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JULY 1968

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**FAST FLUX TEST FACILITY
BACKUP DESIGN PROJECT
THIRD QUARTERLY REPORT
DECEMBER 1967 - FEBRUARY 1968**

D.R. RILEY

U.S. ATOMIC ENERGY COMMISSION
CONTRACT NUMBER BDR-341
BETWEEN PACIFIC NORTHWEST LABORATORY
AND THE GENERAL ELECTRIC COMPANY

ADVANCED PRODUCTS OPERATION

GENERAL ELECTRIC

SUNNYVALE, CALIFORNIA

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THIRD QUARTERLY REPORT
DECEMBER 1, 1967—FEBRUARY 29, 1968

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SECTION I

INTRODUCTION AND SUMMARY

This quarterly report documents the work conducted on the FFTF Backup Design on Contract BDR-341 between Pacific Northwest Laboratory and the General Electric Company during the period December 1, 1967 through February 29, 1968.

The objective of the FFTF Backup Design is to provide the capability of replacing the reference FFTF split conical core design at any point in its evolution up to and including initial operation, with an open lattice vertical core and associated handling system, instrumentation, control rod and top head configuration, making use of the reference design vessel, primary piped coolant system, hot cell configuration, auxiliaries, containment arrangement and central control system.

The major areas of design effort involved delineating interface conditions between the FFTF Reference and Backup Designs, supplying component information for a preliminary cost estimate, describing the specific engineering development work required for the backup design and providing more detailed design work for various reactor components located within the vessel. An optimization study of the in-core location of the six closed test loops with respect to the ease of refueling and providing instrumentation for the open loops and the driver fuel assemblies has resulted in a modified core layout pattern.

Physics analysis of the FFTF Backup design is based on a one-zone, 3-foot-high core composed of 139 in-core hexagonal channels, of which 126 are driver fuel, 3 are in-core control locations, and 10 are closed loop positions. An equilibrium fuel cycle is assumed with an average discharge exposure of 70,000 MWd/Te and a refueling interval of 67.5 full power days, resulting in a burnup

reactivity swing of 2.3 percent $\Delta k/k$. A peak linear power density upper limit of 13.6 kW/ft is met by utilizing partial fuel shuffling.

The presence of reflector control at the beginning of an equilibrium burnup cycle, to compensate for burnup reactivity peaks, the radial power distribution giving a peak neutron flux of 0.99×10^{16} n/cm²-sec. The peak neutron flux at the end of cycle with all control removed is 0.95×10^{16} n/cm²-sec.

Temperature reactivity effects were calculated for the beginning and end of a burnup cycle. Although the individual coefficients changed, it was found that the net cold to hot-operating reactivity effect (-1.75 percent $\Delta k/k$) did not change.

Preliminary diffusion theory studies of the reflector control system indicate the total worth to be \$16 and \$28 for natural B₄C and B₄¹⁰C respectively.

The comparison of a core design using a 0.25-inch fuel pin diameter (rather than 0.21-inch) results in a peak neutron flux 14 percent lower and a reflector control worth 30 percent lower than the core design using 0.21-inch fuel.

A tentative outline of proposed critical experiments to be performed for the backup FFTF design in the PNL Phase-B critical program is presented. The major experiments to be run are critical mass, radial fission traverses, sodium loss reactivity effect, Doppler effect, and reflector control worth.

A preliminary study indicates that measurements of the fission product decay energy from the fast fission of Pu-239 are

as yet quite limited. Comparisons of these limited measurements to the results of extensive thermal neutron fission of U-235 decay heat measurements indicate good agreement. Accordingly, fission product decay curves based on measurements of U-235 thermal fission are recommended for estimating the decay power versus time after reactor shutdown from Pu-239 fast fission. This recommendation applies for preliminary FFTF design work only. Decay curves and a table illustrate these results.

A study was initiated to determine the optimum composition and spatial distribution of shielding above the core. Results from this study indicate that it is both feasible and acceptable to remove all B₄C from the upper axial neutron shield and to reduce the shield thickness from 4.1 to 3 feet. This is a preliminary conclusion subject to review by the reactor vessel and internals design group. The conclusion is based on approximate calculational results which must be verified by more exact methods.

Analyses have been performed on four conceptual arrangements of the head region to establish feasibility of adequate cooling of this region with forced gas circulation, and select one or two arrangements for more detailed design and analysis.

Thermal-hydraulic evaluations were made of four reflector control rod configurations to determine the temperature limits on the poison and cladding that can be met using boron carbide as the poison material. For a 19-pin array, peak temperatures as low as 1250°F appear to be attainable for natural boron carbide.

Parametric studies on pressure drop through the core region have been conducted to guide the core design and permit comparisons with the PNL method of calculation to be made.

The short term (20-30 second) sodium temperature response to normal transients was evaluated. Parameters varied included pump coastdown characteristics, outlet mixing, pump coastdown delay time, and time between scram signal and start of rod drop. Transients analyzed included loss-of-site power, over-power scram, high temperature scram, and pump startup in standby loop. Considerably lower temperature gradients at the core, channel, and vessel outlets were obtained for the FERMI pump coastdown characteristics than for the EBR-II characteristics. The gradients at the vessel outlet were influenced strongly by the outlet mixing time constant.

The IHX model, which uses six nodes and a central difference approximation to the differential equations, was tested for both the steady-state and transient solutions. The model is accurate at steady state for near equal flow rates in the primary and secondary. However, significant deviations from the correct solution at steady state were observed when the primary and secondary flow rates differed by a factor of two. The transient solution was stable for all transients tested, including severe step changes in flow rate.

Fuel and clad damage limits have been tentatively established and the fuel and clad temperature levels set for the design accidents identified.

Abnormal transients for which the sodium-temperature responses are being evaluated include scram with pumps running, loss of flow in one loop without scram, coast-down of all pumps without scram, and 10¢/sec ramp without scram.

A review was made of the design requirements to accommodate the DBA with emphasis on factors contributing to a determination of the vessel head hold down bolts.

SECTION II

PLANT PERFORMANCE

Figure 2-1 is a pictorial of the FFTF Backup Design. The Plant Performance Data shown in Table 2-1 gives the latest design parameters for the reference design.

NOTE: Those items marked with an asterisk (*) have been changed from the values reported in the previous quarterly report. ⁽¹⁾

TABLE 2-1
PLANT DATA SHEET

2.1 TEST LOOPS

Closed Loops	6 (10 core positions)*
Location	1 central (double channel)*
	5 mid-core (3 double channel)*
Open Loops	2
Flux	
Peak Level	1.0×10^{16} *
Fraction above 0.1 MeV	0.66
Loop Cells	
Number	6
Size (ft ³)	8000

Fuel Channel

Pins/Channel	127
Dimension (inches)	
Across Flats	3.365*
Material	Inconel-800
Fuel Pins	
Diameter (inch)	0.210 o.d.
Clad Material	316 Stainless Steel
Clad Thickness (inch)	0.015
P/D (inches)	1.286
Fuel Cycle (days)	90 (Preliminary - based upon 0.8 plant factor and 70,000 MWd/T, 20% of core elements are replaced each cycle.)

2.2 REACTOR

2.2.1 Core

Type	Open Lattice, vertical
Power (MWt)	400
Dimensions (inches)	
Length	36
Equivalent Diameter	42
Composition	
Fuel	PuO ₂ -UO ₂
Structure	Stainless Steel
Coolant	Sodium
Core Channels	
Fuel	126*
Closed Loops	10*
Open Loops	2
Back-up Control	3
Total	141*

2.2.2 Reflector

Dimension (inches)	
Top	12.0
Bottom	12.0
Radial	10.5
Material	Nickel

2.2.3 Control

Primary Control	
Location	Reflector first row
Number	
Single Element	11
Double Element	16
Total	27
Worth	
Single Element	\$0.7
Double Element	\$1.1

FIGURE 2-1. FFTF BACKUP DESIGN REFUELING AND REACTOR SERVICING SYSTEM

Total Worth in Reflector	\$20 to \$30	Grid Plate	<u>Design</u> 1000°	<u>Operation</u> 700°
Material	B ₄ C (natural or partially enriched)	Reactor Vessel	1200°	1000°
Drive	General Electric compact mechanical drive	Vessel Head	1000°	200°
<u>Back-Up Control</u>		<u>2.2.8 Power Distribution</u>		
Location	In-core	Radial Flux Peaking	1.4	
Number	3	Axial Flux Peaking	1.2	
Worth		Over-Power Allowance	1.2	
Each	\$1 - \$2.7	Fuel Burnup	1.05	
Total	\$3 - \$8	Fuel Non-Uniformity	1.04	
Material	B ₄ C			
Drive	General Electric compact drive			
<u>2.2.4 Vessel</u>		<u>2.2.9 Hot Channel Factor</u>		
Diameter (feet)	16.0	Statistical	1.17	
Thickness (inch)	0.75			
Material	Stainless Steel Type 304	<u>2.3 SYSTEM</u>		
<u>2.2.5 Coolant</u>		<u>2.3.1 Primary Coolant</u>		
Flow (lb/h)	15.1×10^6	Number of Loops	3	
Inlet Temperature	700° F	Rating (MWt/ea)	133	
Outlet Temperature	1000° F	Piping		
Core ΔP (nominal plenum to plenum)	100 psi	Material	304 Stainless Steel	
<u>2.2.6 Cover Gas</u>		Size (inches)		
Type	Argon	Reactor Outlet	20	
Operating Pressure (inches, water gage)	±10	Reactor Inlet	18	
<u>2.2.7 Temperature (°F)</u>		Heat Exchanger Rating (MWt/loop)	133	
Peak Channel	<u>100% Power</u>	Coolant Loop Unavailability	7.8%	
Fuel - maximum	4600°*	Temperature (°F)		
Fuel surface - maximum	1875°*	Design	1200	
Clad - maximum (i.d.) (hot spot)	1180°*	Operation	1000	
Sodium - maximum (hot stream)	1125°*	<u>2.3.2 Containment Vessel</u>		
		Type	Cylindrical	
		Diameter (feet)	135	
		Thickness (inches)	1.1	
		<u>2.3.3 Emergency Coolant</u>		
		Type	NaK	
		Heat Load (MWt)	4	

2.3.4 Vessel Cavity

Gas	Nitrogen
Pressure (inches, water gage)	±10
Liner	Steel

2.3.5 Equipment Cell

Gas	Nitrogen
Pressure (inches, water gage)	±10

2.3.6 Loop Cell

Gas	Nitrogen
Pressure (inches, water gage)	±10
Floor Liner	Steel

2.4 HOT CELL

2.4.1 Size (Feet)

Length	84*
Width	30*
Height	78

2.4.2 Transfer Cell

Maximum Size Transfer (feet)	
Diameter	4*
Length	43*

2.4.3 Fuel Handling Method

Closed, finned transfer thimble, sodium filled, natural circulation cooling.

2.4.4 Fuel Storage Method

In-cell, sodium filled decay pools with natural circulation. Secondary cooling by forced circulation.

2.4.5 Refueling Cell Atmosphere

Atmosphere	Argon
Design Temperature - accident (° F)	250
Design Operating Temperature (° F)	150*
Design Pressure	
Estimated Maximum (psig)	10
Estimated Minimum (inches, water gage)	-6
Operating Pressure (inches, water gage)	
High	+2 ±1*
Low	-2 ±1*
Leakage (est. maximum)	1% cell vol/day at design pressure

2.4.6 Refueling Time (Hours)

Shutdown - Preparation	8
Reactor Servicing	44
Driver Refueling	42
Control Rod Replacement	8
Startup Preparation	8
Total	110

2.4.7 Major Equipment

Bridge crane mounted telescoping tube hoists with fuel grapple mechanisms	2
Bridge crane mounted general purpose manipulators	2
Master-Slave Manipulators	3
Traveling Hoist	100 ton*
Bridge Crane	5/20 ton*
Decay and Storage Tanks	7*

2.5 SAFEGUARDS

2.5.1 Containment Double2.5.2 DBA Energy Release 1500
Estimated Maximum
(MW-sec)2.5.3 Credible Accidents

Flow Coastdown

Loss of Single Coolant
LoopPiping Leakage (including
guillotine pipe failure)Single Fuel Bundle
MeltdownControl Rod With-
drawalRefueling Bundle
Drop-In

2.6 NUCLEAR

2.6.1 Core CompositionFuel Channel Volume
Fraction

Fuel 0.326*

Steel 0.154*

Incoloy-800 0.114*

Sodium 0.406*

Experimental Loop
Volume Fraction

Steel 0.487*

Sodium 0.464*

Fuel 0.149*

Back-up Control Rod
Volume

Steel 0.33

Sodium 0.67

2.6.2 Mid-Cycle Fuel Composition
(Atom %)

Uranium 72.6*

Plutonium 23.7*

Fission Products 3.7

Plutonium Isotopic Concentration (%)

Pu-239 88.0*

Pu-240 12.0*

Pu-241 0.5*

Uranium Isotopic Concentration (%)

U-235 0.2

U-238 99.8

2.6.3 Reactivity

Doppler

 $T \frac{dk}{dT}$ (sodium in) -0.006 $T \frac{dk}{dT}$ (sodium out) -0.004

Na Void

Maximum
Positive (\$) +1.5

Total Core (\$) -4.0

Maximum Fuel
Bundle Worth (\$) +2.2Fuel Meltdown
(top 1/3 into
mid 1/3) (\$) +0.52.6.4 Fuel CycleDischarge Burnup
(MWd/kg) 70

Number of Batches 5

Burnup/Cycle
(MWd/kg) 14Fuel Cycle Period
(days) 90Operating Cycle
(days) 722.6.5 MeV/Fission 215Fraction energy
absorbed in fuel 0.91

2.7 RADIATION LEVELS

2.7.1 Dose Rate During Operation
(mRem/h)

Operating Floor 0.2*

Hot Cell Floor 50

2.7.2 Equipment Cells

From Na²² (in
working area) (R/h) 1

2.7.3 Neutron Exposure (nvt > 1 MeV,
20 years)

Vessel Wall Core
Midplane 7×10^{17}

Radial Blast Shield 5×10^{17}
Vessel Cavity Wall 7×10^{16}
Core Support Plate 8×10^{19}
Vessel Head 3×10^{10}

SECTION III

PLANT SYSTEMS

3.1 GENERAL

The plant systems design effort has been directed toward providing system descriptions and design requirements sufficient to define an effective system interface between the reference and backup reactor systems. Functional conceptual specifications have been prepared for the major plant systems peripheral to the reactor which have interface with the reference plant. These specifications represent the initial point of reference from which a compatible interface will evolve. The system specifications provide the following information:

1. System Functional Requirements
2. System Design Criteria
3. System Design Requirements
4. System Description
5. System Operation, Maintenance, and Repair
6. System Availability, Cost Effectiveness
7. System Interface Conditions

Conceptual design work on plant systems has been performed to provide the design information and system requirements necessary to ensure that the backup design reactor system meets the overall broad

objective of the FFTF program. With the completion of this phase of work, an effective interface is defined between the reference and backup designs. Further detail system work on this backup concept will be confined to providing information and establishing design detail within the framework of the interface envelope.

3.2 SPECIFICATIONS

The following specifications have been prepared in draft form, and will be submitted in the interface report:

- 0.2 Reactor Plant Containment System
- 0.3 Building Access and Radiation Control
- 0.6 Plant Availability
 - 2.1.1 Main Primary Sodium System
 - 2.3.0 Safety Tank Cooling System
- 4.0 Experimental Facilities
 - 5.1.1 Argon Supply
 - 5.1.2 Argon Circulating System for Refueling Cell
 - 5.1.3 Argon Purification System
 - 5.2.1 Nitrogen Supply
 - 5.2.2 Nitrogen Cooling System
 - 7.2.0 Emergency Power Supply
- 0.8.0 Reactor Radioactive Gas Waste System

SECTION IV

REACTOR VESSEL AND INTERNALS

4.1 GENERAL

Major design effort was involved with delineation of interface conditions with the reference design and describing the engineering development work specifically required for the backup design. This work is summarized in this report and will be reported under separate cover. A study of the in-core locations of the six closed test loops indicated that a slight change in test loop locations would improve refueling equipment accessibility to the test and driver fuel and provide increased flux in the assemblies that were located at the largest radius. This and other work reported are a part of the continuing design effort directed toward providing additional testing flexibility and improved driver fuel instrumentation.

Nuclear analyses, reported elsewhere in this report, confirm the desirability of the compact vertical core design. These analyses continue to indicate that sufficient nuclear control is available in the reflector. Locating the control rods in the reflector has provided more test volume in the core and decreases outage time for refueling and test rearrangement activities.

Utilization of more than one driver fuel assembly location per closed test loop assembly has been considered, as reported last quarter. At present* the nuclear analyses and reactor layouts are being based on double channel closed loops located in the

center and in three of the five mid-radius locations. Due to the flexibility of the open lattice design either open or closed type multi-channel loops can be installed, provided nuclear worth conditions are accommodated.

Design improvements are reported in the following sections. These include a revised core clamping scheme, a more detailed driver fuel instrumentation rack, a fuel assembly incorporating a removable orifice and some estimates on the life expectancy of the B_4C control rods.

Modified elevation and plan sections of the FFTF Backup Design appear on Figures 4-1 and 4-2.

4.2 HEAD, VESSEL, AND SUPPORT SKIRT

Requirements for the head and vessel penetrations were reviewed in conjunction with the reference design interface study. Additions and modifications are described below. The basic head and vessel design remained the same during this quarter.

Heat transfer and head cooling system analyses have been initiated but are very preliminary due to the dearth of information on the penetrations and plug designs. Argon gas cooling appears to be adequate, however NaK (liquid) cooling is being considered as an alternate for plug cooling. The objective of this work is to cool the vessel and all heads and plugs to about 200° F about 30 inches (one vessel wave length) above the "high," 1200° F, sodium level. The upper part of the plug will be maintained at as uniform a temperature as practical and will contain

*While this report was being prepared the designers of the Reference and Backup cores adopted a set of rules for establishing the nuclear worth and number of in-core loops. Future reports will reflect the adopted rules.

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FIGURE 4-1. ELEVATION VIEW OF BACKUP FFTF REACTOR

FIGURE 4-2. PLAN VIEW OF BACKUP FFTF REACTOR

the major head and plug structure and cover gas seals.

A comparison between the updated vessel and internal elevation drawing, Figure 4-1, and the elevation in last quarter's report⁽¹⁾ indicate the following modifications:

- a. Main outlet piping nozzles were lowered about 30 inches to provide about two feet submergence when the sodium level is lowered for inspection of the top of the fuel and reflector channels.
- b. Reduced the number of radial branch piping and instrument lead penetrations from 16 to 12. (The reduced number are not shown on Figure 4-2.) In addition to the advantage of reducing the number of vessel nozzles this change also provides for a radial path for each set of the branch piping from the six closed test loop tubes to the vessel penetration. The remaining six penetrations for instrumentation leads are believed to be adequate for conceivable experimental instrumentation requirements.
- c. There have been no changes in the branch piping for the central closed test loop, however, a different section has been shown on (the left side of) Figure 4-1 than on previous elevation drawings. All of the elevation drawings have shown two of the following three types of vessel penetrations rotated to the plane of the figure: main sodium inlet piping, the main sodium outlet piping, and the central closed loop branch piping.
- d. Many of the internal reactor components such as the reactor support and high pressure plenum have been updated. These will be described in the following sections.

Comparative elevations of various in-vessel components are summarized in Table 4-1. While all parts are not as interchangeable as is desired in these conceptual drawings, the core location is in about the same location. The significant differences are in the location of the main coolant inlet and outlet nozzles. The inlet nozzles reflect two different schemes of introducing the coolant to the lower end of the vessel. (Both reactor designs can utilize either scheme.) The outlet elevation in the backup design is required to be considerably lower than in the reference design in order to maintain continuity of coolant circulation when the sodium level is reduced for inspection of the top of the fuel and reflector channels or any other above channel experimental or control rod equipment.

The vessel support skirt is a relatively thin cylindrical member that transmits the weight of the vessel, sodium internals, pipe reactions, head and control drives to the ring girder foundation. The dead weight on the vessel support skirt is estimated to be:

	(Pounds)
Vessel	78,000
Sodium	318,000
Internals	425,000
Head and Control Drives	230,000
Stretch Bolts (snug)	<u>10,000</u>
Total Load	1,061,000

The load on the 19-foot-diameter skirt then is about equal to 1500 psi per inch of thickness. For short periods of time during the Design Basis Accident, however, the downward load on the skirt could be many times larger. This occurs when the head and vessel separate (when the hold-down bolts stretch). The added load on the support will be equal to the effective pressure in the vessel times the vessel area. Thus shortly after the vessel starts to deform

TABLE 4-1
COMPARATIVE REACTOR INTERNAL ELEVATIONS
 (Feet-Inches)

	PNL Reference SK · 3 · 12574 (Annular)	PNL Reference SK · 3 · 12876 (8 Spiral Inlet Pipes)	General Electric Backup 277 E 127 (Figure 4-1)
Vessel Foundation to Core Support	37-10	31-8	30-9
Overall Vessel Length (including Head)	46-0	39-0	40-0
Vessel Support to Centerline of Inlet Nozzle	18-4	11-2	32-6
Core Support to Centerline of Inlet Nozzle	19-6	20-6	1-9
Core Support to Centerline of Outlet Nozzle	26-6	21-0	11-0
Vessel Support to Centerline of Outlet Nozzle	11-4	10-8	19-9
Refueling Cell Floor to Bottom of Core	33-4	27-2	32-0
Refueling Cell Floor to Core Support Plate	37-10	31-8	35-0
Refueling Cell Floor to Centerline of Inlet Nozzle	18-4	11-2	36-9
Refueling Cell Floor to Centerline of Outlet Nozzle	11-4	10-8	24-0
Sodium Depth	~38-0	~31-8	~32-0

(equivalent to about 450 psi internal steady-state pressure) and the head has separated from the vessel the skirt load, for a short time, may be as high as 13 million pounds, or over ten times the dead weight load. Even under these conditions the skirt need not be over 1 inch thick. No clamping effect from the head-vessel seal bolts, which would decrease the skirt load, has been assumed in these calculations. Seismic loadings and potential differential pressures across the skirt also have not been included due to the estimated negligibly small additions to the DBA design loadings. The structural design of the skirt and head are being based primarily on integrity during DBA loading.

4.3 CORE SUPPORT AND INLET PLENUM

The core support and inlet plenum provides the support for the reactor internals and flow distribution to the core and reflector. As shown on Figure 4-1, the design of this structure has been improved. The support and plenum consists of a fairly permanent component that attaches to the support brackets on the vessel wall and to the extension pieces of the coolant inlet piping and a removable center section that contains the upper and lower grid plates. This removable feature provides relatively inexpensive replacement if the core lattice design is modified.

The support structure primarily carries the weight of the reactor components. Horizontal loads are relatively low and result from seismic loading and due to pipe reaction

loads that are transmitted through the vessel. Design loads on the support structure under operating and abnormal conditions are itemized in Table 4-2. They are also

TABLE 4-2
DESIGN LOADS ON THE CORE SUPPORT STRUCTURE

1. Normal Operating Conditions

Fuel elements and closed loops (lbs.)	41,400
Reflector elements (lbs.)	59,000
Control rods (lbs.)	18,000
Core clamping ring (lbs.)	400
Radial shield rods (lbs.)	257,000
Support structure (lbs.)	37,000
Cylindrical container (lbs.)	<u>12,000</u>
Total vertical load	424,800 pounds
Total horizontal load	negligible

2. Abnormal or Casualty Conditions (estimated)

Seismic	* vertical * horizontal
Weight of refueling plug and instrument structure (total support) (lbs.)	20,000
Defective control drives (no retarders)	(not yet estimated)
Temperature transient including pipe reactions (cold loop start-up accident)	300° F ΔT between vessel and inlet plenum (horizontal load)

*Consistent with category I of the reference reactor design criteria and in accordance with the intent of AEC 10 CFR 50, General Design Criteria for Nuclear Power Plant Construction Permits.

described graphically on the Core Support sketches shown on Figures 4-3 and 4-4. Since it is not likely that all the loads will act simultaneously, the following design criteria for the load conditions are being considered:

a. The design level seismic loading shall be assumed to occur under any normal operating condition and following an accidental thermal transient but not during the transient.

b. The weight of the refueling plug and instrument structure need be considered only during its removal or replacement (and under refueling temperature and flow conditions). No scram or transient loads are considered to occur at this time.

c. About one-third of the weight of the control rods will be considered under normal conditions; full weight is considered when the rods are disengaged from the drives (refueling conditions).

FIGURE 4-3. CORE SUPPORT STRUCTURE - PLAN

FIGURE 4-4. CORE SUPPORT STRUCTURE AND NORMAL LOADS

The core support structure is centered by the attachment to the vessel. The top surface of the core support plate is about 35 feet below the refueling cell floor and must be horizontal. The vessel lugs include provisions to level this structure during its installation. The design of the vessel and internals allows the entire support structure to be replaced or installed after the reactor has been operated with a different core design. Removal and installation is presumed to occur during a major shutdown and after the sodium has been drained from the vessel. Fixtures, brackets, etc., which overlap the spider supports are removable or do not interfere with the removal operation. The vessel design permits the inspection of critical welds around the vessel inlet piping in the plenum area.

4.4 CORE CLAMPING

The objective of the core tightening system is to provide a means for minimizing the spaces between the core and reflector channels after they have been loosened for refueling or test loop replacement. The core tightening system requirements include that the mechanism operates in a relatively short time, provides a minimum interference with refueling or experiment handling (about 1/3-inch total motion) and provides some positive assurance that the core and reflector channels will not move closer together during operation.

The core clamp (described in last year's tank type vessel design), ⁽²⁾ requires a core inlet to outlet ΔT for the clamping action. Under the ΔT condition the thermal growth above the fuel in the channels will be greater than the thermal growth in the support plate. The results of the differential thermal expansion is to eliminate the clearances between channels. In order for the segmented ring, surrounding the core, to provide radial support, the ring would have

to be operated at or near inlet temperature or be fabricated from some material that has a lower coefficient of expansion than the Incoloy 800 channels or both. This system has the advantage of simplicity and fewer moving parts and the disadvantage of not being in effect during startup or low power level operation.

An attractive core tightening system for the Backup Design consists of a row of special expanding elements between the reflector and shield regions. These elements are designed to apply an inward radial force on the reflector and core. The clamping elements are backed up radially by a circular support ring. This core tightening concept is a combination of the expanding mechanism on each core assembly and the contracting circumferential band concepts that were reviewed last quarter. In Figure 4-2 the new concept is shown as the nearly completely drawn hex rod pattern surrounding the reflector region (the expanding channels) and the circumferential ring with adapter pieces to fit the hex channels. In Figure 4-1 the circumferential rings and vertical supports from the shield cylinder are shown. The rings (and therefore the clamping action) are located at three vertical locations: (1) opposite the axial reflector region of the fuel elements, (2) near the top of the hex section of the fuel assembly, and (3) midway between. More detailed analyses, as provided for the Southwest Experimental Fast Oxide Reactor, will determine the necessity of the mid-span clamp location. Below the core, radial positioning of each subassembly is provided by the core support plate and hydraulic hold-down structure. Radial clamping of this location does not appear to be necessary.

The expanding assemblies are conceived to be identical to the reflector rods except for the addition of the operating

mechanism. Since the mechanisms are relatively far from the core, their structure should not affect the core neutronics. Many expanding mechanisms and operator concepts are being reviewed. At present a hydraulically (sodium) actuated mechanical toggle mechanism concept is preferred. The operating fluid connections can be supplied at either the top of the subassembly or plugged into a socket connection in the nose piece of the assembly. The operating fluid is reactor coolant purity sodium and provides positive or negative pressure to lock or unlock the mechanical toggle. The expanding action is provided by a cam operated by the toggle. This mechanical system provides multi-directional channel expansion toward the circumference ring and toward the neighboring reflector channels. This force deflects the reflector toward the center of the core, thus minimizing inter-channel gaps. In the locked position, the hydraulic pressure can be relieved without loosening the core. To release the clamping action, the toggle is moved in the reverse direction by the operator. Mechanical operation in all three locations of each assembly is provided by one operator. There will be a multiplicity of hydraulic systems for high operating availability. The hydraulic operator can be overridden by mechanical operators located on the top of each clamping assembly. Access to the over-ride operators is available after the refueling plug is removed. Thermal expansion of the core and reflector assemblies is accommodated by spring loading the cam action in the mechanisms.

4.5 DRIVER FUEL INSTRUMENTATION

Effort on driver fuel instrumentation was concentrated in the area of developing a system for removing and replacing defective sensors. In the present concept all the instruments associated with an individual fuel assembly are contained in an instrument package. This unitized package facilitates remote maintenance and handling of the instrumentation and protects the sensors during unit

and plug installation. The concept being considered combines the leads for several instrument packages for a common head penetration. This simplifies the head design and reduces the number of connectors on the refueling plug. (See Figure 4-5.)

The instrument packages associated with each head penetration will be fabricated and handled as a single unit. The units are designed to provide support and protection for the sensors and their leads and are fabricated and tested prior to their assembly with the instrument structure. A sensor protected in this manner is not likely to be damaged during its assembly to the instrument structure or by normal operating conditions.

The instrument units are designed to be removed and replaced on the instrument structure while it is hanging in the cell during refueling activities. The instrument units are sealed and supported from the head penetration and supported laterally at several elevations on the instrument structure. The instrument units are removed by disconnecting the head connection and lowering the unit out the bottom of the plug. The instrument unit must first be raised to clear the instrument packages from the hold-down grid plate and tilted outwards before it can be lowered out through the bottom of the plug. This will require a flexible section in the structure of the instrument unit. An alternate method is to remove the entire grid plate and lower the instrument units directly.

Another concept considered each instrument package having its own head penetration and connector. This method might reduce the time required to replace defective instruments (and their availability is designed to be high), but due to the clutter of over a hundred conduits, it is not obvious there is any advantage to this method.

FIGURE 4-5. REFUELING PLUG AND INSTRUMENT RACK

SCALE: 1 1/2" = 12"

4.6 DRIVER FUEL ELEMENT

Four core designs were reported last quarter which would be used in a parameter study of the backup design vertical core. These were for an active fuel length of only 33 inches. Early this quarter the decision was made to increase the active fuel length to 36 inches to increase the effectiveness of the reflector control rods. Dimensional studies were conducted during the quarter for the longer assembly. As a result, the

fuel pin parameters presented in Table 4-3 have been established for the backup design.

A concept to allow remote changing of the fuel channel orifice in the refueling cell has been incorporated in Section B of the driver fuel layout. (See Figure 4-6.) The design can accommodate either a sharp-edged orifice or a perforated cone orifice. The latter has the advantage of acting as a non-clogging design. This section can also incorporate an in-line strainer, if necessary, although it is not recommended at this time.

TABLE 4-3

FUEL ASSEMBLY PARAMETERS*

Cladding:	
i. d. (inch)	0.180 ± 0.001
Wall thickness (inch)	0.015 ± 0.001
o. d. (nominal) (inch)	0.210
Material	Type 316 Stainless Steel
Fuel Pellet	
Diameter (inch)	0.175 ± 0.002
Length (inch)	~0.2
Density	95 ~2% of T.D.
Smear Density (nominal)	90% of T.D.
Diametral Gap, pellet to cladding (nominal) (inch)	0.005
Fuel Element ΔP (psi)	~100
Pitch/diameter ratio	1.286
Wire wrap diameter (nominal) (inch)	0.060
Fuel pin bundle, flat-to-flat (nominal)	3.365
Channel	
Inside, flat-to-flat (inches)	3.160 ± 0.005
Wall thickness (inch)	0.100
Material	Incoloy 800

*While this report was being prepared, a number of additional common fuel design bases were established for the reference and backup designs. Future reports will reflect these changes.

4.7 B₄C CONTROL ROD LIFE EXPECTANCY

Calculations were made to estimate the factor limiting the life of the B₄C control elements. The factors considered are radiation

damage to the control rod cladding, helium pressure buildup and burnup of the B₄C. The boron carbide and control cladding temperatures were also calculated for the designs considered. The results are shown in Table 4-4.

FIGURE 4-6. DRIVER FUEL ELEMENT

Design Parameters

A list of the designs and parameters used for the calculations is given below.

These numbers are considered typical and are subject to variation for the final design.

Designs Considered

	<u>No. of Pins</u>	<u>B₄C Diameter (inches)</u>	<u>Clad Thickness (inch)</u>
1.	19	0.63	0.050
2.	7	1.038	0.80
3.	1	2.74	0.33
4.	annular	{ 3.0 o.d. 1.216 i.d.	{ 0.080 0.080
Boron carbide		- 59% volume 80% theoretical density 36 inches long	
Clad material		- 304 Stainless Steel	
Clad yield stress		- 17,500 psi at 1100° F	
Allowable stress		- 1/2 σ yield = 8750 psi	
Control channel		- 3.375 inches across flats	
Coolant temperature		- 700° F inlet 1000° F outlet	
Limiting clad temperature		- 1100° F	
Limiting boron carbide temperature		- 4100° F	
Helium generation rate		- $\sim 1.6 \times 10^{17}$ He/sec for a fully inserted element at the average control rod radius and full reactor power	
Total flux at control rod		- 0.25×10^{16} n/sec for a fully inserted element at the average control rod radius and at full reactor power	
Heat generation rate		- 40 W/gm max.	

4.8 RESULTS

The results of the calculations are shown in Table 4-4. The results show that it will be reasonable to design the rods to some total neutron exposure and to make the gas plenum large enough to prevent excessive pressure. Burnup in the boron

carbide is not considered a serious problem since in-core time is limited by the cladding radiation damage or pressure.

Either the 7- or 19-pin design appear to have the most favorable boron carbide and cladding temperatures.

TABLE 4-4
CONTROL ROD LIFE EXPECTANCY FACTORS

<u>Number of Pins</u>	<u>Boron Carbide Diameter (inches)</u>	<u>Clad Thickness (inch)</u>	<u>Gas Plenum Length, (inches)</u>	<u>Allowable Pressure (psi)</u>	<u>Time to Reach Allowable Pressure, Full Power (days)</u>	<u>Time for Total Fluence of 10²³ nvt, Full Power (days)</u>	<u>Clad i.d. Temp. °F Top of Core</u>	<u>Max. B₄C Temp. °F Top of Core</u>
19	0.63	0.050	48	1390	620	460	1040	1460
			18	1390	280	460	1040	1460
7	1.036	0.080	48	1350	600	460	1085	2160
			18	1350	274	460	1085	2160
1	2.74	0.33	48	2110	932	460	-	-
			18	2110	424	460	-	-
Annulus	3.0 o.d.	0.080	48	467	206	460	1150	4100
	1.216 i.d.	0.080	18	467	94	460	1150	4100

SECTION V

REFUELING AND REACTOR SERVICING SYSTEM

5.1 INTRODUCTION

The refueling and reactor servicing system is based upon an open reactor, hot-cell concept similar to that used in the SEFOR facility. This system may be described as visual, remote-manual since operations are performed remotely by operators viewing the handling equipment and upper region of the reactor. Servicing operations and component transfers which involve exposure of radioactive materials are carried out remotely utilizing visual confirmation of events to as great a degree as possible. In addition, direct access to the refueling cell is provided for maintenance and servicing operations which do not involve exposure of radioactive materials. These operations would be performed when shielding is in place over the reactor vessel, fuel storage tanks, and primary coolant lines.

Emphasis is placed on the use of a number of general purpose handling devices and tools within the refueling cell rather than one multi-purpose machine. Included are those items which are necessary, either directly or indirectly, to perform the servicing and handling functions. The "general purpose" features of many of these items also provides the capability for remote maintenance of other in-cell components. A sketch illustrating the major components of this system was shown in Figure 2-1.

A review of the in-core location of the six closed test loops indicated that driver and open loop refueling equipment accessibility would be greatly enhanced with some minor changes in closed loop positions. Refer to subsection 5.6 for more information on this study.

5.2 REFUELING CELL STRUCTURE

The refueling cell located over the reactor vessel, spent fuel storage tanks, and radioactive component transfer lock provides radiation shielding, containment for the inert gas environment, and structural support for the required equipment and operating conditions. Refueling cell dimensions appear to be within the required cell dimensions for the split conical core concept. Even though the visually performed operations may be somewhat compromised, a larger cell is acceptable for the backup design, provided the features illustrated on the arrangement drawings and discussed below are included. Table 5-1 summarizes the design and operating requirements for the refueling cell. Other refueling and reactor servicing system requirements which influence the refueling cell design are described in the following sections.

5.3 CELL PENETRATIONS

Numerous penetrations must be provided in the refueling cell walls and ceiling. A breakdown of types and approximate quantities are discussed in the following paragraphs.

Roughly 200 3-inch-diameter penetrations are required in the refueling cell walls for electrical service. Approximately 25 would be cast in the wall near the cell ceiling to serve the head carrying hoist, general purpose bridge crane and grapples, upper area lighting, and cell gas service equipment. Approximately 50 penetrations are required about 25 feet above floor level to serve the lower area lighting and bridge-mounted handling device operating near that elevation. The remainder would be uniformly

TABLE 5-1

REFUELING CELL DESIGN AND OPERATING CONDITIONSTemperatures (° F)

Ambient outside the cell	>60
Normal operation in cell	70-120
Operating maximum - in cell	150
Design accident - in cell	250
Exposed sodium (maximum)	400

Pressures (relative to operating areas outside the cell)

Design range	-6 inches water gage to +20 psig
Normal operation (prior to alpha contamination)	+1 ± 1/2 inch water gage
Normal operation (after alpha contamination)	-2 ± 1 inch water gage
Normal design maximum	+6 inch water gage
Design accident	+10 to +20 psig

Atmosphere

Argon content	≥95 vol. %
Nitrogen content	≤5 vol %
Oxygen and water vapor content	≤10 ppm
Maximum design leak rate	<1% cell vol./day
Airborne radioactive contaminants	Permissible during man entry

Design Radiation Levels

Design Requirements

Normal along exterior walls where unlimited access is required	<0.5 mR/hr
Maximum along exterior walls where unlimited access is required	<5.0 mR/hr
Normal along exterior walls where limited access is required	<50 mR/hr

Normal Source values within the cell during shutdown:

Over vessel, shield plug on, one day cooldown	10 mR/hr
Over vessel, shield plug on, 10 days cooldown	negligible
Over vessel, shield plug removed, one day cooldown	5×10^4 R/hr
Over vessel, shield plug removed, 10 days cooldown	10 R/hr
Over fuel storage tank, shield plug on	negligible
Over fuel storage tank, shield removed	100 mR/hr
Over unshielded fuel element (10 ft. above and 2 ft. to side, one day cooldown)	10^4 R/hr
Adjacent (10 ft. away) unshielded fuel element (one day cooldown)	10^5 R/hr
Adjacent (10 ft. away) unshielded fuel element (10 days cooldown)	10^4 R/hr
Adjacent to lower structure of reactor vessel shield plug	10 R/hr
Adjacent to lower surface of reactor vessel shield plug	10 mR/hr

TABLE 5-1 (Continued)
REFUELING CELL DESIGN AND OPERATING CONDITIONS

Normal source values within cell during 400 MWt FFTF Operation

Over vessel	50 mRem/hr
Near end of cell where handling equipment is parked	0.5 mRem/hr

spaced along accessible cell perimeter near floor level to provide electrical service for equipment such as control rod drives, clean-up equipment, special handling, positioning, installation, removal and storage devices, etc. Each of these requires relatively simple S- or L-shaped pipe (to prevent radiation streaming) fitted with a cover plate at the cold face end in which a hermetically sealed connector may be mounted. The desirable arrangement for the floor level penetrations is to use the shielding offset directed downward toward the cold face of the cell wall. This enables placing and routing the electrical cable in a trench below floor level.

Approximately 50 instrumentation penetrations are required in the lower region of the cell. These are required primarily for reactor vessel and fuel instrumentation. Therefore they should be concentrated near the reactor vessel cavity (vessel and experiment instrumentation) and midway between the vessel and refueling plug storage position (fuel instrumentation attached to shield plug). These penetrations would be of the same general design as the electrical penetrations.

Approximately 75 1-inch-diameter pressurized gas penetrations are required. Most of these should be located near the reactor vessel cavity to serve gas actuated devices mounted on the vessel head. The remainder should be conveniently accessible to the service and maintenance regions established as the cell arrangement is firmed up.

Refueling cell gas purge lines should be located near the cell floor and connected to headers running along as much of the cell perimeter as practical. Recirculation lines should penetrate the refueling cell ceiling since this is the recommended location for the cell gas cooling or purification equipment or both. Distribution and collection duct work would be mounted within the cell confines.

Most of the gas monitoring equipment will be associated with the recirculation system and thus will be mounted on the top of the refueling cell. However, gas monitoring penetrations are also required to sample the cell gas in the lower regions of the cell where relatively quiescent conditions are anticipated.

A large (approximately 20-foot square) semi-permanent equipment access penetration is required near floor level to satisfy installation and major maintenance of the backup head and reactor structures. This opening should be considered in the cell structure design so that the opening is available and proper wall reinforcing is provided. A second infrequently used equipment removal penetration (approximately 10-foot diameter) is required in the cell ceiling (far from above the reactor) to provide access to the vessel head handling hoist and any cell gas cooling equipment mounted in the upper region of the refueling cell.

Penetrations for man access equipment are required at, (1) floor level for fairly routine servicing and maintenance operations,

(2) twenty to twenty-five foot elevation for access to bridge mounted handling devices, and (3) near the top of the cell for servicing and maintenance of equipment mounted in that region.

Penetrations for dense glass viewing windows are required on all the cell walls. Tentative requirements (pending actual cell arrangement) are as follows:

- a. One centered at the end of the cell nearest the reactor vessel approximately 15 feet above floor level.
- b. Two required on each of the long walls at elevations determined by further cell arrangement studies.
- c. One located about cell floor level to view refueling plug maintenance operations.

Three penetrations (approximately 15-inch diameter) at each window location is a minimum requirement for through wall mounted equipment. These would enable mounting a periscope and two manipulators (either electro-mechanical or master slave) to assist in operations performed in that vicinity. Additional consideration is required following actual refueling cell arrangement studies.

5.4 STORAGE EQUIPMENT

Irradiated fuel and other activated equipment must be stored in shielded and cooled compartments. The most convenient location appears to have the facilities located below floor level. Shield plugs would be placed over these facilities when handling operations are not in process to permit man entry to the cell. Five 4-foot-diameter sodium-filled wet storage tanks and two 4-foot diameter dry storage wells are presently shown on backup design drawings.

Other storage facilities consist of racks, etc., which would be moved into the cell when convenient.

5.5 REMOTELY CONTROLLED HANDLING EQUIPMENT

Interface requirements for remotely controlled handling equipment consist of wall cut-outs for later mounting of tracks on which bridge mounted equipment will be operated. The required elevations and load ratings are as follows:

- a. Two 1-ton rated tube hoists and electro-mechanical manipulators at 20 feet and 40 feet above floor level.
- b. Twenty-ton rated bridge crane at 50 to 55 feet above floor level.
- c. One-hundred ton rated hoist at 70 feet above floor level. Penetrations for electrical leads and through wall mounted manipulators have been discussed previously.

5.6 IN-CORE LOCATION OF CLOSED TEST LOOPS

A review of the in-core location of the six closed test loops has indicated that refueling equipment accessibility would be greatly improved with some minor relocations. This review was carried out with a graphical model showing the core channel arrangement, the refueling plug opening, the size of the telescoping hoist at various elevations and the refueling grapple. The preferred core layout is shown in Figure 5-1. In general, the modifications included radially moving the five mid-core positions to the same radius and locating them such that the radial pipe runs are equidistant apart. The estimated neutron flux at the five locations is 0.7×10^{16} n/cm²/sec compared to 1.0×10^{16} in the center location. Previously

FIGURE 5-1. REVISED IN-CORE LOCATION OF CLOSED LOOPS

the two peripheral loops had lower flux. With the modified layout the reactor tube of any of the closed loop locations can be moved radially outward if the experimenter wants a lower flux facility.

In the refueling mode (with the three backup-safety-rod drives, the transfer thimble reflector piece and the refueling plug removed) the advantages of the present layout include:

- a. Improved accessibility to the transfer thimble location.
- b. Standardized component parts within the vessel for all closed loops except

the longer branch piping between the center reactor tube and coolant disconnects.

- c. Relative ease in "steering" the fuel refueling grapple and hoist between the closed loop facilities due to larger clearances and improved (standardized) turnaround spaces.
- d. Added flexibility in utilizing multi-channel closed loop facilities or locating any of the reactor tubes in a more peripheral position.

SECTION VI

NUCLEAR CHARACTERISTICS

6.1 CORE DESIGN

Analysis of the FFTF Backup Design is based on a core composed of 139 hexagonal channels. The core arrangement is described in Table 6-1 and shown in Figure 6-1. The reactor core is divided into four radial zones:

Zone 1. Closed Loop

Zone 2. Driver Fuel

Zone 3. Driver Fuel + Closed Loops + Secondary Control

Zone 4. Driver Fuel

For analysis, an equilibrium fuel cycle is assumed with an averaged discharge burnup of 70,000 MWd/Te. The refueling interval is 67.5 full power days. It is assumed that all fuel loaded into the reactor contains relatively clean plutonium, i. e., 88% Pu-239 and 12% Pu-240. It is assumed that 25%, 18%, and 11% of the fuel is replaced in zones 2, 3, and 4 respectively at the end of each refueling

interval. In other words, zones 2, 3, and 4 are on 4, 5.5, and 9 batch refueling schedules, respectively. This gives an average discharge burnup of 70,000 MWd/Te and a peak local burnup of about 96,000 MWd/Te. A peak local burnup of 80,000 MWd/Te would give an average discharge burnup of about 55,000 MWd/Te and a refueling interval of 53 full power days for the same refueling scheme.

The peak linear power density allowable is 18 kW/ft. When reduced by a 1.2 over-power allowance and a 1.1 local peaking factor, the peak linear power density for normal operation is 13.6 kW/ft.

Cross sections used in analysis originate from the General Electric 60 energy group cross section file BAS02. Fuel cross sections are self-shielded for the geometry and composition of this design. The higher Pu-239 alpha values are not included in this cross section data. New cross section files are being prepared which include higher Pu-239 alpha values and lower U-238 capture data.

TABLE 6-1
FFTF BACKUP DESIGN CORE COMPOSITION

<u>Core Height (inches)</u>	36
<u>Core Composition (channels)</u>	
Secondary Control	3
Closed Loops	10
Driver Fuel	<u>126</u>
Total	139
<u>Driver Fuel</u>	
Number Fuel Pins/Bundle	127
Fuel Pin o. d.	0.21
P/D	1.286
Bundle Size (distance across flats) (inches)	3.356
Cell Size (includes 0.025 inch sodium gap between channels) (inches)	3.381
V_{fuel}	0.326
V_{sodium}	0.406
V_{steel}	0.154
$V_{\text{Incoloy-800}}$	0.114
<u>Closed Loops</u>	
V_{fuel}	0.049
V_{sodium}	0.464
V_{steel}	0.487
<u>Secondary Control Locations</u>	
V_{sodium}	0.667
V_{steel}	0.333

6.2 EFFECT OF BURNUP CYCLE ON CORE PARAMETERS

6.2.1 Description of Burnup Cycle

Neutronic parameters of the Backup FFTF Design have been calculated for two

core conditions, the beginning of an equilibrium fuel cycle and the end of an equilibrium cycle. The assumption is that once an equilibrium fuel cycle is established, the amount of fuel replaced per zone during each shutdown is constant. In other words, on a zone average basis, all refueling shutdowns on an equilibrium cycle result in the same change in the

REFLECTOR CONTROL

CLOSED LOOP LOCATIONS

SECONDARY CONTROL
LOCATIONS

FIGURE 6-1. CORE COMPOSITION

core loading. Thus, the beginning of cycle (b.o.c.) is defined as the reactor condition just after refueling and the end of cycle (e.o.c.) is defined as the reactor condition just before refueling.

6.2.2 Reactivity Requirements for Burnup Cycle

The burnup reactivity effect is calculated to be 2.3% $\Delta k/k$ for a refueling interval of 67.5 full power days. It is assumed that all control rods will be removed at the end of cycle. Thus, 2.3% $\Delta k/k$ worth of reflector control must be inserted at the beginning of cycle.

For a refueling interval of 53 full power days (peak local burnup of 80,000 MWd/Te with the same refueling scheme) the burnup reactivity is about 1.9% $\Delta k/k$.

6.2.3 Effect of Control Rods on Power and Flux During Burnup Cycle

The presence of B_4C at the core periphery peaks the neutron flux and power distribution, so the beginning of cycle is the limiting condition for the peak linear power density. Figure 6-2 shows the radial power distribution at the beginning and end of cycle. The peak linear power density at b.o.c. and e.o.c. is 13.77 kW/ft and 12.62 kW/ft, respectively. Thus, the peak linear power density at b.o.c. exceeds the 13.6 kW/ft limit. However, fuel shuffling techniques may be employed to remain within the limit. In other words, when a fuel bundle near the core center reaches the goal burnup, it is replaced with a partially burned fuel bundle from another core location instead of a new bundle. This has a flattening effect on the power distribution. Calculations have not been completed for fuel shuffling.

Calculated neutronic parameters for beginning and end of cycle are listed in

Table 6-2. The peak neutron flux is calculated to be 0.99×10^{16} n/cm²-sec at b.o.c. and 0.95×10^{16} at e.o.c. The decrease in peak flux with burnup results from the flattening effect of withdrawing control rods for burnup reactivity compensation.

6.2.4 Effect of Burnup Cycle on Reactivity Coefficients

It is noted in Table 6-2 that the temperature reactivity effect of changing the core from 400° F isothermal to 400 MWt is the same at beginning and end of cycle, $\sim -1.75\% \Delta k/k$. The temperature effect is due to expansion and density changes and the Doppler effect. The expansion and density contribution is calculated to vary from $-1.15\% \Delta k/k$ to $-1.09\% \Delta k/k$ during the refueling interval. The Doppler coefficient, $T \frac{dk}{dT}$, is calculated to vary from -0.0059 to -0.0063 during the refueling interval. The presence of B_4C in the reflector at b.o.c. slightly increases the expansion effect and reduces the Doppler effect.

The relatively large value of the Doppler coefficient, $T \frac{dk}{dT} \approx -0.006$, is partially due to a fuel cycle assumption, i.e., that all fuel is fabricated from "clean" plutonium, 88% Pu-239 and 12% Pu-240. Even when equilibrium fuel cycle is established, the Pu-240 content is low, $\sim 15\%$ of the total plutonium (see Table 6-2). If the plutonium from the FFTF is recycled or if plutonium from power reactors is used in fuel fabrication, the Pu-240 content will be substantially increased and the Doppler coefficient decreased. It is also expected that the Doppler coefficient will be decreased about 10% when the higher Pu-239 alpha values and lower U-238 capture cross sections are used.

6.3 REFLECTOR CONTROL

The reflector control strength for the FFTF Backup Design is the difference in

FIGURE 6-2. RADIAL POWER DISTRIBUTION AT BEGINNING AND END OF CYCLE

TABLE 6-2
CALCULATED PARAMETERS FOR BACKUP FFTF DESIGN

	<u>Beginning of Cycle</u>	<u>End of Cycle</u>
Peak Neutron Flux	$0.99 \times 10^{16} \text{ n/cm}^2\text{-sec}$	$0.95 \times 10^{16} \text{ n/cm}^2\text{-sec}$
U-238 Doppler, $T \frac{dk}{dT}$	-0.0059	-0.0063
Expansion Effect		
400° F - 400 MWt	-1.15% $\Delta k/k$	-1.09% $\Delta k/k$
Total Temperature Effect		
400° F - 400 MWt	-1.76% $\Delta k/k$	-1.75% $\Delta k/k$
Fissile Enrichment	20.56%	19.79%
Core Average Plutonium Isotopic Composition		
% Pu-239	85.2	83.9
% Pu-240	14.4	15.5
% Pu-241	.4	.6
% Pu-242	-	-

reactivity between B_4C and nickel in the radial reflector. Self-shielded nickel cross sections used for analysis of the reflector data obtained from the SEFOR critical experiment in ZPR-3 (Assembly 47) are also used for the FFTF. The SEFOR experimental data appeared to verify the reflector worth predicted when using these cross sections. Because some experimental verification exists, the FFTF reflector strength is also calculated using the SEFOR self-shielded nickel cross sections. Preliminary diffusion theory calculations of control strength are \$16 and \$28 for natural B_4C and B_4^{10} , respectively.

- a. Nickel cross sections are also obtained from the ENDF/B cross sections library and appropriately self-shielded by the ENDRUN code.

Calculations using these cross sections give reflector control strengths about 2\$ greater than the SEFOR nickel cross sections. It does not appear that the nickel resonances above 400 keV are adequately described in ENDF/B so the ENDRUN self-shielding above 400 keV is questionable. ENDF/B cross sections are being prepared which will use Bondarenko self-shielding above 400 keV.

- b. Sixteen-energy group, self-shielded nickel cross sections were used for analysis of the reflector data obtained from the SEFOR critical experiment in ZPR-3 (Assembly 47). These cross sections are self-shielded with Bondarenko self-shielding factors.

Results of preliminary diffusion theory calculations of reflector control strength are listed below.

	ENDF/B (ENDRUN) Nickel	SEFOR Nickel
Natural B_4C	\$17.7	\$15.9
$B_4^{10}C$	\$29.5	\$27.5

Calculations assume the control system is located in 44 channels surrounding the core (see Figure 6-1). As observed above, the SEFOR nickel data predict a control worth about 2\$ smaller than the ENDF/B data. It is not yet known which set of nickel data is best for the FFTF Backup Design reflector control analysis.

6.4 COMPARISON OF 0.21 INCH VERSUS 0.25 INCH FUEL DIAMETER IN BACKUP FFTF DESIGN

The effect on the peak neutron flux of a 0.25 inch versus 0.21 inch diameter fuel pin design was investigated in a parameter study. ⁽¹⁾ It is also of interest to determine the effects of a 0.25 inch versus 0.21 inch diameter fuel pin design on other neutronic parameters in the backup FFTF design.

The reactor designs obtained in reference 1, case 3 and case 2, for the 0.25 inch and 0.21 inch fuel pin diameters, respectively, were used in this study. The two designs are comparable with the design basis listed in Reference (1). The BISYN computer code was used for these calculations. Calculated values for the reflector control strength with natural B_4C and $B_4^{10}C$ and the U-238 Doppler coefficient, $T \frac{dk}{dT}$, for both sodium-in and sodium-out are listed in Table 6-3 for the 0.25 inch and 0.21 inch diameter fuel pin designs. Also listed in Table 6-3 is the peak neutron flux for these two designs calculated during a previous study.

6.5 FFTF BACKUP DESIGN CRITICAL EXPERIMENT IN PHASE B CRITICAL PROGRAM

6.5.1 Program Objective

As part of the FFTF Backup Design effort, a critical experiment is planned as part of the PNL Phase B critical program in ZPR-3. One of the main objectives of this experiment is to verify the feasibility of reflector control for the FFTF Backup Design. Other experiments should be performed to give an experimental comparison of critical mass, Doppler, and sodium loss calculational methods.

TABLE 6-3
COMPARISON OF 0.25 INCH AND 0.21 INCH PIN
DIAMETER DESIGNS

	0.21 Inch	0.25 Inch
Peak Neutron Flux	1.0×10^{16}	14% lower
Reflector Control	20-30\$	70% of 0.21-inch value
U-238 Doppler, $T \frac{dk}{dT}$	Na in 0.006 Na out 0.004	0.0008 greater 0.0007 greater

6.5.2 Experimental Program

Following is a tentative outline of critical experiments proposed for the backup FFTF design in the PNL Phase B critical program. These experiments assume that the 430 kg plutonium limit at ZPR-3 is increased. (Slight modifications to the experimental program will be necessary for the current plutonium limit.) It is also assumed that ~ 300 kgs of B_4C will be available.

Table 6-4 outlines the major experiments in the backup experimental program with a time estimate for each experiment.

a. Load to Critical

It is proposed that the fuel (core) drawers of Assembly IIa be modified so that the composition corresponds more closely to the backup design. This degree of modification will depend on whether the plutonium limit is increased and how accurately the backup design composition is simulated. Calculations will be required to determine what changes are necessary. The reflector geometry should be very similar to that of Assembly IIa. It is assumed that a minimal amount of reloading will be necessary outside of the core.

The initial loading will be designed to go critical with a core volume of about 750 liters with a radial nickel reflector. The core will contain six simulated closed loops and three simulated control rod locations.

b. Radial Fission Traverses

Radial fission traverses will be obtained for U-238, Pu-239, and U-235.

c. Sodium Loss Reactivity Effect

The maximum positive sodium loss reactivity effect should be measured by simulating sodium voiding near the core center and progressively increasing the voided volume until the maximum positive effect is obtained. It is expected the maximum positive effect will occur when sodium voiding is simulated in about the central 30% of the core volume. The maximum effect should be verified by making two measurements where the incremental change in reactivity is negative, due to increasing the volume of simulated sodium voiding.

d. Doppler Effect

The Doppler effect should be measured near the core center in a sodium-voided

TABLE 6-4

OUTLINE OF BACKUP FFTF DESIGN CRITICAL EXPERIMENT

<u>Experiment</u>	<u>Estimated Length (weeks)</u>
a. Load to Critical	4
b. Obtain Radial Fission Traverses	1
c. Measure Sodium Loss Effect	3
d. Measure Doppler Effect	2
e. Measure Reflector Control Worth	6
Total	16

neutron spectrum. The Doppler effect should also be measured in the normal mockup, i. e., with sodium present.

e. Reflector Control

The reflector control worth should be measured with normal sodium conditions. The drawers of the simulated closed loops and control rod locations (loop drawers) should be uniformly distributed in the core.

Nickel in the radial reflector should be replaced with B_4C in small steps so that reactivity effect of each step can be measured with the calibrated ZPR-3 control rods. The negative reactivity

effect of each step should be compensated for by replacing the "loop drawers" with drawers containing fuel. This effect should also be measured with the ZPR-3 control rods. The ZPR-3 control rod calibration should be checked several times during the experiment.

In this manner, nickel in the radial reflector will be replaced with B_4C to simulate the insertion pattern of the backup design control rods. When all B_4C is loaded, the "loop drawers" should be reloaded into their original positions at the start of this experiment and the B_4C worth measured by subcritical multiplication.

SECTION VII

SHIELDING

7.1 FISSION PRODUCT DECAY ENERGY

A very large amount of work has been done on other projects to measure the decay energy from U-235 fission products. Essentially all of these products result from fission reactions with thermal neutrons. In the FFTF core, however, fission products are formed from high energy, or fast, neutron ($E_n > 0.1$ MeV) reactions with Pu-239. The question being asked is, "How does the fission-product decay energy from the fast neutron fission of Pu-239 differ from the thermal neutron fission of U-235?"

A preliminary study indicates that measurements of the fission product decay energy from the fast fission of Pu-239 are as yet quite limited. A relatively old reference notes that within the limits of experimental error (10 percent) there is no perceptible difference between the results for U-235 and Pu-239 as to the dependence on

time after fission, mean energy of the gamma radiation, and total yield of gamma energy per fission.⁽³⁾ It should be noted that the latter reference refers to gamma energy only. Two relative recent References (4 and 5) indicate very good agreement when the total (gamma and beta) fission-product decay power from U-235 thermal neutron fission is compared to the total fission-product decay power from Pu-239 fast neutron fission. However, the latter references include measurements for only relatively long times after fission, i. e., 40 to 150 days.

In summary, a comparison of the limited measurements of fast-neutron-induced Pu-239 fission-product decay energy to the relatively extensive measurements of thermal neutron U-235 fission-product decay energy indicates that differences are not large. Some experimental and analytical data indicate slightly lower decay energy

from Pu-239 fast fission than from U-235 thermal fission. In view of the consistency of the available data, the use of fission-product decay curves based on measurements of U-235 thermal fission is recommended for estimating the decay power versus time after reactor shutdown from Pu-239 fast fission for preliminary FFTF design work.

Figures 7-1a and 7-1b illustrate the recommended total (as well as gamma and beta only) fission product decay power versus cooling time (or time after reactor shutdown). These figures assume infinite reactor operating time. Table 7-1 and Figures 7-2a and 7-2b illustrate fission product decay power versus cooling time for irradiation times varying from 1 hour to infinite time.

In using these curves, it must be noted that there are other important sources of power generation in the FFTF core following reactor shutdown. During the first minute after scram, the fission power due to delayed neutron emission exceeds fission product decay power by a factor of 2 or more. At times ranging from 2 hours to 30 hours after shutdown, Np-239 decay may contribute up to one-half of the heat generated by fission product decay (the former is additional to fission product decay energy).

7.2 UPPER AXIAL SHIELD DESIGN

A study was initiated to determine the optimum composition and spatial distribution of shielding above the core. Figure 7-3 illustrates the schematic shield design for the FFTF core. This optimum study includes consideration of the following factors:

- a. Shielding material costs, including fabrication
- b. Shielding material compatibility

- c. Height of fuel assembly
- d. Reactor vessel height
- e. Maintenance requirements
- f. Neutron activation
- g. Gamma heating
- h. Radiation level in the refueling cell
- i. Coolant pressure drop

As will be illustrated and discussed in the following text and figures, this study indicates that it is both feasible and acceptable to remove all B₄C from the upper axial neutron shield and to reduce its thickness from 4.1 feet to 3 feet. This is a preliminary conclusion subject to review by the reactor vessel and internals design group. The conclusion is based on approximate calculational results which must be verified by more exact methods.

Tables 7-2 and 7-3 illustrate the shielding compositions and thicknesses analyzed in the initial conceptual design study and the alternates evaluated in the current study. As compared to the upper axial shielding in the conceptual design analysis, the first change was a reduction in the neutron shield thickness from 4.1 feet to 3 feet. Figures 7-4 and 7-5 illustrate the changes in the upper axial neutron and gamma flux distributions due to this change. Figure 7-4 indicates an increase in neutron dose rate at the upper surface of the top shield plug from a low <0.1 mRem/h to an acceptable 1.4 mRem/h. Figure 7-5 indicates an increase in the capture gamma dose rate from a low <0.1 mR/h to an acceptable 2 mR/h. For comparison purposes, the dose rate due to Na-24 activity in the primary coolant is also illustrated. The Na-24 dose rate was calculated to be about 20 mR/h at the upper surface of the

**FIGURE 7-1a. ABSORBABLE ENERGY RELEASED BY FISSION PRODUCTS
FOLLOWING INFINITE REACTOR OPERATION**

FIGURE 7-1b. ABSORBABLE ENERGY RELEASED BY FISSION PRODUCTS FOLLOWING INFINITE REACTOR OPERATION

FIGURE 7-2a: ABSORBABLE ENERGY RELEASED BY FISSION PRODUCTS FOR OPERATING TIMES VARYING FROM ONE HOUR TO ∞ VERSUS COOLING TIME

FIGURE 7-2b. ABSORBABLE ENERGY RELEASED BY FISSION PRODUCTS FOR OPERATING TIMES VARYING FROM ONE HOUR TO ∞ VERSUS COOLING TIME

FIGURE 7-3. FFTF CORE SCHEMATIC SHIELDING DESIGN

FIGURE 7-4. TOTAL NEUTRON FLUX DISTRIBUTION VERSUS HEIGHT ABOVE THE CORE-REFLECTOR INTERFACE

FIGURE 7-5. GAMMA DOSE RATE VERSUS HEIGHT ABOVE THE CORE-REFLECTOR INTERFACE

TABLE 7-1
 ENERGY RELEASE RATE FROM FISSION PRODUCT DECAY
 IN A U-235 - FUELED REACTOR AFTER SHUTDOWN

Operating Time, t_o (h)	Shutdown Time, t_s (sec)	MeV/fission ^{a, b}			MeV-sec/fission ^c $\int_0^t s_{p_{\gamma+\beta}} dt$
		P_γ	P_β	$P_{\gamma+\beta}$	
1	1	5.06	4.70	9.76	12
	10	3.90	3.40	7.30	110
	10 ²	2.12	1.86	3.98	500
	10 ³	0.68	0.58	1.25	2,260
	10 ⁴	0.02	0.01	0.03	6,300
	10 ⁵	-	-	-	9,900
10	1	5.80	5.36	11.16	12
	10	4.64	4.06	8.70	110
	10 ²	2.84	2.48	5.32	800
	10 ³	1.32	1.10	2.42	3,760
	10 ⁴	0.40	0.30	0.70	14,800
	10 ⁵	0.046	0.014	0.06	32,400
10 ²	10 ⁶	0.005	-	0.005	47,200
	1	6.08	5.68	11.76	12
	10	4.92	4.38	9.30	110
	10 ²	3.14	2.78	5.92	800
	10 ³	1.62	1.42	3.04	4,960
	10 ⁴	0.66	0.60	1.26	19,200
10 ³	10 ⁵	0.176	0.120	0.296	67,000
	10 ⁶	0.032	0.018	0.050	172,000
	10 ⁷	-	-	-	400,000
	10 ⁸	-	-	-	460,000
	1	6.34	5.78	12.12	12
	10	5.18	4.48	9.63	110
10 ⁴	10 ²	3.40	2.88	6.28	800
	10 ³	1.86	1.52	3.38	4,960
	10 ⁴	0.88	0.72	1.60	27,200
	10 ⁵	0.37	0.246	0.616	103,000
	10 ⁶	0.136	0.088	0.224	406,000
	10 ⁷	0.014	0.010	0.024	1,542,000
10 ⁵	1	6.44	5.88	12.32	12
	10	5.28	4.58	9.86	110
	10 ²	3.50	2.98	6.48	800
	10 ³	1.96	1.62	3.58	4,960
	10 ⁴	0.984	0.824	1.808	27,200
	10 ⁵	0.472	0.344	0.816	137,000
10 ⁶	10 ⁶	0.220	0.176	0.396	520,000
	10 ⁷	0.044	0.056	0.100	3,180,000
	10 ⁸	-	-	-	2,270,000
	1	6.44	5.94	12.38	12
	10	5.28	4.64	9.92	110
	10 ²	3.50	3.04	6.54	800
10 ⁷	10 ³	1.98	1.66	3.64	4,960
	10 ⁴	0.994	0.872	1.866	27,200
	10 ⁵	0.480	0.394	0.874	137,000
	10 ⁶	0.226	0.226	0.452	740,000
	10 ⁷	0.050	0.098	0.148	2,840,000
	10 ⁸	0.006	0.024	0.030	8,300,000
∞	1	0.40	5.98	12.46	12
	10	5.32	4.68	10.00	110
	10 ²	3.44	3.08	6.62	800
	10 ³	2.00	1.74	3.74	4,960
	10 ⁴	1.02	0.93	1.95	27,200
	10 ⁵	0.508	0.450	0.958	137,000
∞	10 ⁶	0.254	0.284	0.538	740,000
	10 ⁷	0.076	0.158	0.234	3,645,000
	10 ⁸	0.030	0.080	0.110	16,105,000

(a) Multiply by (MeV/fission)⁻¹ to obtain fraction of operating power.

(b) Multiply results of (a) by power to obtain units of power.

(c) Multiply by power × (MeV/fission)⁻¹ to obtain units of energy, i. e., power-sec.

TABLE 7-2
CONCEPTUAL DESIGN

<u>Region No.</u>	<u>Name</u>	<u>Region Width (cm)</u>	<u>Material, Volume Fraction, and Density</u>	
1	Core	45.75	Fuel:	0.303 at 10.96 g/cm ³
			Na:	0.468 at 0.8512 g/cm ³
			SS:	0.127 at 7.93 g/cm ³
			I-800:	0.102 at 8.02 g/cm ³
2	Reflector	30.50	Na:	0.449 at 0.8512 g/cm ³
			Ni:	0.359 at 8.90 g/cm ³
			SS:	0.079 at 7.93 g/cm ³
			I-800:	0.113 at 8.02 g/cm ³
3	Gas Plenum	76.20	Na:	0.449 at 0.8512 g/cm ³
			Void:	0.359 at 0.0 g/cm ³
			SS:	0.079 at 7.93 g/cm ³
			I-800:	0.113 at 8.02 g/cm ³
4	Neutron Shield	125.09	Na:	0.498 at 0.8512 g/cm ³
			B ₄ C:	0.297 at 1.85 g/cm ³
			SS:	0.092 at 7.93 g/cm ³
			I-800:	0.113 at 8.02 g/cm ³
5	Fuel Handle	12.07	Na:	0.500 at 0.8512 g/cm ³
			SS:	0.500 at 7.93 g/cm ³
6	Sodium Pool	326.39	Na:	1.0 at 0.8512 g/cm ³
7	Top Shield Plug	61.00	Fe:	1.0 at 7.70 g/cm ³

TABLE 7-3
CURRENT STUDY

Region No.	Name	Region Width (in.)	Material, Volume Fraction, and Density
1	Core	45.75	Same as in Table 7-2
2	Reflector	30.50	Same as in Table 7-2
3	Gas Plenum	76.20	Same as in Table 7-2
No B_4C in Neutron Shield and 0 to 30% SS in Na-Pool			
4-A	Neutron Shield	91.44	Na: 0.498 at 0.8512 g/cm ³ SS: 0.389 at 7.93 g/cm ³ I-800: 0.113 at 8.02 g/cm ³
5-A	Fuel Handles	12.07	Same as in Table 7-2
6-A	Sodium Pool	326.39	a No SS in Na-Pool b 5% SS in Na-Pool c 10% SS in Na-Pool d 30% SS in Na-Pool ρ (SS) = 7.93, ρ (Na) = 0.8512
7-A	Top Shield Plug	61.00	Same as in Table 7-2
B_4C in Upper Part of Neutron Shield and 5% SS in Na-Pool			
4-B I	Neutron Shield I	81.28	Same as in Region 4-A
		10.16	Same as in Table 7-2
4-B II	Neutron Shield II	71.12	Same as in Region 4-A
		20.32	Same as in Table 7-2
5-B	Fuel Handles	12.07	Same as in Table 7-2
6-B	Sodium Pool	326.39	Na: 0.95 at 0.8512 g/cm ³ SS: 0.05 at 7.93 g/cm ³
7-B	Top Shield Plug	61.00	Same as in Table 7-2

top shield plug. Hence, even with the reduced upper axial neutron shielding, the radiation level in the refueling cell is still due mainly to Na-24 decay activity. Another important factor to be noted in comparing the Na-24 gamma dose rate to the neutron dependent radiation levels at the upper surface of the top shield plug is that the latter should include geometric attenuation factors. A factor of 1/25 was estimated for this correction which reduces the neutron dependent radiations to a very low 0.1 mRem/h compared to the 20 mR/h calculated for the Na-24 decay activity. Hence, it is feasible to further reduce the upper axial neutron shield thickness. It may also be advisable to reduce the dose rate due to Na-24 activity by increasing the gamma shielding above the sodium pool.

Additional analyses were made to estimate the shielding effect of the structural material, control rod extensions, instrumentation, coolant loop piping, etc., which will be located in the sodium between the upper neutron shield and the top shield plug. The effect of varying the stainless steel volume fraction from 0 to 30% is graphically illustrated by Figures 7-6, 7-7, 7-8, and 7-9. The base case has the reduced neutron shield thickness assumed for the current design (Figures 7-4 and 7-5). The effect of increasing the stainless content is to reduce dose rates which are acceptable even with no stainless in the sodium. A reduction in neutron activation of components extending below the top shield plug is the most apparent advantage of substituting steel for sodium in the pool. For the 5 volume percent stainless case, neutron activation at the lower surface of the top shield plug is reduced from an estimated dose rate of 2 R/h to about 20 mR/h—a factor of 100. Of course, the gamma radiation in the refueling cell is, as noted in the preceding paragraphs, due to Na-24 activity, and

is not a function of the steel content in the sodium pool (Figure 7-8). It follows that gamma heating also is due primarily to Na-24 activity. As noted by Figure 7-9, an increase in the steel content from 0 to 30 percent in the sodium pool causes essentially no change in the upper axial gamma heating distribution.

A further evaluation was made, primarily to determine the effect on activation of equipment in the sodium pool, of including B_4C in the upper 4 to 8 inches of the neutron shield. The base case selected for this evaluation had a total neutron shield thickness of 3 feet. A stainless steel volume fraction of 0.05 was assumed in the sodium pool. As illustrated by Figures 7-10 and 7-11, total and thermal neutron fluxes are reduced by about a factor of 10 by including B_4C in the upper 8 inches of the neutron shield. At the lower surface of the top shield plug, the activation dose rate is reduced from about 50 mR/h to less than 10 mR/h. Figures 7-12 and 7-13 illustrate the effect on gamma dose rate and gamma heating. Since the dose rate and gamma heating was dominated by gammas from Na-24 decay even in the case with no B_4C , the addition of B_4C has a negligible effect on the gamma dose rate and gamma heating distribution above the core.

Overall, it appears acceptable to reduce the thickness of the upper axial neutron shield from 4.1 to 3 feet. Further, the removal of B_4C from this shielding is also acceptable. Note, however, that the evaluation of results of this upper shield design study discussed in this report considered only neutron activation, radiation level in the refueling cell and gamma heating. The other factors which are pertinent to this optimization study are under evaluation by the reactor vessel and internals design group.

FIGURE 7-6. TOTAL NEUTRON FLUX DISTRIBUTION VERSUS HEIGHT ABOVE THE CORE-REFLECTOR INTERFACE

FIGURE 7-7. THERMAL NEUTRON FLUX DISTRIBUTION VERSUS HEIGHT ABOVE CORE REFLECTOR INTERFACE

FIGURE 7-8. GAMMA DOSE RATE VERSUS HEIGHT ABOVE THE CORE-REFLECTOR INTERFACE

FIGURE 7-9. GAMMA HEATING DISTRIBUTION VERSUS HEIGHT ABOVE CORE-REFLECTOR INTERFACE

FIGURE 7-10. TOTAL NEUTRON FLUX DISTRIBUTION VERSUS HEIGHT ABOVE THE CORE-REFLECTOR INTERFACE

FIGURE 7-11. THERMAL NEUTRON FLUX DISTRIBUTION VERSUS
HEIGHT ABOVE THE CORE-REFLECTOR INTERFACE

FIGURE 7-12. GAMMA DOSE RATE VERSUS HEIGHT ABOVE THE CORE-REFLECTOR INTERFACE

FIGURE 7-13. GAMMA HEATING DISTRIBUTION VERSUS HEIGHT ABOVE THE CORE-REFLECTOR INTERFACE

SECTION VIII

THERMAL HYDRAULICS ANALYSIS

8.1 PRESSURE DROP

Pressure drop calculations were made for 0.210-inch and 0.250-inch pins at various values of P/D , Reynolds number, and number of pins per subassembly. These were performed to investigate the effect of changes in these parameters on fuel pin pressure drop and also permit a comparison between PNL and General Electric calculational methods. Recent communication with PNL indicate that the discrepancies between the methods from earlier reports on this subject do not now appear to be severe and an agreement on the analytical basis for calculating pressure drop is anticipated. Figures 8-1, 8-2, and 8-3 summarize the results of computations. A detailed review of the pressure drop in the shield regions of the fuel pin subassembly is planned for the coming quarter. The allowances based upon an approximate analysis are anticipated to be adequate.

The analysis to date has been based upon the assumption that the flow channel adjacent to the wall is clean. If design features are introduced to reduce the flow adjacent to the channel wall, an approximate estimate of the increase due to this can be obtained by assuming this condition approaches the infinite pin array. From the curve it can be seen that this effect would probably be less than 10 psi.

8.2 REFLECTOR CONTROL RODS

An analysis has been performed on several conceptual configurations of reflector

control rods. The reflector control is comprised of a movable boron carbide poison section with a nickel reflector section each approximately the same length as the active core height. Fission gas plenums are provided above the boron carbide in the poison pins. The purpose of the analysis is to establish the cooling requirements and the corresponding temperatures in the boron carbide, clad, and nickel. The allowable maximum temperature in the boron carbide and the degree of burn-up is not firmly established. The analysis shows that for a 19-pin array boron carbide peak temperatures can be held to about 1460°F with a 300°F temperature rise through the channel, and the corresponding pressure drop is about 7 psi. This temperature level may be sufficiently low that unidentified problems relating to high B_4C temperatures do not manifest themselves. Since the available flow is greatly in excess of that required for a 300°F ΔT , a further reduction of this temperature by increasing the flow is readily attainable. Increasing the flow to give say 100°F ΔT would reduce the peak temperature (for a 19-pin array) to approximately 1260°F at both core midplane and top of the rod, with a corresponding peak clad temperature of approximately 850°F.

The boron carbide and clad temperatures were calculated for four design configurations. The designs considered are:

	No. of Rods	B_4C Diameter (inches)	Clad Thickness (inch)
1	19	0.63	0.050
2	7	1.038	0.080
3	1	2.74	≤0.33
4	Annular	3.0 o.d. 1.216 i.d.	0.080 0.080

FIGURE 8-1. CORE PRESSURE DROP PER FOOT FOR FUEL ELEMENT-DIAMETER = 0.2501 INCH

FIGURE 8-2. CORE PRESSURE DROP PER FOOT FOR FUEL ELEMENT
 DIAMETER-0.210 INCH

FIGURE 8-3. FUEL PIN PRESSURE DROP

Boron Carbide	- 59% volume 80% theoretical density (2.0 gm/cc) 36 inches long
Clad Material	- 304 Stainless Steel
Coolant Temperature	- 700° F inlet 1000° F outlet
Limiting Clad Temperature	- 1100° F
Limiting Boron Carbide Temperature	- 4108° F is an established limit but due to lack of irradiation data, a much lower limit may result from further information.

The following assumptions were used in the analysis:

Film Coefficient - Sodium to Clad	20,000 Btu/h-ft ² -° F
Gap Coefficient - Clad to B ₄ C	1000 Btu/h-ft ² -° F
Thermal Conductivity	
- 304 Stainless Steel	160 Btu/h-ft ² -° F/in.
- B ₄ C	4 Btu/h-ft ² -° F/ft
Coolant Outlet Temperature	1000° F

The total heat generation rate in the B₄C consists of Alpha, Gamma, and Neutron heating. The peak value at the core midplane (peak axial) of 40 watts/gm was used for the analysis.⁽¹⁾ The volumetric heat generation in the stainless steel clad due to gamma and neutron heating was integrated into the calculations.

Thermal Analysis of the B₄C Rod

The results of the thermal analysis are shown in Table 8-1. The table includes the analysis of the midplane and control rod coolant outlet.

The annular and single rod geometries were found to be prohibitive from a B₄C limiting temperature of 4100° F. Also, the cladding temperature limit of 1100° F was exceeded in the single rod analysis. The cladding temperatures may be reduced by increasing the required flow to reduce the outlet temperature to below 1000° F. Both the seven and nineteen-rod bundle design

result in peak temperatures in the B₄C of 1460° F and 2160° F respectively.

Hydraulic Analysis of the B₄C Control Rod and Nickel Reflectors

The required coolant flow in a B₄C-nickel nineteen poison-rod bundle is 4.2×10^5 lb_m/h through the first row adjacent to the core to maintain an outlet temperature not greater than 1000° F. When the nickel segments are adjacent to the core during reactor operation, the coolant flow would be more than adequate and overcooling would occur. This would degrade the reactor outlet temperature slightly.

The total required flow rate through the control rod and nickel segment in all the reflector region is 7×10^5 lb_m/h. This flow rate is predicated on an orificing scheme which maintains a 300° F temperature rise in each channel.

TABLE 8-1

THERMAL ANALYSIS SUMMARY OF THE B₄C CONTROL ROD DESIGNI. At Core Midplane (all temperature in °F)

	<u>Average Bulk Temp.</u>	<u>Clad o.d.</u>	<u>Clad i.d.</u>	<u>B₄C Surf. Temp.</u>	<u>B₄C Centerline</u>
Annular Inner	850	865	1000		
Outer	850	860	990	1300	3960
Single Rod	850	870	1680	2125	-
Seven Rod	850	860	935	1100	2000
Nineteen Rod	850	855	890	1015	1345

II. At Outlet of Control Rod (all temperatures in °F)

	<u>Average Bulk Temp.</u>	<u>Clad o.d.</u>	<u>Clad i.d.</u>	<u>B₄C Surf. Temp.</u>	<u>B₄C Centerline</u>
Annular Inner	1000	1010	1140		
Outer	1000	1015	1155	1430	4100
Seven Rod	1000	1010	1085	1260	2160
Nineteen Rod	1000	1005	1040	1130	1460

For analysis of the expected pressure drop, the following assumptions were used:

Rods/subassembly = 19 rods
P/D = 1.075
D = 0.68
Length (inches) 36 Nickel
36 B₄C
48 Gas Plenum
122 TOTAL

Entrance and Exit Head Losses -
2 velocity heads

The summary of the velocity and pressure drops in a control rod is:

Velocity (ft/sec)	4.1
$\Delta P_{\text{entrance-exit}}$ (psi)	0.4
$\Delta P_{\text{friction}}$ (psi)	6.2
Total (psi)	6.6

8.3 SODIUM-TEMPERATURE RESPONSE TO NORMAL TRANSIENTS

The FFTF system must be designed for stress fatigue and thermal shock resulting from normal transients and for thermal shock from abnormal transients. Normal transients are associated with events for which the design number of cycles is of the order of 10^3 during the plant lifetime. Abnormal transients are associated with low probability events, which may occur one to a few times.

This section describes the short term temperature response of the FFTF Backup System to normal transients for various conditions. The temperature responses of the sodium at the top of the active core, at the channel outlet, and at the vessel outlet are examined for two pump coastdown characteristics. Pump coastdown curve 1 corresponds to the FERMI pumps and represents only the inertia of the pump; it is the

characteristic which is being recommended for the FFTF Backup Design. Curve 2 represents the EBR-II pump which coasts down much more slowly due to additional inertia added to the pump.

The results here apply only to the first 20 to 30 seconds after the start of a transient. The influence of natural convection is not included. Beyond 20 seconds natural convection becomes important and increasingly provides an important pressure driving force.

The following problems were analyzed under the normal transient classification.

- a. Loss of site power
- b. Reactor overpower scram
- c. Reactor high temperature scram
- d. Pump startup in standby loop

The principal variable studied was the comparison of pump coastdown curves 1 and 2. Additional variables studied included (1) outlet mixing time constant, (2) pump coastdown delay time, and (3) time between scram signal and start of rod drop.

8.3.1 Conclusions

1. For the 20-second period following normal scram, the pump coastdown curve 1 (similar to Fermi) is preferable to curve 2 (similar to EBR-II). The sodium temperature gradients (and, hence, thermal stresses) at the core outlet, channel outlet, and vessel outlet are significantly steeper for curve 2 than for curve 1. These gradients are now available to the stress analyst to judge the relative merits of the two coastdown curves, though curve 1 looks more attractive than curve 2 for the first 20 seconds after a transient. No conclusion is yet available concerning the sodium temperature overshoot before natural convection takes over or the ultimate feasibility of transition to natural convection for either coastdown curve.

2. The sodium temperature gradient at the vessel outlet is strongly sensitive to the outlet mixing time constant. Although the gradient with no mixing for curve 1 does not appear to be prohibitive, sufficient reduction in the gradient is obtained for mixing the core outlet sodium with 20% of the sodium in the tank, that an effort to achieve such mixing may be warranted.

3. Curve 1 leads to an initial rapid increase in sodium core outlet temperature for some conditions, but this rise is limited to $\sim 30^\circ\text{F}$ which is believed acceptable. This rise practically disappears for curve 2.

4. Delays of the order of 1 second in starting the pump coastdown after scram produce large initial negative sodium temperature gradients. Hence, this consideration indicates the desirability for minimizing the time delay before starting the pump coastdown.

5. Little difference in temperature gradients outside the core appeared between values of 0.15 sec and 0.25 sec for the time between scram signal and start of rod drop.

6. No important differences in temperature gradients in the regions above the core were observed between the four scram situations investigated. The highest temperatures and positive temperature gradients were reached with the high temperature scram, but these temperatures and gradients do not appear excessive. The largest negative gradients occurred with the pump startup in the standby loop, but again these gradients were not significantly greater than for the other scram situations.

8.3.2 Reactor Model

8.3.2.1 Flow Diagram

The flow diagram for the calculational model is shown in Figure 8-4. Only the

FIGURE 8-4. FLOW DIAGRAM

vessel and vessel internals are included. The vessel inlet is input to the calculation. Excluding the IHX (intermediate heat exchanger) and secondary system is a satisfactory approximation for the conditions analyzed for the first 20 seconds of the transient. The reason for this is that, for constant flow cases, the effect of the IHX is felt only after the transit time of the primary sodium outside the vessel, which is nearly 20 seconds. For pump coastdown, the effect of the IHX is felt after the transit time from the IHX to the vessel, which is 5 seconds at full flow but is increasing rapidly during the flow coastdown.

8.3.2.2 Dynamics Equations

The dynamics equations are given in reference 1.

9.3.2.3 Pump Coastdown Curves

The pump coastdown curves 1 and 2 are shown in Figure 8-5. The solid curves represent the analytical expressions used for the transient analysis, and are of the form:

$$\frac{w(t)}{w_0} = \frac{1}{1 + a(t - t_0) + b(t - t_0)^2}$$

where w_0 is the initial flow rate, and t_0 is the starting time of the coastdown. The dashed lines represent the coastdown curves for FERMI (at 12,000 gpm flow rate) and EBR-II.

8.3.2.4 Control Rod Scram Curve

The shape of the control rod scram curve expected to approximate the FFTF curve was used to describe scram reactivity versus time. For the present analysis, this curve was approximated by the following form up to a total control worth of 10\$:

$$\Delta k(t) = \alpha(t - t_0) + \beta(t - t_0)^2 + \gamma(t - t_0)^3$$

where $\Delta k(t)$ is the control worth inserted, in dollars, and t_0 is the start of the rod drop. Values used for the coefficients were:

$$\alpha = 0$$

$$\beta = -6.085 \text{ \$/sec}^2$$

$$\gamma = -187.35 \text{ \$/sec}^3$$

After -10\$ was inserted, the inserted control worth was held constant at that value.

Sensitivity of the results to the shape of the scram curve has not been evaluated.

8.3.3 Results for Parameter Study

8.3.3.1 Comparison of Pump Coastdown Characteristics

The comparison of pump coastdown curves 1 and 2 is best shown by comparison of transients following simultaneous flow coastdown and scram. The results from this scram condition are typical of all the scram conditions studied.

We assume both the start of coastdown and the start of control rod drop occur at time, $t = 0$. Total control rod worth is 10\$.

Plots of sodium temperature versus time are shown in Figure 8-6. The solid lines are for coastdown curve 1; the dashed lines are for coastdown curve 2. The results at the vessel outlet assume an outlet mixing time constant based on 20% mixing of the core outlet sodium with the sodium in the tank. For no outlet mixing the temperature response at the vessel outlet would be congruent with the channel outlet temperature.

Maximum negative temperature gradients at various system positions are listed in Table 8-2.

FIGURE 8-5. FLOW COASTDOWN CHARACTERISTICS RELATIVE FLOW VERSUS TIME

FIGURE 8-6. COMPARISON OF THE EFFECT OF THE COASTDOWN CURVES ON THE SODIUM-TEMPERATURE RESPONSE

TABLE 8-2
MAXIMUM NEGATIVE SODIUM TEMPERATURE GRADIENTS
FOR SIMULTANEOUS COASTDOWN AND SCRAM
GRADIENT ($^{\circ}$ F/sec)

<u>Location</u>	<u>Coastdown Curve 1</u>	<u>Coastdown Curve 2</u>
Core Outlet	27	87
Plenum Outlet	20	80
Channel Outlet	9	62
Vessel Outlet (20% Mixing)	1	15

The significantly lower temperature gradients for curve 1 makes this coastdown characteristic more attractive than curve 2.

The difference in temperature response for the two coastdown curves can be explained qualitatively. For the rapid coastdown curve 1, the outlet temperature rises initially because the flow drops off rapidly (to 55% in 1 second) while the power stays up for a fraction of a second. Even after scram and rapid power reduction, the energy content of the fuel rod is high enough, the fuel time constant long enough, and the flow low enough to keep the sodium outlet temperature up for several seconds. The sodium outlet then falls due to the loss of power but eventually rises again since both the fission product decay power and the power generated from delayed neutrons are still high after the flow has coasted down to a small fraction of the initial value. At 30 seconds, the fission product power is 2.3% of the original power and the delayed neutron power is 1.4% while the flow rate is 4% of the original value; hence, the sodium outlet temperature has returned to nearly its original value. (Inclusion of natural circulation would increase the flow rate and reduce the sodium outlet temperature at 30 seconds.) The channel outlet lags the core outlet due to the heat capacity of the axial reflector, plenum, and shield, and to the transit time from core outlet to channel outlet. For the coastdown curve 1,

together with 20% mixing, the vessel outlet hardly changes temperature.

For coastdown curve 2, the flow rate remains sufficiently high during the first few seconds while the power is dropping rapidly. This essentially eliminates the initial rise in core outlet temperature which existed for curve 1. All coolant temperatures fall more rapidly for curve 2 since the flow rate remains higher. The core outlet does not rise back to a high value as was the case for curve 1 because the flow rate at 30 seconds is still 13% of the original flow rate.

8.3.3.2 Effect of Outlet Mixing

The results plotted in Figure 8-6 show the advantage of mixing of the sodium from the channels with a significant fraction of the sodium in the tank before leaving the vessel. The vessel outlet curves on Figure 8-6 assumed that the channel outlet sodium mixed with 20% of the tank sodium before leaving the vessel. For no mixing, the gradient at the vessel outlet would be equal to the gradient at the channel outlet.

8.3.3.3 Effect of Delay in Pump Coastdown

The effects on channel outlet temperature of delays in the pump coastdown of 1 and 2 seconds after initiation of scram (i.e.,

start of rod drop) are shown in Figure 8-7. Since scram starts at $t = 0$ sec, the power starts to drop almost immediately. Hence, all sodium temperatures drop rapidly until the pump coastdown starts. Therefore, based on consideration of initial temperature gradients, long delay times for pump coastdown are undesirable.

8.3.3.4 Effect of Time Between Scram Signal and Start of Rod Drop

The effects of channel outlet temperature of the time delays up to 0.25 sec between the scram signal and start of rod motion are shown in Figure 8-8. The sodium temperature response above the core is not very sensitive to this parameter for normal scram conditions.

8.3.4 Results for Response to Specific Accidents Followed by Normal Scram and Coastdown Curve 1

The sodium temperature response at the core outlet, channel outlet, and vessel outlet (for 20% mixing) are plotted in Figures 8-9 through 8-11 for the following cases. All cases shown are compared for coastdown curve 1.

- a. Loss-of-site power
Coastdown starts at $t = 0$.
Rods start to drop at $t = 0.15$ sec.
- b. Overpower scram
Scram signal at 110% overpower from 10%/second ramp.
Rods start to drop 0.15 sec after 110% power is reached.
Coastdown starts at the same time that rods start.
- c. High temperature scram
Scram signal at 30°F increase in temperature at channel outlet.

Temperature increase is caused by instantaneous flow decrease to 80% of normal flow. Rods start to drop and flow coastdown starts 0.15 sec after scram signal.

d. Pump startup in standby loop

Standby loop (sodium at 500°F) starts up at $t = 0$. Flow in standby loop increases instantaneously to $0.75 w_o$. Flow in other 3 loops (vessel inlet at 700°F) drops instantaneously to $0.75 w_o$. Scram signal at $t = 0$, rods start to drop and flow coastdown starts at $t = 0.15$ sec.

Variation in sodium temperature gradients between the four scram conditions is not large. The high temperature scram leads to the highest sodium temperatures, with the channel outlet rising an additional 20°F after the 30°F over-temperature scram is initiated. The core outlet rises a total of 70°F. The high temperature scram is the only case that leads to a significantly positive gradient at the vessel outlet, although this maximum gradient is only 3°F/sec.

The lowest sodium temperatures and largest negative temperature gradients result from the pump startup in the standby loop. However, the difference in gradient between this and the other transients shown is small. The maximum negative gradient at the vessel outlet is -2°F/sec for this scram condition.

The first two scram conditions—loss-of-site power and overpower scram—look similar except for a time delay to reach overpower in the latter case. Small differences in the sequencing of pump coastdown and scram and differences in the time dependence of the power between the two cases account for the small differences shown in the shape of sodium temperature response curves.

FIGURE 8-7. EFFECT OF DELAY IN PUMP COASTDOWN

FIGURE 8-8. EFFECT OF TIME BETWEEN SCRAM SIGNAL AND START OF ROD DROP

FIGURE 8-9. NORMAL ACCIDENT SODIUM TEMPERATURE RESPONSE AT CORE OUTLET

FIGURE 8-10. NORMAL ACCIDENT SODIUM-TEMPERATURE RESPONSE AT CHANNEL OUTLET

FIGURE 8-11. NORMAL ACCIDENT SODIUM TEMPERATURE RESPONSE AT VESSEL OUTLET

8.4 TESTING OF IHX SIMULATION MODEL

The IHX model for the dynamics simulation of the FFTF Backup Design is described in the Second Progress Report. Six nodes are used for the model and the spatial gradient term $\partial T / \partial x$ is approximated by central differencing at all internal points and by backward differencing at the primary and secondary outlets.

The model was tested for both steady state and transient solutions. The model is accurate at steady state for near equal flow rates in the primary and secondary. However, significant deviations from the correct solution at steady state were observed when the primary and secondary flow rates differed by a factor of two. The transient solution was stable for all transients tested, including severe step changes in flow rate.

8.4.1 Steady State Solution

For equal primary and secondary flow rates, the IHX model agrees with the exact solution to the IHX differential equation. For a factor of two difference in flow rate between primary and secondary, however, the steady state sodium temperature as a function of height in the heat exchanger, oscillates around the correct solution. Problems were run for which the primary flow was 0.5 of full flow and also for the primary and secondary at 0.1 and 0.2 of full flow, respectively. For these problems, the difference between the primary inlet and secondary outlet was close to the correct value, but the primary outlet was less than the calculated secondary inlet which is not possible physically. The use of 12 nodes for the case with 0.5 of full flow in the primary

gave results close to the exact solution. Hence, the number of nodes must be increased above 6 if greater accuracy for unequal flow situations is warranted.

8.4.2 Transient Solutions

The 6-node central differencing model appears to be stable for large step changes in both flow rate and temperature. This stability was observed for large differences in flow rate between primary and secondary. Although the transient response has not been compared to an exact transient solution, the temperature responses to large step changes in flow rate and temperature after long time periods asymptotically approach the solution calculated for steady state for the model. These step changes included a simultaneous change in primary flow rate from full to 0.1 flow and a change in the secondary from full to 0.2 flow. For the purpose of this transient evaluation, the results of this simplified model are adequate. A separate GEAP report will be issued on the transient model and results.

8.5 VESSEL HEAD COOLING

The objective in the design of the vessel top shield is to cool the shield with gas and provide a reasonable temperature gradient in the structure. A temperature gradient such that the reactor vessel wall and biological shielding would not undergo adverse thermal stresses.

The approach taken is to moderate the heat removal requirements on the gas system by providing a large thermal resistance on the underside of the head. This is accomplished by using layers of thin plates as reflective insulation on the bottom

surface to reduce the direct conduction over a large area of the head. The determination of the heat load through this, as well as the unavoidable conduction paths, is based upon taking a 1000°F drop through this region.

Four conceptual designs of gas passage configurations, shown in Figures 8-12, 8-13, 8-14, and 8-15, removing heat from the back (top) side of the reflective insulation have been analyzed to determine which show the most promise of providing adequate cooling. The analyses have been taken to the point of demonstrating the heat removal capability of the various configurations. Further detailed analysis is required to demonstrate that the heat can be transferred from the back side of the reflective insulation and the conductive heat paths into the gas stream.

The design criteria are as follows:

Sodium Temperature	1200° F
Cover Gas	Argon
Gas Gap	12 inches
Biological Shielding	24 inches of Stainless Steel or equivalent
Cooling Gas	Argon
Pressure Drop (Max.) (psi)	10
Gas Flow (Max.) lb _m /h	4×10^4

The assumptions made for the analysis are as follows:

Emissivity of Sodium	0.2
Emissivity of Stainless Steel	0.65
Emissivity Factor	0.18
Area Factor	1.0
ΔT Across Reflective Insulation (°F)	1000

Reflective Plates

Thickness (inch)	1/16
C _L to C _L of Plates (inch)	0.5

The heat transfer mechanism across the reflective plates were radiation and conduction transfer.* The heat load as a function of the number of reflective stainless steel plates is shown in Figure 8-16 resulting from radiation and conduction heat transfer for the total top shield inclusive of the small plug. The heat loss in Figure 8-16 excludes the conduction heat transfer of all the components submerged in the sodium pool.

The heat transfer by conduction from the components penetrating the top shield is summarized as:

Number	Heat Path	Transfer (Btu/h)
44	Control Rod Drive Extensions	6200
9	Instrumentation Tubes	875
450	Thermocouple Sheathes (0.020 Clad)	370
6	Closed Loop Support Structure	2365
	TOTAL	9810

Four conceptual designs of the top shield were established for evaluation. The heat load associated with 15 reflective insulating plates was selected as an acceptable value for the design basis. Other considerations in design were pressure drop and flow of argon coolant, location of hardware and capable heat removal potential.

*Free convection was investigated. The conduction transfer was the predominant of the two transfer mechanisms.

FIGURE 8-12. DESIGN 1. DOUBLE PASS; 60 DEGREE SECTOR WITH RECTANGULAR CROSS SECTIONAL AREA PASSAGES

FIGURE 8-13. DESIGN 2. TWELVE SECTOR, DOUBLE PASS, EXTENDED FINNED PASSAGE, RADIAL COOLANT PASSAGE

FIGURE 8-14. DESIGN 3. CONCENTRIC SPIRAL CIRCULAR DUCTS

GEAP-5641

FIGURE 8-15. DESIGN 4. CIRCULAR-TUBE, SINGLE PASS, ONE TUBE ASSEMBLY COOLING

FIGURE 8-16. HEAT LOAD VERSUS NUMBER OF REFLECTIVE PLATES

Design 1

Design 1 is a double pass, 60-degree sector with rectangular cross sectional area passages oriented in a circumferentially baffled passage. The coolant inlet is at the outer periphery, with a circumferential gas passage moving radially inward with each pass, the first pass at one elevation and the second pass at a lower elevation.

The velocity, film coefficient, pressure drop and heat removal capacity as a function of equivalent diameter of the coolant passage are shown in Figures 8-17 and 8-18.

Design 2

This design features 12 equally partitioned sectors, double pass, extended finned passage, radial coolant path of the cooling gas. The inlet is located at the outer periphery with the first pass traversing radially inward toward the center at a high elevation. The gas goes down vertically adjacent to the control rod drives to the second pass passage and traverses radially outward to the outlet at the outer edge.

Since the analysis is preliminary, analysis without extended surfaces was performed. Figures 8-19 and 8-20 show the pertinent parameters for the design.

Design 3

Four concentric spiral circular ducts constitute Design 3. The ducts are cased in lead, with inlets adjacent to the control rod penetrations staggered 90 degrees apart. The spiral section makes two 360-degree revolutions exiting at the outer periphery of the top shield.

The mean length of the duct per single spiral is approximately 62 feet in length. Pertinent parameters are shown in Figures 8-21 and 8-22 for this design.

Design 4

Design 4 consists of four identical circular-tube assemblies, single pass, one tube assembly cooling half of the top shield. The tube pattern is a series of 180-degree bends with straight intermediate sections traversing back and forth with inlet and outlet at the outer periphery. The tube assemblies are staggered 90 degrees apart at two different elevations cast in lead to improve the heat transfer characteristics.

Figure 8-23 illustrates the velocity and film coefficient for this design. The heat load is a function of the length of duct selected. Therefore, Figure 8-24 is presented in Btu/h-ft. To obtain the heat load, one tube assembly duct length should be used.

Figure 8-25 is the pressure drop per foot of a tube assembly.

8.6 DEVELOPMENT REQUIREMENTS

The thermal hydraulic development requirements for the back-up design were established. These have been incorporated in the Back-Up Design Development Program which at the time of this writing is in final preparation for release as a separate document for PNL review.

8.7 FUEL PIN DESIGN BASIS

A design basis for design of the fuel pin covering fuel physical properties, fuel and clad temperature limits, fission gas release, clad swelling and stress limits, pressure drop and heat transfer parameters was established and PNL concurrence obtained. Details are presented in the minutes of the meeting held March 14, 1968 at Advanced Products Operation, Sunnyvale, between General Electric and PNL personnel.

FIGURE 8-17. DESIGN 1. ΔP , VELOCITY AND FILM COEFFICIENT PARAMETERS AS A FUNCTION OF PASSAGE DIAMETER

FIGURE 8-18. DESIGN 1. HEAT REMOVAL CAPACITY

FIGURE 8-19. DESIGN 2. PRESSURE DROP, VELOCITY AND FILM COEFFICIENT VERSUS COOLANT PASSAGE HEIGHT

FIGURE 8-20. DESIGN 2. HEAT REMOVAL CAPACITY

FIGURE 8-21. DESIGN 3. PRESSURE DROP, VELOCITY AND FILM COEFFICIENT VERSUS COOLING DUCT DIAMETER

FIGURE 8-22. DESIGN 3. HEAT REMOVAL CAPACITY

FIGURE 8-23. DESIGN 4. VELOCITY AND FILM COEFFICIENT VERSUS COOLING DUCT DIAMETERS

FIGURE 8-24. DESIGN 4. HEAT REMOVAL CAPACITY

FIGURE 8-25. DESIGN 4. PRESSURE DROP PER FOOT OF DUCT AS A FUNCTION OF DUCT DIAMETER

SECTION IX

SODIUM-TEMPERATURE RESPONSE TO ABNORMAL TRANSIENTS

9.1 GENERAL (Table 9-1)

The sodium-temperature responses above the core and at the vessel outlet to the following abnormal transients were calculated and are currently being evaluated:

a. Reactor Scram with All Pumps Running - Scram rods start to drop at time, $t = 0$ sec; sodium coolant remains at full flow. b. Loss of Flow in One Loop with No Scram - Flow in one loop decreases to zero linearly in 2 seconds; flow in other two loops increases linearly to 1.16 normal flow in 2 seconds. c. All Pumps Trip with No Scram - All pumps trip and start flow coast-down at time, $t = 0$; coastdown characteristics being recommended for the FFTF Backup Design; no scram d. 10%/second Reactivity Ramp with No Scram Nor Flow Coastdown - 10%/sec reactivity ramp is initiated at time, $t = 0$; flow is held constant; no scram	a. The reactor vessel wall be of minimum practical thickness. b. Blast cylinders should surround the vessel wall with a gap between the vessel and the cylinders to permit volume to be created by vessel deformation. This minimizes the water-hammer effects on the head from vapor bubble expansion and reduces the equilibrium pressure in the vessel. c. Internal structure should be designed such that its radial rigidity is minimized - again to moderate water-hammer loadings on the head. d. No gas gap between the impact surface on the bottom of the head. Variations in level accommodated by a porous plate forming this impact surface. e. The head is held down by bolts tying directly into the foundation structure. f. The holddown bolts are sized such that they do not fail under: 1. The dynamic loadings of the "explosion" 2. The quasi-static equilibrium pressure in the vessel following the initial dynamic loading. g. Crush structure be provided beneath the vessel to absorb down-acting dynamics loads which would most probably fail the bottom of the vessel. h. Last ditch core cooling be provided by a meltdown pan below the crush

9.2 DESIGN BASIS ACCIDENT

A review of explosion test data and a discussion of the explosive loadings and containment design requirements was made. This is being included in the interface definition documentation since it influences the geometric interfaces in the reactor cavity, particularly the head holddown requirements. To summarize, the main features recommended are:

9.3 FUEL DAMAGE LIMIT CRITERIA

This criteria established tentative temperature limits for fuel and cladding to be used in Safeguards Evaluations of the FFTF Back-Up Core as given in Tables 9-1 and 9-2. Two types of off-normal conditions must be considered:

- a. Accident power transients in which both fuel and cladding temperatures may rise above the normal rated power conditions.
- b. Accidental flow transients in which cladding temperatures rise above the normal rated power conditions but fuel temperatures are lower than normal power operation.

Transients can be considered in the following categories:

- a. A credible accident is herein defined as one in which the probability of occurrence is sufficiently high that it is desired that no reduction in fuel life shall occur as a consequence of such an accident and the fuel temperatures are limited such that a margin of safety against cladding rupture exists.
- b. An improbable accident is one in which the probability of occurrence is sufficiently low that some fuel damage may be tolerated but the damage is limited such that a catastrophic condition is not approached.
- c. A catastrophic accident is one in which damage is sustained by the permanent structure of the reactor. The conservative upper bound on such accidents is the "Design Basis Accident."

The following list of accidents has been identified to date:

- a. Credible
 1. Power Transients (with scram)
 - (a) Control Rod Withdrawal
 - (b) Single Bundle Meltdown
 2. Flow Transients (with scram)
 - (a) Loss of Site Power
 - (b) Loss of One Primary Loop - Pump Seizure
 - (c) Loss of One Secondary Loop
 - (d) Small Leak in One Loop
 - (e) Non-Shut Off of Pumps Following a Scram Signal
- b. Improbable
 1. Power Transients - Control rod ejection at maximum credible rate with scram
 2. Flow Transients - Guillotine pipe failure in one primary loop with scram
- c. Catastrophic - Loss of flow without scram

9.4 INTERFACE DEFINITION

An approach to assuring that the safeguards requirements of the facility are adequately covered if the back-up core is inserted in the reference arrangement has been established, and a separate report covering this is in process.

TABLE 9-1

FUEL OVERPOWER DAMAGE SEVERITY CRITERIA

	Reactor Power	Description	New Fuel**		(100 MWd/kg) High Burnup Fuel**	
			Fuel T _{max} (°F)	Area Molten* (%)	Fuel T _{max} (°F)	Area Molten** (%)
Catastrophic Threshold	Danger Threshold	Potential DBA initiated by molten fuel injection into sodium.	-	70 [†]	-	50 [†]
	Uncertainty Allowance					
Improbable Accident	Safety Limit	Moderate fuel damage, threshold of cladding rupture, some fission product release.	-	40 [†]	-	30 [†]
	Safety Margin					
Credible Accident Limit	Design Limit (1.2 × normal S.S. power)	No damage, incipient fuel melting	5200	0	4930	0
	Operating Margin					
Normal Operation	Normal Operation	Normal Operation	4450	0	4200	0

* Based on PA10, Task C.

** These values are to be used for current evaluations but are subject to reevaluation as additional test data become available.

† The areas molten are recommendations for a rapid power transient, i. e., a rise and fall in overpower over a period of 1 second or less.

TABLE 9-2

CLADDING OVERTEMPERATURE CRITERIA
(Peak Fuel Temperature less than normal power)

Peak Local Temperature* (Local Hot Spot)
at Inside Diameter of Cladding (°F)

Improbable Accident Limit	1500°F	At low fuel power-sodium boiling controls
Credible Accident Limit	1300°F	No damage
Normal Operation	1200°F	

* These are tentative values subject to change as irradiated cladding damage test data becomes available.

REFERENCES

1. Riley, D. R., "Fast Flux Test Facility Backup Design Project, Second Quarterly Report, September—November 1967," GEAP-5614, April 1968.
2. Clabaugh, W., and Kendall, W., "FFTF Phase II Study, Part I, A Conceptual Design of the FFTF Reactor System Interim Cost Estimate," GEAP-5422, July 1967.
3. Shure, K., "Fission Product Decay Energy," Bettis Technical Review, WAPD-BT-24, December 1961, pp 1-17.

Petrov, Yu. I., "Gamma Radiation from U-235 and Pu-239 Fission Fragments," J. Nuc. Energy, Part A, Reactor Science 12, 129-32 (1960).
4. Johnston, K. (AWRE-UK), "A Calorimetric Determination of Fission Product Heating in Fast Reactor Plutonium Fuel," J. Nuc. Energy, Parts A/B, 19, 527 (1965).
5. Battat, M. E., Dudziak, D. J., and Hicks, H. R., (LASL), "Fission Product Decay from Fast Fission of Pu-239," pp 524-525, Transactions of the American Nuclear Society 1967 Winter Meeting, Vol. 10, No. 2.

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