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FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
JUNE, JULY, AUGUST 1969

OCTOBER 1969

AEC RESEARCH &
DEVELOPMENT REPORT

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PACIFIC NORTHWEST LABORATORY

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BATTELLE MEMORIAL INSTITUTE

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UC-80, Reactor
Technology

FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
JUNE, JULY, AUGUST 1969

By

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October 1969

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FAST FLUX TEST FACILITY
QUARTERLY TECHNICAL REPORT
JUNE, JULY, AUGUST 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 June, July and August 1969.

PRIOR REPORTS IN THIS SERIES

BNWL-501	FFTF Quarterly Technical Report	1-1-67 to	3-31-67
BNWL-541	FFTF Quarterly Technical Report	4-1-67 to	6-30-67
BNWL-567	FFTF Periodic Technical Report	7-1-67 to	12-31-67
BNWL-660	FFTF Quarterly Technical Report	1-1-68 to	3-31-68
BNWL-880	FFTF Quarterly Technical Report	4-1-68 to	6-30-68
BNWL-917	FFTF Quarterly Technical Report	7-1-68 to	9-30-68
BNWL-941	FFTF Periodic Technical Report	10-1-68 to	2-28-69
BNWL-1090	FFTF Quarterly Technical Report	3-1-69 to	5-31-69

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FAST FLUX TEST FACILITY
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INTRODUCTION

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

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

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

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

FFTF ORGANIZATION CODES

AS OF 8-15-69

7N00 - FFTF Division - E. R. Astley, J. C. Cochran
 7N10 - Program Adm. and Control Dept. - F. J. Arrotta
 7N40 - Product Assurance Dept. - R. J. Squires
 7N60 - Construction Dept. - J. S. McMahon
 7N70 - Fuels Dept. - E. A. Evans, W. E. Roake, T. W. Evans
 7N71 - Offsite Fuels Programs - G. A. Last, B. R. Hayward,
 J. F. Williams
 7N72 - Cladding Development - T. T. Claudson, J. C. Tverberg,
 J. J. Holmes, F. R. Shober
 7N73 - Fuel Element Development - C. A. Burgess, R. E. Bardsley,
 J. W. Thornton, E. T. Weber
 7N74 - Fuels Quality Assurance - H. G. Powers
 7N75 - Fuels Evaluation - J. E. Hanson
 7N76 - Material Control - A. W. DeMerschman
 7N80 - Reactor and Plant Technology Dept. - P. L. Hofmann
 7N81 - Reactor and Plant Engineering - L. M. Finch, D. Marinos,
 D. P. Schively, W. L. Bunch
 7N82 - Safety and Systems Analysis - R. E. Peterson,
 D. D. Stepnewski, H. G. Johnson, R. A. Harvey, J. P. Hale,
 J. W. Hagen
 7N83 - Nuclear and Process Technology - R. E. Heineman,
 W. W. Little, P. D. Cohn, R. A. Bennett, D. C. Boyd,
 W. R. Lewis, J. G. Bradley
 7N90 - Engineering Dept. - D. L. Condotta, F. C. Grondemeyer,
 W. B. McDonald
 7N91 - Reactor Design - S. O. Arneson, F. J. Kempf, R. C. Walker,
 C. A. Munro, J. M. Yatabe
 7N92 - Fluid Systems - J. M. Batch, W. R. Wykoff, T. W. Withers,
 P. F. Shaw, R. V. Dulin
 7N93 - Instrumentation Control - C. D. Swanson, C. F. R. Smith,
 D. C. Thompson, M. R. Wood, R. R. Cone, J. W. Mitchell
 7N94 - Facilities and Site - E. M. Johnston, F. H. Shadel,
 C. L. Boyd, G. H. Strong
 7N95 - Engr. Coordination and Planning - J. R. Carrell,
 K. M. Harmon, D. R. Doman, C. W. Higby, H. L. Pringle,
 R. L. Roske, K. G. Toyoda, H. D. Lenkersdorfer
 7N96 - Metals, Materials and Codes - J. C. Tobin, M. T. Jakub,
 D. P. O'Keefe, J. E. Spanner, R. N. Johnson, R. H. Todd,
 J. E. Irvin, D. E. Mahagin, R. A. Moen, J. W. Helm,
 J. H. Cox

When work is actually performed for FFTF by another BNW Division, that code is also included as shown in the following tabulation:

7400 - BNW Safety and Standards Division
 7A00 - Chemistry and Metallurgy Division
 7C00 - Physics and Engineering
 7K00 - Environmental and Life Sciences Division
 7M00 - Systems and Electronics Division

■. OVERALL FFTF PROGRESS

This report is issued by BNW to present technical information of general interest to the LMFBF community. The report is not intended to provide a full description of progress in management of the FFTF Project.

PLANT DESIGN (Chapter II)

The current overall design data summary for the FFTF vertical core concept is given in Chapter II. There has been no significant change in the total project design as previously described.⁽¹⁾ However, an active design optimization program is in progress, and the Preliminary Safety Analysis Report is in preparation.

COMPONENTS (Chapter III)

Much of the current components work is directed toward the heat transport and sodium processing systems. Metallurgical studies have shown that carbon steels will be adequate for sodium piping, pumps, valves and storage tanks where the lifetime steady-state temperature is below 600 °F. New design parameters have been established for the heat transport system including reactor vessel maximum temperature of 1050 °F.

INSTRUMENTATION AND CONTROL (Chapter ■ V)

Successful tests have been conducted at 720 °F on a commercial unbonded strain gage transducer rated by the manufacturer at only 600 °F. New laboratory methods have been developed for calibrating thermocouples, determining thermocouple time constants, and for remote coupling of thermocouples and cables.

1. *FFTF Periodic Technical Report October 1968 Through February 1969, BNWL-941. Battelle-Northwest, Richland, Washington.*

Advanced new laboratory techniques for measurement of coolant mixing and flow distribution, and for measurement of gas entrainment, are being developed. Use of absolute resistance has been demonstrated for determining hydraulic mixing. For determining coolant velocity in subchannels, a new noise analysis technique has given satisfactory results. Successful results have also been obtained in use of ultrasonic techniques for measuring gas entrainment.

SODIUM TECHNOLOGY (Chapter V)

Chapter V presents a brief summary of work being done by the BNW Sodium Chemistry Subdepartment on fission product behavior and control, and on sampling and analysis of sodium and cover gas. Further details are given in BNWL-1200-1. (1)

CORE DESIGN (Chapter VI)

Work directed toward evaluating the effect of stainless steel irradiation-induced swelling on FFTF design and operation is discussed in Chapter VI. Consideration of swelling in constraint mechanisms has resulted in a revised design for core radial constraint. Examination of the effect of swelling on fuel burnup indicates that fuel duct bending resulting from differential swelling is the major potential factor limiting fuel exposure in the current reactor design.

Initial heat transfer tests on fuel subassemblies indicate that excellent cooling occurs at the region under the wire wrap spacer and that the potential cladding hot spot directly under the spacer may not exist. A 3600 hr fluid flow test at 1060 °F and a velocity of 27 ft/sec was successfully concluded on a full scale fuel subassembly.

Development is in progress on the classical problem of remote welding of closed loop piping connections to the reactor.

1. Quarterly Progress Report Sodium Chemistry Subdepartment, July, August, September 1969. BNWL-1200-1. Battelle-Northwest, Richland, Washington.

Initial results on remote welding of a closure seal between the closed loop nozzle plug and the reactor top face spool piece show that complete removal of sodium contamination will be required.

FUELS AND MATERIALS (Chapter VII)

Fuels

Section VII-A presents data relating biaxial stress and temperature to rupture life of irradiated stainless steel. Excellent correlation has been found with the so-called "theta parameter."

Results are given on tests made on EBR-11 driver fuel pin pins which became shortened during irradiation. It was concluded that the observed shortening was the result of irradiation growth of preferentially oriented alpha-uranium.

Hybrid computer equations are being developed to describe fission gas release, pressure build-up, and three-zone sintering of fuel pins. Preliminary results show reasonable agreement between calculated and observed axial hole diameters and the percent of fission gas released.

Materials

Chapter VII-B describes computations which suggest that irradiation-induced swelling of stainless steel may be less at 1000 °F and above, than is predicted by the current swelling equation.

Results of several high strain rate tests completed on annealed irradiated Type 304 SS indicate that at temperatures above 1000 °F reduced ultimate strength is associated with high strain rates.

Postirradiation analysis of capsules BC-1 and BC-4 irradiated in EBR-II showed that the ^{10}B burnup in the boron-carbide was 1.58 at.%. This implies that almost 30% of the helium gas formed during irradiation at 1125 °F was released.

FUELS RECYCLE (Chapter VIII)

In the LMFBR fuel supplier prequalification program Babcock and Wilcox, NUMEC and United Nuclear are processing raw materials. For physics reasons, the aggregate of samarium, europium, gadolinium and dysprosium in FFTF fuels has been limited to 100 ppm.

A number of studies are in progress on methods for fabrication of pellets and fuel pins. In an experiment designed through use of a statistical model, it was found that forming pressures above 30,000 lb/in.² are required to ensure control of sintered density of UO_2 . Other tests showed that milling of PuO_2 powder produces reductions in pellet sintered density.

Studies on fuel pin fabrication revealed, as anticipated, that strength of wire-end termination welds is inversely related to the strength of weld penetration.

In the LMFBR Fuel and Cladding Information Center at BNW, Master File No. 1 now contains results of 2197 tests on stainless steels.

PHYSICS (Chapter IX)

Analysis of the 56B critical experiments discloses that there is still a lack of agreement between calculations and experiments. The calculated worth of ^{239}Pu is too large near the core center and in the radial reflector, and other inconsistencies also exist. The reason for the disagreement has not yet been found.

A comparison of HOMSET cross sections with RNDP cross sections showed that HOMSET provides adequate agreement with measured spectra for use in survey applications, even though some deficiencies exist. ENDF cross sections give somewhat better overall agreement.

Proposed use of a depleted UO_2 pad as part of the reactor shielding structure is expected to cause no heat generation or radiation problems appreciably different from those associated with the natural alpha decay of the uranium.

SAFETY (Chapter X)

Two new computer codes have been developed for calculation of safeguards data. CONRAD is a new code prepared to calculate transient temperatures and pressures within the FFTF inner containment barrier following hypothetical accidents. Another new code, not yet named, is used for computation of long-term pressure and temperature transients following a hypothetical core accident.

In other safeguards work, an extensive analysis of the FFTF Plant Protection System is in progress. A unique new method has been developed to define required testing intervals for the system.

II. PLANT DESIGN

This Chapter presents, as Table 2-I, a current summary of principal FFTF concept characteristics and data.

TABLE 2.I. *Summary of Principal Concept
Characteristics and Data
as of 8/20/69
C. W. Higby (7N95)*

	<u>Units</u>	<u>Values</u>	<u>Status(a)</u>
A. <u>General Plant Data</u>			
1. Core arrangement		vertical	1
2. Design life	years	20	1
3. Peak flux	n/cm ² -sec	7.2×10^{15}	2
4. Total power, initial	MW _t	400	1
5. Reactor coolant		sodium	1
6. Core volumn	liters	1033	2
7. Total reactor flow	gal/min	42,240 init. 43,290 ult.	2
8. Reactor pressure drop design maximum	psi	145	2
9. Reactor bulk inlet temperature initial	°F	600	2
design maximum	°F	800	1
10. Reactor bulk outlet temperature initial core	°F	900	2
design maximum	°F	1050	1
11. Core temperature rise average initial	°F	300	2
design maximum	°F	400	1
12. Reactor cover gas		argon	1

(a) 1 - Firm requirement or definite concept characteristic.
 2 - Current reference design value.

TABLE 2.I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status</u> ^(a)
B. <u>Reactor Vessel</u>			
1. Diameter	feet	17	2
2. Height	feet	50	2
3. Wall thickness	inches	2	2
4. Wall fluence, total (at 1000 °F)	nvt	2×10^{22}	2
5. Material		304 SS	1
C. <u>Core Design</u>			
1. Number of core lattice positions		91	1
2. Number of driver fuel subassemblies		76	2
3. Number of closed loop positions		6	2
4. Number of contact instrumented in-core open test positions		2	2
5. Number of proximity instrumented open test positions		1	2
6. Number of in-core safety rods		3	2
7. Number of in-core control rods		3	2
8. Number of peripheral control rods		15	2
9. Equivalent core diameter	inches	~48	2
10. Active core height	inches	36	2
11. Radial reflector material		nickel	2
12. Reflector thickness			
axial	inches	6	2
radial	rows	3	2
13. Fuel pin heat transfer area	ft ²	2980	2

TABLE 2.I. (contd)

	Units	Values	Status (a)
14. Core coolant velocity, maximum	ft/sec	30	2
15. Direction of coolant flow		upward	1
D. <u>Driver Fuel</u>			
1. Fuel composition		18 to 26 wt% PuO ₂ 74 to 82 wt% UO ₂	2
2. Cladding material		Type 316 SS	1
3. Linear heat generation rate, average (at 90% total power)	kW/ft	7.0	2
4. Overpower factor		1.20	2
5. Peak linear heat generation at over-power (at 90% total power)	kW/ft	14.6	2
6. Target burnup, average	MWd/tonneM	45,000	2
7. Cladding heat transfer coefficient	Btu/hr-°F-ft ²	37,500	2
8. Maximum fuel temperature (steady state)	°F	4230	2
9. Fuel assembly length	feet	12	2
10. Fuel geometry		Hexagonal pin cluster	1
11. Pin diameter	inches	0.230	2
12. Spacer wire diameter	inches	0.056	2
13. Number of pins per assembly		217	2

TABLE 2.1. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status(a)</u>
14. Subassembly cross-section outside dimension (across flats)	inches	4.615	2
15. Lattice spacing	inches	4.715	2
16. Duct wall thickness	inches	0.140	2
E. <u>Physics Data</u>			
1. Delayed neutron fraction		0.003	2
2. Neutron lifetime	seconds	4.0×10^{-7}	2
3. Doppler	T(dk/dt)	-0.004	2
4. Power density, peak	MW/liter	0.75	2
5. Power distribution (peak/average)			
radial		1.40	2
axial		1.24	2
total		1.74	2
F. <u>Test Facilities</u>			
1. Closed loops			
- number of positions		6	2
- location		2 adjacent to core center	2
		1 at mid-radius	2
		3 at core periphery	2
- power handling capability	MW	4	2
- test section outlet temperature	°F	1400 (bypass flow permitted)	1

TABLE 2. I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status (a)</u>
- number of loops installed initially		2 general purpose loops of 2 MW capacity and up to 2 special purpose loops provided by users	2
- test section length	inches	36	2
- test section diameter	inches	2.5-3.0	1
- material (in-core tube)		316 SS	1
2. In-core open test positions			
- number with contact instrumentation		2	2
- with proximity instrumentation*		1	2
- power			
- coolant flow rate			
- test assembly length		Same as driver fuel assembly	
- test assembly cross section		Same as driver fuel assembly	
- coolant		Same as driver fuel assembly	
3. Short term irradiation facility (capability for future installation)			

* *One open test position will have superior accessibility. Driver fuel positions can be used as open test positions with limited accessibility*

TABLE 2.I. (contd)

	Units	Values	Status(a)
- type		trail cable	1
- number		1	2
- location		core periphery	2
- minimum irradiation time at constant flux	minutes	1	2
- sample length	inches	24	2
- sample cross sec- tion, maximum	inches	2.5	2
G. <u>Heat Transport System</u>			
1. Primary loops			
- number		3	1
- primary loop material		304 SS	1
- primary loop flow (per loop) (initial)	gal/min	14,080	2
2. Primary pumps			
- number		3	1
- design pump head	feet	500	2
- available net posi- tive suction head	feet	51	2
- design temperature	°F	1050	1
- speed control		wound rotor motor with liquid rheostat	1
- motor power			
brake	HP	1790	2
rated	HP	2500	2
3. Intermediate heat exchangers			
- number		3	1
- type		vertical shell and tube	1

TABLE 2.I. (contd)

	<u>Units</u>	<u>Values</u>	<u>Status(a)</u>
- LMTD			
initial	°F	85	2
ultimate	°F	99.1	2
- capacity			
initial	MW	133	2
ultimate	MW	157	2
H. <u>Shielding</u>			
1. Within reactor vessel			
- material		stainless steel	2
- configuration		hexagonal assemblies	2
- thickness	inches	21	2
2. Cover shield			
- material		low alloy steel	2
- thickness	feet	4	2
- coolant		argon, helium or Na	2

III. COMPONENTS

A. MATERIALS EVALUATION FOR SODIUM RECEIVING AND PROCESSING SYSTEM (7N96)

R. A. Moen

An evaluation of carbon steels for sodium piping and equipment was performed in support of the Sodium Receiving and Processing System (SRPS). Included in the evaluation was a cost comparison between carbon steel and stainless steel or stainless clad carbon steel, and a cursory review of the contamination and/or cleaning problems associated with a mixed alloy system.

The evaluation concluded that carbon steels should be adequate for use in those areas of the SRPS where lifetime temperatures (steady state) remain below 600 °F. Carbon steel can also be used in large storage tanks where sodium is normally stored at temperatures of 250 to 400 °F, then raised in temperature to a 600 to 700 °F range over a period of a few hours before being transferred into a coolant loop. Precautions will have to be taken during system fabrication where the carbon steel is joined to stainless steel. Chemical cleaning of a carbon steel/stainless steel system with anything but sodium does not appear advisable at this time. This requirement will place additional emphasis on component cleanliness prior to and during fabrication, which would also be the case for an all stainless steel system being constructed for sodium service. By utilizing carbon steel piping, pumps, valves and storage tanks in the fresh sodium receiving and processing subsystem and the contaminated sodium processing subsystem (plus carbon steel storage tanks in the reactor primary and secondary sodium processing subsystems) major savings could be made in material and component costs.

B. HEAT EXCHANGER SIMULATION (7N82)

1. IHX Model Study for Hybrid Simulation

A. L. Gunby

A study of IHX modeling techniques was completed to re-evaluate a versatile model for the FFTF hybrid simulation. Differencing techniques using 2-point back, 3-point back, central, and alternating central and 3-point back approximations were compared to the IANUS approach (Westinghouse) for transient conditions. In addition, all models were compared to exact solutions for steady-state conditions with unbalanced flows.

As a result of the study, the FFTF hybrid simulation now uses the alternating central and 3-point back model (called "mixed difference" model) for the IHX. Specific conclusions are as follows:

- a. For steady-state, unbalanced flows, only the mixed difference and IANUS models provide good energy balances. Of the two, the mixed difference model provides the better temperature distribution.
- b. For flow decays and temperatures, both positive and negative, the mixed difference and IANUS models responded identically and more realistically than the other models.

2. Dump Heat Exchanger Analysis

R. N. Madsen

Six simplified models of the 10-pass DHX have been developed and preliminary comparisons have been made with the reference simulation described in the previous quarterly report.⁽¹⁾ These simplified representations assume that the

1. Fast Flux Test Facility Quarterly Technical Progress Report, BNWL-1090, p. 3.55, Battelle-Northwest, Richland, Washington.

actual cross-flow exchanger may be accurately represented as a counterflow exchanger with only three radial nodes, one each for sodium, metal and air. The axial temperature derivatives were approximated by dividing the DHX total tube length into j nodes and then using alternating central and three-point backward difference methods. The models were then programmed for solution on a dynamic systems analysis code. The number of axial nodes, j , was variously assumed to be 4, 6 and 12. Initially, the appropriate heat transfer coefficients were determined according to local node temperatures; however, subsequent analysis indicated that average coefficients based on average temperatures were sufficiently accurate. The simplified models which have been investigated are: four-nodes with local coefficients; four nodes with average coefficients; four nodes with average coefficients and thermophysical properties; six nodes with local coefficients; six nodes with average coefficients; and 12 nodes with average coefficients.

Comparisons of predicted performance have been made with the TAP reference model for both steady-state and transient behavior with balanced and moderately unbalanced flows. These comparisons have indicated that the simplest model, the four node with average coefficients and properties, agrees very well with the reference and is almost identical with the more detailed six and 12 node models.

C. DESIGN PARAMETERS FOR HEAT TRANSPORT

J. P. Petrek

A new set of design parameters was established for the Heat Transport System. The primary coolant system will be designed for a reactor vessel outlet temperature of 1050 °F. The ultimate core mixed mean outlet temperature will be 1100 °F and core by-pass flow will be used to reduce this temperature to 1050 °F at the reactor vessel outlet.

The system design capability shall be achieved when operated with variable reactor mixed mean outlet coolant temperature consistent with:

1. Initial operation using mixed oxide driver fuel at the following operating conditions:

Reactor Power	400 MW _t
Reactor Core Outlet Temperature	900 °F
Reactor Core Temperature Rise	300 °F
Minimum Bulk Mean Reactor Outlet Temperature at 400 MW _t	850 °F
Allowable Pump Head at Design Flow Rate	500 f t TDH
IHX LMTD	85 °F

2. Ultimate operation using high temperature driver fuel at the following maximum operating conditions:

Reactor Core Outlet Temperature	1100 °F
Reactor Vessel Outlet Temperature	1050 °F
Reactor Core Temperature Rise	400 °F
Heat Transport System Temperature Change	350 °F
Allowable Pump Head at Pump Design Flow Rate	500 f t TDH
IHX LMTD	100 °F

The above maxima are not necessarily coincidental conditions for future operation. These design conditions should provide for reactor operation under the following set of potential core conditions:

Reactor Power	470 MW _t
Reactor Core Outlet Temperature	1100 °F
Reactor Vessel Outlet Temperature	1050 °F
Reactor Core Temperature Rise	350 °F
Allowable Pump Head at Design Flow Rate	500 ft TDH

Section 1.0 of the Heat Transport System (CDD-51) was approved by the CCB and sent to RDT, Westinghouse, and Bechtel.

D. FUEL DUCT FABRICATION DEVELOPMENT (7N96)

W. F. Brown

1. Objective

The objective of this program is to develop and evaluate fabrication processes required to produce fuel ducts for the FTR. Development, evaluation, and qualification of commercial sources of supply is a program goal; and the establishment of two or more competitive and continuing sources of supply is an additional objective.

2. Duct Wear Pad Attachment

In order to determine the feasibility of making simultaneous spot welds on opposite faces of the hex duct, a series of simultaneous welds were made between pairs of nonaged Inconel 718 and Type 304 SS with a 4-in. thick solid copper back-up and current conducting bar. Peel tests and tensile-shear tests were used for preliminary evaluation to establish a starting weld schedule. The results indicated that simultaneous welds can be made. However, a fixture is needed inside the hex tubes to conduct welding current to opposite faces and to prevent tube distortion. An internal adjustable copper back-up and current conducting fixture will be inserted and tested during the next reporting period.

Inconel 718 sheet material and Hynes Stellite Alloy 6B was purchased and received for wear pad attachment studies. The Inconel 718 sheet was sheared and rolled to 0.065-in. thickness to provide samples for forming, heat treating, and welding studies. Press brake forming dies were made and 18 half-hex pads were formed. Prior to forming, the edges were beveled in accordance with the Driver Fuel Assembly drawing SK-3-15149. After heat treatment, attachment will be made by resistance spot, gas tungsten-arc, and electron beam welding processes.

E. HIGH TEMPERATURE ULTRASONICS INSPECTION DEVELOPMENT (7N96)
R. W. Smith and N. L. Johnson

1. Objectives

Work is in progress on the development of an ultrasonic inspection method for potential use at 500 °F and above for detecting materials defects. This work is part of a general program to establish FFTF surveillance and in-service inspection methods. Program objectives include, (a) verification of expected behavior of critical components in the primary coolant boundary and closed loop systems, and (b) early detection of abnormal or unexpected behavior which will have a direct influence on the reliable and safe operation of the FFTF.

A Sperry Model 50D467 high-temperature ultrasonic wheel designed for use at ~250 °F maximum ambient temperature is being tested (a) for resistance to gamma radiation, (b) in dynamic operation at high temperature, and (c) for sensitivity to defects on various surface finishes ranging from 32 μ in. and to 500 pin. if required. It is planned to substitute improved materials where available, and recommend modifications or additional development if deemed necessary to achieve reliable operability under FFTF conditions at 500 °F and above.

IV. INSTRUMENT AND CONTROL

A. SENSOR DEVELOPMENT (7N93)

1. Pressure Sensors for Sodium Service

R. D. Crosier and K. O. Creek

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

Tests of the best available electromechanical type commercial press transducers in static sodium have given the following results:

- a. Unbonded strain gage type transducer rated at 600 °F and 250 psia has been operated successfully for the past seven months. The test temperature has gradually been increased over the rated temperature to 725 °F, and the transducer continues to operate. The test temperature will be increased until the transducer fails.
- b. A linear variable differential transformer type transducer rated at 900 °F and 200 psia failed at 700 °F after a test period of 10 weeks. Post-failure analysis by the vendor showed that the malfunction was caused by insulation failure in a temperature compensation winding which allowed adjacent coils to touch each other.

2. In-Reactor Coolant Temperature Sensors

N. C. Hoitink, R. C. Weddle, and J. L. Jackson

The task focuses on the developmental testing and evaluation of thermocouples to provide reliable, commercially-available sensors for measuring the coolant temperatures in

the FTR thermal and nuclear environment. Pre- and post-irradiation electrical characterization studies provide information about accuracy, reliability, response time and other parameters of candidate sensors exposed to a fast flux reactor (EBR-II) environment.

No thermocouples irradiated in EBR-II were available for testing during this reporting period; thus, efforts concentrated on detailed preparations for the irradiated thermocouples which should arrive soon, and for the ORNL-purchased devices expected early in November. The thermocouples in EBR-II Capsules BT-3 and BT-4 have completed irradiation, and postirradiation tests should be completed prior to arrival of the ORNL thermocouples for preirradiation examination.

A newly designed and fabricated aluminum-bronze equalizing block will permit calibration of approximately 30 thermocouples per cycle. This will greatly reduce the time required for calibration. The new block will also reduce undesirable thermal gradients which existed in earlier calibrations. Laboratory tests of the block indicated acceptable performance.

A newly designed and fabricated instrument for determining thermocouple time constants gave excellent performance when tested with single RC circuits. However, preliminary re-examination of the preirradiation response curves for the thermocouples contained in Capsules BT-3 and BT-4 gave evidence that the 0.0625-in., grounded junction thermocouples have more than one significant time constant for a step change in temperature. No explanation for these dual time constants is offered at this time.

A recently completed mechanical jig using crimp-type connectors is expected to solve the problem of remote attachment of lead wires to irradiated thermocouples.

B. LOW-LEVEL NEUTRON FLUX INSTRUMENTATION (7N93)

C. N. Jackson, N. C. Hoitink, and R. C. Weddle

This task encompasses the developmental testing and evaluation of sensors, cables and associated components for high-sensitivity, low-level neutron counting systems.

1. Tentative Technical Specifications for FFTF Neutron Fission Counters

A new detector specification (B-67398) entitled, Tentative Technical Specifications for FFTF Neutron Fission Counter, has been written. (A companion specification for ^{10}B -lined proportional counters is nearly finished.) Compared to detectors for conventional thermal reactors, the FFTF requires high sensitivity detectors with significantly improved capability for tolerance of combined temperature (900 °F) and gamma (10^6 R/hr) environments, and long cables (~45 ft) which are also subjected to high gamma and temperature levels. In addition, a relatively high neutron sensitivity of 0.4 counts/sec/nv is required for operational control and safety systems.

2. Tests on Cables

a. Thermal "Noise" Induced in Cables and Connectors

Preliminary investigation indicated a possibly serious, temperature-induced "noise" problem with coaxial signal cables used for low-level flux monitoring. Two commercial cables designed for high-temperature (650 to 1000 °F) application generated significant noise at temperatures of 630 °F and above when heated in a furnace. At 1200 °F, noise levels were so high that the ion chamber voltage had to be reduced.

The magnitude of the noise pulse must be reduced in order to have an acceptable low-level neutron counting system for hot startup. Work is continuing on this problem.

b. Effect of Heating Coaxial Cables to 900 °F

Engineering developmental testing concluded a relatively long term (720 hr) temperature examination of high-temperature coaxial cable (silica dielectric) that demonstrated good stability. The cable, heated to 900 °F, transmitted signal pulses without measurable loss throughout the test. During the test, the cable capacitance remained constant at 234 pF, while the insulation resistance held within the range of 2 to 4×10^9 ohms.

C. DETECTION AND LOCATION OF FAILED FUEL

1. Introduction

In the FFTF, a unique and difficult design problem is presented by the necessity to identify the actual fuel assembly in which a cladding failure has occurred. A system called "FEDAL" (Failed Element and Location System) has been conceived and is being developed for this purpose. This system development involves many aspects of sodium technology and hence a general description of the current conceptual design efforts is presented for the information of the technical community.

2. System Requirements

Requirements for the FEDAL system are described in BNWL-955⁽¹⁾ as follows.

- Instrumentation shall be provided for each driver fuel position to detect a fuel pin failure promptly and to locate the affected subassembly.

1. R. C. Walker. FFTF Reference Concept Summary Description. BNWL-955, Battelle-Northwest, Richland, Washington, January 1969.

- Each closed loop, package loop, capsule and at least six open loops to be instrumented for inlet⁽¹⁾ and outlet coolant temperatures, coolant flow and prompt fuel failure detection.
- Fuel will be considered to be failed if the cladding has been violated at any location including the gas plenum region. Locating failed fuel should be possible within the refueling time period for the reactor.

3. General Description of FEDAL Concept

The general concept for the FEDAL system is described in BNWL-955.

The detection of released fission gases was selected as the reference concept for the following reasons:

- This system has the potential for locating a failed element from which only gas is released either continuously or in a burst.
- It has the potential for locating an element with the reactor shut down.
- It has the potential for identifying single or multiple elements which may fail at the same time.

To ensure that the sodium sample stream can obtain gas from any fuel pin in the subassembly and that the amount of gas in the sample stream be maximized, a fission gas concentrator is required. The concentrator (also known as a vortex generator, or a fission gas separator) is located above the fuel pins in the subassembly outlet coolant flow. Gas bubbles are forced to the center of the concentrator by centrifuge action where a perforated tube serves as a collector and the starting point for the sodium sample line.

1. *Bulk reactor inlet temperature for open test positions.*

These sodium samples are removed from the reactor vessel to shielded cells where the gas is separated from the sodium. The sodium is returned to the sodium purification surge tank and the separated gas is delivered to a monitoring location where the GM detectors monitor for an increase in gas activity.

4. Current Definition of FEDAL System Concept (7N81)

G. R. Waymire

To provide the basis for continued conceptual design activity by Westinghouse, a FEDAL monitoring concept was defined. The significant feature of the concept is an integrated FEDAL sample monitoring module that may be combined with other modules in a compact shielded array. The module array has 34 positions in a single cylindrical tank approximately 4 1/2 ft in diameter. The tank and its modules provide FEDAL sample monitoring for one-third of the FTR core; hence, three tanks are required to service the complete core. The monitoring tanks and their connecting piping are located adjacent to and outside of the machinery dome at the level of the reactor head. This location allows the monitoring modules to be at the high point of their individual sodium sampling lines thereby avoiding entrapment pockets for the entrained fission gas in the FEDAL samples.

5. Research and Development

Development work is in progress to demonstrate the practicality of the proposed fission gas concentrator.

D. IN-CORE GAS CONCENTRATOR DEVELOPMENT (7N91)

D. R. Dickinson and J. Muraoka

In order to detect failed fuel assemblies and to determine which assembly has failed, it has been proposed to sample sodium from each individual fuel channel, and to analyze this

sample for the presence of fission gas. To ensure that the sample collected contains a large fraction of the total gas released, it has been proposed to concentrate the gas bubbles at the center of the channel by means of a vortex-type separator device. The proposed device would be located at the downstream end of the fuel assembly and would include a vortex generator, consisting of a set of vanes to impart rotation, a length in which the gas bubbles are forced to the center by the radial pressure gradient, and probably a set of recovery vanes to recover the rotational velocity head. A sample line will draw off sodium plus any gas present from the axis of the channel at the end of the rotational section.

An apparatus for full-scale, full-flow testing of the behavior of vortex generators in a 3.76 in. ID flow tube was completed and put in place in an existing hydraulic loop. This apparatus includes a moveable probe to measure both the axial and tangential components of velocity as a function of radial position. Efficiency in bubble removal was measured by use of hollow silica microspheres having a size range of 30 to 125 μ which simulated small gas bubbles. A known amount was injected upstream of the separator as a dilute aqueous suspension, and the amount removed was collected as a sample.

Two vortex generators were tested, both with similar swirler and deswirler blades. One had eight blades with a 45-degree discharge angle (measured from the axis) on a 1.5-in. diameter streamlined hub; the other had eight 50-degree blades on a 2-in. hub. For each, separations of 12 and 24 in. between swirler and deswirler were studied. All the data were at a flow of 450 gal/min of cold water.

The measured velocity profiles were generally as expected. Both the axial and tangential velocity were maximum at a point about midway between the center and outside of the vortex; were somewhat lower at the outside, and were near zero at the center. The swirl angle in the fluid was about 10 degrees less than the discharge angle of the blades, indicating that more blades or a larger blade length might improve performance. The tangential velocity near the center decayed rapidly with axial distance; otherwise the velocity profiles changed little. The axial velocity dropped to nearly zero at the center, but there was no confirmed instance of reversal of axial flow and internal recirculation near the center, such as is believed to be desirable in achieving good efficiency of sample pick-up.

The overall axial pressure drop was about 2 psi at 450 gpm. The deswirlers tested appeared to be inefficient in recovering rotational energy as pressure; the pressure near the wall was actually less downstream of the deswirlers than upstream of it. The radial pressure depression (pressure at the wall minus that at the sample point with no sample flow) was 7.0, 4.0, and 2.4 psi at axial locations of 3, 12, and 21 in., respectively, downstream of the swirler blades with the 45-degree blades.

The efficiency of removal of microspheres was measured with a sample flow rate of 1.5 gpm. Data have not been completely analyzed, but in all cases removal was poor and was estimated by visual observation of the samples collected to be on the order of 1% of that injected.

A limited series of tests were made which attempted to induce axial flow reversal and internal recirculation near the center in an effort to improve efficiency of sample withdrawal. This desired flow pattern was not obtained and separation efficiencies remained low.

The results of these preliminary tests indicate that significantly higher tangential velocities must be obtained to improve concentration of fission gas bubbles into the sampling tube. The new RPD design has a higher blade angle (e.g., produces higher tangential velocities for the same axial flow) and will be tested during the next reporting period.

E. MEASUREMENT OF COOLANT MIXING AND FLOW DISTRIBUTION (7N91)

R. A. Hildner and T. E. Reihman

1. Objective

The objective of this work is to provide flow distribution data for the FIR fuel assembly design. The major efforts during the quarter were to demonstrate and adapt noise analysis and absolute resistance measurement techniques for use in specific FIR fuel assembly hydraulic studies.

2. Absolute Resistance Techniques

The experimental evaluation of the absolute resistance technique for coolant mixing determination in FFTF fuel assemblies was completed using a full-scale dual channel test section. The purpose of the experiment was to determine the reproducibility with which the solution resistance could be measured with different conductivity sensing probes. Salt solutions of known concentration were circulated in a small volume closed loop system and the change in solution resistance was measured as a function of concentration for eight candidate probes.

Three tests were made in which the salt concentration was varied from 0 to 500 ppm (by weight) and the solution temperature was maintained at $90^{\circ}\text{F} + 1/2^{\circ}\text{F}$. For these three runs, the reproducibility of the data was found to be with +4% for each of the eight individual probes. Most of the data showed reproducibility within $\pm 2\%$. This overall reproducibility included the effects of all pertinent system parameters, e.g., water purity, salt weight, water weight, probe surface characteristics, electrical instrumentation characteristics, velocity (between 11 and 22 ft/sec) and temperature (between 89.5°F and 90.5°F).

Changes in solution temperature were found to have a marked effect on the absolute resistance measurements. To evaluate this effect, a fourth run was made in which the temperature was varied between 80°F and 100°F . A maximum sensitivity of 1.99% of reading per degree Fahrenheit and an average sensitivity of 1.25% of the reading per degree Fahrenheit were found, for the eight probes studied. This sensitivity to temperature implies that temperature control and measurement must be maintained within $\pm 1\%$ during calibration and experimentation to obtain the desired accuracy.

The sensitivity of supposedly identical probes varied by as much as $\pm 40\%$ from the mean sensitivity of the eight probes studied. This implies that each of the probes will have to be calibrated individually. Secondly, it was found that the sensitivity of some of the probes increased slightly with time after probe excitation. To eliminate this effect, data were taken at a set time after probe excitation, the period from 5 to 15 sec after excitation being selected. This timing procedure for excitation and measurement will henceforth be employed in the calibration and experimentation.

The results of the experiment show that this absolute resistance technique is acceptable for measuring local salt concentrations in FFTF-size coolant subchannels. These salt concentration measurements will provide the data for the detailed determination of coolant mixing between fuel bundle subchannels.

3. Noise Analysis Techniques

Experimental evaluation of the noise analysis technique for coolant subchannel velocity determination in FFTF fuel assemblies was conducted using the full-scale channel test section. The purpose of the experiment was to determine the accuracy of coolant velocity measurements in the subchannels using the noise analysis technique. Salt injection and turbulence within the subchannel produced the random variation in conductivity required for sensor operation.

The data from nine probe noise analyses were studied to determine a velocity correction factor, U_Q/U_T , for each probe pair. (U_Q is the velocity determined by flow rate and area and U_T is the velocity determined by time delay and probe separation). U_Q/U_T is the factor that must be applied to the "time delay" velocity to provide the true average subchannel velocity. The spread in the observed values of U_Q/U_T represents a measure of the accuracy of the velocity determinations. The results for the nine probe pairs studied are presented in the tabulation below. Each of these velocity ratios/and the reproducibility limits are based on a minimum of 6 to a maximum of 30 data points spanning a velocity range from 6 to 11 ft/sec.

<u>Probe Pair</u>	<u>Type Channel Modelled</u>	<u>Axial Probe Separation, in.</u>	<u>U_Q/U_T</u>
1A-1C	Wall	36	0.80 ± 0.02
2A-2C	Wall	36	0.79 ± 0.02
3A-3C	Central	36	0.91 ± 0.04
4A-4C	Central	36	0.72 ± 0.04
5A-5C	Central	36	0.96 ± 0.04
5A-5B	Central	12	0.93 ± 0.13
5B-5C	Central	12	1.05 ± 0.07
5A-5AA	Central	6	1.02 ± 0.13
5CC-5C	Central	6	1.06 ± 0.08

The variation of mean U_Q/U_T ratios between different probe pairs in identical channels indicates that each probe pair has unique characteristics which affect response even though the probes are essentially identical. This result implies that each probe pair will require calibration prior to use for velocity measurement in a fuel bundle. Preliminary conceptual design of an instrumented calibration channel for simulated fuel pins containing these conductivity probes is in progress.

Work is continuing in attempting to isolate the causes for the wide variation of results occurring for the smaller probe separations. Two magnetic tapes of probe output were made and processed repeatedly to obtain an estimate of errors in the electronic data reproduction system. Analysis of these tapes showed U_Q/U_T variations of $\pm 1.3\%$ and $\pm 3.9\%$, in correlation performed over identical tape segments.

F. MEASUREMENT OF GAS ENTRAINMENT THROUGH ULTRASONIC TECHNIQUES (7N91)

R. W. Riedeman and J. Spalek

Gas entrainment from the cover gas above the sodium in the reactor vessel could be a problem in sodium cooled

reactors. One of the test series planned for the BNW 1/3 scale outlet region model will be to evaluate the potential for gas entrainment under various conditions of liquid level height, flow rate and outlet nozzle elevation. In order to obtain a quantitative measure of entrainment, an ultrasonic method has been tested. In quantitative tests, known amounts of gas in water were simulated by the injection of continuous streams of small glass particles at known volume rates into stationary water. Qualitative tests were done under actual environmental conditions: real-gas (air) was injected at different rates into water flowing through a pipe at a velocity equal to 1/2 of that expected in a typical hydraulic system.

Ultrasonic tests performed at the Hydraulic Core Mockup Laboratory established: (1) the quantitative proportionality of the attenuation of ultrasonic intensity to the (void) volume of glass particles in water; and (2) the qualitative proportionality of the attenuation to the amount of real-gas (air) continuously injected into water flowing through a 1.125-in. (ID) pipe at 2 msec.

The pulse-transmission technique was used for both quantitative and qualitative measurements. Quantitative tests were made by the introduction, at known controlled-volume rates, of glass particle into a vertical pipe filled with water, where the particles settled under the influence of gravity at known speeds. The void fractions were then calculated for each particle input rate, and graphs of attenuation versus void fraction were generated. During the qualitative tests, it was judged unnecessary to try to inject air at known rates into the flowing water to generate similar quantitative data. The qualitative data positively proved the proportionality of ultrasonic attenuation to different air injection rates into flowing water while the quantitative data showed graphically the functional form of the relationship of the

ultrasonic attenuation to the void fraction. The ultrasonic transmission technique feasibility for the detection of different levels of air content in flowing water was thus experimentally established.

It was impossible with the apparatus used in these measurements, to distinguish any difference in attenuation caused by particles or real-gas bubbles of the same size. However, this limitation could be sometimes used constructively for the additional detection of foreign particles (such as corrosion products, etc.) in reactor coolant stream by the ultrasonic gas entrainment detection apparatus. Also, the transmission technique alone cannot distinguish between the attenuation of ultrasonic energy caused by many small gas bubbles or by one large bubble of equivalent cross-section. But this limitation could be removed by the combination of the pulse-echo and transmission techniques.

The application of the above technique with commercial apparatus for on-line gas entrainment measurements in large pipes would not be elementary. Interface problems with the coupling of an ultrasonic transducer through the pipes (multiple reflection problems) or directly to water (bubble buildup problems) are expected and/or were already experienced. However, the feasibility experimentation results showed that the ultrasonic attenuation technique definitely does work in the real environment of air bubbles in flowing water. Thus, with the calibration experimentation program planned to be executed at the HCM Laboratory upon delivery of commercial ultrasonic equipment, the prospects for on-line gas entrainment measurements are very good.

V. SODIUM TECHNOLOGY

A. SODIUM CHEMISTRY (7A00)

A brief summary of activities in support of FFTF by the Sodium Chemistry Subdepartment during the report period follows. These activities will be described in expanded form in a new series of Sodium Chemistry Quarterly Progress Reports, BNWL-1200-1, -2, -3, -4. The first issue will cover the period of July through September 1969; plus unreported work of May and June 1969.

1. Fission Product Behavior and Control in Sodium Systems

J. W. Bartlett and W. F. Boehm

The Radioisotopes Transport Loop (RTL) was operated for 6000 hr with a 400 °F AT (1300 to 900 °F) to determine transport characteristics of radioactive corrosion products in a sodium system. The loop has been dismantled for inspection. Results of the tests and inspection of the system are being used in several ways:

- a. To anticipate mass transport problems in FFTF. Examination of various portions of the loop has shown surface depletion of Mn, Cr, and Ni in hot leg sections and enrichment of these elements in heat exchanger and cold leg sections. Ferrite formation in the hot leg is suspected but not yet fully confirmed.
- b. Data on transport of corrosion product radioisotopes are being used as basis for investigating the possibility that the reactor-to-cell piping of the FFTF closed loops can be used as an effective trap for some of the isotopes (e.g., ^{60}Co). The data also suggest that other isotopes (^{54}Mn) might be removed in a cold trap.

- c. The radioisotope transport data were used to provide revised estimates of radiation dose rates in the FTR heat transport system. The revised values are lower and more realistic than those originally given (BNWL-808).

2. Sampling and Analysis of Radioactive Sodium

a. Remote Sampling

K. C. Knoll and E. L. Osborne

A remotely operable removable section sodium sampler designed and constructed at BNW was installed on a nonradioactive operating sodium loop and mechanically tested. It is presently undergoing quality assurance tests prior to the admission of sodium. Remote sodium sampling will be carried out during the next report period, followed by an evaluation of sample quality.

b. Remote Analysis

L. F. Lust, J. F. Jarosch, R. F. Keough and
S. E. Nichols

Design of a mockup facility to test procedures and equipment to be used in the FFTF shielded sodium remote analyses facility has been completed. Construction of the air atmosphere portion of the mockup has been completed. Cost estimates have been received on the inert atmosphere portion of the mockup. All of the major equipment items for this portion of the mockup have either been purchased or invitations are out for competitive bids. Mechanical testing of the equipment designed for the remote determination of oxygen by the amalgamation procedure has been completed in the air atmosphere mockup. Equipment for the remote determination of hydrogen has been designed and is presently under construction.

3. Selection and Testing of An In-Line Sodium Analysis System J. J. McCown

Work continued on tests to observe the response of the UNC oxygen and carbon meters to the addition of specific chemical compounds to the sodium purification loop. To date the following compounds have been exposed to 600 °F sodium using bypass addition tube techniques: Na_2O , Na_2CO_3 , NaCN, graphite rod, Na_2C_2 . Data from the meters indicated that only the Na_2O produced predicted and measurable responses on the oxygen meter. Only in the case of the Na_2O was the material very soluble. The Na_2C_2 addition produced a small change for about one hour in carbon meter output, but no increase in carbon (in the sodium) was detected by wet chemical methods. These tests to date have demonstrated primarily the difficulty of making satisfactory chemical additions to the sodium. Plans are being made to move the instruments to another loop which has higher temperature capability (1200 °F) where impurity solubilities will be higher.

4. Cover Gas Systems a. Selection and Testing of Cover Gas Instruments and Sampling Systems G. B. Barton

Laboratory evaluation of the time of flight mass spectrometer as a tool for the analyses of FFTF cover gas has been completed and a topical report is in preparation. The study concluded that the gas chromatographic approach was the best suited for this application and efforts were shifted to the study of sodium vapor removal.

A laboratory study of sodium vapor trap design and performance was completed. An examination of the theory of condensation and vapor removal based on a combination of mass transfer and nucleation thermodynamics was completed and a

vapor trap design equation formulated. Based on these two studies, a prototype vapor trap was designed and is presently undergoing quality assurance tests. During the next report period this trap will be evaluated in the laboratory to confirm that its performance agrees with that predicted by the equation. A laboratory study will also be made to define the effects of sodium vapor removal in the trap upon the validity of the analytical sample produced by the trap.

b. Cover Gas Impurity Reactions Over Sodium

H. P. Maffei

Preliminary studies to measure the inhibiting effect of oxygen on stainless steel nitriding have been completed and have resulted in the recommendation that the oxygen content of FFTF equipment cell nitrogen cover gas be maintained at 5000 ppm for cell temperatures of 1000 to 1400 °F.

Reaction rates between sodium and carbon dioxide in the argon cover gas were measured. The reaction proceeds rapidly with the formation of an inhibiting surface coating. The investigations indicate that carbon dioxide will not exist in significant concentrations in the cover gas over an agitated sodium pool such as that in the FFTF. Detection of carbon dioxide in the cover gas may indicate the presence of a thick crud layer on the sodium surface.

Preliminary specifications for methane and carbon monoxide in FFTF argon cover gas have been established. A formal report is currently being issued.

VI. CORE DESIGNA. EFFECT OF STAINLESS STEEL SWELLING ON FTR FUEL BURNUP (7N83)

R. J. Jackson

Preliminary analyses were performed to estimate the effect of irradiation-induced stainless steel swelling on fuel burnup in the FTR. The interim equations⁽¹⁾ describing the relationship of metal swelling to fast neutron fluence and operating temperature were used. Elastic deformation was assumed for the computations. The effect of irradiation-induced creep was evaluated only in a qualitative manner.

For the assumed case where swelling results in a diameter change in the duct, the mean values of the predictions indicate that fuel burnup to 80,000 MWd/tonneM will not be affected by the swelling. However, based on the 95% confidence limit of the mean, a burnup limitation at about 75,000 MWd/tonneM may be encountered.

The preliminary calculations indicate that differential swelling resulting in duct bending is the major potential fuel lifetime limiting item in the present design. The potential limitations to fuel exposure include:

- a. Duct bending caused by thermal camber if significant creep occurs in-reactor
- b. Duct bending caused by differential volume increase if no creep occurs in-reactor
- c. Core packing (occupying of all available core space by swollen ducts)
- d. Duct packing (occupying of all available duct space by swollen fuel pins).

1. J. L. Straalsund and H. R. Brager, "Damage Analysis", p. 5.16, FFTF Quarterly Technical Report, March, April, May 1969, BNWL-1090. Battelle-Northwest, Richland, Washington.

Much uncertainty exists in the extrapolation of current data on irradiation-induced swelling. More analysis of the duct bending problem is required. This must include a parametric study of the effects of creep (see Section VI-B).

1. Flow Duct Swelling

For irradiation-induced swelling of the diameters of flow ducts the effect of creep on changes in duct cross section has been found to be small compared with dimensional changes from swelling.

Calculations

The environment assumed in these calculations is tabulated below.

Distance from Bottom of Core, in.	Hot Side		Cold Side	
	Fluence, 10 ²²	Temp, °F	Fluence, 10 ²²	Temp, °F
3	12.0	624	9.19	618
7	15.6	655	11.9	642
11	18.2	687	13.9	666
15	19.6	718	15.0	690
19	19.9	751	15.2	714
23	19.0	782	14.5	739
27	17.0	814	13.0	673
31	13.9	846	10.0	786
35	10.0	875	7.6	811

Fluence values corresponding to other fuel burnup values are assumed linearly related to these values. The hot side of the duct is nearer the reactor center.

The swelling model for Type 304 solution-treated SS was used to predict the dimensional increase across the flats of the duct. Because of hot channel factor uncertainties, subsequent check calculations were based upon a maximum duct temperature of 950 °F. The swellings associated with Row 1 and with

Row 5 ducts were calculated for the elevation 27 in. above the bottom of the core. This closely corresponds to the plane of maximum swelling. The calculated dimensional change and burnup for Row 5 was based upon an average dimensional increase or fluence for the hottest side and the cooler side of the duct.

The plot of dimensional change versus fuel burnup for Row 5 almost coincides with the corresponding plot for Row 1. Therefore, Figure 6.1 shows only the dimensional change for Row 1. This indicates core packing⁽¹⁾ would not occur, based upon the mean value prediction. Core packing is predicted to occur, however, based upon the upper 95% confidence limit of the mean at about 75,000 MWd/tonneM. For the check case which considers about a 70 °F increase in maximum duct temperature, core packing is predicted at about 62,000 MWd/tonneM based upon the 95% upper confidence limit on the mean.

2. Flow Duct Bending

The flow ducts are exposed to radial fluence and temperature gradients. This causes more swelling to occur on the hotter, higher fluence side than on the opposite side. The gradients cause both thermal camber and bending from the differential swelling. The temperature and fluence environment used in the duct bending analysis was the same as that used for the initial calculations of swelling.

The Arrhenius type equation for Type 304 solution treated SS was used to calculate the volume increase for the various combinations of fluence and temperature along the length of the duct on both the hot and cold sides. The differences in swelling of opposite sides of the duct cause a difference in axial dimension of the two opposite sides. In the absence of lateral

1. *For the purposes of this analysis core packing is defined as the condition existing when flow ducts have swelled into contact with their adjacent ducts.*

FIGURE 6.1. Duct Diameter Change

restraint this creates a bending deflection. The amount of this deflection can be calculated from geometric considerations of the difference in axial extension on the two sides of the duct. However, for purposes of a computer calculation it is convenient to work in terms of a pseudo temperature gradient across the duct which would cause the same difference in axial extension as the differential swelling. It was found that the uncertainty in extrapolating irradiation-induced swelling to burnup is very large and that additional work is required before firm data can be provided. In particular, the effect of creep must be evaluated.

Creep Effect

Irradiation-enhanced creep is, of course, known and expected. It has not been measured under irradiation conditions prototypical of FTR.

If creep occurs during irradiation swelling it will affect the resultant deflection curve. Figure 6.2a shows that the thermal gradient, alone, causes the duct to bow outward during operation. If no creep occurs (and no plastic flow) the duct will return to a straight configuration when the restraints are released for refueling. Figure 6.2b indicates that if 100% stress relaxation occurs during operation the duct will bow inward at the top when the restraints are released prior to refueling. Intermediate amounts of stress relaxation will cause a free bow condition intermediate between these two cases. Figure 6.2c indicates the duct will bow inward during operation if swelling, alone, occurs. This is because the innermost duct wall axial dimension increases more than the outermost duct wall. If no creep occurs this differential length increase in the core region will cause the deflection shape shown. When the restraints are removed the duct top will bow outward. Figure 6.2d represents the deflection curve resulting from differential expansion if in-reactor creep is very pronounced

FIGURE 6.2. Comparison of Creep Effects

(100% stress relaxation). In the limiting case depicted here the inward deflection of the duct is accomplished without distributing any moment to the adjacent spans. Without the self restraint of the adjacent spans the inward deflection for this case is larger than if no creep occurs. When the restraints are removed the duct retains its same configuration because there are no residual stresses in the duct to cause the deflection curve to change.

It can be noted that creep is helpful in permitting higher burnup for a given differential swelling model in both the restrained and unrestrained cases. Conversely, creep is harmful in the thermal camber case because it causes more free bow. The substantial free bow deflection from thermal camber, if 100% stress relaxation occurred, would seriously affect the refueling operation. Because of these two opposing phenomena, either of which may be burnup limiting, it is not possible to make a conservative assumption regarding creep rate. Clearly, the design analysis must include a parametric study of effects of creep rate.

B. DEVELOPMENT OF BEAM BENDING CODE INCORPORATING EFFECT OF CREEP (7N83)

V. B. Atwood and W. H. Sutherland

Work has been initiated to develop a code which will permit the creep analysis of flow ducts using the same structural idealization as used for the elastic analysis of flow ducts. It is intended that any arbitrary creep law of the general functional form

$$\Delta E_e^C = F(\sigma_e, E_e^C, \phi, T, \Delta t)$$

may be used, where

$$\begin{aligned} \Delta E_e^C &= \text{incremental effective creep strain} \\ \sigma_e &= \text{effective stress} \end{aligned}$$

E_e^C = effective creep strain
 ϕ = flux
 T = temperature
 Δt = increment of time.

The element model will be formulated with the same degree of sophistication as the planar model in AXICRP and in fact the development will parallel that model very closely.

The geometrical generality will be restricted initially to the framework of a two-dimensional beam formulation with axial deformation plus bending in only a single plane. Although restrictive from a completely general point of view this constraint is completely in accord with the present knowledge of loading (radial variation only) and the practical observation that little duct-to-duct interaction will occur.

The code to be employed will be a modification-rewrite of AXISOL-AXICRP. SAMIS was considered in this regard but was judged much too cumbersome for efficient modification.

All of these decisions were made with the objectives in mind to (a) produce useful results as quickly as possible and (b) check the basic validity of the presently envisioned concept of treating creep behavior in beams.

Although the geometrical generality will be restricted in this initial program, consideration will be given to the full three-dimensional beam formulation and stiffness merge problems with the aim of making the extension later as easy as possible.

C. CORE RADIAL CONSTRAINT (7N81)

G. R. Waymire

Core radial constraint must provide a predictable method for controlling core motion to achieve an overall negative reactivity effect during all phases of FTR operation. A description of the requirements for core radial constraint,

and a discussion of the reasoning which led to the selection of mechanical actuators for FTR radial constraint, is given in BNWL-941.(1)

A so-called "compliant" (i.e., spring-like) load must be applied to restrain the core.

The radial restraint design⁽²⁾ initially selected (SK-3-14504, Figure 6.3) applied a compliant load, utilizing beam deflection, to the outer periphery of the core.

Consideration of radiation-induced creep in connection with the design of the restraint system makes use of beams questionable for application of the complaint restraint load. If the magnitude of the creep is significant, the load applied by the beams could be relaxed during the early stages of operation.

Figure 6.4 shows elevation and plan views of an alternative mechanism now being considered which does not utilize beam members. This configuration uses compression members.

D. REACTOR SIMULATION FOR PIPE RUPTURE TRANSIENTS (7N82)

C. D. Flowers

1. Description of Reactor Model

The reactor model for pipe rupture transient studies has been updated to represent the latest FFTF concept for the initial oxide core -- Concept V-A. The model has been simulated and debugged on both the Hybrid computer and the digital systems analyzer code, DYNASAR. The Hybrid simulation is all digital

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1. *FFTF Periodic Technical Report, BNWL-941, UC-80, October through February 1968-1969. Battelle-Northwest, Richland, Washington.*
 2. *R. C. Walker. FFTF Reference Concept Summary Description, BNWL-955, p. 5.5. Battelle-Northwest, Richland, Washington.*

SK-3-14504

FIGURE 6.3. Core Restraint System

RADIAL RESTRAINT
LOAD PADS —————

and for comparison was programmed on the PDP-7 portion of Hybrid-1. Conversion to Hybrid-2 is in process. A brief description of the reactor model follows.

The reactor model consists of 44 separate nodes where reactor temperatures are calculated. The nodal breakdown is shown in Figure 6.5. Inlet and outlet mixing plenums are simulated with four flow paths through the reactor: (a) average driver fuel tube coolant, (b) average leakage coolant outside of driver fuel tubes, (c) average radial reflector coolant, and (d) vessel wall coolant.

The following solid material regions have been simulated either as separate or multiple nodes representing average conditions within the reactor vessel: fuel, fuel cladding, lower and upper axial reflectors, driver fuel duct wall (in-core zone), inlet plenum metal, outlet plenum metal, radial reflectors (such as radial reflectors, shields, and control rods), and an area that represents the fuel storage region outside of the reactor core barrel (simulated as stagnant sodium). Heat transfer between all adjacent regions is indicated by double pointed arrows in Figure 6.5.

Heat generation is included in the following regions: fuel, axial reflectors, and radial reflectors. The total reactor power includes power generated by fast (prompt) neutrons and power generation by beta and gamma decay fission products. The neutron power is calculated by the simple point kinetics equations with the assumption that neutron lifetime is small enough (neutron lifetime $\sim 10^{-7}$ for fast reactors) to make the neutron power calculation algebraic (i.e., neglect $\lambda^* dP/dt$ term - restricts reactivity insertions to less than +1\$). Three reduced neutron decay groups are assumed for input to the neutron power calculation. Three groups are also assumed to represent the fission product decay heat generation and the following reactivity feedback terms, Doppler, sodium

FIGURE 6.5. Reactor Model Node Structure Temperatures and Flows

temperature, and structural effects, are simulated. A reactor power controller is also simulated. Other temperatures that are calculated include average fuel, average driver fuel channel coolant, hot channel core outlet, and hot channel maximum cladding.

The total reactor simulation contains 52 differential equations of which 44 calculate reactor temperatures, six calculate total reactor power and two calculate power controller response. The remainder of the simulation consists of algebraic calculations and function generators.

All sodium, fuel, cladding and driver tube wall material properties are input as a function of temperature at each node point. Reflector and plenum metal properties have been assumed constant.

2. Comparison of Simulation Methods for Transient Conditions

Steady state reactor temperatures and power were found to be identical in both methods of simulation. Therefore, the following four transients were investigated for comparison of simulations:

- HD-1: Full flow reactor scram
- HD-2: Simultaneous reactor scram and primary flow coastdown
- HD-3: Flow transient for double-ended break at reactor vessel inlet, with scram 2 sec after break, and pump coastdown 1 sec after scram. Data were taken from previous pipe rupture studies, Run No. PR-64 with 1 ft³/ft inlet downcomer standpipe.
- HD-4: Flow transient for 1 ft² hole break at reactor vessel inlet with scram 2 sec after break and pump coastdown 1 sec after scram. Data taken from previous pipe rupture studies, Run No. PR-79 with 1 ft³/ft inlet downcomer standpipe.

The results from Runs HD-1 and HD-3 are given in Figures 6.6 and 6.7. Again, the models were found to agree almost perfectly. The core outlet temperatures disagreed by approximately 5 °F during the first two sec for Run HD-3 (not noticeable in plot because of gross plotting scale). The average fuel temperatures also disagreed by approximately 5 °F during the first two sec of the same transient. These differences are considered insignificant and we have concluded that both models perform in essentially the same manner.

E. CLOSED LOOP DEVELOPMENT (7N91)

L. J. Rousseau and J. W. Kolb

1. Closed Loop Tube Weld Closure Studies

Primary objectives of this task are to demonstrate the feasibility of employing remote-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 when the surfaces are contaminated with sodium.

Photos of the sodium-contaminated weld joint are included in this report. Figure 6.8 is a top view of the simulated closed loop weld test specimen showing the extent of sodium contamination. Figure 6.9 is a view of the area where the welding arc was fired and maintained for a short period. This shows a large ball-shaped area at the end of the torch electrode caused by the intense arc heat vaporizing the sodium. Some of the vapors are trapped under the weld puddle, until finally, sufficient pressure is created which literally blows molten metal out of the puddle and onto the electrode. Figure 6.10 shows the extent of sodium vapor accumulation on the upper edge of the spool section. It is evident that weld joint inspection and cleaning methods will be required to prevent such an occurrence in final application.

FIGURE 6.6. Comparison of Results from Hybrid and Digital Models for Full Flow Reactor Scram

FIGURE 6.7. Comparison of Results from Hybrid and Digital Reactor Models for Flow Transient Due to Double-Ended Break at Reactor Vessel Inlet Nozzle

Spool Section

Weld Insert
Filler Ring

Sodium Contami-
nation

Nozzle Plug

Neg 0692819-2

FIGURE 6.8.

*Top View of Simulated Weld Test Specimen
Showing Sodium Contamination*

Neg 0692818-4

FIGURE 6.9. View of Area Where Welding Arc Was Fired

Sodium Vapor
Contamination

Metal Beads
Blown From
Weld Puddle

Neg 0692818

FIGURE 6.10.

*Extent of Sodium Vapor Accumulated in Upper Edge
of the Spool Section.*

2. Nozzle Weld Cutter (7N91)

R. F. Gilmore

The objective of this work is to develop a reliable weld removal mechanism amenable to remote operation and which will produce small chips that can be easily vacuumed away.

A Woodruff key cutter, modified for side cutting, has been shop tested on stainless steel weld samples. This circular cutter, which produces a fine chip, showed there was no visible evidence of wear after one hour and forty-five minutes of operation.

Fabrication on the four-tooth trepanning test module assembly, based on the Do All Company cutter design, has been completed. Preparations are under way for testing this assembly on samples welded by remote application. This cutting technique employs a pulsed feed to eliminate the possibility of long machined stringers clogging the vacuum exhaust passages. Also nearing completion is the side mill cutter test module. This assembly can accommodate four cutters and will be equipped with a vacuum system to remove the chips from the joint area.

3. Comparison of Circular and Hexagonal Tubes for Closed Loops (7N92)

M. K. Mahaffey

In order to determine the upper bounds of closed loop system operation for design purposes, the power capability inherent in the circular closed loop tube design proposed for Concept V-A was investigated and the results were compared to an alternative hexagonal concept. In both configurations, the flow channel and pressure tube are of a double-walled construction. Figure 6.11 shows a schematic of the tube designs characterized in this study.

FIGURE 6.11. Tube Configurations

It was found that no apparent increase in test diameter can be achieved with a hexagonal tube and hexagonal test section compared to a concentric circular tube configuration (see Figure 6.12). This is explained by the greater wall thickness required for hexagonal tubes as compared to circular tubes at the same pressure loading.

Optimizing downcomer dimensions against test diameter to provide 30 ft/sec in the downcomer and maximum flow results in a maximum power capability for a 19-pin bundle of about 5.7 MW at 400 °F AT. (Tests operating above about 2 MW will require ^{235}U enrichment to achieve the necessary kW/ft.) Figure 6.13 shows capabilities for 19-pin, 37-pin, and 61-pin bundles. All tests are operating at 400 °F AT. Pressure drop calculations for the tube and system show that all tests can be performed within the head capability of the present primary pump.

FIGURE 6.12. *Closed Loop Tube Geometry*

FIGURE 6.13. *Tube Capability Concept V-A for Various Pin Bundles at 400 °F AT*

F. HEAT TRANSFER AND FLOW TESTS ON FUEL SUBASSEMBLIES

1. Heat Transfer Characteristics (7N91)

R. E. Collingham and V. R. Hill

The purpose of the task in FY 1970 is to perform sodium heat transfer testing with multi-pin models of FFTF fuel assemblies to determine local cladding temperature profiles in normal and structurally distorted FFTF fuel assemblies. A preliminary 7-pin bundle test, to evaluate the temperature difference between edge and central coolant channels and the potential hot spot under a wire wrap spacer, was completed.

While several thermocouples (mainly cladding thermocouples) failed to operate during the test, sufficient information was obtained to provide the following conclusions:

- The fuel thermal analysis model, COBRA, predicted the average coolant temperature differences over the Pellet Number range of the test with a mixing coefficient of 0.0068. Further water mixing tests are required to verify this value of the mixing coefficient.
- Temperature differences existed between the outer sub-channels themselves and the inner channels themselves, at a single axial position. This is postulated to be caused by forced diversion of coolant, and possibly by tolerance buildup.
- Initial results from limited data indicate that excellent cooling occurred at the region under the wire wrap spacer and that the potential cladding hot spot directly under the spacer may not exist. Further testing will be required to verify this postulation.

2. CCTL Mark I Flow Tests (7N91, 7N75)

M. K. Millhollen

A full-scale model (CCTL Mark I) of the dispersed core reference fuel assembly was exposed for 3600 hours to flowing sodium at 1060 °F and a velocity of 27 ft/sec in the core component test facility. The examination of the assembly has disclosed no significant problems to date and is being reported in detail in a topical report.⁽¹⁾ The test assembly has been reassembled to permit compressibility tests to be performed on it. The results of these tests will provide supporting information to predict the dispersion of the fuel pins during reactor operation.

1. M. K. Millhollen. Sodium Flow Test, BNWL-1075, Battelle-Northwest, Richland, Washington. September 1, 1969.

G. TESTS ON REMOVABLE DRIVER FUEL INSTRUMENT ASSEMBLY (7N91)**R. F. Gilmore**

The purpose of this program is to determine, by testing, the feasibility of remotely removing and re-installing the removable instrument assembly through a curved guide tube during reactor operation.

A mockup simulation of the Westinghouse Driver Fuel Instrument Assembly and curved guide tube has been fabricated (see Figure 6.14), and is currently undergoing testing to determine feasibility of the design concept. The mockup consists of the lower section of the instrument assembly (including a dummy flow meter), five thermocouples, and five ball-and-socket joint sections containing eight electrical leads. The guide tube was made up from 2 in. Schedule-40 pipe with one reverse bend extending into a 90 degree bent section. The bend radius for both sections is 36 in. All the fabricated parts were made from Type 304 SS. Testing is being conducted in air at ambient conditions. The force necessary to pass the partial instrument assembly through the critical guide tube bends is measured by means of a fish scale attached by cable to the test specimen.

Preliminary testing has shown that the insertion and removal loads on the instrument assembly are light, and that the design appears to be feasible. Further evaluation of test data will be necessary to determine if an installation device will be needed.

H. THERMOCOUPLE RESPONSE TIME REQUIREMENTS FOR DRIVER CHANNEL (7N82)**J. J. Regimbal**

A study was performed to define the sensor requirements for protection against flow loss occurrences. Results indicate that satisfactory protection against boiling is obtained using

Neg 0693375-2

FIGURE 6.14. *Simulated Driver Fuel Instrument Assembly*

a thermocouple with a 2.0 sec response time if flow loss is predicted to occur less rapidly than at a rate of 20%/sec. Very little advantage is seen to be afforded by using a thermocouple with a response time improvement to 1 sec. If postulated flow reductions to final levels between 20 and 35% are identified as significantly likely, the flowmeter reaction speed is a valuable asset for rapid blockages. Figure 6.15 provides preliminary results for an analysis of hypothetical coolant channel flow blockage occurrences. The effort was aimed at identifying the performance limitations of thermocouples and flowmeters when signals from these sensors are used to initiate a protection system response to anomalies in flow. Each of the curves represents the minimum conditions for which boiling will begin at the hottest channel position despite sensor-initiated reactor scram.

The values of thermocouple time constant (denoted by τ) are to be understood in the conventional sense. The flowmeters were assumed to have an instantaneous reaction to flow change. Thermocouple trips were set at 30 °F above the steady-state junction temperatures, and the flowmeters were assigned trips at levels of 80 and 60% of full flow. Scram rod motion was assumed to begin 100 msec after trip conditions were reached in all the sensors.

This work provides a partial basis for establishing how signals from in-channel thermocouples and flowmeters should be used. Firm assignment of probabilities that such flow anomalies may occur is not able to be factored into such a choice, at least at this time, and the final disposition of these sensors will undoubtedly require an extensive exercise of engineering judgment.

FIGURE 6.15. Conditions for Boiling of Coolant at Core Exit

VII. FUELS AND MATERIALS

A. FUELS AND MATERIALS EVALUATION

1. Biaxial Stress-Rupture Properties of Type 316 SS (7N72)

R. W. Barker

The objective of this program is to determine the effects of fast reactor irradiation on the biaxial stress-to-rupture properties of candidate fast reactor cladding materials. Biaxial stress-rupture tests have been performed on AISI Type 316 SS irradiated to a nominal fluence of $\sim 1 \times 10^{22}$ n/cm² (E > 0.1 MeV) in subassembly X-022 in EBR-II. Most of the irradiated samples tested were irradiated to a fluence of 0.8×10^{22} n/cm². Test temperatures were 1000, 1200, and 1400 °F. The remainder of the samples were irradiated to higher fluence (1×10^{22} n/cm²) and will be tested at temperatures of 900, 1000, and 1200 °F.

The data obtained from biaxial stress-rupture tests on unirradiated as-received and irradiated AISI Type 316 SS cladding are summarized in Tables 7.1 and 7.11. Plots of stress versus rupture time for the various tests are shown in Figure 7.1 and plots of strain versus rupture time are shown in Figure 7.2. The data points show stress or strain and rupture time for as-irradiated tests while the solid lines are control test results. Some irradiation hardening (strengthening) is evident at 1000 °F for the short rupture times shown. The as-irradiated test data are in fair agreement with the as-received test data to about 10 hr rupture life. At 1200 °F the data fall in a scatter band around the curve formed by the as-received data. At 1400 °F, marked decreases in rupture life are observed with the reduction being a factor of 7 to 10 less than the unirradiated values.

TABLE 7.1. *Biaxial Stress-Rupture Tests on AISI Type 316 SS in the As-Received Condition*

Test Temperature, °F	Hoop Stress, psi	Rupture Time, hr	Rupture Strain, %
1000	52,000	0.1 (Burst)	
	51,500	6	16.8
	48,000	55	14.0
	50,000	78	21.0
	42,000	545	10.5
1200	39,400	0.5	16.5
	31,200	6	6.2
	29,000	15	8.6
	25,500	70	4.8
	19,900	376	1.9
	23,000	880	23.5
	17,000	1236	14.4
1400	25,000	0.4	24.0
	19,800	5.5	19.2
	19,800	6.8	19.2
	17,000	16.5	13.9
	15,200	42	21.5
	9,600	300	9.1

The failure behavior of the irradiated specimens is significantly different from that of the controls. An illustration of typical bursts of irradiated and unirradiated specimens is shown in Figures 7.3 and 7.4. Both specimens were tested at 1400 °F in an argon atmosphere and had rupture times between 5 and 10 hours. It can be observed that the as-received specimen tears apart transversely, a section approximately the same length as the initial longitudinal split (see Figure 7.3), leaving the remainder of the specimen undamaged from which the more precise diameter measurements can be made. The as-irradiated specimens do not show a tendency to tear in the transverse direction. Rather, the longitudinal split becomes more pronounced, often propagating to the entire length of the specimen.

TABLE 7.II. *Biaxial Stress-Rupture Tests on AISI Type 326 SS Irradiated to $\sim 0.8 \times 10^{22}$ n/cm² (E > 0.1 MeV) in EBR-II. The Irradiation Temperature Was 825 °F ± 20 .*

Unirradiated Data ^(a)				As-Irradiated Data			
Test Temp, °F	Hoop Stress, psi	Rupture Time, hr	Rupture Strain, %	Irrad. Test No.	Hoop Stress, psi	Rupture Time, hr	Rupture Strain, %
1000	52,000	Burst		19	58,000	Burst	NRM ^(b)
				26	54,000	2.2	NRM ^(b)
	50,000	~ 10	18	27	50,000	9.6	3.8
1200	36,000	1.0	10	17 ^(c)	36,000	0.6	5.8
	36,000	1.0	10	12	36,000	2.1	1.9
	30,000	10.0	7	13	30,000	32.0	2.4
	25,000	100	5	20	25,000	20.0	0.5
	23,500	150	4.6	24	23,500	70.0	1.9
1400	23,500	1.0	24	18	23,500	0.3	NRM ^(b)
	17,700	10.0	17	21	23,500	1.4	NRM ^(b)
	12,000	100.0	12	22	23,500	8.2	1.4
	7,800	1000	7.5	23	7,800	330	5.3

a. Stress and strain values taken from unirradiated curves at rupture times shown.

b. Not readily measurable.

c. Exposure for this specimen is 1×10^{22} n/cm² (E > 0.1 MeV).

Neg 0693125-1

FIGURE 7.1. The Effect of Biaxial Stress and Temperature on the Rupture Life of AISI Type 316 SS Tested in Inert Gas.

Neg 0693125-2
FIGURE 7.2. The Effect of Rupture Time and Temperature on the Ductility of AISI Type 316 SS Tested in Inert Gas

FIGURE 7.3. *A Typical Burst Failure for AISI Type 316 SS in the As-Received Condition. Test conditions were 1400 °F, 19,800 psi stress, rupture time 6.8 hours.*

FIGURE 7.4. *A Typical Burst Failure for AISI Type 316 SS Irradiated to $\sim 1 \times 10^{22}$ n/cm² ($E > 0.1$ MeV). Test conditions were 1400 °F, 12,000 psi stress, rupture time 8.2 hours.*

Because of the tendency of irradiated specimens to fail in a burst mode, opening more fully from end to end, a good evaluation of rupture strain usually cannot be made. However, where data are available, irradiation is seen to reduce ductility significantly (see Figure 7.2). The lowest ductility in as-received material is at 1200 °F. This also appears to be the case for the irradiated material. Rupture strain at 1000 and 1400 °F test temperatures in the unirradiated and irradiated condition appear: to be comparable.

The length/diameter ratio for the irradiated specimens mounted with mechanical fittings varies from 4 to about 6, depending on the length of cladding after removal of the fittings and on effects of remote assembly such as poor placement of the sleeve (as seen in Figure 7.4). The ratio for the particular specimen shown in Figure 7.4 was 4. Work by Venard⁽¹⁾ has shown that a ratio as low as 3.3 is sufficient to provide valid stress-rupture data.

Time-to-rupture data from AISI Type 316 SS, both as-received and as-irradiated, tested in the Biaxial Stress Test Facility, have been replotted using the "Theta" parameter suggested by other investigators.^(1,2) The master plot, shown in Figure 7.5, shows that biaxial stress-to-rupture data at various test temperatures can be related with the Theta parameter:

$$\theta = \log t r \exp [-Q/RT]$$

which is plotted against $\log \sigma$.

-
- 1, J. T. Venard. Stress-Rupture Properties of Type 304 Stainless Steel Tubing, ORNL-JM 535, June 1963.
 2. R. L. Orr, O. D. Sherby, and J. E. Dorn. Am. Soc. Metals, Trans. Quart. 46, 113-28. 1954.

$\Theta = \text{LOG } t_r \text{ EXP } [-Q/RT]$

Neg 0693098

FIGURE 7.5. Theta Parameter Master Plot of Irradiated and As-Received AISI Type 316 SS Cladding Tested in Inert Atmosphere. [Exposure is $\sim 1 \times 10^{22}$ n/cm² ($E > 0.1$ MeV) at 815 °F in EBR-II.]

where

σ = applied stress (psi),
 t_r = time to rupture (hr),
 Q = activation energy for creep (cal/mole),
 T = absolute temperature ($^{\circ}\text{K}$),
 R = gas constant (cal/mole-degree).

It can be observed that by a value of $Q = 95,000$ cal/mole allows representation of all as-received data, shown in tabular form in Table 7.1 by a single line. Excellent correlation is obtained between tests at various test temperatures.

As-irradiated stress-rupture data from Table 7.11 is also plotted on the Theta master plot. An excellent comparison can again be made between the various test temperatures showing the validity of the parameter.

At low Theta values irradiation hardening is observed which results in an increase in stress to rupture. The increase in rupture stress is consistent with tensile results on AISI 316 SS from the same irradiation experiment. At high Theta values the stress rupture life is decreased. This is probably caused by helium embrittlement dominating irradiation hardening at the elevated temperature.⁽¹⁾ The fractures at high temperatures appear to be intergranular as expected from helium embrittlement processes. At intermediate Theta values an even balance between irradiation hardening and helium embrittlement appears to exist.

During the coming months, attempts will be made to establish further the validity of using the Theta parameter for extrapolations by evaluating correlations of microstructural features by relating different tests via the Theta plot.

1. R. S. Barnes. "Embrittlement of Stainless Steels and Nickel Base Alloys at High Temperature Induced by Neutron Irradiation", *Nature*, vol. 206, pp. 1307-1310, June 26, 1965.

2. Carbide Precipitation

A stabilizing treatment presented in Table 7.111, designed to precipitate carbides on intragrain slip planes rather than on grain boundaries, was given to several specimens of Type 316 SS sheet. The as-treated microstructure, Figure 7.6, shows carbide occurring in about an equal amount on both slip planes and grain boundaries. Examination of TIG welds in treated material discloses increased re-solution of carbides with increasing proximity to the weld bead as illustrated in Figures 7.7, 7.8, 7.9, and 7.10. The usual sensitized structure, consisting of a heavy carbide network in the grain boundaries, did not develop in the heat-affected zone. Further work will determine the resistance of the stabilized structure and the heat affected zone to intergranular attack in hot HNO_3 .

TABLE 7.111. *Heat Treatment for Stabilizing
Type 316 SS*

Solution Anneal, 2000 °F, 15 minutes

Cold Work, 25% Reduction of Area

Precipitation Treatment - 900 °F, 24 hr
 1300 °F, 216 hr

3. Irradiation of Hypostoichiometric UO_2 - PuO_2 Fuel (7N75)

The status of the BNW-1 series of capsules is summarized in Table 7.IV. Two capsules continued their irradiation toward a goal of 100,000 MWd/tonneM, and four were examined in the BNW Radiometallurgical facilities.

a. Compilation of Fabrication and Irradiation Histories

D. W. Dragnich and R. D. Leggett

A complete fabrication history is being assembled for each of the BNW-1 experiments. This is to include analyses of the fuel from the powder stages through to the fuel pin,

469-713

500X

FIGURE 7.6. *Stabilized Grain Structure in Type 316 SS*

469-714 F

500X

FIGURE 7.7. *Stabilized Type 316 SS 0.085 in. Sheet Followed by TIG Welding with Addition of Weld Filler. (Note the presence of prior austenitic grain boundaries.)*

469-714 E

500X

FIGURE 7.8. *Stabilized Type 316 SS in the Heat Affected Zone Approximately 0.050 in. from the TIG Weld Bead. (Again note the prior structure.)*

469-714 D

500X

FIGURE 7.9. *The Same Conditions as Figure 7.8 Except Approximately 0.225 in. from Weld Bead*

469-714 C

500X

FIGURE 7.10. *At a Distance of Approximately 0.190 in. from the Weld Bead There Is No Effect of TIG Welding on the Structure of the Stabilized Type 326 SS.*

the processes involved, certification of capsule components, complete assembly details and all pre-irradiation measurements.

Irradiation histories are also being compiled for each experiment. Plots of temperature versus time are being studied to correlate the thermocouple readings to temperatures obtained from the Tiger V computer program analyses.

b. Gamma Scanning on Irradiated Pins

J. W. Weber and D. E. Christensen

Axial gamma scans for gross activity and for $^{95}\text{Zr}/^{95}\text{Nb}$ have shown several activity peaks on BNW-1-1, 1-2, 1-3 and 1-5. Spectra data collected along the length of the fuel pins including several directly on the activity peaks are being analyzed with a computer program GSSLRN (Guassian-Learn) to determine the major gamma emitting isotopes present. The

TABLE 7.IV. Summary Status of Irradiation Experiments Conducted Under Task FP-6

Test Designation	Description Fuel Form	(a) O/M	Facility	Goal Exposure MWd/tonneM	Actual Exposure MWd/tonneM	Discharge Date	Status 8/8/69
BNW 1-1	Solid	1.93	MTR	100,000	(Est)120,000	2-69	Being Exam. at BNW
BNW 1-2	Solid	1.93	ETR	50,000	(Est) 70,000	2-69	Being Exam. at RNW
BNW 1-3	Solid	1.96	MTR	100,000	(Est)150,000	2-69	Being Exam. at BNW
BNW 1-4	Solid	1.96	MTR	50,000	71,000	6-68	Exam. completed at BNW
BNW 1-5	Solid	2.00	MTR	100,000	(Est)160,000	2-69	Being Exam. at BNW
BNW 1-6	Solid	2.00	MTR	50,000	85,200	10-68	Exam. completed at BNW
BNW 1-7	Annular	1.93	MTR	50,000	<1,000		Lead tube failed. Exam. at BNW
BNW 1-8	Annular	1.93	MTR	100,000	29,000	1-68	Failed. Exam. at BNW
BNW 1-9	Annular	1.96	MTR	50,000	81,800	6-68	Exam. completed at LASL
BNW 1-10	Annular	1.96	ETR	100,000	(Est) 65,000	(Est)2-70	Discharged 2-69 for interim Exam. Recharged 5-69
BNW 1-11	Annular	2.00	MTR	50,000	87,400	8-68	Exam. completed at LASL
BNW 1-12	Annular	2.00	ETR	100,000	(Est) 65,000	(Est)2-70	Discharged 2-69 for interim Exam. Recharged 5-69

a. Fuel

25% PuO₂ - 75% UO₂
 Pellets 0.212 in. OD (0.052 in. ID
 for annular pellets)
 Smeared density - 87.5% TD
 Pellet density - solid pellet 93% TD,
 annular pellet 95% TD
 Fuel column length 13.5 inch

Cladding

OD - 0.250 in.
 Wall thickness - 0.016 in.
 Type 304 SS

activity peaks on the gross gamma scans corresponds to axial positions on the fuel pin neutron radiographs where high density material, presumably fission product ingots, can be seen in the central voids of the fuel. Preliminary analysis from the BNW-1-1 spectra data on the computer program indicates that the fission product ruthenium is the major contributor to the activity peaks along the fuel except for one approximately one mil from the top of the fuel column. The spectra analysis indicates that cesium is a major activity contributor of this position. There is an indication that Cs has migrated more generally to the top of the fuel pin and even into the plenum region. Analysis of the gamma scan data and spectra is continuing.

4. Irradiation Effects on EBR-II Driver Fuel Pins (7N75)

C. R. Hann, R. D. Leggett, and W. J. Gruber

Certain EBR-II driver fuel (U + 5Fs) pins became shortened during irradiation. BNW is assisting ANL in the postirradiation examinations to determine the cause of the anomalous behavior. Four irradiated pins, one nonirradiated pin, and three nonirradiated elements have been examined thus far. One nonirradiated element was produced by ANL at the Fuel Cycle Facility and the rest of the fuel was produced by a commercial vendor.

All of the data obtained to date indicate that the observed shortening of centrifugally bonded EBR-II driver fuel pins is a manifestation of irradiation growth resulting from a preferred orientation of alpha-uranium introduced during fabrication of the fuel elements.

Table 7.V provides a summary of fuel type, casting batches, serial numbers, element numbers, and BNW identification numbers. Table 7.VI summarizes the length measurements for all the pins examined to date. Each pin was sectioned

TABLE 7.V. *EBR-II Driver Fuel-Pin Identification*

<u>Unirradiated</u>					
<u>Type</u>	<u>Casting Batch</u>	<u>Serial Number</u>	<u>Element Number</u>	<u>BNW Number</u>	
AGC	100	01625		A	
AGC	505	24267		B	
AGC	112	01924		C	
FCF	136IH		65	D	
<u>Irradiated</u>					
<u>Type</u>	<u>Casting Batch</u>	<u>Serial Number</u>	<u>Element Number</u>	<u>Subassembly Number</u>	<u>BNW Number</u>
AGC	100	2244	E-21	C-2178	1
AGC	112	1969	E-25	C-2178	2
AGC	119	2147	E-40	C-2178	3
AGC	107	1908	E-88	C-2178	4

TABLE 7.VI. *Lengths of EBR-II Fuel Pins*

<u>Unirradiated</u>				
<u>Type</u>	<u>Casting Batch</u>	<u>ANL Pin Number</u>	<u>BNW Pin Number</u>	<u>Length (In.)</u>
AGC	100	01625	A	13.430
AGC	505	24267	B	13.428
AGC	112	01924	C	13 5/8
FCF	136IH	65	D	13.462
<u>Irradiated</u>				
AGC	100	E-21	1	13.479
AGC	112	E-25	2	12.688
AGC	119	E-40	3	12.509
AGC	107	E-88	4	12.800

into approximately one-inch segments and the length and weight of each segment measured to obtain a M/L (grams/inch) ratio. The density of each segment was determined by immersion density techniques, and swelling values were calculated using the average density of BNW No. C as the preirradiation density. The swelling, mass per unit length, and the diameter change for the segments from irradiated pin PNL No. 4 are plotted in Figure 7.11 as a function of position in the pin. Similar curves for BNW No. 1, 2 and 3 are attached from BNWL-1190-FF (see Figures 7.12, 7.13 and 7.14).

A value of 3%/at.% burnup was used to calculate the theoretical minimum swelling curve. Burnups were estimated from axial gamma scans. Diameter increases calculated from density measurements were subtracted from the measured diameter changes to obtain an estimate of diameter changes caused by irradiation growth. The growth curve is shown on the far right of Figure 7.11. Comparison of this curve with the M/L curve or the measured diameter curve shows that most of the diameter change is induced by irradiation growth. Both the M/L curve and the "growth" curve indicate that irradiation growth is the cause of the length decrease.

5. FFTF Fuels Irradiation Testing

a. Subassembly PNL-2

L. A. Pember

The PNL-2 (X032) subassembly has reached an estimated burnup of 43,000 MWd/tonneM at the end of Run 36 (July 31, 1969). The EBR-II is scheduled to go to 62.5 MW for the first half of Run 38 to check reactor constants and performance. During this period, all experimental subassemblies will be removed from the core. In view of this, it has been requested that PNL-2 be discharged at the end of Run 37. The exposure at the end of Run 37 is estimated to be 47,500 MWd/tonneM.

FIGURE 7.11. Postirradiation Data for EBR-II Driver Fuel Sub-assembly C-2178, Pin No. E-88, BNW Pin No. 4, Batch No. 107.

FIGURE 7.12. Postirradiation Data on EBR-11 Driver Fuel Subassembly C-2178, Pin No. E-21--BNW Pin No. 1, Batch No. 100.

FIGURE 7.13. *Postirradiation Data on EBR-II Driver Fuel Sub-assembly C-2278, Pin No. E-25--BNW Pin No. 2, Batch No. 112.*

FIGURE 7.14. Postirradiation Data on EBR-11 Driver Fuel Sub-assembly C-2178, Pin No. E-40--BNW Pin No. 3, Batch No. 119.

b. Subassemblies PNL-9, 10 and 11

L. A. Pember and E. O. Ballard

Approval-In-Principle for EBR-II irradiation of subassemblies PNL-9, 10 and 11 was received after a change to a smeared density of $85 \pm 2.5\%$ TD and dished-end pellets was made. The dished-end pellet design to be used for these subassemblies is shown in Figure 7.15.

FIGURE 7.25. Dished-end Pellet Design
for Subassemblies BNW-9, 10 and 11.

c. EBR-II Run-To-Failure Tests

M. K. Millhollen and L. A. Pember

Three 40-in. long 37-pin subassemblies of 0.230 in. diameter fuel pins spaced by 0.056 in. diameter wire wrap are being designed for irradiation as Mark E subassemblies in EBR-II. The three subassemblies will contain UO_2 - PuO_2 dished-end pellet fuel of 85% smeared density. The first subassembly will be one with reduced plenum volume to encourage relatively early failure resulting from buildup of internal

pressure. The second subassembly will be a standard subassembly which will be run to determine at what point failure occurs. The third subassembly is planned to be one in which naturally occurring and man-made defects are included in the fuel pin cladding.

d. Subassemblies PNL-3, 4 and 5

L. A. Pember and E. O. Ballard

Three 37 unencapsulated pin experimental subassemblies in EBR-II are operating well. These subassemblies had attained the following burnup at the end of EBR-II Run 36 (July 31, 1969).

<u>Subassembly</u>	<u>Burnup, MWd/tonneM</u>
PNL-3 (X051)	9,500
PNL-4 (X059)	11,000
PNL-5 (X054)	19,500

6. Computer Simulation of Fuel Pin Performance

R. D. Leggett, R. L. Fish and C. R. Cole

. Equations describing fission gas release, pressure buildup and three-zone sintering have been formulated and have been programmed onto the Hybrid Computer. Fuel melting and thermal flux radial power profiles are now being incorporated into the program.

The expected fuel restructuring and fission gas release were calculated for several pins of the PNL-1 series (UO_2 - 25 wt% PuO_2 , Type 304 SS cladding, 0.250 in. OD $\times$ 0.016 in. wall, irradiated in EBR-II) using the two-zone "SINTER" code, assuming an in-reactor sintering temperature of 1700 °C, sintered fuel density of 98% of theoretical, and a gap coefficient of 1200 Btu ft⁻² °F⁻¹ hr⁻¹. Table 7.VII compares the calculated values with the observed values. Except for PNL-1-17, there is reasonable agreement between calculated and observed axial hole diameters and the percent of fission gas released. The agreement for the diameter of columnar

grain growth is poor. These observations indicate that PNL-1-17 operated colder than the calculation predicted.

TABLE 7.VII. *Comparison of "Sinter" Calculations with Post-irradiation Observations of Mixed Oxide Fuel Pins*

	Pin No.			
	PNL-1-3	PNL-1-6	PNL-1-17	PNL-1-19 ^(a)
s, % TD	87.6	84.0	85.6	83.8
Power, kW/ft	8.8	8.1	9.2	9.5
Burnup, MWd/tonneM	8140	7850	8790	9200
Hole Diam, in. $\times 10^3$				
Obs	13	18	13	33
Calc	16	20	30	35
Col. Gr. Diam, in. $\times 10^3$				
Obs	80	80	75	125
Calc	59	54	94	112
Fiss. Gas Rel., %				
Obs	12	11	10	25
Calc	10	9	15	21

a. A gap coefficient of $1000 \text{ Btu ft}^{-2} \text{ } ^\circ\text{F}^{-1} \text{ hr}^{-1}$ was used because of the Zarger fuel-cladding gap in this pin.

7. Tensile Strength of Stainless Steels (7N96)

A. L. Ward

The objective of this program is to provide mechanical property and metallurgical stability information on LMFBF vessel and core structural materials in direct support of the FFTF in order to define adequate lifetime performance under anticipated FTR service conditions.

Tensile test results reported for a series of AISI Type 304 SS specimens irradiated in the EBR-II subassembly X022

suggest a strong sensitivity of tensile properties to irradiation temperature. Some of the data show that much greater irradiation hardening occurred in specimens irradiated to $\sim 6 \times 10^{21}$ n/cm² (total) at 770 °F than in those irradiated to $\sim 4 \times 10^{21}$ n/cm² at 725 °F. The specimens which showed hardening represent a slightly different reactor radial position. Since the present information on the effect of specific pin location on fluence and temperature does not explain the discrepancy, and in view of the FTR vessel design schedule requirements a concentrated effort is in progress to resolve the inconsistencies by careful study of the variation in neutron flux and gamma heating with changes in radial pin locations.

In general, the results reported here are consistent with those given previously⁽¹⁾ for AISI Type 316 SS irradiated under nearly identical conditions. The strong irradiation temperature dependence of the tensile properties is also consistent with results reported for samples of the EBR-II control rod thimble.

Four Mk-B-7 type pins from subassembly X067 have been discharged from the EBR-II following irradiation to a peak fluence of $\sim 1 \times 10^{22}$ n/cm² (total); irradiation temperatures of 700, 800, and 1100 °F are represented. These pins contain a total of 204 tensile and creep-rupture specimens and 20 fatigue specimens. The bulk of the tensile and creep-rupture specimens (159) constitutes the irradiation experiment supporting the FTR vessel weldment study. Except for the austenitic filler materials (308, 308L, and 168-2) in the weldment specimens and 24 annealed, 10% and 20% cold-worked AISI Type 316 SS,

1. FFTF Quarterly Technical Progress Report November 1968 through February 1969, BNWL-1020. Battelle-Northwest, Richland, Washington.

all of the specimens are of AISI Type 304 SS. The testing schedule for weldments calls for tests at 600, 700, 800, 900, and 1100 °F temperatures.

Preliminary results from aged AISI Types 304 and 316 SS indicated that long term (3045 hr) exposure to temperatures in the range 800 to 1400 °F had little effect on the tensile properties of these materials. Aging at 800 and 900 °F had virtually no effect on the yield strength and ductility. Aging at 1400 °F produced generally lower yield strengths and ductilities comparable to as-fabricated values which tend to be retained at higher temperatures. Optical metallography of both alloys indicates that grain boundary carbides appear first in those specimens aged at 1000 °F. At lower aging temperatures there are very few, if any, microstructural features to distinguish the aged material from the nonaged solution treated material. At higher temperatures, the grains tend to lose their sharply angular shape and the boundaries become heavily populated with carbides. At 1400 °F, massive carbides appear in both the boundaries and the matrix.

8. Effect of High Strain Rates on Mechanical Properties of SS (7N96)

J. M. Steichen

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

During the past quarter, several high strain rate tests were completed on the Krafft-Hahn dynamic loader. These tests

were performed on Type 304 SS at strain rates of 0.01, 0.1, 1.0, and 10.0 sec^{-1} in air at test temperatures from room temperature to 1600 °F. The data from each test were recorded on an oscillograph picture, and the information taken from these pictures was processed with existing computer programs. The preliminary results of these tests indicate that the ultimate strength of Type 304 SS is the mechanical property which is most sensitive to strain rate at temperatures above 1000 °F. Below 1000 °F, the ultimate strength appears to be similar at strain rates from 0.00003 sec^{-1} to 1.0 sec^{-1} , but a maximum value is obtained at 0.1 sec^{-1} . At rates above 0.1 sec^{-1} , the ultimate strength falls significantly at test temperatures of 800 °F and above.

The yield strength of Type 304 SS at room temperature appears to increase slightly with increasing strain rate, but at 800 °F and above, maximum yield strengths are obtained at a strain rate of 0.1 sec^{-1} . At higher rates, the yield strength appears to decrease slightly. As the test temperature increases to 1600 °F, the yield strength becomes less sensitive to strain rate.

The total and uniform elongations appear to be fairly insensitive to strain rate at rates of 0.1 and 1.0 sec^{-1} at all test temperatures. The limited amount of data at 0.01 and 10.0 sec^{-1} appears to be in agreement with this, but additional testing will have to be completed before comparisons can be made. In comparing ductility obtained at a rate of 0.00003 sec^{-1} to that of 0.1 and 1.0 sec^{-1} , the ductility decreases at all test temperatures up to 1100 °F as strain rate increases. Above 1100 °F the ductility appears to increase with increasing strain rate.

Since the energy required to cause plastic flow and subsequent failure in the FIR vessel under conditions of rapid loading is of interest, an estimate of the effect of strain

rate on this energy can be obtained by determining the area under a stress-strain curve. Since the stress-strain curves were available at four strain rates (0.00003, 0.1, 1.0, and 10.0 sec^{-1}) at a test temperature of 1000 °F, a rough approximation of the area under each curve was made by multiplying the ultimate strength times the total elongation. The numbers obtained are as follows:

<u>Strain Rate (sec^{-1})</u>	<u>Area (Approximate) ($\times 10^5$ in./lb)</u>
0.00003	2100
0.1	2000
1.0	1705
10.0	1370

These values indicate that the energy required to cause deformation and failure in the vessel at a strain rate of 10.0 sec^{-1} would be about 65% of the energy required at a rate of 0.00003 sec^{-1} .

9. Radiation Effects on Boron Carbide Capsules (7N96)

A. L. Pitner

Additional postirradiation analyses were performed on capsules BC-1 and BC-4. Each of the capsules contained vibrationally-compacted boron-carbide powder ($\sim 80\%$ TD) and was irradiated in EBR-II at 1125 °F. The 6-in. sample in BC-1 was clad in 1/2-in. OD $\times$ 0.065 in. wall Type 304 SS, while the sample in BC-4 was contained in 1/2-in. OD $\times$ 0.035-in. wall Type 304 SS tubing.

Profilometer measurements were made on the cladding of each capsule at three indexed orientations 120 degrees apart. Comparison of the results with similar measurements made before the irradiation showed no detectable changes in the cladding profiles.

Burnup of ^{10}B in the B_4C samples was determined by mass spectrometry. The average burnup and standard deviation of ten samplings was 1.58 ± 0.22 atomic % ^{10}B . Of the calculated amount of helium formed at this value of ^{10}B burnup, the amount of gas released from the B_4C during irradiation is found to be 27.8% for BC-1 and 29.7% for BC-4.

During disassembly and cut-up of the B_4C capsules, the visible appearance of the samples suggested that sintering had occurred. However, particles that were conglomerated broke apart easily when disturbed. The results of a sieve analysis performed on the irradiated material and unirradiated boron-carbide of the same mixture are as follows:

<u>Mesh</u>	<u>wt%, Unirradiated</u>	<u>wt%, Irradiated</u>
10	3.5	2.4
20	43	25.3
28	0.4	4.4
60	10.7	15.2
200	0.6	9.0
325	17.9	10.9
-325	<u>23.9</u>	<u>32.8</u>
	100.0	100.0

The unirradiated results are the average of three tests, and the irradiated values are the average of 10 samplings. Rather than sintering, it appears that the larger-size particles broke up into finer particles during irradiation. Particles in the range of 10 to 20 mesh accounted for 43% of the total materials in the unirradiated sampling, but only 25% in the irradiated specimens. The increased quantity of the finer mesh sizes for the irradiated material can be seen in the foregoing tabulation. Whereas only 24% of the unirradiated powder was smaller than 325 mesh, the irradiated samples were composed of nearly 33% of this finer material.

The irradiation testing of five forms of boron-carbide is presently continuing in a Hanford thermal reactor. Material forms are 60 and 80% TD powder, and 65, 80, and 99% TD pellets. Irradiation temperatures range from 500 to 1225 °F. Subsequent discharges will yield samples irradiated to burnups of about 10 and 15% of the ^{10}B . Postirradiation analysis will include determination of burnup, gas release, sample swelling, and poison-cladding interaction.

10. Subcritical Crack Growth in Type 304 SS (7N96)

L. A. James and E. B. Schwenk

The objective of the subcritical Crack Growth Studies is to determine the effect of characteristic FTR temperature levels and sodium environment on the fatigue crack growth properties of LMFBR alloys that are to be used in the FFTF Primary Piping System. Knowledge of these properties are necessary to estimate the remaining service life or number of load cycles required to grow a crack in a component from a finite size (below an NDT detection limit) to a size that can be detected. The importance of knowing the crack growth properties of the system materials is amplified by the observation that long term subcritical crack growth accounted for about 55%⁽¹⁾ of failures in general primary pressure piping systems where fatigue was a dominant failure mode. The increased fatigue susceptibility of the system was caused by high temperature levels (up to 1200 °F), heavy wall thickness (1/2 to 3/4 in.), high thermal expansion coefficient, and high thermal diffusivity of Type 304 SS.

1. E. B. Schwenk and L. A. James. Estimate of the Fatigue Cracking Behavior of the Fast Test Reactor Primary Piping System, BNWL-1035. Battelle-Northwest, Richland, Washington. (to be printed)

The test program centers on determining the fatigue crack growth behavior of the major system structural material (Type 304 SS) with respect to analysis by the crack propagation power law,

$$da/dN = c(\Delta K)^n$$

where

da/dN = Crack growth rate (in./cycle)

ΔK = Range of stress intensity factor (ksi $\sqrt{\text{in.}}$)

c, n = Constants for a given material metallurgical condition and test state (e.g., high-temperature and sodium environment)

for temperature levels environments and material conditions that are expected to be characteristic of piping system operation. This information will provide a basis for reasonable estimation of the remaining fatigue or service life in piping components using fracture mechanics analysis techniques.

Test Results

Twelve single edge notch (SEN) specimens were prepared from a 1/2-in. thick plate of Type 304 SS that was characterized under the Irradiation Effects on Reactor Structural Materials Program.⁽¹⁾ Specimen size and layout details are shown in Figure 7.16. All chemical analysis, mechanical tests and test specimens fabricated were performed according to ASTM specifications.

Pulsating tension-tension load-cycling tests of SEN specimens were conducted with two Baldwin tensile test machines equipped with load-cycling attachments, and an MTS electrohydraulic fatigue test machine.

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1. T. T. Claudson. Fabrication History of Alloys Used in the Irradiation Effects on Reactor Structural Alloys, BNWL-CC-236. Battelle-Northwest, Richland, Washington, October 1965.

SEN Test Specimen Configuration

SEN Specimen Layout

FIGURE 7.16. Test Specimen Configuration and Specimen Layout

All crack length measurements were made with one of two Gaertner calibrated traveling stage microscopes, each with a least-dial division of 0.0001 in. Magnifications used varied from 23 to 46X on one model and infinitely variable up to 64X on the other. Crack length measurements were made with 0.03 to 0.06-in. increments and were averaged for both sides of the specimen.

After completion of testing the maximum gross stress (σ_G), load ratio ($R = \sigma_{\min} / \sigma_{\max} = P_{\min} / P_{\max}$) specimen width (W), crack lengths (a) and corresponding load cycles (N) information were submitted for computer processing by the SIEDNO (single-edge-notch) program.

Crack growth rate, $\frac{da}{dN}$ was approximated by dividing the incremental extension $\Delta a (= a_n - a_{n-1})$ required to extend the crack Δa for m crack length measurements from $n = 1$ to $n = m$. Corresponding ΔK values were determined by using the mean or average crack length $\frac{a_{n-1} + a_n}{2}$ and the relationship.(1)

$$\Delta K = \Delta \sigma \sqrt{a_{\text{avg}}} \left[1.99 - 0.41 \left(\frac{a_{\text{avg}}}{W} \right) + 18.70 \left(\frac{a_{\text{avg}}}{W} \right)^2 - 38.48 \left(\frac{a_{\text{avg}}}{W} \right)^3 + 53.85 \left(\frac{a_{\text{avg}}}{W} \right)^4 \right]$$

Figure 7.17 shows results of a typical test. Table 7.VIII shows the results of tests on four specimens, RML-2, RML-3, RMT-4, and RMT-5 (all run at room temperature) after application of a least squares regression analysis. A least squares fit of RML-2 will be done after the computer program is modified to fit second order logarithmic data. The data for RML-3

1. Plane Strain Crack Toughness Testing of High Strength Metallic Materials, ASTM Std 410, p. 22.

FIGURE 7.17. Stress Intensity Versus Crack Growth Rate

and RMT-4 are essentially the same indicating no effect of grain direction between the WR and RW orientation. Limited data was obtained from RMT-5 owing to a malfunction of a load control amplifier which crushed the specimen in compression. Fatigue crack growth testing of Specimen RML-6 at 1000 °F is underway. Too few cycles have been applied at this time to produce any significant growth.

TABLE 7.VIII. *Fatigue Crack Growth Power Law Data -
AISI 304 SS*

Spec. No.	Max. Stress, ksi	Stress Ratio	Crack Orient.	Cycle Rate, Hz	Power Law Equation, $\frac{da}{dN} =$
RML-2	18,980	0	RW	2	to be determined
RML-3	9,250	+0.0044	RW	400	$1.3296 \times 10^{-19} \Delta K^{3.0432}$
RMT-4	9,300	+0.0044	WR	400	$2.6236 \times 10^{-20} \Delta K^{3.2070}$
RMT-5	18,480	+0.0022	WR	180	$6.037 \times 10^{-31} \Delta K^{5.5287}$

11. Effects of Irradiation Temperature on Swelling of Stainless Steel (7N75)

J. L. Straalsund, H. R. Brager, and R. L. Fish

Immersion density measurements were recently completed on AISI Type 316 SS tensile specimens which had been irradiated in the EBR-II (x-018 Subassembly) to a maximum fluence of 3.0×10^{22} n/cm² (E > 0.1 MeV) and at temperatures ranging from 700 °F to 1400 °F.

In Figure 7.18 (the swelling values normalized with respect to fluence by dividing each swelling value by $(\phi t \times 10^{-22})^{1.7}$ where ϕt is the fluence) were plotted versus irradiation temperature. For purposes of comparison, swelling values predicted by the currently accepted swelling equation⁽¹⁾ were also normalized and plotted on Figure 7.18. The two curves agree reasonably well at temperatures below 903 °F, but at higher temperatures the data indicate that a maximum occurs near 1050 °F and that no swelling occurs above about 1400 °F. However, these results are to be considered preliminary in nature.

12. Production of Helium in Stainless Steel by Neutron Fluence (7N75)

J. L. Straalsund, H. R. Brager, and R. L. Fish

Actual and calculated values for production of helium in stainless steel by neutron fluence were compared. The calculated concentration of helium produced in the Type 304 SS irradiated near the EBR-II reactor mid-plane was found to be in reasonable agreement with the measured value for neutrons with E > 1.0 MeV. (See Table 7.IX, Sample No. 6) However, helium concentration of a Sample (See Table 7.IX, Sample 7) irradiated in the ETR, a thermal reactor, is more than 20 times the

1. FFTF Quarterly Technical Report, March, April, May, 1969, BNWL-1090, p. 5.18. Battelle-Northwest, Richland, Washington.

FIGURE 7.28. *Temperature Dependence of Swelling Indicated by Initial Results on Solution Treated AISI Stainless Steel*

calculated value of 13 atom ppm (including 3 atom ppm assuming complete boron transmutation). This result is consistent with the data of Jackson and Dahl⁽¹⁾ that the average energy of fast neutrons ($E > 0.1$ MeV) is higher in the ETR than in the EBR-II.

The calculations do not explain the difference indicated in Table 7.IX in helium produced in Type 304 SS and Type 316 SS

1. J. L. Jackson and R. E. Dahl. *"Measurements of the Neutron Flux in the FFTF Using Activation Monitors," BNWL-577, p. 3. Battelle-Northwest, Richland, Washington, October 1967.*

TABLE 7.IX. Helium Production in Irradiated Steel

Sample No.	Irradiated In Reactor	Type of Material	PNL Sample Designation	Neutron Fluence		Helium Concentration (atom ppm)	
				(Total)	(E > 1 MeV)	Measured	Calculated
1	EBR-II	316 SS	14G-16-5	4×10^{21}	---	0.25	---
2	EBR-II	316 SS	14B-16-1	8×10^{21}	---	0.58	---
3	EBR-II	304 SS	3X (Safety Rod Thimble)	10^{20} to 10^{21}	---	0.025	---
4	EBR-II	304 SS	5F (XG05 Thimble)	3.6×10^{22}	9.7×10^{21}	6.9	---
5	EBR-II	304 SS	C (Center Rod Thimble)	3.4×10^{22}	9.1×10^{21}	9.9	---
6	EBR-II	304 SS	3E (Safety Rod Thimble)	7.0×10^{22}	1.9×10^{22}	16.1	15.2
7	ETR	304 SS	GEH-20-34-7	---	6×10^{21}	305.0	13.0

per nvt of fluence. The difference may be caused by the harder neutron energy spectrum near the EBR-II core mid-plane in Rows 3, 4, and 5 where the Type 304 SS was irradiated, resulting in more (n, α) transmutations than in Row 7 locations positioned 6 and 12 in. from the core centerline where the Type 316 SS was irradiated.

Samples 1 and 2 (Type 316 SS) were irradiated in Row 7 of the EBR-II in the same pin, 12 in. above and 7 in. below, respectively, the reactor mid-plane. Sample 3 was taken from the top section of the safety rod thimble about 10 in. above the top of the core and was to be the "control" specimen. Samples 4, 5, and 6 (Type 304 SS) were irradiated within 2 in. of the reactor mid-plane in Rows 4, 5, and 3, respectively. Sample 7 was irradiated in the ETR.

The five elements which were calculated to contribute more than 98% of the helium produced by fast-neutron transmutation in Type 316 SS and Type 304 SS are shown in Table 7.X.

The calculations were based on the fission spectrum average (n, α) cross section of the elements tabulated by Alter and Weber⁽¹⁾ except for nickel where the cross section reported by Weitman and Däverhög⁽²⁾ was used.

The helium concentration in the seven irradiated austenitic stainless steel samples was tested by Dr. H. Farrar at Atomic International. The helium concentration was estimated to have an uncertainty of less than 10%.

TABLE 7.X. *Fast Neutron (n, α) Transmutation of Major Alloy Constituents*

Element	Type 316 SS ^(a)			Type 304 SS ^(a)		
	wt %	Atom He	(atom $\times$ 1 nvt)	wt %	Atom He	(atom $\times$ 1 nvt)
Ni	13.55	539	(-30)	9.36	365	(-30)
Fe	63.64	279	(-30)	70.31	304	(-30)
Cr	17.80	72	(-30)	18.51	74	(-30)
N	0.041	67	(-30)	0.029	46	(-30)
Mo	2.33	10	(-30)	0.10	0.4	(-30)
TOTAL			9.8×10^{-28}			8.0×10^{-28}

a. Boron Content < 5 ppm for both heats of steel.

1. H. Alter and C. E. Weber, "The Production of Hydrogen and Helium in Metals During Reactor Irradiation," J. Nucl. Matl. **16**, 68 (1965).
2. J. Weitman and N. Däverhög, "Helium Production Cross Section Measurements," NBS Spec. Pub 299, 125 (1968).

13. Status of EBR-II Irradiations (7N75)

P. K. McDaniel

The EBR-II irradiations program provides for coordination and direction for the timely accomplishment of fast reactor irradiations of structural materials. Structural materials irradiation experiments are in various stages of development: some in design, some under construction and others under irradiation in the EBR-II. The status of these experiments are described.

A metallurgical variables experiment was being assembled for insertion in EBR-II about September 1, 1969. This pin will be a replacement pin in subassembly X-018. Its goal exposure is 4×10^{22} n/cm² (E > 0.1 MeV). The objectives of this experiment are to provide information concerning the degree to which irradiation-induced swelling in Type 304 and Type 316 SS can be controlled by simple heat treatments, and to establish a relationship between cold work (less than 25%) and irradiation-induced swelling as a function of irradiation temperature.

At the end of run No. 36 (8-1-69) two structural subassemblies (X-019 and X-057) were discharged from EBR-II. Subassembly X-019 consisted of 8-A type experiment pins for BNW. The remainder of the pins are sponsored by General Electric Company or United Nuclear Corporation. The objective of the experiment was to achieve exposure between 1×10^{22} and 1×10^{23} n/cm² fast flux in an environment of bulk sodium and helium at temperatures of 700 °F to 850 °F. The cumulative exposure was 3.8×10^{22} n/cm² (total flux). Postirradiation testing of these materials (Types 304 and 316 SS) will provide data for evaluating the effect of irradiation on design values in support of the LMFBR Program.

Subassembly X-067 consisted of 7 pins; four pins from BNW, two from Argonne National Laboratory and one from General Electric Company.

The four BNW pins contain specimens for a study of the effect of fast reactor irradiations on Type 304 and Type 316 SS weldments in support of the FFTF pressure vessel and fuel ducts.

The accumulative exposure was 9×10^{21} n/cm² (total flux). An updated summary of active and discharged experiments to the end of run No. 36 is given in Tables 7.XI and 7.XII.

Arrangements have been made for dismantling and evaluating flux monitors from these experiments.

TABLE 7. XI. *Summary of Experiments in EBR-II*

Subassembly Designation	Reactor Position	Date Charged	Estimated Discharge Date	Type of Specimens	Material	Accumulative ^(b) Exposure n/cm ²	Goal Exposure n/cm ²
X-020	6B5	1-13-67	July 69	28 Tensile 8 Uniaxial Creep	304 SS 316 SS	3.5×10^{22}	1.9×10^{22}
X-021 ^(a)	2D1	2-27-67	Aug. 70	501 Tensile or Uniaxial Creep 65 Biaxial Creep	304 SS, 316 SS, 348 SS, 321 SS ^(c) Ni Base Alloys	8×10^{22}	8.8×10^{22}
X-027	4B3	11-21-67	Aug. 70	5 Tensile 3 Uniaxial Creep	304 SS	3.9×10^{22}	5×10^{22}
X-041	7C4	7-24-68	May 70	713 Tensile or Uniaxial Creep 74 Biaxial Creep	304 SS, 316 SS, 321 SS, 348 SS	1.7×10^{22}	3×10^{22}
X-057	2B1	2-24-69	March 71	577 Tensile or Uniaxial Creep 164 Biaxial Creep 58 Density	304 SS, 316 SS, 321 SS, 348 SS, Ni Base Alloys ^(c)	1.9×10^{22}	6.0×10^{22}

a. Original space available with two single pin interchanges.

b. Accumulative to end of run 33, ($E > 0.1$ MeV).

c. Variety of other alloys.

TABLE 7.XII. Summary of Experiments Discharged from EBR-II

Subassembly Designation	Reactor Position	Date Charged	Date Discharged	Type of Specimen	Material	Accumulative Exposure ^(d) n/cm ²
X-009 ^(a)	4A2	3-24-66	11-15-66	18 Tensile 10 Uniaxial Creep	304 SS	2.0×10^{22}
X-010 ^(a)	7F3	3-24-66	4-24-69	18 Tensile 10 Uniaxial Creep	304 SS	6.6×10^{22}
X-014 ^(a)	2D1	7-17-66	12-31-66	50 Tensile	304 SS	1.7×10^{22}
X-018	2B1	12-6-66	2-25-69	198 Tensile or Uniaxial Creep	304 SS, 316 SS, 348 SS, 321 SS	4.4×10^{22}
X-019	6D2	1-13-67	July 69	64 Tensile 24 Uniaxial Creep	304 SS 316 SS	2.5×10^{22}
X-021 ^(b)	2D1	2-27-67	9-16-68	127 Tensile or Uniaxial Creep	304 SS, 316 SS ^(c) Ni Base Alloys	3.2×10^{22}
X-021A ^(b)	2D1	9-27-68	11-16-68	86 Density 16 Biaxial Creep 14 Tensile 10 Uniaxial Creep	316 SS ^(c) Ni Base Alloys	5.0×10^{21}
X-022	7C4	2-27-67	6-15-68	595 Tensile or Uniaxial Creep 65 Biaxial Creep	304 SS, 316 SS, 348 SS, 321 SS	1.1×10^{22}
X-030 ^(a)	6E1	10-10-67	3-6-68	5 Tensile 3 Uniaxial Creep	304 SS	5.2×10^{21}
X-067	4F2	1-5-69	Sept. 69	204 Tensile or Uniaxial Creep 20 Fatigue	304 SS, 316 SS, Weldment	0.9×10^{22}

a. Mark A pins irradiated on space-available basis,
all other pins are B-7 type.

b. Single pin interchange.

c. Variety of other alloys also included.

d. $E > 0.1 \text{ MeV}$

VIII. FUELS RECYCLEA. FUEL TECHNOLOGY1. Effect of Dry Pressing Variables on Sintered Properties of UO_2 Pellets⁽¹⁾

J. E. Sammis (7N73)

a. Objectives and Results

The objectives of this pressing test were to determine the effect of forming pressure, preslug pressure, and hold time on the sintered properties of natural Eldorado UO_2 pellets. A $5 \times 3 \times 2$ factorial design was employed. Analyses of variance were used to determine the significant effects.

Sintered density was found to be the most important variable affected by pressing conditions. It may be seen from Figure 8.1 that the sintered density of the pellets is relatively insensitive to forming pressures exceeding 30,000 lb/in.² Figure 8.2 shows that sintered length is insensitive to forming pressures between 30,000 and 40,000 lb/in.² The information obtained will be used as a guide in designing future tests with mixed oxide fuel and to determine the range of the input variables within which optimum properties can be expected to occur.

Dry pressing was conducted with a Western Sintering Company hydraulic press, Model H-10-2.

1. J. E. Sammis. Parametric Study of the Pellet Compacting Operation BNWL-1198. Battelle-Northwest, Richland, Washington.

FIGURE 8.1. *Sintered Density Versus Forming Pressure*

FIGURE 8.2. *Sintered Length Versus Forming Pressure*

b. Design of Experiment

Based on results of preliminary tests, a $5 \times 3 \times 2$ factorial experiment was chosen to examine the effects of pellet press variables (such as forming pressure, dwell time, and preslug pressure) and their interactions on the output variables (green and sintered lengths, diameters, weights and percent of theoretical density). Factor levels are shown in Table 8.1.

All possible combinations of the factors are illustrated in Figure 8.3; each corner represents a set of test conditions, thirty in all. Four pellets were produced for each test condition, making a total of 120 pellets. The groups or "cells" of

four pellets each were produced in random order. Random variability within tests was reflected in an error term in the analysis of variance.

TABLE 8. I. *Levels of the Factors for Pellet Press Parametric Study*

Forming Pressure, lb/in. ²	10,000 20,000 30,000 40,000 50,000
Dwell Time, sec.	0.2 0.4 0.6
Preslug Pressure, lb/in. ²	5,000 10,000

FIGURE 8.3. *Design of the Experiment*

c. An'a'lysis' of Variance

The data from the experiment was analyzed under the following assumed model:

$$Y_{hijk} = \mu + \alpha_h + \delta_h + \beta_i + \gamma_j + (\alpha\beta)_{hi} + (\alpha\gamma)_{hj} \\ + (\beta\gamma)_{ij} + (\alpha\beta\gamma)_{hij} + \zeta_{hij} + \epsilon_{hijk}$$

Y_{hijk} = "measurement" for k^{th} pellet at preslugging level h , hold time level i and forming pressure level j

α_h = effect of h level of preslugging

β_i = effect of i level of hold time

γ_j = effect of j level of forming pressure

$(\alpha\beta)_{hi}$, $(\alpha\gamma)_{hj}$, $(\beta\gamma)_{ij}$ = first order interactions

$(\alpha\beta\gamma)_{hij}$ = second order interaction

δ_h = random component among preslugging levels,
Normal $(0, \sigma_\delta^2)$

ζ_{hij} = random component among groups of pellets treated
alike, Normal $(0, \sigma_\zeta^2)$

ϵ_{hijk} = random component within groups of pellets treated
alike and produced at one time, Normal $(0, \sigma_\epsilon^2)$

Under the parameters of the above model, the three factor (preslug pressure, hold time, forming pressure) interaction mean square was used as an F - ratio denominator. This caused the tests for main effects and first order interactions to yield conservative results.

2. Statistical Analyses of Fuel Pellet Measurements

J. E. Sammis (7N73)

Since the last reporting period, a statistical analysis of sintered fuel pellet dimensions has been completed. Pellets used in this study represent approximately one-half of those pellets produced for the physics test pins. Table 8.11 gives a

TABLE 8.II. Statistical Analyses of Sintered Fuel Pellet Measurements (a)

Batch No.	Approx. Population	End Diameter (in.) D ₁	Center Diameter (in.) D ₂	End Diameter (in.) D ₃	Taper (in.)	Hourglass (in.)	Percent	Theoretical Density
A-5-S18	1,270	0.19689 0.00018	Not Taken	0.19527 0.00030	0.00162 0.00029	Not Taken	93.30 0.36	Ave Std Dev
A-6-S18	2,420	0.19534 0.00020	Not Taken	0.19432 0.00024	0.00102 0.00022	Not Taken	93.44 0.51	Ave Std Dev
A-7-S20	560	0.19499 0.00011	Not Taken	0.19400 0.00012	0.00100 0.00014	Not Taken	93.52 0.18	Ave Std Dev
B-1-S19	1,255	0.19416 0.00016	0.19245 0.00042	0.19313 0.00019	0.00103 0.00024	0.00120 0.00039	93.91 0.56	Ave Std Dev
B-2-S19	1,750	0.19425 0.00019	Not Taken	0.19307 0.00018	0.00118 0.00020	Not Taken	94.50 0.45	Ave Std Dev
B-3-S20	1,875	0.19505 0.00018	Not Taken	0.19375 0.00027	0.00130 0.00022	Not Taken	92.91 0.29	Ave Std Dev
B-4-S21	1,620	0.19627 0.00042	0.19442 0.00051	0.19481 0.00041	0.00146 0.00040	0.00112 0.00037	93.82 0.66	Ave Std Dev
B-5-S21	1,610	0.19613 0.00018	0.19428 0.00024	0.19490 0.00020	0.00123 0.00021	0.00124 0.00022	93.61 1.04	Ave Std Dev (d)
B-6-S22	1,184	0.19782 0.00063	0.19635 0.00057	0.19656 0.00065	0.00126 0.00032	0.00084 0.00035	91.60 0.83	Ave Std Dev
B-7-S22	1,736	0.19603 0.00010	0.19438 0.00016	0.19503 0.00019	0.00100 0.00019	0.00115 0.00017	92.58 0.36	Ave Std Dev
C-1-S23	1,458	0.19585 0.00025	0.19421 0.00031	0.19484 0.00022	0.00101 0.00020	0.00113 0.00027	93.85 0.37	Ave Std Dev
C-2-S23	1,510	0.19609 0.00013	0.19431 0.00014	0.19499 0.00010	0.00110 0.00015	0.00123 0.00012	93.55 0.31	Ave Std Dev

(a) Based on sample sizes of 30.

(b) Taper = $D_1 - D_3$ (c) Hourglass = $\frac{D_1 + D_3 - D_2}{2}$

(d) Large standard deviation, and correspondingly large tolerance interval for this batch was believed due to an incorrect pellet weight reading. After eliminating this pellet, the range of densities was 92.97 to 94.13 percent.

statistical summary of the pellet measurements deemed most important from the standpoint of sintering to size.

3. Evaluation of Pellet Impurities

M. J. Barr (7N73)

Metallic impurities, carbon, fluorine, chlorine, and water content data from 28 development and irradiation test pellet batches were summarized to determine the following relationships:

- a. Impurities in processed powder that are driven off during sintering.
- b. Reliability of the chemical analyses of sintered pellets. (Determined by comparing chemical analyses of sintered pellets from the same batch but analyzed at different times.)
- c. Reliability of chemical analyses of fuel powder lots by comparison of multiple analyses conducted on the same powder lot.
- d. Reproducibility of fuel pellet impurity contents among sintering runs for pellets produced from the same powder lot.

The data are being analyzed to determine statistical significance. Preliminary data concerning possible impurity change through sintering is shown in Table 8.III.

TABLE 8.III. *Effect of Sintering on Impurity Content*

<u>Impurity</u>	<u>Before Sintering (ppm)</u>	<u>After Sintering (ppm)</u>
Al	470	470
Cu	60	18
Fe	72	54
Ni	<44	<16
P	<48	<43
C	340	<36
H ₂ O	3300	5

4. Effects of Milling on PuO₂ Sinterability

M. J. Barr (7N73)

To determine the effect of grinding of PuO₂ powder, a small quantity of oxalate source plutonia was recalcined to 625 °C and subportions of both as-received and recalcined powder were then dry ball milled using Al₂O₃ media for 8 to 20 hours. Pellets were pressed at 40,000 psi and sintered at 1550 °C for two hours in helium bubbled through water (approximately 1500 ppm H₂O). Table 8.IV shows the results obtained.

TABLE 8.IV. *Effect of Milling on the Sinterability of PuO₂*

<u>Treatment</u>	<u>B.E.T. Surface Area (M²/gm)</u>	<u>Carbon (ppm)</u>	<u>Aluminum (ppm)</u>	<u>Green Density %TD</u>	<u>Sintered Density %TD</u>
No Milling					
As Received	37.0	1125	10	48.6	91.0
625 °C Calcine	17.4	590	ND	55.9	92.8
8-hr Milling					
As Received	27.5	ND	10	49.8	88.9
625 °C Calcine	12.1	ND	ND	57.5	85.8
20-hr Milling					
As Received	24.0	ND	10	51.6	86.2
625 °C Calcine	10.0	ND	ND	58.5	85.1

ND Not determined

The foregoing table shows that milling reduced the B.E.T. surface area for both as received and calcined PuO₂. This effect is possibly a result of agglomeration. The difference in sintered densities also implies that the agglomerate condition has been changed. Scanning electron micrographs will be obtained to help explain the nature and state of particle agglomeration. In addition, particle size, and O/Pu ratios are being determined.

5. Standardization of Analytical Procedures

a. Analytical Chemistry Program

B. R. Hayward and F. M. Smith (7N71)

Certain impurity limits imposed on FFTF fuels from the neutronics standpoint are being reviewed. For reasons in the realm of physics it is recommended that the sum of the concentrations of samarium, europium, gadolinium and dysprosium be limited to 100 ppm. At the present time, the BNW Analytical Labs use a semi-quantitative spectrographic method for rare earths in non-nuclear materials. Adequacy of this method is being evaluated.

The potential Am interference with the Pu assay procedure is being studied. Preliminary results from LASL indicate that Am offers no interference in concentrations up to 1100 ppm. LASL is extending this work to include concentrations of Am to about 0.6%.

Offsite work on homogeneity included making some alpha-autoradiographs of both Phase I and Phase II pellets (ME-21, PL-10). These autoradiographs illustrated noticeable differences in the total alpha activities of the two types of pellets. This effect resulted from the filter medium used to collimate the alpha particles. Later total alpha analysis of these pellets at BNW showed them to be essentially identical.

b. Raw Materials Analysis

J. F. Williams (7N71)

Three contractors (Babcock and Wilcox, NUMEC, and United Nuclear) are participating in the program for prequalification as LMFBR fuel suppliers and all three have started processing the raw materials. Babcock and Wilcox and UNC have reported no problems on PuO_2 shipper-receiver assay differences, while NUMEC has reported a difference of 1.5 to 2%. This is currently being resolved through referee analysis by the National Bureau of Standards.

it is important to account for density dependence in the equation of the state of the core. This is especially true for situations in which a significant sodium inventory is present in the core.⁽¹⁾ Exceptionally large pressures for such cases can be generated relatively early in a transient as the result of a small degree of fuel expansion. However, even an infinitesimal amount of core expansion is often adequate to relieve the driving pressure back to the two-phase vapor equilibrium state. Neglect of these effects can cause large errors in the history of a disassembly accident.

The equation of state used for the present study is basically that developed by Harris.⁽²⁾ The approach is to utilize existing density-temperature data for UO_2 to determine the temperature at which the departure point into the single phase liquid region is attained, and then compute the pressure-temperature relationship in this region based on the theorem of corresponding states. For the numerical example reported below, Menzies critical constants⁽³⁾ were employed. Sodium compressibility is accounted for by reducing $\partial P / \partial T$ in the single phase region and adjusting the effective departure point temperature.

The temperature-density-dependent equation of state as formulated above has been successfully incorporated into the VENUS computer program.⁽⁴⁾ It is to be noted that one of the

-
1. A. E. Waltar, R. A. Harris and A. Padilla, Jr. "Considerations of Sodium Conditions and Molten-Fuel/Sodium Interactions for Fast Reactor Excursion Calculations," *Trans. Am. Nucl. Soc.*, 11, 694 (1968).
 2. R. A. Harris. "Preliminary Analysis of Postulated Maximum Accidents for the FFTF: Supplement 1 - Additional Calculations for Sodium-In Conditions," BNWL-760, Supplement 1, Battelle-Northwest, Richland, Washington, December 1968.
 3. D. C. Menzies. *The Equation of State of Uranium Dioxide at High Temperatures and Pressures*, TRG-Report-1119 (D). Technical Research Group, New York, 1966.
 4. W. T. Sha, T. W. Hughes, and P. Margel. "VENUS - Two-Dimensional Neutronic-Hydrodynamic Power Excursion Computer Program," (to be published) 1969.

characteristics of this type of formulation is that the pressure can oscillate between two-phase and single-liquid-phase regions with little density change. This provides a severe test of the program from the numerical stability standpoint, although experience to date has indicated no numerical instability for the problems of interest. The VENUS program employs a mid-point finite difference scheme⁽¹⁾ in the hydrodynamics which is a modified version of Kolsky's method.⁽²⁾ Shock waves are handled by the "shock smearing" technique developed by Von Neumann and Richtmyer.⁽³⁾

Figure 9.2 illustrates the typical pressure-density behavior with a temperature-density-dependent equation of state for sodium-in conditions. The particular example chosen was a rapid power excursion in an FTR type core in which the core (radius ~60 cm, height ~90 cm) was divided into 11 radial and 18 axial nodal points. Values indicated are for the node in the core center after the transient reaches the disassembly phase. If the compressibility of sodium is ignored, the solid lines indicate a sharp rise in pressure when the fuel density drops to the point where all the void space available to the fuel is filled. Nodal expansion quickly follows, causing a density decrease. This relieves the liquid pressure allowing the pressure buildup in adjacent nodes to cause recompression. High pressure again follows and the process oscillates until enough overall core expansion

1. L. Amnrud and R. S. Orr. "A Note on Inverted Centers of Pressure and Crossed Mass Points in a Two-Dimensional Hydrodynamics Calculation," LASL Unpublished Note 1963.
2. H. G. Kolsky. "A Method for the Numerical Solution of Transient Hydrodynamics Shock Problems in Two Space Dimensions," LA-1867 (1955).
3. J. Von Neumann and R. D. Richtmyer. J. Appl. Physics, 21 (1950).

FIGURE 9.2. *Pressure - Density Variations at the Central Node for a Hypothetical 100\$/sec Ramp in a FTR Type Core*

has occurred to terminate the excursion. If sodium compressibility is taken into account, peak pressures are lower because of the increased volume available to the fuel. For the particular case chosen, inclusion of the compressibility effect increased the amount of energy contained in the molten fuel by about 10%.

3. ETOX

J. L. Baker, R. E. Schenter, and D. P. Taylor

Group cross sections generated from the three computer codes: ETOX-1DX, MC², and GAFGAR, using the ENDF/B library as input, were compared for the fast reactor ZPR II assemblies - 52, 50, and zebra assembly - 6A. Results for these calculations for assembly - 52 show that in the energy region having significant fast reactor flux, the group cross sections agree very closely. Discrepancies observed in the overlap regions were eliminated by improving the interpolation methods in ETOX. The interpolation methods have been revised between the resolved and unresolved regions and between the unresolved and smooth regions. Discrepancies in these regions are now less than one percent. Modifications also have been added to improve calculations in the neighborhood of the three resonances in the resolved region for NA-23.

The severe disagreements between the codes in the resolved resonance region casts doubt on the use of any of the codes for analyzing effects caused by reactions in the resolved region. Consequently, effort is underway to resolve these discrepancies.

4. RSYN

K. B. Stewart

RSYN (Reactor SYNthesis) solves the three-dimensional, multigroup reactor diffusion equations by the method of flux synthesis. The two-dimensional diffusion code 2DB is used to generate the trial functions in the horizontal plane, and the

full three-dimensional solution is then synthesized. In the current version, only the x, y geometry option of 2DB has been incorporated into RSYN. Subsequent versions will be extended to include the (r, θ) and hexagonal geometry capabilities of 2DB has been incorporated into RSYN. Subsequent versions will be extended to include the (r, θ) and hexagonal geometry capabilities of 2DB in generating trial functions. Down-scattering from any group to all groups lower in energy is permitted.

RSYN will allow up to 50 energy groups. The maximum number of X and Y intervals is 2,500. The number of Z intervals in mesh used to describe the Z-axis of the reactor must not exceed 50.

The total number of axial layers used must not exceed 10 and the total number of trial functions used to describe these layers cannot exceed 15 per group. Because of variable dimensioning, it is not possible to state categorically which combinations will fit in-core at any one time.

The calculation of the leakage term involves $\text{grad } \phi(X, Y, Z)$. The initial programming effort found that the derivatives could not be calculated across material boundaries and that the surface integral term included in the leakage calculation was especially sensitive to the partial derivative calculations. Originally the derivatives were calculated assuming $\text{grad } \phi$ for the trial functions was continuous across material boundaries but a sample problem with a diffusion coefficient on the order of 10^3 barns painted out the difficulty of correctly calculating a partial derivative over the boundaries.

The second major area of difficulty was the surface integral contribution to the leakage. The effect was much larger than anticipated (approximately 2%) and because of this, the programming necessary to calculate the surface integral was included in RSYN. The calculation of k_{eff} is still effected by this calculation but the effect is on the order of 0.5%.

The next phase of programming will take into account that $D(X,Y) \text{ grad } \phi(X,Y)$ is continuous across material boundaries for the individual trial function in the partial derivative calculation and the calculation of the surface integral contribution should be improved subsequently.

RSYN can be used on a general basis. However, some experience should be accrued on sample problems before it is used to analyze a complete reactor. Generally, the synthesis calculations require about the same amount of computer time as one of the 2DB calculations. Also, the synthesis direction should be the one of least complexity.

5. Cooling of Irradiated Fuel During Examination (7N91)

D. R. Dickinson and T. C. Reichman

The objective of this program is to determine the requirements for gas cooling of fuel assemblies during examination and disassembly following exposure in the FFTF. Before removal of the metal fuel bundle liner during disassembly, cooling will be attained by passing an axial flow of argon through the liner. If the liner cannot be slid off easily it may be necessary to slit it, thereby allowing some of the coolant to escape. The purpose of the work reported below was to determine the additional flow necessary to provide adequate cooling with a slit liner.

Experiments were performed using the electrically heated model containing 169 pins with a 36-in. heated length, 0.250-in. diameter, wrapped and spaced with a 0.030-in. wire on a 12-in. pitch. The pin bundle was surrounded by a closed-fitting hexagonal liner. Two longitudinal slits were cut in the liner near two opposite vertices over the 36-in. heated length. Measurements were also made with slit widths at 0.027 and 0.060 in. A third test with a 1/8-in. slit will be completed soon. Air was passed from a blower, over the unheated (gas plenum) end of the heated bundle, over the heated portion

and discharged into the atmosphere. At each slit width, runs were made over a range of power inputs with the air flow adjusted to provide maximum pin temperatures near 600 °F.

The inlet flow required to maintain all pin temperatures below 600 °F was about 1.4 and 2.0 times as great for the 0.027 and 0.060-in. slits, respectively, as that required with an unslit liner. This agrees fairly well with calculated flow increases by factors of 1.6 and 2.1. This suggests that previously reported calculated flow increases required for slit liners are valid.

B. EVALUATION OF CROSS SECTIONS FOR SHIELD CALCULATIONS (7N81)

D. R. Marr

Calculations of the integral shielding quantities of interest in the design of the Fast Flux Test Facility (FFTF) require a capability to determine the neutron transmission through materials such as iron, nickel, stainless steel, and sodium. In this study two cross-section sets are evaluated by comparing calculated and measured transmission spectra⁽¹⁾ over the energy range 1 eV to 1 MeV for iron, nickel, and stainless steel.

Computer codes being used for shield design require cross sections in a multigroup format for economy of operation. Reduction of experimental cross section data to such a multigroup set of average cross sections is dependent upon the spectrum within the group for all groups in which the cross section is not constant. Because of vast differences in spectra in FTR shield regions, and because the integral shielding quantities of interest have significantly different energy response functions, a universally applicable reduction cannot be made to a few group set. For materials with many resonances, such as iron, it requires hundreds of groups to describe the

1. I. I. Bondarenko, et al. "Measurement of Neutron Spectra in Nickel, Iron, and Stainless Steel," Atomnaya Energiya, Vol. 18, No. 6, June 1965.

known cross section structure. For practical shielding applications, this vast amount of data must be reduced to a tenable structure containing few groups, but one that yields acceptably accurate results. Because of uncertainties in both the method of reduction and in the basic data, the reduced sets need to be evaluated using integral shielding experiments.

Of the two sets evaluated in this study, the 31 group HOMSET cross sections were used in FFTF conceptual design studies and this evaluation provides a measure of validity of the reported results. The 99 group DLC-2 set,⁽¹⁾ derived from the Evaluated Nuclear Data File (ENDF), is evaluated as a basis for a possible successor to the currently used set.

Calculated neutron transmission spectra through iron, stainless steel, and nickel slab shields compare favorably with measured spectra, using both the 31-group HOMSET cross sections and the 99-group DLC-2 set derived from ENDF. All calculations were made using the one dimensional discrete ordinates code ANISN⁽²⁾ with an S_{16} angular approximation. Comparisons were made for thicknesses of 30 and 48 cm of iron, 30 and 49 cm of stainless steel, and 30.6, 51.5 and 77.0 cm of nickel. Typical results are shown in Figures 9.3 and 9.4 where the calculated and experimental spectra for 30 cm of stainless steel and 30 cm of nickel are compared. Absolute magnitudes of the experimental spectra are not known; therefore, the experimental points are normalized in such a manner that the total transmitted experimental flux agreed with the total transmitted flux in the calculation using HOMSET cross sections. The calculation using ENDF cross sections agree very well with the experiment for the stainless

1. RSIC Newsletter No. 48, Union Carbide Corp., Nov., 1968.
2. W. W. Engle, Jr., A User's Manual for ANISN, K-1693, March 30, 1967

30 CM. STAINLESS STEEL TRANSMISSION SPECTRA

FIGURE 9.3. 30 cm Stainless Steel Transmission Spectra

30.6 CM. NICKEL TRANSMISSION SPECTRA

FIGURE 9.4. 30.6 cm Nickel Transmission Spectra

steel, both overall and in detail. The impurities which caused resonance dips in the measured spectra at 0.335, 1.1 and 2.35 KeV for Mn and 132 eV for Co were ignored in the calculations after it was determined that their effects were quite local and would not compromise the overall comparison. The HOMSET calculation agrees well at high energies and conservatively predicts too much transmitted flux at low energies. Agreement for nickel is not as good. Disagreement is primarily at the lower energies where both cross section sets over-predict the transmitted flux.

It is concluded that HOMSET cross sections give adequate spectral agreement for use in survey design applications even though some deficiencies exist. The ENDF cross sections give somewhat better overall agreement, primarily because of the finer energy mesh.

C. IRRADIATION ANALYSIS OF THE UO₂ MELTDOWN PAD
IN THE FTR (7N81)

E. T. Boulette

A design concept of the FTR includes a 3-in. thick, 36-ft diameter depleted UO₂ meltdown pad located on the reactor cavity floor. Between the UO₂ pad and the floor is a 2-in. thick steel plate. Above the pad is a 1/4-in. steel plate followed by seven layers of sand-filled pipes that form a blast shield.

Calculations were made to determine the heat generation in the pad at full core power, and the resulting radioactivity. To obtain lower and upper bounds, calculations were made both with and without the blast shield in position. The neutron flux distribution was assumed to be uniform throughout the pad, and the intensity was obtained by extrapolation of R-Z calculations for Concept V-A. With the blast shield in position, the

total flux at the pad was calculated to be 1.2×10^3 n/cm²-sec; with no blast shield present, it would be about 2.0×10^8 n/cm²-sec.

The UO₂ pad is to be depleted to 0.3 wt% in the ²³⁵U isotope. The resulting mass of ²³⁸U is 6.6×10^7 grams, and the mass of ²³⁵U is 2.0×10^5 grams. All burnup of fissionable material in the pad was disregarded. The formation rate of ²³⁹Pu from the (n,γ) reaction with ²³⁸U was assumed to be constant with time; thus, the number of ²³⁹Pu isotopes at any time may be calculated from:

$$^{239}\text{N}(t) = 5.3 \times 10^7 t \text{ (atoms of } ^{239}\text{Pu)},$$

where 5.3×10^7 is the formation rate and t is the core operation time at design power in seconds.

The fission heat generation of ²³⁸U, ²³⁵U, and ²³⁹Pu were calculated both with and without the blast shield in position. The fission heat generation of ²³⁹Pu was calculated for ²³⁹Pu formation periods of 1 and 10 full power operating years.

In addition to fission heat generation, decay heat and radioactivity calculations were made for the decay of ²³⁸U, ²³⁵U, ²³⁹Pu, and the fission products generated by these isotopes. To obtain the radioactivity of these fission products, it was assumed the fission product decay heat was 1% of the fission heat generated in the pad. This is believed to be a quite conservative value to apply to the pad for consideration of radiological hazards associated with its removal. Fission product radioactivity was based on the fission product decay heat assuming a typical fission product disintegration energy of 1 MeV.

The results of the calculations, given in Table 9.I indicate that α-decay of uranium will be the primary source

TABLE 9.I. Heat Generation and Radioactivity in UO_2 Pad

SOURCE	BLAST SHIELD IN		BLAST SHIELD OUT	
	HEAT GENERATION (WATTS)	RADIOACTIVITY (CURIES)	HEAT GENERATION (WATTS)	RADIOACTIVITY (CURIES)
^{238}U α -decay	5.4×10^{-1}	2.2×10^1	5.4×10^{-1}	2.2×10^1
^{235}U α -decay	1.1×10^{-2}	4.2×10^{-1}	1.1×10^{-2}	4.2×10^{-1}
Fission product decay	1.3×10^{-5}	2.3×10^{-3}	2.3	3.9×10^2
^{239}Pu α -decay(1 yr)	1.2×10^{-9}	4.1×10^{-8}	2.0×10^{-4}	7.0×10^{-3}
^{239}Pu α -decay(10 yr)	1.2×10^{-8}	4.1×10^{-7}	2.0×10^{-3}	7.0×10^{-2}
^{238}U fission	1.4×10^{-9}		2.3×10^{-4}	
^{235}U fission	1.3×10^{-3}		2.3×10^2	
^{239}Pu fission(1 yr)	8.6×10^{-15}		1.5×10^{-9}	
^{239}Pu fission(10 yr)	8.6×10^{-14}		1.5×10^{-8}	

of heat generation and radioactivity if the conceptual blast shield is employed. If no blast shield is employed, fissioning of ^{235}U would become the dominant heat source in the pad during operation, and fission product decay would become the dominant radioactivity source following shutdown. Although the radioactivity (curies) generated by fission product decay in the pad without the blast shield present is estimated to be about 20 times greater than that resulting from ^{238}U α -decay, the biological hazard (per curie) associated with release of fission products is about three orders of magnitude less than that associated with ^{238}U decay. Thus, from a biological hazard standpoint, release of the ^{238}U would be of primary concern.

In conclusion, the positioning of a depleted UO_2 meltdown pad in the radiation field at the cavity floor position will not introduce heat generation nor radioactivity problems significantly different from those associated with the natural α -decay of the uranium.

D. ANALYSIS OF 56B CRITICAL EXPERIMENTS (7N83)

S. L. DeMyer

Calculations have been made of axial and radial activity worth and reaction rate distributions for those materials which were measured in the Assembly 56B experiments. This work essentially concludes the analysis of the Phase B, Part III critical experiments.

The specific measurements which were analyzed were ^{10}B and ^{239}Pu reactivity worths, ^{239}Pu and ^{238}U fission rates and ^{10}B absorption rate. The analysis was made using the 2-D perturbation code PERT V with fluxes and adjoints from the diffusion code 2DB. All cross sections were the 26-group BNWL modified set.

The analytical techniques are different in group structure and in calculation theory (transport versus diffusion) from both the Assembly 48, 48A analysis⁽¹⁾ and the analyses of Assemblies 51⁽²⁾ and 52.⁽³⁾ Yet the results compare with experiment in a similar manner in all of the above. A generalized comparison of calculation and experiment would include:

- a. The calculated worth of ^{239}Pu is too large near the core center and in the radial reflector but agrees closely with the experiment at the core-reflector boundary and in the axial reflector.
- b. The calculated ^{10}B reaction rates, when normalized to the experiment, are too low at the core reflector interface and too high in the radial reflector but agreement is good elsewhere.
- c. The calculated ^{238}U fission rates are in good agreement with the experiment.
- d. The calculated ^{10}B worth and ^{239}Pu reaction rates are in fair agreement with the experiment except in the region of the core reflector interface.

The consistent problem of lack of agreement between calculation and experiment has been studied. Although the reason for disagreement has not been found, it has been concluded that the disagreement was not caused by collapsing of energy groups nor the model used in the calculation of the fluxes.

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1. R. A. Bennett, S. L. DeMyer. Analysis of FTR Phase A Control Rod Experiments in ZPR III Assembly 48 and 48A, BNWL-967, Battelle-Northwest, Richland, Washington, April 1969.
 2. W. R. Young, R. A. Bennett. Analysis of Phase B Critical Experiments, Part 1, ZPR III Assembly 51, BNWL-1138, Battelle-Northwest, Richland, Washington, July 1969.
 3. W. R. Young, R. A. Bennett. Analysis of Phase B Critical Experiments, Part 2, ZPR III Assemblies 52a, b, c, d, e, and f. BNWL-1139, Battelle-Northwest, Richland, Washington, July 1969.

X. SAFETY

A. COMPUTER CODES APPLICATIONS FOR SAFEGUARDS ANALYSIS (7N82)

1. Transient Analysis of FFTF Inner Containment Barrier to DBA R. D. Peak

A digital computer code, CONRAD, has been prepared to calculate transient temperatures and pressures within the FFTF inner containment barrier following hypothetical core meltdown and other disruptive accident potentials which establish the basis for containment. The CONRAD code includes several modifications of the digital computer code CONTEMPT which had been prepared to analyze the temperatures and pressures within an air containment following a loss-of-coolant accident of a water power reactor. The CONTEMPT code is described in IDO-17220. One modification involved the substitution of nitrogen and sodium properties for the air and water properties used in the original program.

Another modification for the CONRAD code provides for the analysis of the energy-mass-pressure-temperature responses of two containment volumes. Each volume is described by a vapor region (nitrogen and sodium vapor) and a liquid region (sodium liquid). The two volumes are connected so that transfer of nitrogen and sodium vapor between the volumes equalizes their pressures. Bulk boiling of the sodium liquid as well as bulk condensation of the sodium vapor are considered in each volume, both depending upon the effects of heat sources and sinks of the two volumes. Typical containment volumes are: (a) the reactor cavity and (b) the three HTS cells combined.

Another modification provides up to 40 heat-conducting structures which can be used as heat sources or sinks with respect to the vapor and liquid regions. The thermal behavior of these structures are described by the one-dimensional, multi-layer, transient heat conduction equation. Typical heat conducting structures are the reactor vessel wall, guard vessel, radial blast shield, bottom blast shield (all of which are in the reactor cavity) the cavity roof, walls, and floor; and the HTS cells' roofs, walls, and floors.

Two basic containment design basis accident cases have been analyzed with the CONRAD code. These two cases assumed that a sodium-out type disassembly accident has occurred. Data obtained will be reflected in the Preliminary Safety Analysis and keV is now in preparation.

2. Post DBA Containment Analysis

G. L. Fox

A digital computer code has been completed for computation of long-term pressure and temperature transients following a hypothetical core meltdown accident. It differs from the CONRAD code described in that it does not treat sodium boiling or condensation. It is a survey code designed for fairly rapid solution of long-time transients. It computes conditions in the containment sphere as well as the reactor cavity-HTS vault complex. The model applies to a hypothetical case wherein all sodium is expelled from the reactor cavity. There is no positive way of showing how much sodium remains in the presence of core debris following a postulated core disassembly accident. The limiting case of no sodium present was evaluated to determine whether or not containment design pressure requirements would be more severe for that condition than for the more likely case where some liquid sodium remains with the disassembled core.

A pictorial description of the computer model is shown in Figure 10.1. The reactor cavity (Cell No. 1) is connected to the equipment cells (Cell No. 2) by a free passage so both are at the same pressure. The reactor fuel decay heat is liberated in Cell No. 1, which is transferred to the wall, floor, and to the internal metal structures within the reactor cavity. Both cells No. 1 and No. 2 have 20% of the decay power proportioned according to the gas mass distribution corresponding to the assumption that 20% of the decay heat is present in the gas phase. Cell No. 2 has concrete walls which allow nonlinear transient temperature distributions. The leakage of gases from Cells No. 1 and No. 2 to the reactor sphere (Cell No. 3) has been calculated based on 10% of the gas mass per day for a design pressure difference of 30 psi. Cell No. 3 is heated by conduction through the roof of Cell No. 2 and by fission products which leak from Cells No. 1 and No. 2.

Heat is produced in the reactor cavity by the fission product decay in a layer of molten fuel on the floor. The heat may be radiated to the steel-gas system within the cavity and conducted down through the floor. The steel-gas structure is represented by a single node which may convect and radiate heat to the cavity walls. The walls consist of 4 in. of insulation which is cooled at the outer surface to a specified sink temperature. The cavity heat transfer model may change as desired without affecting the structure of the rest of the computer program.

FIGURE 10.1. Pictorial Computer Model Dry DBA

It was concluded that this no-sodium or "dry case" causes containment pressures which do not exceed the sodium-present or "wet case." The initial pressure peak resulting from the DBA energy release should be essentially the same for both models. The second peak, which results from core decay energy disposition, is somewhat lower for the dry case as compared to the wet case.

3. Consequences of Sodium Fires

P. R. Shire

A series of time dependent heat transfer and thermodynamic analyses have been completed for hypothetical accidents in the reactor heat transport system (HTS) cells, the main containment sphere, and for both combined. These are loss-of-coolant accidents resulting from either the nuclear design basis accident (DBA) or to non-nuclear causes. Leakage from containment spaces was excluded; this assumption results in conservative conditions for obtaining maximum pressure and temperature versus time relationships.

Two methods of calculating these accident cases have been utilized, both are transient heat transfer computer codes: TAP-3F⁽¹⁾ and SOFIRE.⁽²⁾ The TAP code has been most useful for computing the long term effects (to 5000 hours) while the SOFIRE code performs a more sophisticated analysis of sodium pool burning in the early states of the accident. The cases computed are listed below:

- DBA Loss-of-coolant in HTS Cell (TAP Code)
- Dropped Fuel Element in Main Containment Sphere (TAP Code)

1. B. J. Dray. Thermal Analyzer Program (TAP-3F) Users Manual, NAA-SR-TDR-11812, Atomics International, Canoga Park, California, January 11, 1966.
2. R. P. Johnson. SOFIRE I-B User Package, TI-707-14-001, Atomics International, Canoga Park, California, February 1969.

- DBA Loss-of-coolant in Three HTS Cells Open to Sphere (SOFIRE Code).
- Non-nuclear Loss-of-coolant Accidents Associated with the Sphere (SOFIRE Code)

Data from these studies will be employed in preparation of the PSAR.

B. ANALYSIS OF FFTF PLANT PROTECTION SYSTEM (7N82)

1. Fault Tree Analysis of Plant Protective System

R. J. Schroder

New fault trees have been developed for the PPS to represent failure to achieve a minimum degree of scram success for the specific accidents considered. Minimal rod control for successful scrams for most accidents used in these fault trees are included in Table I.

TABLE 10.I. *Rods Necessary for Successful Scram*

<u>Type</u>	<u>Minimum Rods for Successful Scram</u>
A	2 primary rods
B	2 primary or all three in-core shim rods
C	1 primary rod
D	1 primary or two in-core shim rods
E	1 primary or one in-core shim or four peripheral rods

The scram portion of the Plant Protective System can be schematically represented as follows:

(52 protective inputs)

In all, 22 parameters are monitored. Several of the parameters are measured at more than one point, hence a total of 52 protective inputs are provided. Each protective input requires three or four sensor channels connected to a channel

approximately weekly for the PPS. It was concluded further that the major contribution to subsystem unavailability is the instrument sensors and signal conditioning electronics. Thus, it is required that the low-level neutron monitors will be backed up by the intermediate level neutron monitors, and these in turn will be backed up by the power level neutron monitors.

The range of values for unavailability of each sensor subsystem was determined by considering three testing intervals; 1500 hr, 168 hr, and 8 hr, which correspond to testing each reactor refueling cycle, each week, or each shift, respectively. The range of component failure rates was determined from a "nominal" value, which is a best estimate for the equipment at this time and a pessimistic value; but an optimistic value where the latter two are at least one order of magnitude removed from the nominal value. The nominal failure rate is representative of values that were found in the references for similar equipment. The results of the nominal failure rate, for the three test intervals are given in Table 10.II.

Complete results of this study are documented in a report BNWL-1201 now in publication.

TABLE 10.II. *Plant Protection System Unavailability
Primary or Secondary Rod Scram (at
nominal failure rate)*

	Time Between Test, hr		
	8	168	1500
1. Low Level Neutron Flux (Note 1)	5×10^{-7}	2×10^{-4}	2×10^{-2}
2. Low Level Neutron Flux Period	6×10^{-7}	3×10^{-4}	2×10^{-2}
3. Intermediate Level Neutron Flux	4×10^{-7}	2×10^{-4}	1×10^{-2}
4. Intermediate Level Neutron Flux Period	5×10^{-7}	2×10^{-4}	2×10^{-2}
5. Power Level Neutron Flux	2×10^{-9}	3×10^{-6}	2×10^{-3}
6. Power Level Neutron Flux Rate-of-Change	2×10^{-9}	4×10^{-6}	2×10^{-3}
7. Bulk Outlet Temperature	2×10^{-9}	9×10^{-7}	7×10^{-5}
8. Primary Loop IHX Outlet Temperature (Note 2)	2×10^{-7}	1×10^{-6}	3×10^{-4}
9. Reactor Flux-Flow (Note 3)	5×10^{-9}	2×10^{-5}	5×10^{-3}
10. Reactor Vessel Coolant Level (Note 4)	2×10^{-9}	1×10^{-6}	3×10^{-4}
11. Electrical Power Supply	8×10^{-9}	4×10^{-6}	3×10^{-4}
12. Seismoscope	2×10^{-9}	1×10^{-6}	2×10^{-4}
13. Primary Loop Coolant Flow (Note 5)	2×10^{-9}	1×10^{-6}	2×10^{-4}
14. Containment System Actuation	1×10^{-8}	8×10^{-6}	2×10^{-3}

NOTES

1. Low-level neutron flux: Results are the same for either in-vessel or out-of-vessel subsystem.
2. Primary loop IHX outlet temperature: Results are for one IHX outlet temperature subsystem. Results are the same for closed loop IHX outlet temperature, and closed and open-loop outlet temperature subsystems.
3. Reactor flux/flow: Results are the same for closed loop flux/flow subsystem.
4. Reactor vessel coolant level: Results are the same for closed loop surge tank level subsystem.
5. Primary loop coolant flow: Results are for one loop coolant flow subsystem.

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