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AEC RESEARCH AND
DEVELOPMENT REPORT
OCTOBER 1967

FAST FLUX TEST FACILITY BACKUP DESIGN PROJECT FIRST QUARTERLY REPORT MAY – AUGUST 1967

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PACIFIC NORTHWEST LABORATORY AND
GENERAL ELECTRIC COMPANY
PREPARED FOR
U.S. ATOMIC ENERGEY COMMISSION
BY GENERAL ELECTRIC

ADVANCED PRODUCTS OPERATION
GENERAL ELECTRIC
SUNNYVALE, CALIFORNIA

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SECTION I

INTRODUCTION

This quarterly report documents the work conducted on the FFTF Backup Design during the period of May 16, 1967 through August 31, 1967.

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 accomplishment during the quarter was the completion of the conceptual design studies on the backup design. Concept study design results were reported in two topical reports: "The Selection of a Piped Looped System for the Fast Flux Test Facility" (Reference 1), and "Conceptual Design of the Backup Reactor System for the Fast Flux Test Facility" (Reference 2), published in draft form on August 14, 1967. Because the above noted topical reports describe in detail most of the work accomplished to date, the results of the reports will be summarized herein. Only those topics accomplished during the reporting period and not covered in the topical reports will be covered in depth in this quarterly.

The conceptual design work reported as noted above and summarized herein resulted in the following major features.

1. A total flux of 0.9×10^{16} with a Doppler (sodium-in) of -0.0048 T dk/dt.

2. Elevated primary coolant piping system intended to enhance safety and provide redundant system natural circulation decay heat removal during normal as well as accident conditions.

3. Primary control by means of reflector control rods around core periphery. Reflector-control allows system to remain activated and undisturbed during refueling. Reflector control increases the area over core for instrumentation functions.

4. A small diameter central shield plug with support structure extended to top of core, provides backup fuel assembly holddown and the potential for 100 percent driver fuel instrumentation.

5. Small diameter refueling access portion in top shield allows access to the core, with a minimum of complication.

6. Driver fuel handling is accomplished under sodium. Fuel is transported to refueling cell storage in natural convection, cooled finned thimbles.

7. A cooled, protected, safety tank surrounds the reactor vessel and provides containment for coolant leakage.

8. Blast protection is supplied by radial cylindrical structure, top shield stretch holddown, and crush structure beneath the vessel. Post-incident collection of debris and decay cooling is provided.

The conceptual mechanical and structural design is consistent with an outlet temperature of 1200°F, but operation of the reactor much beyond 1000°F coolant temperature would not be recommended without additional information on large system mass transfer and corrosion characteristics and their dependence on impurities, and thermal effects on radiation damage.

The conceptual design is believed to provide a system that will satisfy the FFTF objectives, however; it has been recognized that there are a number of areas which will require further information and development. Those items noted in the studies to date requiring development are as follows.

1. Knowledge of quantitative effects of radiation on structural materials in and near the core.
2. Reliable in-core instrumentation other than thermocouples.
3. Large system characteristics at 1200°F outlet temperatures (noted above).
4. Normal engineering data development in such areas as channel flow characteristics, reflector control worth, control drive design, etc.

With the completion of the FFTF backup conceptual design, and contingent on PNL concurrence with the concept, present work is being directed toward the next major contractual commitment of establishing, in cooperation with PNL, an outside envelope and a consistent set of functional specifications for the interface between the reference and the backup designs. Definition of the backup design engineering development requirements is also in progress.

SECTION II

PLANT PERFORMANCE

2.1 GENERAL

Figures 2-1 and 2-2, showing the plant plan and elevation, respectively, for the FFTF backup design and the Plant Performance, Table 2-1, are the result of the conceptual studies conducted during the quarter which resulted in the recommended FFTF backup design reported in detail in Reference (1).

2.2 PLANT PERFORMANCE TABLE

Test Loops	
Closed Loops	6
Location	1 central 3 midcore 2 periphery
Open Loops	2
Flux - peak level	0.9×10^{16}
fraction above 0.1 MeV	0.66×10^{16}
Loop Cells	
Number	6
Size	8000 ft^3

2.3 REACTOR

2.3.1 Core

Type	Open lattice, vertical
Power	400 MWt
Dimensions	
Length	33 in.
Equivalent diameter	42 in.

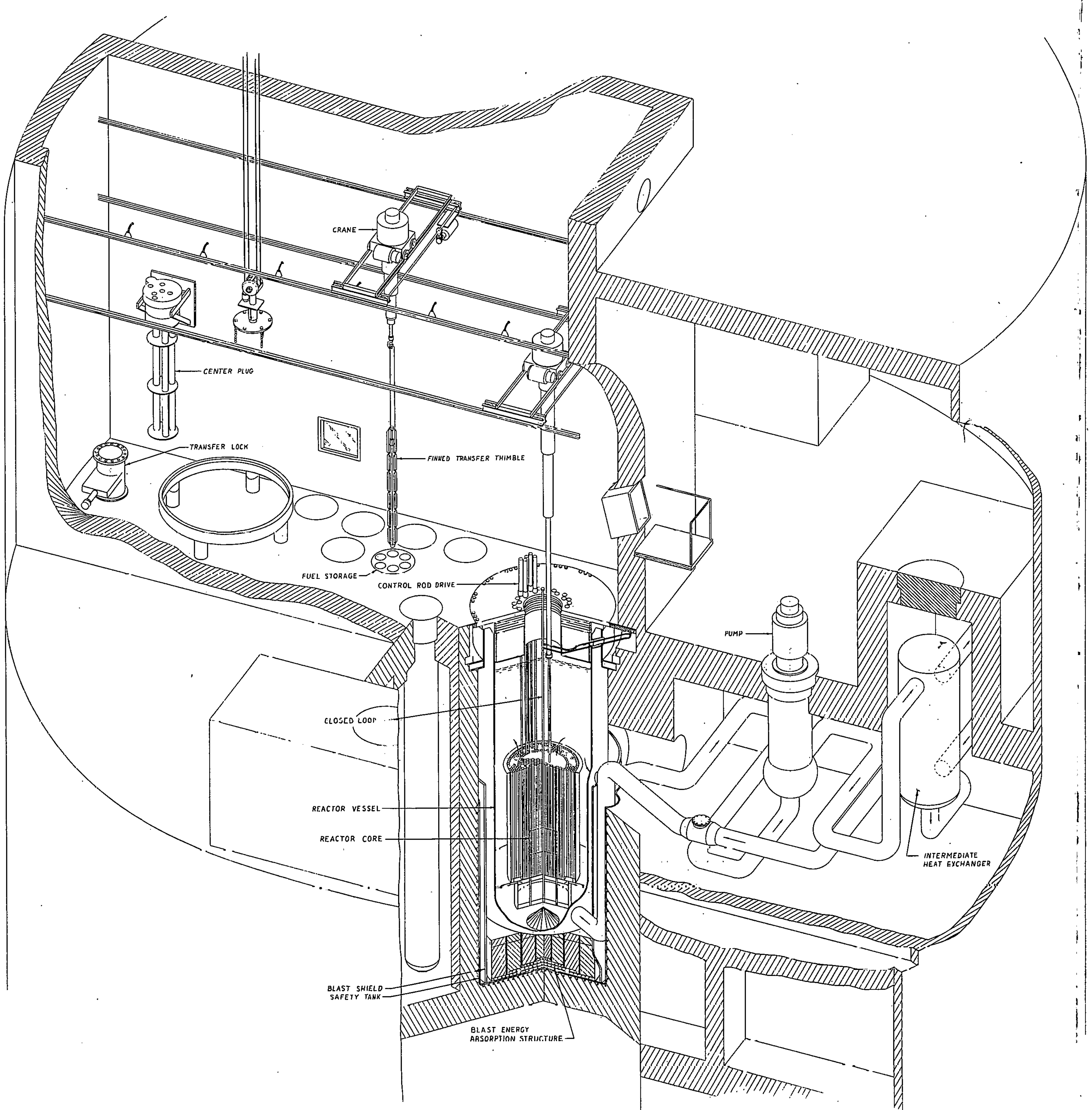

FIGURE 2-1. CONCEPTUAL REACTOR SYSTEMS ARRANGEMENT

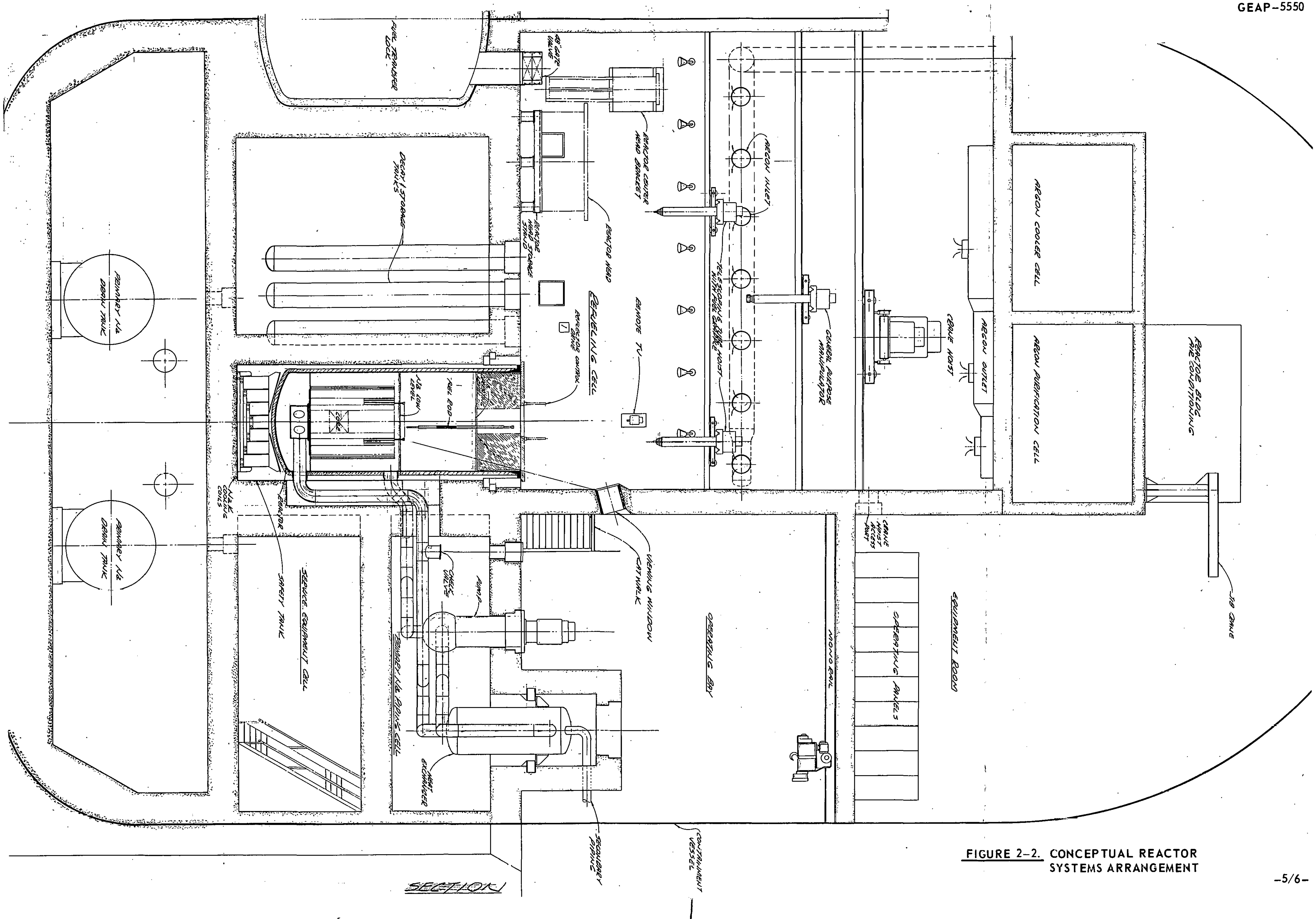

FIGURE 2-2. CONCEPTUAL REACTOR SYSTEMS ARRANGEMENT

Composition:	
Fuel	PuO_2-UO_2
Structure	Stainless steel
Coolant	Sodium
Core Channels	
Fuel	127
Closed Loops	6
Open Loop	2
Backup Control	3
Total	138
Fuel Channel	
Pins/Channel	127
Dimension	
Across Flats	3.375 in.
Channel Material	Inconel 800
Fuel Pins	
Diameter	0.210 in. o.d.
Cladding material	Type 316 stainless steel
Fuel Cycle	90 days (Preliminary - based upon 0.8 plant factor and 70,000 MWd/Te 20% of core elements are replaced each cycle.)

2.3.2 Reflector

Dimension	
Top	12.0 in.
Bottom	12.0 in.
Radial	10.5 in.
Material	Nickel

2.3.3 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$
Total worth in reflector	25$
Material	B_4C
Drive	GE compact mechanical drive
Backup Control	
Location	In core
Number	3
Worth	1$ to 2.7$ ea. 3$ to 8$ total
Material	B_4C
Drive	Fluidic system or GE compact drive

2.3.4 Vessel

Diameter	16.0 ft
Thickness	0.75 in.
Material	Stainless steel Type 304

2.3.5 Coolant

Flow	15.1×10^6 lb/h
Inlet temperature	700°F
Outlet temperature	1000°F
Core pressure (nominal)	112 psi

2.3.6 Cover Gas

Type	Argon
Pressure - operating	±10 in. w.g.

2.3.7 Temperature

Peak channel	(120% operation overpower)	
Fuel - maximum	5000°F	
Fuel surface - maximum	1870°F	
Cladding - maximum	1120°F	
Sodium - maximum	1060°F	
	Design	Operation
Grid plate	1000°F	700°F
Reactor vessel	1200°F	1000°F
Vessel head	1000°F	200°F

2.3.8 Power Distribution

Radial	1.4
Axial	1.2
Total	1.7

2.3.9 Hot Channel Factor

Total statistical	1.1
Overpower allowance	1.2

2.4 SYSTEM

2.4.1 Primary Coolant

Number of loops	3
Rating	200 MWt/ea.
Piping	
Material	Stainless steel Type 304
Size - reactor outlet	24 in.
- reactor inlet	20 in.
Heat exchanger rating	200 MWt/loop
Coolant loop unavailability	2.2%
Temperature	
Design	1200°F
Operation	1000°F

2.4.2 Containment Vessel

Type	Cylindrical
Diameter	135 ft
Thickness	1.1 in.

2.4.3 Emergency Coolant

Type	NaK
Heat load	~4 MWt

2.4.4 Vessel Cavity

Gas	Nitrogen
Pressure	±10 in. w.g.
Liner	Steel

2.4.5 Equipment Cell

Gas	Nitrogen
Pressure	±10 in. w.g.

2.4.6 Loop Cell

Gas	Nitrogen
Pressure	±10 in. w.g.
Floor liner	Steel

2.5 HOT CELL

2.5.1 Size

Length	72 ft
Width	22 ft
Height	78 ft
Volume	124,000 ft^3

2.5.2 Transfer Cell

Maximum Size Transfer	
Diameter	28 ft
Length	37 ft

2.5.3 Fuel Handling Method

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

2.5.4 Fuel Storage Method

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

2.5.5 Refueling Cell Atmosphere

Atmosphere	Argon
Design temperature - accident	250°F
Design operating temperature	100°F
Design pressure -	
Estimated maximum	10 psig
Estimated minimum	-6 in. w.g.
Operating pressure	
High	-2 ±1 in. w.g.
Low	+2 ±1 in. w.g.
Leakage (estimated maximum)	1% cell volume/day at design pressure

2.5.6 Refueling Time

Shutdown - preparation	8
Reactor servicing	44
Driver refueling	42
Control rod replacement	8
Startup preparation	8
Total	110 h

2.5.7 Major Equipment

Bridge crane mounted telescoping tube hoists with fuel grapple mechanisms	2
Bridge crane mounted general purpose manipulators	2
Master-slave manipulators	3
Bridge crane	25/100 T
Man access equipment	
Decay and storage tanks	
Spent fuel transfer thimbles	

2.6 SAFEGUARDS

2.6.1 Containment — Double

2.6.2 DBA Energy Release

Estimated maximum	1500 MW-sec

2.6.3 Credible Accidents

Flow coastdown

Loss of single coolant loop

Piping leakage (including guillotine pipe failure)

Single fuel bundle meltdown

Control rod withdrawal

Refueling bundle drop-in

2.7 NUCLEAR

2.7.1 Core Composition

Fuel channel volume fraction	
Fuel	0.359
Steel	0.079
Inconel 800	0.113
Sodium	0.449
Experimental loop volume fraction	
Steel	0.5
Sodium	0.5
Backup control rod volume	
Steel	0.33
Sodium	0.67

2.7.2 Midcycle Fuel Composition (Atom %)

Uranium	67.6%
Plutonium	28.7%
Fission products	3.7%
Plutonium isotopic concentration	
Pu-239	63.7%
Pu-240	31.5%
Pu-241	3.3%
Pu-242	1.5%
Uranium isotopic concentration	
U-235	0.2%
U-238	99.8%

2.7.3 Reactivity

Doppler	
T-dk/dt (sodium in)	-0.0048
T-dk/dt (sodium out)	-0.0035
Na void	
Maximum positive	+1.5$
Total core	-4.0$
Maximum fuel bundle worth	+2.2$
Fuel meltdown (top 1/3 into mid 1/3)	0.5$

2.7.4 Fuel Cycle

Discharge burnup	70 MWd/kg
Number of batches	5
Burnup/cycle	14 MWd/kg
Fuel cycle period	90 days
Operating cycle	72 days

2.7.5 MeV/Fission — 215

Fraction energy absorbed in fuel	0.91

2.8 RADIATION LEVELS

2.8.1 Dose Rate During Operation

Operating floor	0.5 mR/h
Hot cell floor	50 mRem/h

2.8.2 Equipment Cells

From Na-22 (in working area)	1 R/h

2.8.3 Neutron Exposure

(nvt 1 MeV, 20 yrs)

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

Emphasis during this report period has been upon selection of a piped loop primary coolant system concept. Selection of the primary coolant system has involved the establishment of general design criteria compatible with the testing requirements, and consistent with the design of the vertical core, hot cell refueling concept being pursued in the FFTF backup design.

General design criteria were established in the areas of system safety, performance and feasibility, availability and maintenance, capital cost, and design goals. Development of the detail design criteria is discussed in the loop selection draft report Reference (1), Section 3.0.

Following the development of the general design criteria, several loop-type systems were reviewed conceptually and three of the most promising concepts were selected for further study. Conceptual studies were completed in enough depth to determine feasibility, relative safety characteristics, thermal-hydraulics, component size, piping size and flexibility, and overall building layout.

The three concepts were compared in a relative manner based upon the criteria for safety, performance, reliability and cost. The most promising system was selected for further detailed study. The selection process for the piped loop system is described in detail in Reference (1).

3.2 SYSTEM ARRANGEMENT

The plant and coolant systems arrangement (Figures 2-1 and 2-2) for the plant is described in the report "Conceptual Design of the Backup Reactor System for the Fast Flux Test Facility," Reference (2).

Basically the reactor, primary coolant systems, closed loop cells, sodium service system, refueling cell and auxiliary systems are all housed within a 130-foot diameter cylindrical containment building. The features of the reactor system include: (1) a vertical compact core located within the reactor vessel, (2) a remote manual refueling system operating in the overhead refueling cell, and (3) a piped loop heat transfer system elevated above the reactor core.

The main coolant system consists of three main primary piping loops elevated above the core. Each loop is rated at 50 percent power, or 200 MWt. Three loops may be operated at one-third total power each, or two loops may be operated at one-half power with one shut down. The selection of three main heat transfer loops, each rated at 200 MWt was based upon availability studies conducted by General Electric Research and Development Center consultants. The detail results of this study are included in Reference (2), Section 9.1.

3.3 SODIUM SYSTEM DESIGN

Design effort has been initiated in the areas of thermal hydraulics, and structural design of the main primary coolant system. System pressure drop considerations, and piping flexibility analysis have been emphasized.

3.3.1 Primary Piping Selection

The primary piping size has been selected on the basis of providing a high net positive suction head (NPSH) at the pump suction, to minimize pump costs while providing discharge piping sized to result in reasonable loop pressure drop. Because the pressure available at the pump suction has an appreciable influence upon pump costs, it is desirable to minimize the pressure losses from the reactor to the pump while maintaining atmospheric reactor cover gas pressure. For the 1200°F target design conditions, the following piping size has been tentatively selected. Further study is necessary to determine what economic incentives there are for increased pipe sizes. The selected piping results in a maximum loop pressure drop of about 40 psi with a pump NPSH of about 38 feet.

Pipe Length	System Temperature (°F)	Size (in.)	Schedule	Pressure Drop psi	Pressure Drop Feet of Sodium
1. Reactor to Pump	1200	24	10	3.7	10.9
2. Pump to IHX	1200	20	30	6.3	18.2
3. IHX (estimated maximum)	1050	--	--	15.0	43.0
4. IHX to Reactor	900	20	10	14.0	38.6
	TOTAL LOOP ΔP			39.0	110.7

3.3.2 Primary Piping Flexibility Analysis

Preliminary piping flexibility analysis has been performed to demonstrate feasibility of the present system configuration as described in the conceptual design report, Reference (2). The piping layout used for analysis is illustrated in Figures 2-1 and 2-2 of this report.

Initial computer calculations were performed for the 1000°F design operating conditions. Results of the flexibility analysis indicate the expansion stress is well within the allowable stress range of the code for pressure piping ASA B31.1. Also, piping reaction loads appear to be acceptable, indicating feasibility of the present arrangement at the 1000°F operating condition.

Preliminary analysis has been initiated for the 1200°F target design conditions listed below.

PIPING - TARGET DESIGN CONDITIONS

	Temperature (°F)	Pressure (psig)	Pipe Size (in.)	Schedule
1. Reactor - Pump	1200	20	24	10
2. Pump - IHX	1200	200	20	30
3. IHX - Reactor	900	200	20	10

Initial results indicate that for the 1200°F target design condition, the majority of the loads and piping stresses are within an acceptable range. However, because of reduced material properties, increased thermal expansion, and higher design temperature, some of the reaction loads applied to components appear to be marginal. Further structural analysis will be performed to determine what adjustments are necessary to make the piping configuration compatible with the target design conditions.

SECTION IV

REACTOR VESSEL AND INTERNALS

4.1 GENERAL

Results of the work accomplished in the reactor vessel and internals area during this quarter are reported in detail in Section 4.0 of Reference (2). The concept configuration is shown in Figures 4-1 and 4-2. The vessel and top shield plug of the FFTF backup design are quite different from that described in the Project Agreement 47 interim report[(3)] because of the decision to use the piped system for the FFTF reference design concept which locates the primary pumps and intermediate heat exchangers external to the reactor vessel. The following sections summarize the work involved in defining the conceptual design of the FFTF backup reactor vessel and internals.[(2)]

The design basis for the backup reactor system is an open lattice vertical core compatible with a piped primary coolant loop system. Refueling is accomplished by remote-manual methods over an open pool. The development of the reactor concept has evolved with emphasis upon the following objectives.

1. Provision for 6 closed loop and 2 open loop test positions.
2. Provide a peak flux near 1×10^{16} at 400 MWt power.
3. Provision for 100 percent driver fuel temperature instrumentation capability and future modifications for other instrumentation.
4. Compliance with assumed safeguards requirements.
5. Quick, effective, safe refueling and replacement of closed and open loop tests.
6. Conservative design philosophy: feasibility within framework of known technology and anticipated maintenance.
7. Plant operation at 1000°F. Mechanical and structural design target, 1200°F.
8. High degree of plant availability consistent with testing requirements.

Conceptual design studies of the reactor system evolved several alternate arrangements which were evaluated in terms of the general design objectives. Evaluation of the alternates has revealed several features of the reactor system which improve the effectiveness of the plant as a test reactor facility.

The major features of the concept are:

1. Vertical test and driver fuel assembly orientation.

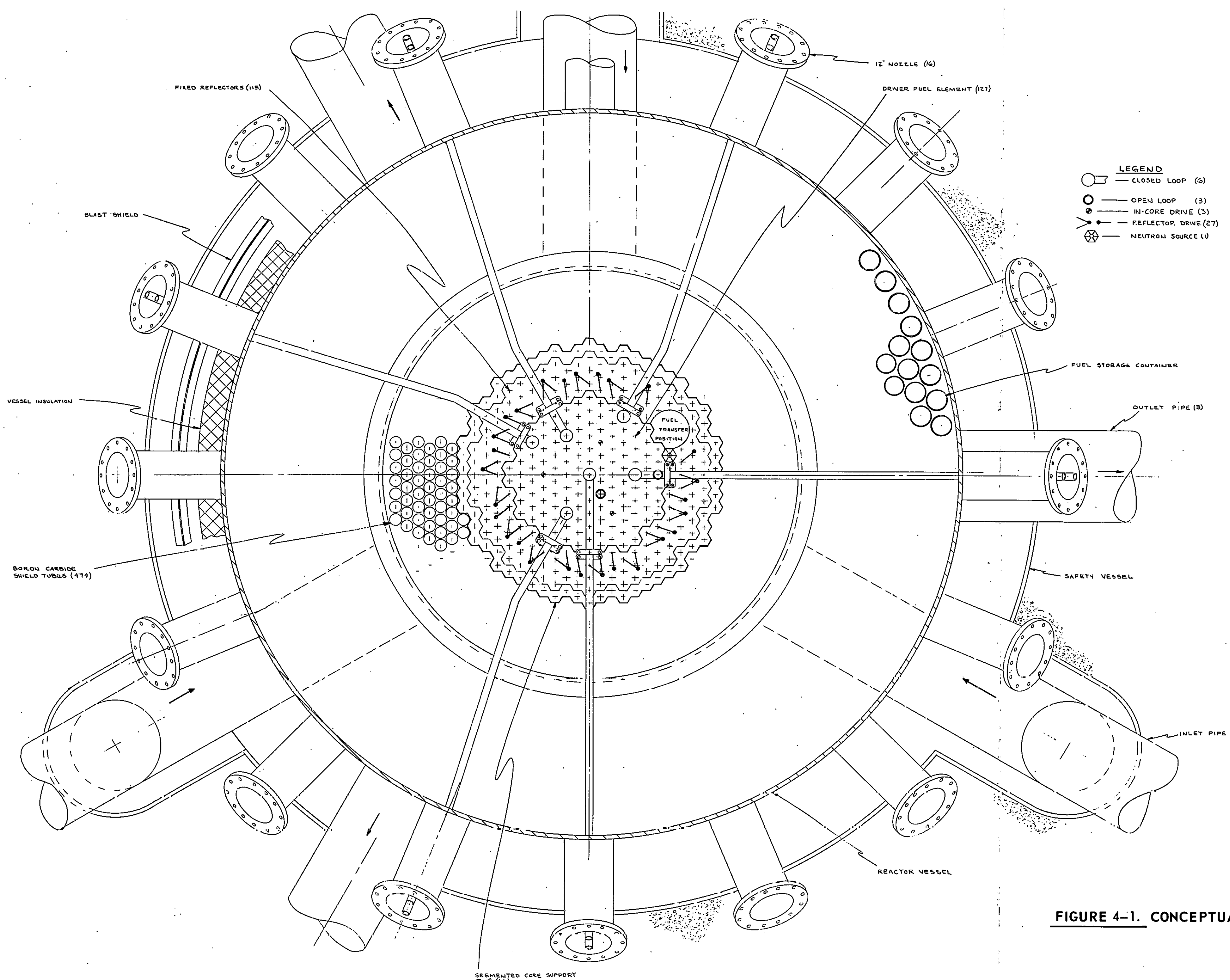

FIGURE 4-1. CONCEPTUAL REACTOR PLAN

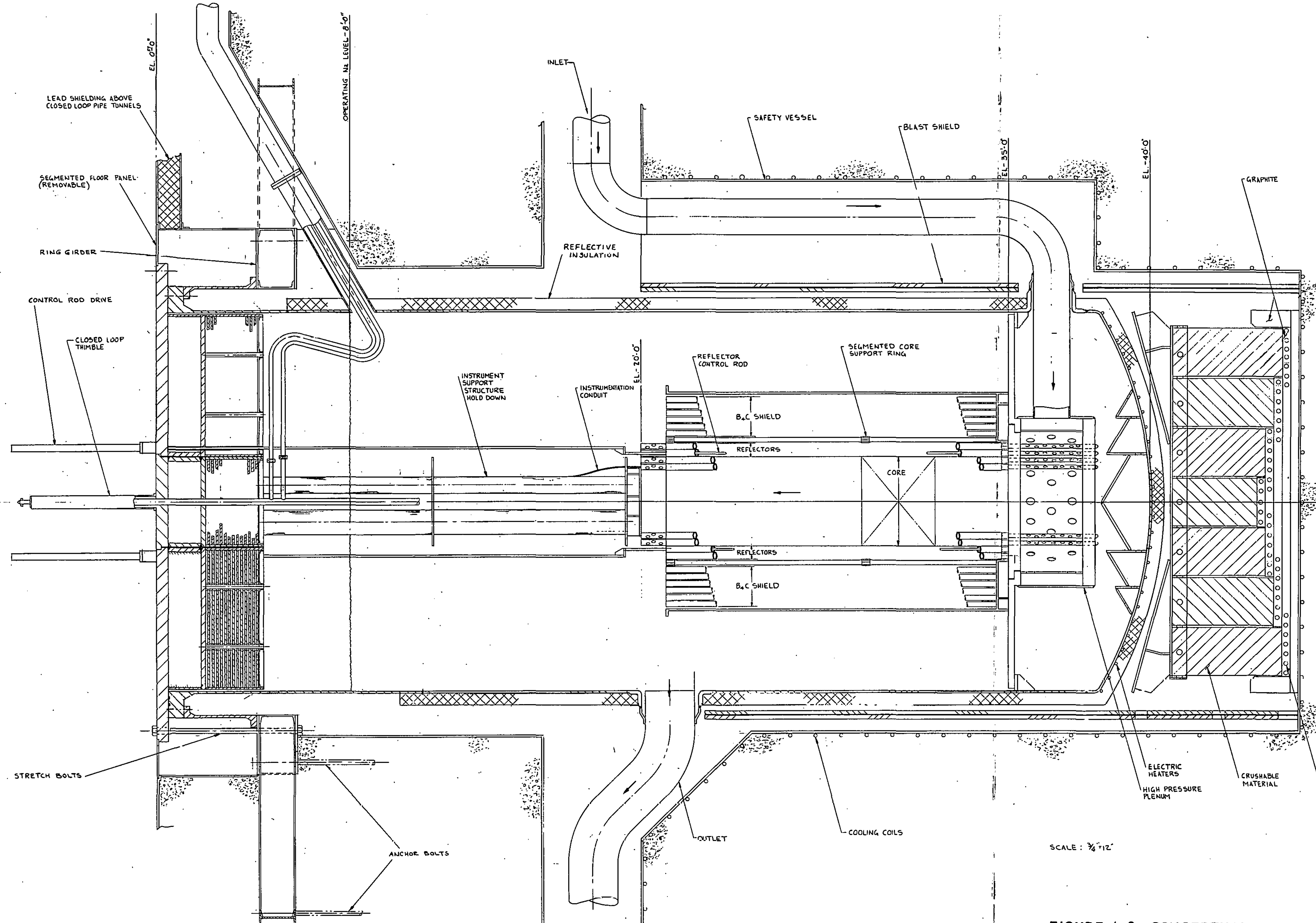

FIGURE 4-2. CONCEPTUAL REACTOR PLAN

2. Test, driver fuel, and reflector channels are the same size for flexibility in core arrangement.

3. Primary control is achieved by movement of the nickel reflector segments surrounding the core. This allows most of the primary control to remain in place and be activated during normal refueling. Backup control is accomplished with in-core poison rods.

4. A small central plug within the main head plug accommodates 100 percent driver fuel instrumentation and allows core access for efficient driver fuel and experiment replacement, a situation enhanced by removing the primary control rods from the core and central plug area.

5. Refueling is accomplished by the remote manual method. After the central plug is removed, fuel is grappled through the sodium pool and transferred through the pool to a finned thimble. The thimble and fuel are withdrawn and transferred to a decay tank (storage pool) within the refueling cell. During transient, fuel decay heat is removed by natural convection within the thimble and is rejected to the cell atmosphere by natural convection and radiation.

6. Design Basis Accident (DBA) blast protection is obtained by a radial cylindrical blast shield which protects the safety vessel, a top shield holddown which utilizes stretch rods, and a crushable structure beneath the vessel.

7. Heat removal following the DBA is accomplished by means of a protected vessel cavity cooling system.

4.2 GENERAL ARRANGEMENT

The reactor vessel arrangement is shown in Figures 4-1 and 4-2. The reactor is essentially an open lattice, vertical core operating within a vertical right circular vessel with a free surface pool of sodium. The reactor core consists of PuO_2-UO_2 fuel, with nickel reflector, and shielded by boron carbide (B_4C). The reactor vessel and internals, with exception of the fuel, are designed for an operating temperature of 1200°F, but are anticipated to be operated at 1000°F.

The reactor vessel is supported from its top flange on a skirt and ring girder, which is supported by the building structure. A shield is provided between the refueling cell and the reactor to provide sufficient biological shielding to permit personnel access into the refueling area during reactor operation. The plug shield is primarily comprised of a steel structure which extends from the refueling cell floor into the reactor vessel to a point just above the sodium coolant level. The top shield plug is provided with a small plug in the center for refueling operations. Surrounding the center plug are the control rod drives.

The sodium coolant enters the vessel near the bottom head through three 18-inch pipes and is directed into a closed plenum at 150 psi. Coolant flows from the lower plenum at 112 psig through each individual fuel channel inlet region, then turns upward passing the lower shield region, lower reflector, 33-inch fuel region, upper gas plenum, reflector, and upper shield, exiting into the upper sodium plenum as it passes the coolant instrumentation. Flow from the upper plenum is directed radially outward through three 24-inch outlet pipes located in the vessel wall below the elevation of the fuel channel extensions. Each loop has a 200 MWt capacity. The reactor core consists of 127 fuel assemblies, 3 safety rods, 6 closed loops, and 2 open loops, for a total of 138 core positions. Design concept of the fuel assemblies has not changed materially from the December concept. The reactor control is provided by twenty-seven control rod drives which move forty-three reflector rods in the first two rows of the nickel reflector. All core and reflector positions are cooled by sodium flow from the high pressure inlet plenum.

Surrounding the reactor core and reflector is a segmented core support ring which provides lateral restraint for the core and reflector elements. The segmented core support ring occupies 66 hexagonal positions. The segments of the ring each fit into a blind hole in the grid plate support ring. Support ring segments are completely interlocked with each other at the top, but are not locked to the reactor vessel.

The neutron shielding consists of 474 boron carbide filled rods, located around the periphery of the segmented core support ring, providing approximately twenty inches of shielding. The neutron shield is enclosed by a stainless steel cylinder with a top flange which serves as a clamping ring for experiment instrument leads. Outside of this cylinder storage facilities are provided. While normal procedures call for direct removal and storage elsewhere of expended core components these storage facilities are intended for temporary removals and for use in emergencies.

4.3 DBA BLAST STRUCTURES AND POST-INCIDENT HEAT REMOVAL

The DBA is currently considered for design purposes as an energy release in the core of 1500 MW-sec of mechanical energy. It is assumed that an upper bound on the damage potential is achieved by considering this energy release is simulated by a TNT explosion yielding the same mechanical energy. (1 lb TNT = 2 MW-sec.)

For a TNT explosion, approximately half of the energy is released in a shock wave and the remaining half is stored in the highly compressed explosion product gas bubble which expands and loads the structure. Safeguards requires that the damage effects of the explosion be limited to ensure that containment leakage barriers are maintained; i.e., an inner barrier being a liner on the refueling cell, primary equipment cell and reactor cavity walls, and an outer barrier being the containment shell. The design approach taken is to protect the refueling cell and reactor cavity walls from damage as a result of the DBA.

In keeping with this, the system is also designed to control the accumulation of fuel and debris and provide decay heat removal capability such that unreasonably high pressures are not generated by gross sodium boiling. The first containment barrier must not be breached, to cause an increase in the containment release from the building. Conceptual design consideration has been given in the following areas.

1. Shock wave damage control on the vessel walls, head and bottom blast pressure on the vessel walls, head and bottom (including water hammer effects on the head shield plugs)

2. Post-incident decay heat removal and safety tank protection.

4.4 REACTOR CONTROL SYSTEM

The control system consists of a primary control system worth 25$, located in the reflector position, and a backup control system, worth up to 8$ and located in the in-core position. By placing the primary control system on the periphery of the core and using movable reflectors with poison followers, sufficient space is available to use a larger number of low worth rods. Further analysis will be made to determine the exact worth of the reflector positions and the minimum number of in-core positions required for the backup system. Figures 4-1 and 4-2 illustrate the conceptual arrangement of control rod positions and drives.

Close centerline-to-centerline space requirements of the primary reflector control rods illustrates a need for a small size drive. Effort was applied to layout and review of the GE compact single screw drive concept for adaptation to FFTF requirements. The layout showed the compact drive can meet the space requirements, including provisions ro reduce drive maintenance time to enhance reactor availability. With the compact drive, the reactor control rods were located with sixteen pairs and eleven single reflector control rods in a circular layout in the inner row of reflector channels. (See Figure 4-1.) Where spacing permitted, the hi-worth single control rods are attached to a single drive mechanism. The low-worth rods are paired by way of couplings which attach two rods to one control rod drive and shaft. Thus, wherever possible the closest balance of the rod worth driven by each control rod drive is maintained.

Backup poison-safety control is provided by fluidic driven control rods mounted in the central core zone. The reference fluidic drives were described in Reference (3). An alternate arrangement utilizes mechanical drives mounted on top of the small central plug. The alternate mechanical drive system adds to the equipment attached to the central (refueling) plug, and lengthens the overall height of the plug and integral instrument support. A modified compact GE drive would be used to actuate the safety rods should the mechanical drive alternate be utilized.

4.5 REACTOR NEUTRON SHIELD RODS

Radial neutron shielding is provided for the reactor vessel. It consists of 474 shield rods located around the core outside of the reflector. The shielding is provided by boron carbide (B_4C) contained in a 3-inch o.d. × 0.090 wall stainless steel tubes. The tubes are located radially by a hole drilled into the support plate and are held in place by gravity. Lateral support is provided a stainless steel shell (~1/2 inch thick) placed around the periphery of the shield rods. The shield cooling is provided by flowing sodium. The peak fast neutron fluence (>1 MeV) based on a twenty-year life at reactor mid-plane are summarized below:

Radially:

1×10^{23}	Reflector (Ni)
2×10^{22}	Segmented core support ring
2×10^{21}	Neutron shielding
7×10^{17}	Vessel wall
7×10^{16}	Cavity liner

Axially:

8×10^{19}	Grid plate
3×10^{10}	Vessel head

The axial shielding is provided by six 19-3/4-inch long B_4C filled tubes in each fuel element channel below the lower nickel reflector and by six 49-1/4-inch long B_4C filled tubes located above the fuel rods. The shielding is illustrated in the figures.

Present knowledge of the effects of radiation on the material properties of the vessel components is limited, particularly about damage caused by high energy neutrons at high temperatures. However, the data available would indicate that degradation in the ductility and creep strength in the reflector, core support ring, and the neutron shield elements (caused also by helium generation) may require their replacement after a few years of reactor operation.

4.6 INSTRUMENTATION

Protection of the core integrity, and prevention or detection of unsafe operation or malfunction is considered the overriding objective of the core instrumentation. Nuclear startup and operating instrumentation will be located external to the neutron shield and probably in the biological shielding.

With the present configuration, it appears highly probable that access tubes can be provided to all driver fuel channels for insertion of an instrumentation package. Instruments such as thermocouples, ion chambers, fission chambers, thermopile or flow probes can utilize these tubes. Locating the instrumentation package for each driver fuel channel in the common plug structure aids in installation and replacement. From the central plug structure, instrumentation leads will be carried to a connector accessible to the in-cell manipulators and capable of remote or manual disconnect, thus allowing for instrumentation fixed in the holddown structure that does not require movement or disconnection in the high temperature sodium environment. Further in depth driver fuel instrumentation studies are reported in Section VI of this report.

Instrumentation for the closed loops will depend on the requirements of the specific experiment. All of a loop's in-core-power or instrumentation leads will pass through a common vessel penetration plug so the vessel seal may be made in one operation. Sodium sample or gas leads, if necessary, will penetrate the vessel through a nearby nozzle. The instrumentation leads terminate in a hermetically sealed connection, sealed to the nozzle. This leaves the mating half of the connector outside the vessel where the connection can be made and broken manually or remotely.

Open loop experiments may occupy almost any core position. The instrumentation facilities must be made flexible enough to accommodate these positions. Allowance must be made for clearance in the fuel holddown structure for the instrumentation leads. This structure cannot be built to accommodate all foreseeable open loop instrumentation at any one time, and may have to be modified during its life to meet the needs. The reference core shows the open loops adjacent to the closed loop piping. With this scheme instrumentation leads in addition to normal driver fuel instrumentation can be brought over the core directly under the closed loop piping and utilize piping supports. There are a sufficient number of closed loop pipes to provide a good deal of flexibility in choosing the open loop positions. The instrumentation penetration may be installed through the vessel wall or one of the vessel heads. Space limitations plus the necessity of disconnecting the lead to remove the central head make it undesirable to have this penetration in this head. A number of nozzles (16) are provided in the vessel wall to accommodate such instrumentation. The penetration plug and connector can be similar to that used on the closed loops.

4.7 THERMAL HYDRAULICS

Conceptual core design is based upon a 3-3/8-inch hexagonal fuel channel containing 127 0.21-inch diameter fuel pins. This was selected during the previous study and reported in GEAP-5422 to give a reasonable bundle worth (2.2$) (from Safeguards considerations) and sodium void fraction (from pressure drop and fuel temperature considerations). The present arrangement of channels is considering a core composed of 127 fuel channels with 2.75-foot active fuel length which results in a nominal core pressure drop of 92 psi. Adjustments for the number of fuel channels and reduced core height will be made in the thermal hydraulics calculations.

4.8 CONTROL ROD TEMPERATURES

The B_4C control rods generate substantial amounts of heat from alpha and gamma irradiation. The limiting temperature of the poison cladding is 1000°F caused by incompatibility between the poison and the stainless steel cladding at higher temperatures. The maximum allowable B_4C temperature is 4100°F. An investigation made of three 0.9 inch B_4C rods enclosed in a single channel showed that with a reactivity worth of 0.85$ each, the temperatures were retained within limits. More rods with lower temperatures may be required to contain the He gas generated.

4.9 RADIATION DAMAGE OF STRUCTURAL MATERIALS

The information that is available on radiation damage to structural materials in a fast reactor indicates that exposure to neutron bombardment is a critical parameter. There appears to be some doubt of the threshold energy of the neutrons that cause damage. There is not available enough information to fully assess the magnitude, the damage mechanism, or optimum solution of the problem. The latest information indicates that an exposure to a total fluence of about 2.3×10^{22} nvt in the EBR-II environment reduces the tensile ductility of the austenitic stainless steels and the high-nickel alloy Incoloy 800 to approximately 1 to 2 percent at test temperatures of 1100 to 1300°F. From this information it also appears that the ductility decreases with increasing test temperature, or with increasing exposure to radiation, or with biaxial stress. Data recently released from BNWL also indicates a severe reduction in ductility with irradiation, but degradation of tensile strength is not severe.

For the unshielded intra-core structure, the degradation of material properties represents a serious design problem. In this region the cladding of the fuel and the hexagonal tubes containing the fuel bundles may be exposed to a fast neutron fluence as much as 2.3×10^{23} nvt (>0.1 MeV). This target fluence represents a burnup of 100,000 MWd/Te with a fuel residence time of 1.5 years. When compared to the most extensive testing reported at this time of a total fluence exposure about 2.3×10^{22} nvt, the expected exposure is probably beyond reasonable extrapolation range. Additional test information is required at the high fluences expected in FFTF. In the meantime, the intra-core elements will be designed to minimize stress levels and forced deformation, cyclic loads and long term loadings. These elements are designed to facilitate removal and replacement. The structure external to the core is shielded to prevent excessive radiation exposure and radiation damage should not represent a serious design problem.

SECTION V

REFUELING CELL AND FUEL HANDLING

5.1 GENERAL

The design of the refueling cell and fuel handling methods has been based upon an open pool reactor, refueling cell concept, similar to that used in the SEFOR reactor. The refueling method may be described as a visual, remote-manual because refueling operations are performed remotely by operators having visual contact with the upper end of the reactor vessel and internals above the sodium level. Emphasis has been placed on the use of a number of easily maintainable, general purpose hoists capable of handling specialized grapples. All component transfers and servicing which involve exposure of radioactive materials will be carried out remotely, utilizing visual confirmation of events to as great a degree as possible. Shielding, applied over the reactor vessel fuel decay tanks, and primary coolant lines make possible direct access to the refueling cell to service any equipment or available power or instrument leads during reactor operation.

5.2 DRIVER FUEL HANDLING

Following reactor shutdown, the primary coolant temperature will be lowered to 350 to 400°F, after which the center shield plug, which includes the fuel holddown and driver fuel instrumentation, will be removed and placed in a storage pit in the cell floor.

The primary coolant level will be maintained approximately eleven feet above the fuel handle elevation. This is done to insure that during the refueling operation the fuel region of the element will remain immersed under sodium. A driver fuel element is removed from the core using the appropriate fuel grapple, and either moved to another core location or placed into the spent fuel transfer thimble. These transfer thimbles are stored within the refueling cell during reactor operation. During refueling they are placed, as needed, into the special refueling position where they can accept spent fuel elements to be removed from the core. The thimble containing the spent fuel is later lifted out of the core and located in one of the decay storage pools located near the edge of the reactor.

The basis of the design of the spent fuel transfer thimble requires that:

1. Fuel temperatures never exceed the maximum experienced in the reactor.

2. Coolants used will in no way change the metal surfaces of the fuel.

3. The transfer be time independent; i.e., steady-state temperature does not exceed the requirements of 1. above.

Based on these requirements, a finned transfer thimble containing primary sodium was chosen for fuel removal. This transfer method maintains the fuel thermal environment below the core operating conditions, and also provides a cooling system which is entirely independent of external power supplies because cooling is provided by natural circulation alone.

The present concept design is based on the removal of 25 kWt. This allows the removal of a driver fuel element 24 hours after reactor shutdown.

New fuel from the storage racks would be inserted into the proper core locations as vacancies are created by fuel removal. Preheating of these elements prior to this operation can be performed in the rack if necessary.

Driver refueling time has been estimated to be approximately 67 minutes per bundle. This estimate is based on refueling time estimates published in Reference (3). Considering the fixed-time jobs which must precede each refueling, and using a thirty-bundle refueling schedule, requires an approximate driver refueling time of 42 hours.

5.3 OPEN LOOP HANDLING

The open loop handling, as well as handling of other experimental devices, will be dependent on the detailed design and purpose of the experiment. In general, open loop test assemblies would be handled in a manner very similar to that for the driver fuel, with appropriate care to protect the instrumentation and leads. Special transfer thimbles may be required in the movement of some of these loops; depending, of course, on the particular design of test loop.

The method and timing of handling of the open loop tests depends primarily upon the element decay heat. Capsules and material sample tests could be removed immediately from the reactor to the examination area. High power prototype driver bundles would require a decay period just as driver bundles do prior to transfer.

5.4 CLOSED LOOP HANDLING

The closed loops are distinguished by their capability to supply environmental conditions independent of the reactor coolant system to the test specimen. Because of this capability handling methods and procedures will be entirely dependent on the loop designs, environment, and length of irradiation. Several promising alternate systems of handling closed loop tests within the refueling cell were studied and reported in Reference(3). Although these concepts were developed for a tank-type reactor, the similarities in refueling methods and cell design make them appropriate to this study also. These alternate handling methods demonstrate that feasible handling methods are available regardless of the final loop design.

Of the alternate handling methods presented in Reference (3), the closed thimble approach to handling was judged to be the most promising. Here the loop experiment is transferred in a closed thimble (actually part of the closed loop internal piping system). Additional cooling being supplied if necessary by a NaK "cold finger." This method would keep the experiment in sodium during all transfers. An alternate handling system would use the facility tube itself, transferred in a finned pot. Because the facility tube must be removed periodically, it is reasonable to remove the test along with it. This would also provide a method of removing a test in which coolant flow restrictions were suspected and isolation of test coolant and primary sodium is desirable.

Thus, in general, the basic handling systems of all fuel assemblies: open loop, closed loop, and driver fuel in essence fall in one general category, that of a sodium filled thimble, relying on natural circulation to remove the decay heat and dissipate it to the cell atmosphere.

5.5 STORAGE FACILITIES

Capabilities will bc provided for the storage of new fuel, spent test and driver fuel requiring decay heat removal, and activated equipment. The used fuel storage capacity will be at least one core load while the new fuel capacity will be sufficient for one refueling, to be consistent with the philosophy that the capability exists for the storage of one refueling load of test and driver fuel in transfer thimbles. This storage allocation will allow refueling to proceed without additional operations of the refueling cell transfers or unloading transfer thimbles.

5.6 FUEL TRANSFER LOCK

All test equipment and fuel to be handled within the refueling cell may be transferred through the containment building and out of the cell via the fuel transfer lock. This lock is a cylindrical container approximately 28 feet in diameter and 37 feet long (high). Two sets of double valves penetrate the upper head. These valves allow 48-inch-diameter components (test loop coolant connections) to pass through. The length of the cell is sized to accommodate the longest expected test element and transfer it in the vertical position within the lock. A trolley transfer system is utilized to shuttle components the short horizontal distance from one set of valves in the refueling cell to the other set with access outside of containment.

5.7 AVAILABILITY OF REFUELING EQUIPMENT

The refueling method presented is judged to yield the highest availability of any of the alternate approaches considered to date. These alternate approaches include:

1. Outer head removal with decay storage in the vessel perimeter.

2. Outer head removal with interim transfer to thimbles taking place in vessel perimeter.

Although other alternate concepts such as rotating storage drums and complex refueling machines may yield refueling times less than the reference method, the simplicity of the proposed scheme is judged to outweigh these from availability considerations.

Final procedures and equipment design will be established to give optimum overall plant availability.

SECTION VI

AVAILABILITY—DRIVER FUEL INSTRUMENTATION

(Contribution of Dr. J. Fleck, GE-Research and Development Center, Schenectady, New York)

6.1 INTRODUCTION

Core instrumentation for the Fast Flux Test Reactor (FFTF) backup design is discussed in Reference (2). In Section 4.8.2 of that report it is stated that thermocouples show the most advanced state of development of all sensors. Hence, present work has been limited to temperature sensing, but what will be presented here is directly extensible to flowmeters.

The purpose of this study is to minimize the reactor unavailability connected with core instrumentation. From this standpoint, the purpose of providing core temperature sensors is to sense incipient meltdowns and scram the reactor before serious damage occurs. Should a meltdown happen, then, of course the reactor would be unavailable for a protracted period. However, the instrumentation itself will be a source of unavailability because of false scrams. The optimum core instrumentation scheme will minimize the total unavailability, which is the sum of unavailability caused by false scrams plus unavailability caused by undetected incipient meltdowns.

Eight schemes for core temperature sensing have been conceived, ranging from a scheme with no redundancy whatever to an elaborate scheme using four thermocouples per channel and extensive failure-detecting circuitry. The unavailability of each scheme has been predicted, and on this basis the arrangement called "Scheme D" is recommended. In brief, this scheme uses four thermocouples per channel, with three of them actively connected and one as a spare. One thermocouple from each channel is connected to each of three "or" circuits, and then the three "or"s are connected to two-out-of three majority logic to provide the scram signal.

6.2 ASSUMPTIONS

Results of this study should be viewed in light of the assumptions used. They are listed below.

1. It is necessary to instrument each driver fuel channel. A thermocouple in one channel is not used to sense overheating in an adjacent channel.

2. Integrated electronic circuits will be used throughout. All electronics will be located in a nonradioactive environment, with suitable air conditioning and access for repair.

3. The failure rate of an integrated-circuit chip is assumed to be 0.3 per million hours, based upon experience with semiconductor devices in "benign" environments.

4. The failure rate of thermocouple joints is 0.1 per million hours, and the failure rate of thermocouple leads is also 0.1 per million hours. This is based upon sodium-cooled reactor history, adjusted for design improvements.

5. Fuel channel meltdowns occur at the rate of 0.002 meltdowns per channel per year (reactor experience).

6. Mean repair times, including time to diagnose failure and to restart the reactor, are as follows:

   False scram — 8 hours
   Replace thermocouple — 300 hours
   Repair after channel meltdown — 4 months longer than repair of plugging

7. At intervals of 1 year, all thermocouples and electronics will be rigorously tested, and any failures not detected during operation will be repaired.

8. All electronic circuits have two equally likely failure modes: false output (leading to false scram), and malfunction (inability to scram when requested).

9. All thermocouple and lead failures result in an abnormally high or low output that can be sensed as a failure.

6.3 SCHEMES FOR CORE TEMPERATURE SENSING

Schematic representations of the eight basic schemes, A through F, are shown generally and respectively in Figures 6-1 through 6-7.

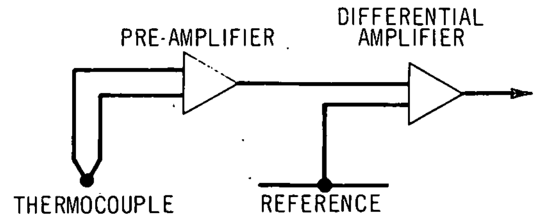

FIGURE 6-1. CORE TEMPERATURE SENSING-GENERAL

6.3.1 Scheme A

Each channel is provided with one thermocouple. As with all schemes each thermocouple output is fed to a temperature-stabilized preamplifier and thence to a differential amplifier, which compares the thermocouple output to a reference termperature limit (see Figure 6-8). Whenever any thermocouple output exceeds the temperature limit, its differential amplifier changes to the alarm state.

Assuming 120 channels to be instrumented, there will be 120 amplifier outputs. These are connected to a 120-input "or" circuit, which outputs a scram signal whenever any one of its inputs is in the alarm state. As an optional feature, the preamplifier or differential amplifier outputs may be connected to individual recorders.

Scheme A is the simplest way to instrument every channel, and thus it is the cheapest from the standpoint of capital cost. However, it incorporates no redundancy, and it is susceptible to both false-output failures and malfunctions.

6.3.2 Scheme B

In this scheme, each channel is provided with three thermocouples. Each thermocouple output is fed through a temperature-stabilized

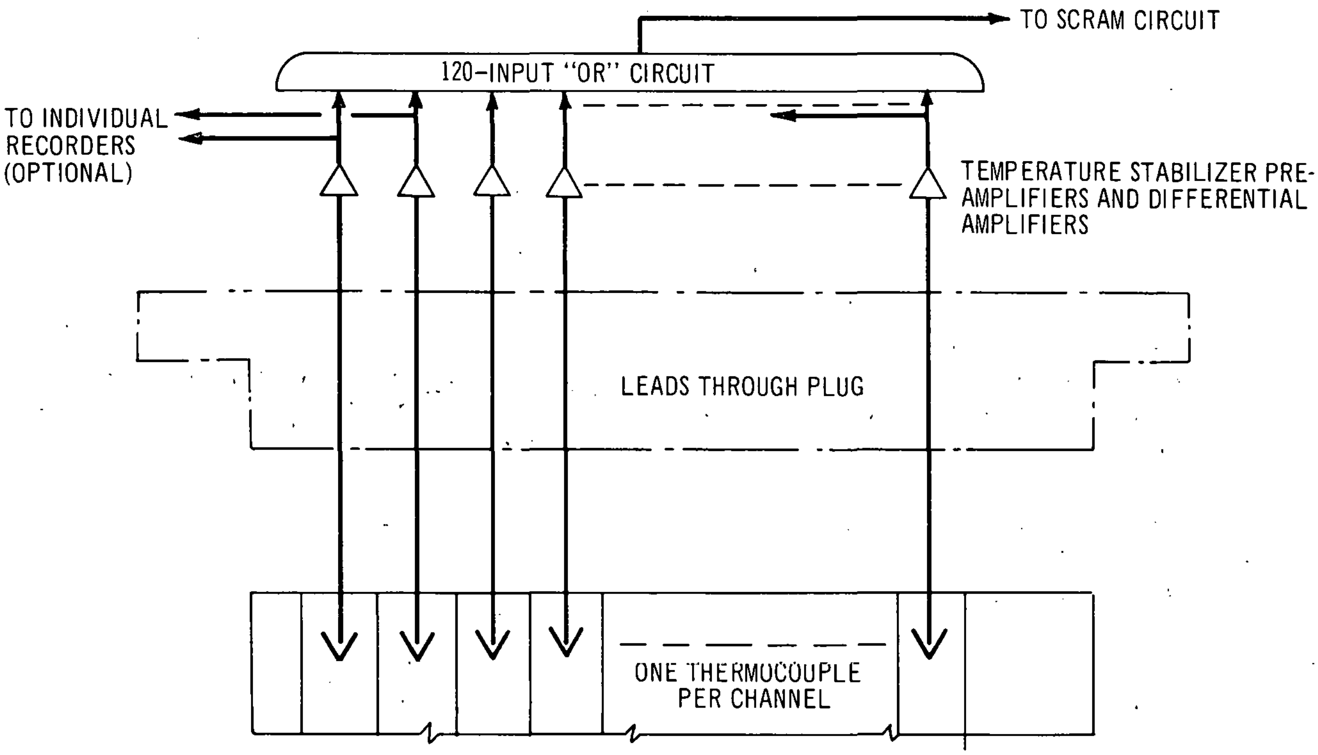

FIGURE 6-2. CORE TEMPERATURE SENSING-SCHEME A, ONE THERMOCOUPLE PER CHANNEL, SCRAM ON ANY HIGH TEMPERATURE

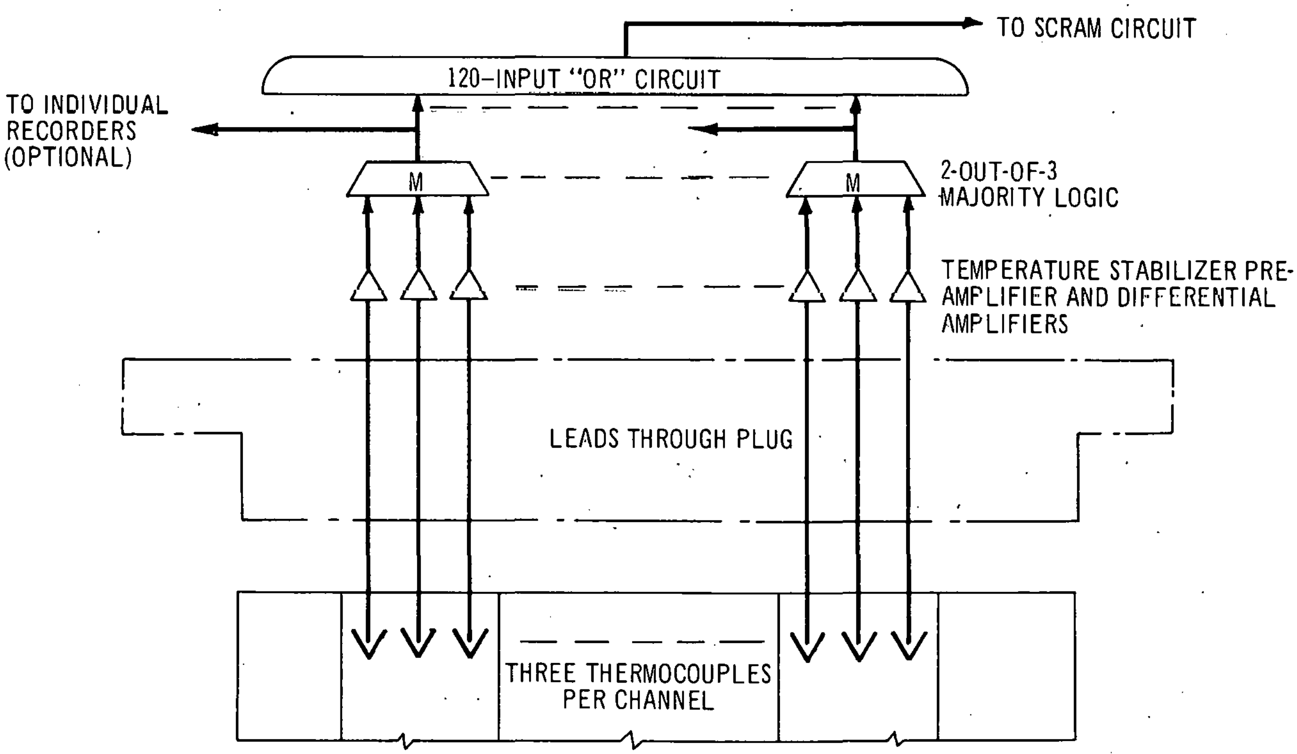

FIGURE 6-3. CORE TEMPERATURE SENSING-SCHEME B, 3 THERMOCOUPLES PER CHANNEL WITH MAJORITY LOGIC IN EACH CHANNEL

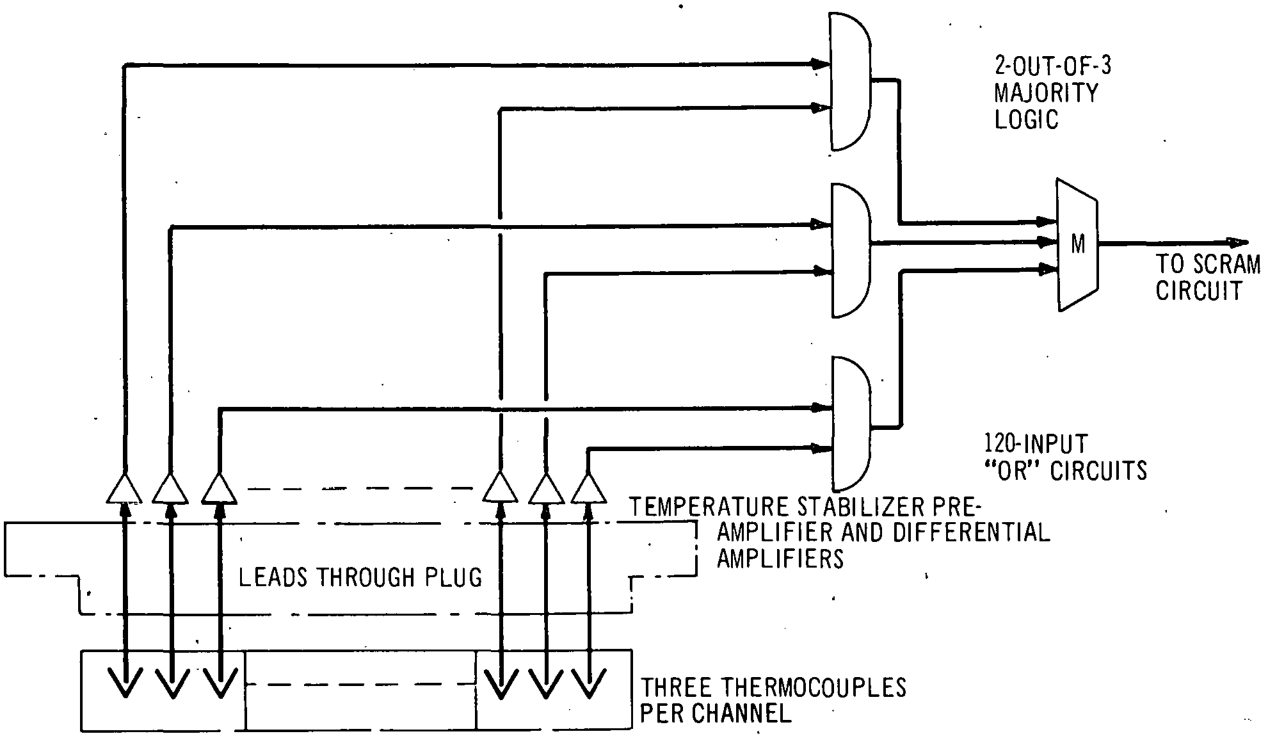

FIGURE 6–4. CORE TEMPERATURE SENSING–SCHEME C, 3 THERMOCOUPLES PER CHANNEL, "OR", IN 3 GROUPS, MAJORITY LOGIC IN EACH CHANNEL

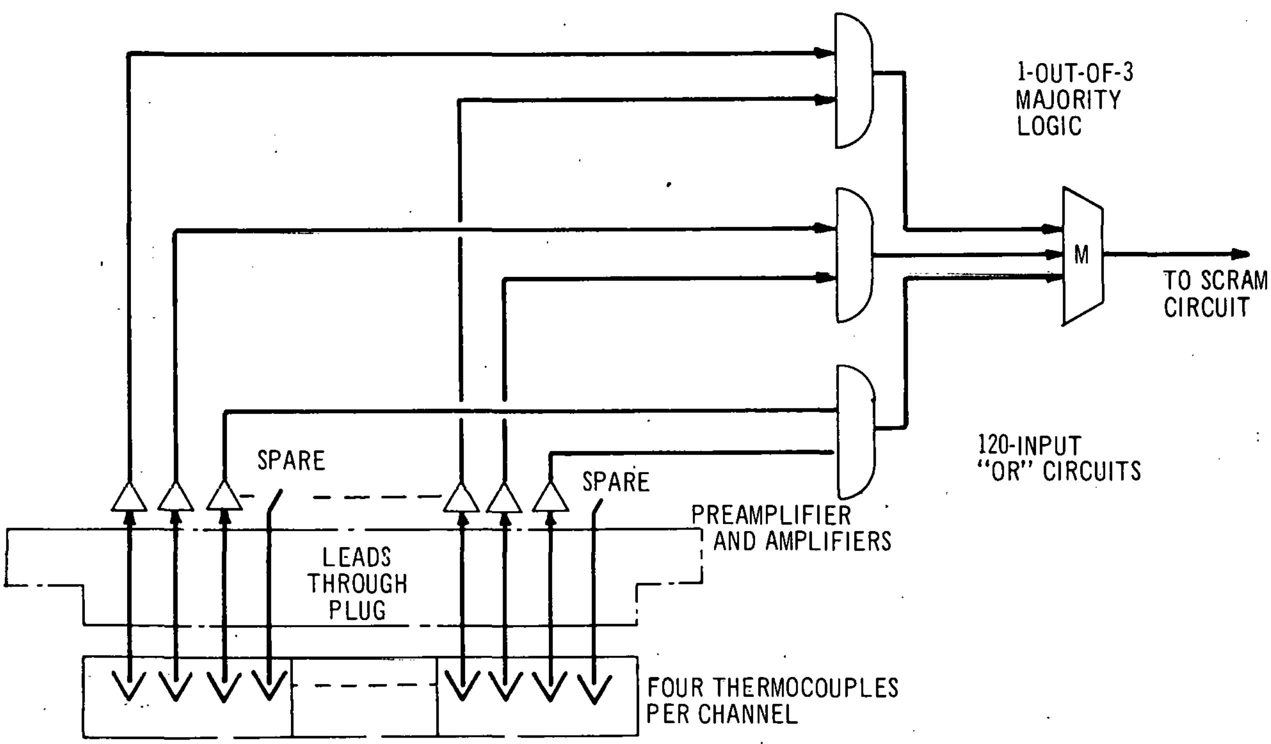

FIGURE 6–5. CORE TEMPERATURE SENSING–SCHEME D, 4 THERMOCOUPLES PER CHANNEL, (ONE SPARE), OTHERWISE SAME AS SCHEME C

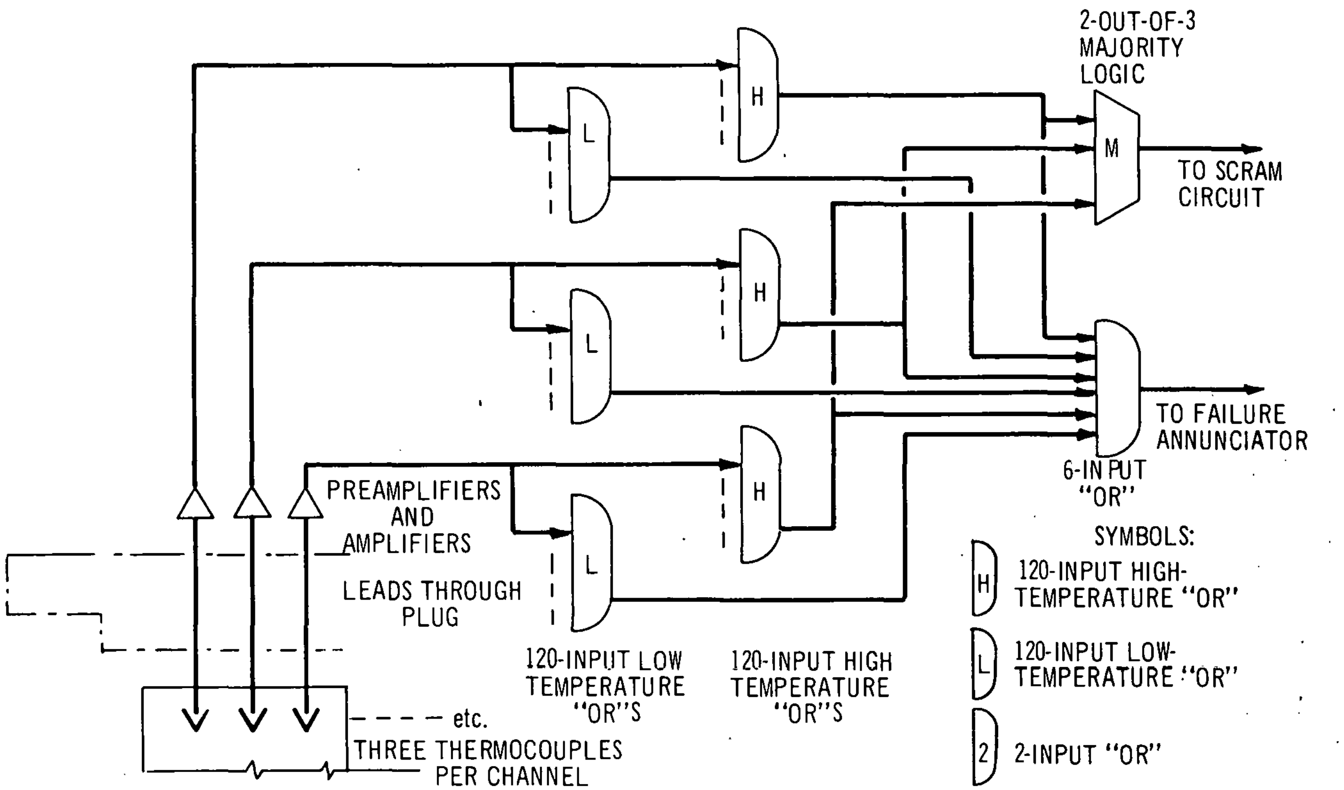

FIGURE 6–6. CORE TEMPERATURE SENSING–SCHEME E, SCHEME C PLUS FAILURE DETECTION

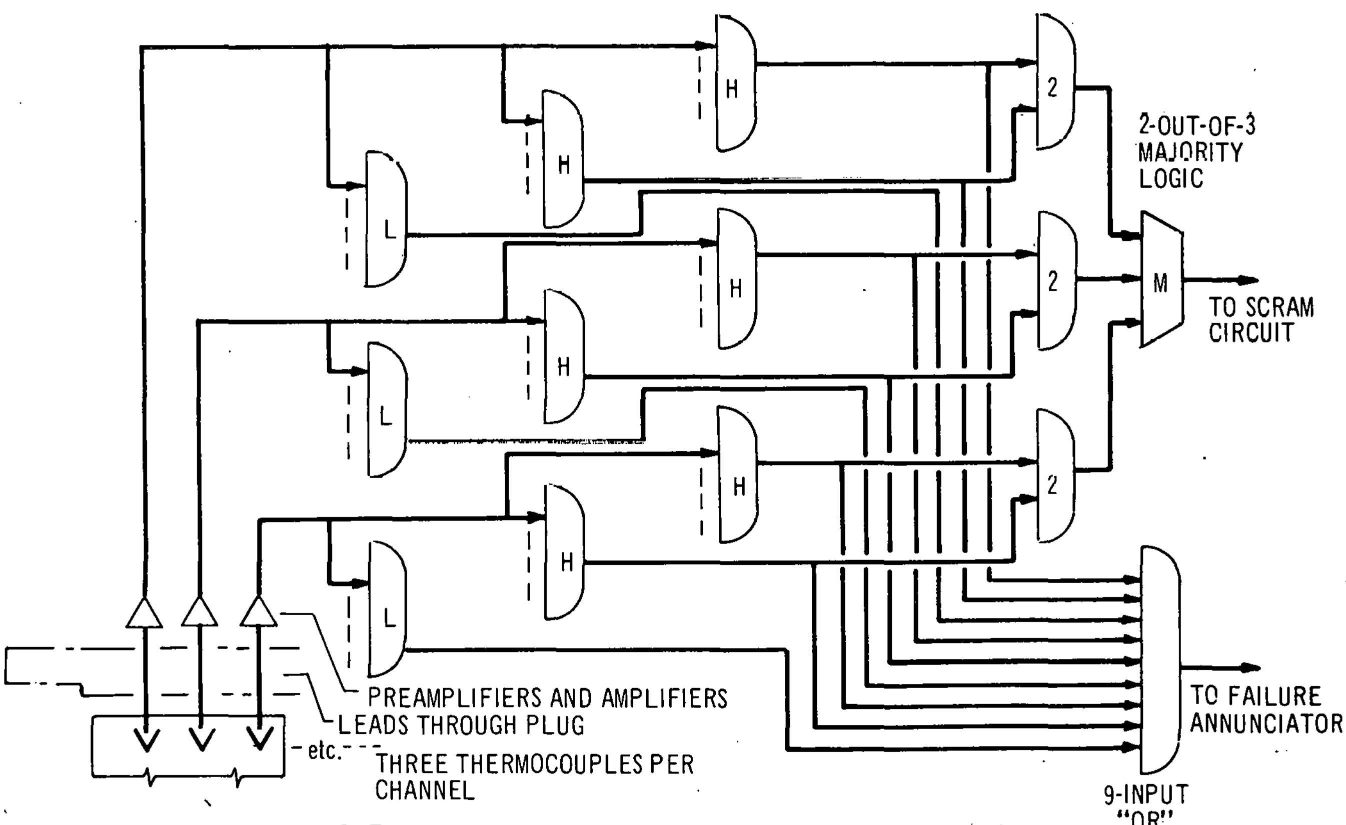

FIGURE 6–7. CORE TEMPERATURE SENSING–SCHEME F, SCHEME E PLUS PROTECTION AGAINST FAILURES OF 120-INPUT "OR"'S

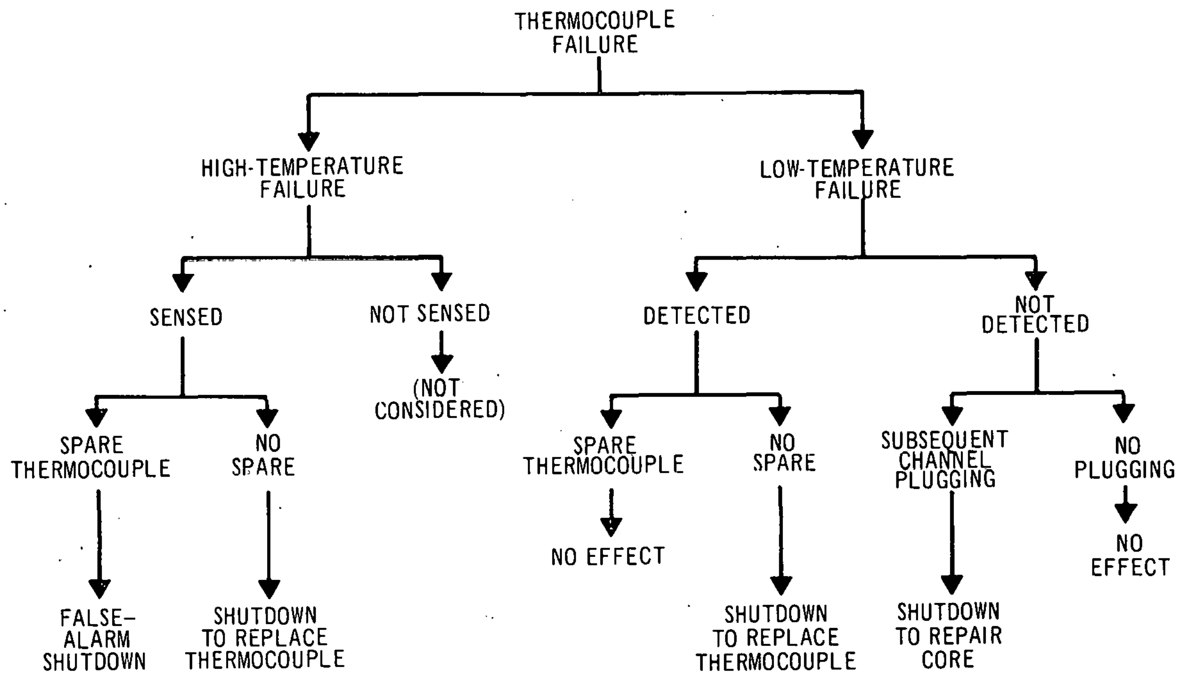

FIGURE 6-8. CONSEQUENCES OF THERMOCOUPLE FAILURE

preamplifier, and then compared with a reference in a differential amplifier. For each channel, the three amplifier outputs are connected to a two-out-of-three majority gate (see diagram for Scheme B, Figure 6-3).

For 120 channels, there will thus be 120 majority gates. The outputs of these majority gates are shown connected to a 120-input "or" circuit, which provides a scram signal whenever any one of the 120 majority gates indicates an alarm.

Scheme B is the most straightforward extension of Scheme A. This protects against failures in thermocouples or in thermocouple amplifiers. It does not, however, protect against failures in the 120 majority gates or in the 120-input "or". A second problem is that one thermocouple or amplifier failure in each channel will go undetected until the yearly inspection.

6.3.3 Scheme C

This scheme is a rearrangement of the electronic logic of Scheme B. As with B, each channel is provided with three thermocouples and associated amplifiers. There are three 120-input "or" circuits; one of the amplifier outputs from each of the 120 channels is connected to each "or" (see the schematic diagram, Figure 6-4, for Scheme C). The outputs of the three "or"s are connected to a two-out-of-three majority gate, which provides the scram signal.

In Scheme B, there are 120 majority gates and one 120-input "or", none of which are protected by redundancy. In Scheme C, by comparison, the only circuit not protected by redundancy is one majority gate. Scheme C requires three 120-input "or"s while Scheme B requires only one of them, but in C they are protected by redundancy.

Scheme C should be less expensive. Its "or" circuits and majority gate require an estimated 46 integrated-circuit chips, while the corresponding circuits in Scheme B will require about 105 chips.

6.3.4 Scheme D

This is a variation of Scheme C, the difference being that a spare thermocouple is provided in each channel for Scheme D.

It is assumed that operating policy will be to shut down the reactor for repair whenever a thermocouple failure is known to exist. With Schemes B, C, and D, a single thermocouple failure in any channel will not be known until a second failure occurs in the same channel. The second failure may be either another thermocouple failure or a thermocouple amplifier failure. Because amplifier failures will probably be more likely than thermocouple failures (see assumed failure rates), it is appealing to have a spare thermocouple in each channel. Thus, the time to replace a bad thermocouple is reduced greatly, from the estimated 300 hours to allow for decay and lift the cover, to an estimated 8 hours for changing leads externally.

6.3.5 Scheme E

This is a different variation of Scheme C, where failure detection is provided.

Referring to the diagram for Scheme E, Figure 6-6, note that it uses three 120-input "high-temperature or" circuits. These are identical to the 120-input "or"s of Scheme C, and it will be noted that they are connected to a two-out-of-three majority gate as in Scheme C. The added equipment in Scheme E consists of three 120-input "low-temperature or" circuits, plus a 6-input "or" which provides failure annunciation. Whenever any one of the 360 thermocouple amplifier outputs is faulty, either high or low, it will be sensed by one of the six 120-input "or"s.

Scheme E will announce failure in any thermocouple or any thermocouple amplifier. Presumably such a failure can be repaired with a short downtime, or perhaps with no downtime at all if its repair can be deferred until refueling (there would still be two good sensors in the affected channel). This scheme does not detect malfunction-type failures in the "or" circuits or the majority gate, but most of the electronics is in the thermocouple amplifiers.

6.3.6 Scheme F

This is an extension of Scheme E, designed to provide protection against malfunction-type failures in the 120-input "or" gates which feed the majority gate.

Referring to the diagram for Scheme F, Figure 6-7, note that three more 120-input "high-temperature or" gates have been added. Each thermocouple amplifier output feeds two "high-temperature or" circuits, as well as one "low-temperature or." There are three sets of "high-temperature or" gates with common inputs (two gates per set), and the diagram shows the outputs from the two gates of each set fed to a two-input "or." The three 2-input "or"s provide the signals for the two-out-of-three majority gate, which in turn provides the scram signal.

With this scheme, any three of the "high-temperature or" gates can fail without subverting the scram function. It is true that only one failure can be tolerated in the 2-input "or"s, but these are much simpler than the 120-input "high-temperature or" gates.

Scheme F has the most complicated electronic logic of all schemes (except H, which is similar to F). It requires about 138 integrated-circuit chips for the logic ("or" gates and the majority gate). This is relatively small, even so, when it is considered that about 720 chips will be needed for the thermocouple amplifiers of all schemes except Scheme A.

6.3.7 Scheme G

No diagram for Scheme G is given in the appendix; it is a straightforward variation of Scheme E. The only difference is that Scheme G has a spare thermocouple in each channel, similar to Scheme D.

6.3.8 Scheme H

This scheme also lacks a diagram because it is a straightforward variation of Scheme F. Scheme H uses a spare thermocouple in each channel; otherwise it is identical to F.

6.4 AMOUNT OF ELECTRONIC CIRCUITS

To predict failure rates for the various schemes, it was necessary to estimate the amount of electronic circuits in each. This was done by consulting Gerald J. Michon of the General Electric Research and Development Center, who gave the following first-order estimates for the circuits required.

6.4.1 Majority Gate (three inputs)

This will be presumed mechanized as three "and" gates with "wired or" outputs. Three-fourths of one integrated-circuit chip will be required.

6.4.2 Thermocouple Amplifiers

Each thermocouple must be provided with a temperature-stabilized preamplifier, a differential amplifier to compare with reference, and fan-out circuitry. The preamplifier corresponds to two transistor stages plus temperature-controlling circuitry, all of which occupies one IC chip. The differential amplifier corresponds to 8 or 10 transistors, and it also occupies one IC chip. The fan-out circuitry corresponds to two transistors, occupying only 1/6 IC chip. Thus, each thermocouple will require a total of 2-1/6 IC chips.

This estimate assumes use of chromel-constantin in thermocouple junctions, which have a sensitivity of 80 microvolts per degree C at a temperature of 450°C. Thermocouples with different sensitivities would require re-examination of the amplifier stabilization circuitry. Chromel-Alumel junctions will be looked at.

6.4.3 "Or" Gates

A 16-input "or" can be constructed from a 4-input "nand" circuit (1/2 IC chip), an inverter (1/6 chip), and a 12-input input expander (1 chip). Thus, a 120-input "or" will require $8 \times (1 + 1/2 + 1/6)$ chips, plus 1 chip to combine the eight outputs; a total of 15 chips.

Estimates have also been made of the amount of discrete-circuit hardware needed. Assuming failure rates of $\lambda = 0.3$ per million hours per chip or per active element group (discrete), the following results are obtained:

Circuit	(per 10^6 h) λ discrete	(per 10^6 h) λ integrated
Thermocouple preamplifiers and amplifiers	4.0	0.6
120-input "or" gate	25.0	4.5
3-input majority gate	1.8	0.3

It is seen that the predictions for integrated circuits are, in general, about six times better than for discrete circuits.

6.5 FORMULAS FOR UNAVAILABILITY

Unavailability is a dimensionless quantity, defined as the ratio of expected downtime per period to total desired operating time per period:

$$U = \frac{E(\text{downtime})}{T} , \qquad (6\text{-}1)$$

Using this definition, it may be readily shown that a suitable approximation for nonredundant systems is

$$U \cong \lambda \mu \quad , \tag{6-2}$$

where λ is the system failure rate and μ is its mean repair time. This also applies to the "i"th system component:

$$U_i \cong \lambda_i \mu_i \; , \tag{6-3}$$

and the system unavailability may be closely approximated by the sum of the component unavailabilities:

$$U \cong \sum_i U_i \; . \tag{6-4}$$

Suitable approximations can also be derived for redundant systems. For example, with two-out-of-three majority logic the unavailability is

$$U \cong 3 \lambda^2 \mu \; T_I \; , \tag{6-5}$$

where T_I is the interval between inspections (assuming that single failures are not detected until periodic inspection). If a system is a serial combination of redundant subsystems, the unavailabilities of the subsystems may be directly summed to approximate the system unavailability.

For the unavailability formulas used for the eight schemes following, the following nomenclature will be used.

U_{FS} = unavailability because of false scrams

U_{TR} = unavailability because of thermocouple replacement

U_{ND} = unavailability because of nondetected failures and subsequent channel plugging

μ_{FS}, μ_{TR}, μ_{ND} = respective mean repair times

N = 120 = number of channels

$\lambda_{amp, F}$ = failure rate for false-output mode of thermocouple preamplifiers and amplifiers

$\lambda_{amp, M}$ = failure rate for malfunction (no output) mode of thermocouple preamplifiers and amplifiers

$\lambda_{or, F}$ and $\lambda_{or, M}$ = failure rates for 120-input "or" gates

$\lambda_{TC, F}$ and $\lambda_{TC, M}$ = failure rates for thermocouples

$\lambda_{lead, F}$ and $\lambda_{lead, M}$ = failure rates for thermocouple leads*

$\lambda_{ML, F}$ and $\lambda_{ML, M}$ = failure rates for 3-input majority gates

$\lambda_{2, F}$ and $\lambda_{2, M}$ = failure rates for 2-input "or" gates

$\lambda_{Hor, F}$ and $\lambda_{Hor, M}$ = failure rates for high-temperature "or" gates (same as $\lambda_{or, F}$ and $\lambda_{or, M}$)

$\lambda_{Lor, F}$ and $\lambda_{Lor, M}$ = failure rates for low-temperature "or" gates (also same as $\lambda_{or, F}$ and $\lambda_{or, M}$)

λ_{plug} = channel plugging rate

T_I = inspection interval

Scheme A:

$$U_{FS} = (N \lambda_{amp, F} + \lambda_{or, F}) \mu_{FS}$$

$$U_{TR} = N (\lambda_{TC, F} + \lambda_{lead, F}) \mu_{TR}$$

$$U_{ND} = N (\lambda_{amp, M} + \lambda_{or, M} + \lambda_{TC, M} + \lambda_{lead, M}) \lambda_{plug} \mu_{ND} T_I$$

Scheme B:

$$U_{FS} = (3N \lambda_{amp, F}^2 T_I + N \lambda_{ML, F} + \lambda_{or, F}) \mu_{FS}$$

$$U_{TR} = 3N \lambda_{TC, F} (\lambda_{TC, F} + \lambda_{amp, F}) \mu_{TR} T_I$$

$$U_{ND} = (3 \lambda_{amp, M}^2 + 3 \lambda_{TC, M}^2 + \lambda_{ML, M} + \lambda_{or, M}) (N \lambda_{plug} \mu_{ND} T_I)$$

*$\lambda_{lead, F}$ and $\lambda_{lead, M}$ are included with $\lambda_{TC, F}$ and $\lambda_{TC, M}$ except for Scheme A.

Scheme C:

$$U_{FS} = \left[3\,(N\,\lambda_{amp,\,F} + \lambda_{or,\,F})^2\, T_I + \lambda_{ML,\,F} \right] \mu_{FS}$$

U_{TR} as for Scheme B

$$U_{ND} = \left[3\,\lambda_{amp,\,M}{}^2\, T_I + 3\,\lambda_{or,\,M}{}^2\, T_I + N\,\lambda_{ML,\,M} + 3\,\lambda_{TC,\,M}{}^2\, TI \right] \cdot N\,\lambda_{plug}\,\mu_{ND}\,T_I$$

Scheme D:

$$U_{FS} = \left[3N\,\lambda_{TC,\,F}\,(\lambda_{TC,\,F} + \lambda_{amp,\,F})T_I + 3\,(N\,\lambda_{amp,\,F} + \lambda_{or,\,F})^2\, T_I + \lambda_{ML,\,F} \right] \mu_{FS}$$

$$U_{TR} = \frac{9}{4}\lambda_{TC,\,F}{}^2\,\lambda_{amp,\,F}\,(\lambda_{TC,\,F} + \lambda_{amp,\,F})\, N\,\mu_{TR}\,T_I{}^3$$

U_{ND} as for Scheme C

Scheme E:

U_{FS} as for Scheme C

$$U_{TR} = 3N\,\lambda_{TC}\,(\lambda_{TC} + \lambda_{amp})\,\mu_{TR}\,T_I,$$

where $\lambda_{TC} = \lambda_{TC,\,F} + \lambda_{TC,\,M}$ = total TC failure rate and similar for λ_{amp}, since any failure is now detected.

$$U_{ND} = \left[3\,(\lambda_{amp,\,M}{}^2 + \lambda_{TC,\,M}{}^2)\,\lambda_{2,\,M}\,T_I + \lambda_{Hor,\,M}{}^2 + 3\,(\lambda_{amp,\,M}{}^2 + \lambda_{TC,\,M}{}^2)\,\lambda_{Lor,\,M}{}^2\,T_I{}^2 \right] \cdot N\,\lambda_{plug}\,\mu_{ND}\,T_I{}^2$$

Scheme F:

$$U_{FS} = 3\,(N\,\lambda_{amp,\,F} + 2\,\lambda_{or,\,F} + \lambda_{2,\,F})^2\,\mu_{FS}\,T_I$$

U_{TR} as for Scheme E

$$U_{ND} = 3N\,\lambda_{Hor,\,M}{}^4\,\lambda_{plug}\,\mu_{ND}\,T_I{}^4$$

Schemes G and H:

U_{FS} as for Schemes E and F, respectively, with an additional term $3N\,\lambda_{TC}\,(\lambda_{TC} + \lambda_{amp})\,\mu_{FS}\,T_I$.

$$U_{TR} = \frac{9}{4}\,N\,\lambda_{TC}{}^2\,\lambda_{amp}\,(\lambda_{TC} + \lambda_{amp})\,\mu_{TR}\,T_I{}^3$$

U_{ND} as for Schemes E and F, respectively.

6.6 CONSEQUENCE ANALYSIS

The preceding formulas were derived by considering the possible paths to each type of system failure. The "consequence analyses" will not be discussed in detail, but two examples will be shown.

Figure 6-8 shows the consequence analysis tree for thermocouple failures. This is general and applies to any scheme, because it has branches where failures are detected or not, and where spare thermocouples are available or not. This figure shows how thermocouple failure can lead to any of the three types of unavailability (false alarm, shutdown to replace thermocouple, or shutdown to repair core).

Figure 6-9 shows, for the specific case of Scheme D, the fault tree leading to the event "shutdown to replace thermocouple." This tree was used to derive the expression for U_{TR} of Scheme D. Note that the upper half of this tree is traversed with probability $\frac{3}{2}\,\lambda_{TC,\,F}\,\lambda_{amp,\,F}\,T_I{}^2$; the factor of 1/2 arises because a specific order is required for the "and"ed events. Likewise, the lower half of this tree is traversed with probability

$\frac{3}{2} \lambda_{TC, F} (\lambda_{TC, F} + \lambda_{amp, F}) T_I^2$.

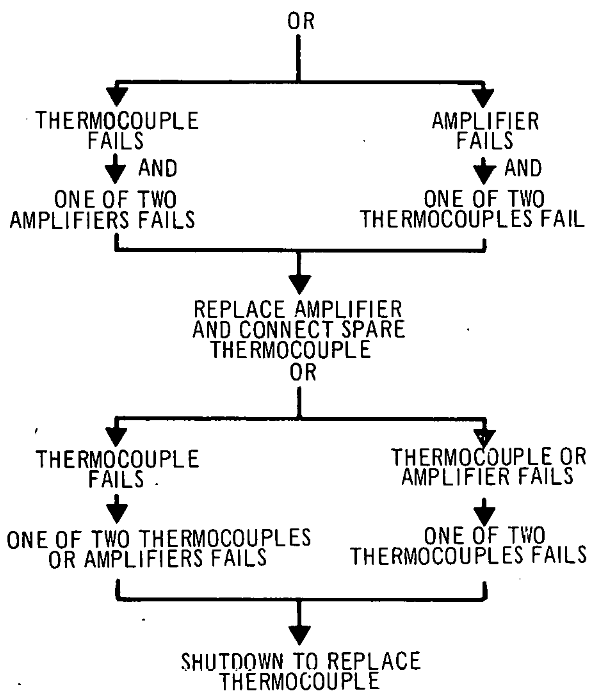

FIGURE 6-9. PATHS TO THERMOCOUPLE REPLACEMENT IN SCHEME D

6.7 RESULTS—DRIVER FUEL INSTRUMENTATION AVAILABILITY

By inserting assumed numerical values in the preceding formulas, the unavailabilities for the various schemes were calculated. Results are shown in Table 6-1.

The following conclusions may be drawn:

1. Any of the redundant schemes is better than Scheme A, the nonredundant scheme.

2. Scheme C is a much better way of applying redundancy than Scheme B. Because C is is slightly less complex, B should receive no further consideration.

3. Schemes D through H, which are more complex than Scheme C, offer little improvement, if any.

On the basis of these results, one would select Scheme C. However, there remains a lingering doubt that the thermocouple failure rate may be higher than assumed here. To satisfy this doubt, one might consider Scheme D, which differs from Scheme C only in adding a spare thermocouple to each channel.

TABLE 6-1

PLANT UNAVAILABILITIES OF VARIOUS INSTRUMENTATION SCHEMES

Scheme	$U_{TC, FS}$	$U_{E, FS}$	$U_{TC, ND}$	$U_{E, ND}$	$U_{TC, TR}$	U_{Total}
A	--	300	70	210	3600	4200
B	--	160	0.6	1700	38	1900
C	--	340	0.6	200	38	580
D	1.0	340	0.6	200	6×10^{-5}	540
E	--	340	0.002	93	150	580
F	--	470	0.002	0.04	150	620
G	4.0	340	0.002	93	0.0014	440
H	4.0	470	0.002	0.04	0.0014	470

$U_{TC, FS}$ = Unavailability due to false scrams caused by thermocouple failures (excludes case of immediate thermocouple replacement)

$U_{E, FS}$ = Unavailability due to false scrams caused by electronics failures

$U_{TC, ND}$ = Unavailability due to nondetected thermocouple failures and subsequent channel plugging

$U_{E, ND}$ = Unavailability due to nondetected electronics failures and subsequent channel plugging

$U_{TC, TR}$ = Unavailability due to thermocouple failures which are detected and immediately replaced

U_{Total} = Total unavailability for each scheme

All unavailabilities are to be multiplied by 10^{-6}.

SECTION VII

NUCLEAR CHARACTERISTICS

7.1 GENERAL

The following nuclear characteristics results are a summary of the detailed results reported in Reference (2).

7.2 MAXIMUM TOTAL FLUX

The neutron flux available at the experimental loop positions is central to the analysis and design of the FFTF. Parameters of importance include the maximum total flux, the flux spectrum, and the spatial flux distribution.

The maximum total flux for the present vertical core design at mid-burnup is 0.9×10^{16} neutrons/cm^2-sec.

Because a design criterion is the power level, which is fixed at 400 MWt, the peak flux is sensitive to the value used for MeV/fission. For the FFTF vertical core design, this value is 215 MeV/fission. The peak flux is inversely proportional to this parameter.

The maximum flux quoted here is the flux at the middle of the equilibrium fuel cycle (or averaged over the fuel cycle). The average fuel burnup at midcycle is 35 MWd/kg. The calculated criticality factor is unity and 1 percent Δk worth of control (B_4C) is present in the reflector. The 1 percent control value at midcycle is consistent with the 2 percent reactivity requirement for the fuel cycle.

The peak flux is nearly constant throughout the fuel cycle. Although the two principal variables which affect the neutron flux change considerably during the cycle, the changes tend to cancel. The two variables are fissile concentration and radial flux shape.

7.3 FLUX ENERGY SPECTRUM AND SPATIAL POWER DISTRIBUTION

The fraction of the flux above 0.1 MeV at the peak flux position, i.e., middle of the central closed loop, is 66 percent.

The present conceptual design is without BeO. Each volume percent of BeO added reduces the percent flux above 0.1 MeV by 1 percent. Future conceptual design work will consider BeO volume fractions up to ~5 percent.

Peak-to-average specific power ratios (peaking factors) are:

Radial = 1.4
Axial = 1.2

7.4 REACTIVITY CONTROL SYSTEM

7.4.1 Reactivity Requirements

The primary-system reactivity requirements for the FFTF vertical core design are 20\$. The contributions to this total are listed in Table 7-1. The effective delayed neutron fraction is 0.003.

In addition, an auxiliary (backup) control system is required. The backup requirements may vary from ~3\$ to ~8\$; 3\$ is required to return the reactor to hot standby (uniform 700°F) and override one reflector control rod, while 8\$ would return the reactor to refueling temperature.

TABLE 7-1
REACTIVITY CONTROL REQUIREMENTS
(PRIMARY SYSTEM)

Fuel Cycle	2%
Cold to Hot	2%
Experiments	1%
Shutdown	1%
TOTAL	6% = 20\$

7.4.2 Reflector Control System Reactivity Primary System

All of the primary-system control requirements of the conceptual vertical core design are met by the reflector control system. The reflector control system is composed of 27 rods located on the core periphery. Sixteen control rods are composed of two cylindrical elements which enter adjacent channels; eleven rods consist of one element each. Nickel is below B_4C in each element; the boron is natural and the B_4C density is 80 percent theoretical, although the use of some enriched boron may be required. In the reactive position, the nickel is adjacent to the core and serves as a radial reflector. To decrease reactivity, the rod is lowered until B_4C portion is adjacent to the core. The movable nickel portion occupies 65 percent of each channel, the B_4C occupies 59 percent, and the B_4C cladding 6 percent.

Preliminary calculations indicate that the calculated values quoted in all primary control requirements can be met by reflector control. The results of five reflector control calculations completed thus far are shown in Table 7-2.

TABLE 7-2

PRELIMINARY CALCULATED REFLECTOR CONTROL WORTHS

Description	Control Worth
24 elements, ρ (B_4C) = 70% theoretical, natural Boron	15$
24 elements, ρ (B_4C) = 80% theoretical, natural Boron	17$
43 elements, ρ (B_4C) = 80% theoretical, natural Boron	20$
24 elements, 80% theoretical, natural Boron; 19 elements $B_4^{10}C$	25$
43 elements, $B_4^{10}C$	33$

7.4.3 Auxiliary Control System

The auxiliary (backup) control system consists of three in-core B_4C poison rods. The allowable worth of each rod can be as high as 3$ because these rods will always be withdrawn from the core during normal operation. Since it is likely that less than 9$ will be required for auxiliary control, smaller rods will probably be used.

7.5 FUEL CYCLE

A 90 day (3 month) fuel cycle has been assumed which includes 72 days operating at full power and 18 days shutdown. Five batch refueling is assumed which means that 20 percent of the fuel assemblies are replaced during each refueling. Total average burnup of an assembly before removal is 70 MWd/kg metal (70,000 MWd/Te); average burnup during a single cycle is 14 MWd/kg. The reactivity swing during a cycle is 2.0 percent $\Delta k/k$.

It was assumed that plutonium available from power reactors would be used in FFTF (~64 percent Pu-239, 31.5 percent Pu-240). The plutonium atom fraction was 28.7 percent. It is possible that cleaner plutonium (~90 percent Pu-238, 10 percent Pu-240) could be made available for this special purpose reactor throughout its life. Future work will assess whether the cleaner plutonium has any advantage for the FFTF over the plutonium from the power reactor industry.

SECTION VIII

SHIELD DESIGN

The results of the shielding design analysis performed to date are described in the backup conceptual design report, Reference (2). The shield design results of Reference (2) are summarized below and on Table 8-1.

The backup design shielding analysis included calculations of neutron and gamma flux levels radially outward and axially upward from the core to assure the feasibility of preliminary shield designs. A neutron flux calculation was also made to determine irradiation levels on structural materials supporting the core. Except for the neutron source from the core, only gamma radiation presents a major shielding problem. Calculations were made to estimate gamma shielding requirements because of the radioactive sodium

TABLE 8-1

SUMMARY OF CALCULATED RADIATION LEVELS

	Dose Rate
Radiation Through Top Shield Plug:	
(1) During Operation	
Fission Neutrons (Neglecting Streaming) 10 nv	< 2 mRem/h
Fission and Capture Gammas (Neglecting Streaming)	< 1 mR/h
Na-24 Activity (Neglecting Streaming)	5 mR/h
TOTAL (Including Streaming)	< 50 mRem/h
(2) After Shutdown	
Na-24 Activity (Neglecting Streaming)	< 5 mR/h
(3) 10 Days After Shutdown, Top Shield Plug Removed	
Na-22 Activity	10 R/h
Radiation Through Concrete Shield Walls from Reactor and Coolant in Tank:	
(1) During Operation	
Fission Neutrons (Neglecting Streaming)	< 1 nv < 1 mRem/h
Fission and Capture Gammas (Neglecting Streaming)	1 mR/h
Na-24 Activity (Neglecting Streaming)	Negligible
TOTAL (Including Streaming)	< 10 mRem/h
(2) After Shutdown	Negligible
Radiation from Na-24 Activity During Operation:	
(1) Inside Heat Exchanger Cells (Including Closed Loops)	< 100, 000 R/h
(2) Operations Floor	1 mR/h

Integral Irradiation Doses on Structural Materials:
(t = 20 years at 75% load factor)

	nvt	
	> 1 Mev	Total
Center of Core	6×10^{23}	4×10^{24}
Reactor Vessel	7×10^{17}	3×10^{20}
Top Shield Plug	3×10^{10}	8×10^{14}
Grid Plate	8×10^{19}	1×10^{21}

coolant in the vessel pool, coolant pipes, and heat exchanger cells. Na-24 constitutes the major radioactive source in the coolant during operation and up to 10 days after shutdown. Then, Na-22 and fission products are expected to be the major radiation contributors. The conceptual design report presents estimates of expected radiation levels from most radioactive isotopes in the coolant. Also, some estimates of neutron activation are provided.

Estimated dose rates in most compartments during and after shutdown are provided for shielding thicknesses assumed to be adequate. These shielding thicknesses and the resulting radiation levels will continue to receive attention in consideration of acceptable industry standards. Preliminary values will be recalculated as the design progresses and design details become better known.

SECTION IX

THERMAL HYDRAULICS ANALYSIS

9.1 GENERAL

Evaluations made during the quarter were directed primarily at supporting the selection of the conceptual design which is described in detail in Reference (2).

The following is a summary of the analyses that were performed during the period:

9.2 SPENT FUEL TRANSFER THIMBLE

The fuel, coolant and thimble temperature distributions were determined for several spent fuel transfer thimble configurations (see Section V of this report). A 21-fin design was presented in the concept selection report[2] as a near optimum configuration. Final temperature distributions were not available at time of printing of Reference (2) and are presented as follows as a supplement to data reported in Appendix B9.3 of Reference (2).

Temperature distribution for a 21-fin transfer thimble having total heat loss is 25 kWt, and is tabulated in Table 9-1.

These results are lower than the conservative estimates presented in the concept selection report, Reference (2).

9.3 EFFECT OF PARTIAL FLOW BLOCKAGE IN FUEL ELEMENT

A heat transfer analysis has been made to determine the core outlet and channel outlet sodium temperature as a function of the flow reduction caused by a blockage area blocked off in a channel. Heat transfer to the six adjoining channels is included.

The analysis has been made to permit an evaluation of thermocouple instrumentation at the top of the channels as a means of monitoring the fuel for incipient failure. The results (Figure 9-1) show that a relatively small amount of blockage will give a measurable rise in temperature, and limiting the fuel cladding should be feasible. Fuel cladding integrity is a major consideration in reactor design, and a detailed evaluation will be conducted on the design of the FFTF fuel pins from considerations of bursting caused by fission gas pressure buildup, swelling strain, and thermal cycle fatigue. In this evaluation, overtemperature caused by potential accidents will be considered. From the present calculations, it appears that for a steady rate of flow reduction, control based on outlet temperature could avoid fuel damage.

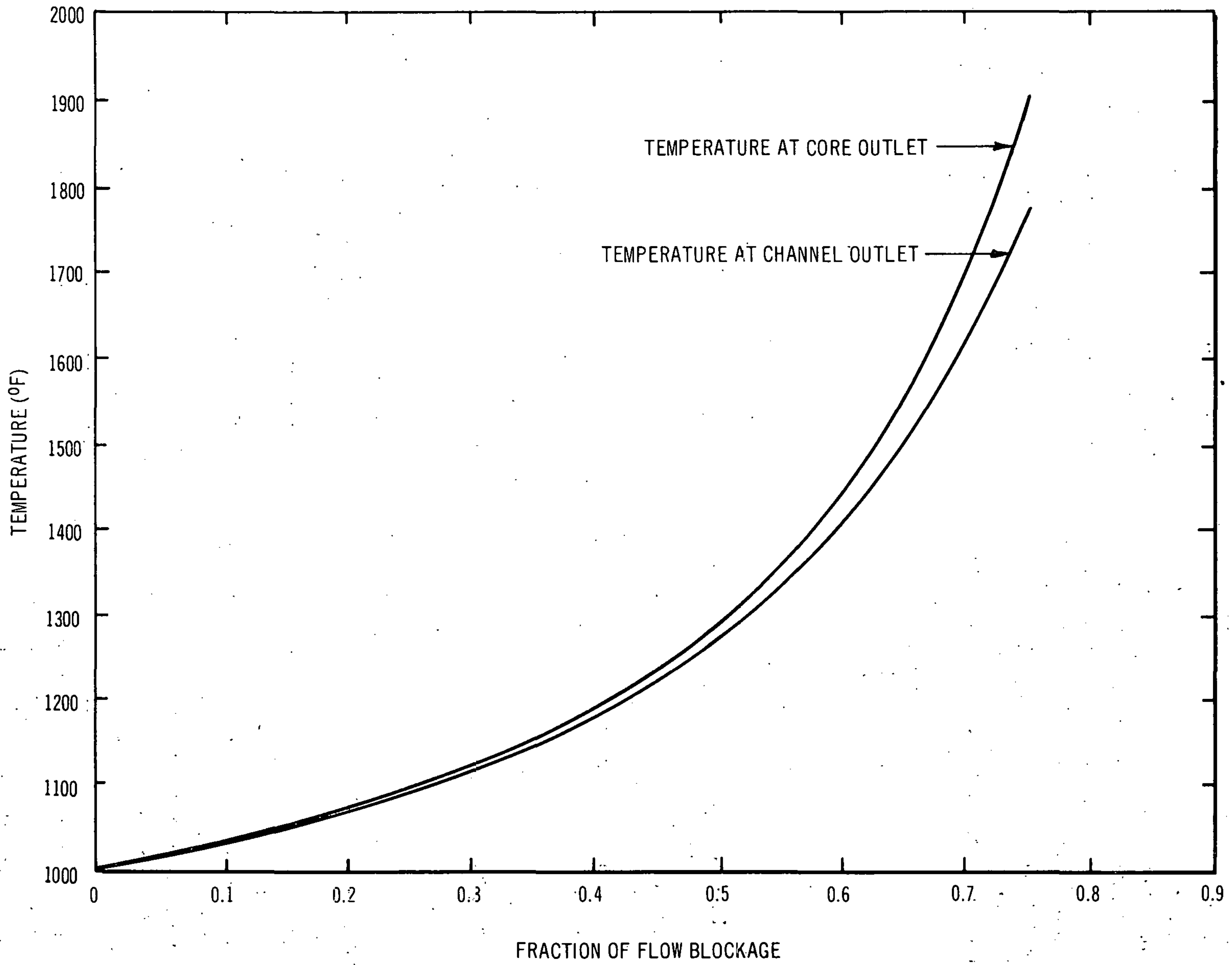

FIGURE 9-1. SODIUM COOLANT TEMPERATURES IN A PARTIALLY BLOCKED CHANNEL

TABLE 9-1

TEMPERATURE DISTRIBUTION FOR A 21-FIN THIMBLE

Distribution Up Thimble (feet)	Temperature °F (bundle)	Temperature °F (annulus)	Temperature °F (next to wall)
0	770	0	770
1	774	0	787
2	786	0	813
3	808	0	848
4	838	0	893
5	878	950	950
6	976	957	969
7	1041	989	984
8	1086	991	975
9	1042	971	952
10	1007	945	930
11	977	926	914
12	953	911	902
13	934	902	896
14	920	898	895
15	911	900	899
16	908	906	909

The six adjacent channels have a negligible response; hence instrumentation in one channel would not be adequate for monitoring such thermal effects in an adjacent channel.

9.4 HEAT REMOVAL CAPABILITY OF PRIMARY COOLANT LOOPS UNDER NATURAL CONVECTION

Calculations reported in the Appendix, Section 8.4 of Reference (1) show the ability of the FFTF backup design primary piping system to sustain the required natural circulation for removal of decay heat with only one of the primary coolant loops operative.

9.5 POST DBA HEAT REMOVAL

Scoping calculations were made on the heat removal problems of the dissociated core which accumulates following the DBA. These led to a recommendation of a collection pan at the bottom of the reactor cavity utilizing graphite with NaK cooling coils embedded in it.

9.6 CORE PRESSURE DROP

For the nominal fuel bundle geometry, the variation of pressure drop with the core height was selected to permit selection of a backup design conceptual core arrangement.

9.7 CONTROL ROD TEMPERATURES

An investigation was made to establish that a 3-rod configuration for each in-core control element did not result in excessive temperatures in the B_4C poison or rod cladding for a 2.50$ element.

9.8 FUEL PIN TEMPERATURE

Previous calculations supporting the selected fuel pin design were rechecked prior to starting further studies which will lead to final pin size, cladding thickness, and pin spacing selection.

9.9 BLAST STRUCTURES

A review of available literature on the blast effects of the DBA was made. A conceptual configuration was established, and preliminary analyses to establish feasibility were performed. These blast structures described in the concept selection report[(2)] can be seen in Figures 4-1 and 4-2 in Section IV of this report.

SECTION X

SAFEGUARDS

10.1 GENERAL

Safeguards criteria establishment and preliminary accident analysis initiated for the FFTF backup design during the quarter are described in detail in References (1) and (2). Safeguards criteria are being developed for the various reactor systems to assure that reactor protection standards can be maintained during all modes of reactor operation. Preliminary accident analysis for loss of coolant flow and reactivity insertion accidents have been initiated. The events leading to the DBA are being studied to determine the sequence leading to core compaction and explosive disassembly.

The following preliminary safeguards criteria developed for the following reactor systems are given in subsections 10.2 through 10.6.

- Core coolant system
- Core and vessel internals
- Control and safety system
- Refueling system
- Containment

The preliminary accident analyses have been based upon a simplified reactor model, using only the Doppler coefficient to determine reactivity feedback, because the purpose is to determine overall system response. Results for two flow coastdown loss-of-coolant accidents and the single bundle meltdown reactivity insertion accident are shown in the FFTF Conceptual Design Report.[2] Studies are being conducted, using the Fore-II computer code, to determine the initiating mechanism for the DBA. The model used is based upon a total loss of core sodium flow, with a reactivity insertion of 0.0037 $\Delta k/k$ in 100 msec (12.3 \$/sec) established by the preferential sodium voiding pattern assumed. The core power, fuel temperature profiles, and reactivity obtained from the Fore-II results are to be used in the modified Bethe-Tait meltdown code to determine the energy released by the core disassembly.

10.2 CORE COOLANT SYSTEM

In general, no operating or accident condition should result in loss of positive cooling capability for the core. Five accident conditions have been identified for the coolant system. The initiating mechanisms and the allowable damage limits are listed below. In each case a reactor scram is assumed to be initiated. The accident consequence limit of no sodium boiling or cladding damage means that continued operation of the system should be possible without having to inspect the driver fuel. The accident limit of no fuel released from the cladding does not preclude damage to the fuel such that inspection of driver fuel and replacement of some fuel elements might be required.

10.2.1 Loss of Site Power

With the power off the main pumps, a flow coastdown must be provided to remove the power produced before scram is completed and the energy stored in the fuel. The capability to remove decay heat is also required. No coolant boiling or cladding damage should result.

10.2.2 Loss of a Single Coolant Loop Without Sodium Leakage

The heat removal capacity of a single loop is lost by mechanical damage to the pump or IHX or by a failure in the secondary loop. The capability to remove the power produced before scram is completed and the energy stored in the fuel with the remaining loop(s) must be provided. The removal of the decay heat load with one loop inoperative is required. No coolant boiling or cladding damage should occur.

10.2.3 Sodium Leakage from a Single Loop

A low leakage rate resulting from the loss of an appendage or a small hole in the piping will lower the coolant level in the vessel. The sodium level should be maintained above the outlet nozzles unless an emergency coolant loop is provided. No siphoning of sodium from the vessel or blowdown expulsion should occur. The power produced before the scram is completed and the energy stored in the fuel must be removed. No coolant boiling or cladding damage should occur.

10.2.4 Guillotine Piping Failure

The consequences are the same as a small pipe break except that the coolant level will be lowered faster, so that more heat will be stored in the remaining coolant. No release of fuel from the cladding should occur.

10.2.5 Vessel Leakage

The loss of sodium should be limited so the main nozzles are not uncovered. Lowering of sodium level will result in more heat being stored in the coolant than occurs during the small pipe break accident. No fuel should be released from the cladding.

10.3 CORE AND VESSEL INTERNALS

In general, the design of the fuel and the components supporting the fuel and controlling the flow will assure that operation over the power range from shutdown to the design overpower condition can be conducted without introducing fuel or component distortion that would result in a positive reactivity addition or inducing either dynamic or power oscillations. The net reactivity feedback for increasing power shall be nonpositive, and the net sodium void coefficient shall be nonpositive.

Positive control of the location of fuel, structural, and fixed neutron absorbing elements is required during all phases of reactor operation. The locking or holddown mechanisms used to fix the positions of the fuel, open loop tests, and any stationary poison elements must be capable of functioning over the entire temperature and coolant flow range, with redundant safeguards available to prevent bundle or poison element floating during high coolant flows or drop-in when coolant flow is reduced.

The accident initiated by blockage of coolant flow in a single channel shall be limited by a low fuel bundle worth and the Doppler effect so the power transient caused by the reactivity addition of the slumped bundle is reversed before coolant boiling is induced across the core. The internal core structure shall be capable of preventing the sodium void in the blocked channel from propagating to other channels, and shall not distort to block flow in the adjacent channels. No fuel is to be released from the cladding except in the bundle with the blocked coolant passage.

10.4 CONTROL AND SAFETY SYSTEMS

In general, the reactivity control and safety system shall be capable of terminating any reactor excursion and return the core to a shutdown condition. The ability of the control system to scram the reactor is a function of both response and insertion time and available reactivity worth.

The shutdown system shall consist of both a primary and a secondary system, each of which will be capable of inserting sufficient control to terminate any credible power excursion. The worth of the primary system shall be sufficient to terminate the maximum credible power excursion and return the reactor to cold standby, or refueling temperature, with one control element stuck in its most reactive position. The secondary system shall be capable of terminating the maximum credible power excursion and returning the reactor to a hot standby or isothermal condition at normal inlet temperature, when the primary system is incapable of operation. The stuck rod criterion is not considered applicable to the secondary system under the single failure theory because the demand for the secondary occurs only after the primary system fails to operate.

The maximum worth of any single element in the primary control system shall be limited so the accidental withdrawal of a single rod at its maximum credible rate will not result in a power transient that cannot be terminated by the balance of the control elements before sodium boiling or cladding damage occurs in the average channel.

If the secondary control system is operated so that the elements are normally withdrawn above the core, then the rod withdrawal limitation on a single element worth is not a requirement, because accidental motion of the rods could only move them to a position of greater control.

10.5 REFUELING

The safeguards guidelines for refueling are established so the possibility of a reactor power excursion during refueling operations is eliminated. The guidelines will require both mechanical or engineered safeguards and procedural controls to insure that incidents do not occur during refueling.

Instrumentation must be provided so that the degree of criticality is always known. During all fuel and experiment loading, the worth of the element to be loaded will be determined and sufficient reactivity control will always be inserted so the core will remain subcritical after the element is loaded. The worth of the reactivity control available must be greater than the total reactivity addition possible during a refueling cycle. Procedures will be established so the proper positioning of all fuel elements and experiments is assured before reactor power operations are initiated, so the possibility of a reactivity change caused by the shifting or settling of a bundle or experiment is eliminated.

10.6 DESIGN BASIS ACCIDENT

The safeguards guidelines for the DBA are intended to specify the limits for the consequences of the excursion, given the maximum accident based upon multiple independent failures in the reactor system.

The DBA analysis will establish the initiating events for the projected explosive disassembly of the core and the magnitude of the energy release. The safeguards protections for the DBA include containment of the radioactive release, blast protection of structure and containment, and post incident cooling of the nuclear debris.

All piping penetrations through the containment barriers will either be equipped with fast acting isolation valves, one on each side of the barrier, or will be operated on a batching basis. All other penetrations, instrument, electrical, etc., will be sealed so their integrity is equivalent to that of the barrier, and shall have further provisions for periodic leak testing incorporated into their design.

Both containment regions shall have provisions for periodic leak testing to assure that leakage rates are less than the allowable rates.

Procedural safeguards shall be established to insure that dual containment integrity is maintained during refueling, fuel and experiment transfer operations, and maintenance operations when fuel is loaded in the core.

10.7 CONTAINMENT DESIGN GUIDELINES

The containment system must provide protection from radioactive release to both the public and operations personnel during all modes of reactor operation. Protection shall assure that dosage rates at the site boundary, the nearest population center, and in the control room will be less than the specified maximum dose rates given in applicable standards following the DBA.

The maximum radioactive release inside the containment and the gas pressures within the building shall be determined from the DBA analysis. These data and the site meteorology shall be used to determine the containment allowable leakage rates. The containment shall have two barriers between the reactor and primary sodium process equipment and the environment. The first barrier shall enclose the primary process system, the reactor core and the refueling cell. The second barrier shall enclose the balance of the reactor system.

In addition to limiting the radiation release rates, the containment structure shall be capable of withstanding the seismic and weather loadings determined for the site, and shall accommodate all calculated equipment static and operating loads. The inner barrier shall also be capable of withstanding the explosive energy release associated with the DBA without loss of containment integrity.

The possibility of sodium chemical reactions shall be minimized by filling the region enclosed by the inner barrier with inert gases and by minimizing the gas volumes contained. Sodium releases in the air filled outer containment region will be of very low probability, because the secondary sodium piping will be enclosed in pipe tunnels. Only limited amounts of sodium shall be available in any other potential sodium spill from the cold traps or other auxiliary sodium systems located in the outer region.

Containment of radioactive release will be accomplished by using a double barrier system. The first barrier shall be the equipment and refueling cells, equipped with steel liners. The second barrier shall be the reactor building. The allowable leakage rates for the containment barriers shall be established so that the radioactive release calculated for the DBA will not result in dose rates in the control room, at the site boundary and at the nearest population center that are greater than those specified by the applicable regulations and standards.

The blast effects of the core disassembly shall be conservatively estimated and the various structures within the reactor cavity will be designed to absorb the blast loading without breaching containment barriers or disrupting the post-incident cooling capabilities. The vessel head shall have an adequate holddown mechanism so that no missiles are generated at the head or pass through it. Adequate blast shielding will be provided so that any explosive energies directed in the radial or downward directions do not generate missiles or damage the post-incident cooling system.

The post-incident cooling system shall be capable of withstanding the blast effects and be capable of removing the stored and delay heat loads of the fuel materials in a disassembled configuration. Provisions shall be made to assure that the fuel debris does not tend to form a critical assembly and the debris temperature does not become high enough to initiate melting of the supporting structure.

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