescope is a better choice.

d. Television

A radiation-tolerant TV camera head 1 1/2 in. in diameter is commercially available. Attached to a flexible, temperature tolerant access cable and looking into a 45° mirror, the unit could follow a twisting path. It could also view underneath the vessel. The camera head is temperature-limited to about 100 °F because of the photoconductive layer on the vidicon faceplate; therefore, a gas-cooling annulus surrounded by insulation must be added to the camera head diameter. In a study of the cooling problem, Southwest Research Institute recommended at least a 3 in. diameter for the cooled camera.

2. Plant Protection System Reliability Analysis

O. B. Monteith (821)

The evaluation of the PPS to investigate false scrams has been completed. Each instrumentation subsystem has been examined for the components and combinations of their failures which would result in a false scram. Some simplifying assumptions were employed:

- The failure of a single channel to a fail-safe condition will be annunciated and repair instituted.
- The average repair time of a channel that has failed safe is 1 hr.
- The fail-safe failure rate of a channel is 50% of the total channel failure rate.
- Component failures are independent.

Employing nominal component failure rates, the following results were obtained for the various PPS instrumentation subsystems.

<u>Subsystem</u>	<u>False Scrams per hr</u>
Power Level Neutron	0.7×10^{-7}
Power Level Flux Rate-of-change	0.9×10^{-7}
Bulk Outlet Temperature	0.05×10^{-7}
Primary Loop IHX Outlet Temperature	0.2×10^{-7}
Closed Loop Coolant Outlet Temperature	0.2×10^{-7}
Reactor Flux/Flow Ratio	3.0×10^{-7}
Closed Loop Flux/Flow Ratio	3.0×10^{-7}
Reactor Vessel Coolant Level	0.2×10^{-7}
Electrical Power Supply	0.6×10^{-7}
Seismoscope	0.1×10^{-7}
Primary Loop Coolant Flow	0.1×10^{-7}
Containment System	0.01×10^{-7}
Open Loop Outlet Temperature	0.2×10^{-7}
Closed Loop Surge Tank Level	0.2×10^{-7}
Closed Loop IHX Outlet Temperature	0.2×10^{-7}

To estimate the overall PPS false scram rate due to instrumentation failures, we arbitrarily establish three heat transport loops, two open loops, and six closed loops. Based on these total subsystems the PPS false scram rate is 3.1×10^{-6} per hr or about 0.03 per year.

This false scram rate, which is very small considering past reactor experience, is the result of the evaluation of equipment failures only. False scrams are also expected from transients and fluctuations of plant parameters as well as from human-induced conditions, and these former kinds of false scrams will constitute the major portion of the total. No attempt has been made to evaluate these latter type scrams in the above results because of uncertainty of data and a suitable analytic technique does not exist.

Because the analytical results above yield such a very low false scram rate from equipment failures, we conclude that the PPS is well designed in this regard and no further evaluation of false scrams is planned at this time.

3. Triangulation for Fuel Failure Location

J. J. Regimbal (823)

Flow patterns in the outlet plenum of the Hydraulic Core Mockup (HCM) were tested by measuring the transport of small air bubbles from various core ducts to monitoring positions on the 8 in. outlet pipes. The results furnish an adequate basis for planning further tests of the feasibility of locating the origin of fission produce gas releases by synchronously monitoring for bubbles moving past ultrasonic sensors on each outlet pipe.

These tests, and others as well using different tagging material, show that flow to each outlet is generally sectorized. Measurements of the arrival time and amplitude of calibrated gas releases also have demonstrated the repeatability of transport characteristics from one case to another for a given gas release.

One parameter, time of transport of a short burst release, is especially useful for estimating the source position.

In Figure 4.8, times of earliest arrival are related simply to shortest distance of travel (plan view) from a given duct to the outlet position indicated. Such data, augmented with amplitude information at each nozzle demonstrates the potential feasibility of locating types of FTR failures which release some 100 cm^3 of fission gas in a short ($\sim 10 \text{ sec}$) burst.

FIGURE 4.8. Air Bubble Transport Times

CHAPTER V. SODIUM TECHNOLOGY

Progress in the Sodium Technology Program for the FFTF is summarized in the Quarterly Progress Report, Sodium Chemistry Subdepartment, October-December 1969, BNWL-1200-2, Battelle-Northwest, Richland, Washington, February, 1970.

REV	PT	DESCRIPTION
X	1	ARRANGEMENT
1	2	GRID PLATE
163	3	FUEL ASSEMBLY SIMULATOR
9	4	TEST POSITION SIMULATOR EXTENSION
1	5	SUPPORT
3	6	SIMULATOR RETAINING BAND
1	7	FRAME
6	8	HYDRAULIC CLAMP ASSEMBLY

APPROVED FOR CONFORMANCE		DATE		U. S. ATOMIC ENERGY COMMISSION	
WITH DESIGN CHANGES				RICHLAND OPERATIONS OFFICE	
APPROVED				PACIFIC NORTHWEST LABORATORY	
TYPE				OPERATED BY BATTHELL MEMORIAL INSTITUTE	
DRAWN				SIMULATED CORE	
CHECKED				MODEL SCM PLAN	
SCALE				FAST FLUX TEST FACILITY	
CLASSIFIED BY				9300G-FF INDEX NO 4703.01	
DATE				DATE	
NONE				SK-3-14762	

REFERENCE DRAWINGS	REVISIONS	DRAWING STATUS
DATE	DATE	DATE

CHAPTER VI. CORE DESIGNA. CORE MECHANICS TECHNOLOGY1. Core Radial Restraint Model

G. R. Waymire (811) and L. R. Besel (913)

Although the core radial restraint problems associated with single core ducts are amenable to analysis using present analytical tools, the problems associated with the array of ducts present a complexity which, while theoretically possible to analyze, becomes complex in implementation.

To circumvent this analysis problem, a structurally accurate model of the core is required which can be used to study the overall structural core problems.

The possibility of using a scale model or section model or a combination has been investigated as a more economical solution to the problem. These alternatives, however, result in problems either with obtaining all desired parameters simultaneously, in the case of a scale model, or with being able to properly mock-up boundary conditions in the case of a segmental model. After reviewing the problems with Westinghouse, it was jointly determined that development effort of the simulated core model (SCM) should concentrate on a full scale model.

The conceptual design for the SCM is shown on SK-3-14762 and SK-3-14763. The model uses mechanical tubing for simulated stiffness with hexagonal pads located at the contact planes.

In the outer rows of the core where extreme deformations occur, the tubing may require bending to the calculated duct deformations. In the inner rows, offset replaceable pads will be used.

SECTION A-A
(SEE SK-3-14762)

0 1 2 3 FT.

DWS NO.		REV NO.	DRAWING TITLE OR INDEX NO.	BY DATE	CHK DATE	REV DATE	DESCRIPTION	REV NO.
NEXT USED ON		REFERENCE DRAWINGS		COMMENT PRT ISSUE NO.		CHECK PRT ISSUE NO.		DATE
DRAWN		WU JACKSON		10-9-69		NOTED		10-9-69
CHECKED		G.R. WAYMIRE		10-9-69		NOTED		10-9-69
APPROVED		U. S. ATOMIC ENERGY COMMISSION		RICHLAND OPERATIONS OFFICE		PACIFIC NORTHWEST LABORATORY		OPERATED BY BATTTELLE MEMORIAL INSTITUTE
CONCEPTUAL		1ST ISSUE		10-9-69		SIMULATED CORE MODEL SCM ELEVATION		FAST FLUX TEST FACILITY
NONE		SK-3-14763		4703.0		C300G-FF		4703.0

For initial tests, hydraulic jacks will apply the restraint load. As the actual restraint design becomes firm, the hydraulic system will be partially replaced by prototype mechanisms.

2. Swelling and Creep Effects upon Fast Reactor Core Structural Design

R. J. Jackson (832), W. H. Sutherland,
and I. L. Metcalf (914)

Because metal swelling is both temperature and fluence dependent, the resulting dimensional change results in both dilation and distortion in fast reactor core components. The most significant effect of the swelling gradient induced distortion is the bending of long slender core components such as fuel subassemblies, control and safety rods, control and safety rod guide tubes, experimental closed loops, reflectors and shields. In extreme cases these bending effects conceivably could cause undesirable interactions between core components.⁽¹⁾ Because of the concern for the operation of the FFTF, analyses have been conducted to determine the distortion shape of these components during refueling. These analyses have considered the combined effects of metal swelling and creep upon the resulting bending distortion. The metal temperature in these core components is too low for thermally activated creep to be effective. However, neutron-induced creep⁽²⁾ has a pronounced effect upon the magnitude of the bending distortion. For example, an unrestrained row six FFTF fuel assembly duct at goal burn up is predicted to bow outward more than one inch as a result of the radial

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1. R. J. Jackson, C. L. Wheeler, G. R. Waymire, and W. W. Eittle. Metal Swelling Effects on FTR Core Components. To be published.
 2. D. Moselkae, G. W. Lewthwaite, G. O. Leete, and W. Sloss. "Irradiation Creep in the Dounreay Fast Reactor," Nature, vol 224. December 27, 1969.

swelling gradient if creep is not considered. However, if neutron-induced creep is considered, the unrestrained duct is predicted to bow inward about 1/4 in. This unrestrained deflection is also very sensitive to the temperature gradient across the duct. This is because the neutron-induced creep causes the thermal stress in the duct to greatly reduce from its initial value at beginning of life. Under most conditions in FFTF, neutron-induced creep is predicted to maintain the bending stress below its initial thermal value even though the swelling gradient is active to increase this bending stress. As a result, when the thermal gradient and the restraints are both removed there is a tendency for a net inward movement of the duct top in contrast to the outward movement if creep is neglected. These calculated unrestrained bending distortions are tabulated in Table 6.1.

TABLE 6.1. Bending Distortion in FFTF Core Components

<u>Core Component</u>	<u>Row No.</u>	<u>Unrestrained Maximum Deflection During Refueling, in.</u>
Fuel S/A Duct	1	0.0126 inward
Fuel S/A Duct	2	0.0214 inward
Fuel S/A Duct	3	0.0185 inward
Fuel S/A Duct	4	0.0094 inward
Fuel S/A Duct	5	0.1464 inward
Fuel S/A Duct	6	0.3100 inward
Safety Rod Thimble	4	0.0172 outward
Control Rod Thimble	7	0.1374 inward
Closed Loop	2	0.0293 inward
Closed Loop	4	0.1467 inward

Note that adjacent ducts will interact in such a way that these components will not actually experience the duct top deflection calculated in Table 6.1. Note also that the

safety rod thimble in row four is predicted to bow outward in contrast to all other ducts. In the safety rod the heat is generated in the poison pins which are above the core during operation. In all other components the heat is generated in the core. This difference in temperature distribution causes the difference in bending distortion which was calculated.

We conclude that neutron-induced creep has a significant effect upon core component bending distortion. We also conclude that the unrestrained bending distortion is very sensitive to the temperature distribution in the core component during operation.

B. CORE PROCESS TECHNOLOGY

1. Fuel Assembly Design and Testing

E. G. Stevens (913)

a. CCTL Mark II

Fabrication of the CCTL Mark II 217 pin full-scale fuel assembly has been completed. Shipment to the test loop at Argonne National Laboratory where the fuel assembly will be sodium life tested will take place in early March. Prototypic FTR environmental conditions of 1100 to 1150 °F (maximum steady-state temperature) at 525 gal/min will provide pin bundle erosion, corrosion, fretting, and vibration effects as well as instrumentation package performance data for 9000 hr.

b. Fuel Assembly Vibration Testing

E. G. Stevens (913) and G. R. Sawtelle, (COO)

A 19-pin, 0.250-in. tube OD, 30 mil wire wrap assembly (B-1 test) instrumentation has been refined to increase its impedance-detecting sensitivity to measure low level pin vibrations (approximately 1 mil amplitudes). A no-flow duct shaker test was run in February as a prelude to water flow tests.

Design of a 19-pin, 0.230-in. OD, 56 mil wire wrap assembly (B-2 test) is nearing completion. Three types of vibration sensors are to be used in the instrumented pins: accelerometers, impedance detectors, and strain gages. Fabrication completion is scheduled for June.

c. Hydraulic Testing of FFTF Fuel Assemblies

J. Muraoka (913)

The objective of this program is to evaluate the performance of FIR driver fuel subassemblies in terms of pressure drop-flow behavior, local flow distribution and coolant mixing. The program is divided into three subtasks to examine 7, 37, and 217-pin subassemblies. The seven-pin-bundle tests will parallel the heated seven-pin-bundle tests. The results of these two tests provide the input for an analytical model. The 37-pin water tests will be used to extend analytical model to predict behavior in a 217-pin subassembly. The final 217-pin test will provide the experimental verification of the analytical model to predict full-size subassembly behavior.

In this report period the following accomplishments were made preparatory for the seven pin tests:

- All design was completed.
- a Construction of the 2 in. instrumented flow loop was completed.
- a Quality Assurance documentation was completed on the instrumented pins of the first seven pin subassembly.
- a Noise analysis calibration test assembly was completed and velocity probe calibrations are about 30% complete.

d. Heat Transfer Characteristics of FFTF Fuel Subassemblies

J. Muraoka (913)

The objective of the program is to define the thermal hydraulic behavior of electrically heated seven-pin subassemblies

in sodium to support the FFTF fuel design. Two seven-pin subassemblies have been examined. The first subassembly test examined a tight, normal pin bundle. The results were reported in the previous FFTF quarterly report.⁽¹⁾ In this report period the second subassembly testing was completed. The second subassembly examined a tight bundle in an oversized duct to examine the effect of manufacturing tolerance accumulations. The significant preliminary conclusions of this test are:

- Increasing the bundle to duct clearance increased the temperature difference across the outer pin row. Increasing this clearance from 3.5 to 15 mils (radial clearance) increased the temperature difference by as much as 30 to 40%.
- The hot spot measured under a wire wrap in the 30° position was not excessive. The data indicated that a wire wrap facing an inner subchannel increases the local clad temperature about 25 °F. The heat flux is a nominal 500,000 Btu/hr-ft² in these tests.
- The clad temperature variation in an outer row pin was significantly less than the maximum coolant temperature variation. These temperature variations indicate a two-fold variation in the local heat transfer coefficient around the pin.

2. Hydraulic Model of Reactor Vessel Outlet Region

H. Leigh (954) and D. L. Ballard (COO)

Tests are being performed on a hydraulic model to study gas entrainment and mixing conditions in the outlet region of the FFTF reactor vessel. Water is being used as the test media.

1. Fast Flux Test Facility Quarterly Technical Report, September, October, November 1969, BNWL-1275. Battelle-Northwest, Richland, Washington, January 1970.

The model is a nominal 1/3 size simulation of the reactor between the core exit and the vessel cover. Figure 6.1 shows an exterior view of the model. The model is connected to a 3600 gal/min capacity loop with flow control and measurement capabilities on the inlet and each of the three outlets. Each outlet is provided with a transparent plastic spool piece for observation of the discharge stream.

A newly developed ultrasonic transmission technique, (1) was used to quantitatively measure gas entrainment on these model tests. The apparatus used in this technique measures attenuation changes of ultrasonic energy caused by gas bubbles in the ultrasonic beam (the apparatus does not detect dissolved gas). The ultrasonic transducer was located on the discharge pipe well downstream of the convergence of the three discharge flows on the model. The system was calibrated at each flow rate and each model configuration to result in a system meter output versus void volume fraction. These calibration data then made it possible to obtain accurate quantitative information on void fraction for each test configuration.

Tests to date, have shown that a problem of gas entrainment does exist for some outlet region configurations. The tests have shown that the model is suitable for evaluation of this problem, and results have indicated that good data can be obtained regarding void fraction for each specific configuration.

One of the design features of the FFTF is the use of "instrumentation support plates" actuated by "instrumentation trees." These items were modeled with respect to melted

2. Fast Flux Test Facility Quarterly Technical Report, September, October, November 1969, BNWL-2275. Battelle-Northwest, Richland, Washington, January 1970.

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FIGURE 6.1. Exterior View of 1/3 Scale FFTF Reactor Vessel Outlet Region Model

surface. The model tests have shown that the position of the instrument support plates above the top of the driver fuel has a significant influence on the surface turbulence and the resultant air entrainment in the effluent stream.

C. IRRADIATION TESTING TECHNOLOGY

1. Closed Loop Tube Nozzle Closure Development

a. Scope and Objectives

C. A. Munro (914)

One of the major problems in assuring individual removal and replacement of test specimens in the operation of a test reactor such as the FFTF is the selection of methods for sealing reactor cover penetrations. The cover penetrations require that a primary refueling closure is made on each closed loop tube. This closure, which is made at the top portion of the tube (nozzle), is difficult to achieve because of the requirement for a small space envelope and the high radiation and temperature levels.

The objective of this development program is to investigate both the mechanical sealing concept and the welding-cutting concept for making this connection.

b. Closed Loop Nozzle Mechanical Closure Studies

R. Kolowith (914) and M. R. Kreiter (COO)

An extensive program of testing available commercial mechanical closures has been completed. None of the closures tested met the requirement for maximum gas leakage of 10^{-7} atm/sec under all the designated conditions. Hence we have concluded that the remotely operable and compact closure required for this application is not within the current state of technology.

c. Closed Loop Nozzle Weld Closure Studies

R. F. Gilmore (914) and L. J. Rousseau (COO)

A new remote weld head has successfully completed a series of six remote seal welds over molten sodium inside the environmental test chamber. Figure 6.2 shows the exterior of the chamber while Figure 6.3 shows the interior of the chamber with the remote weld head in position to perform a nozzle weld.

Prior to weld testing, the chamber was evacuated to around 500 μ and then back filled with helium gas. The recirculating gas purification system was used to maintain the oxygen content at less than 1% and moisture at less than 2%. Molten sodium was admitted into the nozzle spool piece and the weld joint area was stabilized to 475 °F for 2 hr. The sodium was then drained back into the reservoir, and the remote weld was made. Sodium was once again brought up into the test specimen to a level of 1-1/2 in. below the weld area and held at 450 to 500 °F for 2-1/2 hr.

Three of the test specimens were inspected radiographically; both X-ray and irradiation sources were used. The X-ray radiographs showed the weld bead shape to be uniform and of the desired shape. One of the specimens was pressurized to 250 psig and held for 2-1/2 hr at 530 °F to verify structural integrity. Mass spectrometer helium leak detection tests performed on all the specimens have shown leakage less than 10^{-7} atm/sec through the seal weld. Statistical data is being gathered to further evaluate machine durability and weld joint characteristics.

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FIGURE 6.2 Environmental Test Chamber

Neg 51298-3

FIGURE 6.3. Interior Environmental Test Chamber

2. Closed Loop Insulation Studies

K. R. Wheeler, S. W. Binegar (AOO), and S. M. Gill (962)

Thermal stresses in the walls of the closed loop tubes extending through the FIR core will be minimized by means of felted metal insulation. The purpose of this study is to determine the combination of insulated wall structural parameters which give a required thermal conductivity of approximately 2 Btu/hr/ft/°F. Insulated wall sections will be evaluated in air and in flowing sodium over a range of temperatures up to and including 1200 °F.

Commercially produced cylinders fabricated by braze bonding a layer of felted metal* insulation to the OD surface of a 304 SS tube were ordered in July 1969. These test pieces, to be ultimately fabricated into prototypic closed loop wall sections, formed the basis for evaluating: (1) the influence of felted metal density and bonding on thermal conductivity and (2) any fabrication problems associated with the use of this material for small diameter cylinders.

Forming methods employed by the felted metal producer are only capable of fabricating 0.20 density or less material into cylinders relatable to closed loop design. Tensile instability during room temperature bending is observed for higher densities.

A cylinder of 0.20 density felted metal braze bonded to an inner 304 SS shell was successfully fabricated in December 1969. The prototypic sandwich construction of the closed loop wall was completed at BNW with addition of the outer shell. Argon gas was sealed into the felted metal interstices and the composite cylindrical wall section was evaluated for its thermal conductance in an axially heated furnace providing radial heat flow.

* *Chemical makeup as follows: 0.3% C, 2.0% Mn, 1.0% Si, 0.04% P, 0.5% Cu, 0.5% Mo, 0.03% S, 17.0 to 19% Cr, and 7 to 10% Ni.*

Thermal conductance K (Btu/hr/ft/°F) can be calculated from Equation (1)

$$q_k = AT K_k \quad (1)$$

The heat input q_k and the radial temperature drop AT across the composite wall are all readily measurable quantities. Thermal conductivity values for the 0.20 density insulation test section over a temperature range 900 to 1400 °F vary from 0.312 to 0.435 Btu/hr/ft/°F for 900 °F and 1400 °F, respectively. Results indicate that a composite wall must be constructed with greater heat conductance than this first test sample.

Tests planned for next quarter include:

- Use of helium gas in the interstices of 0.20 density felted metal to increase conductance and
- Evaluation of the thermal conductivity of 0.60 density felted metal in a flat sandwich construction. If 0.60 density material proves to be the optimum insulation, a study of forming methods will be undertaken.

D. OTHER CORE TECHNOLOGY

1. Examinations of Test Specimens

C. L. Boyd (942)

a. Disassembly-Reassembly Equipment

Copies of the ORNL report,⁽¹⁾ which describes the concept and development program for equipment needed to remotely position and examine FFTF core components and full length test assemblies up to 43 feet long were issued. The remote examination operations include removal of the pin bundle from the flow duct, visual inspection, photography, measurement of overall dimensions and interpin spacing, disassembly of the

1. J. N. Baird et al. Conceptual Study of Disassembly Equipment for Examination of FFTF Core Components and Test Assemblies, ORNL-TM-2759. Oak Ridge National Laboratory, Oak Ridge, Tennessee, January 2970.

pin bundle, testing and repairing of instrumentation, and reassembly of test assemblies for further irradiation.

As conceived, the FFTF Disassembly-Reassembly Equipment (DRE), shown in drawing M-11325-EM-050 consists of a positioning machine and tooling. The DRE will be capable of being remotely installed, operated, and maintained in a hot cell with a dry inert atmosphere (argon) where it will be subjected to high gamma radiation, residual sodium and sodium vapor, and high radiant heat. Suction-flow cooling of an assembly containing fuel will be provided during dismantling and examination by means of devices built into the positioning machine. A cooling capacity of up to 20 kW will be provided by axial flow of argon coolant through the assembly, and a cooling capacity of up to 10 kW will be provided by transverse flow of argon coolant through the fuel bundle.

The concept of the positioning machine is based on the operating philosophy of vertically maneuvering the assembly to provide optimum viewing and manipulation and upon the need to axially withdraw the pin bundle from the severed flow duct to expose the bundle for examination. The amount of vertical maneuvering required is reduced by the use of three cell operating levels, each of which is equipped with a viewing window, a pair of sealed master-slave manipulators, and a work table. Emphasis has been placed on the use of master-slave manipulators as much as possible for functional operations.

The tooling used for the disassembly and examination operations will be portable items to be positioned and used on the work tables with the manipulators. The items needed include girth cutters and longitudinal cutters to sever and slit the flow duct that forms the jacket of the pin bundle, various tools and cutters needed to disassemble and reassemble

the pin bundle, and measuring equipment, which is not now considered to be part of the disassembly-reassembly equipment.

The structural frame of the positioning machine for 43-ft specimens is approximately 58 ft 6 in. tall, and it consists of a modular column and base that is anchored to the cell floor. The working portions of the machine consist of three separate specific-purpose slide bodies that ride in tracks or ways on the face of the column and hold specialized subassemblies that perform various work operations. The upper slide is equipped to grasp the upper end fitting of a core or test assembly transported to the machine by crane, and the lower slide is equipped to grasp the lower end of an assembly. The lower slide also has a plenum for axial cooling of the assembly and a power-rotated faceplate to turn the assembly during cutting and examination operations. The middle slide is equipped with cooling shrouds for transverse cooling of the exposed fuel bundle, and various clamps and fixtures can also be mounted on its face. These slide assemblies may be moved up and down independently or any two or all three of them may be moved in unison. For disassembly operations, the middle slide moves with the pin bundle and is slaved to the lower slide for downward withdrawal of bottom-supported bundles. Conversely, the middle slide is slaved to the upper slide for upward withdrawal of top-supported bundles.

Recent changes in criteria for the subject equipment requires that additional work be performed to augment the conceptual study. This additional work may be generally divided into three parts as follows:

- Determining the effects upon the disassembly-reassembly equipment and facility related requirements when the lengths of FFTF core components and test assemblies are reduced to 12 feet.

- Preparing new cost and scheduling data for future portions of this project to reflect new criteria.
- Documenting the results of these additional studies in an addendum to the conceptual study report. (1)

b. Decay Heat Removal Development

D. R. Dickenson, T. C. Reihman, and R. A. Hildner (COO)

The major goal of this task is to evaluate the operating conditions (e.g., flow rates, flow split and pressure drop) in order to adequately control the fuel clad temperature of a discharged fuel assembly during fuel examination. The critical problem demonstrated in the FY 1969 testing was hot zones due to local flow stagnation points during removal of the duct from the fuel assembly. The problem was caused by interference between the combined axial positive displacement and transverse suction flow rates during the transition period.

To prevent the hot spots and to assist in the control of in-cell contamination, the decay heat removal method was changed to provide suction rather than positive displacement axial flow. Modifications to the test facility to simulate the new method have been completed.

1. See *Reference 1*, page 6.13.

CHAPTER VII. FUELS AND MATERIALSA. FUELS AND MATERIALS EVALUATION1. B₄C - Stainless Steel Compatibility

L. R. Bunnell (AOO)

The compatibility of unirradiated and irradiated B₄C with Type 316 SS at 550 and 600 °C for 1000 hr was recently investigated. The irradiated B₄C was obtained by scraping high burnup material from the surfaces of 99% TD pellets irradiated in a Hanford reactor. B₄C powder of roughly the same particle size (-325 mesh) was used as a standard. Each capsule used in the test contained an enclosed dead weight to pressurize the stainless steel - B₄C interface to 100 psi. No reaction occurred in each of the samples tested at 550 °C. However, when both samples were heated at 600 °C, they reacted to yield a reaction layer about 2.5 μ thick, as shown in Figure 7.1. There was no discernable difference between reactions of virgin or irradiated B₄C.

Neg 470-36B

500X

FIGURE 7.1. 2.5 μ Reaction Zone on 316 Stainless Steel
Exposed to B₄C Powder for 1000 Hours at
600 °C.

2. Residual Stresses in Irradiated Fuel Cladding

J. F. Bates and R. L. Fish (722)

Work is proceeding to determine if residual stresses in unfueled specimens can be detected by means of optical interference holography. Preliminary tests on unirradiated tubing have demonstrated that interference holograms can be produced through 12 in. of lead glass. The technique appears to be adaptable to hot cell work. Figure 7.2 shows a demonstration of this technique. In this demonstration, a hologram for an unfueled section of cladding was constructed. Subsequently, the specimen was stressed to 2000 psi by internal gas pressurization, and a new hologram was made. The two images were then superimposed which resulted in the interference patterns of Figure 7.2. The magnitude of the stresses calculated from the fringes agreed well with calculations based on elastic theory.

FIGURE 7.2. Holographic Fringes Produced on a Tube Pressurized to 2000 psi

3. Weldment Studies Specimen Size

A. L. Ward, A. J. Lovell, and L. D. Blackburn (A00)

A tensile test study was made to determine whether a larger diameter specimen would improve the reproducibility of test results and also be more representative of the bulk weld material being studied. Results from the larger specimens (0.250-in. gage diameter, 1.125-in. gage length) tested in the range from 600 to 900 °F. are in general agreement with results from the miniature (0.125-in. gage diameter, 1.125-in. gage length) specimen used in the weldment irradiation experiment. No consistent improvement in the magnitude of data scatter was observed upon comparison of the two sets of results. Therefore, present plans for future irradiation experiments involving weldment materials call for continued use of the miniature specimen.

4. Mechanical Testing of Fuel Pin Cladding

R. L. Fish (722), L. A. Pember, J. W. Weber,
R. D. Leggett (750), and E. D. Jensen (A00)

The purpose of this program is to provide a basis for evaluating the combined effects of fuel cladding interaction, fluence and operating temperature on the postirradiation burst and stress-rupture properties of fuel pin cladding.

Burst properties of fueled cladding sections from PNL-1 series fuel pin irradiations at 900 °F have been given in previous quarterly reports. The most important finding in these previously reported studies was a very low strength and ductility region above the reactor midplane. The low strength is thought to be due to grain boundary attack during testing of sensitized cladding. The nature of the corrosive has not been identified.

Additional fuel-containing cladding specimens prepared in a dry, inert atmosphere, but stored for about 10 months in hot

cell air after puncture of cladding for fission gas analysis, have revealed intergranular cracks during leak testing. These specimens were from the upper fueled regions of PNL 1-16 and 1-18 pins.

The origin of the stresses necessary to cause this cracking is not known but the crack orientation suggests the pins may have been bent. The intergranular failure in PNL 1-16 and 1-18 occurred at ambient temperatures which demonstrates the grain boundary weakness effect is not confined to elevated temperatures.

The electron microprobe has been used to examine grain boundaries in failed areas to determine the extent of cladding penetration by fission products. The cladding of fuel pin specimen PNL-1-18 which revealed fissures during leak testing was examined for cesium, iodine, sodium, phosphorus, and rubidium. One side of this section which revealed the typical cladding microstructure at fissures was examined for the above elements. Figure 7.3 shows the area examined. X-ray images for the above elements revealed only one area of sufficiently high concentration to register on the oscilloscope display. Cesium was concentrated at the intersection of two grain boundaries located 50 μ from the fuel-cladding boundary, as shown in Figure 7.4. The area of high cesium concentration also showed barely detectable amounts of iodine, but not enough for an X-ray image display. Cesium occurs in the grain boundaries extending in all three directions from the cesium spot shown in Figure 7.4. This is shown in Figure 7.5 in which the cesium counting rate (corrected for background) is plotted against distance from the grain boundary intersection for each grain boundary. Plutonium was monitored simultaneously with cesium and was not detectable in the cladding in any location sampled, indicating that the cesium is not polishing debris. The concentration of cesium would be valuable information, but cannot

FIGURE 7.3. Area of Cladding of PNL 1-18 Examined in the Shielded Electron Microprobe. Arrow Indicates Grain Boundary Intersection Shown in Figure 7.4

FIGURE 7.4. Specimen Current, Cesium, and Plutonium Distribution in a Grain Boundary Intersection in PNL 1-18. (Scattered dots on cesium and plutonium displays are background. White areas represent the element distributions. Grain boundaries beyond the fissure region are not resolvable. 80×80 microns; 1000X.)

FIGURE 7.5. Variation of Cesium Concentration with Distance from the Maximum Concentration Within the Three Grain Boundaries Shown in Figure 7.4. Plot numbers refer to numbers in Figure 7.4.

be determined without knowledge of the distribution of cesium with depth. This is, the observed counting rates could represent 100% cesium in a very thin layer or a much lower concentration distributed in the grain boundary to a greater depth. Detectable amounts of cesium were also observed in two grain boundaries 25 and 50 μ from the inside of the cladding.

Analysis for phosphorus and rubidium within grain boundaries was attempted and no detectable amounts of either element could be found using both area X-ray displays and point counting within grain boundaries.

Sodium from the coolant was also suspected of having caused the cladding failure; however, area X-ray scans showed no detectable sodium. Point counting across the cladding in 1 μ steps showed one location seemingly containing sodium; however, returning to these areas and making wavelength scans on the suspected areas gave no indication of sodium. Investigation into the history of this pin since removal from the reactor revealed that it had been washed with water and also with alcohol. This would remove sodium on the outside and probably to some extent within the separated grains near the outside of the pin.

The cause of the boundary weakness is very important to fast reactor technology. If the weakness is due to storage environment, then pin handling procedures must be improved in order to preserve data. Alternately, if the effect is inherent in fuel pin performance, then the cladding will not be capable of supporting the loads which appear safe from unfueled irradiation experiments.

In order to clarify the mechanism involved with the grain boundary weakness, a number of definitive experiments have been conducted and additional experiments are planned.

Initial results were obtained from a ring test on a specimen cut from a biaxial test sample which failed prematurely by grain boundary fracture in a biaxial test. In the ring test a 1/4-in. long section of cladding is pulled in a tensile machine in such a way as to create tangential stress in the cladding wall. Fuel and fission products are removed prior to testing at 900 °F. The metallography on these ring test ruptures is shown in Figure 7.6. The good ductility and strength exhibited by this cladding with the fuel and fission products removed contrasts greatly with the very low ductility ($<1.0\% \Delta D/D$), intergranular failures observed at the same test temperature in fueled cladding from the same region of the fuel pin. This indicates that the premature failures are due to a corrosive attack from within the fuel pin and are not a result of external attack or inherent cladding grain boundary weakness. Two burst tests (one fueled and one with fuel removed) are planned to verify the effects observed in these ring tests. Carbon extraction replicas are also being obtained from the ring test samples for further characterization of their microstructure.

- a. Irradiation Conditions:
 $\phi t \approx 3.7 \times 10^{21} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$)
 $T_{\text{ave}} \approx 945 \text{ }^\circ\text{F}$

- b. Irradiation Conditions:
 $\phi t \approx 2.8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$)
 $T_{\text{ave}} \approx 940 \text{ }^\circ\text{F}$

FIGURE 7.6. Microstructure of Ring Test Samples
(Test temperature = 900 °F.)

Fuel pins PNL 1-9, 1-12, 1-13 and 1-14 were helium leak-tested for possible fissures. These pins have not been punctured nor otherwise intentionally compromised. This did not reveal any fissures or leaks. The outside surface of the cladding on these same fuel pins was replicated with cellulose nitrate plastic. This replication did not reveal any intergranular cracking or other cladding perforations. The only significant characteristic revealed by the replicas was a pronounced delineation of the grain boundaries in the upper fueled region of PNL 1-14 (Figure 7.7). This is believed to be heavy carbide precipitation and is not unexpected in view of the calculated temperatures of this cladding during irradiation.

FIGURE 7.7. Precipitate Delineation of Cladding Grain Boundaries in Upper Fueled Region of PNL 1-14

5. Stress Rupture Studies - Effect of Aging
of Type 316 Stainless Steel

R. W. Barker (722)

The objective of this program is to determine the effects of fast reactor irradiation on the biaxial stress-to-rupture properties of candidate fast reactor cladding materials. Emphasis is currently being placed on testing thermal control specimen related to EBR-II Subassembly X-022.

The specimens are made from 0.208-in. diameter, 0.008-in. wall, AISI Type 316 SS tubing in as-received condition, i.e., a small degree of cold work retained from the fabrication processes.

Specimens were aged at 800 °F for 3045 hr in sodium (O_2 -15 ppm; C-29 ppm). Residual sodium was removed by dissolving and rinsing with water and drying with ethyl alcohol. Biaxial stress-rupture tests were subsequently performed at 1000 and 1200 °F by pressurizing the tubular specimens with a high pressure gas and holding the pressure constant until rupture occurred (Figure 7.8). The data show an increased rupture stress for the aged material over that of the as-received material. The strength increase at 1000 °F is not readily apparent at rupture times less than 100 hr, while rupture times greater than 100 hr the strength increase is about 10%. At 1200 °F, the difference between rupture strengths increases with increasing rupture time to about 20% at 1000 hr rupture time. An additional test at 20,000 psi and 1200 °F is unfailed at 2500 hr, which shows that strength increase persists to longer rupture times. Ductility of the aged material (Figure 7.9) is also affected. At 1000 °F, the ductility appears to have been slightly enhanced for rupture times less than 50 hr but degraded for greater times to rupture. Also, the loss of ductility becomes greater with increasing rupture time.

Neg 0690052-2

FIGURE 7.8. The Effect of Aging on the Biaxial Stress-Rupture Properties of AISI Type 316 Stainless Steel. Aging Conditions Were 800 °F and 3045 hr

Neg 0690052-1

FIGURE 7.9

The Effect of Aging on the Ductility of Biaxially Stressed
 AISI Type 316 Stainless Steel. Aging conditions were 800 °F
 and 3045:hr

Ductility of as-received material was lower at 1200 °F than at the other test temperatures, 100 and 1400 °F. However, the aging treatment markedly enhanced the 1200 °F ductility. Ductility increases range from about 50% at 10 hr to 100% at 600 hr.

6. Damage Analysis

J. L. Straalsund (722), H. R. Brager (7A0), and
J. F. Bates (722)

The objectives of this effort are to establish the irradiation-induced swelling characteristics of FFTF alloys, to relate fast-reactor-induced substructural changes in microstructure to corresponding changes in mechanical properties, and to develop means for data extrapolation and interpolation, by using both microscopic and empirical approaches.

Equations are presently being developed empirically to describe the effects of neutron exposure and irradiation temperature on void size and concentration in 304 and 316 SS. The purpose of this work is to develop a set of three self-consistent equations describing the void diameter, void concentration and void volume (swelling) over as wide a range of experimental conditions as possible. A consistent set of equations accounting for the observed trends will provide more confidence in extrapolations to the higher exposures expected in the FTR.

a. Void Concentrations Equations

Best fits obtained to date for the void concentration data developed at BNW are:

$$\rho = (3.31 \times 10^{-5}) \left(\frac{\phi t}{10^{22}} \right)^{0.90} (e^{7030/T-1.0}) \quad (1)$$

$$\rho = (1.21 \times 10^{-5}) \left(\frac{\phi t}{10^{22}} \right)^{-0.45 + 0.0019T} (e^{7740/T-1.0}) \quad (2)$$

where

$$\phi t = n/\text{cm}^2 \quad (E > 0.1 \text{ MeV})$$

$$\rho = \text{voids}/\text{cm}^3 \times 10^{-15}$$

$$T = \text{temperature, } ^\circ\text{K}.$$

These equations are based on transmission electron-microscopy of 17 specimens that had been irradiated over the temperature range between 370 and 620 $^\circ\text{C}$ and to fluences between 0.5×10^{22} and 5.9×10^{22} n/cm^2 .

The limited amount of data presently available and the uncertainty associated with the irradiation conditions make it difficult to determine which equation provides the better fit with the data.

b. Void Size Equations

Several equations were developed which provide good fits to the void size data. Two examples are:

$$d = -417 + 0.87T \quad (3)$$

$$d = 4150 \left(\frac{\phi t}{10^{22}} \right)^{0.082} e^{-2202/T} \quad (4)$$

While different in form, these equations predict similar values over the range of conditions for which we have data. Equation (3) fits over the whole range of conditions better than Equation (4) which was included to emphasize the insensitivity of void size to fluence. The estimated standard deviation of the fluence exponent in Equation (4) is 0.042, i.e., half as large as the exponent itself. This indicates that the fluence dependency of the void size data is almost statistically insignificant.

7. Fuel Clad Interface

R. W. Barker (722)

Further work was conducted toward identifying corrosive constituents at the fuel clad interface in irradiated BNW mixed oxide fuel pins.⁽¹⁾ Capsules containing potential corrosive agents were constructed and thermal aging begun.⁽²⁾

This experiment was removed from the furnace after 519 hr for an intermediate examination. Most capsules were found to be in good condition and were returned for further aging. However, Lot "F" Type 304 and Lot "G" Type 316 capsules containing anhydrous rubidium hydroxide had failed. The condition of the capsules is shown in Figure 7.10.

Optical examination showed an extensive network of intergranular attack as shown in Figures 7.11 and 7.12. Note that whole grains have been dislodged. The similarity between micrographs of rubidium hydroxide attack and that observed in BNW mixed oxide fuel pins is striking.

Calculations indicate that the intergranular attack progressed under a stress of only about 200 psi, thermal stresses excepted. Note that the intergranular attack occurred near the exterior surface, apparently after failure. This observation is important because this type of attack may progress in the absence of significant stresses.

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1. FFTF Quarterly Technical Report June, July, August, 1969, BNWL-1174, Battelle-Northwest, Richland, Washington, (Section VII-A).
 2. Quarterly Technical Report October, November, December, 1969, BNWL-1275, Battelle-Northwest, Richland, Washington, (Section VII-A).

Neg 469-2734-354

1X

Neg 469-2732-33A

1X

FIGURE 7.10. Ruptured Capsules of Lot 'F' 304 (left) and Lot 'G' 316 Stainless Steel (right) Charged with 1 g of Anhydrous Rubidium Hydroxide and Aged 519 hr at 900 °F

Neg 469-2734B

100X

FIGURE 7.11.

Transverse Section of Lot 'F' Type 304 Stainless Steel Charged with Rubidium Hydroxide and Aged 519 hr at 900 °F.

Neg 469-2734D

250X

250X

Neg 469-2732F

Transverse Section of Lot 'G' 316 Stainless Steel Charged
with Rubidium Hydroxide and Aged 519 hr at 900 °F.

100X

Neg 469-2732B

FIGURE 7.12.

B. MATERIALS TECHNOLOGY*1. Uniaxial Creep

A. J. Lovell and L. D. Blackburn (AOO)

The objectives of this program are to evaluate the effects of fast neutron irradiation and environment on the pre- and postirradiation uniaxial creep-rupture properties of FTR vessel and core structural materials and to establish the usable materials limits to assure their safe and reliable performance.

Creep-rupture results on specimens from Subassembly X018 show that large reductions in rupture life of Type 316 SS occur after irradiation to approximately 3×10^{22} n/cm² (total) in the temperature range 1000 to 1100 °F. These losses in rupture life mean that the allowable stresses for highly irradiated components may be significantly lower than for unirradiated components. For example, the stress to achieve a rupture life of 1000 hr in unirradiated material at 1100 °F is about 32,000 psi, while the stress to reach a similar rupture life after irradiation to 3.5×10^{22} n/cm² (total) is only about 21,000 psi.

2. Weldment Studies

A. L. Ward, A. J. Lovell, and L. D. Blackburn (AOO)

The objective of this program is to provide mechanical property and metallurgical stability data for FTR vessel and core structural weldment materials in order to define adequate performance under anticipated FTR service conditions.

Tensile testing of the first FTR vessel weldment irradiation experiment has been completed. Results have been reported for all-weld specimens from the TIG, MIG, submerged-arc, and stick-electrode processes. These results have shown that the tensile properties are essentially unaffected by irradiation to $\sim 1 \times 10^{21}$ n/cm² (total) at 700 and 800 °F, while substantial

* Reported by *J. C. Tobin (960)*

strengthening and ductility loss are observed following irradiation to 3.8 and 4.4×10^{21} n/cm² (total) at 800 °F, except in the case of the stick-electrode weld where ductility is apparently unaffected by the higher fluence irradiation. The weld-deposited materials show smaller irradiation-induced yield strength increases and smaller ductility losses than the base-metal; however, the preirradiation yield values were higher and the ductilities lower in the weld material.

If a total elongation criterion of 10% is applied to FTR vessel materials, it appears that the weldments in this study could be employed safely, provided the maximum fluence (4.4×10^{21} n/cm², total) is not exceeded. This value may well constitute the fluence limit (at 800 °F) to which the TIG and MIG weld processes are useful, since these weldments showed postirradiation total elongation values near 10%.

A study has been conducted to determine whether a larger diameter specimen would improve the reproducibility of test results and also be more representative of the bulk weld material being studied. Specimens with diameters of 0.250-in. (compared to 0.125-in. for the miniature specimen used in the weldment-irradiation experiment) were fabricated from base metal (Type 304) and weld metal from the four weld processes. Tensile tests were conducted in the range from 600 to 900 °F. No well defined improvement in behavior was observed; the magnitude of the scatter band is essentially the same as that shown by the miniature specimens. Therefore, plans for future irradiation experiments call for continued use of the miniature specimen.

Some preliminary results from creep-rupture tests of solution-treated base metal and weld metal specimens from X067 irradiated to approximately 8×10^{21} n/cm² (total) at 1100 °F are now available. The solution-treated base metal and TIG weld specimens show substantial losses in rupture

life, while the sub-arc weld metal shows little change. This variation in response to irradiation is being investigated.

3. High Strain Rate Effects

J. M. Steichen (A00)

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

High strain rate testing on 304 SS was completed during the past quarter at four of the five strain rates of interest. The completed test rates are 0.10, 1.0, and 10.0 sec^{-1} . Testing at the fifth rate, 100.0 sec^{-1} , will be initiated in the near future since the electro-hydraulic test system which is to be used for this work recently arrived on site.

The 0.01 sec^{-1} rate data is presented in Table 7.1. The results obtained at strain rates of 0.1, 1.0, and 10.0 sec^{-1} are not included since they were reported in the previous quarterly report. The information which has been obtained at the four strain rates is presently being prepared for a report which is to be completed in the next quarter.

TABLE 7.1. High Strain Rate Test Results
on 304 Stainless Steel

Strain Rate, Sec^{-1}	Test Temp., $^{\circ}\text{F}$	Strength, $\times 10^3$ psi		Elongation, % (1 in. gage)		Reduction of Area, %
		Ultimate	Yield	Total	Uniform	
0.01	R. T.	85.10	42.56	55.0	48.1	73.5
0.01	600	62.20	30.09	31.0	24.3	70.4
0.01	800	60.60	27.96	33.7	26.1	71.5
0.01	1000	56.33	26.65	29.7	23.1	70.5
0.01	1200	47.65	21.86	28.4	20.8	70.5
0.01	1400	34.41	18.89	37.7	13.9	65.0

4. Notched Tensile Effects

J. M. Steichen (AOO)

The objective of the notched tensile studies will be to evaluate the effects of a sharp notch on the mechanical strength of weldments typical of those found in the FTR vessel and components. An essential portion of this work will include evaluation of the effect of fast neutron irradiation on the notch strength of these weldments.

5. In-Reactor Creep Measurements

E. R. Gilbert (AOO)

Equations were developed for irradiation-induced creep in solution treated and 20% cold.-worked Type 316 SS. The equations were based on: (1) Hesketh's formulation of irradiation-induced transient and steady-state creep, (2) available in-reactor creep data on austenitic stainless steels, and (3) Russcher's damage function to account for neutron spectral differences in the various experiments. Effective strain and effective stress were used to account for the different states of strain and stress employed in the experiments.

The following equations were developed:

$$\bar{\epsilon}/\bar{\sigma} = 0.2 \times 10^{-8} + 1.87 \times 10^{-30} \alpha \phi t$$

for solution treated Type 316 SS, and

$$\bar{\epsilon}/\bar{\sigma} = 0.8 \times 10^{-8} + 1.93 \times 10^{-30} \alpha \phi t$$

for 20% cold-worked Type 316 SS where $\bar{\epsilon}$ is effective strain, $\bar{\sigma}$ is effective stress in psi, $a = \exp(1.405 - 0.0027 T)$ where T is temperature in degrees Kelvin, ϕ is total neutron flux in $\text{n/cm}^2\text{-sec}$, and t is time in seconds. For uniaxial tension $\bar{\epsilon}/\bar{\sigma} = \epsilon/6 \sigma$ and for pure shear $\bar{\epsilon}/\bar{\sigma} = \gamma/3\tau$.

6. Radiation Effects on Absorbing Materials for Control Rods*

A. L. Pitner (AOO)

Corrected ^{10}B burnup levels, determined by mass spectrometric analysis, have been applied to boron carbide gas release data that were obtained from thermal reactor irradiation of the material. The corrected results are given in Table 7.2. These actual burnup values are higher than those which were calculated from reactor operations data, so the reported gas release fractions are correspondingly lower. The general trends indicated previously still hold:

(1) The amount of gas released increases with increasing irradiation temperature. (2) Gas release in the powders appears to be independent of packing density but in the pellets, release varies inversely with sample density.

TABLE 7.2. Corrected ^{10}B Burnup Levels

<u>Sample</u>	<u>Form</u>	<u>Temperature</u>	<u>^{10}B Burnup</u>	<u>Gas Release</u>
1- 1	60% TD Powder	550 °F	4.68%	15.6%
1- 6	60% TD Powder	700 °F	4.24%	28.3%
1- 7	80% TD Powder	600 °F	4.30%	14.9%
1- 8	80% TD Powder	605 °F	4.35%	
1- 2	80% TD Powder	Leaker	3.67%	
1- 9	80% TD Powder	715 °F	4.38%	25.6%
1-10	65% TD Pellet	865 °F	4.44%	54.1%
1- 3	65% TD Pellet	Leaker	3.76%	
1- 4	90% TD Pellet	725 °F	3.84%	27.7%
1-11	90% TD Pellet	860 °F	4.61%	
1-12	90% TD Pellet	1120 °F	4.64%	27.8%
1-13	99% TD Pellet	865 °F	4.67%	11.6%
1-14	99% TD Pellet	870 °F	4.69%	
1- 5	99% TD Pellet	1035 °F	3.91%	20.9%
1-15	99% TD Pellet	1130 °F	4.71%	

* Reported by D. E. Mahagin (724)

The gas release measured for the 65% TD pellet (Sample 1-10) is unexpectedly high, but could not be checked because the capsule containing the other 65% TD pellet leaked water. Two more subassemblies containing similar samples at higher burnups have recently been discharged and will provide additional information on the behavior of boron carbide at elevated temperatures. One subassembly, operated at temperatures of 500-750 °F, is estimated to have ^{10}B burnups of ~7%, and the other, operated at 550-1200 °F, is expected to have reached burnup levels of >10%.

7. Image Enhancement

B. R. Hayward (723)

The purpose of this technique is to increase the data output from ordinary photographic negatives. A topical report on image enhancement results related to the FFTF Fuel Development Program is in the final stages of completion. This report describes three image enhancement processes that show promise (a) photographic (b) optical and (c) computer. Excellent examples are included that show the capabilities of the processes, however all of the examples are not related to materials technology. Since image enhancement has been primarily a space technology program there has been limited examples in the nuclear materials area. Through the efforts of Battelle-Northwest image enhancement is now being successfully used by EBR-II on neutron radiographs of irradiated fuel pins. It is being used to detect relocations of fuel specimens within the clad as well as to register dimensional changes as small as 0.0005 inches.

Additional work is in progress at BNW to evaluate the capabilities of the optical process. The results are preliminary but appear very encouraging.

CHAPTER VIII. FUELS RECYCLEA. FUEL TECHNOLOGY1. Fuel Vendor Prequalification

H. T. Blair and J. E. Sammis (731)

Three contractors, Babcock and Wilcox, NUMEC, and United Nuclear, are engaged in a fuel vendor prequalification program which is directed toward production of small lots of fuel pellets and fuel pins. During the reporting period the work instructions for inspection of fuel pellets and pins at BNW were completed and approved. These instructions require that analyses be randomized in a manner to eliminate any source bias.

A shipment of 50 prequalification fuel pins was received from one vendor and inspection work has begun. Nondestructive testing equipment in the demonstration facility was used to leak check all 59 pins. A statistical sample of 15 pins was gamma scanned and checked with an air gage for outside profile. The fifteen sample pins were then opened for internal component inspection and sampling for chemical analyses. All 30 end cap welds from the sample pins were submitted for metallographic examination.

2. UO₂ Insulator Pellet Fabrication

W. E. Warden (733)

Dimensional requirements for uranium dioxide pellets for pins for subassemblies PNL-9, 10, and 11 are a diameter of 0.194 ± 0.0015 in. and a length of 0.4 ± 0.1 in. The uranium dioxide is specified as natural enrichment.

Two processes which produced acceptable insulator pellets were (1) 2wt% Carbowax added in a slurry of 20,000 psi preslug and 30,000 psi final press, and (2) 20,000 psig preslug of as-received powder, dry addition of 0.3 wt% Sterotex, and 30,000 psi final press (Table 8.1).

TABLE 8.1. Insulator Pellet Processes

Variable	Diameter Hourglassing, in.	Length, in.	Density	
			% TD	g/cm ³
Preslug +				
0.3 wt% Sterotex	0.004	0.466	93.8	10.3
Preslug +				
2 wt% Carbowax	0.005	0.465	93.3	10.2

The taper and hourglassing of the pellets remained approximately constant over the range of variables examined. Therefore, pellets produced by either of the two acceptable processes will require grinding.

3. Off-Gas Versus Density of Mixed Oxide Pellets

M. J. Barr (731)

Analysis of data from earlier experiments and from the recently completed fractional factorial experiment showed that off-gas quantity and the percentage of theoretical density are inversely related. The model equation determined from computer analysis of limited data within the 75 to 89% range of theoretical density is:

$$Y = B_0 + B_1 X \pm E$$

where

Y = off-gas in cm³/g

B_0 = a constant depending on binder type (0-intercept)

B_1 = a constant depending on binder type (slope)

X = percent theoretical density

E = confidence band related to binder type and factor desired.

An example of values obtained for the constants are as follows:

<u>Constant</u>	<u>Value</u>
B_0	22.795
B_1	-0.253
E (95% level)	~ 0.74

However, additional data from other development experiments indicate that the off-gas-density relationship is not linear, and values from sintered pellets in the 90 to 95% theoretical density range do not fit the curve. In addition, variations in binder addition technique and in density control methods (such as high pressure preslug or presintering) are also expected to affect the equation. Additional data are being assembled in an effort to develop a more general equation.

4. PuO₂ Sinterability Testing

M. J. Barr (731)

Table 8.2 shows the relative sinterability of PuO₂ from the three sources currently being considered for fast reactor fuel. Of the three source powders, calcined oxalate PuO₂ was the most difficult to press as a single oxide. Pressing difficulty was evidenced by pellet diameter variation, pellet ejection pressure, and the increased dwell time required.

TABLE 8.2. Sinterability of Various Source PuO₂ Powders

<u>PuO₂ Source</u>	<u>Sintered Density, % TD (11.46 g/cm³)</u>	<u>Density Range</u>	<u>A Diam Max-Min, in.</u>
Recalcined Oxalate Lot #28-3-7	92.30	91.0 - 94.3	0.005
Burned Metal Lot #26-9-5	97.00	96.5 - 97.5	0.002
Calcined Nitrate 400 °C	92.38	91.7 - 92.8	0.0045
650 °C	91.73	90.7 - 93.0	0.003
800 °C	91.54	90.5 - 92.1	0.0035
1000 °C	91.17	90.4 - 91.8	0.003

5. Fuel Fabrication for Test Pins for Irradiation in EBR-II
W. E. Warden (733)

Fuel fabrication for EBR-II Subassembly PNL-9 is in progress. This fuel consists of 30% enriched ^{235}U in the EBR-II test composition of 75% UO_2 -25% PuO_2 . Four kilograms of blended mixed oxide powder were processed to fuel pellet pressing according to the flow chart (Figure 8.1) which was established after analysis of test data from 3 kg of 65% enriched ^{235}U and 1 kg of 30% enriched ^{235}U fuel. Pressing and sintering of approximately 100 test pellets resulted in less densification than anticipated.

B. CLADDING TECHNOLOGY

1. LMFBR Fuel and Cladding Information Center

J. R. Shoher and B. R. Hayward (723)

The LMFBR Fuel and Cladding Information Center, a data storage, retrieval and analysis system, was designed to contain data for all LMFBR type fuel pins and select LMFBR core materials. It presently consists of five major sections: a cladding fabrication file, a fuel fabrication file, an irradiation history file, a postirradiation examination file, and an experimental pin file. Two major and two minor sections will be added later. These will include a fuel pin fabrication file, a fuel pin subassembly fabrication file, a documentation file, and a specifications file.

Cladding fabrication data are being stored in the system as the data becomes available to the center. Cladding data are being supplied to Battelle-Northwest in three categories:

- Cladding purchased after January 1969.
- Cladding now in test that was purchased before January 1969.
- Cladding purchased before January 1969 and used in tests completed in the interim.

FIGURE 8.1. Process Diagram for PNL-9, 10, and 11 Fuel Pellets

An interim fuel pin file was completed and readied to receive data. Data are being compiled from published sources and transferred to data input forms. Thirty additional tests were added to the austenitic stainless steel mechanical properties file bringing its total to 2447. The creep properties data file now contains data from 247 tests, 148 of which were added this past month. An order for tensile data from Atomics International was partially filled. A total of 180 stress versus strain curves was prepared from 90 experiments and forwarded to them.

2. Evaluation of X-Ray Fluorescence Method for Verification of Alloy Composition

The X-ray fluorescence technique was used to verify the classification of stainless steel samples as either 304 or 316. Known alloy samples were used as controls. Comparison characteristic molybdenum Ka peaks were used as the determining factor. Although the analysis is qualitative, efforts are being made to add a quantitative aspect through more stringent calibration against known standards. This method of cursory material analysis will never replace the more thorough, reliable analytical methods; however it adds another tool to the QA function where rapid screening is required. Figure 8.2 shows how comparisons were made.

3. Eddy Current Cladding Tester

H. G. Powers (740)

Proof of principle was achieved for an eddy current cladding tester pancake coil technique with sequential sampling of the coil signals. Nine pairs of pancake coils were used.

The sequential sampling circuitry (counting, decoding and analog gating circuits) was designed and breadboarded. The entire system was activated. The system functioned according to the desired principle. Future efforts will be made to reduce the transients caused by the switching circuits and to develop a suitable readout device.

Known 316 SS Alloy
Note Mo K α Line (2-3%Mo)

Known 304 SS Alloy
Note absence of Mo K α Line

L-37 Sample

L-38 Sample

FIGURE 8.2. Verification That Samples Were Not 316 SS Alloy
Based on Molybdenum Content and Associated K α Lines

4. Cladding Procurement and Development

J. C. Tverberg, R. J. Lobsinger, and R. C. Aungst (721)

Histograms prepared from the eddy current data obtained on lots of tubing produced commercially by two different processes show the benefit of special handling during the fabrication of tubing. Figure 8.3 is a histogram of tubing produced by a conventional tube drawing process while Figure 8.4 illustrates a histogram produced by a process employing thorough cleaning between every fabrication pass and use of fresh drawing lubricants for each operation.

PROCESS: 20 min BRIGHT ETCH, 2 BAR DRAWS ON GROUND
MANDREL, FINAL PLUG DRAW

NORMALIZED INTEGRATED AREA: 33

RATING: 6

FIGURE 8.3. Histogram of SS Tubing Produced by a Conventional Tube Drawing Process

LOT: 9034901

NORMALIZED INTEGRATED AREA: 13.2

FIGURE 8.4. Histogram of Tubing Produced by the Special Process

C. FUEL PIN TECHNOLOGY

1. Fuel Pin End Closure Welding Development

R. M. Crawford (733)

Part of a study to determine the effect of end cap machining tolerances on the finished weld was completed. Three end cap variations were examined as follows:

- The effect of varying the weld lip thickness between 0.010 and 0.020 in. (see Figure 8.5, dimension A).
- The effect of varying the weld lip diameter between 0.208 and 0.228 in. (see Figure 8.5, dimension B).
- The effect of varying the clearance between end cap and tube from 0.0005 to 0.003 in.

All of the samples in each group (a total of 104 welds) were welded, measured, and radiographed in six positions. All of the samples met established FTR pin weld criteria by radiographic examination. The finished diameters ranged from 0.230 in. (the same as the tube OD) to 0.234 in. Two weld samples from each dimensional variation (42 welds) were examined metallographically to determine the effect of the machining tolerances on the penetration and throat thickness.

All of the welding parameters were held constant during the test in order to get only the effect of the dimensional change on the finished weld. All of the samples examined metallographically were acceptable welds and had at least one wall thickness at any point and no indication of cracks, porosity, or other irregularities. Metallographic examination revealed that the penetration is affected by diameter and thickness of the weld lip. As the weld lip diameter increases from 0.208 to 0.228 in. the weld penetration increases from 0.046 to 0.060 in. and the width of the weld decreases from 0.080 to 0.050 in.

FIGURE 8.5. Reference FFTF Fuel Pin End Closure Weld Penetration

The weld configuration varies from 0.033 in. penetration by 0.063 in. wide when the weld lip is 0.010 in. thick to 0.060 in. penetration by 0.056 in. wide when the weld lip is 0.020 in. thick.

D. FUEL SUBASSEMBLY TECHNOLOGY

1. CCTL Mark I Test Assembly Compressibility Tests

D. E. Blahník and R. B. Baker (732)

The objective of the compression tests was to determine the springiness⁽¹⁾ that exists within a Mark I assembly bundle when compressed symmetrically from a maximum duct size to beyond theoretical tightness.⁽²⁾ The load on the simulated duct wall was to be determined as a function of across-the-flats dimensions.

The CCTL Mark I fuel assembly was subjected to compression tests in which all six sides of the bundle were advanced inward in equal increments (maintaining a regular hexagonal geometry) over a four foot span of the bundle. The bundle was compressed while in the vertical position to minimize the effect of gravity. One side of the test apparatus had a two foot load measurement section so that during bundle compression or decompression the side load could be measured as a function of hexagonal size.

The results followed a pattern that was expected based upon previous data. There is a relatively flat slope up to the point where theoretical tightness of the bundle occurs.

Bundle Springiness - When a tight bundle, such as Mark I, is in the vertical position, the fuel pins cannot bow towards the center. Therefore, most of the outer pins are forced outward by a combination of their own natural bow and the forces transmitted by the natural bow of internal pins. When a bundle is compressed to theoretical tightness all of the natural springiness is dispersed. If a bundle is compressed beyond theoretical tightness, the fuel pins act as a combination of leaf and torsion springs.

2. *Bundle Theoretical Tightness - Theoretical point at which all fuel pins and wires are compressed tightly together without any deformation of the bundle components. There is no outward displacement due to natural bow of the fuel pins and no gaps between wires and fuel pins.*

Then, in a narrow band in the theoretical tightness range, there is a transition into higher spring constants which probably extend well beyond 250 lb/ft/side of the bundle. The test apparatus used was designed for a 250 lb/ft/side load limit, so the compression test runs were stopped at that point.

CHAPTER IX. PHYSICSA. CORE PHYSICS1. Stability Analysis of the Fast Test Reactor

James R. Sheff (831)

The stability analysis of the Fast Test Reactor (FTR) merits discussion because features of the reactor differentiate it from other fast reactors. The most significant differences are the large size, 1033 liters, and high power level, 400 MW_t . The Doppler effect replaces the expansion effects as the dominant mechanism providing the prompt shutdown coefficient. The negative Doppler coefficient is -3.2×10^{-6} per $^{\circ}\text{C}$.

A standard lumped feedback path model involving a linearized set of matrix equations describing the neutronics and feedback mechanisms is used for the analysis. The approach is novel, however, in the description of the reactor via more than a single node so that such effects as the spatial dependent sodium void coefficient could be adequately treated. The sodium void coefficient is positive at the core center but becomes negative in the outer regions of the core.

The resulting feedback function for the FTR reference core, the solid line in Figure 9.1, is seen to be stable since multiplication by the zero power transfer function causes only a rather small angle clockwise rotation in the third quadrant. The measurable range from $\omega = 0.006$ to $\omega = 50$ is shown by the bars across the curve.

Extensive parametric studies have proved the system very stable. The Doppler coefficient was reduced by greater than tenfold in all cases showing any tendency toward instability. The only two cases discovered which were absolutely unstable are shown in Figure 9.1. Along with the zero Doppler coefficients, it was necessary to either let the fuel

FIGURE 9.1. Phase Plane Plot of Feedback Functions. $\omega = 0.006$ and 60 rad/sec are indicated by the dashes across the curves.

expansion coefficient go to zero or introduce a positive expansion coefficient for the fuel assembly tubes.

Conditional instability has also been found in a few cases near those producing absolute instability. Power levels of 100 times full power would be required to produce instability.

An analytic fit to the EBR-II experimental feedback function data ⁽¹⁾ is also shown for comparison. One notes that the behavior is quite similar to the zero Doppler coefficient curves. This similarity might be expected since EBR-II has a very small Doppler coefficient.

In conclusion, the stability analysis of the FTR differs from other fast reactors primarily because of effects produced by its larger size. In addition, the systems studied are stable at least to the extent that the model is accurate. Not included at this time are effects of the system external to the reactor.

2. Notes on the Use of the "Engineering Mockup" as a Nuclear Design Tool

R. A. Bennett, S. L. Engstrom, J. V. Nelson, (833)
and P. L. Hofmann (800)

The use of engineering mockup, critical experiments constitutes an important element in the nuclear design of complicated and novel reactors. The goal of the mockup experiment is to verify the design values of the neutronics characteristics of the reactor to be built. The extent to which the physical characteristics of the reactor can be accurately mocked-up in the critical assembly determines its value as a design tool. If only a coarse mockup can be achieved, the required amount of extrapolation and interpolation increases, and the utility of the mockup experiment is greatly diminished.

1. Reactor Development Program Progress Report, ANL-739, October 1967.

In order to assure a useful engineering mockup configuration, detailed experimental planning, including extensive pre-calculation, is necessary. This calculational activity is presently underway in order to plan the engineering mockup experiments for the FTR.

The facilities available for the FTR engineering mockup experiments are the ANL ZPR assemblies, specifically ZPR-9. The matrix structures of these facilities are essentially square and fixed. The platelet inventories of materials are limited and only a discrete spectrum of compositions are possible. Figure 9.2 illustrates one possible ZPR arrangement to simulate the hexagonal FTR geometry, if only whole material drawers can be used. Each zone in the square lattice mockup, although slightly different in volume from its FTR counterpart, has the same mass of materials.

Two-dimensional (x,y), and hexagonal coordinate calculations were carried out for different reactor conditions. The resulting multiplication constants are compared in Table 9.1. A comparison of different control rod worths is also shown.

For the configuration shown, the multiplication constants are consistently higher for the mockup by a few thousandths. Control strengths in the engineering mockup are consistently underestimated by 5 to 10%. These calculational trends will be used in the planning of the engineering mockup experiments in order to increase their accuracy and consequent utility.

3. Inverse Multiplication Monitoring of Subcritical Reactivity Changes in FTR

S. L. Engstrom, R. A. Bennett, and V. O. Uotinen (833)

It is presently planned that reactivity changes during refueling of FTR and those occurring during initial steps in the approaches to critical will be monitored with a portion of the Low Level Flux Monitoring System. Inverse multiplication

ENGINEERING MOCKUP – FTR OVERLAY

FIGURE 9.2. Engineering Mockup-FTR Overlay

TABLE 9.1. Results of FTR/FTR Engineering Mockup
Comparative Calculations

<u>Configuration Description</u>	<u>Multiplication Constants</u>	
	k_{eff}	
	<u>FTR</u>	<u>Engineering Mockup</u>
1. Reactor shutdown - all rods in	0.8858	0.8971
2. All safety rods out	0.9538	0.9606
3. All safety and in-core rods out	1.0087	1.0124
4. All safety rods out, asymmetric in-core control rod in	0.9893	0.9939
5. All safety, control, and peripheral control rods out	1.0445	1.0465

	<u>Calculation Numbers</u>	<u>Worth(%$\Delta k/k$)</u>	
		<u>Reactivity FTR</u>	<u>Engineering Mockup</u>
All safety rods (3)	1 and 2	8.05	7.37
All in-core control rods (4)	2 and 3	5.70	5.20
One in-core control rod	3 and 4	1.94	1.84
All peripheral control rods (15)	3 and 5	3.40	3.21
Total control strength	1 and 5	17.15	15.91
Total in-core control strength	1 and 3	13.75	12.57

techniques are to be used. One can very reasonably expect that changes in fuel loadings of drivers or test loops and control motions near any of the elements of the monitoring system will seriously perturb their responses and ultimately produce results that must be corrected.

This problem has been investigated experimentally in ZPR-III Assembly 56B⁽¹⁾ by Argonne National Laboratory as a part of the Battelle-Northwest FTR Critical Experiments Program. Control rods were inserted at the boundary of the core of the assembly and the inverse multiplication rate was monitored with sensors located at various positions, including the core center and outside the ZPR-III matrix approximately 100 cm from the core centerline and approximately 60 cm from the nearest control zone. The areal profile and counter positions are shown in Figure 9.3. Values of the reactivity worths of rods differed by as much as 12% for the various sensors.

In an attempt to identify which counter was "best" a series of subcritical calculations were performed with a two dimensional diffusion theory model of the various experimental configurations. The distributed external source of neutrons due to a-n reaction in ^{240}Pu was represented to yield absolute flux levels. The actual calculated changes in the subcritical multiplication constants were compared with values inferred from changes in ^{235}U and ^{10}B reaction rates throughout the core. The results are presented in Table 9.2 for the specific experimental sensor locations and are shown graphically in Figure 9.3 for all possible sensor locations. The tabulated values show that sensors located at the core center are "best". The calculated lines of "equal over-estimation," shown in Figure 9.3, illustrate that any other position is worse.

1. ANL Reactor Development Program Progress Report, ANL-7577, April-May 1969.

FIGURE 9.3. Assembly 56B Quarter Core Areal Profile. Lines show the % over-estimation of the worth of four large B_4C rods.

TABLE 9.2. Calculated Reactivity Worths Deduced from Calculated Rates of Various Detectors

Sensor Location	Sensor	Four Small B ₄ C Rods (a)			Four Large B ₄ C Rods		
		Experimental Worth (IH)	Calculated Worth (IH)	R ₁ (b)	Experimental Worth (IH)	Calculated Worth (IH)	R ₂ (b)
Center of Reactor	Reaction Rate in 10B	96 ±10	1005	100 0	1804 ±21	1923	1.003
	Fission Rate in ²³⁵ U		1003	0 998		1915	0.999
	φ Total		1002	0 997		1909	0.996
Standard ZPR-3	Reaction Rate in 10B	1000 ±10	1056	1 051	2021 ±21	2255	1.176
	Fission Rate in ²³⁵ U		1052	1 047		2215	1.155
	φ Total		1045	1 040		2153	1.123

- (a) Small rods were one-fourth the size of the range rods and occupied the quadrant of large rod closest to the core center.
- (b) R is the ratio of the worth deduced from calculated reaction rates to the worth as obtained from the calculated eigenvalues

Comparisons of calculated values with the "best" experimental results show an additional 5% disagreement or c/e ratios of approximately 1.05 for both small and large rods.

4. Central Fuel and Peripheral Control Ring Reactivity Worths in the FTR-2 Critical

R. A. Bennett, and J. V. Nelson (833)

Experimental measurements have been made of reactivity worths of segments of a peripheral control ring of B_4C -steel-Na and large fuel assembly-size plutonium samples from the core of a full scale simulation of the FTR. These experiments were performed by Argonne National Laboratory personnel in ZPPR Assembly-1, the FTR-2 critical in the FFTF Critical Experiments Program.

An areal profile of FTR-2 is shown in Figure 9.4 which illustrates the arrangement of the core, central depleted zone, peripheral control ring, and reflector. Additional detailed descriptions of the experimental arrangements have been reported.⁽¹⁾ Reactivity measurements were made in alternating steps in which fissile plutonium was removed from the core and replaced by fertile uranium. These reactivity decreases were approximately compensated for by the removal of segments of the control ring, which in turn was replaced by reflector composition. The sequence of the experiments is indicated by the numerical scheme shown in Figure 9.4, e.g., fissile material was replaced first in Zone 1 and control material was replaced second in Zone 2. The experimental results and corresponding diffusion theory values were presented in Table 9.3.

Comparisons of the experimental data and calculated values indicate that the neutronics design requirements of the FTR

1. ANL Reactor Development Program Progress Report, ANL-7632, October 1969.

TABLE 9.3 Experimental Results and Corresponding Diffusion Theory Values

Experiment Number	Core Fuel Mass (a) kg	Experimentals		Calculation		Comparison K _E - K _C	Reactivity Worths of Fissile Exchanges				Reactivity Worths of Reflector Exchanges						
		Control (b) Drawers In Place	Ex ss React i %Δk	Multiplication Constants, K _E	Multiplication Constants, K _C		Experimental, %Δk/k(E)		Calculated, %Δk/k(E)		Experimental, %Δk/k(E)		Calculated, %Δk/k(E)				
							Col. 3	Col. 4	Col. 5	Col. 6	Col. 7	Col. 8	Col. 9	Col. 10	Col. 11	Col. 12	Col. 13
0	527.041		240	+0.08±0.001		1.00086		0.98729	0.01357								
1	525.043		240	-0.480±0.030		0.99522		0.98118	0.01404		-0.566±0.030	-0.631	1.114				
2	525.043		224	-0.152±0.005		0.99841		0.98369	0.01472					+0.321±0.030	0.260	0.810	
3	519.050		224	-1.61B		0.98408		0.96677	0.01731		-1.459	-1.779	1.219				
4	519.050		176	-0.800±0.070		0.99206		Asymmetric	-					+0.818	Asymmetric	-	
5	519.050		144	-0.22±0.003		0.99774		Asymmetric	-					+0.574±0.070	Asymmetric	-	
6	515.054		144	-1.360±0.200		0.98658		Asymmetric	-		-1.134±0.200	Asymmetric	-	/	0.753±0.200	Asymmetric	-
7	515.054		112	-0.607±0.027		0.99397		0.97370	0.02027					+0.487±0.027	0.459	0.942	
8	515.054		96	-0.123±0.001		0.99880		0.97807	0.02073								
9	507.064		96	-1.690±0.250		0.98358		0.95959	0.02379		-1.570±0.250	1.967	1.253				
10	507.064		64	-0.840±0.061		0.99158		0.96723	0.02435					+0.841±0.260	0.823	0.978	
11	503.068		64	-1.600±0.250		0.98425		0.95866	0.02550		-0.751±0.260	0.924	1.230				
12	503.068		32	-0.300±0.006		0.99700		0.97037	0.02665					+1.299±0.250	1.259	0.969	
13	495.077		32	-1.520±0.180		0.98503		0.95452	0.03051		-1.219±0.180	1.711	1.404				
14	495.077		0	-0.300±0.007		0.99695		0.96671	0.03024					+1.214±0.180	1.321	1.088	

Mean square deviation = +29%
from unity

Standard deviation = +11%
from unity

Mean square deviation = +29% from unity

Standard deviation = +11% from unity

(a) Fissile material ^{235}U + ^{241}Pu + ^{239}Pu .

(b) Two drawers per matrix location.

should reflect approximately $\pm 10\%$ uncertainty (cf Col.13) in the calculated values of peripheral control rod strengths; and approximately a 30% over-calculation (cf Col. 10) of the reactivity worths of centrally located changes in fissile plutonium densities arising from changes in the compositions of test loops or fuel burnup. These analyses also indicate that the discrepancy between calculated and measured multiplication constants, $(K^E - K^C)$ (cf Col. 7) increases linearly with the mass of the plutonium test sample. One may infer from this dependence that the troublesome fractional over-calculation of reactivity worths of plutonium samples is approximately constant for sample sizes from zero to 32 kg, the limit of these experiments.

B. RADIATION AND SHIELDING

1. Effect of Cobalt Content in Steel on Shield Requirements C. A. Mansius (813)

It is anticipated that a number of radioactive components from the Fast Test Reactor will require replacement at some time during the lifetime of the facility. Replacement of large components such as the core support structure will probably require fabrication of a shielded container. The purpose of this study was to determine the effect of the cobalt content of the stainless steel used in the fabrication of the core support structure on the shielding requirements during the removal process.

Only long-lived radioisotopes will influence the shield requirements, since replacement of the core support structure would require an extended shutdown period. At the location of the support structure, the neutron spectrum is relatively degraded in energy and (n,γ) reactions are dominant. Based on half life, decay energy, and production rate, ^{60}Co and ^{59}Fe will be the predominant radioisotopes emitted by the support structure

The half life of ^{60}Co is 5.2 years and it emits both a 1.17 and 1.33 MeV photon per decay. The half life of ^{59}Fe is 45 days and it emits a 1.10 MeV photon in 56% of the decays and a 1.29 MeV photon in 44% of the decays. Because of the comparable photon energies from the two isotopes, the relative gamma intensity will be essentially independent of the shield thickness.

The relative radiation intensity from ^{60}Co and ^{59}Fe depends on the cobalt content of the steel together with the operating and decay history. Figure 9.5 gives the relative intensity of the two isotopes as a function of operating time and cobalt content. It is assumed the reactor operates at 400 MW for 75 days and is then shutdown for 25 days. For this assumed cycle, ^{59}Fe attains 87% of its saturated value whereas ^{60}Co would reach a maximum of about 60% of its saturated value (continuous cyclic operation). Figure 9.6 presents the decay curves for the two isotopes.

Calculations indicate that about 5 to 6 in. of lead will be required to reduce the radiation intensity from the core support structure to a level of about 25 mrem/hr at the surface of the cask, based on the maximum relative ^{59}Fe activity shown in Figure 9.5. About 0.4 in. of lead is required to effect a factor of 2 change in the radiation intensity for either the cobalt or iron radiations. The data in the two figures can be used together with this basic shield information to estimate the required shield thickness as a function of operating time, cobalt content, and shutdown time. In most cases of interest it is found that shielding requirements will depend directly on the cobalt content of the steel.

2. Radiation Levels in Heat Transport Cell

W. L. Bunch, C. A. Mansius and D. R. Marr (813)

Access into the FFTF heat transport system (HTS) cells is desirable to permit inspection and maintenance. Estimates were

FIGURE 9.5. Relative ^{59}Fe - ^{60}Co Gamma Intensity

made of anticipated radiation levels within the HTS cells as a function of various operating conditions to provide information for the conceptual design. The basic conditions considered are: (1) entry into the cell following shutdown of the reactor, and (2) reducing the reactor power and valving off the single HTS cell of interest to permit decay of ^{24}Na , with entry into the cell following shutdown of the reactor.

After 12 days delay time following shutdown of the reactor to permit decay of ^{24}Na , it is estimated the average radiation level in the cell will be about 30 mrem/hr because of the long-lived (2.58 yr half life) ^{22}Na in the system. The radiation level could be significantly higher as a result of corrosion and fission products in the system. However, it will take significant operating time for transport of corrosion products to the cells, and fission product contamination will depend on the fuel failure history that is experienced.

By operating the reactor at reduced power while one HTS cell is valved off, it would be possible to reduce or eliminate the 12 day wait to permit ^{24}Na decay. Table 9.4 indicates the operating-shutdown intervals that would be employed as a function of valve leakage to maximize plant efficiency. It is assumed there would be no incentive to continue to operate at reduced power once the radiation level due to ^{24}Na had achieved the equilibrium value for the specific leakage rate.

To estimate the radiation level in the HTS cell as a function of valve leakage, it was assumed the point of interest within the cell was equidistant from all of the sodium within the cell. The activity of the sodium at any point in the system was related to valve leakage or flow rate based on time since leaving the high pressure inlet plenum. The most important single factor becomes the decay time in transit from the inlet plenum to the HTS cell. It is obvious that within the cell

significant radiation gradients will exist at low flow (leakage) rates. Methods are available to include the exponential variation in source strength from straight pipe sections; however, it was not deemed necessary to include such refinement in this study. Detailed calculations can be made for specific component arrangements and specific points of interest.

TABLE 9.4. Operating and Shutdown Times to Maximize Plant Efficiency as a Function of Valve Leakage*

Valve Leakage			Days at 2/3 Power	²⁴ Na	Days Down	²⁴ Na
%	gpm	gph		Intensity, R/hr		Intensity, R/hr
0.01	1.5	90	2	3000	10	0.03
0.0025	0.38	23	4	300	8	0.03
0.001	0.15	9	6	30	6	0.03
0.0005	0.075	4.5	8	3	4	0.03
0.0002	0.03	1.8	12	0.03	0	0.03
0	0	0	12	0.03	0	0.03

* Assumes 75,000 gallons in one HTS cell, 15,000 gpm full flow rate

3. Neutron Attenuation Characteristics of Stainless Steels D. R. Marr (813)

Summary

The decision to replace Type 304 SS with Type 316 SS in the duct material as well as in the cladding indicated a need to examine the relevancy of the existing neutron shielding calculations which are based on the use of Type 304 SS. This study provided the following results:

- Type 316 SS is a slightly more effective shielding material than Type 304 SS. Largest differences (up to a factor of 2) are seen in the lower energy reactions.

- Previously reported results based on the use of Type 304 SS are thus conservative.
- The higher shielding effectiveness of Type 316 SS is due to molybdenum, which is present only in small percentage amounts, and which has a much higher absorption cross-section than any of the other components of stainless steel.
- The shielding characteristics of stainless steel are not sensitive to the chromium-nickel content within the ranges established for Type 316 SS and Type 304 SS.
- Most of the difference in neutron attenuation between Type 316 SS and Type 304 SS is effected in transmission through the reflector and about half way into the shield.

Discussion

The compositions of Type 316 SS and Type 304 SS⁽¹⁾ are shown in Table 9.5 together with the composition assumed in this study. Type 316 SS differs from Type 304 SS in higher nickel content offset by a corresponding decrease in the chromium content and by the inclusion of about 2.5% molybdenum, which replaces some of the iron. Type 316 SS also has a slightly higher density.

For the purpose of this comparison, a one-dimensional spherical geometry model based on the radial configuration of concept V-A was selected. This configuration is shown in Table 9.6. The core barrel dimension was arbitrarily thickened to 1 ft to study the attenuation in pure stainless steel.

1. J. G. Yevick. East Reactor Technology: Plant Design. M.I.T. Press, Cambridge, Massachusetts. p. 483.

TABLE 9.5. Weight Percent composition of Type 316 and Type 304 Stainless Steels

	Type 304 SS (7.90 g/cm ³)		Type 316 SS (7.98 g/cm ³)	
	Yevick ⁽¹⁾	This Study	Yevick	This Study
Fe	64.92-70.92	68.92	61.92-68.92	66.42
C	<0.08	0.0	<0.08	0.0
Cr	18.00-20.00	19.00	16.00-18.00	17.00
Ni	8.00-12.00	10.00	10.00-14.00	12.00
Mn	<2.00	1.50	<2.00	1.50
Si	<1.00	0.50	<1.00	0.50
Co	<0.20	0.08	<0.20	0.08
Mo	~0	0.0	2.00-3.00	2.50

TABLE 9.6. Model Configuration and Composition

Zone	Outside Radius	Material Composition (Vol%)				
		Na	SS	Fuel	Void	Ni
Core I	38.6 cm	38.89	29.40	25.91	3.53	2.27
Core II	60.2	38.89	28.06	26.26	3.58	3.12
Reflector I	72.14	35.00	37.20	-----	----	27.80
Reflector II	96.0	33.00	33.70	-----	----	32.30
Shield	144.8	34.00	66.00	-----	----	-----
Core Barrel	179.3	-----	100.00	-----	----	-----

Twelve integral quantities of possible shielding interest were calculated using the converged fluxes from the discrete ordinates transport code ANISN. An S_g calculation was made with ANISN using the HOMSET cross sections on each of the four following cases:

- Case I (304) - All stainless used in this case was Type 304 SS as described above.

- Case II (316) - All stainless used in this case was Type 316 SS as described above.
- Case III (316A) - All stainless used was modified Type 316 SS having a modified chromium and nickel content (10% Ni, 19% Cr).
- Case IV (316B) - All stainless used was Type 316 SS, but without the molybdenum.

The results of this study are shown in Table 9.7 where, taking Type 316 SS as a base case, the effect of replacing it with each of the other materials is shown at four selected points.

Several items should be noted:

- There is little effect in the core.
- The 316A case shows there is little effect due to changing the chromium-nickel ratio over the range that can be expected to encompass variations within each type steel or between the two steels.
- Having excluded an appreciable effect due to the change in chromium-nickel ratio, case 316B (316 with no Mo) shows that the primary difference due to replacement of Type 304 SS by Type 316 SS is due to the molybdenum content of the Type 316 SS. A close check of the **HOMSET** absorption cross sections for molybdenum shows values which are slightly high with respect to the latest experimental data, a bias not thought to grossly affect the results of this study.
- Reactions with a significant high energy contribution such as ^{238}U (n,f) or biological dose do not show a large effect due to Type 316 SS replacement.
- For those quantities such as ^{59}Co (n, γ), which have an elevated low energy response, there is a point approximately midway in the shield where subsequent attenuation is relatively independent of the steel used.

TABLE 9.7 Ratio of Reaction Rate to Reaction Rate in Type 316 SS Case

Reaction Rate	Core r = 1.28 cm			Reflector r = 70.1 cm			Shield r = 115.8 cm			Core Barrel r = 145.8 cm		
	304	316A	316B	304	316A	316B	304	316A	316B	304	316A	316B
$^{239}\text{Pu}(n, f)$	1.00	1.00	1.01	1.21	1.00	1.21	1.70	1.01	1.71	1.93	1.04	1.93
$\text{Na}(n, \gamma)$	1.00	1.00	1.01	1.14	1.00	1.14	1.49	1.02	1.49	1.69	1.04	1.69
$^{235}\text{U}(n, f)$	1.00	1.00	1.01	1.19	1.00	1.19	1.66	1.01	1.66	1.86	1.04	1.86
$^{238}\text{U}(n, f_0)$	0.99	1.00	1.00	1.01	1.01	1.02	1.03	1.02	1.08	1.04	1.02	1.14
Biological Dose	0.99	1.00	1.00	1.01	1.00	1.01	1.06	1.02	1.08	1.12	1.03	1.16
$^{59}\text{Co}(n, \gamma)$	1.01	1.00	1.01	1.26	1.00	1.26	1.82	1.01	1.82	2.09	1.04	2.10
$^{55}\text{Mn}(n, \gamma)$	1.02	0.99	1.03	1.10	1.00	1.10	1.35	1.01	1.34	1.49	1.04	1.49
$^{58}\text{Fe}(n, \gamma)$	1.00	1.00	1.01	1.08	1.00	1.08	1.33	1.02	1.33	1.46	1.05	1.46
$\text{p}(n, \alpha)$	1.01	1.00	1.01	1.24	1.00	1.24	1.77	1.01	1.78	2.02	1.04	2.02
Total Flux	1.00	1.00	1.00	1.04	1.01	1.04	1.20	1.02	1.20	1.31	1.05	1.31
Flux Above 0.8 MeV	0.99	1.00	1.00	1.01	1.01	1.01	1.02	1.01	1.07	1.04	1.02	1.13
Flux Between 0.07- 0.08 MeV	1.00	1.00	1.00	1.01	1.01	1.01	1.05	1.02	1.06	1.09	1.03	1.12

4. ZPPR/FTR-2 Shield Experiments

E. T. Boulette (813)

Experiments in support of the design of the Fast Test Reactor (FTR) shields were conducted at Argonne National Laboratory in the Zero Power Plutonium Reactor (ZPPR). The experimental arrangement approximated the basic features of the FTR: a 1000-liter, two-zone core surrounded by a 1 ft thick nickel reflector and a stainless steel-sodium shield. Although the FTR shield is surrounded by an annular sodium zone in which fuel and other reactor components may be stored, the ZPPR facility is not large enough to incorporate this feature. Provisions were therefore made to include a stored fuel zone in the experimental shield. Neutron and gamma distributions were measured throughout the core, reflector, and shield, with and without stored fuel in position.

Neutron Distribution

The neutron distribution was obtained by measuring the reaction rate traverses of ^{239}Pu and ^{238}U with fission counters. and of ^{10}B with a BF_3 counter. In addition, Na and Mn foils were dispersed throughout the system and irradiated. These foils were selected not only because of their different response functions, but also because of the impact these reaction rates have on the FTR design.

The two-dimensional diffusion theory code 2DBS⁽¹⁾, and the one-dimensional discrete ordinates code ANISN⁽²⁾ were used to calculate the reaction rate distributions for comparison to the

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1. D. R. Marr. A User's Manual for 2DBS, A Diffusion Theory Shielding Code, BNWL-1291. Battelle-Northwest, Richland, Washington. February 1970.
 2. W. W. Engle. A User's Manual for ANISN, A One-Dimensional Discrete Ordinate Transport Code with Anisotropic Scattering, K-1693. Union Carbide Corp., Oak Ridge, Tennessee. 1967.

experimental results. The nuclear data used was extracted from the HOMSET⁽¹⁾ 31-group transport-corrected cross section set. The measured and calculated ^{10}B reaction rate distribution with no stored fuel in the shield are compared in Figure 9.7. Similar agreement was found for the other reaction rate distributions.

Gamma Distribution

The gamma distribution was measured by irradiating lithium fluoride thermoluminescent dosimeters (TLD's), which were dispersed radially and axially throughout the core, reflector and shield. Dosimeters were irradiated to a suitable integrated exposure, one set with and one set without stored fuel in the shield. Light output of each TLD was measured subsequently and compared to calibrated standards.

The discrete ordinates transport theory codes ANISN and DOT⁽²⁾ were employed to calculate the gamma distribution for comparison to the experimental results. A ten-group gamma transport cross section set was generated employing the MUG⁽³⁾ code. The energy range of the calculation extended from 10 MeV down to 0.015 MeV with equal (one MeV) group widths except for the last group. The volume distributed

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1. D. R. Marr and M. G. Zimmerman. FTR Shield Design Cross Sections--A Partial Evaluation, BNWL-1197. Battelle-Northwest, Richland, Washington. 1969.
 2. F. R. Mynatt. Unpublished Data. Union Carbide Corp., Oak Ridge, Tennessee. [User's Manual for DOT, A Two-Dimensional Discrete Ordinate Transport Code with Anisotropic Scattering. (to be published).]
 3. J. R. Knight and F. R. Mynatt. Unpublished Data. Union Carbide Corporation, Oak Ridge, Tennessee. [MUG - A Program for Generating Multigroup Photon Cross Sections, (to be published)]

FIGURE 9.7. A Comparison of Calculated and Measured $^{10}\text{B}(n,\alpha)$ Radial Reaction Rate Distribution in the ZPPR/FTR 2

gamma source was calculated from the neutron flux solution and included prompt fission gammas, fission product gammas, and neutron capture gammas. A ^{235}U prompt fission gamma spectrum⁽¹⁾ was employed in the calculation; however, evidence suggests this spectrum to be a realistic approximation for ^{239}Pu fission also. The fission product gamma spectrum was generated using the computer code ISOSHL⁽²⁾ based on a fission product inventory obtained from the code RIBD⁽³⁾. The neutron capture gamma ray spectra were extracted from data on thermal neutron capture reported in the literature.

A comparison of the calculated and measured radial gamma dose distribution is shown in Figure 9.8, based on a S_8-P_3 ANISN cylindrical calculation with no stored fuel present. The calculated ten-group gamma fluxes were converted to roentgens⁽⁴⁾ and normalized to the integrated experimental exposure. Similar agreement was found at the other elevations.

Overall, the agreement between the calculated and experimental results is believed adequate to permit use of these techniques and nuclear data compilations in the preliminary design of the FTR shields. Basis of the relatively consistent difference between the experimental and calculated gamma results is being sought.

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1. R. W. Peele, W. Zobel, and F. C. Maienschein. "Spectrum of Prompt Gamma Rays from Thermal Fission of ^{235}U ," Trans. Am. Nucl. Soc., vol. 12, p. 384. 1969.
 2. C. A. Mansius. A Revised Photon Probability Library for Use with ISOSHL-III, BNWL-236, Supplement 2. Battelle-Northwest, Richland, Washington, April 1969.
 3. R. O. Gumprecht. Mathematical Basis of Computer Code RIBD, DUN-4136. Douglas United Nuclear, Inc., Richland, Washington, 1968.
 4. Reactor Physics Constants, ANL-5800. Argonne National Laboratory, Argonne, Illinois, July 1963. Second Edition.

FIGURE 9.8. A Comparison of Calculated and Measured Radial Gamma Dose Distribution in the ZPPR/FTR 2

CHAPTER X. SAFETYA. SAFETY ANALYSES1. Sodium Fire Studies Involving Outer Containment

P. R. Shire (821)

A series of sodium fire studies have been updated for an outer containment vessel 135 ft in diameter with a volume of 1.3 million ft³. In each case the outer containment vessel was conservatively assumed to be completely closed with no leakage; the initial condition was air at atmospheric pressure and ambient temperature at 80 °F.

The following accident conditions were investigated:

- Closed loop tube rupture in reactor top head box open (33 ft²) to outer containment vessel.
- Same as above, except 1000 ft² vent opening represents head box cover completely removed.
- Sodium pipe rupture in heat transient cell open (100 ft²) to outer containment vessel.
- Fuel handling machine spray release, adiabatic conditions.

A second accident condition was found to represent the most severe problems. Data from the study will be employed in preparation of the Plant Safety Analysis Report (PSAR).

2. FFTF Containment Analysis - CACECO Code

R. D. Peak (821)

The digital computer code, CACECO, has been prepared and used in the safety analysis of the FFTF containment design. This code predicts the transient temperatures and pressures in the reactor cavity, reactor head box, HTS cells, and containment building following any accident that breaches the primary sodium system. The CACECO code (~~cavity-cells-containment~~ transient code) is an expanded version of the CONRAD code

described in BNWL-1174. The new code has the capability of computing containment transients in four containment volumes connected by vent or leakage paths. The code also has restart capability so that long time transients can be computed in stages.

The CACECO code maintains accurate material and energy inventories of the nitrogen-oxygen-sodium vapor-fission product atmosphere and of the sodium pool in each containment volume and thereby determines the total pressure in the volume, the pool temperature, and the mixed atmosphere temperature. The code provides for bulk boiling of the sodium pool as well as for bulk condensation of the sodium vapor from the atmosphere, each depending upon the effects of heat sources and sinks in the volume, the sodium vapor-oxygen chemical reaction, and atmospheric leakage flows between volumes.

The leakage analysis in the code uses a predictor-corrector technique. In this analysis, the pressure differentials between the several containment volumes and the differential between the building volume and outside cause nitrogen, oxygen (when present), sodium vapor (when present), and fission product (when present) flows between the volumes and to the outside which tend to reduce the pressure differentials. This analysis is based upon the turbulent flow equation and accounts for atmospheric inventory changes due to flows of material and energy. The areas for leakage between the reactor cavity and HTS cells, the six pipeways with 34.4 ft² total opening in the present design, and between the cavity and head box (an assumed 10 ft² opening) allow huge flows at very small pressure differentials. For example, the leakage between the cavity and cells with only 0.01 psi differential would be about 1250 lb/min when the cavity inventory is only 2140 lb. Any pressure differential is rapidly dissipated by such flaws.

The CACECO code is being applied to the safety analysis of the FFTF containment design. For these cases, the FFTF containment design was modeled with four volumes and 39 heat structures as summarized below.

TABLE 10.1. FFTF Containment Model for the CACECO Code

Reactor Cavity:	volume	31,200 ft ³
	number of heat structures	15
	surface area of insulated structures	7,030 ft ²
	surface area of steel structures	4,520 ft ²
HIS Cells:	volume	142,800 ft ³
	number of heat structures	13
	surface area of insulated structures	8,748 ft ²
	surface area of steel structures	21,940
Reactor Head Box:	volume	10,200 ft ³
	number of heat structures	7
	surface area of steel structures	4,320 ft ²
Containment Building:	volume	1,118,000 ft ³
	number of heat structures	4
	surface area of steel structures	53,640 ft ²
	surface area of concrete structures	6,420 ft ²

Although most of the heat structures act as heat sinks for the sensible heat, reactor accident energy, sodium-oxygen chemical reaction heat, and decay heat energies, the steel structures, the liners in the cavity, cells, and head box and other exposed steel, are the initial sinks because of their high internal thermal conductivities. The concrete structures are the next effective sinks and the insulated structures are

the last effective sinks. In all recent cases, the atmosphere-to-structure heat transfer has been based upon a temperature-dependent heat transfer coefficient based on natural convection of nitrogen. Also, this coefficient is enhanced by local condensation when sodium vapor is present in the atmosphere.

Data from these studies are being employed in preparation of the PSAR.

3. Post DBA Containment Transients - SOHOT Code

G. L. Fox (821)

A digital computer code, SOHOT, has been prepared for computation of long term pressure and temperature transients following a hypothetical core meltdown. This code differs from the previously described CACECO code in that it does not model the rapidly occurring events immediately following a DBA, and also differs in the level of detail in the heat transfer structure.

The SOHOT code models the type of hypothetical accident where fuel is assumed to melt through the bottom of the reactor vessel and guard vessel and release both sodium and molten fuel to the floor of the reactor vault. The fuel produces fission decay heat from solid and gaseous products which will heat the sodium and the cavity until sodium boiling occurs. The sodium vapor transfers the decay heat from the reactor cavity to the connecting equipment cells and raises the temperature and consequently its pressure. The pressure buildup is relieved by gas leakage from the reactor cavity and equipment cells to the containment sphere. The containment sphere gas is heated by the sodium-oxygen reaction and fission gas products which raises its pressure. The code estimates temperature and pressure in the various subdivisions of the containment system. The mathematical modeling is described below:

a. Fission Product Decay Heat

The total fission product decay heat is input as a table of power versus time. A specified fraction of this power (20% for present work) is assumed to be a gaseous fraction which diffuses throughout the reactor cavity and equipment cells. The remaining solid fraction is assumed to stay in the reactor cavity where it heats up the walls and sodium.

b. Containment Geometry

The physical layout for the system which is being modeled is shown in Figure 10.1. There is free passage of gas between the reactor cavity and the equipment cells. The leakage rates, L2 and L3 are specified as percent flow rate of gas from the cell under accident conditions. The other points listed on Figure 10.1, starting with the letter T, represent temperature points. For the present studies, the concrete wall was considered 5 ft thick in all locations.

c. Heat Transport Equations

This model now requires a list of assumptions. The notation will be simplified by identifying the reactor cavity as cell 1, equipment cells as cell 2, and containment sphere as cell 3.

- The sodium vapor content in cells 1 and 2 is determined by equilibrium conditions at the cell gas temperature.
- The time periods are divided into three phases:
 - (1) Initial phase where sodium is being heated to boiling point.
 - (2) Boil phase where boiling occurs.
 - (3) Dry phase where no sodium remains in the reactor cavity.
- Sodium boiling occurs when all nitrogen has been transported from cell 1 to cell 2.

FIGURE 10.1. Containment System Schematic for SOHOT

- Pressure in cell 1 and cell 2 are equalized by open ductwork.

Since the pressure is equal in cells 1 and 2, the equation of state which governs the system is

$$P1A + PS1 = P2A + PS2 \quad (1)$$

Let 1 designate the particular cell being considered:

$P1A$ = Cell 1 - partial pressure of nitrogen

$PS1$ = Cell 1 - partial pressure of sodium.

This expression may be differentiated with respect to time and, using the gas law, provides an equation of state which relates temperature and pressure changes within the cells.

The equation of state coupled with an energy balance and gas leakage equations for the cells provides sufficient equations to solve for temperature and pressure history for all three cells.

The sodium transported from cell 1 to 2 during the "Initial" phase is coupled to the mass of nitrogen transport. This is represented by:

$$\frac{dMS}{dt} = \frac{dM1}{dt} \frac{PS1}{P1A} \frac{R1}{RS}$$

where $M1$ = Nitrogen mass in cell 1

MS = Sodium vapor transport from cell 1 to cell 2

$R1$ = Gas constant for nitrogen

RS = Gas constant for sodium.

During the "Boil" phase, the quantity of sodium transported results from an energy balance since there is no further nitrogen transport.

The quantity of heat lost from cell 1 by sodium transport is represented by:

$$(H1SG - H1SL) \frac{dMS}{dt}$$

where $H1SG$ = Enthalpy of sodium vapor at cell 1 temperature

$H1SL$ = Enthalpy of liquid sodium at cell 1 temperature.

The quantity of heat gained by cell 2 by sodium transport is represented by:

$$(H1SG - HZSL) \frac{dMS}{dt}$$

where H2SL = enthalpy of liquid sodium at cell 2 temperature.

d. Heat Transport to Walls

The heat is removed from the cells by quickly accessible and slowly accessible heat sinks in addition to conduction to the cell liner and into the concrete wall. The concrete wall is modeled by six nodal elements of constant thermal properties to produce the transient temperature distribution.

Data from these studies are being employed in preparation of the PSAR.

4. A New Computational System for Fast Reactor Accident Investigation

Alan E. Waltar (831)

Major fast reactor excursion calculations are characteristically laden with numerous parametric studies due to uncertainties in the state of the reactor during the extreme conditions hypothesized. An improvement over earlier methods (1,2) has been developed which permits elimination of a substantial degree of arbitrariness in the range of parametrics by providing a consistent basis for establishing the reactivity ramp rate, effective Doppler coefficient, and appropriate equation of state during the disassembly process.

1. D. E. Simpson, J. W. Hagan, A. E. Waltar, R. A. Harris, A. Padilla, and G. L. Fox. Preliminary Analysis of Postulated Maximum Accidents for the FFTF, BNWL-760. Battelle-Northwest, Richland, Washington, November 1968.
2. D. R. MacFarlane, N. A. McNeal, T. J. Heames, W. T. Sha, and C. K. Youngdahl. "Reactivity Transients Leading to Disassembly in Oxide-Fueled Fast Reactors", Trans. Am. Nucl. Soc., vol. 12, p. 342. 1969.

Basically, the method consists of a coupling of the multi-channel neutronics-heat transfer computer program, MELT-II⁽¹⁾ with the two-dimensional disassembly computer program VENUS.⁽²⁾ The MELT-II phase includes feedback effects due to Doppler broadening, sodium voiding, and fuel relocation whereas the VENUS phase includes Doppler broadening and the reactivity feedback due to core disassembly. The accident sequence is followed in MELT-II until fuel temperatures are elevated to the point where sizeable disassembly pressures are attained. At this time the fine structure temperature information is volume averaged over the R-Z spatial mesh structure used in VENUS, and this input, along with the power distribution and all reactivity feedback components, is used directly as the starting point for the disassembly phase in VENUS. In addition, the core is scanned for liquid sodium content to determine the effective Doppler coefficient and to establish whether a sodium-in or a sodium-out equation of state should be employed at each particular VENUS mesh point.

An example of the type of problem which can be treated by this coupled system is the highly asymmetric situation arising by passing a large sodium bubble through the core. The particular example presented is based on a bubble 162 liters in volume passing through an FTR-like core (radius ~ 60 cm, height ~ 90 cm). Reactivity, power response, and energy yield resulting from upward movement of the bubble through the lower reflector and into the core are illustrated in Figure 10.2. Disassembly conditions are established when

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1. *Alan E. Waltar, Andrew Padilla, Jr., and Richard J. Shields. [MELT-II, A Two-Dimensional Neutronics-Heat Transfer Code for Fast Reactor Safety Analysis, Battelle-Northwest, Richland, Washington. (to be published)]*
 2. *W. T. Sha and T. H. Hughes. Unpublished Data. Argonne National Laboratory, [VENUS - Two-Dimensional Neutronic-Hydrodynamic Fast Reactor Power Excursion Computer Program. (to be published)]*

FIGURE 10.2. Transient Response for a Large Sodium Bubble Passing Through a Fast Reactor Core.

the center of the bubble reaches approximately the lower third of the reactor. The ramp rate is nearly 80 \$/sec and the effective Doppler coefficient has dropped from an initial value of -0.005 to approximately -0.0038. The amount of energy contained in the molten fuel at the end of the nuclear excursion is calculated to be 1120 MW-sec.

Tight coupling between the predisassembly and disassembly phase is essential to provide a realistic appraisal in energy release, particularly for an accident of this type because of the core inhomogeneity (variation of sodium content) involved. For example, if analysis of the above accident is attempted using only a disassembly code, parametrics including a range of ramp rates from 50 to 100 \$/sec, Doppler coefficients from -0.003 to -0.005, and equations of state varying from essentially full sodium-in to full sodium-out are necessitated to bound the problem. Such calculations have been performed and the energy contained in molten fuel ranges from 181 to 2407 MW-sec.

Acknowledgement

The author is indebted to Dr. William T. Sha, Argonne National Laboratory, for his keen interest and collaboration in this study.

5. On the Treatment of Spatial Feedback Effects in Fast Reactor Accident Analyses

Alan E. Waltar and Winston W. Little, Jr. (831)

Numerous papers^(1,2,3) have recently appeared which point out the need for using a space-time kinetics package when analyzing fast reactor transients. Although the desirability of such techniques cannot be argued in an absolute sense, they introduce some very practical limitations in terms of both programming and computation time, particularly for two or three dimensional computations. The purpose of this work is to emphasize that the discrepancy between lumped parameter and space dependent models is frequently caused by differences in the important feedback terms, and that these differences can be greatly reduced by using a lumped nuclear model coupled with a space dependent feedback model.

An example of an attempt to account for spatial feedback effects and yet retain the simplicity and speed associated with point kinetics⁽⁴⁾ is the treatment of Doppler feedback in the MELT-II code. Assuming T_{dT}^{dk} remains constant over the temperature range of interest for an oxide fueled core, Doppler feedback is computed according to the following expression:

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1. G. Kessler. "Space-Dependent Dynamic Behavior of Fast Reactors Using the Time-Discontinuous Synthesis Method," Trans. Am. Nucl. Soc., vol. 11, p. 569. 1968.
 2. James F. Jackson and William E. Kastenberg. "Space-Time Dynamic Studies in Large LMFBR's with Feedback," Trans. Am. Nucl. Soc., vol 12, p. 705. 1969.
 3. D. A. Meneley and K. O. Ott. "Spectral Effects in Liquid-Metal Fast Breeder Reactor Transients," Trans. Am. Nucl. Soc., vol. 12, p. 706. 1969.
 4. Alan E. Waltar, Andrew Padilla, Jr., and Richard J. Shields. [MELT-II, A Two-Dimensional Neutronics-Heat Transfer Code for Fast Reactor Safety Analysis, Battelle-Northwest, Richland, Washington. (to be published)]

$$k_{\text{Doppler}} = \frac{K_D \sum_{i,j} W(i,j) V(i,j) \ln \frac{T(i,j)}{T_o(i,j)}}{\text{TOTAL FUEL VOLUME}} \quad \text{for } T_o \leq T \leq T_1, \quad (1)$$

$$A + \frac{K_D \sum_{i,j} F W(i,j) V(i,j) \ln \frac{T(i,j)}{T_1(i,j)}}{\text{TOTAL FUEL VOLUME}} \quad \text{for } T_1 < T$$

where K_D = original whole core Doppler coefficient

$W(i,j)$ = local Doppler weighting factor

$V(i,j)$ = local nodal fuel volume

$T(i,j)$ = local nodal fuel temperature

$T_o(i,j)$ = original local nodal fuel temperature

$T_1(i,j)$ = local nodal fuel temperature at the time local voiding occurs

F = effective normalized change in the Doppler coefficient for spectrum hardening due to sodium voiding

A = feedback value evaluated at $T = T_1$.

Figure 10.3 illustrates the change in the effective Doppler coefficient (D_{eff}), calculated by dividing the change in the Doppler feedback of Equation 1 by the natural logarithm of the core average fuel temperature change over the same time interval. The particular example is a guillotine break in one inlet pipe of an FTR-like core (radius ~ 60 cm, height ~ 90 cm). The overall drop in D_{eff} up to the time of initial coolant voiding results from the skewed fuel temperature buildup towards the top, lower worth region of the core. Channel 1 (central subassembly) is assumed to begin voiding at the core due to cladding failure and fission gas release. Initial reactivity effects are negative, and the attendant drop in power accentuates the relative temperature peaking at

FIGURE 10.3. Change in the Effective Doppler Coefficient During a Pipe Rupture Accident

the core periphery. As the void moves downward into the core, however, the reactivity effects are positive and the power surge peaks temperatures toward the central regions. Hence, D_{eff} increases. This phenomenon continues although the effects of spectrum hardening resulting from coolant voiding (the dashed line) become dominant and D_{eff} has dropped from -0.005 to approximately -0.0035 at the time the core nears a disassembly condition. For this type of accident, Doppler feedback is shown to be fairly small until the latter stages of the accident. Even so, however, the inclusion of a spatial Doppler feedback according to Equation 1 results in a substantially higher power and integrated energy ($\sim 15\%$) relative to the lumped model which uses a Doppler feedback based on the core average temperature.

B. OTHER SAFETY TECHNOLOGY

1. Use of Delay Beds for Radioactive Gas Decay Storage

C. J. Foley (925)

Contaminated gases must be stored until radioactive decay processes permit release at levels allowed by Federal regulations. A delay bed utilizing a sorbent material is being considered for FFTF. A delay bed is a volume of sorbent material such as charcoal packed into a container such as piping. Contaminated argon gas passes into the bed where xenon and krypton are sorbed more strongly than the argon carrier; this, in effect, delays the passage of the xenons and kryptons while the argon passes through relatively quickly.

Such a delay bed would be preferable to tanks for several reasons. The delay bed envisioned would operate at ambient pressure and temperature and require only modest flows of cooling water and no gas compressor. A bed that will delay passage of noble gases for 87 days has been preliminarily sized.

The bed envelope is approximately 7 ft in diameter by 10 ft long and has a pressure drop of approximately 2.5 in. of water through the sorbent for a gas flow of 0.6 scfm. In contrast, pressurized tanks to perform the same function would consist of three pressure vessels each 10 ft in diameter by about 20 ft long. These vessels would store the contaminated cover gas at about 10 atm pressure. We believe delay beds offer significant advantages over pressure tanks because the space requirements for the delay bed are less than the pressure vessels and because the radioactive gases are stored in the sorbent material without significant pressure buildup. Thus leakage of radioactive gases from a delay bed is very unlikely.

APPENDIX A

ORGANIZATION CODES FOR FFTF PERIODIC TECHNICAL REPORTS

(Effective 1-1-1970)

<u>Code</u>	<u>Author's Organization</u>
A00	BNW Chemistry & Metallurgy Division
C00	BNW Physics & Engineering Division
K00	BNW Environmental & Life Sciences Division
M00	BNW Systems & Electronics Division
N00	FFTF Division, B. Wolfe, J. C. Cochran
TCD	BNW Technical Communications Department
390	FFTF Administration Department - H. E. Little
391	Contract Negotiations - J. C. Richardson
392	Procurement - R. J. Gandy
393	Data & Support Services - J. R. Boldt
394	Reports & Publications - J. F. Erben
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400	FFTF Quality Assurance Dept. - R. J. Squires
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600	FFTF Construction Department - J. S. McMahon
610	Construction Engineering - C. E. Love
620	Engr. Administration & Control - H. E. Hylbak
630	Construction Engr. & Inspection
700	FFTF Fuels Dept. - E. A. Evans, W. E. Roake, T. W. Evans
710	Offsite Fuel Programs - G. A. Last
711	LMFBR Fuel Dev. - B. R. Hayward
712	FFTF Fuel Procurement
720	Cladding Development - T. T. Claudson
721	Cladding Dev. & Procurement - J. C. Tverberg
722	Cladding Evaluation - J. J. Holmes
723	Fuel & Cladding Info. Center
730	Fuel Element Dev. - C. A. Burgess

	731	Process Dev. & Demonstration - R. E. Bardsley
	732	Subassembly Dev. - J. W. Thornton
	733	Special Products Fab. - E. T. Weber
740		Fuels Qual. Assurance - H. G. Powers
750		Fuels Evaluation - J. E. Hanson
760		Fuels Cycle Analysis - A. W. Demerschman
800		FFTF Reactor & Plant Technology Dept. - P. L. Hofmann, D. E. Simpson
810		Reactor & Plant Engr. - L. M. Finch
	811	Reactor Engineering - D. Marinos
	812	Plant Engineering - D. P. Schively
	813	Radiation & Shield Analysis - W. L. Bunch
820		Safety & Systems Analysis - R. E. Peterson
	821	Safety Analysis - D. D. Stepnewski
	822	Systems Analysis - H. G. Johnson
	823	Control Analysis - R. A. Harvey
	824	Rad. & Environ. Safety
	825	Safety Tech. & Practices
830		Nuclear & Process Tech. - R. E. Heineman
	831	Nuclear Analysis - W. W. Little
	832	Core Analysis - P. D. Cohn
	833	Exper. Physics - R. A. Bennett
	834	Operations - D. C. Boyd
	835	Process Analysis
900		FFTF Engineering Department - D. L. Condotta, F. C. Gronemeyer, W. B. McDonald
910		Reactor Design - S. O. Arneson
	911	Specialists
	912	Reactor Components - R. C. Walker
	913	Core Engineering - J. F. Wett
	914	Equipment Engineering - C. A. Munro
920		Fluid Systems - J. M. Batch
	921	Specialists

	922	Sodium Technology - W. R. Wykoff
	923	Heat Removal System - T. W. Withers
	924	Test Systems - P. F. Shaw
	925	Auxiliary Systems - R. V. Dulin
930		Instrumentation & Control - C. D. Swanson
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970		Engineering Q.A. & Standards - J. A. Perry

APPENDIX B

FFTF BNWL REPORTS ISSUED - DECEMBER 1, 1969 - FEBRUARY 28, 1970

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