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FAST FLUX TEST FACILITY
MONTHLY INFORMAL
TECHNICAL PROGRESS REPORT
MARCH 1969

AEC RESEARCH & DEVELOPMENT REPORT

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BATTELLE MEMORIAL INSTITUTE
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UNITED STATES ATOMIC ENERGY COMMISSION UNDER CONTRACT AT(45-1)-1830

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BNWL-1043

FAST FLUX TEST FACILITY
MONTHLY INFORMAL TECHNICAL PROGRESS REPORT

MARCH 1969

E. R. Astley
Project Manager

C. P. Cabell

by C. P. Cabell

SUBMITTED TO AEC APR 16'69

April 7, 1969

BATTELLE MEMORIAL INSTITUTE
PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON 99352

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FFTF MONTHLY INFORMAL TECHNICAL PROGRESS REPORT
MARCH 1969

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 March 1969.

INTRODUCTION

This Informal Technical Progress Report is divided into five chapters. Chapter I presents a brief summarization of FFTF overall program progress during the reporting period. Chapter II ("Systems") describes the conceptual features of the FFTF design and summarizes design management progress. Chapter III (General Technology) and Chapter IV (Development) describe work done by the Reactor and Plant Technology Department and the Development Department, respectively, in support of FFTF design. Chapter V describes progress in the FFTF Fuels Program.

The various chapters of the report reflect, as appropriate, results of progress made by offsite supporting organizations.

I. OVERALL FFTF PROGRESS

This chapter summarizes overall FFTF progress made during March 1969. To facilitate readership and interpretation, the chapter has been arranged by the compilers into 10 sections as follows:

1) Program Management and FFTF Operations
2) Plant Design
3) Components
4) Instrumentation and Control
5) Sodium Technology
6) Core
7) Fuels and Materials
8) Fuel Recycle
9) Physics
10) Safety.

These sections are not intended to represent FFTF organizations.

The summary writeups given in this chapter carry references to report sections where additional details are given. For example, the notation "(IIA)" refers to Chapter II, Section A, of this report.

1. PROGRAM MANAGEMENT AND FFTF OPERATIONS

The program for completion of the CSDD's based on the FFTF vertical core reference concept is being vigorously pursued, and separate status reports are being issued on this subject.

The BNW preliminary design work packages for Bechtel were revised and transmitted to RDT and Bechtel. RDT approval of the work packages was received, and preparation of preliminary design work plans by Bechtel is scheduled to be completed by early April.

All practicable steps are being taken to reduce FFTF construction costs by use of available excess equipment. A large 115 to 138 kV power transformer is being held (IIA) at the Savannah River Plant while its possible use for FFTF is being investigated.

The Sodium Facilities Building (BAP-027) will be used to house several nonradioactive sodium loops and other test equipment. Lump sum bids to construct the 60 ft by 100 ft by 24 ft high building were opened on March 12, 1969. The apparent low bid was within $1,000 of the Government estimate and indicates that the project will be constructed for the amount authorized.

FFTF operating plans (IIIB) are proceeding; definition of an operator training program is now in progress.

2. PLANT DESIGN

A. Heat Removal

Investigation of improved air dump heat exchanger simulation techniques for transient analysis continued with the completion of a reference multinode cross-flow model using the TAP computer code. (IIIC)

Studies showed that advanced reactor cores can be protected against a double-ended pipe rupture of a reactor vessel inlet nozzle by the use of three 10-inch inlet downcomers for each primary loop. (IIIC)

Pipe crack failure data are being reviewed and correlated to establish a definition of a "intermediate size" leak for the FFTF Heat Transport System. This definition will be applied in evaluating the severity of such an occurrence and the corresponding effect upon engineered safeguards. (IIID)

B. Fuel Handling

Radiation fields associated with unirradiated FTR fuel assemblies have been calculated to be sufficient to influence the design of the unirradiated fuel handling system. (IIIB)

3. COMPONENTS

A. Vessel and Shield

Westinghouse work on preparation of the RFP for the reactor vessel and shield (IIA) is proceeding satisfactorily. Babcock and Wilcox study results are being issued as report No. BAW-1280-59.

Agreements with RDT on the interim design code for the vessel will permit use of austenitic stainless steel in accordance with present code rules if the irradiation damage to the vessel material does not reduce the predicted end-of-life ductility below that represented by a total tensile elongation of 10%. (IVC)

B. Sodium Pumps

The Westinghouse primary pump design study is proceeding according to the new schedule based on the change of conditions from the cold leg to the hot leg location. (IVA)

C. Control Drive Mechanisms

Testing of the modified Fermi latch mechanisms continued. This mechanism is submerged in water for these initial tests. Examination of the latch fingers after 1651 cycles indicated that the mating surfaces of the fingers and spool were not contacting properly. The cam stroke is being modified for further tests. (IVA)

D. Heat Dumps

The final copy of the Test Request for Testing of Commercially Available Fin Tubing (for FFTF Heat Dumps), Test T-2, was transmitted to LMEC on March 4. The LMEC Test Plan is expected to be prepared and transmitted to BNW/FFTF in about 30 days. (IVA)

4. INSTRUMENTATION AND CONTROLS

An extraneous neutron source is not required for subcritical and startup monitoring of the FTR; however, such a source could be used to verify the operability of the low level neutron monitoring equipment. (IIIB)

Tests of a Consolidated Electrodynamics Corporation electromechanical transducer at 575 °F have been completed and the test temperature has been increased to 600 °F. The Barton-Statham hybrid NaK capillary system continues to operate successfully after 6 months of testing. (IVB)

5. SODIUM TECHNOLOGY

Considerable insight into the transport and deposition of activated corrosion products is being obtained from high resolution gamma scanning of the dismantled radioisotope transport loop. Results indicate that preferential deposition of activity in preferred locations may possibly be designed into the heat transport system. (IVB)

Improvement in the performance of the mass spectrometer being evaluated for analysis of the cover gas was obtained by using a ratio measurement technique. (IVB)

6. CORE

A. Reactor Control

Evaluation of the significance of various scram system parameters continued. The effect of the normal position of the safety reactivity-control elements with respect to the core top face has been investigated and results are given in terms of the effective time delay between release and insertion of 10% of the system worth. (IIID)

Parametric nuclear transient analyses are in progress; data are given on the magnitude of reactivity insertion which is within the negative feedbacks of the core. (IIID)

Parametric analyses indicate a tight-packed (P/D ≤ 1.1) 61-pin bundle is a preferred control rod design. The potential increase in achievable rod worth may permit a significant reduction in the number of rods required. (IIIC)

The feasibility of available locations for flux monitoring is still under study. (IIIA) Optimum positions that will present a neutron field sufficient to measure and still be far enough from the core to minimize gamma interference are in the general vicinity of the core restraint mechanisms.

Hot-pressed boron carbide pellets with nominal densities of 75, 85, and 95% TD have been subjected to sodium (oxygen content 3 ppm) in 100-hr static tests at 300 and 500 °C. Postmortem examination of the pellets revealed that very little damage occurred as a result of the exposure. (IVC)

B. Fuel Design

(1) Fuel Pin Spacers

Efforts are being accelerated (IIA) to reach a decision on the FFTF Fuel Pin Spacer Design. It has been agreed that the tight wire wrap method will be given prime emphasis, but Westinghouse work will also continue on a grid-spacer method.

Analysis of hard point support fuel pin spacer grids indicates unacceptably high forces between support dimples and fuel cladding. (IIIC)

(2) Fuel Subassembly Duct

It has been agreed by Battelle-Northwest and Westinghouse (IIA) that a one-piece type 304 austenitic stainless steel duct is the recommended reference design.

Welding parameters for pipes of typical fuel duct sizes are being developed to aid in the design of adaptors for the automatic welding of hexagonal duct sections. (IVC)

Three fabrication processes for the application of wear pads to the fuel ducts are being investigated. Of the three processes, the welding attachment of preformed wear pads appears to offer more promise than either weld-overlaying or flame-spraying the coatings. (IVC)

C. Core Power and Thermal Design

Methods for achieving a total core power of 400 MW_t while maintaining a goal peak flux of 0.7×10^{16} n/cm^2-sec are under investigation. (IIIA)

The method for computing hot channel factors has been defined and engineering factors have been established. (IIA)

7. FUELS AND MATERIALS

Tests on as-fabricated weld metal and heat-affected-zone specimens confirm that the ductility of the weld metal can be substantially lower than that of the base metal adjacent to the welds. This was found to be true for welds made by either the submerged-arc or the manual stick electrode process. (IVC)

Production of 4,000 mixed oxide pellets for the Analytical Standards Program has progressed to the sintering stage. All major steps in the production of mixed oxide pellets (with the exceptions of the PuO_2 calcining and binder removal) were accomplished within the new Fuel Fabrication Demonstration Facility.

The first programmed sintering run on No. 2 sintering furnace was completed on March 3. The furnace and all associated systems performed well. (VA)

Thirty-seven pins for EBR-II irradiation subassembly PNL-4 were shipped to the reactor. (VA) In the Quality Assurance program a method to measure nitride nitrogen in mixed oxides based upon the conversion of nitrides in a sample of ammonia by caustic fusion, was set up. (VB) In the Damage Analysis program a section of the safety rod thimble, which had been irradiated in the EBR-II at 730 °F to the peak fluence of 8.1×10^{22} n/cm^2 (E > 0.1 MeV) was examined. Preliminary analysis indicates a void density of about 8×10^{15} /cm^3 and a lineal averaged void diameter of 170 Å. This value is in good agreement with an apparent increase of some 3.00% in the volume of the same sample. (VC)

BNW was requested to examine further the 7-rod cluster cladding which has been exposed to 9,000 hr of flow in 1,060 °F sodium to determine if preferential surface damage could be detected in the cladding directly under the wire wrap. Typical surface damage is shown by penetration along the grain boundary in all of the photomicrographs developed, but the damage is neither better nor worse at the wire wrap location. (VD)

8. FUEL RECYCLE

No report.

9. PHYSICS

Plans for the second series of physics experiments in a larger critical facility are essentially complete with concurrence between Battelle-Northwest and Argonne National Laboratory. (IIIB)

Code ETOX has been modified (IIIB) for improved calculation of inelastic scattering transfer matrices.

ZPR III-566 critical assembly has been cylindricized. Average atom densities are given. (IIIB)

Evaluation of the effect of sodium density on neutron attenuations (IIIB) has shown that this factor is not a major parameter in shield design.

10. SAFETY

The energy release for accidents which are terminated by disassembly of the core have been determined for various initial fuel temperatures (prior to mixing with the sodium in the core) and for various expansion modes of the molten fuel-sodium mixture. (IIIC)

Through use of the fault tree technique, six events have been identified which could, during the assumed existence of highly improbable conditions including protective system failure, lead to core disassembly. These events are: (1) core clampage failure, (2) spectrum shift, (3) stuck poison rod, (4) fuel charging at criticality, (5) dropped fuel assembly, and (6) pump startup accident. (IIID)

A list was compiled of current FFTF fault trees and additional fault trees required. The fault tree for radioactive release was revised to incorporate events not previously included. A new fault tree was started for reactor unavailability. A program to be used with the CalComp digital plotter for automated fault tree drawing is now usable and a specimen is presented. (IIID)

II. SYSTEMS

W. B. McDonald

A. REACTOR PLANT DESIGN

S. O. Arneson

1. Reactor

R. C. Walker

a. Reactor Core

J. F. Wett

BNW-W discussions during meetings conducted at Richland on March 4-6, resulted in a number of conclusions and agreements which pertain to the reactor core and its conceptual design description. Conclusions of the meetings are highlighted as follows:

(1) Fuel Pin Spacer.

- A grid spacer design which grips the fuel rod at three rigid dimples is unsatisfactory. Westinghouse proposes a spacer grid operating on the "twig" theory as a basis for an acceptable design. (The "twig" theory describes a system in which the fuel pins are positioned by random contact along their length. The primary compliance in this support system is supplied by the fuel pin. An auxiliary compliant element ensures pin location within the grid.)
- A grid spacer design decision should be made prior to fabrication of a flow test assembly which is to be initiated by May 15, 1969.
- BNW and Westinghouse agreed that the tight wire bundle should be viewed as the prime contender and efforts

focused on establishing an agreed BNW-W burnup capability of a tight wire bundle by May 15, 1969. This burnup value and its uncertainty would then be assessed for overall design and performance acceptability. It was also agreed that within the context of the above work, efforts on grid design and analysis, and loose wire bundle analysis would proceed so that estimates of performance of these spacer systems would also be available by May 15, to be considered for assessment of alternatives.

- BNW-W agreed that the reference material for the fuel subassembly duct should be Type 304 austenitic stainless steel based primarily on fabrication considerations.

(2) Fuel Duct. An agreement on the duct design was reached between Battelle-Northwest and Westinghouse to use the BNW duct design for the reference core.[1]

(3) Core Power. Several items are currently being investigated as methods for improving core power and flux:

(a) Developing better $\int kdt$ data

(b) Examining 0.050 in. pin spacing

(c) Replacing 3 in-core control positions with 3 driver fuel positions

(d) Adjusting core ΔT, T_{in} and T_{out} to improve flux

(e) Evaluating reduction of the over-power factor (OPF).

(f) Investigating the effects of:

1) Holding the present configuration, but improving the calculational methods.

2) Changing conditions and limits, such as reducing coolant outlet temperature, cladding hot spot, core ΔP, coolant ΔT and OP factor.

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1. *For further details see Section III.A.1.b.*

3) Changing configuration, of such items as flow baffles, wire wrap sizes, number of control rods, cladding thickness, and tighter core pack.

(g) Orificing scheme and core power map.

(h) Determining a better value for the hot channel factor. A mathematically and physically precise method for application and combination of the hot channel factor utilization was agreed upon by BNW and Westinghouse. The combination of these factors include the individual effects of each factor on the fuel $\int kdt$ and configuration (i.e., the sintering factor). The engineering factors agreed upon are given in Table 2.A.I.

b. Reactor Vessel and Shield

R. C. Walker and O. W. Priebe

Preliminary design work by Westinghouse on the preparation of the Request for Proposal (RFP) for the Reactor and Guard Vessels is progressing satisfactorily but slightly behind schedule. The schedules established in the initial work plan are being revised and the date now planned for issuing the RFP is established as May 2. General agreement has been reached on manpower requirements for the work of preliminary design effort through the remainder of this fiscal year. Review Board meetings were held during the month on submitted sections of the RFP Package. Comments on these sections have been made and presented to W.

The Conceptual Component Design Description for System No. 32, Reactor Vessel and Shield, has been reviewed by W and RDT. These comments have been received and applicable information is included in the document. The CCDD has received approval of the Technical Evaluation and Configuration Control Boards and is being prepared for submittal on the scheduled date of April 1 for RDT approval.

TABLE 2.A.I. Engineering Factors (Radial Factor = 1.40)

Item	Region: Coolant	Film	Cladding	Gap	Fuel
A. Direct					
1. Inlet Flow Maldistribution	1.05	1.02	-	-	-
2. Intra-Subassembly Flow Maldistribution	1.15	1.06	-	-	-
3. Inter-Channel Coolant Mixing	0.99	-	-	-	-
4. Power Control Band	1.02[a]	1.02[a]	1.02[a]	1.02[a]	1.02[a]
5. Wire Wrap Temperature Peaking	-	2.00[b]	-	-	-
Product-Fuel: Overpower	1.195	2.162	1.00	1.00	1.00
Product-Fuel: Steady State	1.219	2.206	1.02	1.02	1.02
B. Statistical					
6. Fissile Fuel Maldistribution	±2.3%	±3.3%	±3.3%	±3.3%	±3.3%
7. Power Level Measurement	9.00	9.00	9.00	9.00	9.00
8. Burned Fuel Reload	1.2	1.2	1.2	1.2	1.2
9. Rod Diameter Pitch and Bow[c]	2.0[b]	0.9	-	-	-
10. Film and Gap Conductivity	-	3.3	-	40	-
11. Fuel and Cladding Conductivity and Thickness	-	-	12	-	10
	13.54	34.40	15.41	41.15	13.90
(Product-Direct and Statistical)					
Fuel: Overpower	1.357	2.906	1.154	1.412	1.139
Fuel: Steady State	2.965	2.965	1.177	1.440	1.162

a. *Goes to 1.0 at Overpower*
b. *For cladding temperature only; is less for full temperature*
c. *Bow not considered yet.*

Babcock and Wilcox conceptual study work performed during this fiscal year is being issued in topical report No. BAW-1280-59.

2. Heat Transport and Irradiation Testing

J. H. Westsik

a. Irradiation Test Facilities

M. A. Vogel

(1) System 61, 61-P - Closed Loop System.

(a) Project Management (M. K. Mahaffey and R. R. Wyer). The AI work plan for CSDD review was received and approved by BNW. The AI review effort is to concentrate in the following areas:

- Suitability of component operation to 1200 °F.
- Maintainability of primary loop components.
- Failure modes.
- System performance.

Scheduled completion date for the AI review is June 30, 1969.

(b) Components (IT-7) (R. R. Derusseau). A Survey of components included in the closed loops and their purification systems has been completed. Agreements were reached between BNW and Westinghouse on the individual components that needed development and/or testing and the degree of each program necessary to ensure reliable components for the reactor closed loops.

(2) System 68 - Short-Term Irradiation Facility (STIF) (R. R. Derusseau). The draft of the Conceptual Facility Design Description was completed and distributed for comments. The first draft is presently being updated to reflect these comments and those of the technical evaluation board. The test specimen requirements have been changed from baseline to reference to allow more flexibility in the way the in-reactor tube is to penetrate the reactor cover and the way the transfer duct is routed. Test specimen size dictates the duct curvature and, therefore, the routing plan.

The design safety criteria document for STIF has been approved by its technical evaluation board and has been incorporated into the baseline section of the CFDD.

3. Fuel Handling and Radioactive Maintenance

W. E. Cawley

a. Reactor Refueling

E. J. Ruane

(1) CSDD Preparation (E. J. Ruane). Work was completed at AI and BNW on the typed draft copy of the CSDD. Preliminary draft copies were distributed to RDT, BNW Safety Section, BNW Operations Section and Westinghouse for comments. All applicable comments received to date were incorporated into the document which has been issued within BNW for review and comment prior to the Technical Evaluation Board Meeting on March 31, 1969.

b. FFTF Gas-Cooled Loop-Handling Machine Study

C. L. Peckinpaugh

A statement of work was issued in March to Atomics International to develop a gas-cooled, loop-handling machine. The design would embody the capability of achieving direct gas cooling of closed and open loop test assemblies, and indirect gas cooling of closed-loop, in-reactor tubes. The purpose of this study is to determine the impact on testing capability and plant design in the event that the reference sodium-cooled, loop-handling machine proves to be undesirable or unachievable. The work plan, schedule, and cost estimates are scheduled to be transmitted by the end of March.

c. Test Train Length and Standardization Studies

C. L. Peckinpaugh

A statement of work was issued to AI to determine which parts of the FFTF test train could be shortened and the manner in which configuration and characteristics could be standardized. The study is to determine practicality, limitations and incentives for shortening test trains and to determine if the inventory of spare parts can be reduced. The work plan, schedule, and cost estimate are scheduled to be transmitted to BNW by early April.

d. Nonirradiated Fuel Handling

E. D. Grazzini and M. A. Fischer

Work was completed on the typed draft copy of the CSDD's for the Nonirradiated and Irradiated Fuel Handling Systems. Draft copies of the documents were to be distributed to Aerojet, Westinghouse, RDT and BNW for comments prior to the Technical Evaluation Board Meetings scheduled for early April.

e. Radioactive Maintenance

C. R. Nash

AGC, AI and BNW are preparing Functional Analysis Sheets and a Functional Flow Diagram for the CSDD. The scope of work for preparing the CSDD was agreed upon by all principals and work is progressing on the first draft copy due for a BNW Technical Evaluation Board review in April.

4. Reactor Plant Instrumentation

E. M. Johnston and C. D. Swanson

a. Reactor Instrumentation and Control

R. R. Cone

(1) Reactor and Vessel Instrumentation System (W. Dalos and L. E. Fort).

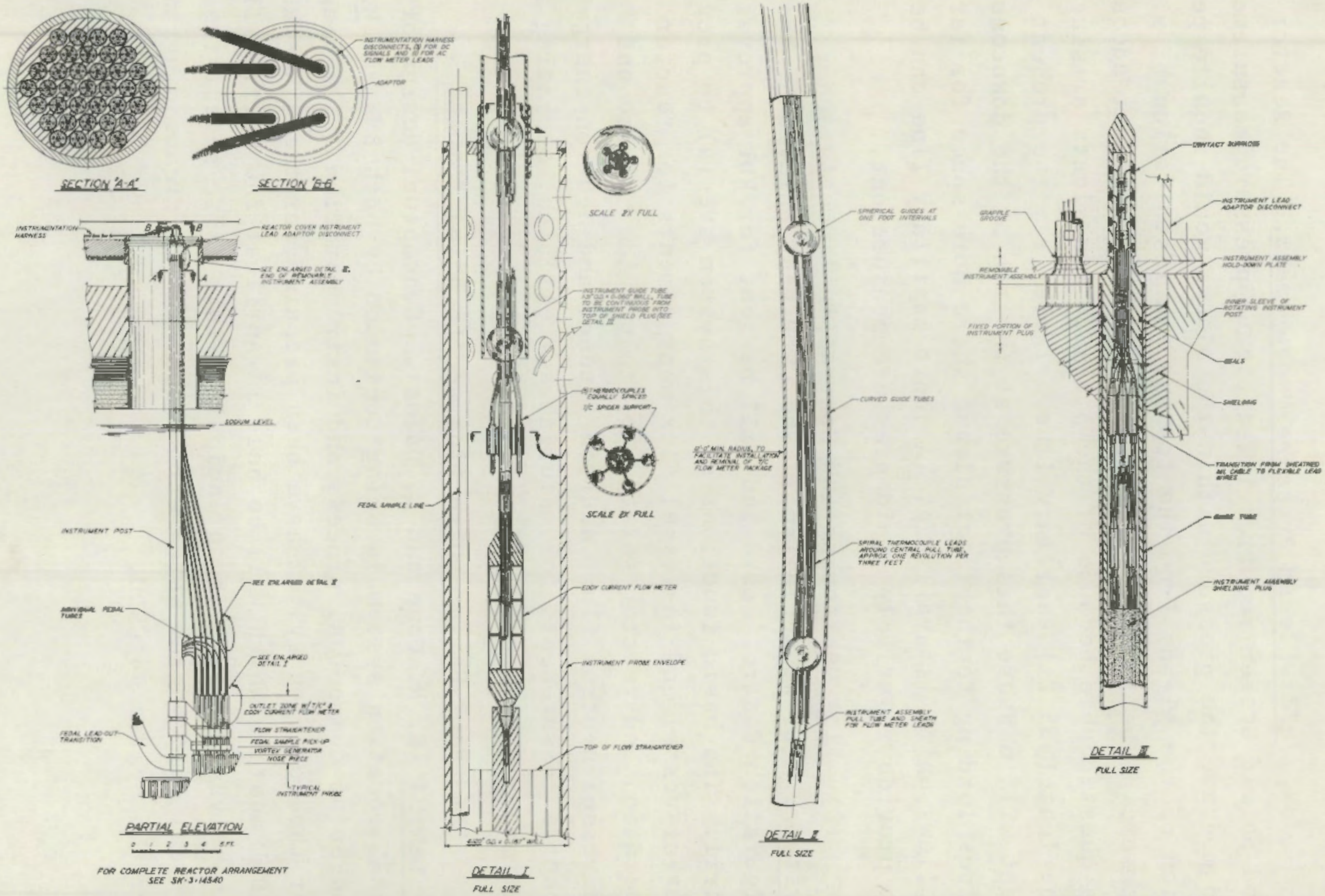
(a) System 92 CSDD. The CSDD has been revised to satisfy RDT requests for additional detail, and to reflect the changes required by the adoption of the Reference Concept. The revised

CSDD will include 1) a table of measurements to be made, their end use and a partial listing of instrumentation performance requirements, 2) an expanded glossary and 3) design safety criteria requirements.

(b) *Driver Channel Inputs to PPS.* Tentatively, driver channel thermocouple signals will not be provided for the Plant Protection System (System 99) circuits. A BNW study has been initiated to determine whether driver channel thermocouples and driver and open-loop channel flowmeters should provide input to PPS to initiate scram or whether they should be used to initiate other action such as controlled power reduction. The study is to be completed in June.

(c) *Safety Rod Instrumentation.* An instrument probe similar to the type used to measure driver position flow and temperature was selected for these measurements on the control and safety rods. The instrument assembly will be removable from the instrument tree in the same manner as the driver fuel instrument assembly. The flowmeter and thermocouples are offset in the control rod channels to provide adequate clearance for the hanger rod as shown in Figure 2.A.1 (SK-3-12896, second issue). Thermal conductance from the coolant in the driver channels to the slower flowing coolant in the control and safety rod channels requires that thermocouples for the control/safety rod channels be located as close to the duct outlet as mechanically possible. The conceptual design for this instrumentation is now being developed.

(d) *Development Task IC-11, Pressure dP Flow and Level Sensors for In-Reactor Sodium Service.* This task has been re-directed to develop induction coil level probes instead of small pressure transducers. Little justification can be found for the development of improved pressure transducers since commercially available pressure transducers appear to meet FFTF requirements except as described in (e).



SK-3-12896

FIGURE 2.A.1. Removable Instrument Assembly Driver Fuel Position

(e) *Inlet Plenum Pressure Measurement*. The Reactor Vessel and Shield System requires inlet plenum pressure measurement. To measure the pressure within the plenum would require penetration of the plenum from the top and the installation of a commercial bellows-NaK capillary-transducer unit but adequacy is questionable because of the pressure transients arising from the long (∿35 ft) capillary line. Therefore, the present concept will measure this pressure at the top of the downcomer pipes leading to the inlet plenum where more space for larger bellows units is available. A short capillary line can be used to provide relatively rapid pressure measurement.

(2) Flux Monitoring and Control Instrumentation - System 95 (L. W. McClellan). The CSDD was approved by the Configuration Control Board and will be sent for RDT approval in April. The basic functions of this system are (a) to provide out-of-vessel and in-vessel flux measurements for reactor operation and reactor safety, (b) to provide in-core and in-vessel neutron flux, neutron fluence and neutron energy spectra measurements for material surveillance, operating information, and NPTR correlation.

(3) Fuel Failure Detection and Location (FEDAL) - System 94 (R. R. Cone and W. Dalos). Conceptual design of the FEDAL location system is being continued by both BNW and W. A number of conceptual process and instrumentation (P&I) diagrams for the location system have been prepared for the BNW/W/RDT FEDAL meeting which will be held in April at the Waltz Mill, Pennsylvania site. The principal of this is to select the conceptual P&I diagram which will become the reference diagram for the System 94 CSDD.

B. GENERAL PLANT DESIGN

E. M. Johnston

1. General Plant Instrument and Electrical

C. D. Swanson

a. Plant Instrumentation and Control

J. W. Mitchell

(1) Central Control and Data Handling - System 91 (G. L. Waldkoetter and G. G. Thieme).

(a) Sizing of Digital Control Hardware. A summary tabulation of variables from several FFTF systems has been prepared to assist in estimating the digital hardware size and configuration. Table 2.B.I. is a summary from the detailed lists. The variables sources have been grouped to correspond to three major multiplexer (MUX) sections. Although information from some FFTF systems providing input to the computer system are not available for inclusion in the tabulation, the systems listed have the major impact on the computer system size. The estimated totals for each MUX section are as follows:

- Core, Vessel and Shield - 670 MUX channels with an average scanning rate of 570 points/sec.
- Process and Experiment Systems (out-of-core) - 850 MUX channels with an average scanning rate of 355 points/sec.
- Pipe Preheat Temperature - Estimated 1500 MUX channels with an average scanning rate of five points/sec.

The computer system will also have available, on operator demand, an estimated 700 trends, and will assemble an operating history from about 1000 variable items of input. These totals include estimates from the Sodium Receiving and Processing System and the Inert Gas Receiving and Processing System although detailed lists of these systems have not beem completed.

TABLE 2.B.I. *Summary Tabulation of Computer System*

Variables Source	MPX Channels[a] High	MPX Channels[a] Low	Average Scan Rate	CRT Trends	System History Input	SK-3-14110
Core, Vessel and Shield						
In-Core Driver Fuel Channels	440	0	440/sec	228	295	Sheet 1
In-Vessel Out-of-Core	13	0	7/sec	10	10	Sheet 1
Vessel & Shield Process	0	68	7/sec	48	64	Sheet 1
Open Test Facility - 2 Loops	14	26	4/sec	18	32	Sheet 1
In-Core Closed Loop - 6 Loops	42	36	78/sec	54	54	Sheet 2
- (per Loop)	(7)	(6)	(13/sec)	(9)	(9)	Sheet 2
Flux Monitoring and Control	30[b]	0	30/sec	31	52	Sheet 2
Pipe Preheat Temperature						
Reactor Heat Transport System	0	300	1/sec	0	0	Sheet 3
Closed Loop Process System (6)	0	600	2/sec	0	0	Sheet 5
Sodium Receiving and Processing	0	600	2/sec	0	0	Estimate
Process and Equipment Systems						
Reactor Heat Transport System	96	48	98/sec	70	100	Sheet 3
Out-of-Core Closed Loop	336	96	192/sec	192	264	Sheets 4 & 5
System (6) (per Loop)	(56)	(16)	(32/sec)	(32)	(44)	Sheets 4 & 5
Inert Gas Rec. and Processing	TBD		TBD	TBD	TBD	-
Sodium Rec. and Processing	TBD		TBD	TBD	TBD	-
Atmospheric Variables	29	34	35/sec	25	63	SK-3-14538
Radiation Monitoring	61	50	10/sec	4	111	SK-3-14080

a. *Does not include digital multiplexer inputs*
b. *Does not include 33 inputs which may be digital with a rate of 330 pts/sec*

(2) Development Task IC-51, Process Data Display (M. D. Erickson).

(a) Process Data Display. Design of the PDP/9 Computer - D-30 CRT Display interface has been completed. This display system will be used to develop FFTF control and data system display requirements. The CRT manufacturer indicates that the display will consist of a TV set with an alphanumeric keyboard below and in front of the screen. A small TV camera will be used to display video signals from transparencies for permanent background displays on the display.

(3) Plant Instrumentation System - System 93 (M. O. Rankin). A list of containment variables to be measured in the machinery and containment domes has been defined. Control actions which will be initiated when predetermined limits of a combination of these variables are exceeded have been established.

b. Electrical Systems

G. H. Strong

The Conceptual System Design Descriptions for the following Electrical Systems have been revised to include the appropriate Design Safety Criteria requirements:

- Primary Electrical Power System - System 11.
- Building Electrical Power System - System 12.
- Communications System - System 15.
- Lighting System - System 16.

(1) Primary Electrical Power System - System 11 (M. F. Wiitala). One 115 kV to 13.8 kV power transformer rated at 20 MVA has become available at the AEC-Savannah River Plant and is being held while it is determined whether it can be used for the FFTF Primary Power System. The AEC-Richland Office has

been requested to determine whether a second transformer of the same rating will become available. If two such transformers are available and are found to meet the Quality Assurance requirements, new 115 kV power transformers for the Primary Electrical System will not have to be purchased, and this could save as much as $100,000.

2. Other General Plant Facilities

a. Fuel Examination Facilities

C. L. Boyd

(1) Conceptual Design (D. J. Meyers and E. B. Ramsey). A BNW/Bechtel design review meeting was held to discuss System 71 problem areas resulting from the Bechtel Conceptual Design Review effort. Those areas identified as requiring resolution were:

- Criticality Control - Design Safety Criteria, Paragraph 27.1.9
- Radiation Dose Rates - Design Safety Criteria, Paragraph 27.1.1
- Specific Emergency Power Requirements - CFDD No. 71, Paragraph 1.2.1-AD.

(2) Examination Equipment (E. B. Ramsey). Completion of the GE/IPO study report on the conceptual design of the FFTF Radiography Facility[1] is expected by early April. Follow-on development effort required to define neutron radiography design requirements and criteria (as they relate to imaging FFTF fuel pins and bundles) will begin after receipt of RDT approval.

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1. *FFTF Monthly Informal Progress Report, BNWL-990. Battelle-Northwest, Richland, Washington, January 1969.*

(a) Disassembly and Reassembly Equipment. The conceptual design study by ORNL is continuing in the Phase II effort of the FFTF Disassembly-Reassembly Equipment.[1] Their report will be based on a conceptual design description format with drawings of the equipment; its remote equipment, installation, operation, and maintenance; the proposed programs for development and tests; and the projected schedules and cost estimates. The projected schedule (in a broad form of critical path schedule) includes preliminary and final design, machine and tooling development, procurement, fabrication, inspection, functional and performance testing, and all necessary engineering laison and documentation. The cost estimate for the projected scheduled work resulted in an overall figure of $1,595,000 to acquire two disassembly-reassembly machines. This cost assumes carbon steel construction and a prudent and practical amount of quality assurance surveillance.

1. *FFTF Monthly Informal Progress Report, BNWL-990. Battelle-Northwest, Richland, Washington, January 1969.*

III. GENERAL TECHNOLOGY

P. L. Hofmann

A. DESIGN EVALUATION

L. M. Finch

1. Reactor Evaluation

D. Marinos

a. Core Concept Studies

G. R. Waymire

Further work to resolve problems in core design affecting preparation of the CCDD No. 31 continued during March. Areas investigated and either partially or completely resolved included:

(1) Determination of the number of fuel assemblies. Tentative resolution with Westinghouse (W) on peaking factors indicates that additional driver fuel will be required. Methods for achieving a total core power of 400 MW_t while maintaining a goal peak flux for the first core of 0.7×10^{16} nv are presently under investigation. Methods being considered include:

- Increase fuel volume fraction by reducing the pin spacer diameter.
- Increase the core height.
- Provide additional assemblies to the periphery of the present core map.
- Adopt a totally different core map.

(2) Required In-Core Control. Further neutronic calculations of worth as a function of radial distance from the core centerline, indicate that by utilizing a smaller pitch to diameter ratio, and increasing the density of the poison, rod worths can be attained which will produce the required margin with three intermediate control positions removed. In addition to adding three more full assembly positions to the

core, this change frees the tight packing at the reactor cover. The removed control positions are noted in Figure 3.A.1.

(3) Flux Monitor Positions. The in-core flux monitor positions require three additional penetrations in the reactor cover. Two possible positions include:

(a) A location at an unspecified distance above the core axial midplane with support provided by the instrument tree.

(b) An additional location along the test position radius with probe guidance provided by a tube with configuration similar to an open test position.

Position (a) would require utilization of the STIF for flux monitoring during startup when two out of three logic is required. Position (b) appears to present minimum perturbations to the overall core design. Further definition of possible locations is still required, however, to prove feasibility.

(4) Design Definition of All Major Components. Routing of the FEDAL transition lines coupled with restraint design has required that the top of the shield assemblies be lowered from 18 in. to 32 in. below the top of the fuel assemblies. The impact of this change on required shielding has not yet been determined. Definition of the in-core portion of the instrument tree is nearing completion. Design of the instrument tree plug will be initiated as soon as the in-core portion is complete.

b. Fuel Assembly Duct and Radial Restraint Design

G. R. Waymire

Joint meetings were held between Westinghouse and BNW to discuss the duct design for the fuel assembly and its interface with the radial restraint system.

CORE MAP

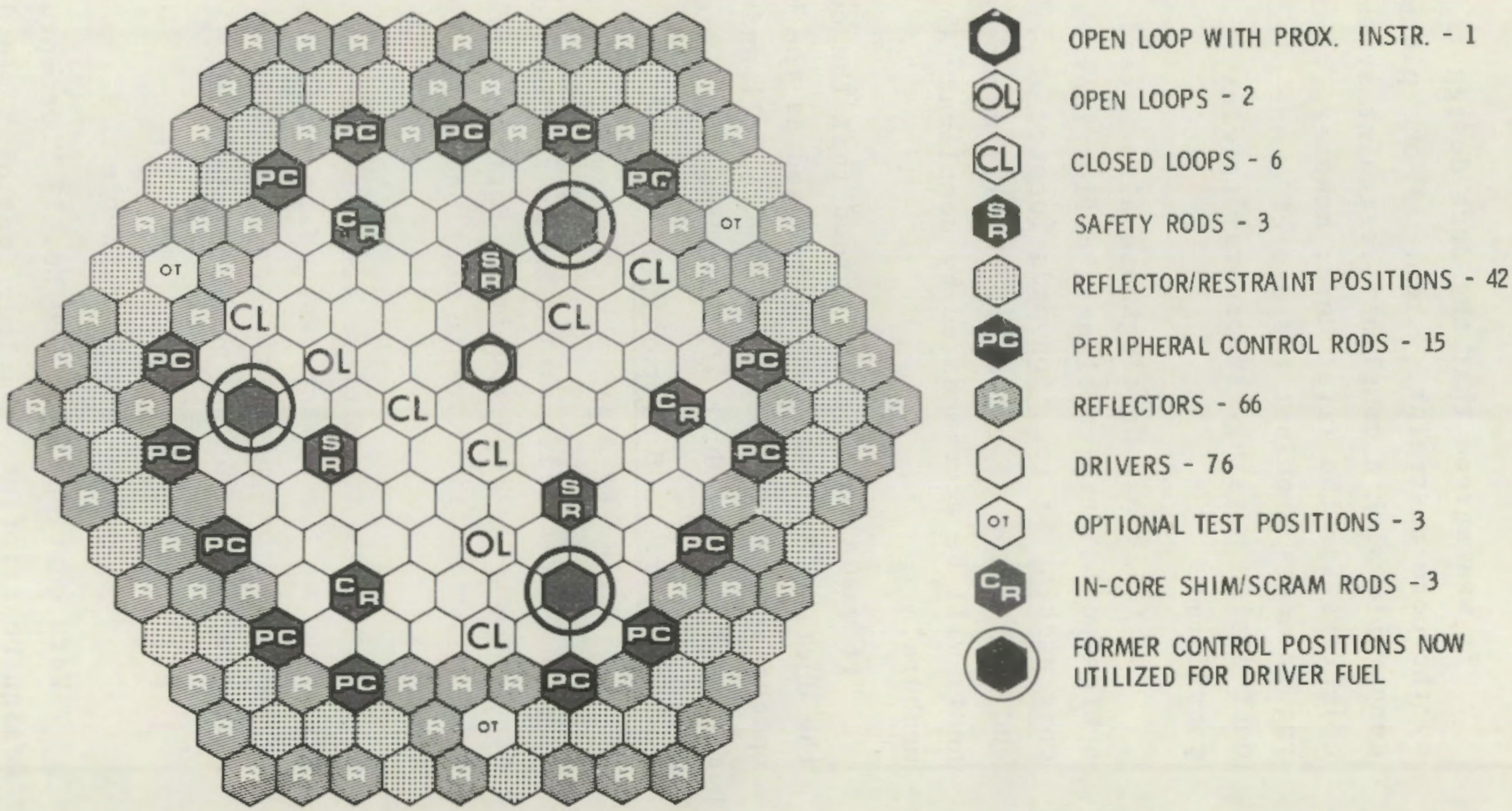

FIGURE 3.A.1. Core Map

It was agreed that the duct design is strongly influenced by the core restraint system. A one-piece structural duct compatible with a compliant restraint system will be the design objective for the first core; however, if core restraint requirements prohibit the use of a one-piece duct, alternative duct design possibly incorporating flexure members will be considered.

Battelle and Westinghouse also agreed that the restraint system would utilize elastic deflections which are limited by core packing at reaction pads located above and below the core and at the top of the duct. Radial expansion of core components will be accommodated by compliancy in the loading members.

It was also mutually agreed that the conceptual drawing of the duct, SK-3-14581, would be used as the basis for further development and for inclusion in CCDD No. 31 and 35. The reference material for the duct will be Type 304 austenitic stainless steel.

c. Core CCDD Preparation

G. R. Waymire

The first draft of the major sections of the description are complete. Areas requiring further work include the addition of parametric data in Section II and completion of the appendix.

The bases document is complete and ready for TEB review. Revisions to the fuel peaking factor will definitely require revision of the parametrics document and may require some revision of the nuclear performance document.

d. Nuclear Control

D. Marinos and W. M. Noble

Work continued on CCDD No. 33 "Conceptual Component Design Description - Nuclear Control." To date, a rough draft of Section 1 is in the final stages of completion. Sections 3 and 4 are complete and Section 2 is presently undergoing major rewrite to apply the latest core requirements mentioned in Item a of this section. Final determination of rod worth for selected P/D ratios as well as thermal hydraulic verification for these conditions is in progress.

e. Irradiated Fuel Handling
M. A. Fischer

The FH-7 Task Description for Design Evaluation of the Fuel Handling and Radioactive Maintenance Systems was completed. Effort continued on the CSDD for the Irradiated Fuel Handling System (System 43). Final flow diagrams and functional analysis sheets have been completed. The detailed outline including descriptions of appendices and a drawing list has been completed.

f. Reactor Vessel and Shield Conceptual Design
D. P. Schively

Preparation of the finalized version of the Reactor Vessel and Shield Conceptual Component Design Description, No. 32, is proceeding on schedule. The draft document has been reviewed and approved, after appropriate modifications, by the Technical Evaluation Board. Consideration and approval of the CCDD by the Configuration Control Board is scheduled for the last week of March. No major problems are foreseen in obtaining technical concurrence by Babcock and Wilcox for the conceptual configuration, but the configuration of the inlet plenum has yet to be resolved with Westinghouse.

B. REACTOR PHYSICS AND OPERATIONS

R. E. Heineman

1. Nuclear Analysis

W. W. Little

a. ETOX[1] Modifications

R. B. Kidman and J. L. Baker

Initial experience with ETOX, which generates group average cross sections from the ENDF/B library for use in fast reactor codes, has revealed certain areas where improvements are desirable. The calculation of inelastic scattering transfer matrices has been reprogrammed to better account for the following occurrences:

(1) When an inelastic partial energy distribution begins or ends within a group, or

(2) When discontinuities appear in the inelastic probabilities.

This reprogrammed portion is more accurate and significantly faster.

Also added to ETOX is an option to calculate the fission neutron source for each group. Since this information is necessary input to 1DX[2] and FCC-IV,[3] the results are punched on cards in an appropriate format for use by the two codes.

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1. *R. E. Schenter, J. L. Baker, R. B. Kidman. ETOX, A Code to Calculate Group Constants for Nuclear Reactor Calculations, BNWL-1002. Battelle-Northwest, Richland, Washington, 1969.*
2. *To be published.*
3. *W. W. Little, Jr., R. W. Hardie. FCC-IV-A, Revised Version of the FCC Fundamental Mode Fast Reactor Code, BNWL-450. Battelle-Northwest, Richland, Washington, 1967.*

b. Plotting Code, ϕMAP

R. W. Hardie

A computer code, ϕMAP, designed to plot data from a 1DX or a 2DB flux tape is now operational. The program will plot the following items:

(1) Integeated flux versus ln(E) for a specified row and column,

(2) Flux per unit lethargy versus ln(E) for a specified row and column,

(3) Flux or ln(flux) versus R for a specified group (or total) and row,

(4) Flux or ln(flux) versus Z for a specified group (or total) and column.

This code expedites the interpretation of physics analyses for the FTR.

2. Experimental Physics

R. A. Bennett

a. Phase B Critical Experiments Status

R. A. Bennett

Critical experiments being conducted in ZPR III, Assembly 56b are presently expected to be completed by March 28, 1969. The program was extended approximately one week to include central fission ratio measurements and axially-integrated core and poison worth measurements. Approximately a one-week delay in the program has resulted from experimental equipment malfunction.

Plans for conducting the second series of experiments (FTR-2) in a larger facility are complete with two specific exceptions and Battelle-Northwest has received Argonne National Laboratory's concurrence. Additional detailed plans for neutron lifetime and fuel storage experiments remain to be completed. However, because of early dates for initiation of procurement of the FTR vessel and head it may be necessary to modify the schedule. Shielding experiments may have priority over core physics experiments.

b. Cylindricization of ZPR III-56b

S. L. DeMyer

The critical assembly ZPR III-56b, previously referred to as FTR I of the FTR Critical Experiments Program, is to be included in an assessment of calculational models presently being used in analyses of FTR core design. For purposes of this assessment, the somewhat irregular core coundary has been cylindricized and gross in-core heterogeneities have been removed. Reactivity and fissile mass changes involved in the cylindricization and homogenization are given in Table 3.B.I together with the dimension of the equivalent right circular system. The average atom densities of the system are given in Table 3.B.II.

c. Doppler Effects

W. R. Young

Calculated values of the negative Doppler effects in ZPR III, Assembly 51 are consistently less in absolute magnitude than the measured values,[1] indicating that estimates of Doppler effects currently used in FTR design and safety analyses are conservative. However, analysis of these experiments is necessarily based on a model more complex than that used for FTR Doppler calculations.

Because the relationship between the experimental Doppler effects and the FTR Doppler reactivity coefficient is not simple, there is a need to devise new Doppler experiments for future FTR criticals. A desirable objective of any new experiment would be a measured quantity from which the FTR Doppler could be inferred to within ±25%.

1. *Reactor Development Program Progress Report*, *ANL-7460*, *p. 20. Argonne National Laboratory, Argonne, Illinois, June 1968.*

TABLE 3.B.I. Critical Parameters ZPR III-56b

Initial Conditions:

216 "A" drawers

203 "B" drawers

17 Safety and control drawers fully inserted

1 Control drawer withdrawn 8.8 in.

k_{eff} = 0.999964

Temperature = 37.6 °C

Mass = 33.938 kg $^{239+241}Pu + ^{235}U$

	Changes:	% ΔK/K	Mass, kg
1.	Fully insert control drawer	+0.0719	333.415
2.	Temperature increase 37.6→40.0 °F	-0.0056	333.415
3.	Cylindricize Core	+0.034	332.970
4.	Convert safety and control drawers to "B" drawers	+0.0534	333.656
5.	Remove interface gap void between halves	+0.063	333.656
6.	Return k_{eff} to 1.00000	-0.212	328.703

Final Critical Parameters:

k_{eff} = 1.00000

Mass = 328.7 kg $^{239+241}Pu + ^{235}U$

Core Radius = 45.49 cm

Reflector Radius = 65.08 cm

Core Height = 91.59 cm

Axial Reflector Thickness = 27.90 cm

NOTE: The probable error in this assessment is taken to be ±0.010% ΔK/K, which is largely associated with the interface gap void width.

TABLE 3.B.II. Average Atom Densities, Assembly 56b

Isotope	Atom Densities, atoms/b-cm Core	Axial Reflector	Radial Reflector
Na	0.008669	0.01346	0.006567
Fe	0.013757	0.008825	0.007586
Cr	0.002492	0.002188	0.001880
Ni	0.001090	0.01948	0.047599
Mn	0.0001041	0.000181	0.000289
Si	0.0001221	0.0001072	0.000123
^{238}Pu	0.0000006		
^{239}Pu	0.001332		
^{240}Pu	0.0001770		
^{241}Pu	0.00002515		
^{242}Pu	0.00000259		
^{235}U	0.00001352		
^{238}U	0.006193		
Am	0.000001233		
Mo	0.0003433		
O	0.015192		
C	0.001030		

d. Neutron Source for the FTR

Q. L. Baird

The potential need for an extraneous neturon source to supplement the fuel source was reviewed. For a fully loaded "green" FTR core containing approximately 650 kg of plutonium, the source strength resulting from spontaneous fission and the (α,n) reaction will be about 1.4×10^8 n/sec. The source strength will depend primarily upon the ^{240}Pu content of the fuel. The source strength also will increase a few percent as the fuel is irradiated, because of the change in the isotopic composition.

The neutron flux at the source range sensor when the reactor is subcritical with a k_{eff} of 0.9 is calculated to be about 10^4 n/cm^2-sec from the fuel source alone. With the sensitivities of currently available fission chambers, this will result in counting rates greater than 30 counts/sec. Such count rates are quite adequate for startup and shutdown monitoring. For example, when 10% subcritical, reactivity changes can be determined to the desired accuracy of ±1$ with counting times of less than 10 minutes. Therefore, no extraneous neutron source is required. However, an extraneous neutron source could be used to verify the operability of the low level neutron monitor.

e. FTR Testing Requirements
Q. L. Baird

Preliminary physics testing requirements for definition of neutronics environment of tests within FTR and for operation of FTR have been determined and transmitted to the FFTF Development Department for review.

3. Radiation and Shield Analysis
W. L. Bunch

a. In-Vessel Storage of Fuel
E. T. Boulette

Irradiated fuel is to be stored within the vessel in the sodium annulus surrounding the core, reflector, and shield. Calculations indicate that the neutron spectrum incident on the vessel will be hardened significantly in the vicinity of the stored fuel. Experiments are planned in a zero power critical facility to validate these calculations and provide a basis for further design details. The experiment will also investigate the impact of stored fuel on the reactor instrument system.

b. Radiation Fields from Unirradiated FTR Fuel

C. A. Mansius and W. L. Bunch

The isotopic composition of the plutonium used in FTR fuel will depend on its previous irradiation history. Because of the decay characteristics of the various isotopes, the radiation fields encountered by personnel while processing fuel will depend on its isotopic composition. Preliminary calculations were made to estimate the exposure rates associated with fuel meeting the specifications for the initial loading, and for typical fuel that might be used at some later date. In both fuels the gamma intensity was calculated to be dependent primarily upon the ^{241}Pu content and its daughter products ^{237}U and ^{241}Am, whereas the spontaneous fission of ^{240}Pu is expected to contribute most of the neutrons. At a distance of two feet from a fuel assembly, the calculated rate is 7 mrem/hr from the initial fuel and about 23 mrem/hr from potential recycle fuel. These preliminary values are to be used in the conceptual design of the unirradiated fuel handling system.

c. Effect of Sodium Density on Neutron Attenuation

E. T. Boulette

Initial sodium inlet temperature for the FTR is 600 °F, whereas it is to be designed to operate at 900 °F inlet. In both cases the temperature increase is to be 300 °F, giving outlet temperatures of 900 °F and 1200 °F, respectively. The density of sodium decreases with temperature. For a typical conceptual arrangement of the radial shield, calculations indicate that a sodium density change from 0.83 to 0.79 g/cm^3 will increase the total neutron flux to the vessel less than 10%. In the case of the pool above the FTR, the decrease in sodium density will increase the flux to the cover by about a factor of two. It is concluded that sodium density, within the range of interest, is not a major parameter in shield design even though the effect must be included.

4. Reactor Operations

D. C. Boyd

a. Training Program

An operator training program is being defined for the FFTF operating staff. Planning for this program continued as follows:

- Information was requested from MIT regarding their Reactor Technology Course.
- The AEC-NE&T Department was contacted concerning a Fast Reactor Study Course for the Center for Graduate Study.
- Information was requested from SEFOR on the training input to their safety analysis reports.
- Discussions were held to obtain viewpoints on training requirements and qualifications.

C. REACTOR AND SYSTEMS ANALYSIS

R. E. Peterson

1. Reactor Core Analysis Section

P. D. Cohn

a. Energy Release for Postulated Maximum Accidents

A. Padilla, Jr.

In BNWL-760 (Preliminary Analysis of Postulated Maximum Accidents for the FFTF, November 1968), the maximum potential to do work (energy release) was calculated by expanding the fuel vapor (sodium-out case) or the sodium vapor (sodium-in case) in a thermodynamic reversible process. The results for the sodium-in case were somewhat limited and the effect of some of the important parameters on the results was not fully assessed. Therefore, this study was undertaken to investigate the importance of some of the assumed parameters in the BNWL-760 analysis. There are slight differences between this study and the results in BNWL-760 even for the same conditions because the analytical method has been refined.

The analytical model assumes that at disassembly the molten fuel and sodium in the core become mixed and reach thermal equilibrium instantaneously. This equilibrium temperature is above the critical point of sodium for the range of conditions expressed in BNWL-760. The sodium is then expanded in a reversible process and the maximum work for this expansion can be calculated by thermodynamic principles. However, there are two extremes for the expansion process. If there is no heat transfer between the sodium and the fuel during the expansion, then the process is an adiabatic expansion. On the other hand, perfect heat transfer between the sodium and the fuel corresponds to the sodium and fuel being in thermal equilibrium throughout the expansion.

Figure 3.C.1 shows the energy release for both types of expansions for various initial fuel temperatures. In BNWL-760, the average temperature of the molten fuel at disassembly was 3600 °K for the accident initiated from full power and 3800 °K for the accident initiated from shutdown conditions. The sodium-to-fuel mass ratio of 0.05 corresponds to the fuel design for the split conical core which was used as the basis in BNWL-760. The figure depicts that higher initial fuel temperatures significantly increase the energy release for the thermal equilibrium expansion whereas the energy release is constant for the adiabatic expansion. All expansions were carried to a final pressure of 1 atm.

Figure 3.C.2 shows the effect of expanding down to pressures other than 1 atm. Note that the initial fuel temperature used is 4000 °K which is higher than both cases listed in BNWL-760. It is seen that expansion to 1 atm increases the energy release tremendously for the thermal equilibrium expansion. Inasmuch as the final pressure of 1 atm was used arbitrarily, higher pressures can probably be justified rather easily. Instead of expanding down to higher pressures, it may be more realistic to expand to certain volumes corresponding to specific containment structures. Figure 3.C.3 shows the pressure as a function of expansion volume. For example, expansion to the volume corresponding to the machinery dome (estimated to be 8600 ft^3) gives a pressure of almost 9 atm. This reduces the energy release for the thermal equilibrium expansion by more than 30% over the 1 atm case. The actual pressures and temperatures to which the containment structures will be subjected also depend upon interaction of the expanding sodium bubble with the gas originally in the containment structure. Figure 3.C.3 also shows these pressures and temperatures assuming that the containment structures contain air at 100 °F and 1 atm (0 psig). However, the effect of a sodium-air reaction has been disregarded.

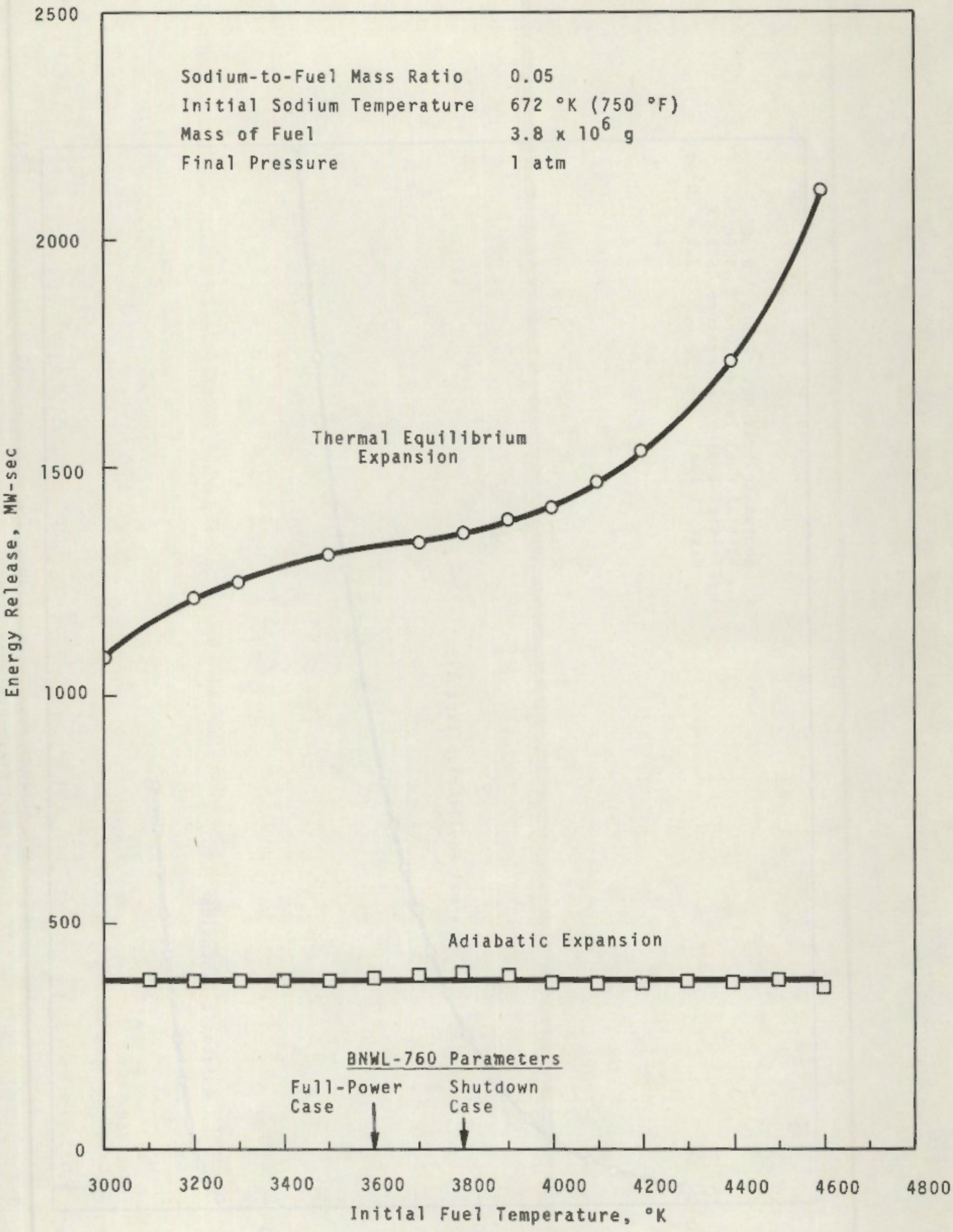
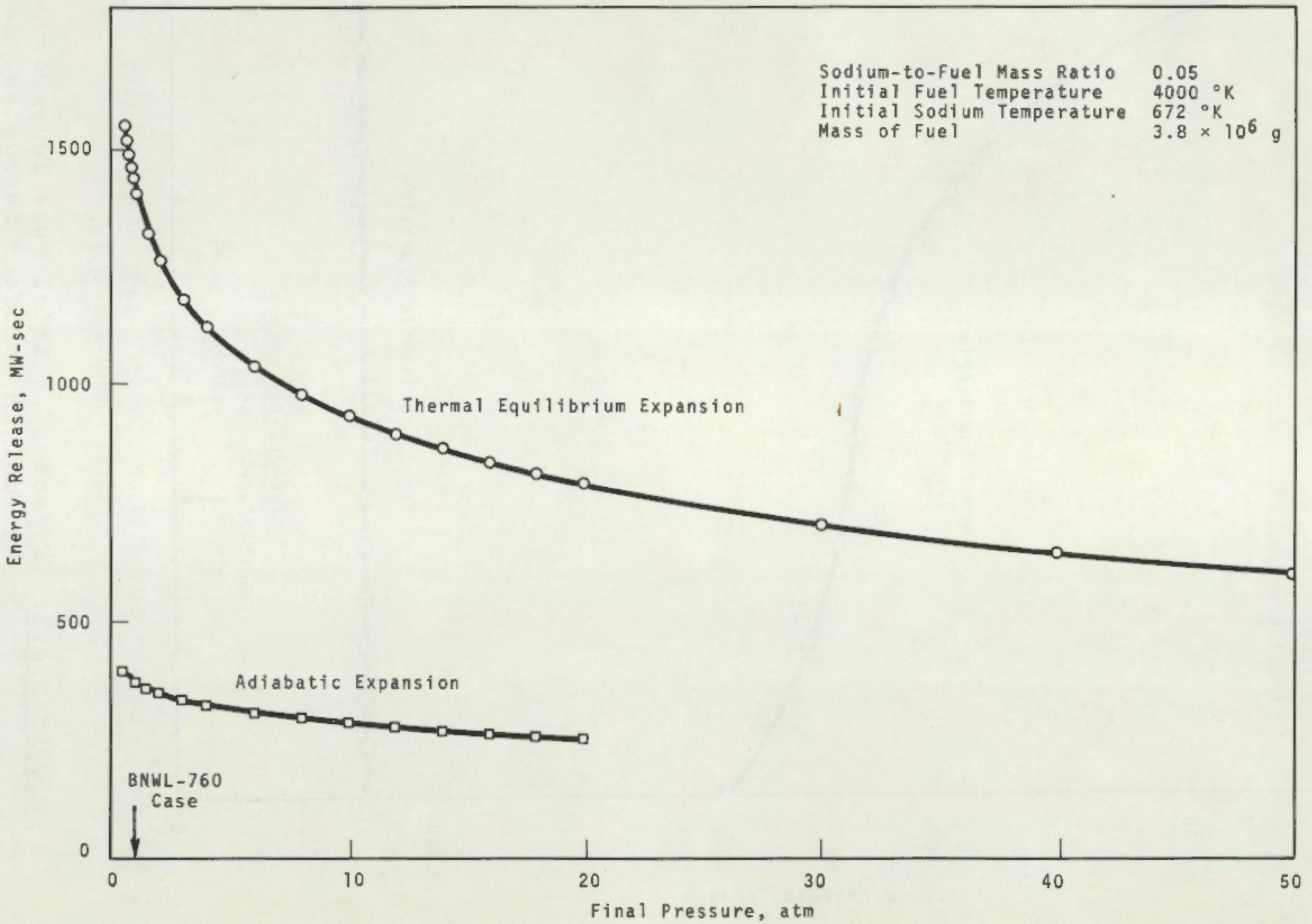


FIGURE 3.C.1. *Effect of Initial Fuel Temperature on Maximum Accident Energy Release*



<u>*FIGURE 3.C.2.*</u> *Effect of Final Pressure on Maximum Accident Energy Release*

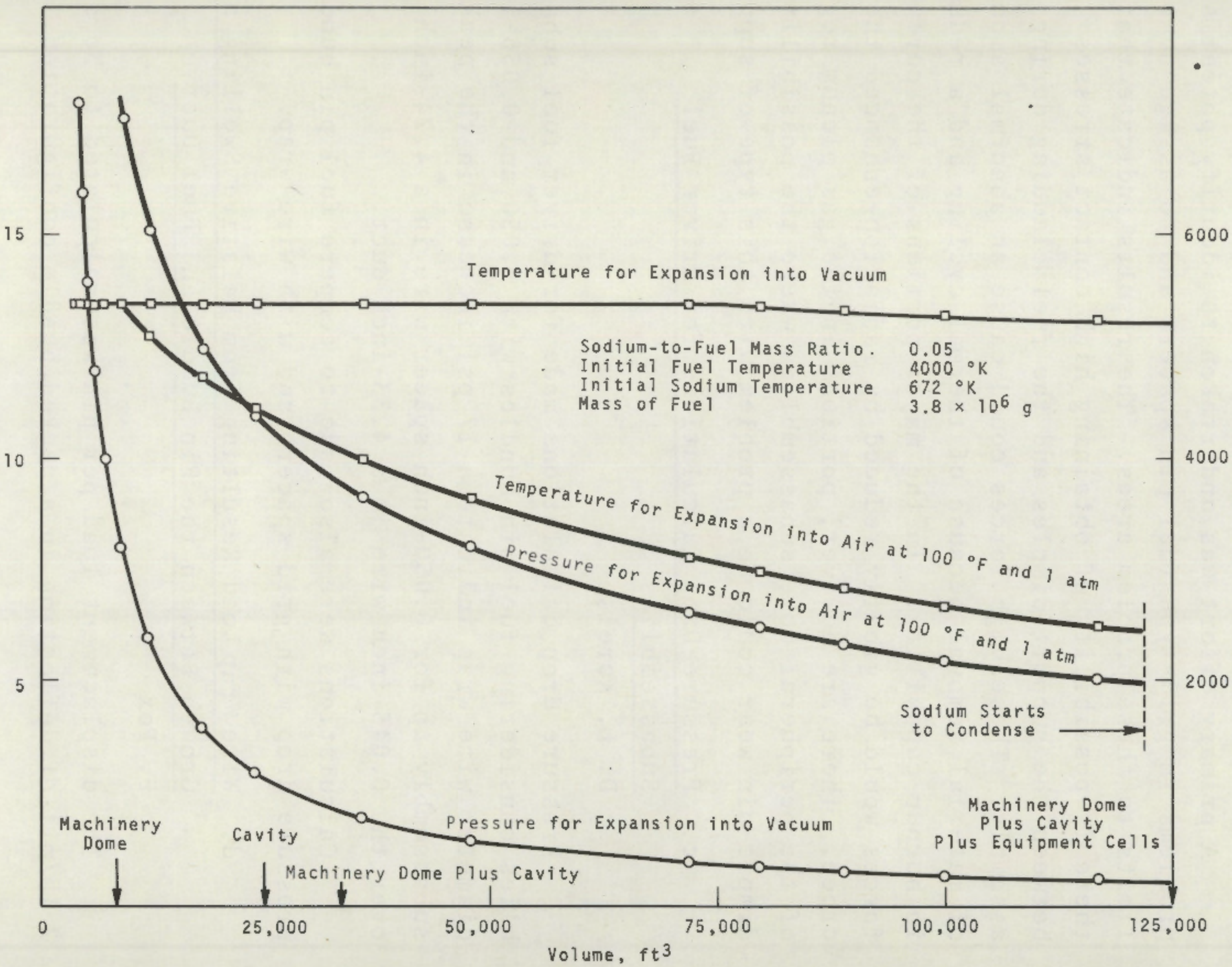


FIGURE 3.C.3. Expansion of Sodium in Thermal Equilibrium with Fuel

b. Preliminary Hard Point Grid Analysis

C. L. Mohr

A primary effort was undertaken to identify potential problems in grid-type fuel pin spacers and to assign numerical limits to these problem areas. The results indicate that there is possibility of obtaining high contact stresses between the support dimples and the fuel cladding during assembly. These high forces could cause an abnormal amount of material removal because of thermal cycling and a reduction in burnup capability. In the major portions of the core these forces would be greatly reduced by radiation-enhanced thermal creep. There are however, portions in the gas plenum regions of the peripheral core subassemblies where the possibility of dimple-pin wear could be a problem with this type of support.

c. Pressure Drop Calculations for Driver Fuel Subassemblies

D. L. Koreis

Pressure drop calculations made for driver fuel subassemblies considering fuel pin bundles with 0.050 and 0.056-inch diameter wire wrap indicate a 17 psi increase in the peak flow subassembly Δp for 0.050-inch spacer wire in a 4.24-inch duct over the 0.045-inch wire in a 4.33-inch duct.

Calculations were also made to compare fuel pin bundle pressure drop with grid spacers and with wire wrap.

d. Wire Wrap Load Resulting from Relative Swelling and/or Growth Between the Pin Bundle and Fuel Duct

F. Fox

The displacement load per pin was calculated to be about 2 lb/mil of deflection in a packed bundle configuration.

2. Systems and Instrument Analysis

H. G. Johnson

a. Emergency Core Cooling Analyses

C. D. Flowers, H. G. Johnson, and R. N. Madsen

Additional computer runs were made to show the effect of the number and size of multiple parallel reactor inlet downcomers on core coolant temperature responses to double-ended pipe breaks at a reactor inlet nozzle for an advanced core (533 MW, 800 to 1200 °F, 3-loops). Data were obtained for two 12-in., three 10-in., and five 8-in. downcomers per primary loop. It was determined that hot channel coolant boiling could be prevented by using three 10-in., or five 8-in. downcomers with the heat transport system arranged in accordance with the emergency cooling concept described in the November 1968 report.[1]

All of the analytical work completed over the past eight months on emergency core cooling systems, which are designed to limit core damage to below that of a disruptive accident for double-ended pipe failures, is being summarized for presentation as a topical report. A first draft of the report was completed.

b. Hybrid Simulation of the Reactor and Heat Transport System

R. D. Benham and A. L. Gunby

Simulation Development. The modular programming approach (MMP) to simulation on the Hybrid-2 machine (DEC-PDP-9, AD-4) is 95% complete. Only the digitally-simulated DHX and the primary flow calculation need to be debugged. In all, the digital portion of the simulation is broken into 27 separate

1. *FFTF Monthly Informal Technical Progress Report, November 1968, BNWL-937. Battelle-Northwest, Richland, Washington, December 1968.*

modules with the analog computer performing as the 28th. Support programs for heat balance calculations and process variable monitoring are operational. Coolant transport delay programs have been altered to allow the operator to set the delay to zero, and speed up the process of reaching new initial conditions, particularly at low flows.

c. Heat Transport System Analysis

Dump Heat Exchanger Analysis (R. N. Madsen). Verification and checkout of the DHX model to which reference was made in the previous monthly report has been completed. Both steady-state and transient test cases have been investigated and the satisfactory performance of the model has been demonstrated. The transient test cases demonstrated that in order to avoid an excessively small and restrictive time increment the heat storage capacity of the air coolant must be disregarded. This is not an unreasonable approximation since in a given section of the model the heat storage capacity of the air is only about 1/400th of the sodium heat capacity and 1/1700th of the metal capacity in that section. Making this assumption now permits transient cases to be simulated in about ten times actual time; that is, a transient of about one minute duration requires about ten minutes of computer time. This is rapid considering the great detail of the model. Evaluation of the performance of DHX simulations for which this model was created is now in progress.

3. Plant and Facilities Analysis

R. J. Hennig

Control Rod Analysis

A. F. Lillie

A parametric study of the FTR control rod absorber section assembly design has yielded the following tentative results.

- A 61-pin absorber section geometry with pin pitch-to-diameter ratios less than 1.1 employing 90% TD B_4C pellets appears to be preferred design considering maximum operating B_4C temperatures, total attainable individual rod worths, and rod scram dynamics.
- A reduction in the total number of control rods used in the present reference core arrangement may be possible using this rod design.

The study is endeavoring to verify these tentative results.

D. SAFETY RELIABILITY ANALYSIS

D. E. Simpson

1. Reliability Analysis

R. A. Harvey

a. Fault Trees

R. J. Schroder

A list has been compiled of current FFTF fault trees and additional fault trees required by those which exist. The list is given in Table 3.D.I. The fault tree for radioactive release has been revised to incorporate events not previously included. A new fault tree has been started for reactor unavailability.

Repair time scaling has been incorporated into the fault tree simulation program and significant reduction in run-time has resulted (about 20%).

A program to be used with the CalComp digital plotter for automated fault tree drawing is now usable although additional revisions are being made. Examples of the required input data and the resulting output are given in Figure 3.D.1. The cost is estimated at approximately $35.00 for a 100 event fault tree.

b. Effectiveness Goals

H. C. Martin

The <u>System Effectiveness Goals</u> document was completed, submitted to the Technical Evaluation Board, approved with the Board's comments incorporated, and submitted to the Configuration Control Board.

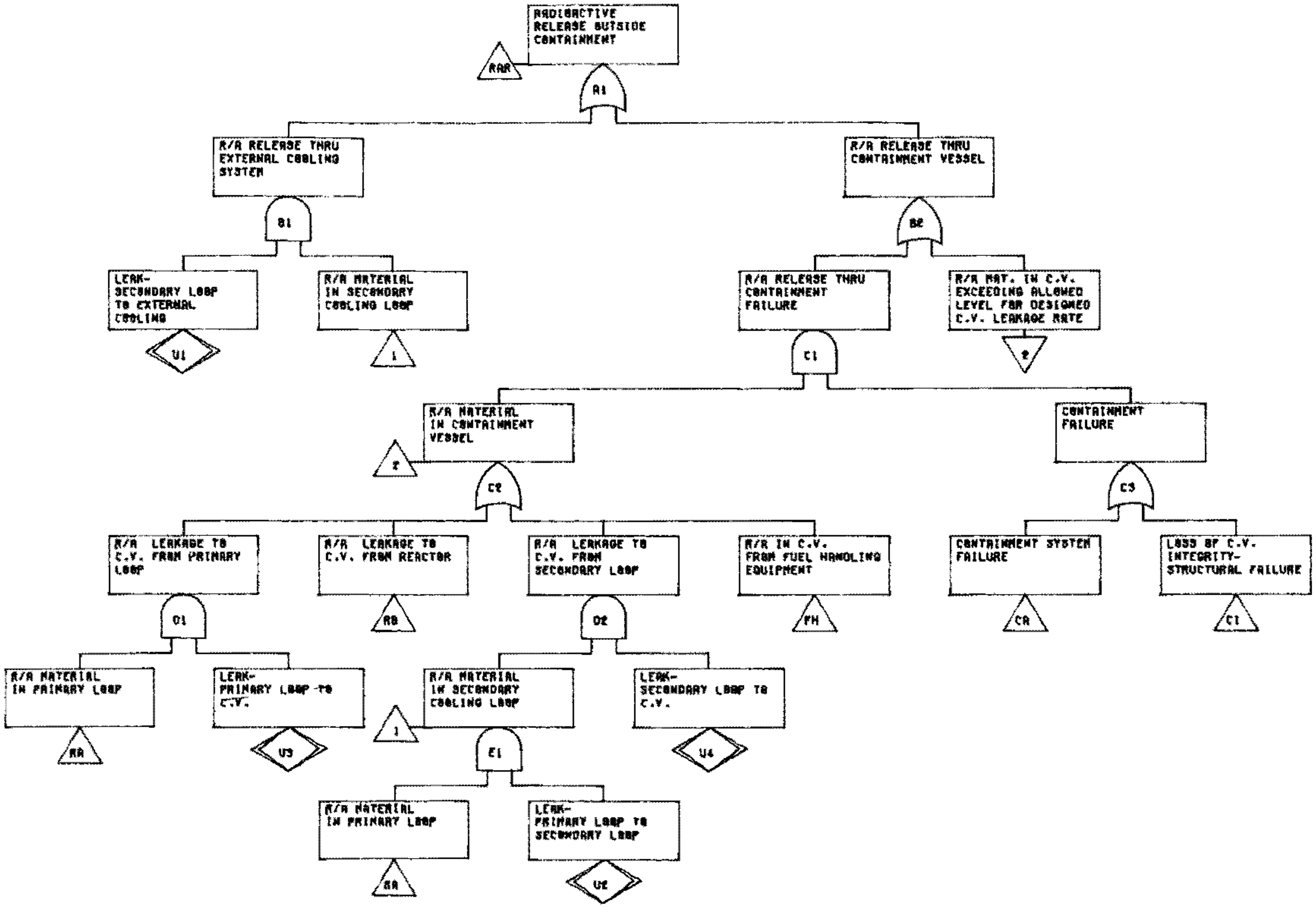
TABLE 3.D.I. Fault Tree Status - 3/69 (Fault Trees Required/Constructed)

System	Symbol	Description	Drawn by
Reactor			
	RI	Reactivity Insertion	McLaughlin & Schade
	RD	Core Disassembly	Schade
	RA	Fission Products in Primary Loop	McLaughlin (on RAR)
	RB	Fission Products Leak - Core to Containment Vessel	
	RF	Fire in Reactor Na Cavity	
	RG	Cover Gas Vented to Reactor Cavity	
	RP	Fuel Pin Cladding Failure	
	RS	Core Structural Failure	
	RC	Control Rod Failure/ Error	
HEAT TRANSPORT			
	HB	Heat Removal Imbalance	McLaughlin
	HP	Failure of Na Purification Loop	
	HV	Coolant Vaporization	
	HL	Coolant Leaks	
	HC	Coolant Controller Failure	
	HR	Coolant System Rupture	
PROTECTION			
	PS	Scram System Failure (about 26 separate trees)	Monteith
	PD	Detector Fails (about 120)	

TABLE 3.D.I. (contd)

System	Symbol	Description	Drawn by
FUEL HANDLING			
	FL	Gross Fuel Loading Error	
	FD	Fuel Element Dropped	
	FH	Radioactive Matter in Containment Vessel from Fuel Handling Equipment	
CONTAINMENT			
	CI	Loss of Containment Integrity	Hagan & Schroder
	CA	Containment System Failure	
	CF	Na Fire in Containment Vessel	
ELECTRICAL			
	EL	Loss of Electrical Power	
FFTF			
	RAR	Radioactive Release (outside containment)	
	DBA	Design Basis Accident	

NOTE: No entry in the "Drawn by" column indicates a tree has not been drawn.




```
A1      0  B1     B2
B1      1  U1     1
B2      0  C1     2
C1      1  C2     C3
C2      0  D1     RO     D2     FH
C3      0  CA     CI
D1      1  RA     U3
D2      1  E1     U4
E1      1  RA     U2
END    -0
A1     -0  RADIOACTIVE       RELEASE OUTSIDE   CONTAINMENT
B1     -0  R/A RELEASE THRU  EXTERNAL COOLING  SYSTEM
B2     -0  R/A RELEASE THRU  CONTAINMENT VESSEL
C1     -0  R/A RELEASE THRU  CONTAINMENT       FAILURE
C2     -0  R/A MATERIAL      IN CONTAINMENT    VESSEL
C3     -0  CONTAINMENT       FAILURE
D1     -0  R/A  LEAKAGE TO   C.V. FROM PRIMARY LOOP
D2     -0  R/A  LEAKAGE TO   C.V. FROM         SECONDARY LOOP
E1     -0  R/A MATERIAL      IN SECONDARY      COOLING LOOP
U2     50  LEAK-             PRIMARY LOOP TO   SECONDARY LOOP
U1     50  LEAK-             SECONDARY LOOP    TO EXTERNAL       COOLING
U3     50  LEAK-             PRIMARY LOOP TO   C.V.
U4     50  LEAK-             SECONDARY LOOP TO C.V.
CI     -0  LOSS OF C.V.      INTEGRITY-        STRUCTURAL FAILURE
CA     -0  CONTAINMENT SYSTEMFAILURE
RA     -0  R/A MATERIAL      IN PRIMARY LOOP
RO     -0  R/A  LEAKAGE TO   C.V. FROM REACTOR
FH     -0  R/A IN C.V.       FROM FUEL HANDLINGEQUIPMENT
1      -0  R/A MATERIAL      IN SECONDARY      COOLING LOOP
2      10  R/A MAT. IN C.V.  EXCEEDING ALLOWED LEVEL FOR DESIGNEDC.V. LEAKAGE RATE
END    -0
A1     20  RAR
E1     20  1
C2     20  2
END    -0
```

FIGURE 3.D.1. Representation of Computer-Drawn Fault Tree

2. Safety Analysis

D. D. Stepnewski

a. Scram System Parametric Study

C. L. Fies

Analysis of the trade-off between various Scram System parameters such as time delay, rod acceleration, and initial position of the safety rods continued. Figure 3.D.2 shows distance from the top axial face of the core versus increase in time delay for rod accelerations of 1, 2, and 3G. All results are plotted for a 10% insertion of safety rod worth

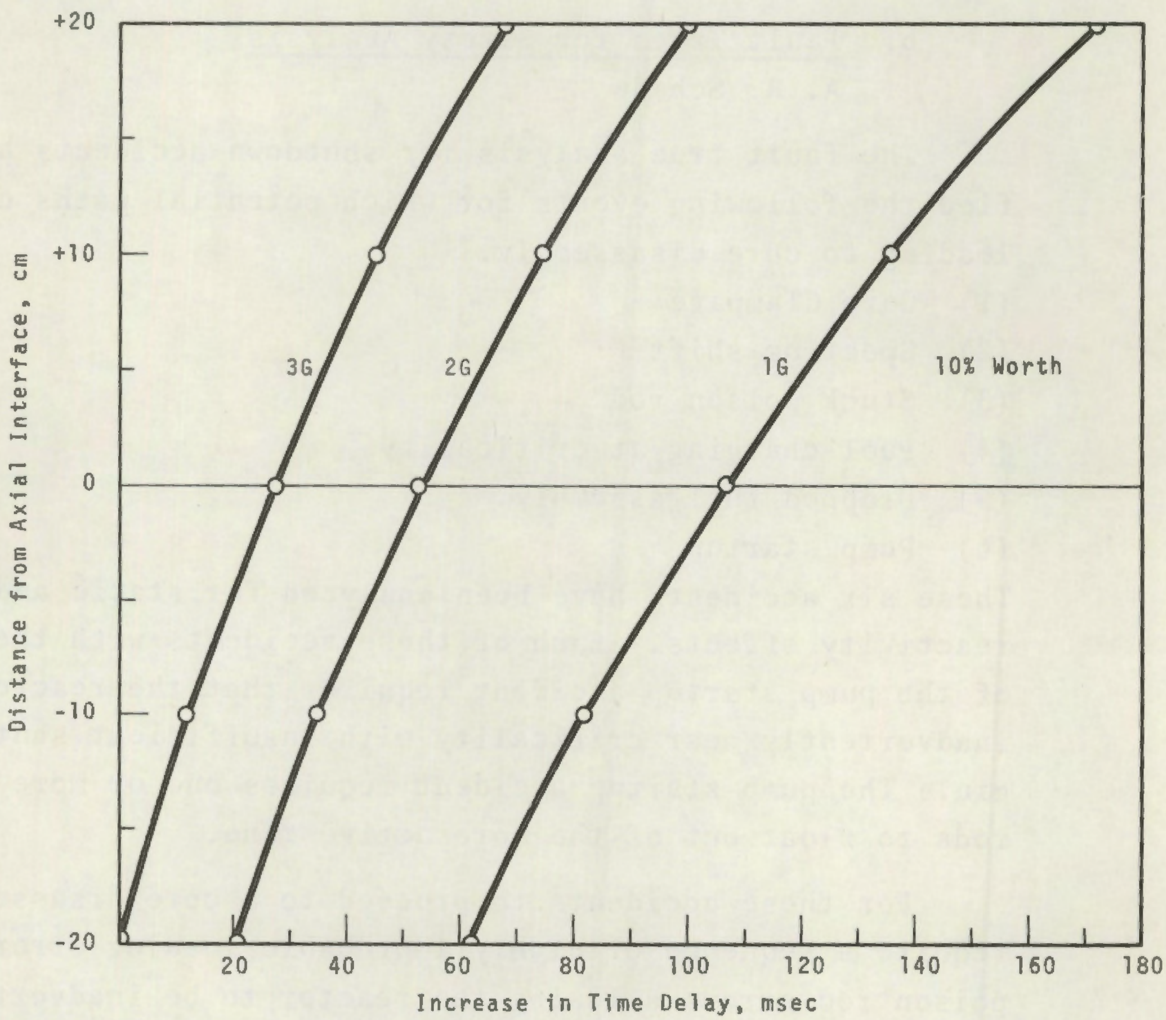


FIGURE 3.D.2. Initial Position of FTR Safety Rods Versus Increase in Time Delay for Rod Accelerations of 1, 2, and 3G.

(less one stuck rod) from the initial position. If, for example, the total safety rod worth was 12$, the curves show the decrease in time delay for a 1.2$ negative reactivity insertion. The results are normalized to the 3G-20 cm below the axial interface case. The curves, therefore, do not show the total time delay but only the changes as axial position or distance from the top axial face of the core is varied. The results show, for example, that for a 2G rod acceleration moving the safety rods from the top axial face to 20 cm above the axial face increases the time delay by 47 msec. For a 1G rod the same movement causes a 65 msec increase in delay time.

b. Fault Trees for Safety Analysis

A. R. Schade

The fault tree analysis for shutdown accidents have identified the following events for which potential paths do exist leading to core disassembly.

(1) Core Clampage
(2) Spectrum shift
(3) Stuck poison rod
(4) Fuel charging at criticality
(5) Dropped fuel assembly
(6) Pump startup

These six accidents have been analyzed for static and dynamic reactivity effects. Each of these accidents with the exception of the pump startup accident requires that the reactor be inadvertently near criticality with insufficient shutdown margin. The pump startup accident requires one or more poison rods to float out of the core active zone.

For these accidents to proceed to a core disassembly would require a sequence of highly improbable loading errors and/or poison rod errors allowing the reactor to be inadvertently near the critical point. In addition, the reactivity status of the reactor must not have been adequately monitored. The fault must occur concurrently with the failure of the protective system before the accident can proceed.

c. Parametric Nuclear Transient Analysis

A. R. Schade

Parametric analyses have been performed to determine what magnitude of reactivity inserted at a rate of 40$/sec is within the inherent negative feedbacks of the reactor. The core model was the reference vertical core[1] with orifeed coolant flow at 10% of the normal operating flow rate. Assuming only a feedback of -0.004 Doppler:

(1) For a reactivity insertion of >1$, Na will reach vaporization temperatures eventually if no corrective action is taken.

(2) For a reactivity insertion of >3$, fuel will melt before Na vaporizes.

If radial expansion feedback is included about 1$ of additional negative feedback is available. Na density changes give a positive feedback of +10¢ for transients of this type.

d. Design Safety Criteria

K. L. Berrett, C. L. Fies, J. W. Hagan, R. D. Peak

Configuration Control Board approval has been obtained on three additional systems criteria. These are: Structures - 21, Containment - 22, and the Electrical Group consisting of Systems 11, 12, 15, and 16. Three additional systems criteria have been submitted for CCB action. These are Short Term Irradiation Facility - 68, Sodium Receiving and Processing - 81, Fuel Assembly Component - 35. All currently planned DSC's have now been submitted for CCB action (or have been previously approved).

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1. *Memo, A. E. Waltar to Distribution, "Reference Vertical Core for Transient Studies," December 19, 1968.*

Definitions of "Size" of Leaks in the Reactor Heat Transport System (R. D. Peak). Quantitative definitions of a "small," "intermediate size," and "large" leaks are required as a basis for the design of emergency core cooling provisions in the FFTF. A preliminary definition of the "intermediate size" leak is being made by correlating pipe failure data. The present correlation covered pipe crack failures, not corrosion failures, in view of the seemingly adequate corrosion resistance of Type 304 SS for piping in sodium systems. Also excluded from this correlation were cases involving cracks longer than the circumference of the pipe, that is, severance failures and their longitudinally cracked kin are excluded.

The crack failure data were obtained from three secondary sources.[1,2,3] Reference 1 provided 66 cases having an extensive definition of material, pipe geometry, crack characteristics, and service exposure. Reference 2 provided one case and Reference 3 provided one case. Austenitic stainless steel was involved in eight of the 68 cases, carbon and low-alloy steel in the remaining cases.

The cracks were circumferentially orientated in 50% of the 68 cases, longitudinally orientated in 16% of the cases, and indefinably orientated in the remaining 34% of the cases. A

1. *W. S. Gibbons and B. D. Hackney. Survey of Piping Failures for the Reactor Primary Coolant Pipe Rupture Study, GEAP-4574. General Electric Company, San Jose, California, May, 1964.*

2. *J. A. Prestele and G. F. Froshauer. "Operating Experience at Indian Point Nuclear Power Station," Nuclear Safety, vol. 6, no. 3, pp. 292-298. Spring 1965.*

3. *Summary of KER Stainless Steel Loop Piping Failures and Status of Investigation on Causes, HW-68125. Available from Division of Technical Information Extension, Oak Ridge, Tennessee, March 1961.*

preliminary data reduction was made to obtain the following parameters for correlation (data range as indicated);

Da Average pipe diameter (range: 1- to 24-in.),

t/Da Wall thickness/average pipe diameter (range: 0.027- to 0.245-in.),

S Pressure stress in pipe wall based on mean diameter formula (range: 460 to 67,200 psi),

Cl/Da Crack length/average pipe diameter (range: 0.020- to 2.72-in.), and

Cl/t Crack lenth/pipe wall thickness (range: 0.27- to 56.0-in.).

Figure 3.D.3 is a histogram of the 68 cases according to the pipe parameter t/Da to show that most of the failure data involves thicker wall piping than is being considered

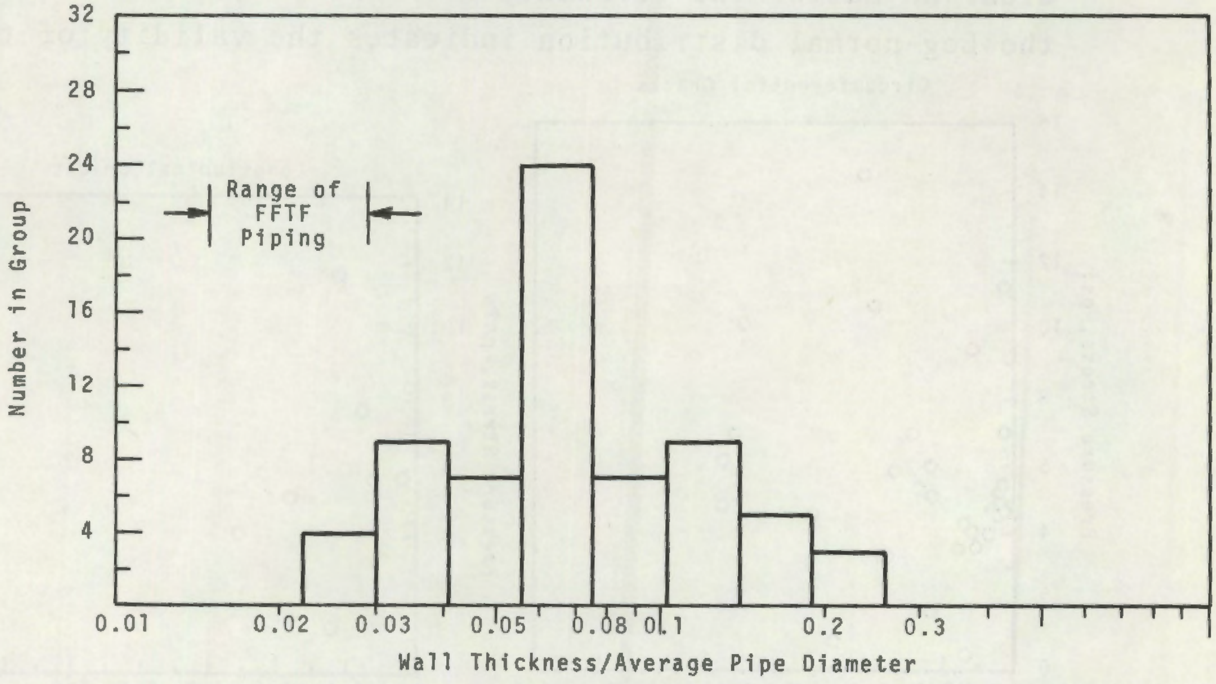


FIGURE 3.D.3. *Histogram of Pipe Failure Cases Wall Thickness/Average Pipe Diameter*

for the FFTF primary sodium system (typically 18-in. OD with 0.50-in. wall for pump discharge piping and 26-in. OD with 0.375-in. wall for suction piping). Figure 3.D.4 shows that the crack parameter Cl/Da is not correlated to pressure stress, S, for either circumferential or longitudinal cracks. Figure 3.D.5 shows that the crack parameter Cl/t is not correlated to stress either.

Figure 3.D.6 is a histogram of the 68 cases according to the crack parameter Cl/Da and Figure 3.D.7 is a histogram according to Cl/t. These two histograms show an approximate normal distribution with respect to the logarithm of the parameter. The proof that these two crack parameters follow the Log-normal distribution, each independently, is given in Figures 3.D.8 and 3.D.9. The general trend of the data is clear no matter the orientation of the cracks. The trend of the Log-normal distribution indicates the validity of the

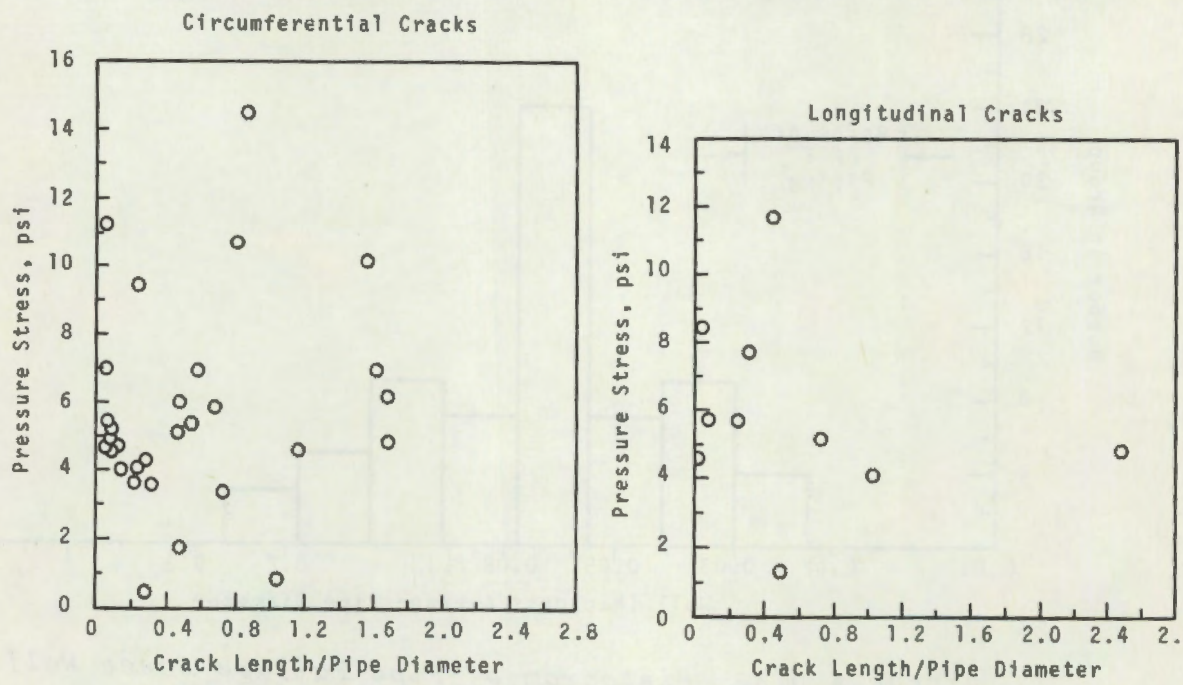


FIGURE 3.D.4. Pressure Stress Versus Crack Length/Diameter

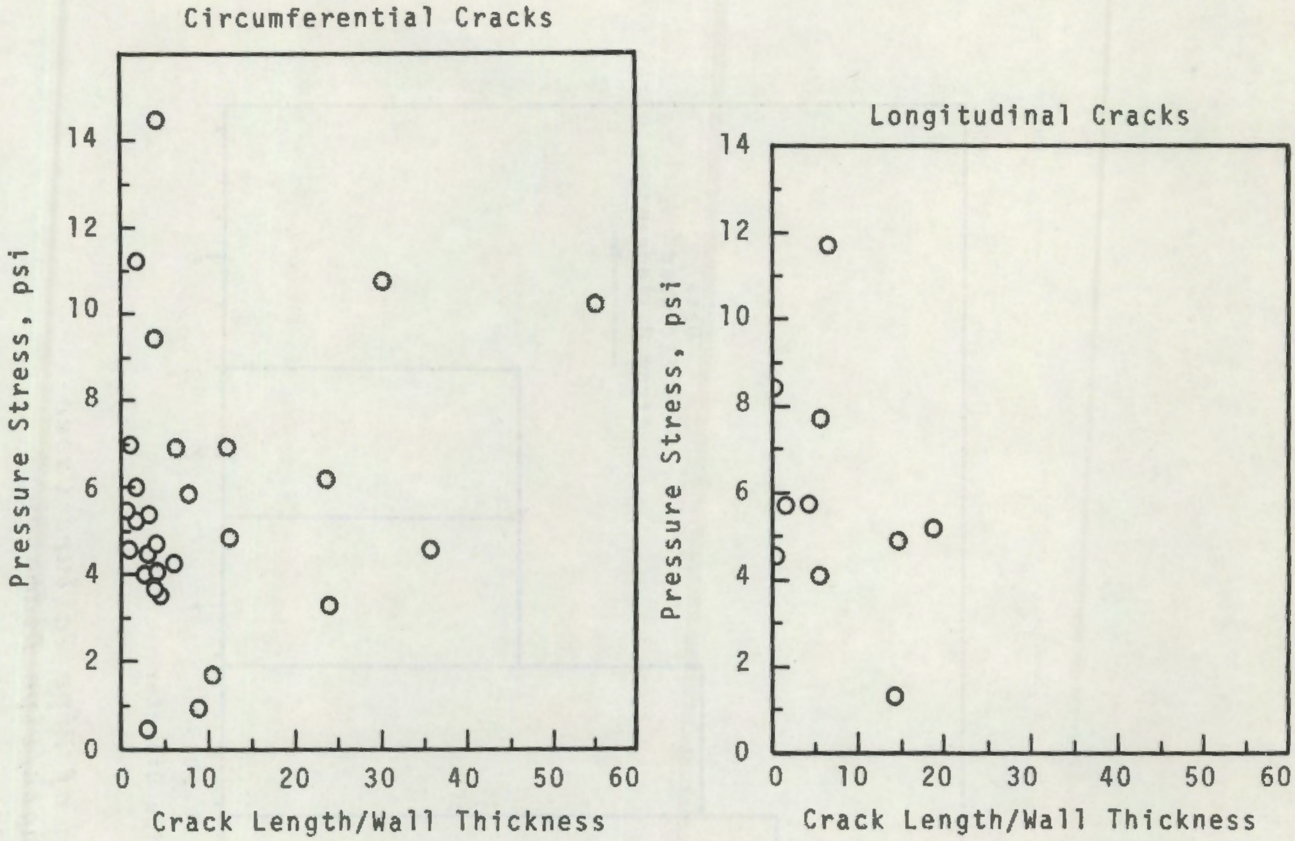


FIGURE 3.D.5. Pressure Stress Versus Crack Length/Wall Thickness

"proportional-effect" theory for correlating these data rather than the "weakest-link" and "parallel-strand" theories of failure.(1)

These correlations coupled with theoretical or experimental data for crack openings will provide estimates of flow area of the leak implied by the crack, a topic for study in the next month.

1. *W. G. Ireson, ed. Reliability Handbook, McGraw-Hill Book Company, New York, New York, 1966; pp 2-6 to 2-8.*

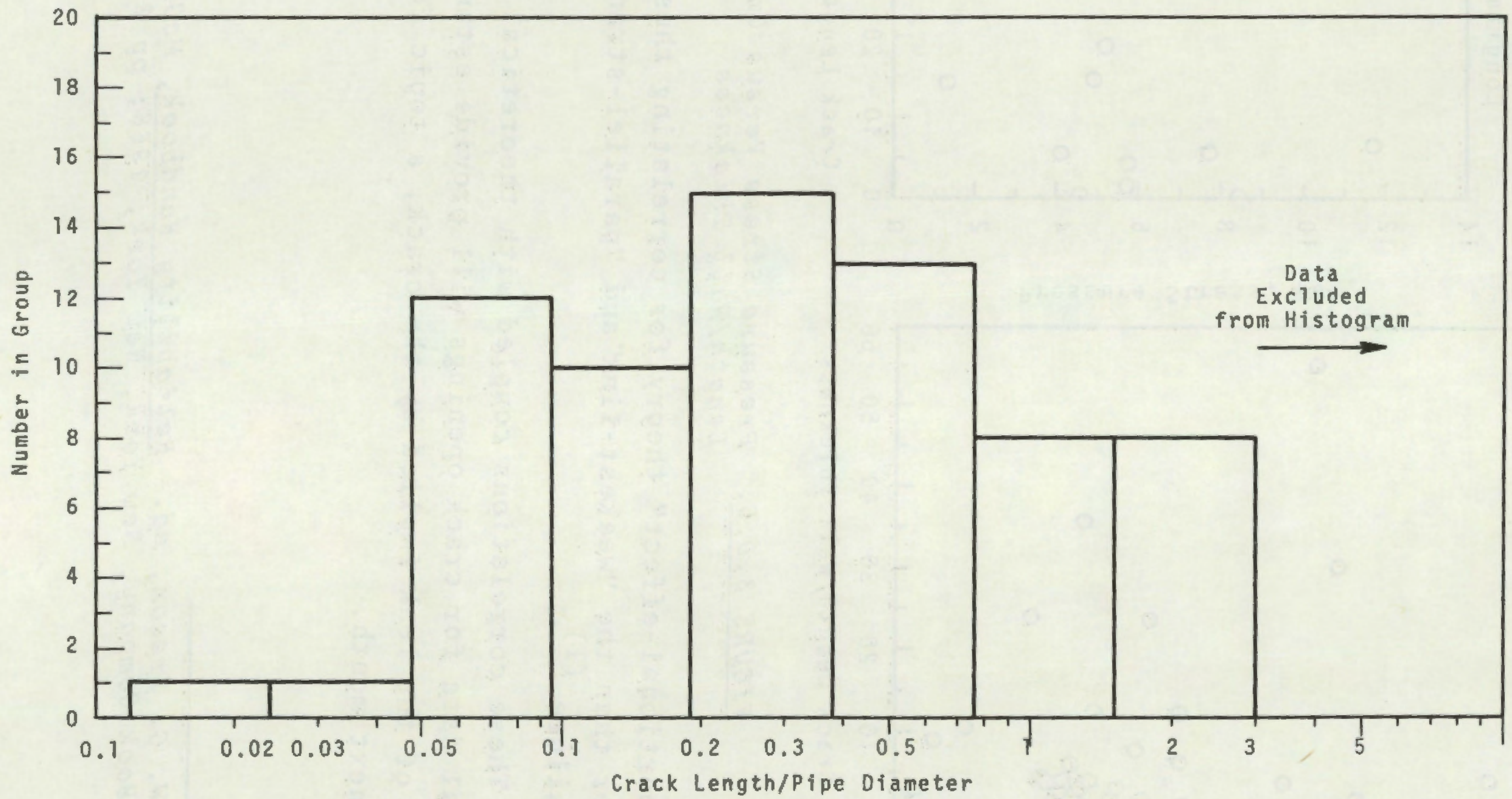


FIGURE 3.D.6. Histogram of Pipe Failure Cases Crack Length/Pipe Diameter (68 Cases)

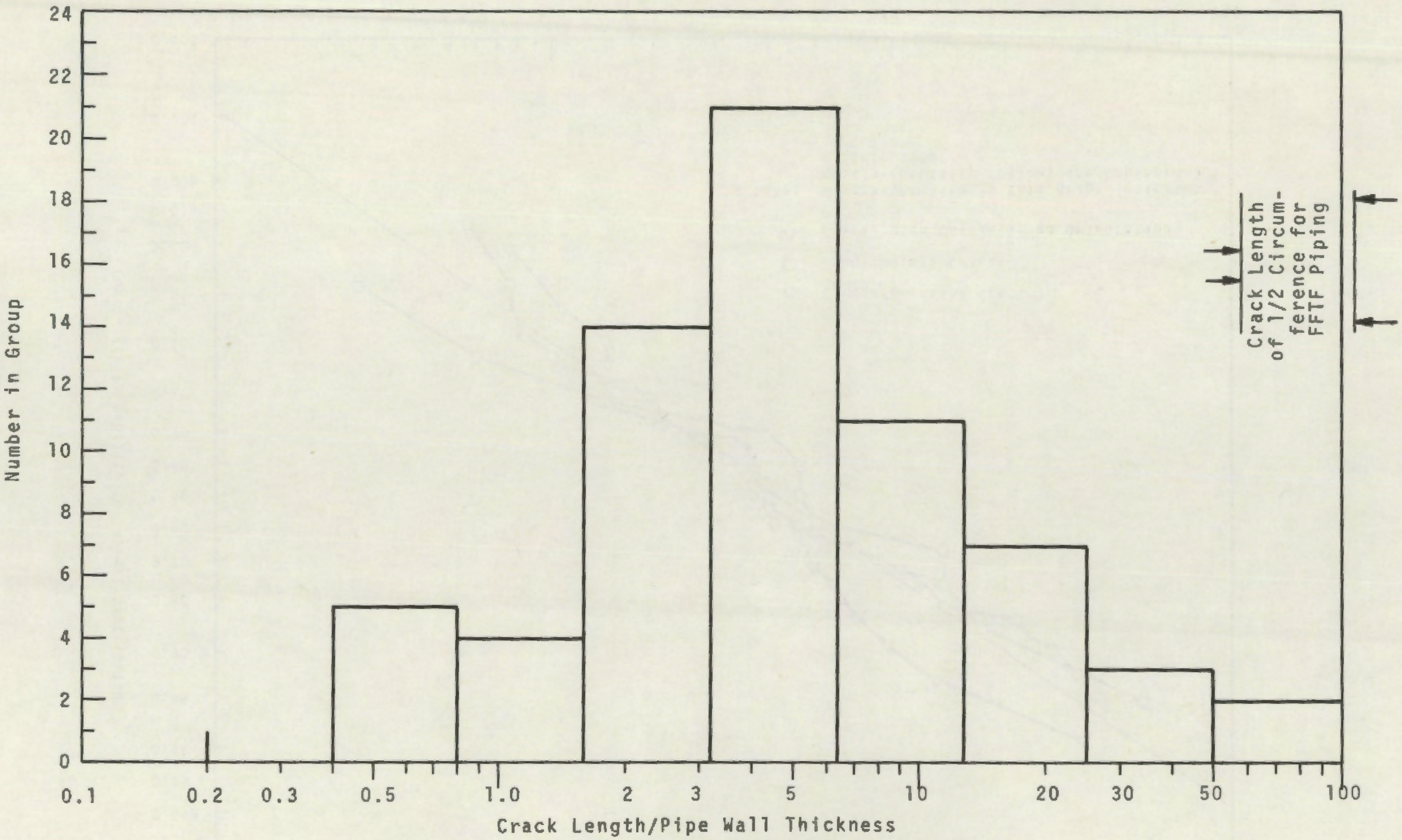
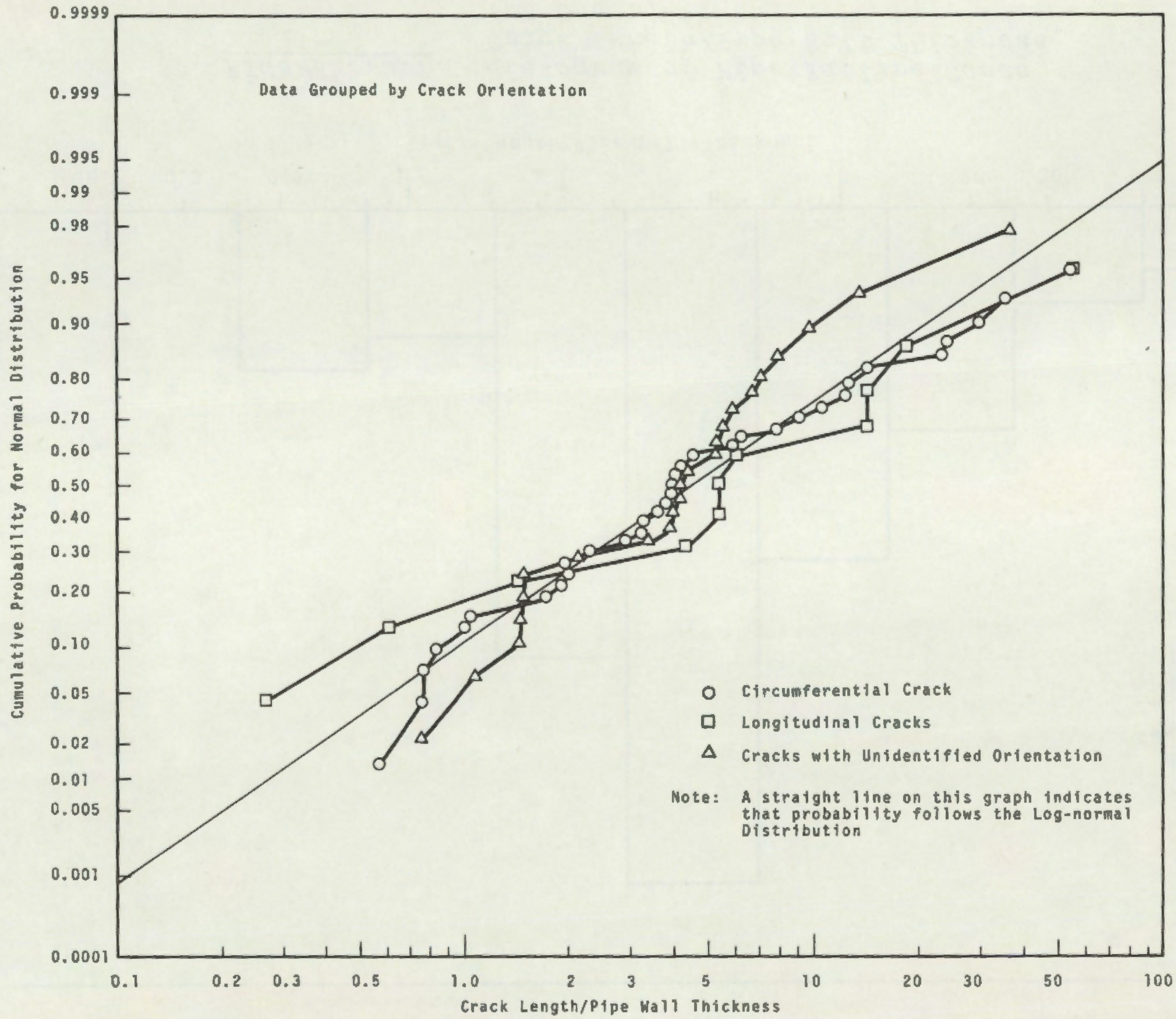


FIGURE 3.D.7. Histogram of Pipe Failure Cases Crack Length/Pipe Wall Thickness (68 Cases)



FFIGURE 3.D.8. *Probability Correlation for Ratio: Crack Length/ Pipe Wall Thickness*

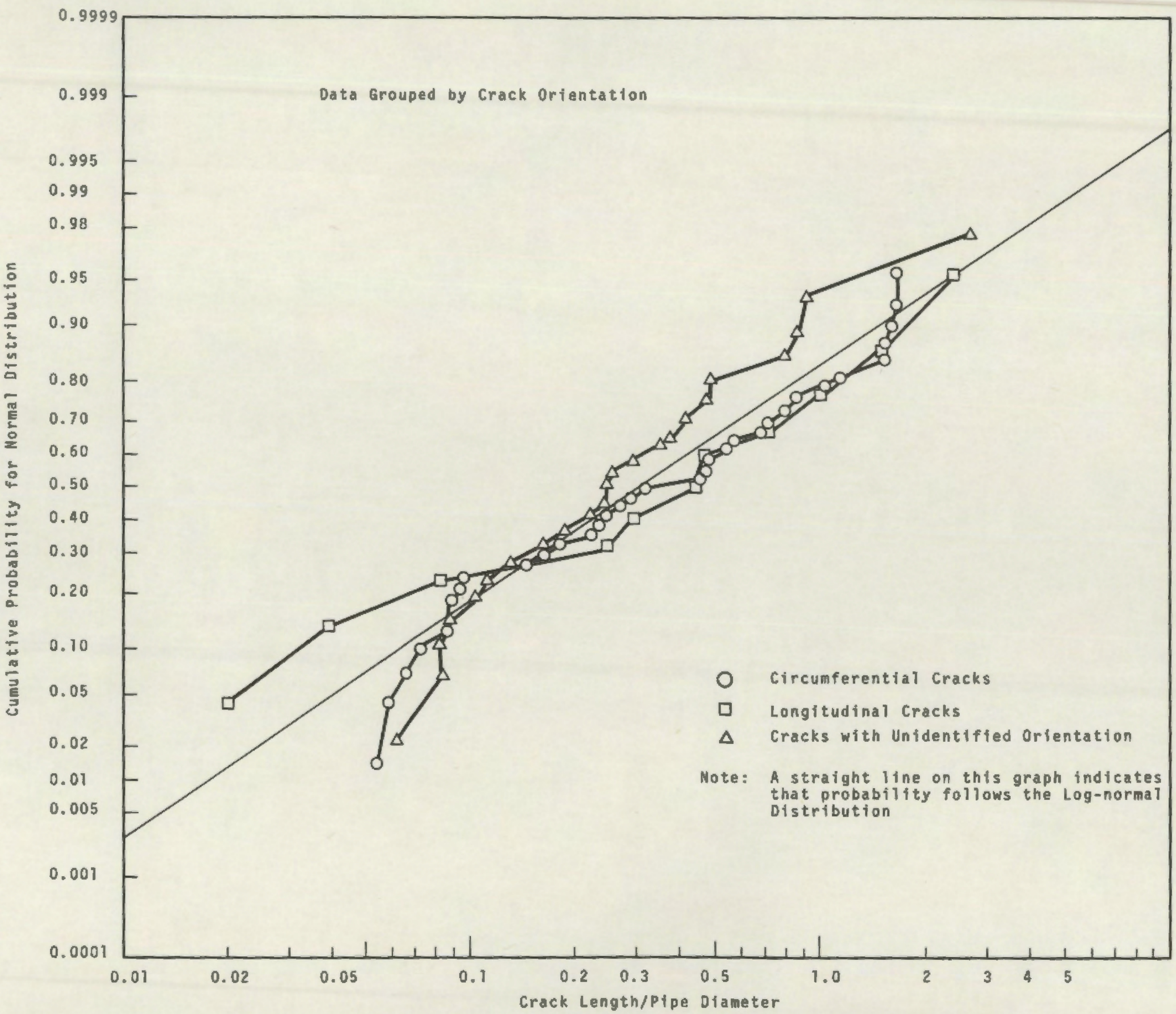


FIGURE 3.D.9. *Probability Correlation for Ratio: Crack Length/Pipe Diameter*

IV. DEVELOPMENT

D. L. Condotta

A. ENGINEERING DEVELOPMENT

J. R. Carrell

1. Sodium Components Development

T. W. Withers

a. Sodium Pumps

W. Babcock

The Westinghouse primary pump design study proceeded according to the new schedule based on the change of conditions from the cold leg to the hot leg location. Drafts of the following Westinghouse reports and specifications were received for comment (key BNW comments are parenthetically inserted):

(1) Revised report on the extrapolation study of the Hallam and Fermi pumps. (The Fermi type sump pump is recommended as the FFTF primary pump.)

(2) Dynamics side study on shaft dynamics and vibration. (A shaft with one sodium bearing and one anti-friction bearing above the mechanical seal is recommended for the FFTF pump.)

(3) Corrosion study report. (300 series stainless steel is suitable for a 20-year pump life at 1200 °F.)

(4) Driver side study report. (A wound rotor motor with liquid rheostat is recommended for the FFTF pump.)

(5) Radiation shielding report. (A design for a nitrogen cooled shield plug has been developed as well as a design for a handling cask.)

(6) Preliminary equipment specification. (A pump and driver preliminary specification is developed to assist in initiation of prototype procurement.)

Pump bearing and seal test requirements and facilities were discussed in a meeting at BNW with representatives from LMEC and RDT present. There appeared to be no near-term bearing or seal test facility at LMEC which could prove FFTF sodium pump bearings or secondary pump seals in time to be useful to the prototype pump designs. LMEC recommended a longer-term facility using the Hallam pumps which they have. BNW recommended that the drivers of the Hallam pump be changed to a speed simulating FFTF operating conditions.

BNW attended a meeting at RDT-HQ, Germantown, on the SPTF. The SPTF facility is expected to be available on a schedule which will permit testing of the FFTF prototype pumps.

2. <u>Equipment Development</u>

C. A. Munro

a. <u>RS-3 -- Control Rod Drive Mechanism Development</u>

J. C. Noakes, P. K. Telford, J. R. McBride, B. C. Fryer, D. E. Rasmussen, and R. Kolowith

The development of the nuclear control rod drive mechanism is directed at providing reliable, predictable, and efficient devices for control and operation of the FTR. Studies and component tests, currently in progress, are being made to support the design effort on the reference drive concepts.

(1) <u>Modified Fermi Latch Tests</u>. Testing of the modified Fermi latch mechanism continued.[1] This mechanism is submerged

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1. *FFTF Monthly Informal Technical Progress Report, BNWL-1006. Battelle-Northwest, Richland, Washington, February 1969. (See this reference for a description of the test arrangement.)*

in water for these initial tests. Figure 4.A.1 shows a cross-section of the control rod latch mechanisms. Examination of the latch fingers after 1651 cycles indicated that the mating surfaces of the fingers and spool were not contacting properly. It was theorized that the cam stroke was too great, causing the fingers to be cammed too far in. This would result in point contact between finger and spool as indicated in the insert of Figure 4.A.1 and by the wear pattern on the fingers shown in Figure 4.A.2. To correct this, the cam stroke was reduced from 0.750 in. to 0.711 in. The latch was then reassembled and operated for 525 cycles. Figure 4.A.3 shows the wear pattern that resulted. It appears that better contact is being made between the fingers and the cam. Cam stroke was reduced to 0.700 in. which should bring the fingers and spool into uniform contact. The latch was reassembled and will be operated for 1000 cycles to evaluate the 0.700-in. stroke setting.

(2) Control Rod Poison Section Dynamic Analysis. FTR Dynamic Analysis estimates on time and cost requirements for modifying the RODDYN computer code to accommodate dynamics analysis of the FTR Concept V-A and experimental verification were completed. Only minor modification to the code will be necessary and will require about four man-days. Experimental verification will require about four months to fabricate and install the required test stand. One month should suffice to complete the verification tests after the test stand has been installed.

b. IT-6 -- Closed Loop Tube and Hardware Development

J. W. Kolb, B. G. Smith, and R. Kolowith

(1) Prototype Closed Loop Tube Design. The prototype detail design drawings of the closed loop re-entrant tube have been circulated for review. The detail design of the prototype was performed to enable prototype tube fabrication development to proceed on schedule.

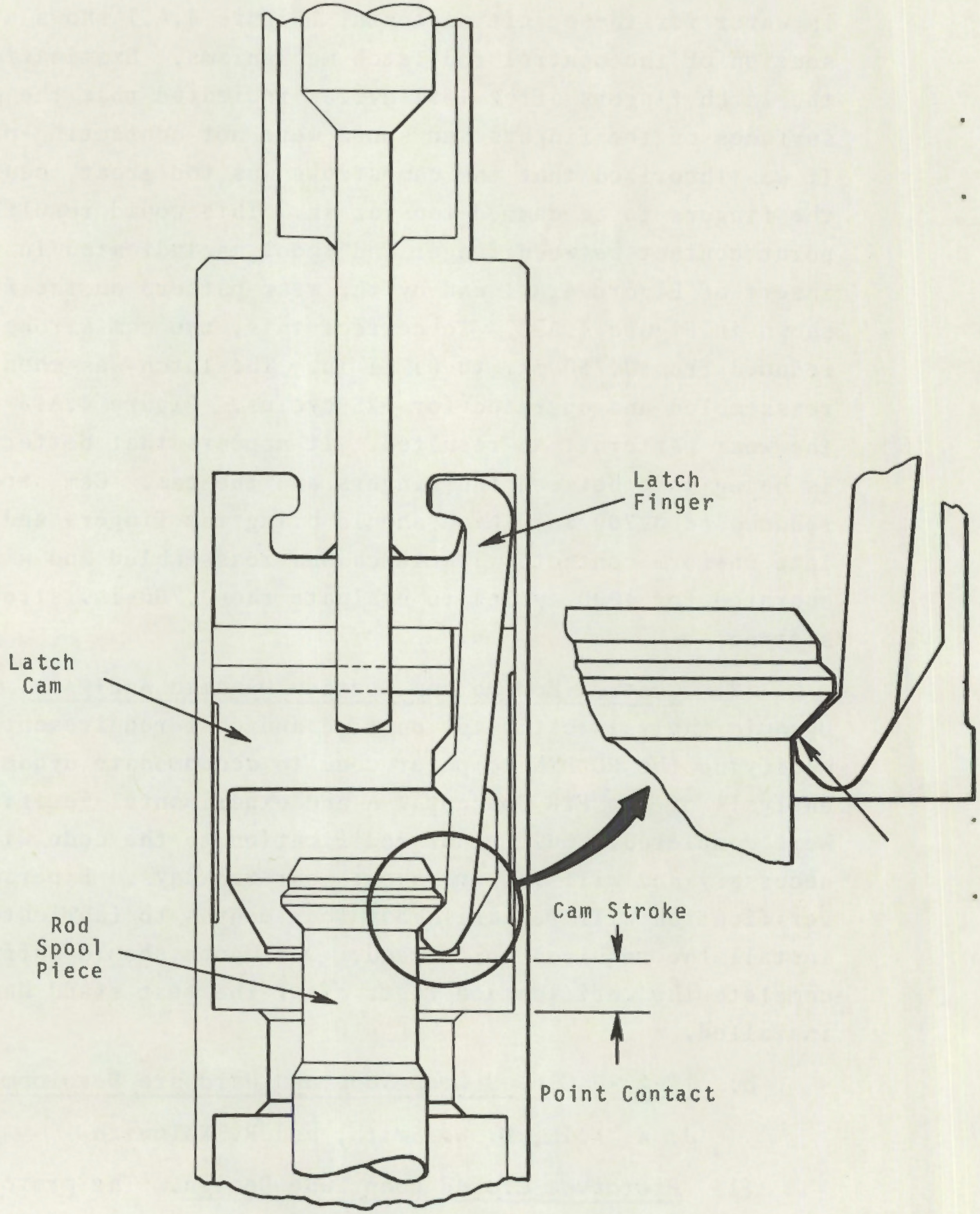
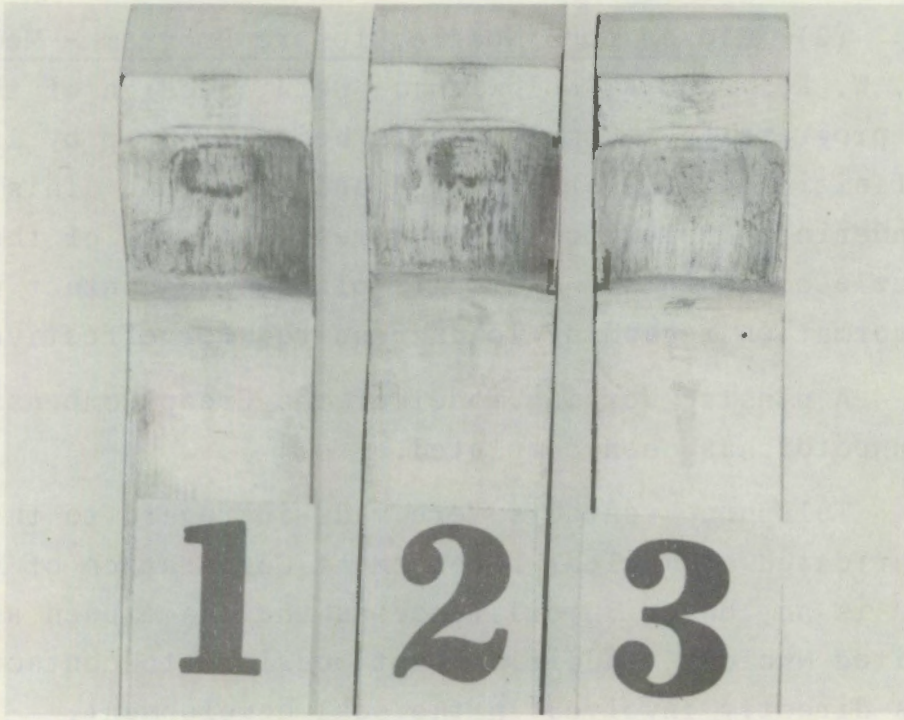
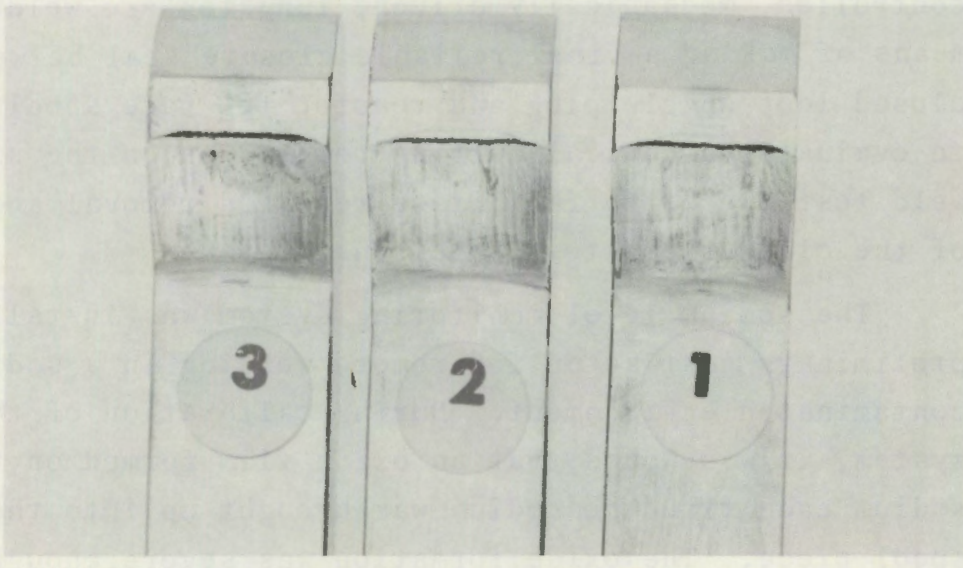


FIGURE 4.A.1. Modified Fermi Latch Cross Section



Neg 0690343-3

FIGURE 4.A.2. *Latch Finger Wear Pattern After 1651 Cycles*



Neg 0690549-2

FIGURE 4.A.3. *Latch Finger Wear Pattern After 2176 Cycles*

(2) <u>Closed Loop Nozzle Closure Program - Mechanical Seals</u> (J. W. Kolb and R. Kolowith). Detail design of the test nozzle is progressing. Completion is being held up by lack of final definition of the spring load and function. This is also hindering completion of the stress analysis of the various nozzle components. A trip is planned to contact vendors to get information regarding loads required for effective sealing.

A plastic working model of the Creep Compensating Connector has been completed.

Telephone contacts were made in regard to the 4-point seal fabricated by Nuclear Development Corporation of America (NDA). NDA is no longer in existence as they have been assimilated by United Nuclear, and it was not possible to contact anyone who was directly involved in the seal development.

(3) <u>Closed Loop Nozzle Closure Program - Welding</u> J. W. Kolb and R. Kolowith). Primary objectives of this task are to demonstrate the feasibility of employing remotely controlled, mechanically driven, tungsten arc welding as a means of making a final reliable closure seal between the closed loop nozzle plug and reactor top face spool piece, and to evaluate mechanical cutting techniques on the simulated weld test pieces to facilitate repeated removal and replacement of the closed loop test specimens.

The sodium level monitoring system was installed in the preliminary test setup for remote welding in a sodium-contaminated environment. During calibration of the level system, it was noted that an oxide film formed on the molten sodium each time the sodium was brought up into the simulated spool piece. The oxide formation was severe enough to affect read-out of the level monitor after repeated operation. The film formed from moisture and oxygen contaminants in the inert gas which were not adequately removed by the chamber purge method. A recirculating gas purification system has been installed in the test system. This will eliminate the necessity of venting the chamber to the atmosphere, which should ensure a higher purity of cover gas and reduce greatly the amount of inert gas required to provide a suitable atmosphere to protect the molten sodium.

The second remotely welded closed loop test specimen has been inspected using ultrasonic testing methods. Nondestructive testing personnel are confident that reliable techniques can be established to determine weld bead integrity. However, it will be necessary to use one welded specimen as a gage, in which holes of various diameter and depth will be drilled into the weld area to provide a basis for calibration of the ultrasonic test equipment.

Future development work requires a large inert gas chamber in which a full-scale mockup of the nozzle spool assembly can be installed and filled with sodium to test prototype welding and cutting equipment under the conditions which will exist in the FFTF. A chamber has been procured from excess CANEL equipment, and detailed drawings of the modifications to the chamber which are necessary to facilitate the prototype test pieces and equipment are being made. These modifications include: a sump to accommodate the test piece and its heating equipment; a "birdhouse" to accommodate a hoist/trolley system to remotely handle the test piece and welding and cutting equipment; and an airlock through which the test pieces and equipment can be passed without disturbing the atmosphere within the chamber.

c. IT-8 -- Closed Loop System Development

R. Kolowith

(1) Prototype Closed Loop Test Plan (R. Kolowith). A preliminary test plan is being prepared for the Closed Loop Prototype. This test plan will include pre-operational,

steady-state, transient, remote operations, accelerated life, and decontamination testing. This preliminary test plan is scheduled for completion by the end of FY-69.

(2) Prototype Closed Loop Development Installation (PCLDI) Project Proposal (K. D. Hayden). A project proposal is being prepared to incorporate the PCLDI in the High Temperature Sodium Facility (Line Item FY-70).

3. Development Facilities

K. G. Toyoda

a. Hydraulic Core Mockup Design and Construction - CM-1

H. Leigh, C. H. Henager, L. M. Polentz, J. J. Dorgan, and L. R. Pierce

The objective of the Hydraulic Core Mockup Design and Construction Program is to provide a facility for evaluation of the FTR core hydraulic and mechanical design problems. The immediate efforts are directed to the installation of a one-half scale reactor model and associated water circulating system. Concurrent with the construction design activities, studies have been initiated to define the operating characteristics of the Hydraulic Core Mockup system. Evaluation of problems associated with transients during such events as startup and system heatup indicate that system pressurizing techniques using nitrogen in the mockup vessel and the storage tank may lead to fluctuation of liquid level in the mockup vessel. It will be necessary to bypass a large volume of flow at startup to allow the nitrogen pressure control system to compensate for the pressure changes.

A major step in the preparation of the pit to house the mockup vessels was completed with completion of pile driving in the 321 Building.

b. Hydraulic Core Mockup Test Program - CM-2

H. Leigh, J. Muraoka, D. L. Ballard, P. M. Jackson, L. R. Sweetin, J. Spalek, and G. W. Riedeman

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

(1) Inlet Region Model. Approximately 25 test runs have been made to define inlet conditions and core flow distribution using the one-third scale inlet model with the toroidal enclosure and dispersed core inlet plenum arrangement. The test examined the effects of various parameters on the flow distribution. The parameters were:

- Total core flow - 10 to 75% of design
- Number of inlet nozzles - 4, 6, and 8 in service
- Discharge angle of inlet nozzle - 0, 45, 67.5 and 90 degrees.

Inlet pressure-drop data were not recorded except in the most recent tests. However, individual duct flows were recorded which provide a good basis to evaluate the effects of these parameters. The data indicate that the effect of the parameters on the core flow distribution is in the order of 1 to 2%, or essentially negligible. In other words, the pressure drop in the torus is small and, consequently, the variations in the pressure drop have a negligible effect on the duct flow distribution.

Flow mixing within the plenum was observed by injecting dye into individual downcomer streams. There was basically no mixing in the inlet plenum with the downcomers in the 0 degree (directed toward the core center) position. The coolant spread is about 30 degrees toward the adjacent paired inlet and about 60 degrees toward the next pair of inlets. Varying the number of downcomers in operation had little effect on the coolant mixing as evidenced by the injected dye.

Dye injected with the downcomers rotated to the 90 degree position demonstrated a significant degree of mixing. In this test there was a decided split in the coolant: a portion (approximately 1/3) moved upward in the torus chamber into a low velocity zone and immediately entered the inlet plenum about 90 degrees downstream. The remainder of the colored coolant remained in the higher velocity zone at the bottom of the torus where it immediately was swept the full 360 degrees through the torus. From here, this fully mixed coolant was fed into all portions of the inlet plenum. Shutting off the flow of adjacent downcomers had little effect on the mixing.

(2) Instrumentation - Conductivity Measurements. Conductivity measurements provide a means to determine fluid mixing and flow distribution characteristics in the hydraulic models. Conductivity measurements may be quantitative or used with noise analysis techniques for application to the HCM program. Much of the basic sensor and circuitry work has been established by Atomics International. However, the differences in the sizes of the models (i.e., the A-I test models versus the larger HCM models) require demonstration of satisfactory application of techniques and circuitry. As part of this effort, BNW Instrument Research and Development Department has fabricated a breadboard of their new conductivity signal conditioning circuit. Although no quantitative measurements have been made, the circuit appears superior to the original one now in service. Electrolyte ($NaNO_3$) concentration variations of 0.3 ppm were easily detected and, at 6 ppm electrolyte injection, the output signal voltage was approximately 100 times the overall system noise voltage. Several new circuit modifications are now being tested to optimize the particular circuit best suited for measuring absolute conductivity at a single electrode position. Some circuits appear promising but work continues.

Experiments with single electrode sensors for measuring single point solution conductivity are underway. This technique utilizes the phenomenon that solution resistivity is a function of current density. The current density is much greater around a small surface electrode than one with large surface for a given cell current. In this system, the sensor is one electrode, and the system is the other. The sensor responds mainly to variations in its immediate vicinity. The technique works, but problems in nonuniform sensor-to-sensor sensitivity and response characteristics may limit its usefulness. Work continues because if the technique can be made practical, vessel instrumentation can be greatly simplified.

(3) Outlet Region Model - Gas Entrainment Techniques. Two methods for measuring gas content in the coolant (gas entrainment) during outlet region model tests were investigated as follows.

(a) Periodic sampling method, using mamometric and gas chromatography techniques.

(b) On-line (real-time), continuous gas detection method. This method is functionally more attractive since it does not involve the problem of selecting a "representative" fluid volume sample with inherent analysis time delays, and it provides a continuous (on-line) survey of the gas content of the coolant stream.

The specific techniques investigated were the standard nondestructive testing ones -- real-time X-ray and neutron radiography and ultrasonics. X-ray radiography is impractical for scanning through more than $\sim$10 in. of water because of high Compton scattering of X-rays in water in the energy domains accessible with economical X-ray machines. Fast neutron radiography was eliminated only because of the high cost and unavailability of intense fast neutron sources.

Finally, the ultrasonic method proved to be both technically feasible and economical. An up-to-date literature survey of ultrasonic nondestructive testing methods was made, as well as a survey of ultrasonic commercial equipment manufacturers. The latter survey yielded an economical, commercially available cavitation meter, suitable for use on the HCM.

c. Sodium Core Development Facilities - CM-3

K. D. Hayden

The objective of the sodium development facilities program is to establish the technical basis and requirements of sodium core development facilities responsive to the long- and short-range needs of the FFTF program, and to provide the necessary technical support for acquisition of facilities. The project proposal for the Fast Reactor Thermal Engineering Facility (BCE-037) was sent to RL-AEC on February 27, 1969. Total project cost was estimated at $876,000 plus $95,000 transferred capital equipment. The sodium loop will provide the capability for evaluating the heat transfer characteristics of various types of LMFBR test fuel assemblies. Loop operating conditions are expected to be 600 gal/min, 250 psi head, 1200 °F and 1-1/2 MW electrical heating capacity to the test section.

Design and related service funds of $80,000 were authorized February 11 for the LMFBR Core Segment Development Facility (BAP-032). Work Authorities have been issued to both BNW and HES.

The Sodium Facilities Building (BAP-027) will be used to house several nonradioactive sodium loops and other test equipment. Lump sum bids from six firms to construct the 60 ft by 100 ft by 24 ft high building plus a lunchroom and restroom were opened on March 12, 1969. The estimated low bid, which was within $1,000 of the Government estimate, puts the total project costs within the total amount authorized and the building costs (as defined in AEC Manual 1301) below the $100,000 limitation.

d. Sodium Facilities Operation - TF-3

J. D. Berg and L. J. Defferding

The objectives of this section and description of equipment were given in BNWL-915.[1] Status of facilities is given in the Table 4.A.I.

TABLE 4.A.I. Sodium Facilities Status - March 1969

Small Component Evaluation Loop	Following an operational checkout, the loop was shut down. The modification required to tie-in the Sodium Sampler Loop is in progress.
Radioisotope Transport Loop	The sodium leak (reported last month) at the inlet header of the sodium-to-air heat exchanger developed from a thermal stress crack. This loop is being completely disassembled for gamma scanning and future upgrading.
Small Heat Transfer Loop	This loop operated ∿300 hr during this report period. The purpose of this run was to check the integrity of the "Gray-Loc" mechanical seals in the test section. The temperature was cycled from 800 to 1200 to 800 °F with temperature changes limited to 100 °F/hr. The mechanical seals were inspected for leakage but none were observed. A work schedule and CPM chart were prepared for the modification to increase loop flow and thermal capacities for FFTF 7-pin testing.

1. *FFTF Monthly Informal Technical Progress Report, BNWL-915. Battelle-Northwest, Richland, Washington, October 1968.*

TABLE 4.A.I. (contd)

Static Sodium Pots	The testing of pressure transducers in Pot No. 3 continued. Pots No. 1 and No. 4 were used intermittently while Pot No. 2 was idle.
Sodium Purification Loop	The loop was shut down 2-7-69 to install an in-line carbon meter probe. Following this installation, the loop was returned to operation with both the carbon meter and an oxygen probe in service.
Fission Product Screening Loop	The checkout of the primary loop continued up to 1200 °F. The loop was shut down on 2-7-69 to correct minor operational problems and to install calibrated flowmeter (to be utilized to calibrate the normal loop flowmeters).
Cover Gas Evaluation Loop (Title changed from Nitriding Test Loop)	Fabrication continued and installation was initiated.

e. Fin Tube Testing - T-2

D. J. Lagrou

The final copy of the Test Request for Testing of Commercially Available Fin Tubing (for FFTF Heat Dumps), Test T-2, was transmitted to LMEC on March 4. The LMEC Test Plan is expected to be prepared and transmitted to BNW/FFTF in about 30 days.

The fin tube test specimen pipe, 1.5-in., Sch. 10S, Type 304 seamless austenitic stainless steel, was received during this reporting period and is currently being inspected by liquid penetrant and ultrasonic tests to determine the integrity of the pipe wall and surface condition.

f. Test Status

Table 4.A.II lists tests administered or processed by Engineering Testing Services in which the status changed during the reporting period. Objectives were given in BNWL-915.[1]

TABLE 4.A.II. Engineering Testing Services - FFTF Tests

Test No.	Title	Status
T-2	Fin Tube Test - Heat Transport DHX - Proposed LMEC (LCTL)	Test request sent to LMEC for their acceptance as test performer.
T-3	Single Duct Test - Reactor Core - BNW	Test program being re-evaluated. Selection of the new reference core concept has eliminated the need for this test as originally scoped.
T-8	Material Interface Effects in Sodium/ PMIS - Reactor Core - LMEC	Rev. 1. to the test request sent to LMEC for their acceptance.

1. *FFTF Monthly Informal Technical Progress Report, BNWL-915. Battelle-Northwest, Richland, Washington, October 1968.*

B. COOLANT AND PROCESS DEVELOPMENT

B. M. Johnson

1. Corrosion and Chemistry

D. W. Shannon

a. Evaluation of Radioisotope Transport and Deposition

W. F. Brehm, E. A. Kovacevich, and J. J. McCown

The purpose of this task is to determine the magnitude and location of radioactive material deposits that will be produced in FFTF operation, and to investigate possible control measures.

(1) Radioisotope Transport Loop (RTL). Extensive gamma scanning studies are being made of the cutup sections of the RTL, which was shut down after 754 hr operation with a high ^{60}Co source[1] and these operating conditions: 1300 °F specimen temperature; 900 °F minimum temperature; 1.5 gal/min sodium flow; and 10 ppm oxygen in the sodium. The most important features shown by the counting to date are these:

- ^{60}Co and ^{51}Cr were found mostly in the hot leg of the loop, their concentrations decreasing with distance from the source. Smaller concentrations were found in the heat exchanger and in the regions between the HX and secondary inlet. Practically no radioactive cobalt or chromium was found in the secondary or the return leg to the heater.
- Both cobalt and chromium, like manganese,[2] are deposited preferentially at higher-velocity sections in both hot and cold legs.

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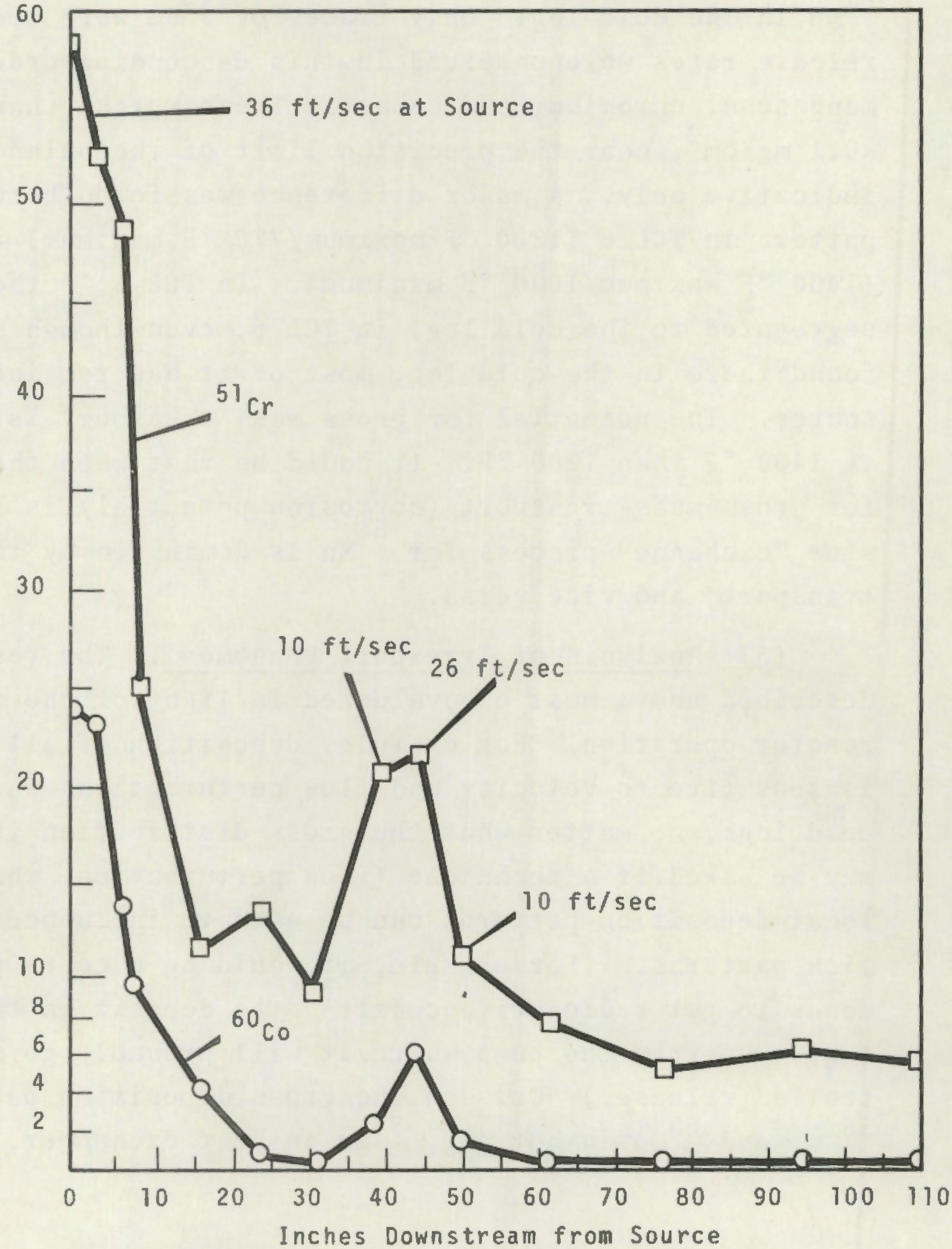
1. *FFTF Monthly Informal Technical Progress Report, BNWL-990. Battelle-Northwest, Richland, Washington, January 1969.*
2. *W. F. Brehm, E. A. Kovacevich, and D. W. Shannon. Radioisotope Transport and Deposition in Flowing Sodium, BNWL-969. Battelle-Northwest, Richland, Washington, February 1969.*

- Much more ^{51}Cr than ^{60}Co is released. This agrees with G.E. and Brookhaven data. The chromium migrates farther in the loop than does cobalt. Deposition profiles of the isotopes, showing the preferential transport of chromium and the preferential deposition at higher velocity, are given in Figure 4.B.1.
- Increased deposition of both ^{51}Cr and ^{60}Co near flow perturbations was seen. More detailed counting is being done in those regions.
- The ^{54}Mn that had been absorbed in the pipe walls in previous loop tests continued its migration to the cold leg and coldtrap system. Practically all the ^{54}Mn has left the hot leg as a result of corrosion of the pipe walls. (No ^{54}Mn was present in the source used in the last run.)
- Only traces of ^{59}Fe were found in the hot leg. The $^{60}Co/^{59}Fe$ ratio was about 250/1 compared with about 50 in the source. The lack of significant iron transport remains a puzzle.
- Sections of the loop were counted both before and after cleaning residual sodium from the pipes. There was no difference in the counting rate, indicating that the deposited activity is firmly attached to the steel and cannot be removed by sodium cleaning operations.

Sections of the specimen holder, hot leg, heat exchanger, and cold leg, were submitted for metallography and microprobe examination.

(2) Thermal Convection Loops. TC Loop 6, run 816 hr at 1200 °F maximum/770 °F minimum, with ^{51}Co, ^{54}Mn, ^{60}Co, and ^{59}Fe in the source, was counted. The ring specimens were weighed. The isotopes ^{51}Cr, ^{60}Co, ^{54}Mn were found in the loop, with only



Uncertainty - ±10%. Temperature - 1300 °F. Test time 754 hr. Velocity 2.5 ft/sec except where noted. Cr/Co ratio in source - 1.2

FIGURE 4.B.1. Comparison of ^{51}Cr and ^{60}Co Activity in RTL Hot Leg

^{54}Mn in the cold leg. Only traces of ^{59}Fe were seen. Relative release rates were observed in this descending order: manganese, chromium, and cobalt. These weight changes were <0.1 mg/cm^2, near the precision limit of the balance, hence are indicative only. A major difference was found between the ^{54}Mn pattern in TCL 6 (1200 °F maximum/770 °F minimum) and TCL 5 (1400 °F maximum/1000 °F minimum). In TCL 5[1] the ^{54}Mn was segregated to the cold leg; in TCL 6, even though some ^{54}Mn was found there in the cold leg, most of it had remained near the source. The potential for gross mass transport is much greater at 1400 °F than 1200 °F. It could be that when the tendency for gross mass transport (corrosion potential) is high, the slow "exchange" process for ^{54}Mn is dominated by the mass transport, and vice versa.

(3) Analysis of Transport Phenomena. The results described above must be evaluated in light of the impact on reactor operation. For example, deposition of all three isotopes is sensitive to velocity and flow perturbations in both hot and cold legs, no matter what the gross distribution is. Hence, it may be asked if a technique (flow perturbation) which affects local deposition patterns can be used to influence gross deposition patterns. (For example, it would be nice to provide some means to get radioactive cobalt[2] to deposit in the FTR cold leg, away from the pump where it will probably go in an uncontrolled release.) Or, can the gross deposition patterns (^{51}Cr, ^{58}Co, and ^{60}Co in hot leg, ^{54}Mn in heat exchanger, cold leg, and

1. *W. F. Brehm, E. A. Kovacevich, and D. W. Shannon. Radioisotope Transport and Deposition in Flowing Sodium, BNWL-969. Battelle-Northwest, Richland, Washington, February 1969. p. 14.*
2. *Both ^{58}Co and ^{60}Co will be produced in FFTF. The ^{60}Co is easier to obtain for experiments so it is generally the one used.*

cold trap) be accepted, and design deposition traps be put in the appropriate locations? The latter approach accentuates the natural deposition tendencies and appears more promising.

Some questions about "natural deposition tendencies" of ^{54}Mn also remain. The deposition preference of ^{54}Mn seems to be influenced by loop temperature and/or corrosion potential. Is the deposition of ^{54}Mn in the hot leg favored by the reduction of temperature from 1400 to 1200 °F. Or does the corrosion potential change in the low-velocity TC loops over that temperature range, produce an effect which would not be seen in larger loops and reactors even at much lower temperatures? To answer these questions with regard to FFTF application, release and deposition tests need to be done in a high velocity loop at 900 to 600 °F, which the FFTF temperature is at the start of operation. Such tests are proposed for the rebuilt RTL, with the following test sequence:

(a) One month at 900 to 600 °F to determine radioisotope deposition patterns at FFTF startup conditions.

(b) One month at 1300 to 1200 °F to check deposition conditions in the absence of large corrosion potential. This will check out the question of ^{54}Mn deposition raised above.

(c) One month at 1300 to 900 °F with cold trap flow reduced to 1% of total (as in FFTF closed loops) to determine the effectiveness of the cold trap in retaining manganese.

This information is necessary for successful trap design for large reactor systems.

b. <u>Selection and Testing of Radioactive Sodium Analysis Methods</u>

(1) <u>High Level Analytical Facility Design</u> (J. F. Jarosch, L. F. Lust, and J. J. McCown). A preliminary draft of the System 85 CSDD for the FFTF analytical facility was received

from Bechtel for review and comment. An extensive review was completed and the major comments transmitted to Bechtel by John McCown.

BNW did not concur with the Bechtel CSDD draft and a meeting was held in San Francisco to resolve the problems. An acceptable layout for the sampling cell, analytical cells and support labs was presented to Bechtel by BNW. Bechtel concurred with the changes and is revising the draft.

The review emphasized the urgent need for hot cell mockup studies at BNW under Task SG-3 by pointing out a number of decisions which must be made based on the results of these tests.

c. <u>Selection and Testing of FFTF Sodium Sampling Methods</u>

K. C. Knoll, T. J. Owen, and J. Gibson

The installation of the remotely operable removable section bypass loop sampler on the 314 Building sodium loop is about 75% completed. Completion is awaiting the receipt of replacements for a number of trace heaters which were received in an incorrect size.

Crimp sealing of 1/2 in. diameter nickel sample tubing is under investigation. Room temperature seals of as-received tubing have not yet been produced. Room temperature crimp sealing of empty annealed nickel tubing has been successful, provided an annealing temperature above 600 °C is maintained.

Crimp sealing of a 1-foot section of annealed nickel tubing filled with sodium has been attempted. The tube did not rupture but the section between the crimps increased 0.04 in. in diameter. The seals obtained in this initial attempt were not hermetic. It is hoped that the proper preconditioning of the inner surface of the nickel will permit sealing.

d. Selection and Testing of Cover Gas Instruments and Sampling Systems

G. B. Barton

(1) Mass Spectrometry. A ratio measurement, based on the use of argon as an internal standard, has been evaluated for the measurement of impurities in argon. Improvement in noise level and detection limit has been obtained. It appears possible to measure methane down to approximately 5 ppm with an uncertainty of 5 ppm. A greater uncertainty has been observed for hydrogen. This represents an improvement by a factor of 10 over previous tests.

The source of much of the noise (and the resulting uncertainty) has been traced to the 300 V power supply of the instrument. It apparently is not adequately compensating for variation in the line voltage. Efforts are underway to overcome this malfunction.

2. Thermal Hydraulics

J. M. Yatabe

a. Heat Transfer Characteristics of FFTF Fuel Assemblies - Task CTH-2

The FY-1969 objective of this program is to develop bundles of electrically heated pins exactly modeling FFTF driver fuel assemblies and perform first sodium and then thermal-hydraulic testing in support of FFTF design and for general use in the LMFBR Program. The current effort is concentrated on developing improved quality assurance procedures for each pin, and in developing the capability and quality assurance for bundle testing.

Development of quality assurance methods to determine the boron nitride density and heat flux uniformity is in progress. The attached photograph, Figure 4.B.2, shows the results

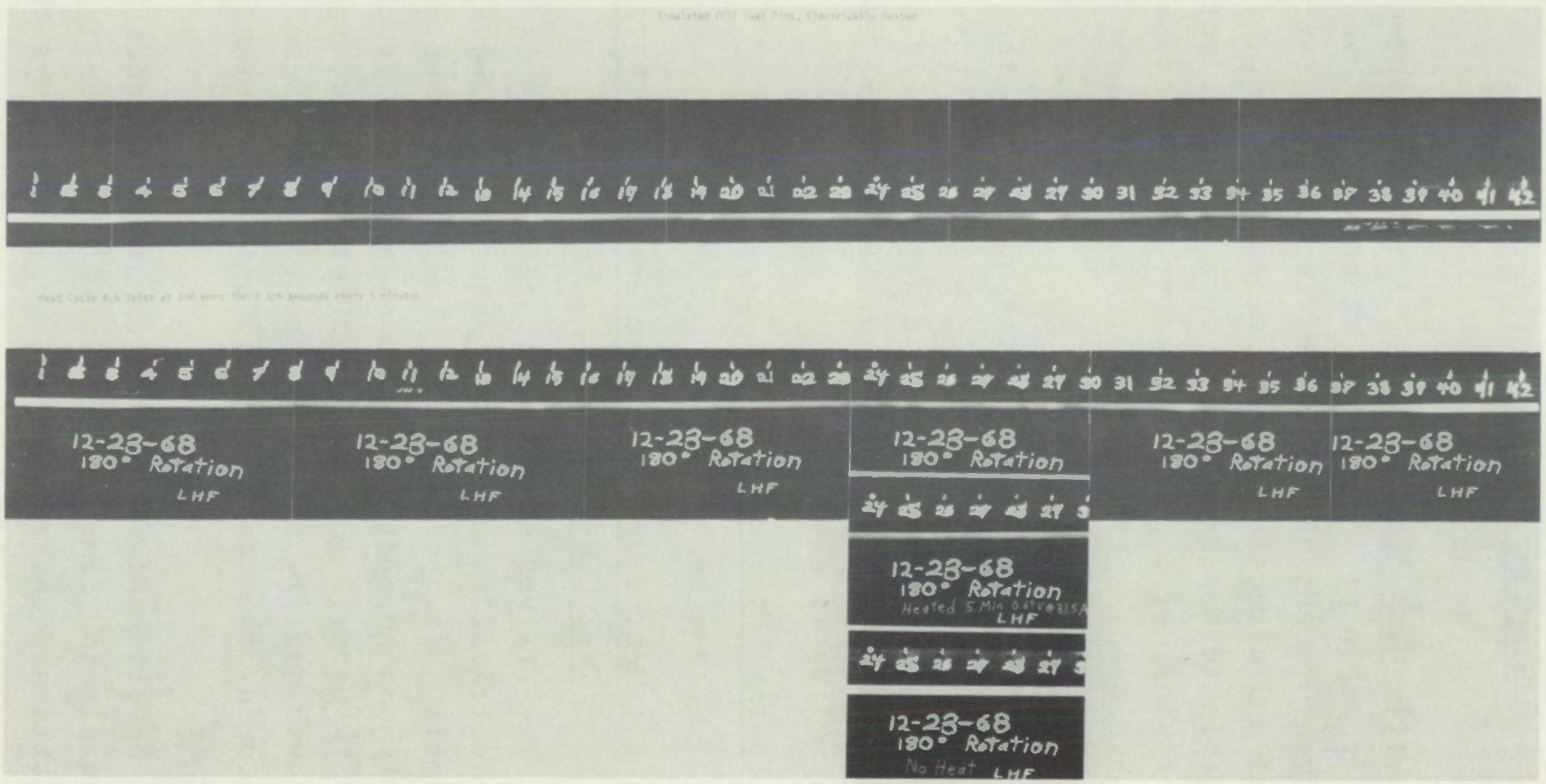


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FIGURE 4.B.2. Results of Phosphorescent Test to Determine the Uniformity of Heat Flux From an Electrically Heated Fuel Pin Model

achieved with the phosphorescent technique. The pin was coated with a sensitized phosphor and the center heater element was then given a pulse of electric current. The photograph was taken 3.5 seconds after the pulse was initiated. The light areas at 14.5, 25, 29, and 33 in. indicate poor heat transfer and agree with boron nitride density measurements made from neutron radiographs of the same pin. Further development is continuing to determine quantitative results.

Fabrication of the 7-pin bundle flow housing, electrical connections, and instrumentation leads is approximately 40% complete.

The circle generator required for the electron beam welder to permit seal welding of the pins into the electrical head of the bundle is complete.

The upgrade of the Small Sodium Heat Transfer System is in progress. All components are scheduled for delivery and initial construction work orders and material purchase orders issued. Specific orders for material items were delayed pending AEC approval of expenditure. A quantitative estimate of the delay in loop modification due to this extra approval step will be made based on the critical path method used for detailed schedule planning.

In the development of high heat flux pins for specific accident tests (e.g., $Q/A \geq 1{,}000{,}000$ Btu/hr-ft^2 with surface temperature up to 1800 °F), BNW is considering the use of heaters developed at Atomics International as back-up to the refractory metal heaters being developed under this program. Specifications for these heaters have been formulated and forwarded to the AEC/RDT for approval before detailed negotiations are initiated to purchase three prototypes for evaluation in sodium.

b. Sodium Flow, Hydraulic and Mechanical Testing of FFTF Fuel Assemblies - Task CFD-3

(1) Coolant Mixing and Flow Distribution. The FY-1969 objective of this task is to provide flow distribution and coolant mixing data for the thermal analysis of the fuel design. The activities during this reporting period include:

- Installation of a full size prototype fuel assembly. The assembly has been modified to permit local direct sampling of injected salt solution to measure coolant mixing. Test plans have been prepared and testing will begin during the next reporting period.
- Completion of the dual channel test assembly to perform final verification of the capability of noise analysis and absolute resistance techniques to make detailed determination of local coolant flow distribution and mixing. Testing has been started. Electrical parallel path problems (e.g., leakage through the conducting solution to ground) have affected the operation of the conductivity probes (the basic test sensors). The problem is being solved with the assistance of the noise analysis off-site consulting group.

(2) Sodium Flow Testing. The purpose of the sodium flow tests is to evaluate the effect of potential long-term failure modes on FFTF fuel assemblies. The Mark I assembly, a full sized fabrication prototype, has now been tested to approximately 2900 hr at full flow and temperature conditions (400 gal/min, 1060 °F) in the Core Component Test Loop at Argonne National Laboratory.

Postinspection tests of the 7-pin bundle of the same configuration which was exposed to flowing, heated sodium for over 9,000 hr have been completed and compiled into a draft report.

(3) Mechanical Testing. It has been postulated that wire wraps may loosen during reactor operation as a result of differential swelling. If movement of this slack wire could occur relative to the bundle, the coolant channel configuration could change and cause hot spots. In addition, increased vibration amplitudes in the assembly could result, leading to more rapid wear rates.

As the first step in the evaluation of the problem, a demonstration 19-pin bundle with wires of varying degrees of tension has been fabricated. The bundle will be used to identify qualitatively the potential for bundle deformation. The next step will be to consider small bundle sodium life tests with slack wire wraps.

c. Channel Instrumentation Development - Task IC-23

The major FY-1969 goal is to develop and evaluate individual fuel assembly, fission product gas disengagement devices to be used in the FEDAL system to locate failed fuel assemblies. The present concept calls for the use of a vortex generator to concentrate the released gas bubbles to the sampling core located in the center of the duct.

In the last reporting period, details of the state-of-the-art survey and the feasibility analyses were described. During this period:

- The design of several prototype vortex generator designs to be used in the test program has been 60% completed.
- The techniques of fabrication and fabrication quality assurance are being developed.
- Modifications of the flow system and data collection system to perform the first tests is about 40% complete.

In addition, other Commission funded programs which may provide supporting information are being actively surveyed.

d. Gas Cooling of FFTF Fuel Assemblies During Inspection - Task FE-7

The objective of this program is to determine the requirements for gas cooling of discharged FFTF fuel assemblies during examination and disassembly. The cooling will be performed by axial flow, transverse flow, and a combination of the two.

The studies with pure axial flow and pure transverse flow on the electrically heated 169-pin model fuel assembly with 0.250 in. pin diameter and 0.030 in. spacing have been completed. Data analysis of these tests has not been completed. The experimental study of transition from axial to transverse flow on the same fuel assembly model has been started.

3. Instrumentation Development

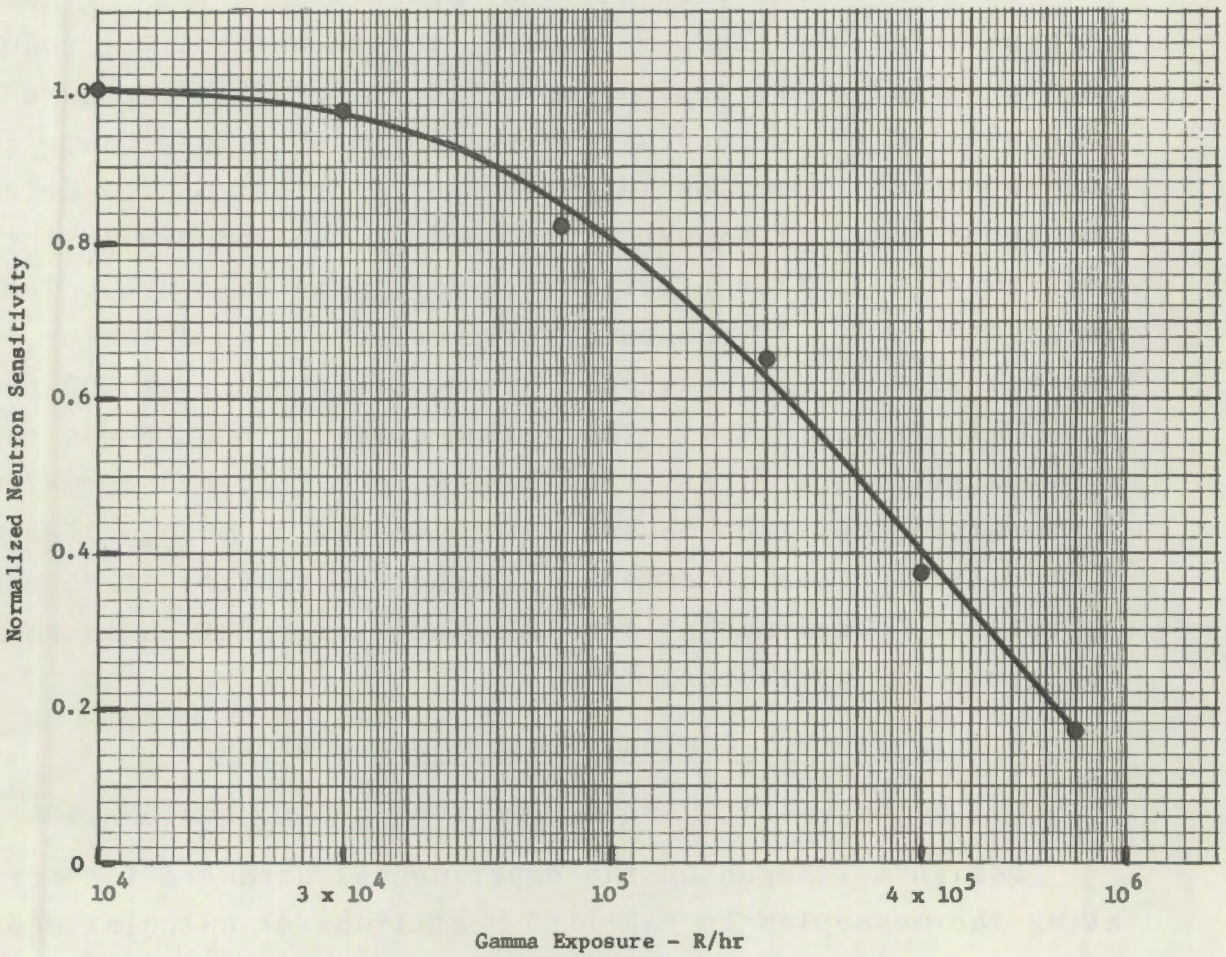
M. R. Wood

a. Low Level Neutron Flux Instrumentation

C. N. Jackson, N. C. Hoitink, and D. C. Thompson

The objective of this task is to test and evaluate nuclear components of a counting system in an adverse gamma/neutron ratio flux environment for the detection of delayed neutrons for failed fuel monitoring and for the reactor startup instrumentation channels.

A detector filled with a P-10 (10% methane and 90% argon) collection gas failed to achieve improved gamma pulse pileup performance as compared with an identical argon-nitrogen filled detector. Figure 4.B.3 depicts gamma effects on the neutron sensitivity of the fission chamber over a range of relatively high gamma exposures. Compared to the argon-nitrogen filled detector, the P-10 filled detector yielded only about 80% as much neutron sensitivity when exposed to intense gamma background in the 10^5-10^6 R/hr range. Apparently, the higher gas pressure (2.25 atmospheres) of the P-10 detector more than



A ratio of high exposure output to low exposure count rate output yields the normalized neutron sensitivity. The fission counter, incorporating fast P-10 collection gas, was a model WL-23604, Westinghouse Electric Corporation. Instrumentation includes a charge sensitive preamplifier HP-5554A, and 50 nsec pulse shaping; neutron count corresponded to 1 CPS background on integral bias curve.

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FIGURE 4.B.3. Gamma Tolerance of Fast Collection Time Fission Counter

offsets gamma pulse pileup reduction attributable to the faster signal collection characteristics. Measurements reported last month indicated a signal collection time of about 125 nsec for the P-10 chamber, as compared to 200 nsec for the usual design.

As the investigation progressed, other deficiencies in the P-10 chamber became evident. An exposure of 2×10^8 R gamma, attained over a period of 22 days, caused an apparent reduction in the signal pulse collection efficiency. This exposure necessitated a 22% increase in system gain to compensate for the corresponding reduction in output pulse height of the detector. The upper curve of Figure 4.B.4 depicts the comparable integral bias characteristics after the 22-day gamma exposure. Temperatures beyond 300 °F caused drastic reduction in detector signal characteristics, as indicated by the performance at 500 and 600 °F in Figure 4.B.4. The second curve at 600 °F was run ten minutes after the first. Apparently, the combined gamma and temperature condition caused a leak in the chamber, allowing the collection gas to escape. Subsequent testing at room temperature confirmed a large permanent degradation in the integral bias characteristics.

b. <u>In-Reactor Coolant Temperature Sensors</u>

N. C. Hoitink, J. L. Jackson, and D. C. Thompson

Design has begun on the experimental hardware for irradiating thermocouples in EBR-II. Heat transfer calculations to optimize internal pin hardware have been completed for Row 4 and are continuing for Row 7. Preliminary design sketches are being prepared.

Two complete B-7 subassemblies will be needed for the test, one in Row 4 and one in Row 7. This will require the fabrication and construction of 14 pins containing 198 thermocouples. Present design calls for an irradiation temperature of 1200 °F

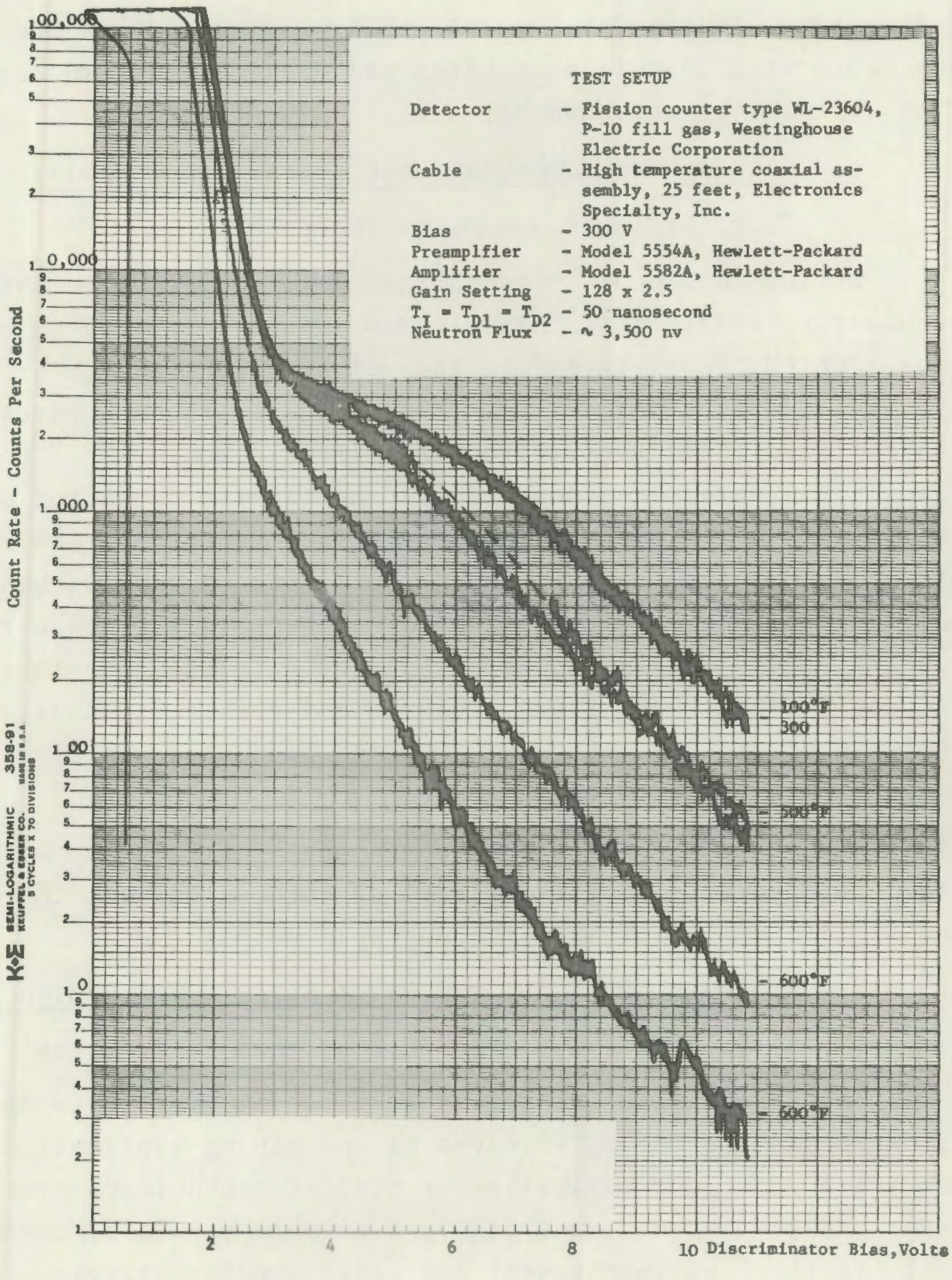

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FIGURE 4.B.4. *Combined 10^5 R/Hr and High Temperature Characteristics of Fast Rise Time Fission Counter (Preliminary Data)*

and a total fluence of 5×10^{22} in Row 4 and 10^{22} in Row 7. Each pin will contain melt wires and flux monitors to determine the irradiation environment.

c. Reactor Environmental Effects on Signal Cables

J. L. Stringer and D. C. Thompson

During this month irradiations were performed on the two-conductor, sheathed (twinaxial) MgO insulated cables in both the ^{60}Co PNL facility and in the pulsed TRIGA reactor at Washington State University. The cables, with both parallel and twisted conductors, were first irradiated in the ^{60}Co facility. Data were recorded for the twisted conductor cables which will allow calculation of the induced current and insulator conductivity. It was found that a much better mechanical design would be required for holding the parallel conductor cables before meaningful data could be obtained, due to the induced current variations with field-to-cable orientations. Because of this problem, only twisted conductor cables were used in the TRIGA test.

The maximum induced current in the ^{60}Co facility was less than 10^{-9} amperes. At this level of induced current there is no significant error in a thermocouple output.

d. Pressure, DP Flow, and Level Sensors for Sodium Service

R. D. Crosier, V. F. Sedlecak, and K. O. Creek

The objective of this task is to develop pressure and level instrumentation for FFTF in-vessel and piping applications. Activities include evaluation of present technology, procurement and testing of commercially available products, component modification and improvement and final proof testing.

(1) Pressure Transducer Testing. Having completed the 575 °F testing runs on the Consolidated Electrodynamics electromechanical unit model 4-361-0001 rated at 600 °F and 250 psi, tests will be run at 600 °F.

Arrangements have been completed with the vendor to perform a post-failure analysis of the Consolidated Controls model 412M44-200.

Kaman-Nuclear models K-1709 and K-1730 units were returned to the vendor for post-failure analysis. These units were rated to 900 °F.

One Barton-Statham PA 822 unit has indicated a negative zero shift. Thermocouples have been placed on the NaK capillary and transducer body to examine the cause of this shift. The other unit was returned to Barton for post-failure examination. Since the NaK system was found to be sound, the Statham unit will be sent to Statham for examination.

e. Driver and Test Position Connector Concepts

R. D. Crosier, S. K. Meyers, and K. O. Creek

The object of this task is to provide the basis for the development of an instrumentation connector capable of surviving in the FFTF environment.

Drawing SK-3-15073, Seal Leak Test Facility - Task IT-6 and IC-17 Interconnection Diagram, was completed to provide definition of the equipment and instrumentation required for the seal and connector test stand. Procurement of equipment has been initiated.

Details of the measurement and control circuits shown on Drawing SK-3-15056 are being prepared. These details will provide the wiring diagrams and specifications for hookup of the data acquisition and test mode control.

Drawings of the closed loop connector for the vertical core reactor concept have been completed. Fabrication of machined parts will start on March 17. The prototype will be completed by April 18, 1969. Testing with the Digital Data System is scheduled to begin September 1, 1969, with testing performed manually in the interim.

C. METALS AND MATERIALS DEVELOPMENT

J. C. Tobin

1. Materials Properties

J. E. Irvin

The primary purpose of this program is to obtain materials properties data for use in the selection, design, fabrication, and assessment of performance of FFTF components. Effects of irradiation, temperature, environment, and other major parameters will be determined as required.

a. Materials Design Data

G. A. Dunagan

A literature search has been completed on the effects of neutron irradiation on austenitic welds. Very little information exists on the irradiation effects of welds per se, and most of the irradiations were performed in thermal reactors, at temperatures less than 100 °C (212 °F), and in water. A continuing review is being performed to obtain specific information on welds of either Type 304 or 316 SS as they relate to potential applications in the FTR.

b. Weldment Studies

L. D. Blackburn

Characterization of the tensile properties of two as-received weldments is in progress and test results to date are summarized in Table 4.C.I. Weld specimens (100% weld metal) and fusion line specimens (weld metal, fusion line, and heat affected zone all contained in the gage length) were tested for both the submerged arc and the stick electrode processes. A major feature of the results is the low ductility of the welds. Although base metal data is not yet available, total elongation is expected to be about 50%. In comparing the weld and fusion line specimens for the submerged arc process, the similar ductility values suggest that failure is occurring in the weld metal portion of the fusion line specimen. However, the yield strength of the fusion line specimen is higher than that of the weld metal specimen. Since the base metal yield strength would be expected to be about 25,000 to 30,000 psi, the high strength of the fusion line specimens requires clarification. For the stick electrode process, both strength and ductility values suggest that the fusion line specimen properties may be controlled by the weld metal. Final verification of the flow and fracture behavior of the fusion line specimens will require testing of base metal and selected metallographic observations.

TABLE 4.C.I. Tensile Properties of As-Received Weldments

Specimen	Test Temp, °F	0.2% Yield Strength, psi	Ultimate Tensile Strength, psi	Uniform Elongation, %	Total Elongation, %	Reduction of Area, %[a]
Submerged Arc Process (308 Filler)						
Weld	700	35,300	66,900	22	30	~45
	900	34,800	63,300	23	25	~40
Fusion Line	700	45,100	63,500	21	25	~70
	800	44,300	65,100	21	25	~60
	900	52,300	63,300	13[b]	15[b]	~35[b]
Stick Electrode Process (16-8-2 Filler)						
Weld	700	49,500	67,200	16	18	~50
	900	48.600	67,000	16	17	~40
Fusion Line	700	48,000	66,100	17	20	~70
	900	43,800	59,200	15	19	~55

a. *Reduction of area values in Table 4.C.I are only approximations because of irregular circumferences of fracture.*
b. *Shear-type fracture*

c. Chemical Compatibility of Shield and Control Rod Materials Compatibility of Boron Carbide with Sodium and Stainless Steel

L. R. Bunnell

Hot-pressed boron carbide pellets with nominal densities of 75, 85, and 95% TD have been subjected to sodium (oxygen content 3 ppm) in 100-hr static tests at 300 and 500 °C. Post-mortem examination of the pellets revealed that very little damage occurred as a result of the exposure.

An interesting feature of the results is the good performance of the low density pellets. If the general trends for ceramics (mostly oxides) in contact with liquid metals held true, material with a density in the range 60 to 90% TD would be expected to perform badly due to penetration of the liquid sodium into open porosity. Metallographic examination confirmed the apparent absence of attack, and, except for some slight pitting in the surface of the 75% TD pellet, the 75% and 85% TD pellets were free from noticeable effects. The 95% TD pellets were cracked near the outer surfaces, which had also occurred in previous tests on high-density pellets. Thermal cycling may increase cracking and lead to other problems in the less-dense pellets. Other pellets are currently being thermally cycled in sodium.

The Type 316 SS used in the capsules showed no metallographically observable carbon pickup. Results from spark-source spectroscopy runs are not available, but will be included in the next monthly report.

2. Fabrication Development

R. N. Johnson

a. Automatic Duct Welding Development

W. F. Brown

A series of circumferential butt joints are currently being made in circular Type 304 SS tubes, 5-1/8 in. OD × 0.150 in. wall thickness. These joints are rotated under an automatic welding head. The welding parameters developed will be applicable to hexagonal sections and will provide parameter data for the hexagonal tube welding machine design.

A purchase requisition for automatic duct welding equipment has been submitted for AEC approval. This equipment, as purchased, will be capable of automatically welding a range of round pipe sizes. As soon as this equipment is received, design efforts can proceed to adapt the systems for hexagonal configuration welding.

b. Fuel Duct Wear Pads

G. S. Allison

Visits were made to two companies specializing in metalizing by flame and plasma spraying. The goal of the discussions was to determine the practicality of applying the fuel duct wear pads by these techniques. (Candidate wear pad materials are Inconel 718 and Stellite 6B).

(1) Metalizing Company of Los Angeles. This company operates a job shop specializing in the flame spraying of metals and ceramics onto a variety of substrates. Their opinion of efforts to apply wear pads by flame spraying is not very encouraging. They visualize the major problems as:

- The tube will undoubtedly warp from the heat of application.
- The 0.050 and 0.090 in. thicknesses required are considerably above the normal application and will cause cracking of the coating.

They recommended the following process as the best for this application, but could not guarantee complete success without extensive testing and development.

(a) Prepare surface by blasting with No. 25 steel grit.
(b) Flame spray coat with metal powder within 4 hr.
(c) Flame fuse the coating.
(d) Slow cool in asbestos powder bed.
(e) Machine.

Tight fitting mandrels and/or rugged external fixturing would be necessary to prevent warping. Also, the heating of the duct followed by the slow cool in the asbestos powder bed would probably sensitize the Type 304 SS, leaving the duct in an unacceptable condition.

(2) Plasmadyne Division, Giannini Corporation. This firm manufactures and markets a line of plasma and arc metal spraying equipment. They also have a laboratory for developing applications of their product for customers. They were less encouraging than Metla about the fuel duct wear pads. The pad thickness is too great to be practicable and the tube will probably warp from heat of application. Their coating is not fused, has approximately 90% density, and approximates 10,000 psi bond strength. This process has been eliminated from further consideration at this time.

Two further processes for attaching wear pads to the fuel ducts are being investigated. One process is the weld overlay method, which has the disadvantage of being the process most likely to cause an unacceptable distortion in the duct.

The other process is the welding of preformed wear pads onto the duct. This process is the one being most actively pursued since it offers the following advantages:

(a) With proper welding technique, the process should cause the lowest duct distortion of any of the processes.

(b) Inconel 718 wear pads can be heat treated for optimum properties before attaching them to the duct. (The precipitation hardening heat treatment for Inconel 718 would cause sensitization of the duct if heat treatment was required after attachment.)

(c) The thickness of the wear pad causes no problem for this process.

c. <u>Closed Loops - Decontamination Recommendation</u>

S. M. Gill

Stainless steels in sodium reactor service may become surface-contaminated to fractional mil depth from deposition and diffusion of radioactive metal isotopes. Currently, the Coolant Chemistry organization is studying methods of decontamination for this situation. It was recommended that in addition to the methods under study, they also study the molten sodium hydroxide-sodium hydride scale removal process. This idea presumes that the stainless surface can be oxidized to a predetermined depth with hot air. Subsequent treatment with the molten caustic-hydride would remove the oxide without attacking the base metal. Intergranular penetration has not been observed with this process. Traces of the molten base left in the decontaminated unit would presumably be removed in the system cold traps, so no extrusive cleanup after decontamination would be required.

3. Component Surveillance

J. W. Helm

The objective of this effort is to establish a program for surveillance and inservice inspection. Objectives include early detection of abnormal or unexpected behavior and verification of expected behavior of critical components in the primary coolant boundary and closed loop systems which have a direct influence on the reliable and safe operation of the FFTF.

(a) Signature Analysis

J. W. Helm

A meeting was attended at USAEC Headquarters on February 19 in which General Electric Company representatives discussed the techniques and application of Mechanical Signature Analysis (MSA). MSA makes use of interpretation and analysis of external measurements of sound or vibration signals to diagnose internal conditions or malfunctions and to detect incipient failure. MSA was recommended as being pertinent to the LMFBR and FFTF programs in the areas of in-core sodium boiling, water-sodium leak detection, system inspection of critical components, control rod drive mechanisms, component development, and neutronic noise cross correlation.

FFTF will not initiate action on the recommendation pending establishment of inspection requirements upon completion of the first phase of the effort by Southwest Research Institute.[1] A trip report summarizing the meeting was issued.[2]

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1. *Contract BDR-570 with Southwest Research Institute, Conceptual Design of An Inservice Inspection System.*

2. *Trip Report, J. W. Helm, Meeting on Mechanical Signature Analysis at AEC Headquarters, February 19, 1969, BNW/FFTF.*

4. Design Codes for FTR Vessel Piping and Components

M. T. Jakub

A meeting was held at RDT on February 28, 1969 to determine the interim design code for the vessel. The conclusions are listed below.

(a) Agreement reached on allowable values for primary membrane and primary plus secondary stress from 100 to 1200 °F.

(b) Secondary stress can be based on elastic analysis without the necessity for considering creep if vessel is cooled to 1000 °F or less.

(c) The Manson method for combining cumulative creep and fatigue damage will be used, with modifications.

(d) It was affirmed that austenitic stainless steel may be used in accordance with present code rules if the irradiation damage to the vessel material does not reduce the predicted end-of-life ductility below that represented by a total tensile elongation of 10%.

V. FFTF FUELS DEPARTMENT

E. A. Evans

A. FUEL ELEMENT DEVELOPMENT

C. A. Burgess

1. Process Demonstration and Development Section

R. E. Bardsley

a. Processing and Equipment

H. T. Blair and J. E. Sammis

Production of 4,000 mixed oxide pellets for the Analytical Standards Program has progressed to the sintering stage. All major steps in the production of mixed oxide pellets with the exceptions of PuO_2 calcining and binder removal were accomplished within the new Fuel Fabrication Demonstration Facility. Test pellets having the required density of 90 to 92% of theoretical with an oxygen-to-metal ratio of 1.94 were produced to demonstrate sintering capability prior to sintering the entire batch of pellets for the Analytical Standards Program.

Several sintering tests with mixed oxide pellets indicate that density and oxygen-to-metal ratio do not vary with radial position within the sintering furnace.

The 12,500 UO_2 pellets for irradiation test fuel pins for EBR-III Test Assemblies PNL-6, 7, and 8 were reduced to the desired diameter by centerless grinding.

Sampling for acceptance verification of a 35 kg PuO_2 shipment from ARHCO was completed. Net weights were checked, and the samples were submitted to Analytical Chemistry.

The first programmed sintering run on No. 2 sintering furnace was completed March 3. The furnace and all associated systems performed well.

b. Process Development - UO_2, PuO_2 and Mixed Oxide Powder Characterization

M. J. Barr

(1) Process Variables Study - Fractional Factorial Experiment. Evaluation of the fuel pellet process variables test using a fractional factorial design for investigating major fuel pellet pressing and sintering variables was continued. All pellets for the 82 cells of the fractional factorial test were pressed, weighed, measured, and made ready to be sintered. Data for the as-produced green pellets were submitted for computer analysis.

(2) Detailed Variables Study - Block Experiments. The test to determine the effect of wet milling mixed oxide powder (using tungsten carbide balls as the grinding media) on homogeneity, density, microstructure, and impurity level was partially completed. However, the data have not been evaluated for statistical significance.

2. Subassembly Development Section

J. W. Thornton

a. FFTF Fuel Fabrication Demonstration Facility

M. D. Jackson

The design phase of Project BCP-034 (Fuel Subassembly Demonstration Facility) is progressing well. The layout drawing of Room 138 was completed and approved. The cleaning and rinsing tanks and the pin dryer were designed and are in the approval stage.

3. Special Products Fabrication Section

E. T. Weber

a. Irradiation Test Assembly Fabrication - Task FP-8

R. M. Crawford

Thirty-seven pins for EBR-II Subassembly PNL-4 were shipped to the reactor.

b. Fabrication of Irradiation Test Pins

R. M. Crawford

Two attempts to fabricate four small xenon test capsules for Task FP-3 were insufficiently successful. It was impractical to control diffusion of the xenon gas from the pin to the welding chamber because of the small gas plenum in these pins (less than 1 cm^3). The xenon test capsules are currently being made using a small chamber where it is possible to fill the entire chamber volume with the desired xenon-helium ratio. This is not expected to be a problem with the EBR-II irradiation test pin with larger gas plenums.

A device for sampling gas from dummy fuel pins attached directly to the gas mass spectrometer was developed for providing analyses of xenon-helium ratio. This system will be used in evaluating and testing the present xenon tagging method and apparatus. Testing will be completed prior to final tagging and welding operation on pins for EBR-II irradiation.

c. Fuel Fabrication for Irradiation Test Pins

W. E. Warden

Fuel pellet fabrication for EBR-II Subassembly PNL-8 is in process. Six kilograms of pellets were pressed and four kilograms sintered. This fuel consists of 93% enriched ^{235}U in the standard EBR-II test composition 75% UO_2—25% PuO_2. Standard process procedures applied in fabricating this fuel yield densities on the low side of the acceptable range (93 ± 2% TD). Analysis of the microstructure and a review of process conditions is underway to define process alternatives or controls which will increase yield in sintering enriched fuel.

4. Fabrication of Irradiation Test Pins

E. M. Crawford

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A device for sampling gas from dummy fuel pins attached directly to the gas mass spectrometer was developed for providing analyses of xenon leakage rates. This system will be used in evaluating and testing the present xenon tagging method and apparatus. Testing will be completed prior to final tagging and welding operation on pins for EBR-II irradiation.

5. Fuel Fabrication for Irradiation Test Pins

[illegible]

Fuel pellet fabrication for EBR-II Subassembly PNL-8 is in process. Six kilograms of pellets were pressed and four kilograms sintered. This fuel consists of 93% enriched UO_2 in the standard EBR-II test composition 75% UO_2-25% PuO_2. Standard process parameters applied to fabrication will yield densities on the low side of the acceptable range (90 ± 2% TD). Analysis of the microstructure and a review of process conditions is underway to define process alternatives or controls which will increase yield in sintering enriched fuel.

B. QUALITY ASSURANCE - FUELS

H. G. Powers

1. Development and Control of Inspection and Testing

a. Eddy Current Tester

J. Ryden, Jr., and A. C. Callen

Evaluation of the eddy current technique that uses two encircling coils as drivers, and two or more search coils as pickups, was continued. The results were encouraging.

b. End Closure Test

J. L. Thompson, J. Ryden, Jr., and A. C. Callen

Evaluation of the ultrasonic end weld test was continued. The high incident angle, pulse-echo ultrasonic test has demonstrated exceedingly good sensitivity to a small region of the weld which includes the original tubing ID-to-end cap interface. It has great sensitivity to defects in this region, principally lack of penetration, while rejecting to a large degree surface signals resulting from such items as weld puddle, ripple, or shoulder. The surface conditions mentioned here rendered all more conventional test approaches invalid because of confusing echos or poor transmission of ultrasound.

A preliminary experiment was performed to determine the feasibility of employing acoustic emission as another nondestructive test technique for the end closure weld. The defects formed were of gross nature, but significant acoustic emission was detected.

c. Nitrogen

W. L. Delvin

A nitrogen method based upon the conversion of nitrides in a sample to ammonia by a caustic fusion was set up in a glove box. The ammonia is collected and measured spectrophotometrically. The equipment is being used to measure nitride nitrogen in mixed oxides. However, no standards exist and thus determining the presence or absence of a bias such as low nitrogen recoveries is difficult. Work is in progress to determine if a bias exists.

d. High Temperature Water Analyzer

W. L. Delvin

Presently, the water analyzer in use is capable of heating a sample only to 200 °C to evolve water. Evidence exists that water may be retained in FFTF fuel pellets at that temperature. All of the water may not be evolved until 800 °C is reached. Another water analyzer was returned to the manufacturer for modification to give a heating capability of up to 1000 °C.

2. Special Studies

a. Quality Assurance Comparison of Grid and Wire Wrap Fuel Spacer Systems

J. D. Schaffer

Comparative analyses of fuel spacer systems and associated Quality Assurance considerations were continued because of a change in the grid spacer design by Westinghouse.

Inspection costs incurred in support of grid fabrication and assembly, even with efficient mechanization, are estimated at 45% of direct labor. This figure indicates that quality controls will be a significant contributor to the overall cost of a grid spacer system.

C. CLADDING DEVELOPMENT

T. T. Claudson

1. Cladding Development and Processing

J. C. Tverberg

a. Procurement Activities

R. J. Lobsinger

A quantity approximating 260 pounds of 0.250-in. diameter rod was obtained from Crucible Steel as part of the conversion of 4 1/2 in. diameter double vacuum melted Type 316 SS rounds. The physical certification received for this material, however, indicates the rod was cold-worked after the final annealing. Discussions are in progress with the vendor to resolve the problem which will require the furnished material to be re-annealed.

b. Vendor Development

J. C. Tverberg, R. J. Lobsinger, R. C. Aungst, and R. R. Studer

Proposals were received from five of the firms to whom RFP packages were sent. None of the proposals could be considered completely responsive based upon the criteria established by the Cladding Evaluation Board. Therefore, a visit was made to each of the five plants (Carpenter Technology Corp., Universal Cyclops, Sandvik, Wolverine Tube, and Superior Tube Co.) by members of the Board and their proposals were discussed in detail. On the basis of this visit the Product Description and the Statements of Work were rewritten to reflect the changes in the proposals.

2. Cladding Evaluation

J. J. Holmes

a. Acceptance Testing and Cladding Characterization

M. M. Paxton

Lot-to-lot variations in selected mechanical properties of AISI 304 and 316 SS seamless tubing are being determined.

A check chemical analysis was completed on five tube lengths from "G" lot tubing. The analysis agrees reasonably well with the analyses supplied by the vendor except for phosphorus. The phosphorus values supplied by the vendor are lower by a factor of 2 than the values received on the check analysis.

Metallography of five "G" lot tube lengths was completed. Figures 5.C.1 and 5.C.2 are typical transverse and longitudinal photomicrographs from 100X to 1000X. The grain size as determined from five samples varied between ASTM grain size 7 to 8.

b. In-Reactor Measurements

N. E. Harding and E. R. Gilbert

Testing of the mechanical strain measuring system for EBR-II in-reactor cladding creep tests was terminated after 2000 hr. Examination of the components indicate that the dry film lubricant is satisfactory, but there is still a problem of relaxation of the spring used in the micropositioner.

c. Damage Analysis

J. L. Straalsund and H. R. Brager

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

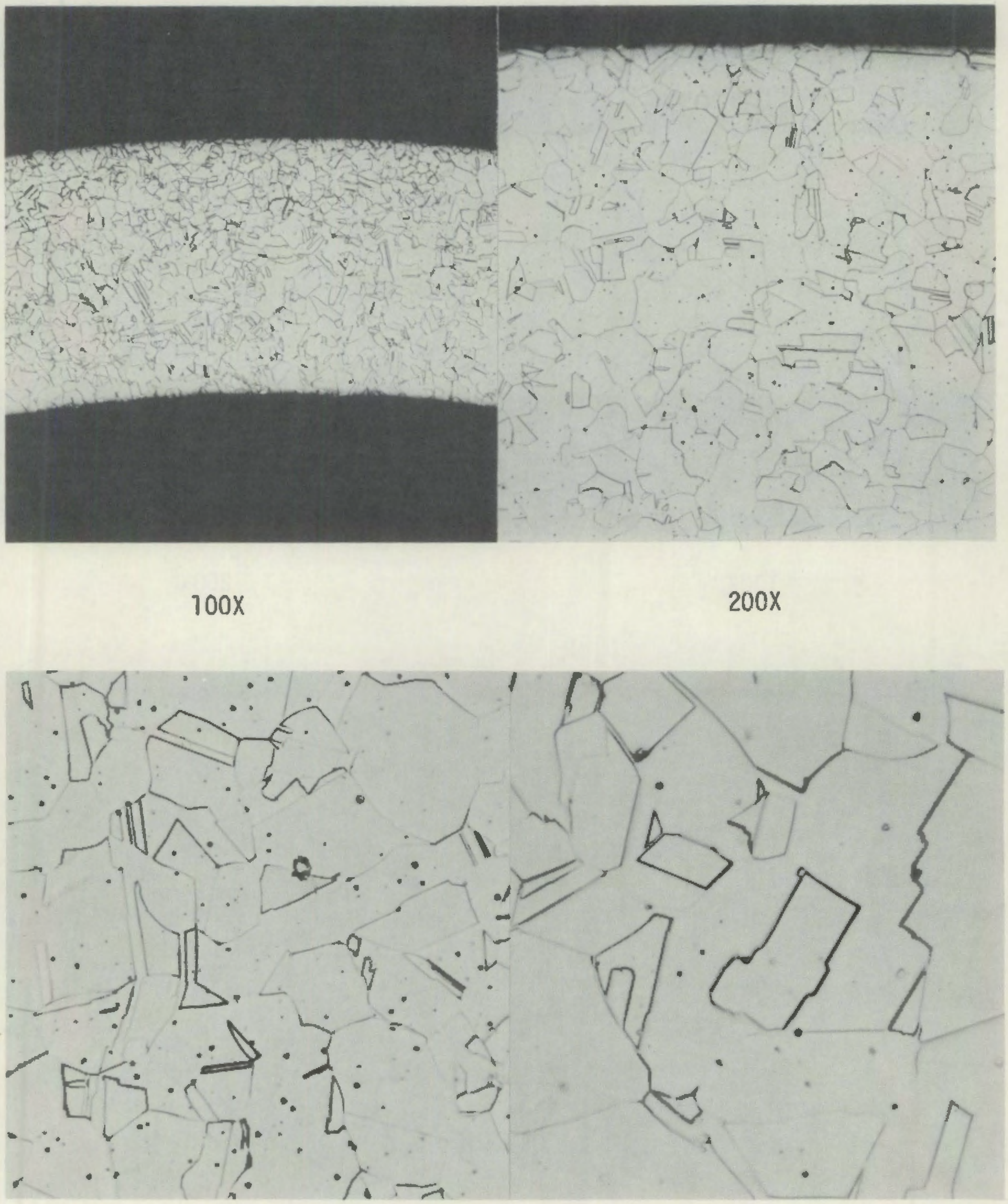

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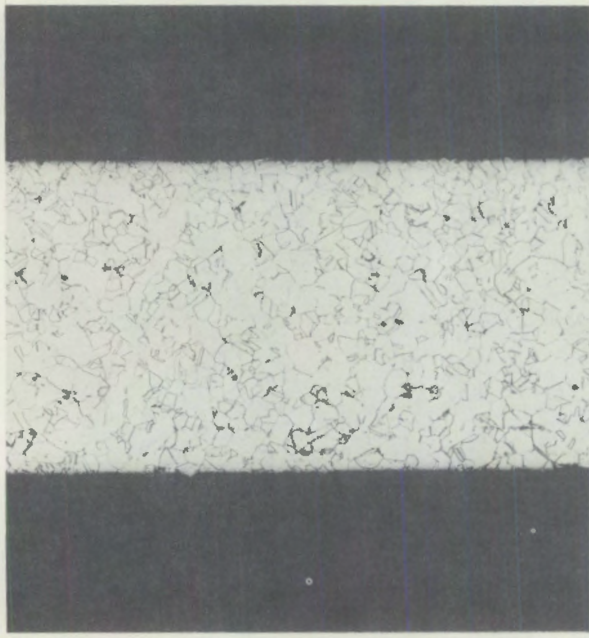
FIGURE 5.C.1. Typical Transverse Microstructure for Tubing Lot "G", 0.250 OD × 0.016 ID, Mill Annealed Type 316 SS



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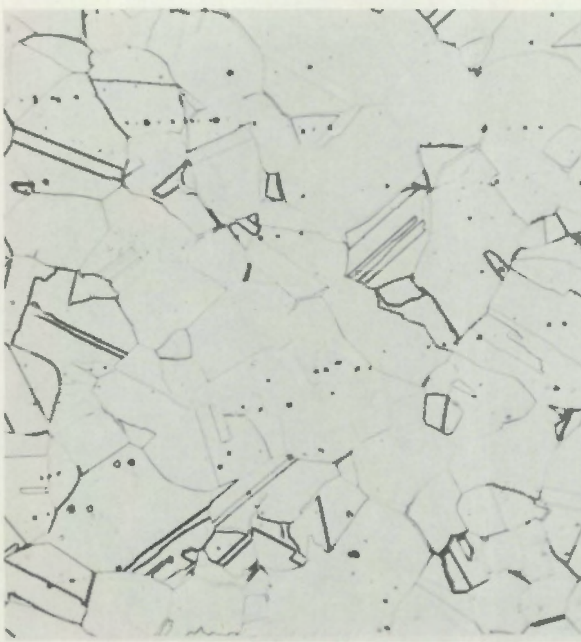

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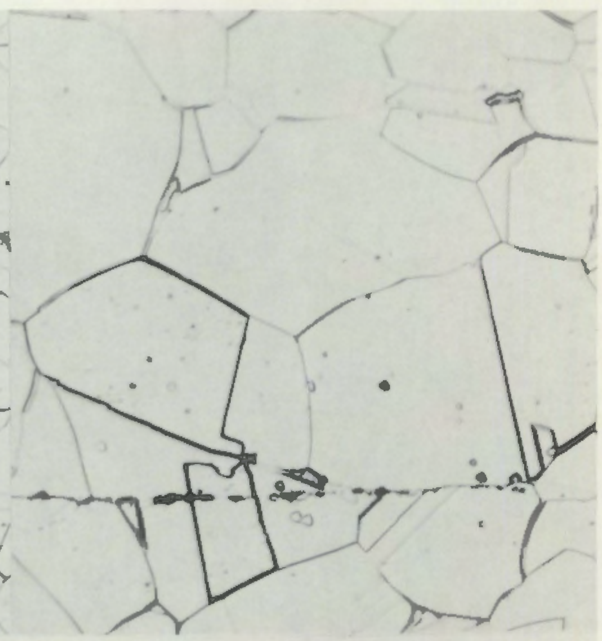


1000X

FIGURE 5.C.2. *Typical Longitudinal Microstructure for Tubing Lot "G", 0.250 OD × 0.016 ID, Mill Annealed Type 316 SS*

Density measurements were completed on the EBR-II safety rod thimble. This thimble had accumulated approximately 9.6×10^{22} n/cm^2 total fluence.

Linear regression analysis was used to fit an equation of the form

$$\frac{\Delta V}{V} = A(\phi t)^m e^{-Q/RT} \tag{1}$$

to the available data. Because of the reduced accuracy in flux and temperature calculations for regions outside of the core, only the data taken from points 10 in. or less from the midplane were used in the analysis. Also, since it had been shown earlier[1] that there appears to be a fluence-independent component of measured swelling values of about 0.05%, this amount was subtracted from each of the swelling values, before making the analysis. The resulting swelling equation is:

$$\left(\frac{\Delta V}{V} - 0.05\%\right) = 2.82 \times 10^{-40} \; (\phi t)^{1.86} e^{-\frac{7060}{RT}} \tag{2}$$

where

$\Delta V/V$ = swelling in %

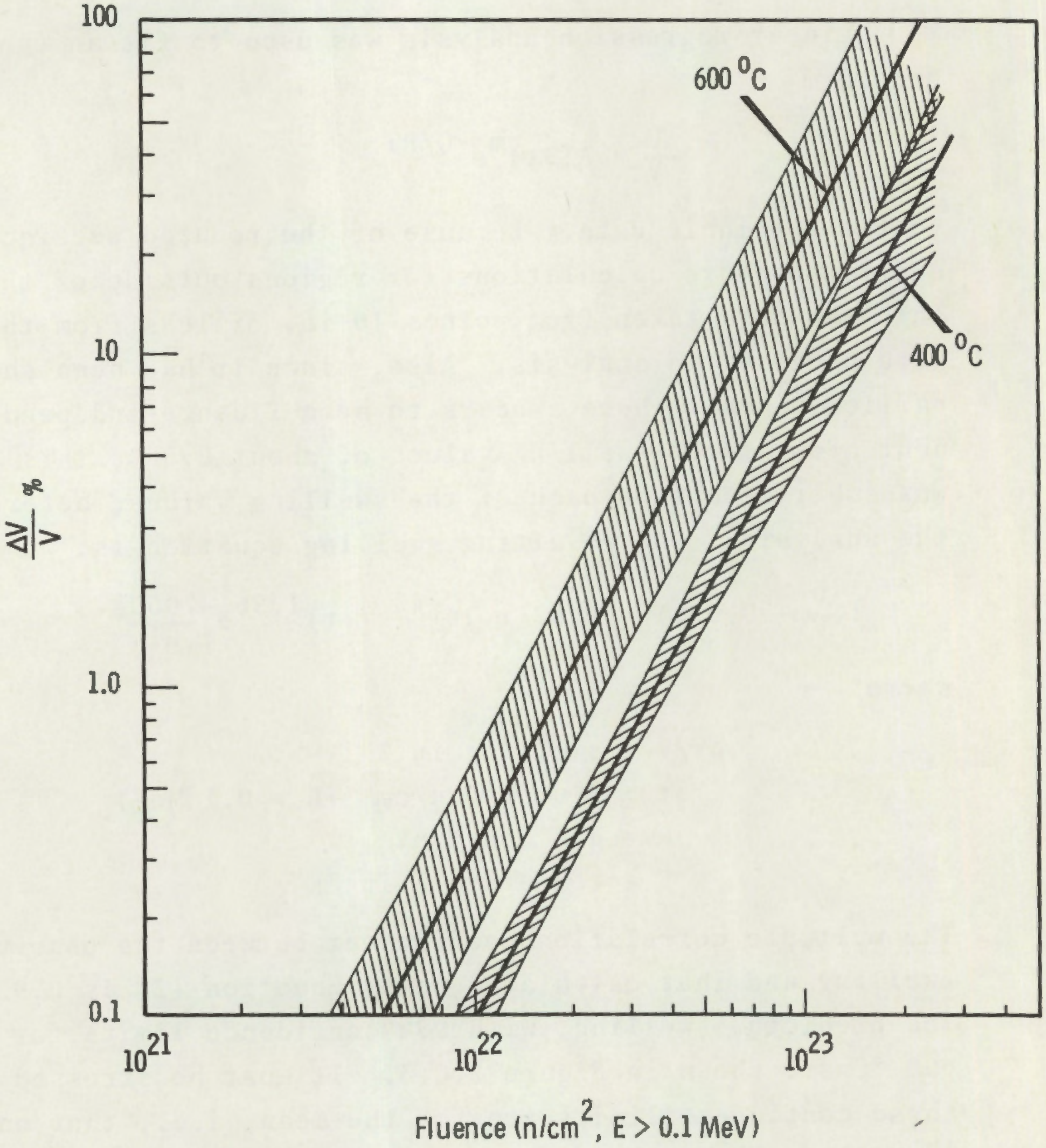
ϕt = fluence, n/cm^2 ($E > 0.1$ MeV)

R = gas constant

T = temperature in °K

The multiple correlation coefficient between the observed swelling and that calculated using Equation (2) is 0.934. The predicted swelling, with 95% confidence limits for 400 and 600 °C are shown in Figure 5.C.3. It must be stressed that these confidence limits are for the mean, i.e., that one may be "95% Certain" that the mean of further swelling measurements

1. *J. J. Holmes. Fast Reactor Induced Swelling in Austenitic Stainless Steel, BNWL-SA-2126. Battelle-Northwest, Richland, Washington.*



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<u>*FIGURE 5.C.3.*</u> *Swelling in 304 Stainless Steel (Annealed) with 95% Confidence Limits for Mean*

will lie within the limits. The 95% prediction intervals for individual points are given in Figure 5.C.4.

A plot of ΔV/V versus distance from reactor midplane for each of the components studied to date is shown in Figure 5.C.5. As can be seen from the plot, the swelling for the safety rod thimble is symmetric about the midplane whereas the swelling for previously studied components is not.

Density measurements were completed on specimens of Nickel-270 (99.98% Ni), Nickel-200 (99.6% Ni) and Inconel 600 (73 Ni—17 Cr—8 Fe) in the annealed and 50% cold worked condition. These specimens were irradiated in Row 2 of the EBR-II to a total peak fluence of about 4.3×10^{22} n/cm^2 at 800 ± 50 °F. Various fluence levels were obtained by taking advantage of the axial flux gradient in the EBR-II. Specimens were spaced so as to accumulate from 100% to 2% of the peak fluence. The irradiation environment was static sodium.

Volume changes induced by the irradiation are given in Figure 5.C.6 as a function of neutron fluence. The effect of purity is clearly evident. The highest swelling rate among these alloys occurs in Nickel-270 with Nickel-200 being intermediate between the Nickel-270 and Inconel-600. The Inconel-600 shows no tendency for a fluence dependence of swelling over the range of fluences studied. There is, however, a small but consistent decrease in volume for the Inconel-600 at all fluences studied and in Nickel-200 at the lower fluences only.

It was previously reported that solution treated Type 316 SS which was irradiated at 900 °F to 0.8×10^{22} n/cm^2 ($E > 0.1$ MeV) contained small polyhedral and rod-shaped precipitates. In addition, an unirradiated sample of the same heat of solution treated steel had been annealed 3050 hr at 900 °F, the same time at temperature as the irradiated specimen. The unirradiated thermal control sample did not contain any

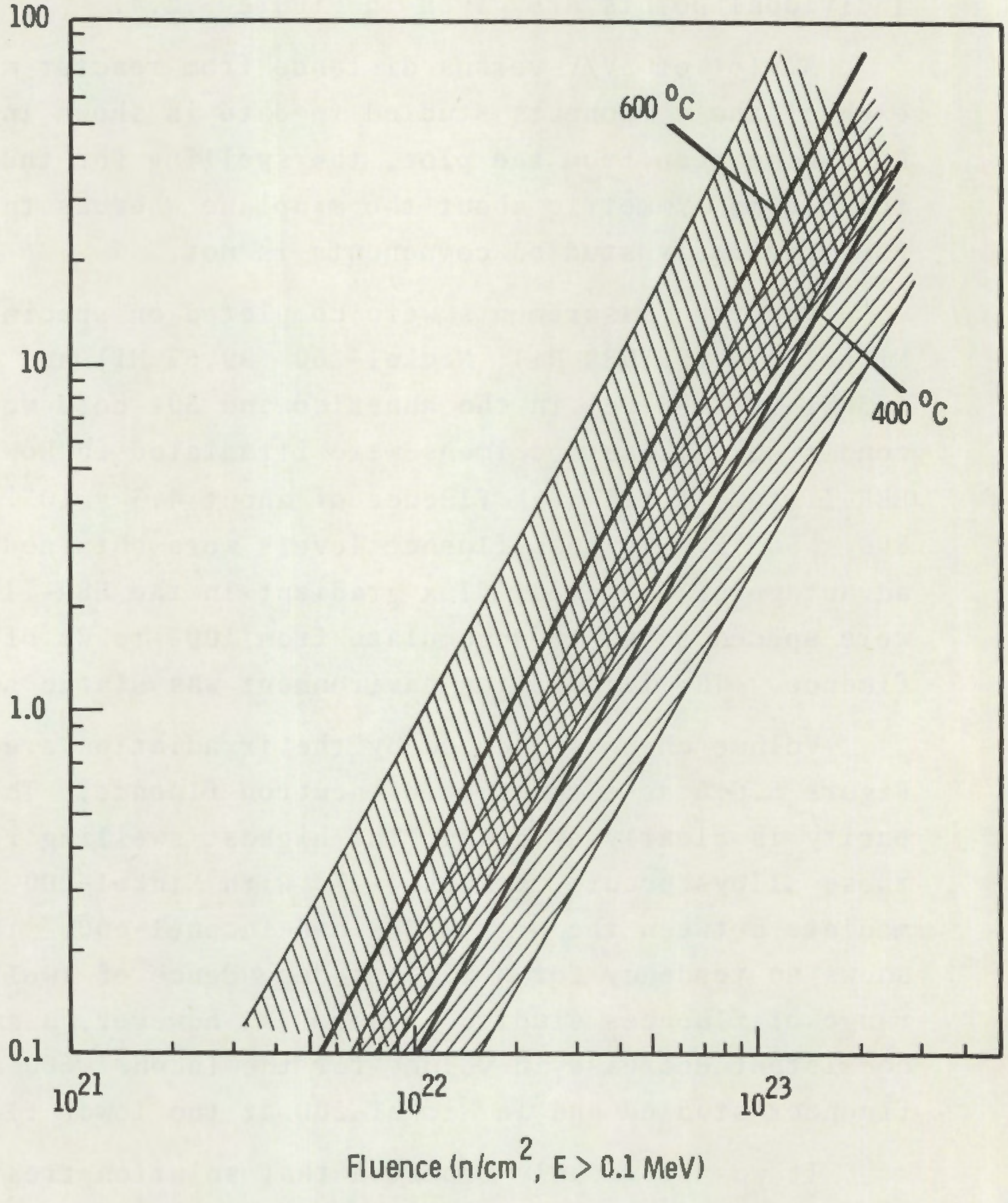

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FIGURE 5.C.4. Swelling in 304 Stainless Steel (Annealed) with 95% Prediction Interval for Individual Points

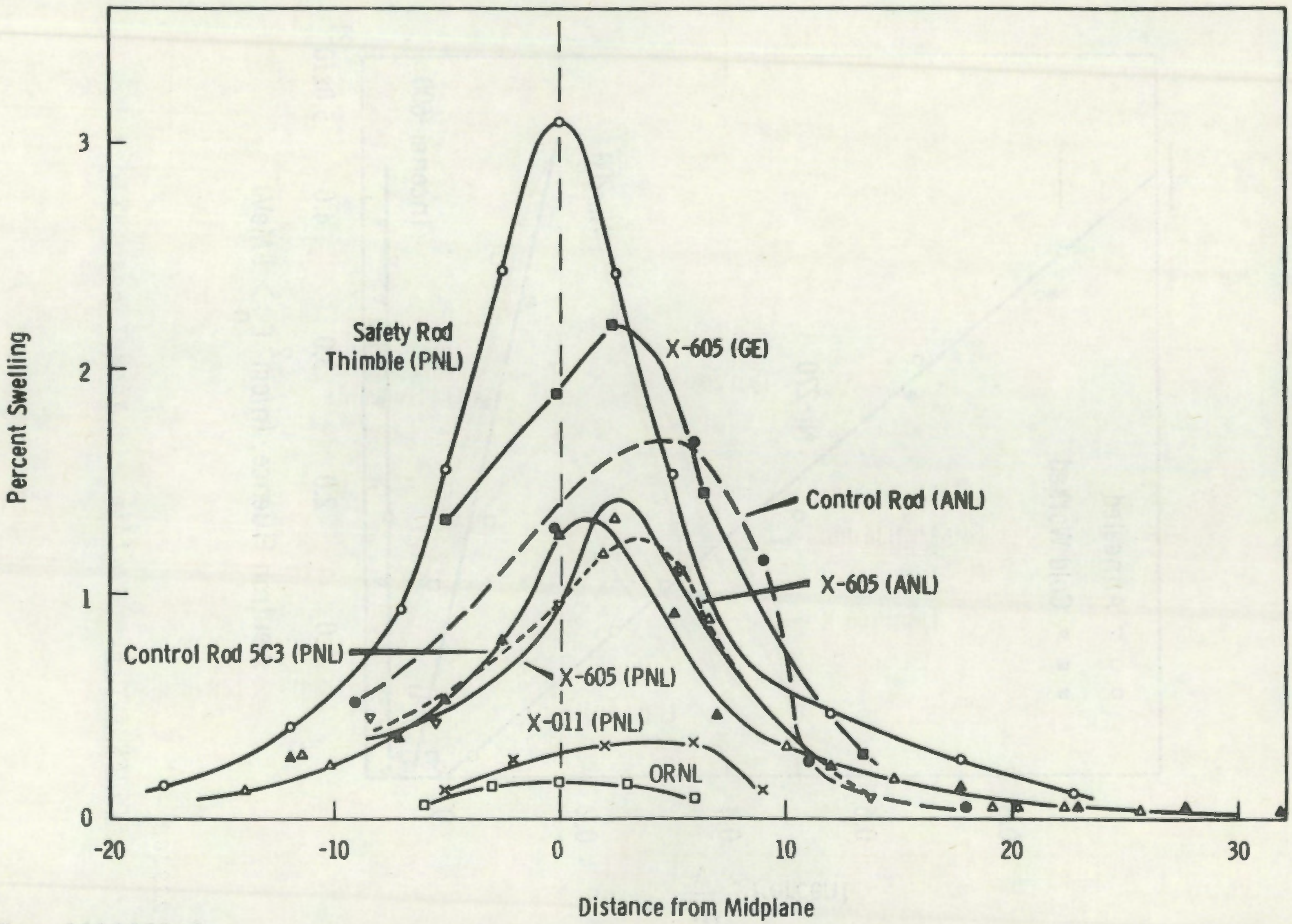

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FIGURE 5.C.5. *Swelling in Various EBR-II Components as a Function of Fluence*

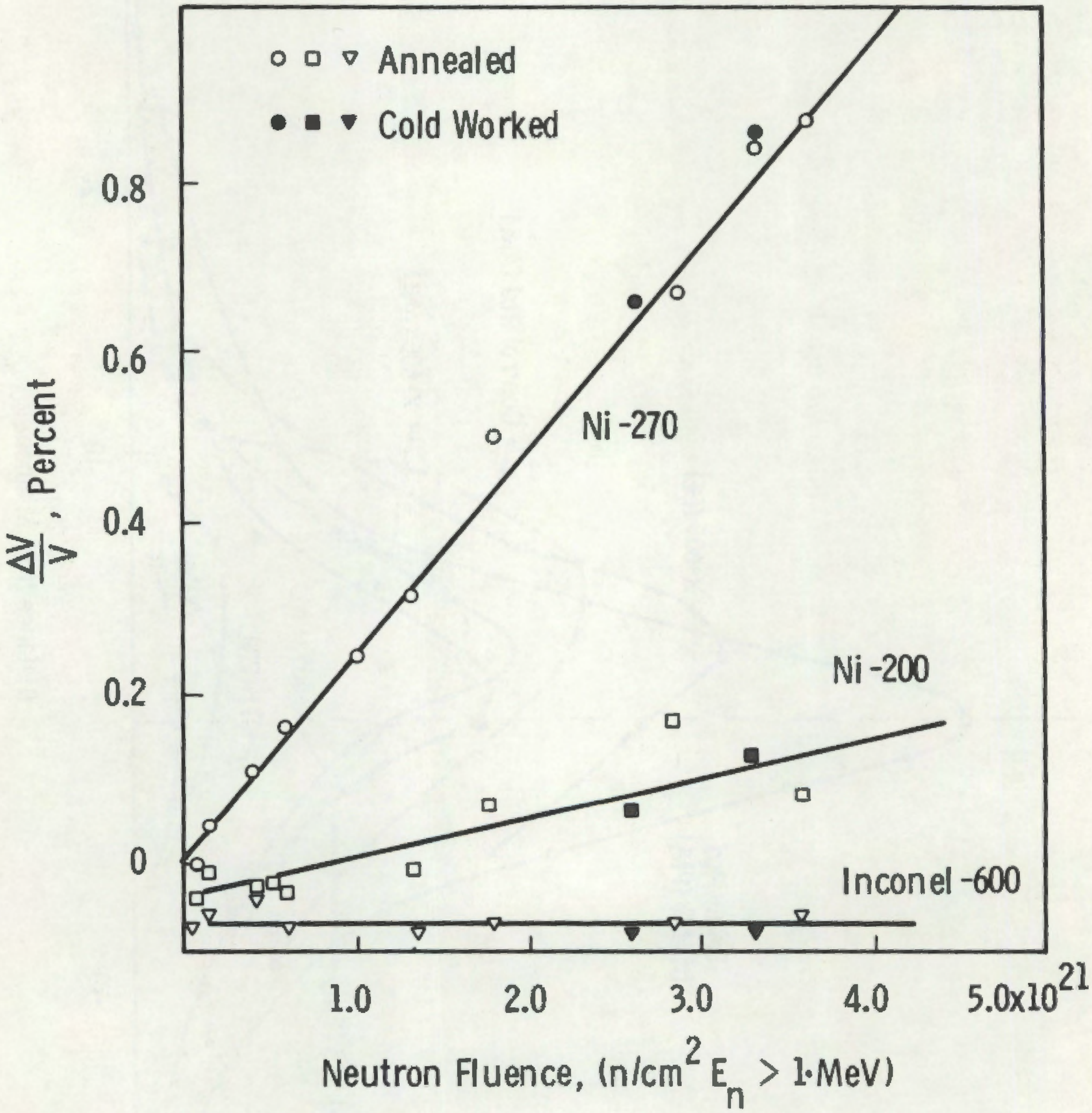

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FIGURE 5.C.6. Swelling in Nickel Base Alloys

visible precipitates, excluding a few large inclusions normally found in commercial steels. An extraction replica was made of the irradiated steel. The replica contained both individual cuboids and stringers of cuboids as well as precipitates (Figure 5.C.7). Electron diffraction identified only $M_{23}C_6$ carbide particles. X-ray analysis of the same precipitates identified only $M_{23}C_6$ carbides. Therefore, it appears that fast neutron irradiation at 900 °F induces $M_{23}C_6$ carbide precipitates having two morphologies in solution treated Type 316 SS.

The section of the safety rod thimble which had been irradiated in the EBR-II at 730 °F to the peak fluence of 8.1×10^{22} n/cm^2 (E > 0.1 MeV) was examined. Preliminary analysis indicates a void density of about 8×10^{15} /cm^3 and a lineal averaged void diameter of 170 Å. The volume occupied by the voids was calculated to be about 2.7%. This value is in good agreement with the 3.09% decrease in bulk density of the same sample.

3. LMFBR Fuel and Cladding Information Center

F. R. Shober

Data for 19 tensile tests were placed in the REM file, bringing the total tests in the system for austenitic stainless steel to 1971.

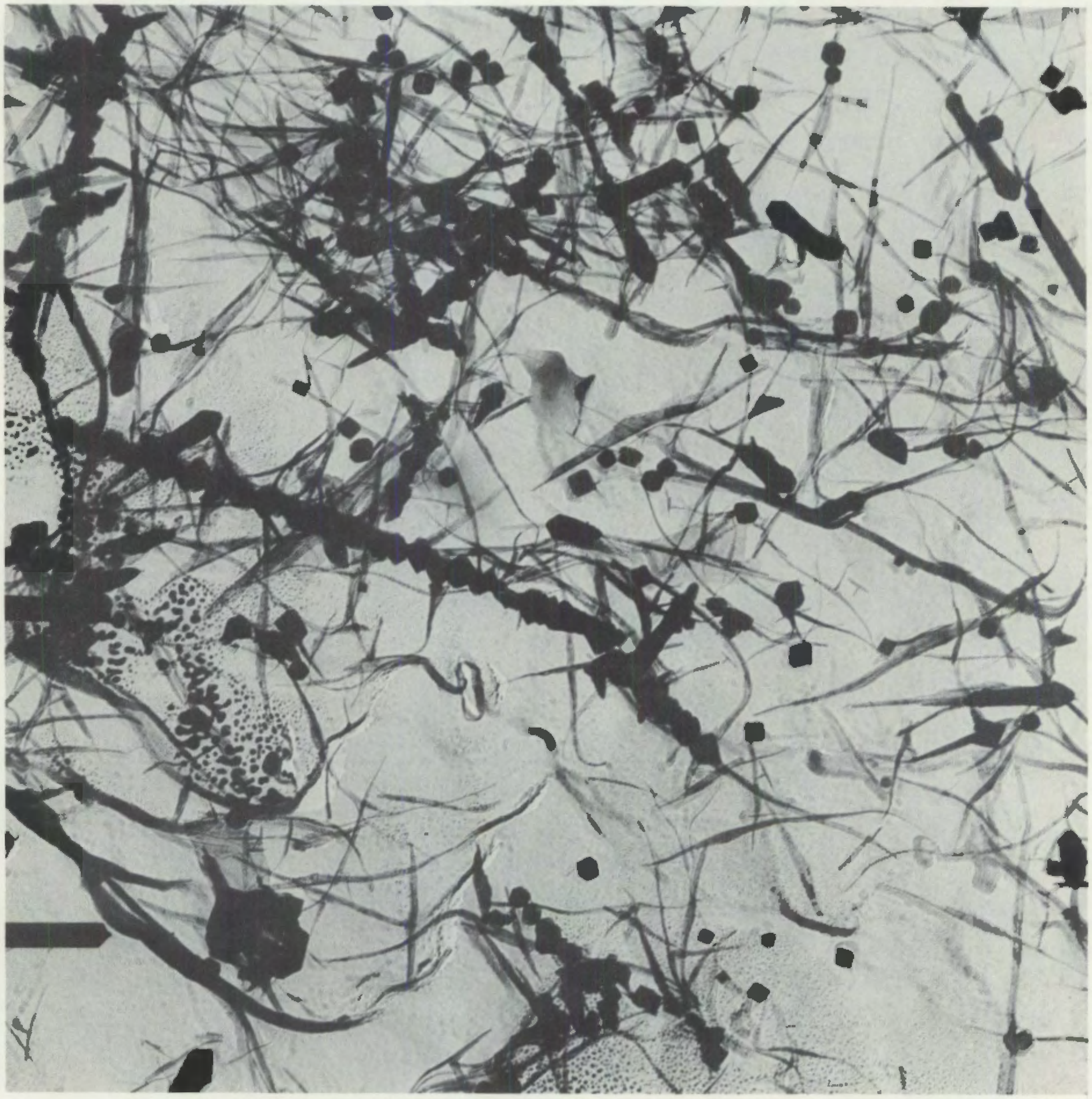

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FIGURE 5.C.7. Extraction Replica Showing Needle and Cuboid Morphology of $M_{23}C_6$ Precipitates from Type 316 SS Irradiated in the EBR-II at 900 °F to 0.8×10^{22} n/cm^2 (E > 0.1 MeV)

D. FUELS EVALUATION

J. E. Hanson

1. Thermal Irradiations of UO_2 + PuO_2 Fuel Pins

G. E. Culley, E. O. Ballard, S. H. Christensen, and N. D. Mills

a. Irradiations in GETR

Irradiation of four capsules containing prototypical FTR length fuel column pins in GETR continued. The goal burnup for these capsules is 50,000 MWd/tonneM. The approximate burnup for capsules PNL-59-7 and 8 is 45,000 MWd/tonneM and 36,000 MWd/tonneM for capsules PNL-59-9 and 10. All capsules operated at design power during this reporting period as a result of improved reactor loading characteristics. The predicted discharge date for capsules 7 and 8 is April 13, 1969, while capsules 9 and 10 will be discharged on June 22, 1969.

b. Transient Irradiations

The TREAT calibration test capsule is 90% completed and the data package has been drafted. A check of the physics calculations confirmed that the fully enriched and normal fueled sections in the fuel pin will operate at about the same power because of the different thermal dam materials (nickel versus aluminum and stainless steel).

2. Thermal Flux Irradiation of Hypostoichiometric UO_2-PuO_2

R. D. Leggett, L. A. Pember, and J. W. Weber

a. Postirradiation Examination of BNW-1 Fuel Pins

The measured changes in diameters for BNW-1-4, 6, 9 and 11 fuel pins are summarized in Table 5.D.I. BNW-1-11 which had the highest burnup (85,000 MWd/tonneM) showed only 0.0008 in. or 0.3% increase in diameter of the pin in the fuel region compared with the plenum section. The largest increase occurred

TABLE 5.D.I. Average Cladding Outside Diameters - Inch[a]

Axial Location Inches from Bottom of Pin	BNW 1-11	BNW 1-9	BNW 1-4	BNW 1-6
Top of Pin				
21	0.2498+	0.2503	Fuel Pin Cut - No Measurements Nominal 0.250	0.2493
19	0.2498	0.2504		0.2490
17	0.2499	0.2504		0.2489
15	0.2500	0.2505		0.2490
14	0.2502	0.2506		0.2491
13	0.2503	0.2508		0.2492
12	0.2504+	0.2509		0.2493
11	0.2504	0.2516		0.2494
10	0.2505	0.2502		0.2494
9	0.2505	0.2526	0.2535	0.2494
8	0.2505	0.2527	0.2549	0.2494
7	0.2505	0.2531	0.2527	0.2494
6	0.2504+	0.2532	0.2524	0.2496
5	0.2504+	0.2531	0.2527	0.2497
4	0.2504	0.2539	0.2528	0.2501
3	0.2504	0.2538	0.2529	0.2503
2	0.2505	0.2538	0.2559	0.2508
1	0.2507	0.2537	0.2545	0.2510
Bottom of Pin				
Max. diam increase compared to plenum.	0.0007	0.0035	0.0059	0.002
	0.3%	1.4%	2.3%	0.8%

a. *Average of four diameter measurements around circumference at 0°, 45°, 90°, and 135° for BNW 1-6, 1-9, and 1-11 and two diameters 0° and 90° for BNW 1-4.*

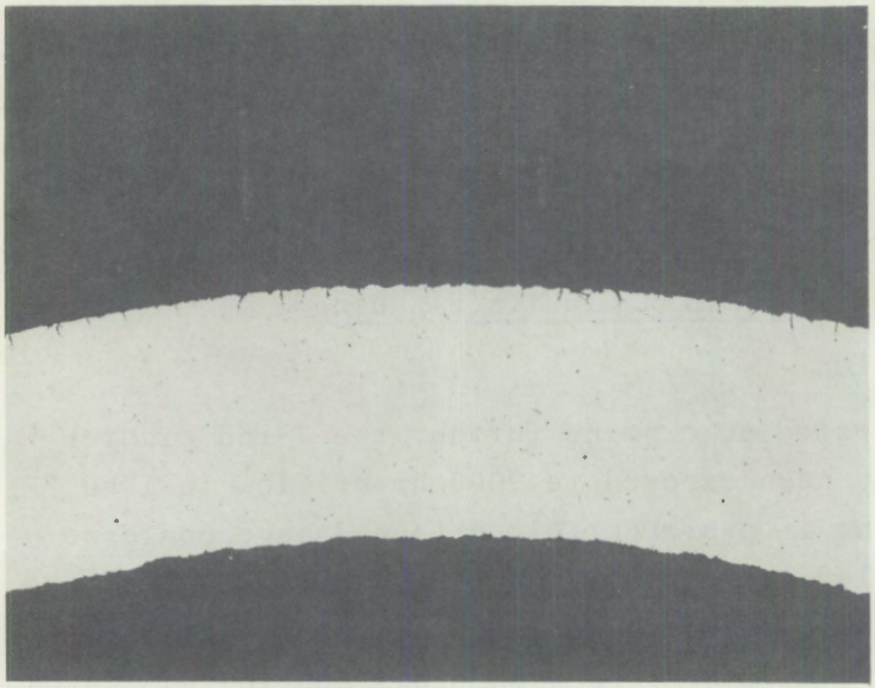
in the failed pin (BNW-1-4) which showed 0.0059 or 2.3%. BNW-1-6 and 9 showed increases of 0.002 in. (0.8%) and 0.0035 in. (1.4%), respectively. Since these fuel pins were irradiated in a thermal flux, the diameter increases are attributed to creep strain of the cladding fuel swelling and to the internal buildup of fission gas pressure.

3. Examination of 7-Rod Flow Cluster Cladding

M. K. Millhollen

BNW was requested to examine further the 7-rod cluster cladding which had been exposed to 9000 hr of flow in 1060 °F sodium to determine if preferential surface damage could be detected in the cladding directly under the wire wrap. Three cross sections of the cladding were mounted and polished and the location of the wire wrap on each cross section was carefully marked. The three polished cross sections were carefully examined under the microscope and although slight surface damage was visible on all surfaces exposed to sodium, no particular damage oriented to the wire wrap location could be detected.

One typical cross section was selected from which to prepare photomicrographs for record. The section of the cladding directly under the wire wrap is shown in Figure 5.D.1. This figure, as do the others, shows the cladding in the unetched as well as the etched condition. Three other locations on the cladding circumference are shown in Figures 5.D.2, 5.D.3, and 5.D.4. Typical surface damage is shown by penetration along the grain boundary in all the photomicrographs developed, but it is neither better nor worse at the wire wrap location.



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100X

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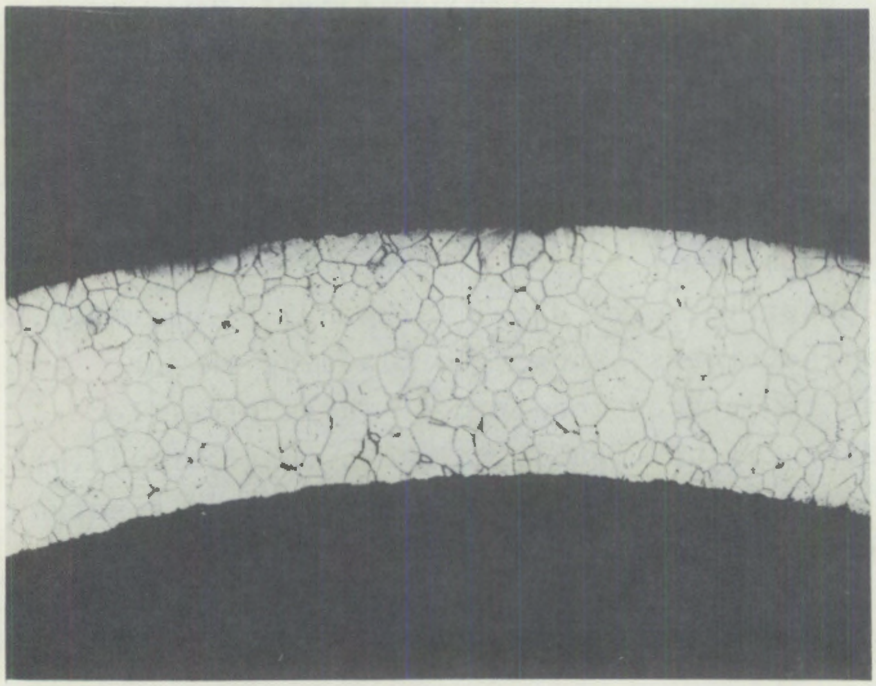

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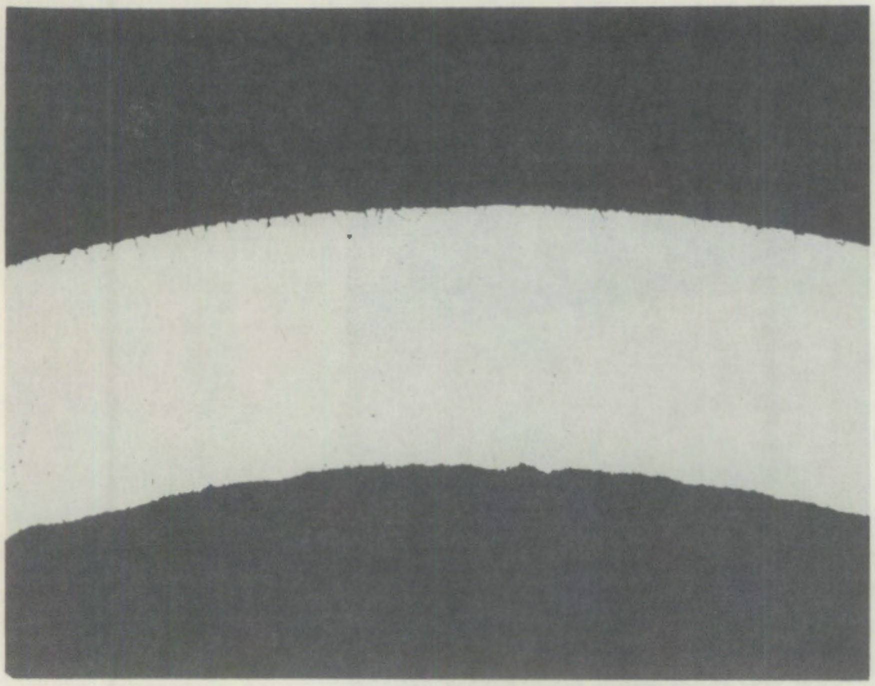
FIGURE 5.D.1. Cladding Surface Under Wire Wrap from Fuel Rod in 9000 hr, 1060 °F Sodium Flow Test



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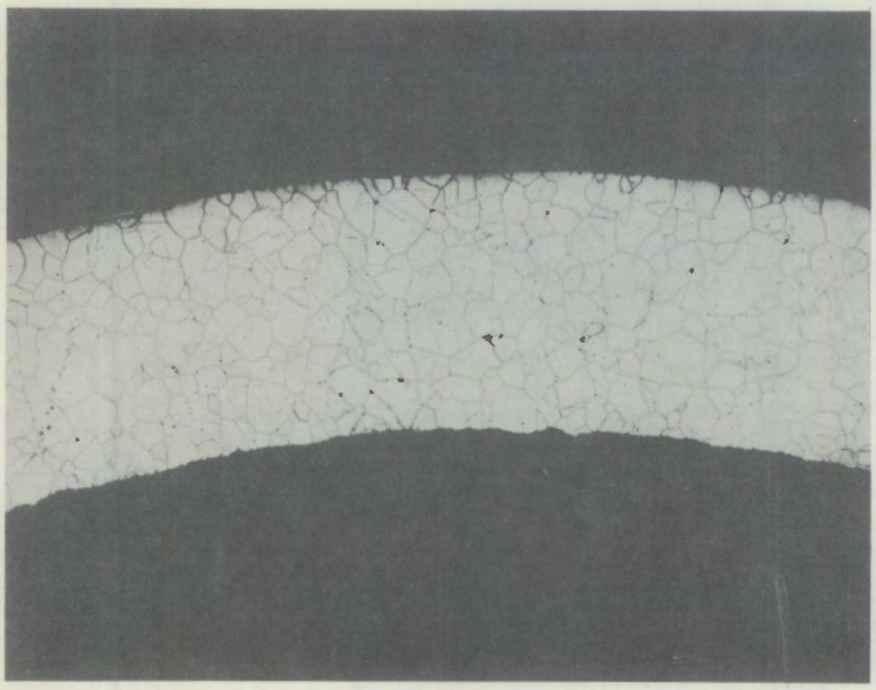

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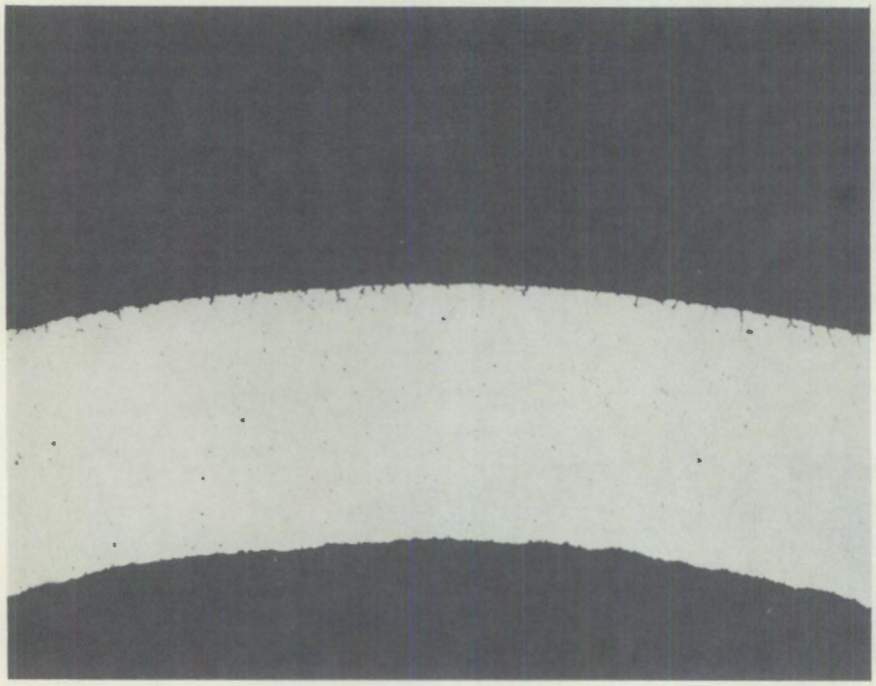
<u>*FIGURE 5.D.2.*</u> *Cladding Surface at 1st Random Position on Fuel Rod in 9000 hr, 1060 °F Sodium Flow Test*



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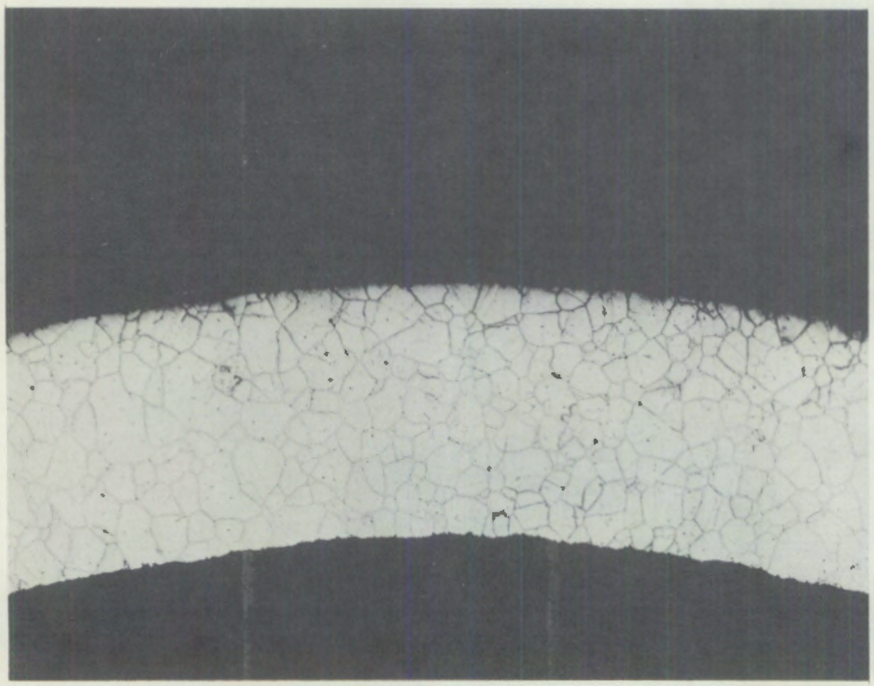

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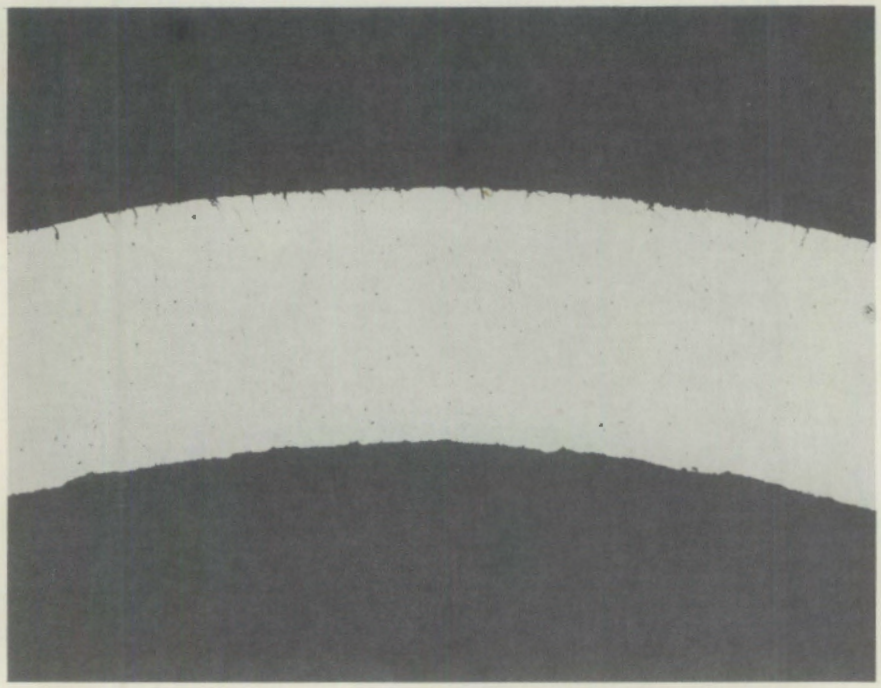
<u>*FIGURE 5.D.3.*</u> *Cladding Surface at 2nd Random Position on Fuel Rod in 9000 hr, 1060 °F Sodium Flow Test*



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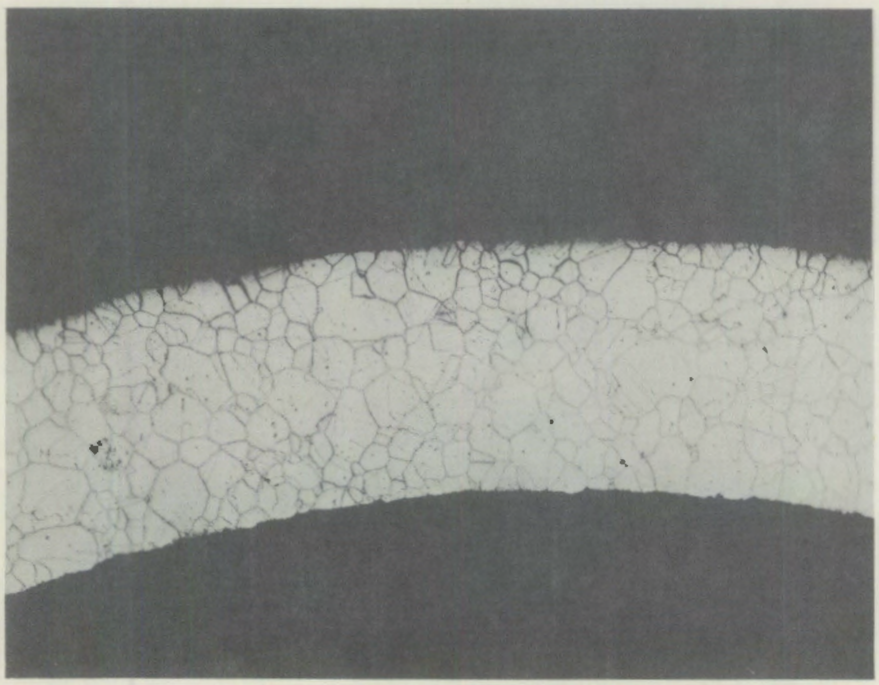

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FIGURE 5.D.4. *Cladding Surface at 3rd Random Position on Fuel Rod in 9000 hr, 1060 °F Sodium Flow Test*

E. OFFSITE FUEL PROGRAMS

G. A. Last

1. LMFBR Fuel Development

B. R. Hayward and F. M. Smith

Contracts are being prepared for Phase II of the Analytical Chemistry Program scheduled for completion by June 30, 1969. The analyses requested in Phase II are:

Uranium Assay	Carbon
Plutonium Assay	Nitrogen
Spectrographic Impurities	Density (geometric)
O/M Ratio	H_2O Content
Cl and F	Homogeneity
Gas Evolution	Tungsten

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