ANL-7168
Volume I)

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ANL-7168
(Volume I)

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Argonne National Laboratory

FAST REACTOR TEST FACILITY (FARET)

Volume I—Description and Program

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ANL-7168 (Volume I)
Reactor Technology
(TID-4500)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

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FAST REACTOR TEST FACILITY
(FARET)

Volume I—Description and Program

Edited by J. D. Geier

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IN NUCLEAR SCIENCE ABSTRACTS

March 1966

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PREFACE

Volume I of this report describes the FARET Facility and its related experimental program as they existed at the time of project termination. At that time the project, which was initiated in late 1962, had reached a status wherein the general Title II design, as performed by the Bechtel Corporation, was essentially complete. The ANL portion of the Title II design, which included the reactor core and the special purpose remote controlled equipment necessary for reactor operation and fuel handling, had reached the advanced prototype development stage. Also at that time, the reactor vessel preliminary design under contract to the Babcock and Wilcox Company had been completed. The Construction Manager, United Engineers and Constructors, Inc., was ready to initiate field construction activities and procurement of major long delivery equipment items such as Liquid Metal Heat Exchangers and Pumps.

At the time of project termination, the then current schedule anticipated construction completion by July 1969. Facility checkout and sodium filling operations would have followed directly.

Volume II of this report describes the safety oriented considerations in the FARET design and operational program. It is a summary of the Preliminary Safety Analysis Report and subsequent reviews of that document.

ACKNOWLEDGMENTS

This document is a collection of material which was written by the ANL staff in connection with their work on the FARET Project. Principal authors include the following:

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I. INTRODUCTION

This report consists of a description of the FARET Project and Program at the time of Project termination. It describes the salient features of the facility, reflecting details and changes which have developed during the conceptual and definitive design phases. The experimental program and associated experimental facilities are described. This report emphasizes the reactor and its associated equipment as well as the experimental and operational capability of the reactor system. It presents a description of normal facility systems. It should not be considered per se as a design report.

The FARET Project was initiated in December 1962, with a Feasibility Study to establish the basic design concept and to provide preliminary cost and schedule estimates. The highlights of project activities since that time are:

- (1) The Feasibility Study performed in conjunction with Atomics International and Shaw Metz and Associates was completed April 1963.
- (2) The Title I design was initiated by the Bechtel Corporation in June 1963.
- (3) The Title II design was started in November 1963.
- (4) The construction contractor was selected and key personnel were assigned to the Project in April 1964.
- (5) The Site Preparation Package was issued for start of construction in May 1964.
- (6) The Preliminary Safety Analysis Report was submitted for Commission approval in May 1964.
- (7) The reactor vessel procurement was initiated in August 1964.
- (8) The principal long lead procurement packages such as, liquid metal pumps and heat exchangers and control and instrumentation were issued by the Architect-Engineer for Laboratory and Commission review and approval during September, October and November 1964.
- (9) Reactor vessel vendor selection was approved in April 1965.
- (10) The Bechtel part of Title II design was essentially completed in June 1965.
- (11) The Core I Design Study was completed in September of 1965.
- (12) The preliminary reactor vessel design by Babcock and Wilcox was completed in November of 1965.

(13) The final meeting with the ACRS was held in November 1965. Project approval was indicated at that time.

(14) The Project was terminated by the AEC in December 1965.

At the time of Project termination the Title II design of the facility proper (the A-E design) was essentially complete. All A-E drawings and specifications had either been signed off by the Laboratory and the Commission or had been submitted for signature. The reactor and reactor component design (ANL design) was approximately 50 percent completed with work proceeding on each system and component. The overall concepts were developed and established, and detail design of components was in varying degrees of progress. As a consequence of the status of design of these more special reactor components, their description will not be completely consistent with respect to definitive detail. Where such detail is lacking, criteria and plans are presented.

With project termination there was initiated a series of Project terminal efforts. These consisted primarily of preparation of in-house terminal memoranda related to individual work phase-out, and the following topical reports:

- (1) "FARET Containment Structure Analysis and Design," ANL-7154, by R. Seidensticker et al. (to be published).
- (2) "Design, Performance and Evaluation of Shielding Window for FARET Containment Cell," ANL-7157, by T. W. Eckels and A. Smaardyk, July 1966.
- (3) "FARET Control Rod Drive Mechanisms," ANL-7158, by T. E. Sullivan, E. Hutter and G. Giorgis, June 1966.
- (4) "Fuel Handling Machine Design," ANL-7159, by A. P. Grunwald (to be published).
- (5) "FARET In-core Instrument Development," ANL-7161, by G. F. Popper et al., July 1966.
- (6) "Stress Analysis of a Reactor Core Support Structure Consisting of Two Interconnected Multiregion Plates," ANL-7162, by A. Marchertas (to be published).
- (7) "FARET Reactor Vessel Design," ANL-7163, by R. Seidensticker et al. (to be published).

Some work which was being conducted in the FARET Project is expected to be continued because of its general applicability to the LMFBR Program. This work includes the following:

1. Completion of the control rod drive components test to evaluate a number of improved design features. These include the increased stroke over the EBR-II design and the ability to operate in a 1200°F sodium environment.

2. Completion of certain portions of the fuel handling machine testing program. These are related to mechanical handling functions involving installation or removal of subassemblies from the sodium and sodium vapor environment.

3. Continuation of the in-core instrument development program. This includes work on subassembly connectors, thermocouples, pressure transducers and flowmeters for experimental fast reactor in-core instrumentation.

II. SUMMARY

FARET was designed as an experimental 50 MW(t) sodium cooled fast reactor facility. Its salient feature is the capability to irradiate a broad variety of fuels over a wide range of operating conditions. It was designed to permit simultaneous experimental irradiation of large numbers of different fuel elements under controlled variable conditions. Another important feature is the capability to provide instrumentation within the reactor core to measure the important irradiation parameters. The broad flexibility in operating temperature, flow rate, flux and power density, which is provided permits achieving operating conditions in FARET comparable to those which will be encountered in power reactors of the future. The wide latitude in experimental loading permits experiments to be conducted for the determination of operating parameters related to stability and safety of fast power reactors. Table 1 lists the important parameters of operation established for Core I of the FARET reactor. See Vol. II for other core data.

The primary objectives of the FARET facility are to provide the loading flexibility required to conduct the desired experiments; to provide the appropriate operational environment of flux, power density, and sodium temperature, flow and pressure; and to remove and dissipate the energy produced in the reactor. The facility design reflects these objectives both for the experimental programs which were originally planned as well as those which could arise in the future. In many instances, capability has been provided even though a specific experimental requirement was not developed. In other words, operational and experimental flexibility were provided in excess of established requirements for known usage. This broad and long range approach reflects the experience the Laboratory has had with other experimental reactors and reflects the established need for utilizing these facilities to maximum advantage.

TABLE 1. Summary: FARET Design Data

CORE ICore Dimensions

Equivalent diam, in.	17.5
Height, in.	18
Total volume, liters	70

Core Composition, vol-%

	Metal	Ceramic
Fuel	30	37
Structure	22	22
Sodium	48	40
Void	0	1

Subassemblies

Mixed oxide	24
Mixed carbide	16
Pu,U alloy	8
Control rods (oscillator rod included)	12
Special subassemblies	3
Radial reflector	238
Total	301
Configuration	Hexagonal
Dimension across flats, in.	2.290
Lattice spacing (pitch), in.	2.320
Tube wall thickness, in.	0.040

Subassembly Data

Fuel Material	Subassembly Designation	Fuel Type	Clad Material	No. of Subassemblies	Instrumented Subassembly	Maximum Linear Heat Flux (Design), kW/ft	Maximum Clad Temp, °F (°C)	Maximum Fuel Temp, °F (°C)	Burnups* at Removal, a/o	Maximum Clad Thermal Stress, psi	Coolant Velocity, ft/sec
Oxide	O1	Pellets	304	4	(1)	15	1250 (675)	3220 (1770)	5, 9, 12, 14	13,300	16
	O2	Pellets	304	4	(1)	22	1290 (700)	4220 (2330)	5, 8, 10, 12	19,400	23
	O3	Vipak	304	4	(1)	15	1250 (675)	3220 (1770)	5, 9, 12, 14	13,300	16
	O4	Vipak	304	4	(1)	22	1290 (700)	4220 (2330)	5, 8, 10, 12	19,400	23
	O5	Pellets	HSA	4	(1)	22	1340 (725)	4250 (2340)	5, 9, 12, 15	-	20
	O6	Vipak	HSA	4	(1)	22	1340 (725)	4250 (2340)	5, 9, 12, 15	-	20
Carbide	C1	Pellets	304	4	(1)	15	1250 (675)	2030 (1110)	5, 9, 12, 14	13,300	16
	C2	Vipak	304	4	(1)	22	1290 (700)	2520 (1380)	5, 9, 12, 14	19,400	23
	C3	Pellets	HSA	4	(1)	29	1340 (725)	3050 (1680)	5, 9, 12, 15	-	30
	C4	Vipak	HSA	4	(1)	36	1340 (725)	3390 (1870)	5, 9, 12, 15	-	39
Metal	M1	U-Pu-Zr	HSA	4	(1)	14	1250 (675)	1580 (860)	5, 9, 12, 14	-	20
	M2	U-Pu-Ti	V-Ti	4	(1)	14	1250 (675)	1580 (860)	5, 9, 12, 14	-	20
Advanced	C4S	-	-	3	(3)						
				51	(15)						

NOTES: 1. Vipak means "vibratory compacted;" HSA means "high-strength alloy," not necessarily the same for oxide as for carbide, etc.

2. Coolant inlet temperature is 810°F. Average outlet temperature from each subassembly is 1050°F. (Exception: outlet temperature from O5 and O6 subassemblies is 1085.) Axial peaking factor = 1.2.

3. Maximum clad temperature include both a 1.33 factor for flow and power maldistribution effects on coolant temperature rise, and a 1.10 uncertainty factor. Maximum fuel temperatures include the 1.33 factor but no uncertainty factor.

*Approximate burnup intervals.

TABLE 1 (Contd.)

Fuel Element Data

Fuel Material	UO ₂ -15 w/o PuO ₂	UC-15 w/o PuC	Metal ^b
Elements per subassembly	37	37	61
Fuel height, in.	18	18	18
Clad outside diam, in.	0.290	0.290	0.220
Clad thickness, in.	0.020	0.020 ^a	0.016
Spacer wire diam, in.	0.057	0.057	0.052
Axial blanket height (each end), in.	6	6	6
Gas reservoir, in.	25	25	25
Element height, in.	55	55	55
Effective fuel density, %	80	80	70
Bond material	Helium	Helium	Sodium
Fuel U/Pu atom ratio	6/1	6/1	5/1
Fuel details	Coprecipitated; O/M 1.97-2.00	Single Phase; Solid Solution	

^a0.015 in C4 subassemblies.^bU-15 w/o Pu-6.5 w/o Ti; U-15 w/o Pu-10 w/o Zr. (See Subassembly Data.)Upper Reflector

Height, in. (each end)	6
Equivalent diam, in.	17.5

Radial Reflector

Height, in.	80
Equivalent outside diam, in.	42.3
Radial thickness, in.	13.1

Control Rod Composition, vol-%

Boron carbide	55
Stainless steel	20
Sodium	25

Upper Reflector Composition, vol-%

Stainless steel	48
Sodium	52

Radial Reflector Composition, vol-%

Stainless steel	82
Sodium	18

Control Rod

Total (includes one oscillator rod)	12
Operating drive	Ball nut and screw
Velocity, in./min	6
Total movement, in.	24
Scram drive assist.	Pneumatic

TABLE 1 (Contd.)

Nuclear Data

Neutron flux at center of core, $n/(cm^2)(sec)$	
above 1.35 MeV	0.8×10^{15}
below 1.35 MeV	2.8×10^{15}
Total	3.6×10^{15}
Prompt neutron lifetime, μsec	0.21
Effective delayed neutron fraction	0.0058

Reactor Control (Average Reactivity Coefficients),
 $\Delta k/\Delta T ^\circ C$

Overall Doppler at 750°K	
Uranium	-0.45×10^{-6}
Plutonium	$< +0.1 \times 10^{-6}$
Radial core expansion	-10×10^{-6}
Axial fuel expansion (row)	
Oxide fuel (1 and 2)	-0.55×10^{-6}
Carbide fuel (3)	-1.63×10^{-6}
Oxide fuel (4)	-1.11×10^{-6}
Metal fuel (5)	-1.80×10^{-6}
Sodium expansion (row)	
Oxide region (1 and 2)	-1.1×10^{-6}
Carbide (3)	-2.5×10^{-6}
Oxide (4)	-4.1×10^{-6}
Metal (5)	-1.6×10^{-6}
Total core	-10×10^{-6}
Core and reflector	-14×10^{-6}

Estimates of Attainable Worth of Control Rods

Composition (vol-%): B_4C /coolant/structure	55/25/20
No. of rods	12
70% B^{10} enrichment $\Delta k/k$	0.075-0.085
95% B^{10} enrichment $\Delta k/k$	0.085-0.10

Excess and Shutdown Reactivity, $\Delta k/k$

Heavy atom burnup	0.035-0.060*
Operational Temperature	
Sodium ($\Delta T_{avg} = 500^\circ C$ core and reflector)	0.007
Radial core expansion ($\Delta T_{avg} = 500^\circ C$)	0.005
Axial fuel expansion ($\Delta T_{avg} = 1500^\circ C$ oxide)	
($\Delta T_{avg} = 1100^\circ C$ carbide)	0.004
($\Delta T_{avg} = 500^\circ C$ metal)	
Total operational temperature	0.016
Excess reactivity	0.051-0.076
Shutdown reactivity	0.020
Design Worth of Control System	0.071-0.096

TABLE 1 (Contd.)

Power density	
Average, MW(t)/liter	0.71
Overall flux peaking factor	1.6-1.7
Axial flux peaking factor	1.2
Radial flux peaking factor	1.3-1.4

MAIN PRIMARY COOLANT SYSTEMCoolant Data--50 MW Operation

Number of loops	1
Primary coolant	Sodium
Flow, lb/hr	2.991×10^6
Flow, gpm at 960°F	7210
Core pressure drop, psi	80
Reactor core flow, lb/hr	2.374×10^6
Reactor core bypass flow, lb/hr	0.617×10^6
Temperatures (max), °F	
Reactor core outlet	1200
Reactor core inlet	960
Heat exchanger inlet	1150
Heat exchanger outlet	1050

Piping Data

Piping material	Type 304 SS
Reactor outlet piping	
Design pressure, psig	-15, 50
Design temperature, °F	1150
Size, diam, in.	24
Heat exchanger to pump piping	
Design pressure, psig	-15, 50
Design temperature, °F	1050
Size, diam, in.	20
Reactor inlet piping	
Design pressure, psig	-15, 150
Design temperature, °F	1050
Size, diam, in.	14

Pumping Equipment

Main primary pump	
Number of pumps	1
Type	Mechanical seal free surface, vertical centrifugal
Speed control	Eddy current coupling

TABLE 1 (Contd.)

Main primary pump (Contd.)

Turndown ratio	10/1
Design pressure, psig	-15, 150
Design temperature, °F	1075
Design flow, gpm at 960°F	7500
Total dynamic head, ft at 960°F	306
Motor horsepower, hp	700

Main primary shutdown pump

Number of pumps	1
Type	Stainless steel a.c. electromagnetic
Design pressure, psig	-15, 100
Design temperature, °F	1075
Design flow, gpm at operating temperature	490
Operating temperature, °F (max)	1050
Total dynamic head, ft at 1050°F	30
Power input, kW	22

Intermediate Heat Exchanger

Number of units	1
Material	Type 304 SS
Type	Shell and tube
Tube side	Secondary sodium
Shell side	Primary sodium
Design pressure	
Tube side, psig	-15, 75
Shell side, psig	-15, 75
Design temperature, °F	1175

MAIN SECONDARY COOLANT SYSTEMCoolant Data--50 MW Operation

Number of loops	1
Secondary coolant	Sodium
Flow rate, lb/hr	2.97×10^6
Flow rate, gpm at 870°F	7100
Temperatures (max), °F	
Heat exchanger inlet	1060
Heat exchanger outlet	870
Air cooled heat exchanger inlet, °F	
Maximum	91
Minimum	-20

TABLE 1 (Contd.)

Piping Data

Piping material	Type 304 SS
Design pressure, psig	-15, 50
Design temperature, °F	1150/1050
Size, diam, in.	14

Pumping Equipment

Number of pumps	1
Material	Type 304 SS
Type	Mechanical seal, free surface, vertical centrifugal
Speed control	Eddy current coupling
Turndown ratio	10/1
Design pressure, psig	-15, 75
Design temperature, °F	1075
Design flow, gpm at operating temperature	7500
Total dynamic head, ft at 960°F	137
Motor horsepower, hp	300

Heat Exchanger

Number of units	1
Type	Sodium, air cooled fin tube
Material	Type 304 SS
Design pressure, psig	-15, 75
Design temperature, °F	1175
Control	
Fans	Variable speed coupling
Dampers	Pneumatically operated
Maximum fan input, hp (each of two units)	500
Tube bank preheating	Electrical

AUXILIARY PRIMARY COOLENT SYSTEMCoolant Data

Heat removal capacity, MW	1
Number of loops	1

TABLE 1 (Contd.)

Coolant Data (Contd.)

Coolant	Sodium
Flow rate, lb/hr	6160/196,000
Flow rate, gpm at 960°F	24/475
Temperatures, °F	
Heat exchanger inlet	1018
Heat exchanger outlet	960

Piping Data

Material	Type 304 SS
Design pressure, psig	
Pump suction	-15, 75
Pump discharge	100
Design temperature, °F	1050
Size, diam, in.	6
Concentric containment piping	
Piping material	Type 304 SS
Design pressure, psig	-15, 75, 100
Design temperature, °F	1050
Size, diam (nominal), in.	10
Annulus fluid	Eutectic NaK

Pumping Equipment

Number of pumps	1
Type	a.c. Electromagnetic
Material	Type 304 SS
Design pressure, psig	-15, 100
Design temperature, °F	1100
Design flow, lb/hr	6160/196,000
Total dynamic head, ft at 1050°F	2.5/30
Total energy input, kW	22

Heat Exchanger

Number of units	1
Type	Shell and tube, fully contained
Material	Type 304 SS
Design pressure, psig	
Tube side	-15, 75
Shell side	-15, 75
Annulus NaK containment	-15, 50

TABLE 1 (Contd.)

Heat Exchanger (Contd.)

Design temperature, °F	
Tube side	1100
Shell side	1175
Annulus NaK containment	1175

AUXILIARY SECONDARY COOLANT SYSTEMCoolant Data

Heat rejection capacity, MW	1.6
Number of loops	1
Coolant	Sodium
Flow rate, lb/hr	185,000
Flow rate, gpm at 960°F	450
Temperatures, °F	
Heat exchanger inlet	948
Heat exchanger outlet	852
Air cooled heat exchanger inlet	
Maximum	91
Minimum	-20

Piping Data

Piping material	Type 304 SS
Design pressure, psig	
Pump suction	-15
Pump discharge	50
Design temperature, °F	1050
Size, diam, in.	6

Pumping Equipment

Number of pumps	1
Type	a.c. Electromagnetic
Material	Type 304 SS
Design pressure, psig	-15, 75
Design temperature, °F	1100
Design flow, lb/hr	185,000
Total dynamic head, ft at 1050°F	60
Total energy input, kW	41

TABLE 1 (Contd.)

Heat Exchanger

Number of units	1
Type	Sodium, air cooled fin tube
Material	Type 304 SS
Design pressure, psig	-15, 75
Design temperature, °F	1100
Control	
Fans	Variable speed coupling
Dampers	Pneumatically operated
Tube bank preheating	Electrical

GENERAL BUILDING DIMENSION AND GROSS SPACE ALLOCATION

<u>Reactor Wing</u>	<u>Length in ft</u>	<u>Width in ft</u>	<u>Height in ft</u>	<u>Area ft²</u>	<u>Gross Volume ft³</u>
			124.5	(~2 levels)	
Overall	168	84	42 below Grad. 82.5 above Grad.	28,250	1,800,000
Cell	68	20	37	1,360	57,700*
Annex cell	37	24	27.5	900	21,340 [†]
Vault	No continuous dimension			2,650	75,400
Front cell area					
Basement	168	23	17	3,600	61,300
1st floor	168	23	19	3,900	73,500
2nd floor	168	23	28	3,900	108,200
Rear cell area					
Basement	No continuous dimension			(2 levels) 9,350	112,200
Ground floor	168	60	82.5	7,750	781,000 ^x
Loft	24	60	-	1,440	-
Support wing					
Basement	144	50	15	7,200	108,000
1st floor	144	50	12	7,200	86,400

*Exterior volume = 110,000 ft³.[†]19.5 ft head room.^xDoes not include cell area or volume.

A particularly important design objective is the provision for instrumenting in-reactor components. The FARET design permits instrumented subassemblies and other components to be installed in the reactor vessel. Provisions for bringing the in-core instrument leads out of the vessel to appropriate and continuously accessible readout locations in the plant are incorporated. (Mechanical connectors for use in a 1200°F sodium vapor environment are still being developed to facilitate the installation and removal of instrumented subassemblies. This effort is being continued because of its continued importance to fast reactor development even though the FARET project has been terminated.)

In-reactor loops were planned at the time of project termination. These loops would permit the testing of advanced fuels and the performance of safety experiments that have a higher than average risk factor.

The in-reactor loop program included the irradiation of advanced fuels at high burnup rates, claddings to high integrated neutron fluxes, and vented fuels. In addition fuel meltdown tests to study resulting fuel configuration, fuel failure propagation, and loop contamination were being considered. The re-entrant type loops would occupy the space of one or more subassemblies. It was planned to have a small loop installed in a single reflector subassembly location during operation of the first core.

Core I was established for the first core loading of the reactor. The facility requirements, however, were established for a wide range of core loadings and for experimental and operational capability well beyond those required for the first core. The total available (useable) core volume for irradiation experiments is about 500 liters (slightly greater than 30,000 cubic inches).

The FARET reactor design resembles the EBR-II in geometry of the core and subassemblies. The FARET reactor design is the outgrowth of many years of ANL participation in the AEC fast reactor development program. Much of the technology which was being incorporated into the FARET design had evolved from the EBR-II and EBR-I projects and the numerous liquid metal experiments which had been conducted at the Laboratory. Consequently, the details related to subassembly sizes, proportions, materials selection, and general arrangement of in-core components such as thimbles, control rods, etc., were the product of continuing experience and design evolution. Appropriate considerations regarding self-welding of materials, clearances, compacting requirements, fuel and subassembly bowing, warping, sticking and other deformation tendencies related to thermal cycling and in-core environmental problems were implicit throughout the design. This permits common utilization of many systems and components in support of EBR-II and FARET. The principal difference between the EBR-II and FARET subassemblies is their overall length. The FARET subassembly is

some seven feet longer because each subassembly contains an upper shielding section which, when installed in the reactor, comprises the vertical shield for the reactor. This feature is very much like the system incorporated in EBR-I and provides two very important experimental capabilities: (1) the top of each subassembly is readily accessible since the shielding is incorporated in the subassembly rather than in plugs or in heavy shielding above the reactor; and (2) the shielding section provides "a handle" for each subassembly through which instrumentation leads and other forms of communication can be provided from the active core to the upper end of the subassembly. The use of common dimensions and configurations for both FARET and EBR-II subassemblies also simplified some of the expected fuel and subassembly development, and the procurement of hardware common to both reactors. It also makes possible more definitive proof testing of FARET reactor components in EBR-II prior to operation in FARET.

The primary objective of the planned initial use of FARET was to develop the maximum possible information relevant to potential fuel loadings for the Prototype Reactor in the AEC's Fast Reactor Development Program. It was believed that core and fuel specifications would be developed for the FARET Core I before those for the Prototype Reactor were established. Therefore, the Core I design, fuel element specifications, and reactor operating conditions were planned so that they "bracketed" the corresponding design, specifications, and operating conditions for the Prototype Reactor to the extent they could be anticipated. Core I would test several different fuel element designs, in order to provide information which could be used to evaluate, define, or verify operating limits for the first core of the prototype. It would also yield information related to the choice of a second core for the prototype.

It was planned to achieve the above objectives with a core consisting of 51 subassemblies, having a height of 18 in., and a core volume of about 70 liters. Twenty-four of the subassemblies would contain Pu-U oxide fuel, 16 Pu-U carbide fuel, eight Pu-U metal fuel, and three would be available for special fuels. This distribution and allocation of fuel could be changed as more becomes known about the relative availabilities and/or potentialities of the various fuels and as plans for the Prototype Reactor develop. Core I also includes an assumed mixed mean outlet sodium temperature from the reactor of 1050°F. The linear heat fluxes, temperatures, coolant velocities, etc., calculated for Core I are summarized in Table 1. The FARET primary sodium coolant system is designed for a capability of 50 MW(t) to operate over a temperature range of 790 to 1150°F mixed mean outlet temperature from the reactor vessel. Heat is transferred in an intermediate heat exchanger to a secondary sodium system from which it will dissipate to the atmosphere in an air to sodium heat exchange system.

The selected FARET Facility site is located in the East Area of the National Reactor Testing Station, Idaho about 4000 ft southeast of the EBR-II. The facility comprises a single building of conventional construction and a few miscellaneous structures as shown in the frontispiece. The building consists of two principal areas: a reactor building and a support wing. The reactor building houses the reactor vessel and reactor coolant systems, the fuel handling equipment, and various auxiliary systems. The support wing contains the offices, personnel facilities, plant support utilities, and the reactor control room. The reactor vessel is located in a shielded cavity which is interconnected with the vault containing the primary coolant systems and with the cell containing the fuel handling and reactor control systems. These three interconnected spaces constitute the containment structure. They are arranged to permit an inert gas environment for both fuel handling and reactor access and to minimize the hazards involved in the event of a sodium leak. Although this basic arrangement is the same as shown in earlier descriptions of the facility, certain significant changes occurred during the development of the detailed design. These are: (1) The reactor cavity arrangement was changed to permit installation of the neutron shield within the reactor vessel rather than external to the vessel. This change simplified significantly the shield cooling system by placing the major heat generation portion of shielding within the vessel where the heat is readily removed by sodium. The resulting heat generation external to the vessel is quite small and easily handled by the gas circulation shield cooling system. The vessel is still surrounded by a concrete biological shield as originally contemplated. (2) Within the cavity region a reinforced concrete energy absorption structure surrounds the reactor. This structure can absorb the radial component of energy which could accompany a nuclear accident. (3) The cell geometry was changed to permit the use of a vertical cover handling system. This change significantly improved access to the reactor and the control drives while simultaneously permitting improvement in the reliability of the cover handling system and the control rod drives.

III. OPERATING AND EXPERIMENTAL CAPABILITY

A. General Features

The fundamental objective of FARET is to provide the capability to perform reactor experiments at operating conditions which can be widely varied to obtain the maximum of experimental data. To achieve this objective the "reactor-in-a-cell" arrangement is utilized (Fig. 1). This arrangement permits access to the interior of the reactor vessel for in-core instrumentation, for in-core re-entrant type loops, and for other experimental devices or components. This feature, coupled with the internal, integral, upper neutron shield provides considerable access to the immediate vicinity of the top of the reactor vessel and permits viewing of the upper regions of the vessel--both direct and remote. These features provide broad experimental capability and flexibility which are extremely important when dealing with experimental equipment and programs that have constantly changing requirements. When the cell is clean, and therefore accessible, the installation of experimental equipment in the reactor cell or vault is accomplished manually. When contamination exists in the cell, remote handling procedures may be utilized. The installation of experimental equipment is facilitated by the minimal equipment shielding requirements permitted by the FARET design. This facilitation is made possible because the containment structure provides the massive shielding normally included in the fuel handling equipment. The result is lighter, more easily controlled machinery.

In order to maintain a clean cell environment, operations in the cell are limited to fuel handling, storage and subassembly cutoff operations. The operations involving handling of spent fuel are designed to limit the inadvertent spread of contaminated material to the cell. Removal or opening of fuel cladding are not intended to be done in the FARET cell.

Locks provide personnel access to the cell when containment may not be breached or when a hostile environment exists therein. The latter could be from either a high contamination level or a low oxygen content in the cell atmosphere. Life-support equipment is provided for cell entrance and controlled occupancy under these conditions.

In the event the cell environment prohibits personnel entry for a significant period of time, the cell design permits the use of remote handling devices to perform experimental operations. Two sets of rails located at elevations 111 ft-4 in. and 124 ft-0 in. are available for remote manipulative devices. One set would be shared with the fuel handling machine, and a spare set of rails is specifically provided for remote handling equipment. In addition, floor or wall mounted manipulators can be added. These remote handling devices are not included in the design of the facility, but provisions have been made to permit the addition of such equipment in the future if required in support of the experimental program.

B. Operating Capability

The operating capability of the principal reactor process systems permits a wide range of reactor conditions to be achieved. At a mixed-mean sodium outlet temperature of 790°F from the reactor vessel, the maximum reactor power capability is 50 MW(t). At the maximum permissible sodium outlet temperature from the vessel of 1150°F, a somewhat greater power capability is available due to the greater temperature differences across the heat exchangers. Stress considerations permitting, power levels of 65-70 MW(t) may be possible. The coolant inlet temperature to the reactor can be controlled over the range from 600 to 960°F. The reactor is designed for a maximum permissible coolant temperature rise across the core of 240°F; local temperature rises in individual sub-assemblies can be larger, if necessary, with coolant outlet temperatures

TABLE 2. Operating Parameters

<u>Power, MW(t)</u>	
@ 790°F outlet temp	up to 50
@ 1150°F outlet temp	up to 70*
<u>Coolant Temperature, °F</u>	
Core inlet	600-960
Core outlet	840-1200
Subassembly outlet	840-1300**
Vessel outlet	790-1150
Fuel handling	250-350
<u>Flow Rate, gpm</u>	
Main primary	750-7500
Core	600-6000
Core bypass	150-1500
<u>Power Density, kW/liter</u>	up to 1700
<u>Specific Power, kW/kg</u>	up to 700

*Reactor coolant system. Stress considerations permitting.

**Dependent upon subassembly design and maintaining a mean 1200°F core outlet.

from a subassembly of 1300°F or greater. At full power and maximum temperature conditions, a mixed-mean core coolant outlet temperature of 1200°F exists and a mixed-mean reactor vessel coolant outlet temperature of 1150°F is maintained by attenuating the core coolant outlet temperature with coolant bypassed around the core.

A maximum coolant flow rate of 7500 gpm is provided at core pressure drops of up to 85 psi. At maximum core coolant outlet conditions of 1200°F, 6000 gpm flows through the core and 1500 gpm is bypassed to the core outlet mixing chamber. The coolant flow rate can be controlled over the range from 10 to 100 percent for cores with pressure drops ranging from 40 to 80 psi at rated flow.

A summary of principal FARET operating parameters are given in Table 2.

C. Experimental Capability

A primary design objective has been to provide a capability to perform a wide variety of experiments. The experimental requirements may be unclear yet, or even possibly unknown. To meet this situation, the vessel arrangement readily permits assembly of a wide variety of core configurations. Within the subassembly configuration, various fuel element or other

test component arrangements can be utilized. Coolant flow rate control to a subassembly is provided by fixed orifices in the subassembly lower end fitting, thereby permitting a predetermined flow rate in each subassembly. A twelve control rod arrangement is provided initially. Different arrangements of control rods and drives can be used, if desired, with modification of the reactor vessel cover.

1. Reactor

The characteristics of the FARET systems permit considerable latitude in the selection of core configurations and fuel element geometries. The reactor can accommodate core arrangements up to an assembly of 301 subassemblies. With the exception of rows 8 and 9, each subassembly location is a potential test position since it has the same flow capability, i.e., each subassembly has access to the high pressure or low pressure plenums. Even rows 8 and 9 which are only somewhat limited by grid plate tie tubes have a high degree of flexibility. Cores as large as 500 liters can be accommodated with core heights up to 3 ft. The length of fuel elements in an individual subassembly is limited only by the test section length of the subassembly. Uniform or composite core arrangements can be installed.

An arrangement of a composite or zoned core is illustrated in Fig. 2. The core has a relatively small, central test zone surrounded by a buffer (filter) zone, a driver zone, and a reflector. This core is typical of the arrangement which could be used for measurement of the following parameters:

- (a) Doppler coefficient over a fuel temperature range typical of large power reactors and as a function of sodium temperature.
- (b) Spectral component of sodium temperature coefficient for large power reactors as a function of sodium and/or fuel temperature.
- (c) Time constants associated with the above reactivity coefficients for the power reactor mocked up in the test zone.
- (d) Integral reaction rates, such as fission/capture ratios, threshold reactions, etc., in a neutron spectrum typical of a large class of fast power breeders.

The subassemblies are fueled with $(\text{Pu}, \text{U})\text{O}_2$, while those in the buffer zone contain depleted UO_2 . The buffer zone serves to equilibrate the neutron spectrum in the test zone to that of a corresponding zone in a large, dilute system of interest. The driver zone is composed of metallic uranium alloy.

Typical parameters for small, uniformly loaded cores containing 37 or 61 subassemblies are given in Table 3. These values show that a FARET core can accommodate fuel elements operating at conditions similar to those chosen in the recent 1000 MW(e) fast reactor studies as illustrated

in Table 4. The approximate experimental range indicated shows that wide variations from nominal conditions are possible for specific fuel element designs.

TABLE 3. Core Parameters for 50 MW(t)
(Fuel-Oxide, Carbide and Metal)

	37 Subassemblies	61 Subassemblies	Approximate Exp Range
Height, in.	14.8	20	up to 78*
Core Diameter, in.	14.8	19	up to 42
Volume, liters	41.7	93	up to 500
Lattice Spacing, in.	2.320	2.320	-
Power Density, kW/liter	1200	500	up to 1700
Flow Rate, gpm/sub	160	100	-
Core Outlet Temp, °F	1100	1100	840-1300**
Reactor Inlet Temp, °F	860	860	600-960
Temperature Rise, °F	240	240	-
Core Pressure Drop, psi	80	40	up to 80
Sodium Velocity, ft/sec	30	19	-

*Individual subassembly. Maximum core height about 36 in.

**The maximum value can be tolerated on a few subassemblies.

TABLE 4. Comparison of Large Fast Reactor Core Parameters

	1000 MW(e) Studies				
	A-C	GE	CE	WE	ANL
Fuel	Oxide	Oxide	Carbide	Carbide	Metal
Active Height, in.	48	24	30	75	36
Fuel Clad OD, in.	0.30	0.25	0.30	0.30	0.21
Clad Thickness, mils	28	15	11	10	18
Power Density (avg), kW/liter	280	365	690	310	455
Power Density (max), kW/liter	430	790	1060	710	705
Linear Power (max), kW/ft	13.1	22.7	43.7	29.1	10
Linear Power (avg), kW/ft	8.6	10.4	28.2	12.0	7
Sodium Velocity (max), ft/sec	24.7	13.6	23.4	31.8	26
Sodium ΔT , °F	250	300	250	220	280

The neutron spectrum can be varied, depending on core size, fuel arrangement, fuel composition, etc. Table 5 provides some nuclear data for the smallest and the largest cores which have been studied to date. The small core consists of a uniform array of 37 stainless steel-clad, plutonium-uranium-carbide-fueled subassemblies similar to the

37 subassembly core described in Table 3. The large core is the same as illustrated in Fig. 2. Neutron fluxes above 1.35 MeV of 1×10^{15} are readily attainable.

TABLE 5. Calculated Nuclear Data for Two Typical Cores

<u>Small Core (41.7 liters)</u>	
Average Fission Density at Core Center, $f/(\text{cm}^3)(\text{sec})$	3.6×10^{13}
Neutron Flux at Core Center, $n/(\text{cm}^2)(\text{sec})$	
Above 1.35 MeV	1.0×10^{15}
Below 1.35 MeV	4.3×10^{15}
Total	5.3×10^{15}
Prompt Neutron Lifetime, sec	6.55×10^{-7}
<u>Large Zoned Core (466 liters)</u>	
Test Zone	
Average Fission Density, $f/(\text{cm}^3)(\text{sec})$	1.8×10^{12}
Neutron Flux, $n/(\text{cm}^2)(\text{sec})$	
Above 1.35 MeV	0.7×10^{14}
Below 1.35 MeV	7.3×10^{14}
Total	8.0×10^{14}
Driver Zone	
Average Fission Density, $f/(\text{cm}^3)(\text{sec})$	1.1×10^{13}
Neutron Flux, $n/(\text{cm}^2)(\text{sec})$	
Above 1.35 MeV	1.0×10^{14}
Below 1.35 MeV	5.3×10^{14}
Total	6.3×10^{14}
Prompt Neutron Lifetime, sec	8.90×10^{-7}

2. Subassemblies

The geometrical configuration of an individual subassembly is illustrated in Fig. 3. Potentially, the entire cross-sectional area inside the hexagonal tube, for a length of about 78 in., is available for fuel elements or components to be tested. (At present the usable length is restricted to about 65 in., in order to permit disassembling the subassembly in the Fuel Cycle Facility.) Coolant enters the lower end of the subassembly and exits via the reduced section above the test section. The upper end of the unit is designed to accommodate an instrument lead connector. Each subassembly is capable of accommodating instrumented test elements, but not all subassemblies will be instrumented simultaneously because the instrument leads can interfere with fuel handling operations.

With hexagonal subassembly tubes, it is convenient to install 7, 19, 37, 61, or 91 cylindrical fuel elements per subassembly. These numbers of elements permit the use of an equilateral triangular lattice which

distributes the subassembly coolant flow quite uniformly among the fuel elements. When these numbers of fuel elements are used, corresponding permissible fuel element diameters are approximately 0.50, 0.37, 0.29, 0.22, and 0.17 in., respectively. The fuel elements are not restricted to cylindrical shapes; any fuel or component geometry can be used that can be accommodated within the hexagonal tube and is compatible with FARET design parameters.

The FARET resembles the EBR-II in geometry of the core and subassemblies. This permits the common utilization of many systems and components in support of EBR-II and FARET. The principal difference between the EBR-II and FARET subassemblies is their overall length. The FARET subassembly is about seven feet longer because each subassembly contains a five foot upper shielding section which, when installed into the reactor, comprises a portion of the vertical shield for the reactor.

The use of common dimensions and geometrical configurations for both FARET and EBR-II subassemblies permits use of the EBR-II inter-building coffin, the Fuel Cycle Facility (FCF) disassembly equipment and other components. This permits ready transfer of subassemblies to the Fuel Cycle Facility or the EBR-II and the disassembly of these units with existing equipment. Fuels irradiated in the FARET reactor can be more readily tested in EBR-II and handled in the FCF because of the equivalence of subassembly designs. For example, an EBR-II Mark II experimental irradiation subassembly has a usable test section length of about 60 in. FARET fuel can readily be accommodated. In addition, the development of the FARET fuel element and subassembly designs is simplified since the extensive information already developed for EBR-II can be used. Moreover, procurement of component hardware common to both reactors is simplified.

3. In-core Instrumentation

One of the more important capabilities of the facility is the degree of freedom afforded for access to the core for instrumentation of in-core components. The reactor vessel extension contains penetrations for the leads from all in-core devices, such as, instrumented subassemblies and re-entrant loops. Penetrations in the cell walls and floors permit the leads to be brought through the containment structure and space is provided for cable trays and terminal boards for up to 1200 pairs of thermocouple leads, 200 coaxial cable leads, and 300 shielded copper conductors. In addition a 15-ft long panel board space is provided in the control room for installing instruments to measure, record, control, etc., the signals. The most important link in this chain is the connection of the lead to the subassembly within the vessel, accomplished by a mechanically coupled connector under development when the project was terminated. This is recognized as a difficult development since the connector must operate in a 1200°F sodium environment. A welded connector was utilized within the vessel as a backup design.

Since it was considered desirable to monitor temperatures, fission gas release, and coolant flow rate in the test subassemblies, instruments for these purposes were under development. While this is a difficult and challenging area of research, the results at the time of project termination were encouraging. Also, developments in other fields and other programs were expected to contribute to the solution of these problems. Each successive achievement was expected to further increase the usefulness of the facility.

4. Reactor Vessel

The reactor vessel is designed to extend the ultimate usefulness of the facility as much as feasible. The reactor vessel is provided with a separate, replaceable extension. This extension permits access to the vessel for in-core instrumentation to be varied or to replace the upper flange of the extension should it become damaged during operations. The vessel cover, separate from the vessel, can be replaced if a different arrangement of control rods and drive mechanisms is required. In addition, it is also possible to replace the internals of the reactor vessel. This permits variation in lattice dimension, subassembly size and arrangement, flow arrangement and even neutron shield design.

D. In-reactor Loops

Certain types of irradiation experiments require isolation from the reactor and primary coolant system. Included are experiments which require different operating parameters than exist in the reactor and experiments which present some risk to the reactor. Isolation from the reactor can be achieved with closed, in reactor experimental loops. Although in-reactor loops are not included in the facility design space provisions have been made to accommodate as many as three such loops in FARET simultaneously. It was planned to have at least one loop installed in the reflector region before or during the operation of the first core.

Experimental in-reactor loops would permit irradiation experiments involving fuel elements and structural materials which require individual control of environmental parameters such as temperature, power density, flow, and coolant chemistry and where the risk of fuel failure may be higher than thought permissible in the reactor. Fast reactor safety experiments, such as fuel meltdown tests, also could be conducted in such loops.

1. Description of Loop Concepts

a. In-pile Thimbles

Two general types of experimental in-reactor closed loops have been evaluated for potential use in FARET. They have been designated as small loops and large loops. The in-reactor thimbles (each is a small loop) would occupy the space of one subassembly in the FARET core and/or

reflector. The large loop thimble would occupy the space of more than one subassembly. Typical locations of FARET loops in a 61-subassembly core are shown in Fig. 4. Cross-sectional views of typical test fuel subassemblies which could be tested in the small or large loops are shown in Fig. 5.

A conceptual design of the re-entrant type in-pile thimble in the FARET reactor is shown in Fig. 6. Sodium from the external loop system enters and leaves the reactor vessel through a thermally sleeved nozzle flange in the reactor vessel extension. Sodium flow is down through the annulus between inner and outer tubes and up through the inner tube which contains both the in-core test section and an out-of-core test section for control specimens. The lower portion of the inner tube has a gas-gap insulator to isolate the in-core test section and to maintain the fuel-exit sodium temperature through the out-of-core test section. Dilution of exit sodium by inlet sodium flowing through the bypass ports near the top of the inner tube can be adjusted for each experiment by a fixed valve sleeve on the upper shield of the test fuel subassembly. High pressure flexible metal hose on the inlet and outlet coolant lines to the loop provide for differential expansion between the loop thimble and the core support structure. Conoseal or DSD couplings on each end of a flexible metal hose and on the top cap of the thimble permit installation and removal of experiments, hoses, and the in-pile thimble. Each loop test subassembly is instrumented. Instrument leads are brought into the reactor vessel through seals on the same nozzle flange used for the sodium lines. Flexible instrument leads pass through a detachable seal in the top of the in-pile thimble to a sealed connector on top of the test fuel subassembly.

The potential operating parameters of temperature, power density, and neutron flux are illustrated in Table 6. Higher test fuel exit sodium temperatures (up to $\sim 1500^{\circ}\text{F}$) can be achieved by bypassing part of the inlet sodium at the top of the in-pile thimble directly to the exit. A regenerative heat exchanger is incorporated in the upper portion of the in-pile thimble. The heat exchanger also aids in reducing the sodium exit temperature below 1200°F .

b. External Loop System

A schematic diagram of an in-reactor loop system is shown in Fig. 6. The particulate filter in the sodium outlet line from the in-pile thimble is designed to trap fuel particles in the event of a fuel failure or deliberate meltdown. Not shown is a shock tank, connected to the outlet sodium line in the reactor cell, which would absorb the pressure surge during a rapid fuel meltdown experiment. The heat exchanger utilizes the reactor main secondary sodium system for cooling.

A separate purification system would normally be provided for each in-reactor loop. This system includes a hot trap to obtain a low oxygen concentration in the loop system sodium if required.

TABLE 6. Preliminary Design Criteria: FARET In-reactor Loops
(at 50 MW(t) Reactor Power)

Parameter	Small Loop Core Center	Small Loop Reflector	Large Loop Reflector
Loop power (2 ft high fuel), MW	1.0	1.0	3.0
Temperature, °F	1200	1200	1200
Pressure, psig	300	300	300
Total Dynamic Head, psig	200	200	200
Na Flow Rate, gpm	100	100	250
Max ΔT across Test Assembly, °F	360	360	360
Power Density in Test Fuel (2 ft high)			
MW/l	1.5	1.5	1.5
kW/ft of pin length	40	40	40
<u>Total Neutron Flux, n/cm²-sec</u>			
<u>5 Ring Core (61 Assembly, 20-in. high)</u>			
Loop Location, Radius cm	0	25.4	35.4
Ring No.	1	6	8
Max flux	3.0×10^{15}	1.6×10^{15}	1.07×10^{15}
Avg flux	2.6×10^{15}	1.4×10^{15}	0.86×10^{15}
<u>4 Ring Core (37 Assembly, 24-in. high)</u>			
Loop Location, Radius cm	0	21.0	25.4
Ring No.	1	5	6
Max flux	3.6×10^{15}	2.2×10^{15}	1.7×10^{15}
Avg flux	3.0×10^{15}	1.9×10^{15}	1.5×10^{15}

The drain tank for each loop also serves as a storage tank and sump for the pump. Pump head for the multistage, centrifugal pump in both large and small loops is estimated to be 200 psi. The variable flow pump is manually or automatically controlled to achieve desired loop temperature conditions. Shutdown cooling of the loop test fuel is accomplished by a small electromagnetic auxiliary coolant pump in each loop which operates continuously at 10% of design flow. Emergency power is furnished to these pumps from the diesel-generator system and from a separate emergency battery which maintains adequate pump flow for one-half hour.

A sodium fill system, blanket gas supply and vent system, and a sodium sampling system are also provided for each loop.

Both primary and secondary electrically-heated sodium systems are located in the lower level at the north end of the vault and are thus within the reactor containment structure. Pump motors, batteries, rectifiers, and all reach-rod valve operators for primary and secondary system bellows-seal sodium valves are located in the lower level of a loop equipment compartment to the north of the vault outside the containment. In the upper level of this compartment is the control room for each loop.

Loop instrumentation would measure such parameters as flow, sodium and pipe temperatures, drain tank level, loop pressure, sodium activation, and fission products in the sodium. In addition to conventional indicators and multichart recorders, a small data logging system was considered for loop data handling. Certain parameters such as high sodium temperature and low flow in the loop would be interlocked with the reactor scram system.

2. In-reactor Loop Experiments

A variety of experiments were identified which might be conducted in experimental in-reactor loops in FARET. These may be broadly classified as (1) irradiation experiments and (2) fast reactor safety experiments. A tabulation of possible experiments follows:

a. Irradiation Experiments

1) Irradiation of advanced fast reactor fuel elements at intermediate and high burnup rates (5 to 10 a/o per year), at high surface temperatures (650 to 800°C), at high power densities (up to 1.5 MW/l), and at a high risk of failure. (Loop only will be contaminated if failure occurs.)

2) Irradiation of fast reactor fuel cladding and structural materials at high temperature and to high integrated neutron fluxes for non-fissile irradiation damage studies (where control of operating parameters--such as sodium quality or temperature--different from those in the reactor is required).

3) Irradiation of vented fuel concepts to investigate coolant and loop contamination.

4) In-reactor corrosion experiments in which the sodium chemistry is controlled.

5) In-reactor measurement of the mechanical properties of irradiated cladding and structural materials in a controlled environment.

b. Safety Experiments

1) Conduct slow and rapid meltdown experiments on pre-irradiated fuel by gradually reducing or suddenly stopping loop coolant flow.

Attempt to measure reactivity feedback. Study resulting fuel configuration, loop contamination, and fuel transport in coolant.

2) Conduct fuel meltdown experiments on clusters of fuel elements to study fuel failure propagation.

3) Develop and test instruments to detect the onset of sodium boiling and the inception of fuel failure.

In addition to the above proposed experiments, possible future loop experiments in FARET might include testing of coolants other than sodium, the testing of direct conversion fuel elements, etc.

IV. CORE I DESIGN

A. General

The planned initial use of FARET was to develop the maximum possible information relevant to the fuel loading for the Prototype Reactor in the AEC's Fast Reactor Development Program. It was intended that core and fuel specifications be developed for the FARET Core I before those for the Prototype Reactor were established. Therefore, the Core I design, fuel element specifications, and reactor operating conditions were planned to "bracket" the anticipated design, specifications, and conditions for the Prototype Reactor to the extent necessary for FARET design purposes. Core I was to test several different fuel element designs, in order to provide information to evaluate, define, or verify operating limits for the first core of the prototype. It was also to yield information related to the choice of later cores for the prototype.

B. Core I Description

Numerous fuels were expected to be tested in Core I under operating conditions appropriate to the Prototype Reactor. By selection of reactor position and/or enrichment, environments could be achieved which meet the desired conditions and permit assessment of fuel operating capability.

The Core I design consists of carbide, metal, and oxide fuels in sub-assemblies driving each other and can test simultaneously ten different fuel-element designs with good statistics. The fuel-element types, and designs, and the tentative conditions of operation are presented in Fig. 7. The elements are typed according to fuel material, fabrication method, cladding material, and/or thermal conditions.

The assumptions used in the thermal calculations, basic to the core design and thermal analysis, are shown in Table 7. The calculated parameters for Core I are listed in Table 8.

TABLE 7. Assumptions for Thermal Calculations
(Based on a sodium film coefficient of $20,000 \text{ B hr}^{-1} \text{ ft}^{-2} \text{ }^{\circ}\text{F}^{-1}$)

Fuel Material	Clad Thermal Conductivity, $\text{B hr}^{-1} \text{ ft}^{-1} \text{ }^{\circ}\text{F}^{-1}$	Fuel-clad Interface Conductance, $\text{B hr}^{-1} \text{ ft}^{-2} \text{ }^{\circ}\text{F}^{-1}$	Fuel Thermal Conductivity, $\text{B hr}^{-1} \text{ ft}^{-1} \text{ }^{\circ}\text{F}^{-1}$
Oxide	13.5	1,500	1.7 ^a
Carbide	13.5	1,500	.10
Metal	14.5	40,000 ^b	9 ^b

^aThe fuel is assumed to have a central hole large enough to produce an "effective" fuel density of 85%, provided that the remainder of the jacket interior is filled with fuel of 100% density. This condition reduces the temperature rise through the fuel to 67% of that through a solid fuel pellet. The central hole is expected to form in oxide at centerline temperatures above 1600-1800°C.

^bThe fuel is considered to have expanded against the jacket, displacing the bond sodium, and to have an accompanying reduction in thermal conductivity.

TABLE 8. Calculated Parameters for Core I

Nuclear Data			
Neutron Flux, $n/cm^2 \cdot sec$			
Energy Range	Radial Position		
	Core Center	Core Edge Loop	Reflector Loop
>1.35 MeV	0.8×10^{15}	0.3×10^{15}	0.3×10^{14}
100 keV - 1.35 MeV	2.4×10^{15}	1.5×10^{15}	5.7×10^{14}
1 - 100 keV	0.4×10^{15}	0.5×10^{15}	3.9×10^{14}
<1 keV	6.5×10^{11}	1.8×10^{13}	1.2×10^{14}
Total	3.6×10^{15}	2.3×10^{15}	1.1×10^{15}
Prompt neutron lifetime, μsec			0.21
Effective delayed neutron fraction, β_{eff}			0.0058
Reactor Control Average Reactivity Coefficients, $\Delta k/\Delta T^\circ C$			
Overall Doppler @750°K Uranium			-0.45×10^{-6}
Plutonium			$<+0.1 \times 10^{-6}$
Radial core expansion			-10×10^{-6}
Overall axial fuel expansion			-4×10^{-6}
Sodium expansion coefficient			
Overall core region avg			-10×10^{-6}
Core and reflector			-14×10^{-6}
Estimated Range of Total Worth of Control Rods			
Composition (v/o): B_4C 55/coolant 25/structure 20			
No. of rods			12
70% B^{10} enrichment			0.075 - 0.085
95% B^{10} enrichment			0.085 - 0.10
Excess and Shutdown Reactivity			
Heavy atom burnup			0.035 - 0.060*
Operational temperature			
Sodium $\Delta T_{avg} = 500^\circ C$ core and reflector			0.007
Radial core expansion $\Delta T_{avg} = 500^\circ C$			0.005
Axial fuel expansion $\Delta T_{avg} = 1500^\circ C$ oxide			0.004
$\Delta T_{avg} = 1100^\circ C$ carbide			
$\Delta T_{avg} = 500^\circ C$ metal			
Total operational temperature			0.016
Excess reactivity			0.051 - 0.076
Shutdown reactivity			0.020
Design worth of control system			0.071 - 0.096

*Depending on manner of compensating for burnup.

The core configuration is shown in Fig. 7. Several conditions were considered in the positioning of the subassemblies. Symmetry was maintained as closely as possible to simplify analysis and to subject the several elements of each type to similar power conditions throughout the experiment. Also, to achieve the linear heat fluxes, the uranium enrichment and positions in the configuration were varied.

The positioning was also influenced by the sequence of removal and replacement. Instrumented subassemblies are usually located so that the loading and unloading of other subassemblies is not made cumbersome by having instrumentation leads located over the subassemblies to be removed. It is necessary to remove the instrumented subassemblies in the latter stages of burnup for each type element.

The radial power distribution within a subassembly varies about 10% from the average (see Fig. 8). With the axial maximum to average power ratio of 1.16, the overall maximum to average power density is about 1.28. This factor was used in the approach to the design heat fluxes listed in Fig. 7. The uranium enrichments planned include the effect of control rods on the power density distribution. Partially inserted control rods could reduce the average power distribution in the outer rows by 10%. Therefore, the subassemblies in the outer row, when all factors are applied, approach the design heat flux.

Computations, along with related ZPR-3 critical experiments for FARET, indicate that a 10% reactivity-control system may be attained with rods employing highly enriched boron carbide. With this margin of control along with 2% shutdown margin and 2% margin for the operational swing, a 6% excess reactivity can be used for burnup.

The percent atom burnup as a function of full-power operating days is given in Fig. 8. The estimated percent atom burnup as a function of time is plotted for the twelve types of fuel elements. The types of subassemblies to be removed after each irradiation period are encircled. There are only two metal M2 assemblies in Row 4, so that after 9 a/o burnup, the burnup experiment for this type of fuel is to be followed with the M2 assemblies in Row 5. After 780 days of full-power operation, all but three types will have exceeded 14% burnup.

At the end of a 430-day operating period at full power (1.5 years, assuming a plant factor of 0.75), the performance of close to 1000 fuel elements (22 subassemblies) would have been investigated at advanced thermal and burnup conditions.

In order to go to burnups of 15 a/o, the necessary reactivity could be obtained by the replacement of some partly burned-up subassemblies with new subassemblies of the same design or of a new design. Alternatively,

a few additional subassemblies could be added at the core edge. The sequence of removal and the replacing of subassemblies could continue until all types, or all those that have performed satisfactorily throughout the course of the experiment, reach a burnup of 15 heavy atom percent.

The downtime due to removal and replacement of the subassemblies can be maintained at a minimum, depending on the degree of confidence obtained from previous research and development. Inspection of removed subassemblies gives further confidence of satisfactory performance. The estimate of the time required for the usual loading and unloading operation is two weeks. Using an average plant factor of 75% over the duration of this experiment, the time to complete the program as outlined is three years.

C. Future Programs

Fuel for the first loading of the Prototype Reactor was expected to be specified in time to permit fabrication and testing in a uniform loading for FARET Core II. Operation of Core II would thus be a final and detailed proof test of the prototype fuel, verifying the adequacy of the specifications and firmly establishing the fuel operating limits. As performance capabilities of the fuel were more firmly established, safety and performance limit tests were to be performed either in Core II or subsequent cores.

Later cores would have tested improved fuel elements which evolved from experience gained with the earlier cores. It is likely that these later cores would be smaller, to emphasize high rates of burnup, high flux, and high specific performance parameters. Special fuel design features to enhance safety, such as controlled expansion elements, could also have been tested. In the later stages of fuel development, it was anticipated that fuel elements would be vented directly to the reactor coolant, to gain experience with the problems involved in operating a large system in that manner. Safety experiments were expected to be interspersed between various irradiation runs.

V. FACILITY AND SITE

A. General

The FARET site is located in the East Area of the National Reactor Testing Station (NRTS) as shown in Fig. 9. The station is about 30 miles due west of Idaho Falls, Idaho. The East Area includes the EBR-II and TREAT reactors and the Fuel Cycle Facility. It is about 11 miles west of the NRTS east boundary and 4 miles north of U. S. Highway 20. The FARET site is about 4,000 ft southeast of the existing EBR-II Facility. A new asphaltic concrete roadway, about 3,400 ft long, connects the existing Taylor Boulevard to the west with the FARET site. Taylor Boulevard connects with U. S. Highway 20 between Idaho Falls and Arco, Idaho.

B. Site

The FARET Facility site (Fig. 10) covers about $2\frac{1}{2}$ acres. The site is a plains area generally level at elevation 5170 ft and covered with sparse sagebrush. A thin layer of silty loam and silt 2-10 ft thick covers the lava rock. The water table is approximately 650 ft below grade. Climate varies from heavy snow with low temperature in winter to hot and dry in summer.

On-site roads consist of a service road along the south and east side of the building area with an access driveway to the rear cell area truck doors in the east wall of the reactor building. To the south of the support wing is a parking area for cars and space for a bus terminal. General personnel access to the reactor area is through the support wing.

The yard area has a lighting system and includes the main sodium-air heat exchanger, auxiliary Na-air heat exchanger, NaK-air heat exchanger for the wet fuel storage tank, water cooling unit, inert gas supply tank, reactor building exhaust stack, well and pump, substation, septic tank and leaching field, boiler fuel oil storage tank, and diesel fuel tank.

C. Reactor Building

1. General Arrangement

The reactor building is a conventional structural steel frame building with concrete substructure. It provides a full basement 25 ft below the operating floor for most of the area. The foundation is on lava rock. Overall, the building is 87 ft wide by 172 ft long; Figs. 11 through 16 show its general arrangement.

The superstructure width includes two bays: (a) the high or rear cell bay, 60 ft wide and 84 ft high, which is serviced with a 40/10 ton crane, and (b) the low or front cell bay, 24 ft wide by 44 ft high, which has a full length second floor. The high bay has a second floor at the north end.

Masonry and insulated metal panel walls, steel roof deck and built-up roofing on rigid insulation enclose the building. The design maintains the slightly negative pressure required of the ventilation system. Bituminous waterproofing protects the substructure from moisture. Structural excavation, foundation preparation and basement details include provisions for future extension of the cell in a southerly direction within the limits of the reactor building.

2. Building Contents

The reactor building houses the reactor, radioactive and non-radioactive reactor cooling systems, the cell, vault, and reactor cavity with their individual gas cooling systems, fuel transfer facilities and auxiliary systems. The cell, vault and cavity provide containment, controlled inert argon or air atmospheres and biological shielding for the reactor and all radioactive systems; including primary cooling, fuel handling, fuel cut-off and new and spent fuel storage.

The principal auxiliary systems in the reactor building are the radioactive vent and waste, inert gas purification and handling, cooling water, radiation monitoring, instrument air, normal electrical and emergency electrical power, and the reactor building ventilation systems.

3. Main Operational Features and Areas

A shielded containment cell in the high bay area contains a controlled inert atmosphere for operations related to the reactor, refueling, or fuel cut-off. The cell provides shielding, containment, and environmental control. Shielding windows in the cell wall permit observation of "in-cell" operations from the operating areas. At the north end of the cell, the ceiling is raised to form a cell annex. The reactor cover and the control rod drives can be lifted into the annex for fuel transfer and to provide accessibility to this equipment for maintenance. Access to the reactor head in place can be gained from the annex through a bellows which seals to the reactor head. The annex has the same shielding and containment rating as the cell below. The design of the facility is such that the process flow and equipment arrangement of these systems radiate outward in different directions from the reactor. The reactor coolant systems are located in the vault to the east of the reactor, the systems for handling fuel and scrap are located in the cell to the south, and the control mechanisms are fixed to the reactor vessel cover with all normal movements being in a vertical direction. Thus, the cell annex over the reactor vessel is devoted to cover handling and control drive mechanism operations; the cell area to the south of the reactor to fuel and scrap handling; and the vault to reactor cooling.

Two personnel air locks, one at each end of the cell allow access from the rear cell area operating floor. Fuel and equipment transfer can be

made through the concentric fuel and equipment transfer ports in the cell floor to a tunnel below. Other equipment too large to be carried through the air locks or through the transfer ports can be passed through the large normally closed rectangular opening in the south end of the cell roof.

The reactor vessel and the wet fuel storage tank are recessed into the concrete substructure in steel lined cavities which also provide shielding for adjacent basement areas and containment. No access is provided to these cavities. Thimbles for nuclear instrumentation extend from near the reactor cavity to the basement beneath the front cell area. The thimbles are located in the concrete biological shield and external to the cavity liner.

The primary sodium systems and the experimental in-core sodium loops are installed in a shielded and steel lined vault in the basement under the rear cell area. Personnel and equipment access to the vault is through the various shield plugs in the ceiling or the north vault wall at elevation 78 ft. The vault is part of the containment system and is designed for intermittent personnel occupancy.

The operating floor, excluding that occupied by the cell, is divided into two areas: the front cell area and the rear cell area. The front cell area includes space for instrumentation for in-core experiments, room for control of personnel access to the rear cell area, space for electrical gear, and the emergency diesel at the north end. The rear cell area includes operating space adjacent to the cell and above the vault. Principal access to the cell and vault and access to the new fuel storage cells embedded in the vault wall and the fuel transfer tunnel are from the rear cell area. Space is also included for equipment laydown and for truck loading or unloading in this area. The secondary sodium system pump and main sodium piping are on the operating floor along the east wall. The refrigeration equipment is located at the north end of the operating floor.

The basement north of the vault in the rear cell area contains the argon purification equipment, radioactive vent and liquid waste system and experimenters' space. The basement south of the vault in the rear cell area contains the secondary sodium auxiliary systems. The fuel transfer tunnel extends south from the cell in the basement to the outside wall of the building. Access to the fuel transfer tunnel for cask removal is through a hatch in the operating floor in the rear cell area. The basement in the front cell area contains the battery room, instrument air compressors, cooling water pumps, the biological shield cooling system, the continuous power motor-generator set, and an instrument thimble access area.

The second floor in the front cell area serves as space for reactor building ventilation air intake equipment and for electrical gear. The north end of the second floor rear cell area provides space for building air exhaust equipment.

Normal access to the reactor building is through the support wing for all personnel. Truck doors in the east wall provide for bringing materials and equipment to the rear cell area; a restricted access door in the partition wall permits equipment and material access to the front cell area. Personnel doors on the north, east, south, and west wall permit emergency exit from the reactor building to the yard area. A door, normally kept locked, provides an emergency exit from the operating area under the front cell area to the basement of the support wing.

D. Support Wing

The support wing is a conventional single-story and full-basement building extending west from the reactor building, as shown in Figs. 12, 13 and 15. The building area is about 144 ft long by 50 ft wide. The main floor, corresponding to the reactor building operating floor, is 13 ft high and the basement floor is $14\frac{1}{2}$ ft below the main floor. The boiler room floor at the west end is lower to accommodate the boiler headroom requirements. The superstructure is masonry block with long span decking and rigid roof insulation and built-up roofing. The basement is of reinforced concrete.

The support wing includes the control room, offices, health physics room, file room, conference room, lunch room, janitor's closet, and toilets on the main floor. In the basement are the cable spreading room under the control room, a locker, shower and men's toilet area, instrument shop, storage room, boiler room, janitor's closet, service equipment for the support wing, and an area for general plant service. Support wing finish is minimum. No ceilings are provided except for the luminous ceiling in the control room.

E. Utilities

The primary water supply is a 4000 ft long 8 in. buried pipeline from EBR-II which supplies domestic water, service water makeup, and fire protection water. For reliability, the existing FARET site test well will be used for backup. It is 675 ft deep with a 10 in. casing and is fitted with an existing test pump available at EBR-II, having a capacity of 150 gpm at 745 ft head. The pump is provided with about 40 ft of additional column sections to adjust for the difference in well depth. This pump has an electric motor and gasoline driven auxiliary. A pump house is provided for the pump and drivers. No additional treatment is required for the water supply.

The domestic water system which supplies the toilets and the washrooms is pressurized by the EBR-II system. An electric heater supplies hot water for domestic use. Electric drinking water coolers are located on each floor and the basement of the support wing.

Drains other than sanitary in the access control room are connected to the radioactive waste system. The sanitary drains are connected to the

building sanitary drain system. Sanitary wastes from the support wing flow by gravity through a septic tank to a leaching field just north of the site. The effluent of the septic tank is chlorinated. Roof drainage is piped to the site surface drainage system. Yard drainage is handled in ditches with a minimum of culverts where required to supplement surface drainage from the graded site area.

F. Containment

1. Structure

The containment complex is comprised of the cell, annex, vault and reactor vessel cavity. Normally, the cell atmosphere is isolated from the annex cavity-vault atmosphere by providing light gage metal or fabric bellows seals at penetrations communicating between these areas. In the event the pressure differential across these seals is of the order of 2 psi, the seals will rupture and containment atmosphere will be common to the connected regions.

The inner liner which encloses the entire containment volume is a continuous welded steel plate, and is structurally backed up with two different types of construction--reinforced concrete for the cavity and vault, and steel plate and girder structure for the cell and annex. The basic design stresses for both types of structures are 80% of those specified in the appropriate codes.

In the reactor vessel cavity and the equipment vault the steel liner plates function only as membrane barriers to the flow of gas into and out of the volumes enclosed, both during normal operation and during postulated accident conditions. In the cavity and vault the structural resistance for the 30 psig internal pressure (and the loadings) is provided by the reinforced concrete floors, walls and ceilings. No credit is taken for the steel liner. Since the liner plate is in intimate contact with the concrete the strains in the liner will be substantially equal to those in the concrete, thus resulting generally in low stresses for the liner plate.

The steel liner in the cell and annex, however, provides the structural resistance to internal pressure as well as providing the leaktight membrane. The cell structure is formed by the inner steel plate, a continuous steel liner plate on the outside of the concrete and a series of vertical and horizontal webs of steel plate connecting these two liners. This steel box-girder structure provides the necessary strength of the 30 psig internal pressure; the concrete acts only as biological shielding. An exception to this is at the juncture between the cell roof and cell annex, where the concrete and the steel plates act as a composite beam framing the floor of the cell annex.

The values of allowable stresses for the various structural components for the several combined loadings is shown in Table 9. The values shown for the steel liners apply where the liner is part of the structural section (cell and cell annex only) or acting only as the leaktight membrane.

TABLE 9. Allowable Stresses for Containment Structure for Various Loading Conditions

Loading Condition	Allowable Stresses (psi)		
	Steel Liner Plates (Tension or Compression) ** $f_y = 36,000$ psi	Steel Reinforcing Bars (Tension) ** $f_y = 40,000$ psi	Concrete* (Compression) ** $f'_c = 3,000$ psi
(1)	$f_s = 0.8 \times 0.6 f_y$	$f_s = 0.8 \times 0.5 f_y$	$f_c = 0.8 \times 0.45 f'_c$
Normal	$= 0.48 f_y$	$= 0.4 f_y$	$= 0.36 f'_c$
Operating	$= 17,600$	$= 16,000$	$= 1,080$
(2)	$f_s = 0.8 \times 0.6 f_y$	$f_s = 0.8 \times 0.5 f_y$	$f_c = 0.8 \times 0.45 f'_c$
30 psig Internal	$= 0.48 f_y$	$= 0.4 f_y$	$= 0.36 f'_c$
Pressure	$= 17,600$	$= 16,000$	$= 1,080$
(3)	$f_s = 1.33 \times 0.8 \times 0.6 f_y$	$f_s = 1.33 \times 0.8 \times 0.5 f_y$	$f_c = 1.33 \times 0.8 \times 0.45 f'_c$
Accident Conditions	$= 0.64 f_y$	$= 0.53 f_y$	$= 0.48 f'_c$
(30 psig and Temp) plus Seismic	$= 23,500$	$= 21,300$	$= 1,440$

*Allowable stresses for shear, bond, etc., are 80% of the values specified in the Building Code Requirements for Reinforced Concrete (ACI 316-63) for loading conditions (1) and (2), and 1.33 times 80% of these values for loading condition (3).

** f_y = yield strength for steel.

f'_c = ultimate compressive strength (at 28 days) for concrete.

f_s = allowable stress (tensile or compressive) for steel plate or reinforcing bars.

f_c = allowable compressive stress for concrete.

In addition to the combined loadings shown in Table 9 the cell structure is designed to absorb the effects of the most energetic missile possible. It is not convenient to show, in the table, the limiting factors and criteria used in this case, since the energy absorption capability is determined by dynamic analysis of the structural response to this impact loading. The criteria used for the impact loading of the cell annex ceiling were:

- (1) Impact Location: Anywhere on the underside of the cell annex roof in a rectangular area bounded by lines located 2.5 ft inside of and parallel to the inner faces of the cell annex walls.
- (2) Missile Weight: 100,000 lb.

- (3) Impact Energy: The kinetic energy of the impact load, as it strikes the underside of the cell annex roof structure, is 0.83×10^6 ft-lb.
- (4) Loaded Area: Uniformly distributed over a circular area 2 ft in diameter.
- (5) Design Loading Conditions and Bases:
 - (a) Steady internal pressure of 30 psig in cell; design on elastic basis using 80% of normal allowable stresses.
 - (b) Impact load (50-ton missile) concurrent with 30-psig pressure; design on plastic basis. Plastic action of the cell annex structure shall be limited to that portion of the cell annex structure above a line 3 ft above the top of the cell annex access door. Welds subjected to bending or shearing stresses shall be located away from locations of plastic hinges, wherever possible.
 - (c) Cell structure to be capable of sustaining a steady-state pressure of 30 psig after impact, with no further plastic deformation.

Similar criteria have been applied to the design of the cell floor.

2. Radial Energy Absorbing Shield

The FARET containment complex is designed to satisfy three major requirements. First, it is designed for a static internal pressure of 30 psig which is intended to provide for the effects of a sodium-air reaction caused by the ejection of sodium into the cell when an air atmosphere exists. Secondly, the entire inner surface of the containment is lined with a continuous steel plate membrane to prevent leakage of significant amounts of fission products into the building from the cell, cavity or vault. The third requirement is the provision of sufficient energy absorption in the containment structure surrounding the reactor cavity to withstand the dynamic effects of an energy release at the center of the core equivalent to a bare charge of 40-lbs of TNT.

Normally, the cell atmosphere is isolated from the annex and cavity-vault atmospheres by light gage metal or fabric bellows seals at penetrations which communicate between these volumes. If the pressure differentials across these seals exceed 2 psi, the seals will rupture and containment atmosphere will be common to all the enclosed volumes--cell, annex, vault and reactor cavity.

An all-welded steel plate is used for the inner liner, which is continuous throughout the containment complex. This inner liner plate is

structurally backed up with two different types of construction--reinforced concrete for the cavity and vault, and steel plate and girder structure for the cell (and annex).

In the reactor cavity and vault the steel liner plates function only as membrane barriers to the flow of gas into and out of the volumes enclosed; both during normal operation and during postulated accident conditions. All structural design resistance for the 30 psig internal pressure (and other loadings) is provided by the reinforced concrete floors, walls and ceilings. No design credit is taken for the steel liner.

In the cell, however, the steel liner participates in providing the structural resistance to pressure loads as well as providing the leak-tight membrane. The cell structure consists of the inner steel plate liner, a continuous steel liner plate at the outer surface of the cell, and a series of steel web plates welded to each of these liners; thus forming a continuous array of structural steel frames. This steel box girder structure provides the necessary strength for the 30 psig internal pressure; the concrete acts only as biological shielding. The values of allowable stresses for the various structural components for the several combined loadings are shown in Table 9. The values shown for the steel liners apply whether the liner is part of the structural section (cell and cell annex only) or acts only as the leaktight membrane.

Missile protection for the inner steel liner plate which is the containment envelope is provided in several ways. The wall liner of the cavity is protected by a radial energy absorber shield of concrete which absorbs a significant amount of the blast energy by the stretching of steel reinforcing hoops placed around the concrete cylinder. The energy absorbing shield can move outward radially under the design load, but should not impinge on the cavity liner. The part of the containment liner below the reactor vessel is protected by an overlay of steel faced concrete.

The radial energy absorbing shield is an annular structure which surrounds the reactor cavity (Fig. 16). The shield consists of closely spaced steel bar hoops wrapped around a concrete cylinder in the configuration of an open-ended basket. Thus, the array of bars and the steel shell enclosing the concrete cylinder form a pseudopressure vessel which is open at the top and bottom. The radial blast energy is then absorbed by significant elasto-plastic deformation of these bars as the "pressure vessel" expands under the influence of the blast forces. The concrete serves as a piston system pushing radially outward on the steel bars.

In the radial direction the shield has a thickness of 8 in. of concrete (including an inner steel liner plate of 5/16 in. thickness) and an outer

liner of 9/16 in. steel plate. The hoops are $1\frac{3}{8}$ in. square bars spaced vertically at 3 in. centers. The outer diameter of the hoops is 16 ft 0 in., and the shield extends from the floor of the cavity to 9 in. below the cell floor shielding for an overall height of 22 ft 9 in. (This gap is provided for the shield cooling system gas.)

In order to utilize the plastic straining capability of the steel shell encasing the outer surface of the concrete one-half of all the radial blast energy is assumed to be absorbed in the hoops and the remaining one-half by the steel shell. This is considered a reasonable assumption, since the hoops cannot expand unless the concrete, and hence, its steel container, expands.

The concrete enclosed by the steel hoops will, of course, tend to move radially outward as the steel undergoes plastic deformation. Since this concrete serves as a piston or loading device, on the steel bars, it is necessary to permit this concrete to slide outward. This is accomplished by eliminating any structural ties or continuity between the shield concrete and the rest of the containment structure.

A generous gap of 12 in. is provided between the hoop steel of the shield and the containment steel liner plate on the cavity walls. Thus, even though the radial energy absorbing shield undergoes significant deformation, the 1/2 in. liner plate which forms the ultimate containment leaktight barrier will not be subjected to any significant radial blast loading due to the energy released during a nuclear accident.

3. Penetrations

The containment structure has numerous penetrations to accommodate personnel access or pass through of viewing devices, instrumentation lines, electrical cables, pneumatic lines, etc. All of these penetrations are specially designed to assure containment integrity. The seals and welds of each penetration are designed either for initial testing with the overall structure or, in the case of the penetrations which are expected to be opened for operational functions, locally retested without repressurizing the structure. Table 10 lists all FARET containment penetrations. Figures 17, 18 and 19 illustrate the location of each penetration. Figures 20 through 25 show details of the various types of penetrations which are used for the purposes listed in Table 10.

4. Cell Shielding Windows

In order to observe and control operations within the cell six identical viewing windows are provided. The overall thickness of each

TABLE 10. Penetration List

DESCRIPTION	DESIGNATION*	NO. REQ'D.	REF. TYPE	STRUCTURAL TEST AT 30 psig		NORMALLY PRESSURIZED WITH ARGON	PERIODIC LEAK TEST
				INTEGRAL WITH CONTAINMENT	INDEPENDENT		
CELL							
Electrical (6 in. Stepped)	CP-FH-1 thru CP-FH-6 CP-FD-1 & CP-FD-2 CP-SE-1 thru CP-SE-4 CP-S-1 thru CP-S-6 CP-WS-1 thru CP-WS-12 CP-C-1 thru CP-C-4 CP-CE-1 thru CP-CE-3 CP-EX-3 & CP-EX-4	39	A	No	Yes	Yes	Yes
Gas Tube (6 in. Stepped)	CP-FD-3 thru CP-FD-8	6	A	No	Yes	Yes	Yes
Electrical (3 in. L-Type)	CP-WF-1 & CP-WF-2	2	D	No	Yes	Yes	Yes
Gas Tube (3 in. L-Type)	CP-WF-3	1	D	No	Yes	Yes	Yes
Instrumentation (6 in. S-Type)	CP-Cl _a -1 & CP-Cl _a -10	10	C	No	Yes	Yes	Yes
Instrumentation (6 in. L-Type)	CP-Cl _b -1 & CP-Cl _b -2 CP-Cl _b -4 thru CP-Cl _b -10	9	D	No	Yes	Yes	Yes
Tube (6 in. L-Type)	CP-Cl _b -3	1	D	No	Yes	Yes	Yes
MI Cable	P-1 thru P-10	10	B	Yes	No	No	No
Gas Cooling (Pipe)	CP-FD-9 & CP-FD-10 CP-FH-9 & CP-FH-10	4	J	No	Yes	Yes	Yes
Personnel Lock	A-1 & A-9	2	P	Yes	No	Yes	Yes
Window	CP-W-1 thru CP-W-5 & CP-W-9	6	Q	No	Yes	Yes	Yes
Equipment Transfer Port	H-17	1	S	Yes	No	Yes	Yes
Fuel Transfer Port	H-18	1	S	Yes	No	Yes	Yes
Equipment Hatch	H-15	1	N	Yes	No	Yes	Yes
Transit Holes (Tool)	CP-FH-7 & CP-FH-8	2	L	Yes	No	No	No
Emergency Crane (Tool)	CP-C-5 thru CP-C-7	3	L	Yes	No	No	No
Small Tool Pass Thru (Equip.)	CP-ST-1	1	G	No	Yes	Yes	Yes
Bubble Crane Access (Equip.)	CP-C-8	1	G	No	Yes	Yes	Yes
Pressure Transmitters (Equip.)	CP-VI-1	1	G	No	Yes	Yes	Yes
Pipe	P-66, P-81, P-82, P-84 P-92 thru P-97	10	J	Yes	No	No	No
ANNEX							
Personnel Hatch	A-10	1	R	Yes	No	Yes	Yes
Service Penetration	CP-AN-1	1	F	Yes	No	Yes	Yes
VAULT							
Valve Operator (Shaft)	V-1 thru V-3, V-8 V-13 thru V-27 V-34 thru V-37	23	H	Yes	No	No	Yes
Hatch	H-5 Thru H-10 H-21	7	M	Yes	No	No	Yes
Pipe	B-1, B-3, B-5	3	K	Yes	No	No	No
Electrical & Instrumentation	B-300 thru B-316 B-330 thru B-332 Y-2	20	E	No	Yes	No	Yes
Mechanical Equipment (Shaft)	V-5 thru V-7, V-9 thru V-12 V-29 thru V-33	12	H	Yes	No	No	Yes

*Location of Penetration Designations are shown on Figs. 17, 18, 19

window is five feet. Vertical and horizontal sections of the cell window are illustrated in Fig. 26. There are four subassemblies:

(1) The fabricated steel liner imbedded in the concrete cell wall and welded to the inner and outer steel shells of the cell wall.

(2) The tank unit containing glass slabs T-1 through T-6. All free space between glass slabs is filled with a clear mineral oil.

(3) The "B" assembly consisting of a steel frame unit containing glass slabs B-1 and B-2.

(4) The "A" assembly consisting of a steel frame containing glass slab A-2, and the glass plate A-1 attached to the frame facing the cell. The "A" assembly will be hinged to simplify replacement, should that be necessary at some later date.

To maintain cell gas purity, and to contain all radioactive particulate matter within the cell, the window will be a highly leak resistant assembly. The primary cell atmosphere gas seal will be established by gasketing the glass slab B-2 directly to the liner. Since the slab B-2 is wider than slab B-1, the steel frame is permitted to exert pressure on the B-2 glass along two sides. Cushion gaskets over the bearing area of the glass prevents direct contact between the glass and the steel frame. A secondary cell atmosphere gas seal will be provided by a peripheral gasket at the operator's face of the window bridging the gap between the tank unit and the liner. The peripheral gasket will be designed to withstand a pressure of 60 psig across its effective area. The annular cavity between the tank unit and the liner will normally be pressurized to about 12 in. W.G. with argon gas. In case of minor leakage of the primary seal only argon gas can pass into the cell and thus purity of the cell atmosphere is maintained.

The primary seal gasket will be protected from cell gamma radiation by 20 in. of 3.3 density glass. In addition the gasket material which is specified as Koroseal 116, a polyvinyl chloride material being quite resistant to gamma damage is expected to last the life of the window. The window, as an integral part of the reactor containment structure will be designed to withstand a temperature and pressure excursion to 1900°F and 30 psig without failure as a containment component, although extensive damage to the visual qualities of the window is to be expected. A shop proof test of 60 psig hydraulic pressure will be applied to the B-2 slab and frame and liner assembly before slab B-1 is installed during fabrication.

G. General Access Control

The facility has been divided into four radiation zones, based on a combination of anticipated radiation levels during normal operation and possible contamination during certain maintenance, refueling, and operating conditions. The FARET shield design is based on the general criteria for keeping the radiation dose rate at or below 2.5 mr/hr outside the cell and vault during normal facility operation. Under certain circumstances, the radiation dose rate in some regions may exceed this level. When these

conditions prevail, it may be necessary to temporarily limit access. Figures 27-30 show the zoning breakdown of the facility.

1. Zone I: Reactor Building Front Cell Area

Shielding is provided to maintain the front cell area radiation level at or below 2.5 mr/hr at all times during normal reactor operation. Basement shielding is based on full power reactor operation; first floor shielding is based on refueling operations in which no more than the equivalent of one spent fuel subassembly source is withdrawn from the reactor or wet fuel storage tank at any one time.

Access to the front cell area is provided from the support wing at the first floor level. Egress is provided at both the basement and first floor levels. An audiovisual alarm at the guard desk actuates when the basement door is opened. The location of the guard's desk permits him to observe the entrance to the corridor which leads to the control room and door to the front cell area.

2. Zone II: Reactor Building Rear Cell Area

Zone II is divided into three subzones, reflecting the possibility that at certain times the radiation level in various areas of the zone may exceed 2.5 mr/hr. Zone IIa is that portion of the rear cell area which will have radiation level not exceeding 2.5 mr/hr during normal operations, but in which contamination and opening of cell and vault shield access plugs may occasionally raise the radiation above this level. Zone IIb is the region above the cell and cell annex; during some refueling operations, the radiation level in this region may be as high as 50 mr/hr. Zone IIc consists of the fuel transfer tunnel, the radioactive liquid waste holdup tanks, radioactive gas holdup tanks, and the radioactive vent compressors. Because the holdup tanks and compressors may contain radioactive materials at various times, shielding is provided to segregate them from the rest of the rear cell area and from each other. (Entry to these areas is through doors which are normally locked.) The fuel transfer tunnel may become contaminated during transfer operations. In addition, fuel transfer casks may be decontaminated and washed in the tunnel prior to shipment. The tunnel is therefore, separated from the rest of the rear cell area by a waterproof partition and locked door to facilitate decontamination operations and insure that contaminants are not spread beyond the fuel transfer tunnel area.

Access to Zone II is normally from Zone I through the change room on the first floor. Additional access is available, when required, through truck doors to the fenced building exterior on the first floor and

doors to the front cell area on the basement and first floors, all of which are administratively controlled and normally locked. Emergency exits to the building exterior are provided on the first floor. The guard controls access to the fenced area by means of an electric gate visible from his station.

3. Zone III: Vault and Reactor Cavity

The radiation level in the vault is governed by the sodium activity level. Access to the vault, which is limited to those times during shutdown when the sodium activity has decayed sufficiently to permit entry, is through a shield plug in the vault ceiling. The plug can be moved only by the crane, preventing inadvertent exposure of personnel to vault radiation. The reactor cavity is not accessible at any time.

4. Zone IV: Cell Annex and Cell Access Locks

The radiation level in Zone IV is far above human tolerance during reactor operation and some fueling operations. Early in the facility's operating life, access to the cell during shutdown will be possible. Over the lifetime of the facility, however, contamination of Zone IV may well increase, making access more difficult as time progresses. Access to the cell is normally through two locks, which are included in Zone IV. The locks are within shield. Labyrinth or removable shielding of the outer doors is provided. The doors are normally locked. Additional access for equipment handling is available through a hatch in the cell ceiling. The hatch can be moved only by the crane.

H. Ventilation

1. General

The design criteria for the ventilating systems are based on established safety practices for the protection of personnel wherever radioactive materials are encountered. The application of these safety practices as they apply to the FARET Project include: (a) prevention of the spread of activity, by controlling pressures within the building so that air flow is always from clean areas toward areas of potential contamination, (b) maintaining subatmospheric pressures in the reactor building to reduce the possibility of radioactivity out-leakage, (c) restriction of radioactive gases or particulate matter discharges from the 200 ft high ventilation stack to allowable levels.

2. Support Wing

The support wing air supply unit (Fig. 31) is located in the support wing basement and consists of a preheat coil, filters, air washer, and

a 14,000 cfm centrifugal blower. Reheat coils are installed in the distribution ductwork. A centrifugal blower and duct system are provided to exhaust air from the building or to return this air to the supply unit. The percentage of outdoor and return air can be manually selected (from 100% F. A. to 20% F. A.).

The control room air conditioning unit is also located in the support wing basement and consists of a filter, a multizone type air handling unit with heating coil, cooling coil, and centrifugal blower, a return air fan, and associated ductwork. This unit serves both the control room and the cable spreading room. An air handling unit consisting of a filter, heating coil and centrifugal blower provides 3,200 cfm of combustion and ventilation air for the boiler room.

Supply and exhaust air flows are balanced to maintain the support wing at a slightly positive pressure with respect to outdoor ambient and to maintain the control room pressure slightly positive with respect to the remainder of the support wing.

3. Reactor Building

From the standpoint of ventilation, the reactor building has been considered to consist of three areas. These are: (1) front cell area, (2) rear cell area, and (3) containment areas (cell, cell annex, vault-cavity). The areas are listed in order of increasing potential contamination. See Fig. 32.

The ventilation systems provided are common to both the front cell and rear cell areas. The containment areas normally operate with an argon atmosphere but provisions are made to change to an air atmosphere during periods of personnel access (reactor secured); a system is provided to ventilate these areas during these periods.

A. Description

The ventilation system consists mainly of four subsystems; supply air system, building exhaust system, emergency exhaust system, and the containment area ventilation system.

1) Supply Air System

The air supply equipment is located on the second floor of the front cell area and consists of two units each with preheat coil, air filter, air washer and 30,000 cfm capacity belt-driven centrifugal blower. The two blowers are interconnected and then connected to the distributing duct system. Automatic control dampers are installed in the supply main to the front cell area and the supply main to the rear cell area.

2) Building Exhaust System

The building exhaust system consists of two (2) belt-driven centrifugal fans of 35,000 cfm capacity each located on the mezzanine at the north end of the rear cell area, a 200 ft high ventilation stack, and a system of ductwork. Motorized dampers are provided in the discharge from each blower.

3) Emergency Exhaust System

The emergency exhaust system is located on the mezzanine at the north end of the rear cell area near the building exhaust blowers and consists of two (2) 5,000 cfm direct-driven centrifugal blowers, ten (10) final filter assemblies, interconnecting ductwork and motorized dampers to automatically divert flow to this system during emergency conditions.

4) Containment Area Ventilation System

The containment area ventilation system is also located on the north mezzanine and consists of one (1) 1,000 cfm centrifugal blower, one (1) final filter assembly, and ductwork connections from each of the containment areas. All air both into and out of these areas is filtered before being discharged to the suction of the main building exhaust fans. All containment area ventilation openings are automatically closed (by sealed automatic butterfly valves) in case of a rise in pressure. These valves are normally closed to failsafe in case of loss of instrument air pressure. Manual butterfly valves are installed in series with the automatic valves.

B. Design Requirements

(1) Pressures within the building will be controlled such that air flow is always from clean areas toward areas of potential contamination in order to protect operating personnel and prevent the spread of activity.

(2) Pressure in the reactor building will be maintained subatmospheric in order to preclude the possibility of out-leakage and thus prevent a ground release of radioactivity.

(3) Discharge of radioactive gases or particulate matter from the 200 ft high ventilation stack must not exceed allowable levels.

C. Operation

The following pressures have been selected for the various building areas in order to ensure air flows in the proper direction.

- (1) Front cell area--0.15 in. W.G. negative (with respect to atmosphere).
- (2) Rear cell area--0.35 in. W.G. negative.
- (3) Cell--2 in. W.G. negative to 4 in. W.G. negative.
- (4) Cell annex-- $1\frac{1}{2}$ in. W.G. negative to $3\frac{1}{2}$ in. W.G. negative. (Annex pressure is controlled to be $1/2$ in. W.G. positive with respect to the cell.)
- (5) Vault-cavity-- $1/2$ in. positive to 4 in. positive (with respect to atmosphere).

A motorized damper automatically regulates the air flow to the front cell area in order to control the pressure in this area. Another motorized damper similarly controls the pressure in the rear cell area. These automatic pressure control dampers are located so as to control the supply air flow rather than the exhaust air flow. In this way the building is maintained at subatmospheric pressure despite the failure of one exhaust fan or the opening of outside doors.

Since only one pressure controller is provided for the front cell area, relief openings are provided to equalize pressures between the various floors and confined spaces within the area. The same is true for the rear cell area. These relief openings are fitted with fire dampers where they penetrate fire walls. The ventilation system is balanced manually to minimize flow through these relief openings.

Control of pressure within the containment areas during normal operation (argon atmosphere) is accomplished by the radioactive vent system in conjunction with the argon supply system. Pressure control is normally not required in these areas during operation with an air atmosphere (except to keep cell and annex slightly negative with respect to the rear cell area) as barometric pressure changes are communicated from the rear cell area through the ventilation air intakes. However, pressure control in the annex is required during operation with an air atmosphere, if the cell is operating with an argon atmosphere, in order to reduce the pressure difference across and consequent leakage through the head lifting bellows. This control is provided by a modulating valve in the annex ventilation inlet.

The vault-cavity is maintained at a positive pressure with respect to the surrounding areas. This prevents contamination of the vault-cavity argon atmosphere due to air inleakage. The normal outleakage from the vault-cavity does not represent a hazard to personnel working in adjoining areas from the standpoint of radioactive argon.

The responsibility for ensuring that the reactor building pressure remains subatmospheric, to prevent a ground level release of radioactivity,

is the prime function of the emergency exhaust system. A high degree of confidence in the ability of this system to remain continuously operable and capable of performing its task has been the main design objective. This confidence stems from the following design features.

- (1) Two direct drive blowers are provided, each with sufficient capacity to maintain the building at a subatmospheric pressure when operating alone. It is estimated that this building in-leakage will be well below the 5,000 cfm blower capacity based on the use of a metal wall panel with a gasketed interlocking side joint.
- (2) Both blowers are operated from the diesel-generator emergency power source in case of failure of the normal power supply.
- (3) At the time of project termination two schemes were being investigated to insure the operability of the emergency exhaust system in event the diesel-generator fails to start immediately on main power loss. In one of the schemes, one of the emergency exhaust fans would be connected to operate for a period of about 1/2 hour from the continuous power source consisting of batteries plus a motor-generator set. This capability to operate on the continuous power system would be for a considerably greater period of time should the emergency occur at any time other than during fuel handling. It is assumed that operating personnel would manage to start the diesel during this period. In the other scheme a small continuously operating diesel would be provided for the emergency exhaust fans. These fans could then operate on main power, the main emergency diesel or the auxiliary diesel.

The adopted scheme would be reported in the final safety analysis report.

It should be noted that the following measures to enhance reliability are incorporated into the design and operating procedure of the diesel-generator. These are:

- (1) Use of high quality fuel (No. 1 oil) for easier starting;
- (2) Use of a reliable starting method (compressed air);
- (3) Periodic operation of diesel to assure that it is in operating condition.

The emergency exhaust system is normally at standby condition but is automatically activated on any of the following occurrences.

- (1) High radioactivity in the ventilation exhaust flow to the ventilation stack.
- (2) High pressure in the cell.
- (3) Failure of the main power supply.

Activation of the emergency exhaust system results in the following automatic operations but not necessarily in the order shown below.

- (1) The supply fans are de-energized.
- (2) The building exhaust fans are de-energized and exhaust fan discharge dampers are closed. These dampers are installed to minimize backflow from the stack during periods when the emergency exhaust system is operating and to minimize recirculation in case of the failure of one building exhaust fan during normal operation. They are normally closed to be failsafe.
- (3) The emergency exhaust fans are energized, from the main power source, the diesel-generator emergency power source, or the continuous power source depending on the cause of the emergency condition and on diesel startup as previously discussed.
- (4) The filter inlet valve and the emergency exhaust fan discharge dampers are opened. The filter inlet valve is a leaktight rubber seated butterfly type to prevent backflow through the filters during normal operation. The valve and dampers are normally open to be failsafe.

Absolute type filters are used in the emergency exhaust system filter assemblies to prevent the discharge of radioactive particulates to the ventilation stack. All filters are specified to have an efficiency of 99.97% for 0.3 micron diameter particles of DOP, and to be tested at Hanford, in accordance with Military Standard 282 Method 102-91 dated May 28, 1956. Following the tests and on arrival at NRTS all filters will be visually inspected for possible tears and other signs of damage due to shipping. Filters will be rejected if damage is found.

Following installation, the filters will be tested using a portable DOP aerosol generator which produced particles having a mean diameter of 0.9 microns. Filter efficiencies of less than 99.9% measured by this method will cause filter rejection. Subsequently, the filters will be routinely tested in place periodically by the same method.

The requirement for halogen filters is not yet clear. Various experimental programs are expected to yield data directly applicable to this problem. In addition, the leakage rate of the containment, as built, is an important factor in this analysis. Therefore, the final safety analysis report is expected to pursue this question in more detail than is presently available. Provisions have been included in the design for the installation of activated carbon halogen filters should such an installation become necessary.

VI. THE REACTOR AND COMPONENT SYSTEMS

The reactor and component systems are comprised of the reactor subassemblies, vessel, internal shielding and components, control systems, and the fuel and scrap handling systems. These systems are located both within and without the containment structure, e.g., the vessel and core are located within the containment structure whereas the control instruments and thimbles are located externally.

A. Reactor

1. Reactor Vessel

a. General

The reactor vessel assembly (Fig. 33) includes the double-walled reactor vessel, core support structure, neutron shield, vessel extension, and cover. The assembly is located within the reactor vessel cavity and is thermally insulated on all sides. External to the vessel assembly and insulation and within the cavity liner is a radial energy absorbing structure. This is an annular ring of concrete reinforced on the outer periphery with vertical and circumferential steel. This structure also serves to moderate the neutron flux leaving the reactor vessel. The cavity is cooled by an argon cooling system.

The vessel assembly is specified as Type 304 stainless steel with a composition range of 18-19% chromium, 11.5-12% nickel, 1-2% manganese, 0.5% max. silicon, and 0.04-0.06% carbon. The vessel is supported at the bottom and guided laterally at the top by the cavity expansion joint seal. The inner vessel carries the internal load and is free to expand in the vertical direction. The annular volume between the inner and outer walls contains NaK with provisions for monitoring the NaK for change in volume. The principal piping connections to the vessel are (1) the 24-in. main outlet, (2) 14-in. main inlet, (3) 6-in. auxiliary outlet, (4) 4-in. overflow outlet, (5) 3-in. overflow return, and (6) 2-in. auxiliary bypass. The main inlet, auxiliary outlet, and auxiliary bypass nozzles are doubly contained. There are no piping connections to the vessel below the top of the core.

The core support structure is comprised of the coolant inlet lines; high and low pressure inlet plenums and core support grid plates; the core cylinder; and an upper coolant mixing chamber and exit ports. This structure, which is located on a support at the bottom of the vessel, is attached to the vessel by a single, flanged and bolted, inlet piping connection. The core support cylinder outlet coolant connection consists of a loose sleeve which is bolted to the internal extension of a thermal sleeve in the reactor vessel outlet nozzle. These sleeves are not physically connected to the support structure or to the outlet pipe.

The reactor sodium coolant enters the vessel through the 14-in. inlet line. This line divides within the vessel into two 10-in. inlet lines which connect to a chamber surrounding the high pressure inlet plenum. A circumferential baffle plate is provided between the chamber and the plenum to promote uniform radial flow into the plenum. The inlet piping, chamber, and high pressure plenum operate at pressures approximating the discharge pressure of the main reactor coolant pump. A low pressure inlet plenum receives coolant from the high pressure plenum through orificed passages in the subassemblies.

The reactor uses hexagonal subassemblies arrayed on a triangular lattice. These subassemblies are located by and supported on the upper grid plate of the core support structure. The lower portion of each subassembly is cylindrical and penetrates the upper and lower grid plates. The high-power subassemblies receive coolant from the high pressure plenum and low-power subassemblies from the low pressure plenum. In addition, there is a controlled flowrate, of up to 20% of the total, which bypasses the core and provides cooling for the reactor vessel and the neutron shield. Additional vessel cooling is provided by the vessel overflow system.

The coolant outlet plenum formed by the subassemblies is located about 5 ft below the top of the subassemblies. Most of the core coolant emerges from the subassemblies into the outlet plenum, passes across the core into the mixing chamber, flows under a mixing chamber baffle, and then passes upward and out through the main and/or auxiliary coolant outlet nozzle. Some coolant passes upward from the outlet plenum through the annular spaces between the 5 ft shielding extensions of the hexagonal subassemblies. This coolant and the coolant that bypasses the core and enters through radial openings in the lower part of the mixing chamber, mix with the main coolant stream in the mixing chamber. The combined flow then enters the vessel outlet piping.

The upper four feet of the reactor vessel consists of a removable extension which contains a number of penetrations for in-core instrumentation and miscellaneous in-vessel components. The lower flange of the conical extension is bolted and seal-welded to the vessel. Members attached to the conical extension guide the upper portion of the core support cylinder.

The reactor vessel cover rests on the top flange of the extension and is removable for periodic access to the core. Shielding is provided both within and external to the cover. Penetrating the cover are nozzles and sleeves for the reactor control drive mechanisms which are attached to and supported on the cover. The cover with attached drive mechanisms is handled by a separate hoist.

b. Steady-state Operating Conditions

The steady-state operating conditions are as follows:

1) Power Operation

14 in. main inlet flow:

Maximum 7500 gpm

14 in. main inlet floor

Minimum 750 gpm

Power loss* 450 gpm

Core bypass flow:

Maximum 1500 gpm

Minimum 150 gpm

Power loss* 90 gpm

Flow through core:

Maximum 6000 gpm

Minimum 600 gpm

Power loss* 360 gpm

2 in. auxiliary loop bypass maximum flow 50 gpm

3 in. overflow and make-up return,
maximum flow 100 gpm

4 in. overflow, maximum flow 100 gpm

Maximum mixed mean coolant temperature rise after mixing with reactor
bypass flow 190 °F

2) Isothermal Operation

14 in. main inlet flow:

Maximum 7500 gpm

Minimum 750 gpm

4 in. overflow and make-up return,
maximum flow 100 gpm

*Power loss condition is loss of main primary pump. An E-M pump in series with main primary pump is operable, providing 6% of full flow.

Maximum temperature for isothermal operation while maintaining system temperature with external heat supply 600°F

Maximum temperature for isothermal operation while maintaining system temperature with nuclear (core) power and external heat supply 1050°F

3) Auxiliary Loop Operation (No Main Loop Flow)

14 in. inlet flow, maximum 450 gpm

6 in. auxiliary outlet flow, maximum 450 gpm

4) Internal Heat Generation

The reactor vessel is designed for radiation heating of 0.02 watts/gram in the inner vessel wall and 0.01 watts/gram in the outer vessel wall at core mid-plane.

5) Transient Operating Conditions

A comprehensive mathematical model and digital computer program were developed to simulate operation of the reactor and coolant systems. The model simulates operation of the reactor, reactor kinetics, the coolant paths inside the reactor vessel, main primary coolant flow and main secondary coolant flow. The applicable controls for these primary coolant systems and for the air flow through the main air cooled heat exchanger are included.

Although the initial efforts on simulation were to provide a mathematical model for analog solution of transient problems, subsequent work indicated that digital computer solutions of the problem were generally more satisfactory. The final transient values presented herein were programed FORTRAN II for use on an IBM 7094 with a 32 K memory. The model involves simultaneous solutions (approximate) of differential equations for each variable. The program starts with initial input assumptions for each variable, calculates derivatives, takes a time step, and calculates a new value for the variable at the end of the time step. Many runs were made to evaluate effects of control alternates, and high or low reactor decay power effects. Certain runs were programed to stop at a specific point to insert dominant effects of the auxiliary coolant system prior to continuation.

For accuracy the model was built to contain multiple zones in the main heat exchangers and certain pipe sections. The differential temperatures between outer and inner walls of the doubly contained

reactor outlet piping were investigated by including modeling of the heat transfer to the inner wall, and conduction to NaK and the outer wall. The design transients are shown in Table 11 and by Figs. 34, 35, and 36.

TABLE 11. Temperature Changes for Reactor Vessel Design

Nature of Change	No. of Events	Temperature			Remarks
		Start of Change, °F	Rate of Change, °F/hr	Range, °F	
Dry Heatup	50	70	50	280	1. Vessel contains argon gas and is heated by circulating NaK in outer vessel annulus in 150 hr or less.
Normal Heatup	1000	250	50	350	2. Primary sodium coolant circulated through core and heated by external source within 25 hr.
Nuclear Heatup	250	600	50	450	3. Same as (2) except heat source is nuclear.
Normal Cooling	1000	1050	-50	700	4. Primary sodium coolant flow through core is cooled externally.
Refuel Cooling	1000	700	-50	400	5. Same as (4).
Temperature	a) 3500 b) 3500	600-inlet 960-inlet	50-inlet -50-inlet	360 360	6. Reactor power 50 MW; outlet temperature 1150°F, or less.
Flow	1000	600-inlet	+50-outlet	190	7. Inlet temperature and power constant.
Flow	1000	960-inlet	-50-outlet	190	8. Same as (7).
Power	1000	600-inlet	+50-outlet	190	9. Full flow rate and inlet temperature constant. Power change through 0 to 50 MW range.
Power	1000	960-inlet	-50-outlet	190	10. Same as (9).
Emergency Shutdown	400	1150-outlet	Fig. 34		11. Reactor power from 50 MW to decay heat. Full flow rate reduced to 10%.
Emergency Shutdown	50	960-inlet 1150-outlet	Fig. 36		12. Same as (11) except flow rate reduced to 6%. (Loss of Primary Pump.)
Emergency Shutdown	50	960-inlet	Fig. 35		13. Same as (12) and coolant level lowered to bottom of 24 in. outlet nozzle. (Loss of Main Coolant.)

2. Neutron Shield

a. General

The neutron shield surrounds the core support cylinder, within the reactor vessel, as shown in Fig. 37. The shielding material is borated-graphite. To protect the graphite and avoid contamination of the primary sodium, the graphite is hermetically sealed in stainless steel containers. The stainless steel containers are designed to withstand a positive internal pressure 50 psi greater than the external pressure. Special cans are required to fit around the inlet and outlet piping and thimbles within the reactor vessel. Also, to facilitate handling and installation, various lengths of cans are required.

The cans are placed in four rows around the core support cylinder. Each row is held in place by a system of stainless steel bands. Adjacent rows of the graphite shield cans are staggered to reduce neutron streaming.

External to the reactor vessel and within the reactor cavity liner (and hence, within the containment) is a radial energy absorbing shield which consists of an annular ring of ordinary concrete reinforced with steel hoops. This structure also serves to further moderate the neutrons leaving the reactor vessel.

Outside the cavity liner is the biological shield of ordinary concrete. The nuclear instrument thimbles are located at the inner face of this shield. In order to assure that the fluxes at the instruments will be adequate, a "window" is provided in the concrete energy absorbing shield, spanning the location of instrument thimbles. As a result of increasing the neutron flux at the detectors, local high heat generation rates will be produced in the detector regions and water cooling internal to the concrete in the general area of the thimbles is required. The thimbles, however, do not require internal cooling. Concrete temperatures are kept at or below about 200°F. In most areas the concrete is cooled with argon from the vault.

Heat generation rates in the graphite have been estimated and temperature calculations indicate that the graphite will be of the order of 100°F higher than that of the surrounding sodium. Thus, the pressure criterion for the can has been based on a temperature of 1150°F and the can thickness will be determined accordingly. Calculations of the pressure buildup due to helium formation have been made on the basis of boron content of 1 w/o in the graphite for the innermost row at the core elevation. It appears feasible that the design of the cans can be optimized with respect to boron content, can thickness, and pressure to the expected life of the reactor which is 15 yr at a 50% load factor.

Long term temperature effect experiments conducted on EBR-II type canned graphite in the range of 1000°F have shown satisfactory results. By the use of procedures and materials similar to those used for the fabrication of EBR-II shield cans for the FARET shield components, and by proposed tests at FARET temperature conditions, it is anticipated that satisfactory can performance will be achieved and demonstrated.

The literature deals extensively with the performance of nonborated graphite materials exposed to reactor environment. For example, in stainless steel-sodium-graphite systems there has been concern about carburization phenomena. It is reported that moderate carburization begins at temperatures of the order of 600°C or higher. For the FARET conditions of 570°C the effect should be minimal. The borated graphite has essentially identical properties as the nonborated type. There is considerable developmental experience with the EBR-II, 3 w/o B₄C graphite and there is confidence therefore that the existing information on both nonborated and borated graphite can be applied safely to the FARET conditions.

b. Design

1) Neutron Flux Calculations

Neutron fluxes were calculated using the FAIM code and the ANL 16-group cross section set number 200. The combination removal-diffusion method was used, except where straight diffusion gave more conservative results. Calculations were performed at various angles

about the core horizontal midplane in order to investigate local "hot spots." For each calculation, the core used was that core which gave the most pessimistic fluxes. For example, in the radial direction, the largest contemplated core was used, while in the downward axial direction, the small, high power density core was used.

The calculation of fluxes at the instrument thimbles was made on the basis of the core giving the least fluxes. In addition, the computed fluxes were reduced by an arbitrary factor for added conservatism.

2) Gamma Flux Calculations

The gamma fluxes and heating rates resulting from the (n,γ) reaction in all materials in the shield were calculated using the GRACE I code and the neutron fluxes generated as described above. For conservatism, the computed heating rates were increased by 50%.

3) Heating Calculations

The gamma heating rates computed as described above were used in the GHT Code which calculates temperatures in solids in two dimensions, allowing internal heat generation and convection boundary conditions. Calculations were performed on the horizontal midplane of the core in order to design the argon cooling system. In addition, temperatures were calculated through the neutron window in the blast shield concrete with and without water cooling coils embedded in the concrete.

4) Results

a) Neutron Flux Calculations

The radial flux throughout the shield is given in Fig. 38, and the flux beneath the core is given in Fig. 39. In each case the fluxes shown are the fast removal flux and thermal flux, and were calculated on the horizontal and vertical centerlines, respectively. The configurations for each calculation are superimposed upon the flux plots. Both figures are for the case where there is no neutron window.

b) Gamma Flux Calculations

Figure 40 gives the radial gamma heating rates throughout the shield without the neutron window. The heating rates in the biological shield in the vicinity of the thimble locations were approximated by the heating rates at the same relative location in the blast shield concrete.

c) Heating Calculations

Using the information from Fig. 40, a centerline temperature calculation was performed through the radial shield. The resulting temperature distribution is shown in Fig. 41. Note that the maximum concrete temperature is about 216°F in the blast shield concrete. This would increase about 16°F at the end of life due to B¹⁰ depletion in the graphite neutron shield.

The peak temperature in the biological shield opposite the neutron window without internal cooling coils was calculated as 540°F and with cooling coils less than 200°F.

3. Core

The basic core arrangement consists of 301 EBR-II type hexagonal subassemblies. These subassemblies are arranged in a close packed geometry on a 2.320 in. triangular lattice in the core support cylinder having a diameter of about 4½ ft (Fig. 7). A single subassembly size, 2.290 in. across flats by 177 in. long, is employed (Fig. 42). A common size bottom end fitting is employed for all subassemblies, except those located in the outer three rows, to permit flexibility of core arrangements. Likewise, the upper end fitting of each subassembly is designed so that all subassemblies may be manipulated with the same handling and transfer devices. Each subassembly contains fuel, core components, or other test or instrumentation components. The leads for instrumentation located in a subassembly terminate in the upper end fitting of the subassembly. A connector, either mechanically-coupled or welded to the upper end fittings, with attached lead is used to bring the signal through a port in the vessel extension, through a penetration in the cell, to a terminal location external to the cell for display of the sensing element signals on panels located in the control room or other appropriate locations. A nominal clearance of 0.030 in. between subassemblies is provided to facilitate removal or insertion of the subassembly. For closer packing, each subassembly makes contact with adjacent subassemblies by means of buttons formed in the hexagonal subassembly tubing.

The control subassembly consists of a control rod and a fixed guide thimble. The guide thimble is hexagonal in cross section and of the same dimensions as the subassembly tubes. Each thimble, therefore, occupies a unit lattice that is identical to those occupied by the various subassemblies. The control rod consists of a hexagonal subassembly that is 1.827 in. across flats. The control rod contains B₄C material encased in stainless steel in the active region. The remaining sections of the rod are similar to the other subassemblies. Bearings are provided on the lower section which provides the guide between the control rod and the guide thimble.

B. Control Rod Drives and Cover Handling

1. Control Rod Drives

Operation of the reactor is controlled by eleven control rod drives and one oscillator. Each rod is driven independently by an electrical-mechanical drive mechanism that is essentially the same as used in EBR-II (Fig. 43). The control rod drives are identical and so arranged that only one drive may be operated at a time (with the exception of "scram" when all operate simultaneously or when the reactor is being manually "shimmed" to keep within operating limits prescribed by the automatic flux control system).

The rod drives are mounted on a lifting mechanism located on the vessel cover. For removal of the vessel cover the control drive assemblies first are placed in the "down" position. Then the drive assemblies are disengaged from the rods. After all control rods are released the control drive lifting platform is raised several inches. During this operation each gripper sensing device will indicate whether separation of the control rod from the drive assembly has been effected. When all control rods have been released and the platform raised, the reactor vessel cover can be raised.

The major portion of each control rod drive mechanism exists above the reactor vessel cover. This portion consists of, in the main, a pneumatic scram accelerator and shock absorber assembly, a motorized lead screw and ball-nut yoke assembly, a yoke-mounted electromagnet and a motorized gripper actuator. The piston of the scram accelerator and shock absorber assembly forms the top end of the vertical drive shaft which extends downward through the reactor vessel cover into the vessel interior. The bottom end of the drive shaft is fitted with a gripper for connecting to the control rod. The drive shaft's armature, when magnetically attached to the vertically driven electromagnet, permits the shaft to be actuated through its full control stroke.

The opening in the reactor vessel cover for each vertical drive is sealed by a concentric welded-disc bellows whose top end is sealed to the drive shaft and bottom end to the vessel cover nozzle. The placement of the component subassemblies in each control rod drive mechanism is dictated by the following conditions: (1) close spacing of control rods, (2) lateral space restrictions and (3) limited height allowance.

The control rod drive mechanisms are mounted in an approximately circular pattern on a horizontal platform fitted with auxiliary vertical supporting members. The platform can be raised or lowered over a range of several inches by its integral motorized jackscrew mechanisms, thereby enabling adjustment of the basic "Down" elevations of the control rod grippers. This is done to effect engagement of the grippers with the

control rods in the downward movement and to obtain clearance between the empty grippers and the control rods in the upward travel.

The drive mechanism-platform complex is mounted atop a base-plate located just above the insulation of the reactor vessel cover. The function of the base-plate is to isolate the thermally expanding cover from the fixed-mounted drive components.

a. Control Rod Gripper

The functions of the gripper (Fig. 44) are to mechanically couple or uncouple the drive shaft and the control rod by remote actuation and to provide remote indication of the action. The control rod's upper adapter consists of a short, vertical 1/2-in.-dia rod topped with a 1-in. dia, 90° cone similar to the one used in the EBR-II reactor. The gripper, also, is similar except that Haynes 25 alloy metal is used in place of stainless steels 304 and 420. The basic component of the gripper is a pair of hinged, single-toothed jaws whose teeth are drawn together underneath the control rod's cone in the gripping action and spread apart in the releasing process. A prerequisite for the gripping function is that the cone must be fully inserted in the guide chamber of the gripper body. The back sides of the jaws have angular cam surfaces on both sides of the hinge pin. The jaws are opened or closed when the cams are acted upon by a pair of parallel bars integral with a jaw-encompassing sleeve. The sleeve's 1 1/2 in. axial movement is imparted to it by a concentric shaft within the drive shaft that extends upward to a point within the control rod drive mechanism above the reactor vessel.

A device which senses the presence of a control rod's cone is normally located in the gripper guide chamber. When the cone enters the guide chamber, the sensing device is displaced, resulting in its upward movement of 3/4 in. The motion is transmitted via a vertical concentric shaft within the gripper actuating shaft to a point in the control rod drive mechanism above the reactor vessel cover.

b. Scram Accelerator and Shock Absorber

The scram accelerator (Fig. 45) consists of a 4-in.-dia pneumatic cylinder assembly whose piston is connected to the control rod drive shaft. A constant downward pressure is maintained on the piston through an enlarged gas inlet at the top of the cylinder. The inlet is connected through a "quick-disconnect" fitting and short length of hose to a torus-shaped gas accumulator common to and encircling all twelve scram accelerators. Gas pressure of only 12 psig is required to achieve a 1.5 g acceleration. The accumulator is equipped with an over-pressure relief valve and a check valve to prevent pressure loss in case of a failure in the constant-feed supply piping.

The shock absorber's hydraulic dash-pot is located immediately below the scram accelerator cylinder. It is concentric with the piston rod, and its straight-walled oil-filled annulus is designed to receive the dash-pot plunger. The cylindrical plunger is attached to the underside of the pneumatic piston. The plunger is also concentric with the rod and is of a stepped-taper design on its outer surface, resulting in a slightly smaller diameter at its bottom end. As the plunger progresses toward the bottom of the dash-pot during a scram stroke, the oil is displaced and forced upward through the decreasing annulus created by the plunger's stepped diameters. The result is a gradual decrease in velocity of the plunger before it comes to a stop. The displaced oil is collected in a reservoir concentric with the dash-pot. The reservoir contains a float-operated magnetic switch that can be connected to an annunciator to indicate a loss of oil.

c. Gripper Actuating Mechanism

The motorized gripper actuating mechanism is used to open or close the control rod gripper jaws. The assembly is mounted on the lifting platform so that its vertical output shaft is directly underneath and aligned with the short shaft emerging from the gripper shaft-actuating jackscrew housing on the control rod drive shaft. Coupling the output shaft with the jackscrew shaft can be achieved only when the control rod drive shaft is in its lowest position. Under this condition the jackscrew shaft is only a fraction of an inch above the hub of the flexible coupling attached to the mechanism's output shaft. The gearmotor's 30 rpm output shaft drives a torque limiting device which in turn drives a spur gear. The spur gear's center has a square hole through which can slide a squared portion of the vertical output shaft. The output shaft is raised 1 in. by a solenoid to effect engagement of the flexible coupling with the jackscrew shaft.

d. Gripper and Sensing Data Transmission

The gripper data transmitting equipment consists basically of a pair of electrical switches and a Size 18 synchro transmitter. A vertical, spring-loaded plunger designed to travel downward over a $1\frac{1}{2}$ in. stroke has its bottom end connected to a gear rack. The rack pinion is fastened to the synchro's shaft. Thus conversion of the $1\frac{1}{2}$ in. linear motion to about 238° of synchro shaft rotation provides a continuous indication of the gripper shaft's motion. A horizontal bar attached to the plunger actuates a limit switch at each end of the stroke for travel limit indication and control of the motor in the gripper actuating mechanism.

The sensing data transmitting equipment is identical to the gripper's except that the plunger is designed to move $\frac{3}{4}$ in., and the rack pinion is necessarily smaller to obtain 238° of synchro shaft rotation.

Both the gripper and the sensing data plungers are positioned underneath and are acted upon by the vertical movements of the

horizontal levers attached to the gripper and sensing shafts and which protrude through the wall of the control rod drive shaft.

e. Lifting Platform and Support Structure

The lifting platform and support structure (Fig. 46) provides mounting and support surfaces for all twelve control rod drive mechanisms. It is equipped with a motor-powered mechanism for raising or lowering itself from the reference operating elevation over a total range of several inches. The purpose of this is to permit adjusting the elevations of all the control rod grippers simultaneously.

f. Environment

The control drive mechanisms are within the area enclosed by the cover bellows and cell annex, as shown in Fig. 47. The system is designed so that the atmosphere in this area can be air. This would permit the drives and cover handling mechanisms to operate in a more acceptable atmosphere than high purity argon gas which requires derating of the electrical equipment. Moreover, access to this area for equipment maintenance is then facilitated.

Initially, it was planned to operate this region with an argon atmosphere since the safety aspects of an air atmosphere as related to a sodium-air reaction in event of a nuclear incident have not been resolved. The argon atmosphere would be maintained slightly less pure than in the cell since very small additions of oxygen and moisture materially help the operation of electrical equipment.

2. Cover Handling

To permit access to the reactor, the cover complex which consists of integral or attached shielding, thermal insulation, control drive mechanisms, control cables and the cover closure mechanisms, is raised and lowered by means of a cover handling system.

The cover handling system (Fig. 48) is comprised of a large bellows, cover hoist, lifting fixture, support structure, support ring, cable take-up mechanism, pulleys and associated control and appurtenances.

The control drive cluster and cables are isolated from the cell atmosphere by means of a large bellows and provides a gastight flexible connection between the reactor cover and the annex floor cylindrical sleeve. A cover hoist mounted to the underside of the annex ceiling raises and lowers the reactor cover by means of a lifting fixture fixed to the reactor cover. The control cables are attached to the reactor cover. The cables run over pulleys located in the annex and are guided into a gastight tank located on the outside of the annex wall.

To provide access to the reactor vessel and to permit unobstructed movement of the in-cell equipment to complete fuel handling, the cover is raised vertically into the cylindrical sleeve mounted to the annex floor directly above the reactor, hereby compressing the large bellows. During the vertical lift of the cover complex the control cables are taken up and kept taut by a loop formed by the pulley counterweight located inside the tank mounted on the outside annex wall. In its fully raised position the reactor cover complex with attached support structure is secured to the support ring at the top of the cylindrical sleeve in the annex floor by means of locking bolts. A double inflatable seal between the cover support structure and the cylindrical annex floor seal permit bellows maintenance to be performed without breakdown of the atmospheric barrier between the cell and annex regions.

A shutter mounted on a carriage and supported on rails from the cell walls forms a closure under the vessel cover when it is in its "up" position. The shutter carriage is driven by a motorized chain drive. Inflatable seals are employed to effect the seal between the shutter and the lower flange of the cylindrical annex floor sleeve. This permits an argon atmosphere to be maintained around the underside of the vessel cover in the event an air atmosphere in the cell is desired.

With the vessel cover secured to the support ring the cover hoist and lifting fixture can be disconnected to permit access to the control drives for adjustments, repairs and replacement. The bellows can be disconnected to permit repairs and maintenance.

Normally the cell and annex contain an argon atmosphere. As noted previously the system is designed to permit using an annex atmosphere of air. This would provide for personnel access to the drives, bellows and control cables without the need of breathing apparatus.

With the vessel cover in place (attached to the reactor) the cell and annex atmospheric separation is effected by the large bellows. Personnel is lowered inside the bellows to perform adjustments on the control drives.

The bellows is constructed of a double layer of a Hypalon neoprene coated nylon fabric. Aluminum slats are bonded between the two layers of the elastomer-coated nylon fabric. The aluminum slats provide rigidity to the convolutions yet permit the fabric to flex during movement of the bellows. In addition the slats reduce the effective leakage and/or diffusion area through the bellows. Nylon tapes fixed to the inside vertex of each convolution and to the metal flanges limit the travel of each convolution and support the weight of the bellows in its extended position. The bellows are equipped with collars at each end for attaching to the proper gasketed metal flanges.

During the movement of the vessel cover, the atmospheric pressure in the cell and annex is controlled to maintain a few inches (water) pressure differential between the two regions.

C. Fuel and Scrap Handling

The fuel and scrap handling systems perform all handling operations for new and spent subassemblies in the cell, provide sodium-cooled storage, cut the spent subassemblies to acceptable out-of-cell handling dimensions, handle scrap, and perform other miscellaneous tasks. The principal components of these systems are the fuel handling machine, the wet fuel storage tank, the subassembly cutoff machine, the scrap handling equipment, the in-cell crane, and the out-of-cell subassembly and equipment transfer system. The arrangement of these components is shown in Fig. 49.

1. Fuel Handling Machine

The fuel handling machine performs several tasks: (1) moves new fuel subassemblies from the storage tank or fuel transfer port to the reactor vessel and installs them, (2) removes irradiated subassemblies from the reactor and transfers them to the wet fuel storage tank, (3) takes subassemblies from the wet fuel storage tank to the subassembly cutoff machine (which removes the fuel containing section from the rest of the subassembly), and (4) moves the fuel section or scrap to the transfer port and deposits its load in a transfer cask for shipment. The machine also can handle reflector subassemblies and, with special grippers, it can handle control subassemblies, special fuel subassemblies, sources, thimbles and other components. In addition, it provides a mobile base for an auxiliary manipulator arm to manipulate instrument connectors inside the reactor and for emergency use within the cell.

The fuel handling machine (Figs. 50-52) consists of a gripper mechanism and a drive which are parts of a barrel assembly that is mounted in a sleeve and trolley. The barrel assembly (including gripper and gripper drive mechanisms) is raised and lowered in the sleeve by another device. The trolley is supported by a travelling bridge that spans the reactor cell. The gripper grasps or releases subassemblies and is operated through a chain linkage. The gripper drive, is powered by a torque motor, a modified three-phase squirrel cage machine capable of maintaining stalled torque for extended times. This device moves the gripper assembly up and down relative to and within the barrel. The operator can regulate the pulling or lifting force on the gripper by monitoring and adjusting the motor current from outside the cell. Downward force is limited by a fixed weight acting on the gripper chain. The hoist drive, sprockets and chain retainer are at the upper end of the barrel; at the lower end are two pairs of shutters for directing gas coolant into the fuel subassembly's lower adapter. The gripper-barrel assembly has been made compliant and is fitted with appropriately shaped leads so that subassemblies can be engaged and disengaged smoothly even under misalignment conditions of as much as 1/2 in.

The barrel of the fuel handling machine is carried by the trolley on cylindrical bar rails on the bridge. A large pipe serves as structural backbone for the bridge. This arrangement minimizes interference with other in-cell components. The bridge itself rolls on rails like the ones of the trolley. Movements of trolley and bridge are normal to one another. Thus the position of the gripper over the reactor grid, fuel port, wet fuel storage tank or the subassembly cutoff machine is controlled by the "X-Y" movements of bridge and trolley. An eddy current drive assembly propels the bridge. A similar unit operates the trolley. Positioning of the machine over a specific location is done by manually operated controls from outside the cell. Position indication is derived from a stationary rack and readout devices are provided for achieving the desired accuracy of position.

A closed-loop, 3-kW, argon, circulating system is provided for cooling a fuel subassembly during all transfer steps. The compressor, filters, vapor traps, gas purifier and the heat exchanger are located underneath the cell in an extension of the fuel and equipment transfer tunnel. The vapor traps and the argon gas returned to the compressor are cooled by refrigerant from the cell atmosphere cooling system.

The machine operates in the following manner to pick up a subassembly from the reactor:

1. After the vessel cover is raised, the machine is located over the selected subassembly.
2. The barrel and gripper assembly is lowered into the open reactor vessel and over the subassembly with the barrel assembly eventually resting on the surrounding six subassemblies.
3. The gripper is lowered and engaged with the upper adapter of the subassembly.
4. As soon as complete engagement occurs, the gripper mechanism is mechanically locked. (Should the gripper fail to lock, it would not be able to raise a subassembly but would simply disengage again when elevating.)
5. The subassembly is raised into the barrel assembly, passing through the two compliant shutters at the lower end of the barrel.
6. When the subassembly is completely within the barrel and the shutters have engaged the lower adapter, argon circulation through the subassembly begins. At this time, the barrel with the subassembly inside is raised until it clears the upper flange of the reactor vessel, and moved to the wet fuel storage tank or to any other desired location in the cell.

Irradiated fuel elements handled by this system are cooled during transfer operations in the fuel handling machine by forced argon convection. This medium was selected because there will be no significant change in the composition of the cell atmosphere or of the coolant should some interchange of gas between atmosphere and coolant circuit occur. The coolant circuit is a closed, isolated loop to minimize spread of sodium vapors, radioactive contamination, sodium oxide, etc., in the cell.

A gas cooling and purification system is used in which heat from the (argon) gas is rejected by direct contact with NaK through which the gas is bubbled. The NaK in turn is kept at a low temperature ($\sim 45^{\circ}\text{F}$) by a simple ammonia cooling coil which is immersed in it. Thus the gas is cooled and at the same time Na vapors are condensed and washed from the gas by the liquid NaK, instead of using a conventional trapping on wire mesh or other solid surfaces.

The composition of the bath may vary widely, starting clean with a near eutectic composition. It therefore has to be replaced or reconditioned when the Na content has reached approximately 50% due to adding sodium to the mixture by condensation and dissolution of vapor from the gas stream.

The cooling unit together with associated blowers, coolers, heaters, and filters, as well as valves and other controls are located outside the cell for easier access.

2. Wet Fuel Storage

a. Storage Tank

To reduce the problems of handling and shipping intensely radioactive and thermally hot fuel, a sodium-filled, double-walled tank is provided for storage of spent subassemblies until the fission product activity and the heat generation decays to handling levels. The tank is shielded and sealed by a rotatable cover and is cooled by eutectic NaK that is circulated in the tank wall annulus. An external electric heater on the NaK piping system keeps the tank sodium molten when sufficient decay heat is not present. Figure 53 shows a schematic of the NaK system.

The tank has storage positions for 151 fuel subassemblies in a configuration that will prevent criticality regardless of type or number of fuel subassemblies and whether or not sodium is present. The reactor and storage tank are nuclearly isolated from each other by concrete and also by shielding inside the reactor vessel. The capacity was chosen based upon reactor core size, time at rated power, cooling time in the reactor, and dead time between shutdown of one core and startup of another.

The storage tank is double-walled to prevent inadvertent loss of sodium, which could lead to fuel meltdown. Also, there are no piping connections to the sodium volume below the top of the fuel section of the subassembly. To drain the tank, a cap must be installed manually on an antisiphon gooseneck on the sodium inlet line. Level gages and high and low level alarms on the sodium and NaK surge tanks indicate changes in coolant levels.

Both the wet fuel storage tank and its NaK-filled annular tank are rated at an internal pressure of 30 psig and an external pressure of 15 psig. The vessel is rated at 600°F, which is the calculated maximum coolant temperature for removal of 80 kW of heat by natural circulation.

The auxiliary primary system provides sodium makeup to the storage tank. When the added sodium reaches the operating level in the storage tank it overflows into the primary drain tank. This overflow procedure permits removal of sodium for purification. It also permits cooling the storage tank in emergencies by routing precooled sodium from the auxiliary primary system.

b. Storage Tank Cover

The cover of the wet fuel storage tank provides gamma shielding, a barrier between the cell and tank atmospheres during nonfuel handling operations, and access to individual subassembly storage grid locations in the tank (Fig. 54). By rotating the cover, the operator can position one of two access ports over any one of the 151 storage positions in the grid. After the cell crane removes a shield plug from the gasketed access port, the fuel handling machine is used to insert or remove core subassemblies as required.

The cover mounts on a single-row angular-contact ball thrust bearing and is rotated at about 1/4 rpm by a reversible ratio motor with an integral braking device. The drive gear pinion, which engages a large gear on the cover, is fitted with a torque limiting clutch (mounted on the output shaft of the ratio motor) that prevents damage to the drive motor if the cover fails to rotate. During cover rotation, a position decoder in mesh with the cover gear transmits the cover position to a readout instrument at the fuel handling console. In addition, the operator can look through the cell window to observe position as indicated by a stationary pointer and a graduated index ring on the cover.

When fuel is being handled the cell atmosphere is maintained compatible with sodium at fuel handling temperatures. But when the cover is stationary, two inflatable seals isolate the storage tank argon gas blanket from the cell atmosphere. The inflatable seals, each of which has an independent argon pressurizing system, are made of high strength fabric--coated with silicone and molded into a continuous loop. When the

seals are inflated, the pressure forces the seal striker face against the circular striker surface on the cover, forming a gastight seal; the annulus between the two inflatable seals can be pressurized periodically to determine seal adequacy. Argon gas for the two seals is supplied through independent systems from the main argon (100 psig) line; the operator adjusts each system to proper pressure by means of regulating valves at the fuel handling console. The seals are depressurized to permit cover rotation and the argon is exhausted to the radioactive vent system. To replace the inflatable seals or the cover support bearing, the lower shielding section of the cover can be detached and lowered onto a compression gasket to form a gas seal with the storage tank.

3. Subassembly Cutoff and Scrap Handling

a. Subassembly Cutoff

The subassembly cutoff machine has been designed for the purpose of cutting the all welded 178-in. long fuel subassemblies to a 96-in. long fuel-containing section that allows for transfer out of the cell by means of the EBR-II Interbuilding Coffin and subsequent transfer to the EBR-II Fuel Cycle Facility for further dismantling. Also, scrap sections are cut to less than 65 in. to fit into containers that will be used with the EBR-II Waste Transfer Cask for subsequent disposal at the EBR-II radioactive waste burial ground. All nonfuel bearing subassemblies are cut into lengths compatible with scrap handling.

The subassembly cutoff machine (Fig. 55) is located in a 20-ft deep floor pit near the reactor and wet fuel storage tank, and in front of an east cell wall window.

The fuel handling machine is used to place a subassembly into the cutoff machine. For fuel subassemblies, the transfer will probably be from the wet fuel storage tank to allow for fission product decay. The maximum heat generation rate for the cutoff machine is 1000 W and corresponds to that of the EBR-II Interbuilding Coffin. The fuel handling machine also transfers the fuel bearing section of the subassembly from the cutoff machine to the transfer coffin after the subassembly has been removed.

Core components, other than fuel bearing, may be transferred directly from the reactor to the cutoff machine and cut into scrap lengths since heat generation is not a problem. The cutoff portion of the fuel subassembly, other than fuel, and cut-up core subassemblies are handled by an integrally mounted manipulative device for transfer out of the cutoff machine and into adjacent scrap containers.

The procedure used with an instrumented fuel subassembly is as follows:

1. The fuel handling machine picks up the subassembly from within the wet fuel storage tank and transfers it to the subassembly cutoff machine where it is lowered through the hexagon sleeve (conforming with the hexagon fuel subassembly) in the rotational drive unit and deposited on the machine elevating platform.
2. The cutoff machine cooling system is started and the fuel handling machine can be removed.
3. The elevating platform lifts the subassembly to the cutting position.
4. The subassembly is rotated and the outer hexagon tubing is cut by use of roller type cutters.
5. The subassembly is raised to the instrument lead cut-off position.
6. The integral manipulative device clamps on to the top severed section of the subassembly and elevates it enough to allow the instrument lead snipper to complete the separation of the top end of the fuel subassembly. The top end is scrap and transferred to the adjacent scrap containers by the manipulative device.
7. The fuel handling machine is returned to the cutoff machine and transfers the fuel section to the EBR-II Interbuilding Coffin using the integral fuel handling lifting fixture at the cutoff position.

Fuel subassemblies without instrument leads do not require the snipping operation; core subassemblies not containing fuel do not require the transfer operation to the shipping cask since all sections are scrap.

b. Scrap Handling

Except for core components manipulated by the fuel handling machine, material and equipment are transferred within the cell primarily by the 25/5 ton bridge crane. The crane is used to install equipment inside the cell and reactor, replace equipment modules as a means of remote or expedient maintenance, transport containers (e.g., scrap), support tools and fixtures for standard and improvised operations, uncover pits and ports, move shield sections (including in-cell window slabs) and generally provide for handling functions that are required to operate the facility.

The 25-ton hoist on the crane moves the equipment transfer port cover, wet fuel storage cover, reactor vessel head cover, and large

reactor vessel internal components. The 5-ton hoist transfers the fuel transfer port plugs, wet fuel storage tank access plugs, radioactive scrap storage hole plugs, material containers, and miscellaneous machinery or parts of equipment. The crane hoists are equipped with hooks or grapples to accept the particular fixtures necessary for various types of equipment. The 25-ton hoist has a standard hook to engage eye bolt type fixtures. The 5-ton hoist has a special hook that accepts different types of lifting fixtures, e.g., a T-bar type or the EBR-II subassembly top fixture type as shown in Fig. 55.

Scrap containers for both radioactive and nonradioactive waste are slightly modified versions of the 12-in. dia by 12-in. long EBR-II waste containers. This permits the EBR-II Waste Transfer Cask to be used for transfer to the burial site. These containers are flat bottom cylindrical cans closed and sealed with a standard 5-gal pail 16 crimp lid. Containers that are filled with radioactive scrap and sealed are stored in scrap storage pits in the cell floor until transferred from the cell to the EBR-II Waste Transfer Cask. Nonradioactive scrap is stored in cans at various locations on the cell floor until it is convenient to remove them from the cell through the personnel access lock or with the waste transfer cask.

4. Transfer System for Fuel and Equipment

To move fuel and other materials in and out of the reactor cell the operator uses two concentric floor ports at the south end of the cell. These ports open into the ceiling of the transfer tunnel beneath the reactor cell. Figure 56 shows the transfer system, tunnel, and ports. The fuel transfer port is 14-in. in diameter and the equipment port is a $9 \times 8\frac{1}{4}$ -ft oval.

The system operates in the following manner:

1. The empty spent fuel transfer cask (EBR-II Interbuilding Coffin) is placed by the building crane on a support frame which is supported by the transfer carriage.
2. The carriage transfers the cask and support frame to a position directly beneath the transfer port over the hydraulic hoist.
3. The hoist raises the support frame and cask until the fuel port-to-cask seal engagement is made.
4. The screw jack supports are raised and locked in place to support the load. (The hoist hydraulic pressure is then released.)
5. The cask air atmosphere is purged with argon gas.

6. Concurrently the port-to-cask seal is pressure tested to ascertain leak tightness.
7. The fuel transfer port covers are removed by the in-cell crane.
8. The fuel handling machine deposits a fuel assembly in the cask.
9. The fuel port covers are replaced and their seals pressure tested for leak tightness.
10. The above procedure is reversed to allow for disposition of the spent fuel subassembly.

This same procedure is used for transfer of the new fuel container, waste cask, small equipment container, and other containers deemed desirable for use with the small fuel transfer port.

The large equipment transfer port is sealed by bolting a large container to the gasketed flange after the equipment port shield plug is removed. Handling of the container, purging, and seal testing is effected in the same manner as described above.

VII. PRINCIPAL HEAT TRANSFER SYSTEMS

A. System Scope and Design

The reactor is cooled by two systems. The normal cooling of the reactor is provided by the main cooling system which is capable of dissipating 100 percent of the reactor power (50 MW) with a coolant outlet temperature from the reactor vessel of 790°F. Shutdown cooling (1 MW) of the reactor is effected by the auxiliary cooling system. Each system is comprised of primary and secondary heat transport loops and the heat is dissipated to the atmosphere via the secondary air blast-type heat exchangers.

Sodium is the heat transfer medium for the main primary and secondary reactor cooling systems, the auxiliary primary and secondary cooling systems, and the fuel storage cooling system. NaK is used in the reactor annulus cooling system, in the secondary loop of the wet fuel storage tank, and in the cooling system for the coolant purification systems. Argon serves as the inert blanket gas in all liquid metal systems.

Figures 16, 57, and 58 show the general interrelation between the auxiliary primary and the main primary loops. The main sodium stream leaves the reactor vessel and flows through the shell of the main IHX (5) and then to the main primary pump (4). The flow then enters the high pressure plenum of the core support cylinder, from where it passes through the core and out of the vessel. Under normal operating conditions, the auxiliary primary sodium flows from the vessel sodium annulus into a circumferential header, then to the auxiliary IHX (7), and then to the auxiliary primary pump (8). The discharge head of the main primary pump (4) is such that the discharge head from the auxiliary primary pump cannot open the auxiliary primary check valve when the main pump is operating. Thus, the auxiliary primary sodium bypasses the main primary sodium inlet pipe (and hence the core) and enters the vessel sodium annulus via an orifice.

Under scram conditions not caused by a loss of primary flow, the primary sodium flow rate is reduced to about 10 percent, which is adequate to cool the reactor. The auxiliary system remains on low flow standby.

Under emergency operating conditions, i.e., loss of main primary flow, power to the auxiliary primary pump (8) automatically switches from low to high. As the main primary sodium stream decelerates, the auxiliary primary loop sodium begins to accelerate, so that at some point in time after the scram, the check valve begins to open and the auxiliary primary sodium flows into the main primary piping and through the core. The flow leaves the core and enters the vessel sodium annulus through holes in the core support cylinder, mixes with the lower temperature sodium in the annulus, and leaves the vessel through the auxiliary system outlet header.

From the junction with the auxiliary primary system, the main primary inlet pipe to the reactor vessel is doubly contained. The entire auxiliary primary loop is doubly contained. The relative head of sodium at the highest elevation at which the main primary loop is doubly contained is greater than the discharge head of the auxiliary primary pump. Thus, it is impossible to drain the auxiliary primary pump, and it is impossible to drain the auxiliary primary loop through a break in a single pipe section of the main primary loop.

A shutdown pump is included in the main primary loop. This pump provides flow in the event that the main primary pump fails, whereas the rest of the system is still serviceable and available to dissipate reactor decay heat. The pump is an electromagnetic type and is always operating. On loss of power, it is energized from a floating battery.

B. Main System Components

1. Main Primary Sodium Pump

The main primary sodium pump is a vertical centrifugal single stage pump of the free surface type. This eliminates the need for liquid sodium seals. The surface of the liquid in this barrel is at the same level as the sodium in the reactor vessel.

This pump is designed to handle 3,100,000 lb/hr of sodium at 960°F (specific gravity 0.83). The suction pressure at these conditions is 8.9 psia and the discharge pressure is 118.9 psia; the net positive suction head available is 26.4 ft of sodium absolute. Under normal operating conditions the maximum shut-off head is 340 ft of sodium. The liquid metal exposed portions of this pump are constructed of Type 304 stainless steel. It is equipped with argon gas type shaft seals. The suction connection is 20 in., the discharge connection 14 in., the overflow connection if external is 6 in., and a 4-in. inert gas connection is provided.

The pump has an electric motor drive and variable speed coupling for control down to approximately 10 percent of full speed. Biological shielding provided within the pump barrel is cooled by circulating the argon gas atmosphere within the vault across cooling fins on the barrel surface. The motor and coupling are located above the vault in the rear cell area and are air-cooled.

2. Main Intermediate Exchanger

The main intermediate heat exchanger is a shell-and-tube design with a removable tube bundle. It is arranged for primary sodium in the shell and secondary sodium in the tubes. The exchanger is mounted vertically, entirely within the vault. Fluid flow paths are arranged for

natural convection flow. All materials for both shell and tubes are Type 304 stainless steel. The design conditions for this heat exchanger are: 75 psig, shell design pressure; 75 psig, tube design pressure; 1175°F, shell-and-tube design temperatures; 3 psi, overall pressure drop on the shell side; and 7 psi on the tube side (in both cases the minimum is to be no less than 75 percent of the maximum). Case I of the following table refers to highest temperature conditions at maximum load; Case II refers to the lowest temperature conditions at maximum load. Temperature differences between shell-side fluid and tube-side fluid are maximum normal values at the design flow rates indicated.

	Case I		Case II	
	Shell	Tube	Shell	Tube
Temperature In, °F	1150	870	790	510
Temperature Out, °F	960	1060	600	700
Heat Transfer (nominal), MW	50	50	50	50
Flow, 10 ⁶ lb/hr	2.99	2.97	2.92	2.89

3. Main Secondary Pump

The main secondary pump is a free surface pump similar to the main primary pump except that there is no requirement for shielding and the entire assembly is located in the rear cell area.

4. Main Air-cooled Exchanger

The main air-cooled heat exchanger is a finned-tube sodium-to-air exchanger located outside the reactor building. The unit includes structural steel supports, finned-tube coils, two half-capacity fans each complete with constant speed electric motors, variable speed couplings for control of fan speed, ductwork, dampers, and control devices. The unit is fully enclosed and insulated to minimize heat losses when operating in the system under isothermal conditions. Supplementary electric heating coils maintain sodium temperatures up to 600°F under all shutdown conditions.

This heat exchanger is designed for outside air conditions of -20°F minimum and 91°F maximum normal dry bulb temperatures and a 55°F maximum wet bulb temperature. Heat exchanger design pressure is 75 psig, inlet temperature is 1175°F, and overall maximum pressure drop is 20 psi.

The steady-state operating conditions provided for in the design of this heat exchanger are as follows:

	Case I	Case II
Temperature In, °F	1060	700
Temperature Out, °F	870	510
Heat Transfer (nominal), MW	50	50
Flow, 10 ⁶ lb/hr	2.97	2.89

Note that Case I refers to the highest temperature conditions at maximum load and Case II refers to the lowest temperature conditions at maximum load.

5. Piping and Tanks

All main primary and secondary piping is Type 304 stainless steel. The main primary cooling system is arranged within the confines of the vault to obtain the maximum practical natural convection cooling. On normal scram the main pumps reduce their flow in a period of time related to the closing time of the main air-cooled heat exchanger dampers, the main primary shutdown pump remains in operation, and the auxiliary primary system remains on a low-flow standby condition. On an abnormal scram such as loss of electrical power, the main primary shutdown pump continues to operate on limited battery power, and the auxiliary primary system pump increases to full capacity utilizing the emergency power source. The main primary shutdown pump has adequate capability to cool the reactor at any scram without any other supplemental cooling. Credit for natural convection cooling capability is not considered in any of the above. Although it is available, it cannot be considered adequate immediately after scram. However, natural convection cooling is adequate to cool the reactor after the first minute or so following scram. Vents and drains are installed where required; they are arranged with provisions against accidentally draining the reactor vessel.

All piping connected to the reactor vessel which, in event of a rupture would drain the sodium below core level, is double contained. Failure of any single wall sections of the main primary piping will not adversely affect the reactor auxiliary cooling system capability.

Provisions for heating the system to 600°F for isothermal operation without nuclear heating are made by the pipe and equipment heating system.

The primary system drain tank can contain the entire inventory of sodium in the primary systems or the maximum sodium inventory of the wet fuel storage tank, of which the former is greater. The secondary system drain tank can contain the entire inventory of sodium in either the main or the auxiliary secondary system.

C. Auxiliary System Components

1. Auxiliary Primary Pump

The pump is an electromagnetic pump, hermetically sealed, having no moving parts, and no direct electrical connections to the liquid metal-carrying components. Flow is controlled to either of two settings by varying the voltage supplied to the pump windings. The pump is cooled by circulating vault argon gas through the stator. All materials in direct contact with the sodium are Type 304 stainless steel.

This pump is designed to handle a maximum sodium flow of 196,000 lb/hr at 1050°F (specific gravity 0.83). The differential head across the pump under these conditions is 30-ft sodium. The pump design pressure and temperature are 100 psig and 1100°F. The pump is equipped with preheat elements to increase its temperature from 100 to 350°F in less than 150 hr with a maximum rate of 50°F/hr. The suction connection is 6 in., as is the discharge connection. This pump is designed to operate in an argon-gas atmosphere at temperatures of 100°F normal and 160°F maximum. There are three temperature detectors in the windings and the pump is double-walled throughout. The pump will operate from 240 volt, three-phase, 60 cycle power with a maximum specified energy input to the coolant of 28 kW. A maximum shut-off head of 90 ft of sodium at 1050°F is specified.

2. Auxiliary Intermediate Exchanger

The auxiliary intermediate heat exchanger is a vertical shell-and-tube unit with a removable tube bundle, similar to the main intermediate heat exchanger.

a. The heat exchanger design temperatures and pressures are:

	<u>Shell, Na</u>	<u>Tube, Na</u>	<u>Annulus, NaK</u>
Design Pressure, psig	75	75	50
Design Temperature, °F	1175	1175	1175
Overall Pressure Drop (max.), psi	5	5	*

*The NaK fluid is normally stagnant.

b. The operating design conditions are:

	Case I		Case II	
	Shell	Tube	Shell	Tube
Temperature In, °F	1018	870	656	510
Temperature Out, °F	960	931	600	570
Heat Transfer (nominal), MW	1	1	1	1
Flow, 10 ³ lb/hr	196	185	196	185

Case I refers to highest temperature conditions at maximum load; Case II refers to the lowest temperature conditions at maximum load. Indicated temperature differences between shell-side fluid and tube-side fluid are maximum allowable values at the design flow rates indicated. Mixing of the reactor outlet sodium with the sodium in the reactor vessel annulus reduces the thermal duty from the peak reactor cooling requirement of 3 MW.

3. Auxiliary Secondary Pump

The auxiliary secondary pump is an air-cooled electromagnetic pump. It has one normal flow value. This pump is designed to handle 185,000 lb/hr of sodium at 1050°F (specific gravity 0.83). The specified maximum shut-off head is 120 ft of sodium at 1050°F and the differential pressure developed by this pump is 60 ft of fluid at 1050°F. The design pressure is 50 psig and design temperature is 1100°F. The pump is made entirely of Type 304 stainless steel and is equipped with preheat elements. Its temperature rise is from 100 to 350°F in less than 150 hr at a maximum rate of 50°F/hr. The suction and discharge connections are 6 in. The pump is designed to operate in air at an ambient temperature of 100°F. It is equipped with three temperature detector windings and operates from 480 volt, three-phase, 60 cycle power. The specified maximum energy input to the coolant is 36 kW.

4. Auxiliary Secondary Exchanger

The auxiliary secondary sodium air-cooled heat exchanger is a finned-tube natural convection unit located outside the reactor building. It is designed for outside conditions of -20°F minimum and 91°F maximum normal dry bulb temperatures and 55°F maximum wet bulb temperature. In addition, the following design criteria apply to this heat exchanger (Case I and II are as previously described):

Design Pressure, psig	75
Design Temperature, °F	1100
Overall Pressure Drop (max.), psi	5

	<u>Case I</u>	<u>Case II</u>
Temperature In, °F	948	566
Temperature Out, °F	852	506
Heat Transfer (normal), MW	1.6	1
Flow, 10 ³ lb/hr	185	185

5. Miscellaneous

All auxiliary primary and secondary piping is of Type 304 stainless steel. The auxiliary primary system can be drained into the main primary drain tank with the main primary system. Purification is concurrent with the main system. The auxiliary secondary system is filled with sodium and provision for routine draining is included.

D. Wet Fuel Storage Tank and Cold Trap Heat Removal Systems

1. Storage Tank

The wet fuel storage tank system schematically shown in Fig. 53 consists mainly of a stainless steel (Type 304) vessel filled with sodium and a NaK system. The sodium-filled vessel is used for temporary storage of up to 151 irradiated fuel subassemblies prior to their removal from the facility for examination or processing. The NaK system is used for temperature control of the sodium contained in the tank.

Irradiated fuel subassemblies freshly removed from the reactor can have significant decay heat generation and high radiation levels so that the handling of such elements in transport casks would be extremely difficult and rather impractical. Therefore, the storage tank is provided to allow decay until the heat and radiation levels are at reasonable levels.

An annulus encompassing the tank is filled with eutectic NaK and connected to an air-cooled heat exchanger with piping and a pump. When heat is rejected from decaying fuel within the tank, it will be transferred to the circulating NaK for rejection to the atmosphere via the air-cooled heat exchanger. The NaK system is provided with an electric heater to permit addition of heat to the sodium within the vessel during periods when there are an insufficient number of decaying elements present to keep the sodium molten.

The configuration of stored fuel within the tank is such that criticality will not be possible regardless of type or number of elements stored or whether or not sodium is present. However, the decay heat could lead to fuel meltdown if sodium were not present. To preclude inadvertent sodium loss, the NaK cooling annulus is designed to serve as double containment for the vessel to avoid sodium loss through vessel failure.

The wet fuel storage tank has nuclear isolation from the reactor vessel due to the reactor internal shielding and the concrete between the two vessels.

It will be impossible to drain the tank unless a cap is manually installed on a built-in siphon breaker. The loss of sodium or NaK from the systems will be detectable on level indicators in the expansion tanks for each system. Each level indicator is provided with high and low level alarms.

The total heat rejection capability of the wet fuel storage tank is 80 kW. If an entire freshly irradiated core loading is placed in the vessel, a suitable short decay period will be necessary in the reactor before complete transfer to the storage tank is effected.

The wet fuel storage tank and its NaK filled annulus each have internal and external pressure ratings of 30 and 15 psi, respectively. The internal rating is based on the cell containment design pressure so that the same pressures up to 30 psi can be tolerated. The external pressure rating is based on internal pressures which could be anticipated if total evacuation would be prescribed for prefiling operations. The vessel temperature rating of 600°F corresponds to the maximum temperature which could be anticipated when a full heat load is applied concurrent with the loss of the NaK circulation pump when natural NaK circulation will be utilized for heat rejection.

Sodium makeup to the vessel is from the auxiliary primary system. Sodium, when added from this source, will overflow to the primary drain tank after the storage tank has reached the operating level. This feature permits displacement of contained sodium for subsequent purification via the cold traps. Further, emergency cooling of the storage tank can be performed by routing precooled sodium from the reactor flange cooling system with continuous overflow as will be done for purification.

2. Cold Trap Cooling

The NaK cooling system for the wet fuel storage tank is integral with the cooling system for the sodium purification system cold traps. Valving in the NaK piping permits heat rejection from either or both of the two systems. In the event of power outage NaK will flow through the wet fuel storage tank annulus and the air-cooled heat exchanger by natural convection. Under these conditions the sodium temperature in the wet fuel storage tank will not exceed 600°F.

The air-cooled heat exchanger is designed for both natural convection and forced circulation of ambient air. The forced circulation mode permits the lower NaK temperatures that are required for operation of the

cold traps. NaK temperatures leaving the heat exchanger are controlled by controlling the air flow through the unit. During natural circulation cooling of the fuel storage tank, air flow dampers control the sodium temperature out of the heat exchanger and consequently maintain proper temperatures in the wet fuel storage tank. During this mode of operation the natural convection flow of NaK to the storage tank bypasses the cold traps. However, during cold trap operation the NaK temperatures out of the heat exchanger are reduced to 250°F as required for cold trap operation, and forced convection of both air and NaK is employed. The heat exchanger is provided with electrical heating during extremely cold weather.

E. Liquid Metal Purification

1. General

The primary objective of the liquid metal purification systems is control of oxygen at or below the permissible levels for operation. The primary and secondary sodium systems are provided with separate recirculating cold traps. NaK systems are provided with natural circulation diffusion-type cold traps. The primary purification system is also used for purifying the sodium in the wet fuel storage tank.

Provisions are made for the future installation of one additional cold trap and two hot traps for the main primary system and a hot trap for the main secondary loop.

Maximum design limits of oxygen (present as Na_2O) for the sodium systems are 10 ppm for the secondary and the primary systems. A limit of 5 ppm is an anticipated lower capability level. The cold traps are capable of reducing oxygen concentration from 30 to 10 ppm in about 24 hr. The diffusion-type cold traps in the NaK systems are designed to maintain the oxygen levels below 10 ppm.

2. Primary Sodium Purification System

Two individually shielded regions are provided in the vault to contain one primary cold trap initially installed and one future primary cold trap. These units may become contaminated from fission product accumulation during service. The design will permit one set of units to be serviced while the other set operates.

Primary coolant to be purified is pumped from the primary drain tank through the purification system, to the reactor flange cooling heat exchanger, and to the main primary coolant stream in the reactor vessel via the reactor make-up inlet line. The flow enters the cold trap from the economizer. The sodium temperature is reduced to about 250°F in the trap where the precipitated oxides are trapped in a wire mesh. The coolant is then reheated in the economizer and returned to the main stream.

The cold trap system is designed to operate over a range of NaK coolant inlet temperatures from 200 to 350°F. The cold trap is designed for a sodium flow rate of 30 gpm. The expected oxide capacity of the cold trap is 200 lb.

A manually controlled plugging meter in the primary system determines oxide content; it is cooled by vault argon gas. Two sampling connections, one upstream and one downstream of the purification system, are provided.

3. Secondary Sodium Purification System

The secondary sodium purification system, which is similar to the primary system, is located in the rear cell area. This system has an ambient air environment and is always fully accessible. The secondary sodium purification system serves both the main and auxiliary secondary systems. Continuous purification takes place by pumping sodium from the drain tank through the cold trap, and to the main and auxiliary systems. Sodium returns to the drain tank via the overflow in the expansion tanks.

4. Initial Purification

The sodium is purified initially by mechanical filtration through a strainer placed in the temporary fill lines. The sodium is further purified by circulating through the primary or secondary cold traps during filling of the primary and secondary systems.

F. NaK Annulus System

The reactor vessel, the entire primary portion of the auxiliary loop and portions of the main reactor inlet piping have an annulus system filled with NaK. This NaK annulus system:

- a. Prevents loss of sodium from the core region through failure of the reactor vessel walls or of any piping that could act as a siphon.
- b. Detects sodium leakage from any of the aforementioned regions (and from the NaK system) so that orderly remedial action can be taken.
- c. Transfers nuclear-generated heat from the outer vessel wall to the sodium in the vessel, and
- d. Serves for initially preheating the reactor vessel before filling with sodium.

During normal reactor operation, the NaK in the system is stagnant. The NaK system (Fig. 53) has a heater and a circulating pump for occasional reactor vessel preheating--and for maintaining the vessel at about 300°F during periods when all other sodium-circulating systems are inactive. This eliminates the need for electrical heaters in the reactor vessel. Because of the presence of NaK, the temperature differential between the inner and outer vessel walls will be minimal. Heat generated from gamma captures in the outer vessel wall will be transferred by the NaK to the sodium inside the reactor vessel.

Leakage--either loss of NaK to the sodium, loss of NaK to the surrounding environs, or a gain of sodium in the NaK system--will be detected as a level change in the NaK expansion tank. When the expansion tank sodium level changes (~100 gal equivalent), an alarm registers, requiring an orderly shutdown of the reactor and an appraisal of the situation. If a gain or loss of sodium occurs to such a magnitude that the NaK annulus expansion tank level approaches the point of overflow or to an abnormally low reactor level, the reactor scrams automatically. An abnormally low level in the NaK expansion tank, resulting from loss of NaK to the sodium or surroundings, will also cause a reactor scram. To provide level monitoring confidence, the alarm and scram level monitors will be independent instruments.

The inert gas in the NaK expansion tank and the NaK annulus drain tank (to which any overflow will occur) are interconnected; normal gas supply or vent provisions prevent excess pressures in the NaK annulus system. The calculated pressures in the NaK system and the associated sodium system at various locations during normal operation are shown in Fig. 59.

The reactor inlet piping will be subjected to the pump discharge pressure (~85 psig). Since the associated guard piping is connected directly to the reactor vessel annulus, steps have been taken to preclude the pressurization of the vessel annulus due to sodium piping failure through the use of restrictions where the guard piping joins the annulus. These restrictions are sized to limit the pressure buildup in the annulus to below the external design pressure rating of the vessel (15 psig). Sodium leakage into the annulus will be directed to the NaK expansion tank where it will overflow to the NaK drain tank. The NaK piping which joins the annulus to the expansion and drain tank is traced with resistance heating to permit the free flow of sodium. In such an event, flow in the main primary system would be reduced to minimize or stop the leakage (by reducing the main pump discharge head) until the condition could be assessed.

All of the NaK annulus system is fabricated of Type 304 stainless steel by procedures identical to those for the primary sodium system. Design pressure and temperature ratings are consistent with credible operating and accident conditions. Inventory adjustment to the NaK system

(to and from its drain tank) are made by use of remotely operated valves. These valves, as well as several other valves in the system, are of the same bellows-sealed type as those used in the sodium systems.

Although little NaK oxidation is anticipated because the NaK system is closed and its tanks blanketed with argon, any oxidation products will be removed with a built-in diffusion cold trap. Periodic forced circulation of the NaK displaces NaK from regions which normally are stagnant.

G. Transient Design

The design of FARET has given careful attention to the temperature transients which can occur in the liquid metal systems. Of particular importance are the major liquid metal systems located in the vault, where space is at a premium. These systems are designed to operate for long periods of time without maintenance.

Tables 12 thru 21 list transient temperature and flow conditions in some of the major liquid metal system components. This material is arranged to emphasize two kinds of conditions--namely, normal operational

TABLE 12. Main Intermediate Heat Exchanger, X-1, Transient Operating Conditions

Source of Transient	Max. Rate of Temp. Change	Amplitude of Temp. Transient	No. of Occurrences	Temp. at Start of Transient
Initial dry heatup ¹	To be set by detail design	+280°F	50	70°F isothermal
Normal heatup ²	+50°F/hr	+350°F	1000	250°F isothermal
Nuclear heatup ³	+50°F/hr	+450°F	250	600°F isothermal
Normal cooldown ⁴	-50°F/hr at tube-side inlet	-350°F	1000	600°F isothermal
Temperature ramp ⁵	a) +50°F/hr at tube-side inlet	+360°F	3500	510°F at tube-side inlet; 790°F at shell-side inlet
	b) -50°F/hr at tube-side inlet	-360°F	3500	
Power ramp ⁶	a) +50°F/hr at shell-side inlet	+190°F	1000	600°F isothermal
	b) +50°F/hr at shell-side inlet	+190°F	1000	960°F isothermal
	c) -50°F/hr at shell-side inlet	-190°F	1000	790°F
	d) -50°F/hr at shell-side inlet	-190°F	1000	1150°F
Emergency shutdown (forced convection in main loop) ⁷	See Fig. 60		400	1150°F at shell-side inlet; 870°F at tube-side inlet
Emergency shutdown (loss of electric power) ⁸	See Fig. 61		50	1150°F at shell-side inlet; 870°F at tube-side inlet

¹Heatup within 150 hr by external electrical heaters; exchanger filled with argon.

²Heatup within 24 hr by external electrical heaters with circulating coolant.

³Heat supplied by low power reactor operation with or without electrical heating and with circulating coolant.

⁴Cooldown with circulating coolant; heat dissipated at X-2.

⁵Tube-side coolant inlet temperature is increased or decreased at 50°F/hr while at full power by varying X-2 air flow; exchanger continues to operate at design duty; flow rates are held constant.

⁶Power is ramped from isothermal operation to full ΔT (190°F) operation (or vice versa) with maximum temperature rate-of-change of 50°F/hr.

⁷Due to system malfunction, reactor power suddenly drops from 50 MW to decay power.

⁸Due to loss of electric power, reactor power suddenly drops from 50 MW to decay power.

TABLE 13. Main Air-cooled Heat Exchanger, X-2, Transient Operating Conditions

Source of Transient	Max. Rate of Temp. Change	Amplitude of Temp. Transient	No. of Occurrences	Temp. at Start of Transient
Initial dry heatup ¹	To be set by detail design	+280°F	50	70°F isothermal
Normal heatup ²	+50°F/hr at sodium inlet	+350°F	1000	250°F isothermal
Nuclear heatup ³	+50°F/hr	+450°F	250	600°F isothermal
Normal cooldown ⁴	-50°F/hr at sodium outlet	-350°F	1000	600°F isothermal
Temperature ramp ⁵	a) +50°F/hr at sodium inlet	+360°F	3500	700°F
	b) -50°F/hr at sodium inlet	-360°F	3500	1060°F
Power ramp ⁶	a) +26°F/hr at sodium inlet	+100°F	1000	600°F isothermal
	b) +26°F/hr at sodium inlet	+100°F	1000	960°F isothermal
	c) -26°F/hr at sodium inlet	-100°F	1000	700°F
	d) -26°F/hr at sodium inlet	-100°F	1000	1060°F
Emergency shutdown (forced convection in main loop) ⁷	See Fig. 62		400	1060°F sodium inlet
Emergency shutdown (loss of electric power) ⁸	See Fig. 63		50	1060°F sodium inlet

¹Heatup within 150 hr by external electrical heaters; tubes filled with argon.²Heatup within 24 hr by external electrical heaters with circulating coolant.³Heat supplied by low power reactor operation with or without electrical heating and with circulating coolant.⁴Cooldown with circulating coolant.⁵Temperature is increased or decreased at 50°F/hr while at full power by varying heat exchanger air flow; sodium flow is held constant.⁶Power is ramped from isothermal operation to full ΔT (190°F) operation (or vice versa) with maximum rate-of-change of 26°F/hr.⁷Due to system malfunction, reactor power suddenly drops from 50 MW to decay power; blower power tripped off.⁸Due to loss of electric power, reactor power suddenly drops from 50 MW to decay power; blower power tripped off.

TABLE 14. AUXILIARY INTERMEDIATE HEAT EXCHANGER, X-3, Transient Operating Conditions

Source of Transient	Max. Rate of Temp. Change	Amplitude of Temp. Transient	No. of Occurrences	Temp. at Start of Transient
Initial dry heatup ¹	To be set by detail design	+280°F	50	70°F isothermal
Normal heatup ²	+50°F/hr	+350°F	1000	250°F isothermal
Nuclear heatup ³	+50°F/hr	+450°F	250	600°F isothermal
Normal cooldown ⁴	-50°F/hr	-350°F	1000	600°F isothermal
Temperature ramp ⁵	a) +50°F/hr at shell-side inlet	+360°F	3500	600°F
	b) -50°F/hr at shell-side inlet	-360°F	3500	960°F
Auxiliary loop startup ⁶	See Fig. 64		200	946°F primary inlet; 970°F secondary inlet

¹Heatup within 150 hr by external electrical heaters; exchanger filled with argon.²Heatup within 24 hr by external electrical heaters with circulating coolant; exchanger jacket annulus filled with NaK.³Heat supplied by low power reactor operation with or without electrical heating and with circulating coolant.⁴Cooldown with circulating coolant; heat dissipated at X-4.⁵Shell-side coolant inlet temperature is increased or decreased at 50°F/hr while maintaining constant tube-side flow of about 52 lb/sec and shell-side flow of approximately 2.3 lb/sec.⁶Auxiliary loop to remove reactor decay heat upon loss of main coolant loop; flange cooling system operating. In addition to transient conditions given by Fig. 64, the heat exchanger shall be designed for a temperature step change of +20°F in the primary sodium inlet followed by a step change of +20°F in the secondary sodium inlet; these temperature changes will occur within the first 3 min after a scram from low power plant operation.

TABLE 15. Auxiliary Air-cooled Heat Exchanger, X-4, Transient Operating Conditions

Source of Transient	Max. Rate of Temp. Change	Amplitude of Temp. Transient	No. of Occurrences	Temp. at Start of Transient
Initial dry heatup ¹	To be set by detail design	+280°F	50	70°F isothermal
Normal heatup ²	+50°F/hr	+350°F	1000	250°F isothermal
Nuclear heatup ³	+50°F/hr	+450°F	250	600°F isothermal
Normal cooldown ⁴	-50°F/hr	-350°F	1000	600°F isothermal
Temperature ramp ⁵	a) +50°F/hr at sodium inlet	+360°F	3500	600°F
	b) -50°F/hr at sodium inlet	-360°F	3500	960°F
Auxiliary loop startup ⁶	See Fig. 65		200	960°F at sodium inlet; 933°F at sodium outlet

¹Heatup within 150 hr by external electrical heaters; exchanger filled with argon.

²Heatup within 24 hr by external electrical heaters with circulating coolant.

³Heat supplied by low power reactor operation with or without electrical heating and with circulating coolant.

⁴Cooldown with circulating coolant.

⁵Sodium temperature is increased or decreased at 50°F/hr while at full power by varying heat sink in the main coolant loop; sodium flow rate is held constant.

⁶Auxiliary loop to remove reactor decay heat upon loss of main coolant loop; sodium flow and air inlet temperature remain constant; flange cooling system operating. In addition to transient conditions given by Fig. 65, the heat exchanger shall be designed for a temperature step change of +20°F in the sodium inlet; this temperature change will occur within the first 3 min after a scram from low power plant operation.

TABLE 16. Reactor Flange Cooling Heat Exchanger, X-43, Transient Operating Conditions

Source of Transient	Max. Rate of Temp. Change	Amplitude of Temp. Transient	No. of Occurrences	Temp. at Start of Transient
Initial dry heatup ¹	To be set by detail design	+280°F	50	70°F isothermal
Normal heatup ²	+50°F/hr	+350°F	1000	250°F isothermal
Nuclear heatup ³	+50°F/hr	+450°F	250	600°F isothermal
Normal cooldown ⁴	-50°F/hr	-350°F	1000	600°F isothermal
Temperature ramp ⁵	a) +50°F/hr at shell-side inlet	+355°F	3500	687°F
	b) -50°F/hr at shell-side inlet	-355°F	3500	1050°F
Auxiliary loop startup ⁶	See Fig. 66		200	933°F at tube-side inlet; 1050°F at shell-side inlet

¹Heatup within 150 hr by external electrical heaters; exchanger filled with argon.

²Heatup within 24 hr by external electrical heaters with circulating coolant; exchanger jacket annulus filled with NaK.

³Heat supplied by low power reactor operation with or without electrical heating and with circulating coolant.

⁴Cooldown with circulating coolant.

⁵Shell-side coolant inlet temperature is increased or decreased at 50°F/hr while maintaining constant tube-side flow of about 52 lb/sec and shell-side flow of approximately 11 lb/sec.

⁶Auxiliary loop startup due to loss of main coolant loop; flows on both shell and tube sides remain constant. In addition to the transient conditions given by Fig. 66, the heat exchanger shall be designed for a temperature step change of +20°F in the secondary (tube side) inlet; this step change will occur within the first 3 min after a scram from low power plant operation.

TABLE 17. Main Primary Pump, P-1, Transient Operating Conditions

Source of Transient	Max. Rate of Temp. Change	Amplitude of Temp. Transient	No. of Occurrences	Temp. at Start of Transient
Initial dry heatup ¹	To be set by detail design	+280°F	50	70°F isothermal
Normal heatup ²	+50°F/hr	+350°F	1000	250°F isothermal
Nuclear heatup ³	+50°F/hr	+450°F	250	600°F isothermal
Normal cooldown ⁴	-50°F/hr	-350°F	1000	600°F isothermal
Temperature ramp ⁵	+50°F/hr -50°F/hr	+360°F -360°F	3500 3500	600°F 960°F
Emergency shutdown (forced convection in main loop) ⁶	See Fig. 67		400	960°F
Emergency shutdown (loss of electric power) ⁷	See Fig. 68		50	960°F

¹Heatup within 48 hr by external electrical heaters; pump filled with argon.

²Heatup within 24 hr by external electrical heaters with circulating coolant.

³Heat supplied by low power reactor operation without electric heating and with the coolant circulating.

⁴Cooldown with the coolant circulating; the heat is dissipated at X-2.

⁵Coolant temperature is increased or decreased at 50°F/hr while at full power and full sodium flow by varying the air flow through X-2.

⁶Step change in reactor power from 50 MW to decay heat; flow cutback to 10% of full flow.

⁷Due to loss of electrical power, reactor power suddenly drops from 50 MW to decay power; forced circulation in main primary loop is maintained by pump P-35 on emergency power and main secondary loop circulation is maintained by natural convection.

TABLE 18. Main Secondary Pump, P-2, Transient Operating Conditions

Source of Transient	Max. Rate of Temp. Change	Amplitude of Temp. Transient	No. of Occurrences	Temp. at Start of Transient
Dry heatup ¹	To be set by detail design	+280°F	50	70°F isothermal
Normal heatup ²	+50°F/hr	+350°F	1000	250°F isothermal
Nuclear heatup ³	+50°F/hr	+450°F	250	600°F isothermal
Normal cooldown ⁴	-50°F/hr	-350°F	1000	600°F isothermal
Temperature ramp ⁵	a) +50°F/hr b) -50°F/hr	+360°F -360°F	3500 3500	510°F 870°F
Power ramp ⁸	a) -24°F/hr b) -24°F/hr c) +24°F/hr d) +24°F/hr	-90°F -90°F +90°F +90°F	1000 1000 1000 1000	600°F isothermal 960°F isothermal 510°F 870°F
Emergency shutdown (forced convection main loop) ⁶	See Fig. 67		400	870°F
Emergency shutdown (loss of electric power) ⁷	See Fig. 68		50	870°F

¹Heatup within 48 hr by external electrical heaters; pump filled with argon.

²Heatup within 24 hr by external electrical heaters with circulating coolant.

³Heat supplied by low power reactor operation without electric heating and with the coolant circulating.

⁴Cooldown with the coolant circulating; the heat is dissipated at X-2.

⁵Coolant temperature is increased or decreased at 50°F/hr while at full power and full sodium flow by varying the air flow through X-2.

⁶Step change in reactor power from 50 MW to decay heat; flow cutback to 10% of full flow.

⁷Due to loss of electrical power, reactor power suddenly drops from 50 MW to decay power; forced circulation in main primary loop is maintained by pump P-35 on emergency and main secondary loop circulation is maintained by natural convection.

⁸Power is ramped from isothermal temperature conditions to full ΔT (190°F) conditions (or vice versa) with a maximum temperature rate-of-change of 24°F/hr.

TABLE 19. Auxiliary Primary Pump, P-3, Transient Operating Conditions

Source of Transient	Max. Rate of Temp. Change	Amplitude of Temp. Transient	No. of Occurrences	Temp. at Start of Transient
Initial dry heatup ¹	To be set by detail design	+280°F	50	70°F isothermal
Normal heatup ²	+50°F/hr	+350°F	1000	250°F isothermal
Nuclear heatup ³	+50°F/hr	+450°F	250	600°F isothermal
Normal cooldown ⁴	-50°F	-350°F	1000	600°F isothermal
Temperature ramp ⁵	a) +50°F/hr	+360°F	3500	600°F
	b) -50°F/hr	-360°F	3500	960°F
Auxiliary loop startup (loss of main loop) ⁸	Step change of 30°F	+30°F	200	960°F

¹Heatup within 48 hr by electrical heaters; pump filled with argon.

²Heatup within 24 hr by circulating externally heated coolant through system.

³Heatup by circulating primary coolant heated by reactor operation.

⁴Cooldown by convective cooling.

⁵Primary coolant temperature is increased or decreased at 50°F/hr while at power by varying heat sink.

⁸Auxiliary loop startup; primary loop cold leg temperature control set point set at prestartup reactor inlet temperature; step change in auxiliary primary flow; auxiliary secondary cold leg temperature gradually decreases as duty is imposed on loop.

TABLE 20. Auxiliary Secondary Pump, P-4, Transient Operating Conditions

Source of Transient	Max. Rate of Temp. Change	Amplitude of Temp. Transient	No. of Occurrences	Temp. at Start of Transient
Initial dry heatup ¹	To be set by detail design	+280°F	50	70°F isothermal
Normal heatup ²	+50°F/hr	+350°F	1000	250°F isothermal
Nuclear heatup ³	+50°F/hr	+450°F	250	600°F isothermal
Normal cooldown ⁴	-50°F/hr	-350°F	1000	600°F isothermal
Temperature ramp ⁵	a) +50°F/hr	+360°F	3500	582°F
	b) -50°F/hr	-360°F	3500	942°F
Auxiliary loop startup ⁸	-90°F at -6°F/min	-90°F	200	942°F

¹Heatup within 48 hr by electrical heaters; pump filled with argon.

²Heatup within 24 hr by circulating externally heated coolant through system.

³Heatup by circulating primary coolant heated by reactor operation.

⁴Cooldown by convective cooling.

⁵Primary coolant temperature is increased or decreased at 50°F/hr while at power by varying heat sink.

⁸Auxiliary loop startup; primary loop cold leg temperature control set point set at prestartup reactor inlet temperature; step change in auxiliary primary flow; auxiliary secondary cold leg temperature gradually decreases as duty is imposed on loop.

TABLE 21. Main Primary Shutdown Pump, P-35, Transient Operating Conditions

Source of Transient	Max. Rate of Temp. Change	Amplitude of Temp. Transient	No. of Occurrences	Temp. at Start of Transient
Initial dry heatup ¹	To be set by detail design	+280°F	50	70°F isothermal
Normal heatup ²	+50°F/hr	+350°F	1000	250°F isothermal
Nuclear heatup ³	+50°F/hr	+450°F	250	600°F isothermal
Normal cooldown ⁴	-50°F/hr	-350°F	1000	600°F isothermal
Temperature ramp ⁵	a) +50°F/hr	+360°F	3500	600°F
	b) -50°F/hr	-360°F	3500	960°F
Emergency shutdown (forced convection in main loop) ⁶	See Figs. 67 and 69		400	960°F
Emergency shutdown (loss of electric power) ⁷	See Figs. 67 and 69		50	960°F

¹Heatup within 48 hr by electrical heaters; pump filled with argon.

²Heatup within 24 hr by circulating externally heated coolant through system.

³Heatup by circulating primary coolant heated by reactor operation.

⁴Cooldown by convective cooling.

⁵Primary coolant temperature is increased or decreased at 50°F/hr while at power by varying heat sink.

⁶Step change in reactor power from 50 MW to decay heat; flow cutback to 10% of full flow.

⁷Due to loss of electric power, reactor power suddenly drops from 50 MW to decay heat; main primary loop circulation maintained at 6% by P-35 on emergency power.

transients and post-scrum transients. The latter group is plotted on the curves, Figs. 60-69. While these transients are often more severe, they also are generally less common than the normal operational transients which arise from routine operating conditions such as system preheating, normal shutdown, and temperature ramps. Both transient conditions must be carefully considered in the design of a system.

VIII. INSTRUMENTATION AND CONTROL

A. General

This section describes the instrumentation provided for the nuclear, plant protective and plant control systems, and defines the instrumentation requirements for the radiation monitoring, sodium pipe and equipment heating control and plant auxiliary service systems.

The plant control room (see Fig. 70 for arrangement) accommodates all instruments, controls and switches essential to operate the facility and to provide safe shutdown and startup. Instruments, controls, annunciators and switches of the highest priority are located on or immediately adjacent to C-04, which is directly in front of and beside the operator.

Instruments are assigned by system to control boards in the control room as follows:

- C-02 Nuclear Instruments
- C-04 Reactor Controls
- C-05 Liquid Metal Systems
- C-06 Radiation Monitoring
- C-07 General Services
- C-08 Temperature Monitoring and Preheat Systems
- C-09 Auxiliary Electrical Systems
- C-10 Relay Racks and Auxiliary Equipment
- C-21 ANL Experimental Equipment

B. General Reliability and Testing

All equipment, instruments, switches, wiring, and component parts furnished as part of any instrumentation system conforms to the applicable requirements of the latest ASA, ASTM, AIEE, NEMA, ISA, and JIC standards and the National Electric Code. Signal transmission and control system electronics are solid-state to the maximum extent practicable.

In addition to these requirements, the following steps have been or are planned:

1. All components for installation in the control room are specified for operation in an ambient temperature range of from approximately 5°C to 50°C, and a maximum relative humidity of 90%. Confirmation of the capability of the instrumentation to meet these criteria will be based on the testing of individual representative chassis for operation within specifications of these temperatures and humidity.

The normal temperature in the control room will be approximately 25°C. Ambient temperatures within the control panels will be surveyed during initial operation and one or more representative "hot spot" temperature sensors will be used to actuate an alarm, if excessive temperatures are reached. Surveillance of humidity is not specified since it is expected that the specified limit will not be reached during the operation of the facility.

Thermal characteristics of the control room will be tested by measuring the temperature as a function of time following shutdown of the air conditioning system. The results of these tests will aid in determining the most appropriate settings for the alarm temperature switches. It is expected that the alarm settings for the temperature switches will be of the order of 40°C. Corrective action will be taken following an alarm.

2. Instrument transmitters located elsewhere in the plant are similarly specified for operation in ambient temperatures between 5°C and 50°C, and a maximum humidity of 90%. Although they will, in general, not be located in air-conditioned spaces, representative temperatures are monitored in the control room. These temperatures will be observed closely during initial operation of the plant. At this time, corrective action will be taken as needed to adjust environmental temperatures to acceptable values for all phases of normal plant operation. Failure of any significant portion of the plant cooling systems such that environmental temperatures can be expected to rise beyond acceptable values will be considered cause for appropriate adjustment of operating conditions of the reactor plant.

3. The nuclear instrument thimbles will be maintained at less than 200°F by the shield cooling system. Failure of this system will result in an alarm. The computed rate of rise of temperature in the thimbles following failure of the cooling system is 5°F per hour at 50 MW reactor operation so that several hours will elapse before the coaxial cable limiting temperature of 250°F is reached. Temperatures within the thimbles will be measured and recorded regularly.

4. Wherever possible all nuclear and process instrument channels that have reactor shutdown functions for an excessively high, or low, value of the measure variable are arranged to actuate a channel failure alarm for apparent extreme values at the opposite end of the range. This is based on the premise that most instrument failures result in either maximum output or zero output.

5. Component chassis that are associated with alarm or scram functions are provided with interlocks so that removal of one of these chassis will automatically generate the alarm or scram associated with that channel.

6. A "channel failure" signal in a channel employed in two-out-of-three logic will always leave the remaining two channels in a one-out-of-two configuration.

7. Each channel that is employed in two-out-of-three or one-out-of-three trip logic is fed from a separate power supply.

8. Redundant instrument channels are isolated as much as possible physically in order to reduce the probability of simultaneous failure.

9. In cases where the trip level of a channel depends on a range switch an alarm is provided to indicate the use of an incorrect range.

10. Where deliberate bypassing of interlocks is permitted under certain operating conditions, it can only be accomplished by means of key switches. Keys for such switches will be under strict supervisory control.

11. Bypassing of the scram circuits associated with the startup nuclear instruments is accomplished automatically by contacts associated with the intermediate range instruments. Upon subsequent reduction of reactor power to the startup range the bypass is automatically removed. Failure of the operator to re-energize the start-up channels prior to this time will result in a reactor scram.

12. The nuclear instrumentation is provided with the built-in test circuits listed below. These are individual, so that redundant channels need not be connected together for the test operation. For testing trip circuits the test signal can be superimposed on the existing reactor-derived signal. For calibration the test signal only is applied. In any case, on-line testing preserves the two-out-of-three logic so that tripping of one of the two remaining channels during a test will generate a scram signal. The test signals will be injected at such a point as to include as much of the system as possible. Testing will be done on a regularly scheduled basis and results will be recorded.

a. Channels 1, 2 and 3 (Start-up range)

- (1) Variable-slope ramp signals for testing period circuits.
- (2) Variable-frequency pulse signals for testing steady-state calibration.

b. Channels 4, 5 and 6 (Intermediate range)

- (1) Same as a(1).
- (2) Variable-current source for testing steady-state calibration.

- c. Channel 7 (Linear, wide range)
Same as b(2).
- d. Channel 8 (Automatic flux control)
No built-in test features.
- e. Channels 9, 10 and 11 (Power range)
Same as b(2).

13. Where on-line testing is not feasible, as with single-channel process instruments having a scram function, regularly scheduled recorded surveillance will be performed with the objective of detecting unsafe failures (those that do not result in a channel trip signal). These protective channels will be tested periodically during shut-down, and under conditions simulating as nearly as possible the actual measured parameter.

C. Nuclear Instrumentation (see Fig. 71)

The nuclear instrumentation system will measure and indicate neutron flux level and period in overlapping ranges from source level up to 300% at full power. The system will initiate reactor scram or fuel handling inhibit whenever the neutron flux level or period exceed the preset limits, and will provide a signal for automatic neutron flux level controller. All nuclear system components, except the detectors, are located in the reactor control room and are designed to operate in an ambient temperature up to 120°F. The detectors are located within thimbles, adjacent and external to the reactor vessel. An individual thimble is provided for each detector. The detectors and associated components that are located within thimbles (connectors and cables) are suitable for operation at 250°F.

Adjustable trip units that are composed of a bi-stable amplifier in conjunction with a relay(s) are utilized. The relays are de-energized in a tripped (abnormal) condition.

Since most failures of a channel components will result in the output signal either going up to full scale or down to zero, maximum fail-safeness is achieved by providing bi-stable trip units that will operate if the signal exceeds the preset limits in either direction. In certain cases a trip unit that is provided for the monitoring of a reactor system abnormality that results in an excessively high, or low value of measured variable, will also provide the monitoring of a channel failure that results in the change of output signal in the same direction. In other cases, special trip units are provided for failure monitoring.

"Channel trips" are established for each channel by arranging the contacts of system abnormality monitors and of channel failure monitors (including chassis interlocks, calibration switches, etc.), in a series circuit, through which power is supplied to the "channel relays." The "channel relay" contacts are utilized in the plant protective system circuits as needed.

Redundant channels are provided to increase serviceability and to allow on-line testing. The redundant channels are kept independent from each other by supplying the needed d.c. power from a separate unit for each channel. Separate voltage supplies are provided for each detector. A failure of a channel detector voltage will trip that channel.

The nuclear instrumentation system comprises eleven channels which are divided into five groups as follows:

1. Startup Channels (LCR-Period Channels)(See Fig. 72).

Three identical channels (Nuclear Channels 1, 2 and 3) are provided for the startup range. Fission counters are utilized as detectors. A counting rate of approximately 10 c/s is produced when operating in a neutron flux of approximately 15 nv with a maximum gamma background at approximately 10^4 R/hr. Each of the channels will cover reactor power range from source level to 5 kW. An additional half decade can be covered above 5 kW but with an approximately 20% loss of the counting rate.

Pulses from the detector are amplified by a linear pulse amplifier and fed into a pulse height discriminator. Discriminator output is applied to a scaler (which can be controlled by a timer or manually), and to a logarithmic count rate meter. Period is computed by differentiating the signal from the count rate meter. A recorder (one only for three channels) is provided for recording the log count rate. A selector switch provides for recording of any one of the three channels. In a similar manner the period at any one of the three channels can be recorded.

The main function of each channel is to provide protection against abnormally short period. High flux protection is provided during fuel handling. A minimum counting rate of 10 c/s is required and a channel will be tripped if the counting rate drops below this value.

Built-in signal sources are provided for each channel for calibration and for trip testing.

2. Log N-Period Channels (See Fig. 73)

Three identical channels (Nuclear Channels 4, 5 and 6) are included in this group. Compensated ionization chambers are utilized as detectors. Each of the channels will cover reactor power range from

approximately 500 W to 150 MW (300% of fuel power). The lower limit is based on operating with a gamma background of approximately 10^4 R/hr. The channel range will be extended when operating in a lower gamma field.

Current from the detector is fed into a logarithmic current (Log N) amplifier. The period is computed by differentiating the signal from the Log N amplifier. A recorder (one only for three channels) is provided for recording the Log N. A selector switch is provided for recording the Log N of any one of the three channels. In a similar manner the period of any one of the three channels can be recorded.

The main function of each channel is to provide protection against an abnormally short period in the intermediate and/or power ranges. The channels are also utilized to provide a bypass of nuclear channels 1, 2 and 3 interlocks in the scram circuit, when the reactor power reaches a preset level.

Built-in signal sources are provided for each channel for calibration and trip testing.

3. Linear N-Channel (See Fig. 74)

A single linear channel (Nuclear Channel 7) is provided for the intermediate/power ranges. A compensated ionization chamber is utilized as detector. The channel will cover reactor power range from 500 W to 150 MW. The lower limit is based in operating with a gamma background at approximately 10^4 R/hr. The channel range will be extended when operating in a lower gamma field.

Current from the detector is fed into a linear current amplifier. A manually operated range switch provides for range switching in steps of 3X and 10X. A recorder is provided for recording the linear power level.

This channel will provide a high flux trip at power levels lower than the power range protective channels. The channel will also serve a main indicating instrument at the neutron flux level in the intermediate and/or power ranges.

4. Automatic Flux Level Control Channel (See Fig. 74)

A single linear channel (Nuclear Channel 8) is provided. A compensated ionization chamber is utilized as detector. The channel will cover reactor power range from 500 kW to 75 MW.

The current from the detector is fed into a linear current amplifier. The output signal from this amplifier is supplied to NC-108 automatic neutron flux controller. Low detector voltage trip will result in the change over from the automatic to manual control mode.

A built-in calibration signal source is provided.

5. Power Range Protective Channels (See Fig. 75)

Three identical channels (Nuclear Channels 9, 10 and 11) are included in this group. Compensated ionization chambers are utilized as detectors. Each of the channels will cover the reactor power range from 500 kW to 75 MW.

The current from the detector is fed into a linear current amplifier. A recorder is provided for each channel for recording the linear power level.

The main function at each channel is to provide protection against an abnormally high flux level. A signal from the amplifier of each channel is also fed into flux/flow comparators to generate "Flux Flow ratio high" trips.

Built in signal sources are provided for each channel for calibration and for trip testing.

D. Plant Protective System (See Fig. 76 and 77)

1. General

The plant protective system includes provisions to protect the reactor and the associated equipment in the event certain abnormal conditions occur. The main action taken is that of shutting down the reactor (scram) or of inhibiting the fuel handling. Shutting down of the reactor is accomplished by interrupting the power to control rod clutch magnets and dropping in the rods. Simultaneously, the scram relays are de-energized. The relay contacts are utilized to initiate other action as needed, e.g., shutting down the pumps, starting up other equipment, etc.

Insofar as practicable, circuits will be designed and components will be applied in such a manner that no failure of a single component or interconnecting wire can compromise the safety system. This means that component parts are applied in such a way that their most likely mode of failure results in a trip of the channel involved. The method of failure monitoring and the establishing of the "channel trips" is discussed in more detail in paragraph B, Nuclear Instrumentation, above. The philosophy was also followed in the design of the plant protective channels that receive information from two or more process or nuclear channels, such as neutron flux/sodium flow ratio, primary/secondary flow ratio, etc. Such a plant protective channel would be tripped if a failure occurred in any of the participating channels. The development of plant protective system channels is shown in detail in Figs. 78-81. Three redundant channels are

provided and two-out-of-three scram logic is used with those parameters where spurious trips are expected; also to increase serviceability and allow for on-line testing.

2. Abnormalities

The abnormalities that result in reactor scram or fuel handling inhibit are compiled in the following list. The list also includes some abnormalities that result in an alarm only. Other abnormalities to be automatically identified are given by the annunciator displays Figs. 82 and 83.

Abnormality	Reactor Scram	Alarm Only	Inhibit Fuel Handling
1. Flux level low: Nuclear Channels 1, 2 & 3	X ^(1,2)	X ⁽³⁾	X ^(3,7)
2. Period short:			
(a) Nuclear Channels 1, 2 & 3	X ^(1,2)	X ⁽³⁾	X ⁽³⁾
(b) Nuclear Channels 4, 5 & 6	X ⁽⁴⁾	X ⁽³⁾	X ⁽⁴⁾
3. Flux level high:			
(a) Nuclear Channels 1, 2 & 3	X ^(1,2)	X ⁽³⁾	X ⁽³⁾
(b) Nuclear Channel 7	X ⁽⁵⁾		X
(c) Nuclear Channels 9, 10 & 11	X ⁽⁴⁾	X ⁽³⁾	
4. Detector Voltage low:			
(a) Nuclear Channels 1, 2 & 3	X ^(1,2)	X ⁽³⁾	X ⁽³⁾
(b) Nuclear Channels 4, 5 & 6	X ⁽⁴⁾	X ⁽³⁾	X ⁽³⁾
(c) Nuclear Channel 7	X ⁽⁵⁾		X
(d) Nuclear Channel 8		X ⁽⁶⁾	
(e) Nuclear Channels 9, 10 & 11	X ⁽⁴⁾	X ⁽³⁾	
5. Flux/Flow ratio high: (Flux signals from Nuclear Channels 9, 10 and 11)	X ⁽⁴⁾	X ⁽³⁾	
6. Nuclear Channel 7 range switch not in "correct" range			X
7. Nuclear instrumentation d.c. power supply failure		X	
8. Any control rod unlatched	X ⁽⁸⁾		
9. Any control rod not fully down	X ⁽⁹⁾		
10. Control rod scram assist argon pressure low	X		
11. Automatic flux control rod down	X		

Abnormality	Reactor Scram	Alarm Only	Inhibit Fuel Handling
12. Reactor inlet coolant flow low	X ⁽⁴⁾	X ⁽³⁾	
13. Reactor inlet coolant flow rate at change high positive or negative	X ⁽⁴⁾	X ⁽³⁾	
14. Main primary-secondary coolant flow ratio high or low	X ⁽⁴⁾	X ⁽³⁾	
15. Reactor outlet temperature high	X ⁽⁴⁾	X ⁽³⁾	
16. X-1 outlet temperature rate at change high positive or negative	X ⁽⁴⁾	X ⁽³⁾	
17. X-2 sodium outlet temperature rate of change high positive or negative	X ⁽⁴⁾	X ⁽³⁾	
18. Reactor blanket gas pressure high	X		X
19. Reactor sodium level low	X ⁽⁴⁾	X ⁽³⁾	
20. NaK annulus expansion tank level high or low	X		
21. Earthquake trip	X		X
22. Main primary pump motor power off	X		
23. Main secondary pump motor power off	X		
24. Main secondary heat exchanger blower; one fan off above 60% reactor power level	X		
25. Main secondary heat exchanger blower; both fans off below 60% reactor power level	X		
26. Manual scram	X		
27. Auxiliary primary pump flow low or auxiliary secondary pump flow low; and shutdown pump power off		X	
28. Fuel handling incomplete	X		

Abnormality	Reactor Scram	Alarm Only	Inhibit Fuel Handling
29. Cell access air locks "A" or "B" manual latch open or cell annex closure open	X		
30. Main supervising key switch off	X		
31. Reactor operate-fuel handling selector switch in "Fuel Handling" position	X		
32. Reactor operate-fuel handling selector switch in "Reactor Operate" position	X		
33. Fuel handling supervisory key switch off			X ⁽¹⁰⁾
34. Run-startup bypass key switch in "Bypass"	X		

¹ Reactor startup is prevented if any one out of three channels is tripped. Two-out-of-three scram logic is effective after the scram bus is energized.

² Above a present flux level, nuclear channels 4, 5 and 6 will provide bypass of nuclear channels 1, 2 and 3 scram functions.

³ One-out-of-three channel trip.

⁴ Two-out-of-three channel trip.

⁵ May be bypassed with a key switch.

⁶ Also will switch from automatic to manual control mode.

⁷ May be bypassed with a key switch for source removal.

⁸ May be bypassed with a key switch for rod drop test.

⁹ Effective only for reactor startup. Bypassed when the scram bus is energized.

¹⁰ Effective only for energizing of fuel handling inhibit bus. Bypassed when fuel handling inhibit relays are energized.

3. Annunciator System (See Figs. 82 and 83)

The control room annunciators provide the following functions and alarm sequence:

a. Alert the attendant to an off-normal condition by means of an audible signal (horn) and flashing light.

b. Identify off-normal conditions by means of back-lighted nameplates.

c. Permit the attendant to acknowledge each off-normal condition by silencing the horn while retaining the alert indication (i.e., steady light).

d. Identify when each off-normal condition returns to normal ("ring back" -- visual only, flashing).

e. Require manual reset to clear annunciator.

f. Permit the attendant to check the annunciator lamps for burnout, as well as operation of the audible signals (horn and bell).

In addition to these functions, the scram annunciators identify the initial off-normal condition causing scram. The annunciator system operates with normally closed trouble contacts which open on off-normal condition.

E. Plant Control System

1. General

The plant control system accommodates manual startup or shutdown and manual or automatic steady-state operation.

Normal operating parameters are:

a. Reactor power--0 to 50 MWt

b. Reactor coolant

Temperature difference, °F (Δt across vessel)	0 to 190
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Temperature, inlet, °F	300 to 1050
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Temperature, outlet, °F (mixed with by-pass flow)	300 to 1150
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Flow, gpm	750 to 7500
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2. Control Functions

Figures 84-106 (Piping and Instrumentation Diagrams) show the principal plant control functions. These are:

a. Reactor Power (Flux) Control

This function provides two methods of control rod regulation--manual or automatic. Manual control rod regulation will be normal operation for start up, power level change, or shut down. Automatic control rod regulation may be used optionally to control neutron power to a pre-selected set point. On SCRAM, from either method of operation all rods will be automatically driven down into the core by gravity and a pneumatic pressure assist when the electromagnet that couples the lead-screw assembly to the drive shaft is de-energized.

Reactor neutron flux measurement is derived from Channel 8 (NE-108) of the nuclear instrumentation system. Under automatic control the measured neutron flux signal is transmitted by NT-108 to flux controller NC-108 where it is compared with the selected neutron flux demand set point. Any difference between the measured flux and the flux set point greater than $\pm 1\%$ (of set point) shall initiate rod raising or lowering signals. Any difference greater than $\pm 2\%$ (of set point) trips, NC-108 out of automatic to manual and annunciates.

Final control action which drives the control rod up or down is accomplished by closing contacts functionally in parallel with the control rod drive switch, HS-117.

HS-116 provides for selection of any one of the control rods for automatic control. HS-115 provides for the selection of any one of the rods for manual control of rod position during either the automatic or manual mode of operation. Interlocks allow only one rod to be selected for automatic operation (from HS-116) and one rod for manual operation (from HS-115) at any one time. HS-117 provides manual demand (up or down) of the control rod selected by HS-115. If HS-115 position is coincident with the rod selected by HS-116, actuation of HS-117 results in switchover from automatic to manual control. HS-118 provides manual demand (up or down) of the control rod selected by HS-116. Prior to NC-108 being placed in manual, HS-118, NC-108, and HS-117 are interlocked so that no two manually driven rods can be driven in the same direction at one time. Fine and coarse indication of rod position is shown on PnI-118-1 through 12 double scale indicators.

The relationship of measured neutron flux and main primary coolant mass flow is monitored by flux-flow comparators NFX 109, 110, and 111. The outputs are fed to the plant protective system for scram if two out of three comparators indicate excessive flux relative to flow rate.

b. Main Primary System Coolant Flow Control

This function provides two methods of control for sodium flow--manual or automatic pump speed demand. A servo loop around the pumps automatically maintains speed. On SCRAM, the pump is automatically ramped down from existing speed to a speed corresponding to 10% flow.

c. Main Secondary System Coolant Flow Control

As with the primary system, two types of control are available--manual or automatic pump speed demand. A servo loop around the pumps automatically maintains speed. On SCRAM, the pump is automatically ramped down from existing speed to a speed corresponding to 10% flow. Also, on SCRAM, X-2 outlet temperature at that time is held automatically as a reference.

d. Auxiliary Primary System Coolant Flow

Normally the electromagnetic pump will run continuously at approximately 10% of rated flow (not controlled). Scram will not affect this pump unless caused by or accompanied by loss of main system coolant flow (primary, secondary, or both). If this situation does occur, the pump will be automatically switched to full flow operation.

e. Auxiliary Secondary System Coolant Flow

The electromagnetic pump will always run at 100 percent of rated flow (not controlled).

f. Main System Temperature Control

Primary sodium reactor inlet temperature is controlled via manipulation of the main heat exchanger X-2. There are two sodium loops, two operating modes, and two controlled temperature variables. There is only one point of entry of "main" primary sodium; line 3-14 inches.

The following schedule of actions is automated:

Reactor Operating Mode	Primary Sodium		Secondary Sodium	
	Controlled Variable	Setpoint Origin	Controlled Variable	Setpoint Origin
Normal Operation	Temperature of (primary) sodium leaving X-1. TE-739	Manually se- lected TC-739	Temperature of sodium leaving X-2.	Output of TC- 739. (Cascaded control loops)
Scram	None	None	Temperature of sodium leaving X-2.	Output of TC- 739 at instant of scram. (Hold)

During normal operation, primary sodium temperature setpoint is manually selected at controller TC-739. This temperature is measured at TE-739 (line #2--20 inch), and deviations, (past, present, and "future") as operated upon by controller TC-739, produce controller output. This output feeds to another controller (furnished with X-2) as the set point value for that controller.

A thermocouple (furnished with X-2) measures X-2 outlet temperature, and deviations (past, present, "future") as operated upon by the X-2 controller produce signals to manipulate X-2 mechanisms (such as blower speed) and regulate X-2 outlet temperature. This multi-variable, multi-loop arrangement is known as cascade control.

After the instant of scram, primary sodium temperature is no longer directly controlled. The X-2 controller set point is held at the value existing at the instant of scram, and the X-2 controller and manipulated X-2 mechanisms seek to maintain a steady value of X-2 outlet temperature.

g. Auxiliary Primary System Temperature Control

Auxiliary sodium reactor inlet temperature is controlled via manipulation of the auxiliary heat exchanger X-4. There are three sodium loops, three operating modes, and three controlled temperature variables. There are two points of entry of "auxiliary" primary sodium; line #4--6 inch into line #3 and line #68--3 inch.

The following schedule of actions is automated:

Reactor Operating Mode	Primary Sodium		Secondary Sodium	
	Controlled Variable	Setpoint Origin	Controlled Variable	Setpoint Origin
<u>Normal Operation</u>	Temperature of (primary) sodium leaving X-43. TE-1300	Measured temperature of (primary) sodium leaving X-1. TE-739.	Temperature of sodium leaving X-4.	Output of TC-1300. (Cascaded control loops)
<u>Scram</u> (Not caused or accompanied by loss of main system primary or secondary system coolant flow)	Temperature of (primary) sodium leaving X-43. TE-1300	Measured temperature of (primary) sodium leaving X-1 at instant of scram. TE 739. (Hold)	Temperature of sodium leaving X-4.	Output of TC-1300. (Cascaded control loops)
<u>Scram</u> (Caused or accompanied by loss of main system primary or secondary coolant flow)	Temperature of (primary) sodium leaving X-3. TE-471	Measured temperature of (primary) sodium leaving X-1 at instant of scram. TE-739. (Hold)	Temperature of sodium leaving X-4.	Output of TC-1300. (Cascaded control loops)

During normal operation, auxiliary primary sodium temperature setpoint is automatically slaved to the temperature of primary sodium measured at TE-739 (line #2--20 inch). Auxiliary primary sodium temperature is measured at TE-1300 (line 68--3 inch) and deviations, as operated upon by TC-1300, produce controller output. This output feeds to another controller (furnished with X-4) as the setpoint value for that controller. A thermocouple (furnished with X-4) measures X-4 outlet temperature, and deviations as operated upon by the X-4 controller produce signals to manipulate X-4 mechanisms (such as damper position) and regulate X-4 outlet temperature. This, again, is cascade control.

After the instant of scram (not caused or accompanied by loss of main system primary or secondary coolant flow), auxiliary primary sodium temperature at TE-1300 is controlled as above except that the setpoint for controller TC-1300 is held at the value existing at scram. During such a scram the main coolant system will be supplying adequate cooling, and the auxiliary system nearly preserves the status quo of reactor flange cooling temperature control.

After the instant of scram (caused or accompanied by loss of main system primary or secondary coolant flow), auxiliary primary sodium temperature at TE-1300 is no longer controlled. Control is shifted to the temperature of auxiliary primary temperature measured at TE-471 (line #4--6 inch). Controller TE-1300 setpoint is held as before, but TC-1300 measurement is now taken from TE-471. TC-1300 cascades to the X-4 controller as before. During such a scram, the main coolant system is disabled, and the auxiliary coolant system must take over emergency cooling.

F. Radiation Monitoring System (See Fig. 107)

The plant radiation monitoring system monitors the radiation levels in various locations to provide basic radiation protection for plant personnel. The system also monitors and indicates the level of radiation present in stack effluent released to the atmosphere. The system is divided into the following main categories.

1. Area Radiation Monitors

a. General Area Monitoring

Eleven channels and four spares are provided. Either halogen quenched G-M counters or gamma sensitive ionization chambers will be used as detectors. The range of each channel is from 1 mR/hr to 100 R/hr. Indication of the radiation level is provided both at the detector and in the reactor control room. Adjustable high and low level alarms are provided for each channel. High level indicating lights and audible alarms

are provided both at the detector and in the reactor control room (audible alarm by annunciator). A low level alarm indicating light is provided in the reactor control room only. A solenoid operated check source is provided at each detector. Check source controls are located in the reactor control room. A 16-point recorder is provided in the reactor control room for recording the radiation levels of each channel. Recorder charts and scales are direct reading. All necessary power supplies are located in the reactor control room.

b. Controlled Access Area Monitoring

Eight channels and two spares are provided. Gamma sensitive ionization chambers, suitable for operation in argon atmosphere, are utilized as detectors. The range of each channel is from 100 mR/hr to 10^4 R/hr. Each channel is provided with adjustable high and low level alarm units. Alarm indicating lights are provided for both high and low levels in the reactor control room only. High level alarm circuits are also arranged to provide audible alarm on the control room annunciator. A 10-point recorder is provided in the control room for the recording of radiation levels of each channel. Recorder charts and scales are direct reading. All necessary power supplies are located in the reactor control room.

2. Process Gas and Airborne Particulate Monitors

Three channels are included as follows: (a) stack gas activity, (b) stack airborne particulate activity and (c) building vent exhaust gaseous activity.

Two gas sampling stations are provided. One is utilized for the building vent exhaust channel and the second will be utilized by both stack channels. Each of the samplers include an isokinetic sample nozzle, pumping system, flow adjustment and flow indicator. Each detector assembly includes a thallium-activated sodium-iodide crystal and a preamplifier. The stack particulates are collected on a moving filter. Provision is made for periodic gas flush cleaning of the sampler systems, using air or bottled gas. An absolute fixed filter is provided in the building vent exhaust sampler for removing airborne particulates from the sampled gas. A solenoid operated check source is provided for each detector. All controls for operating the sampler mechanisms and the check source are located in the reactor control room. An adjustable high level alarm with indicating light is provided for each channel. Alarms are also provided for low-airflow and for failures in the moving filter system. All alarms are located in the control room. Ratemeters with logarithmic scale indicators are used for indication of radiation level. Two recorders are provided. One two-pen recorder is utilized for reading the output of the channels associated with the stack activity monitors. The second recorder is single-pen and is used for recording of the building vent exhaust channel. Charts and scales are logarithmic, direct reading.

3. Waste Gas Monitors

Two identical channels are provided for monitoring of: (a) blanket waste gas activity and (b) cell, cell annex and vault waste gas activity.

A gamma scintillation detector assembly that includes a thallium-activated sodium-iodide crystal and a preamplifier is provided for each channel. A solenoid operated check source is provided at each detector. Source controls are located in the reactor control room. Rate-meters with logarithmic scale indicators are used for radiation level indication. Spectrometer input circuits are included with provisions for both differential and integral analysis modes. An adjustable high level alarm with indicating light is provided for each channel. One two-pen recorder is provided for both channels. Charts and scales are logarithmic, direct reading.

4. Miscellaneous

Cooling water and liquid waste activity will be checked periodically by radiochemical analysis. A portable gas monitor will be provided in the reactor wing for monitoring the activity at various locations. Personnel monitors will be utilized for individual protection of personnel.

G. Pipe and Equipment Heating Control

Controls will maintain system(s) during startup, shutdown, and isothermal operation at 300-600°F. The control system comprises the control board C-08, on which are mounted five 60-point recorder-controllers, one switch cabinet assembly, and one relay and selector switch cabinet assembly for remote mounting in the cable spreading room below the control room.

Each temperature-recorder control system is designed to perform the following functions:

1. Measure each temperature point through the remote cabinet-mounted selector switch.
2. Annunciate and record each off-normal point.
3. Control each measured temperature by actuation of the appropriate heater circuit through the recorder controller set point switch, remote cabinet-mounted selector switch, latching relay and auxiliary relay.

A spare thermocouple is installed for each temperature measurement point in the system which is considered inaccessible for normal maintenance once the plant has accumulated significant power operation.

H. Plant Auxiliary Service

1. General

Instrumentation for the sodium, argon, radioactive vent, heating and ventilation, air cooling water, and miscellaneous service systems is provided as shown on various P and I diagrams. In general, instrumentation is provided to monitor temperatures, pressures, levels and flows, and control these variables as dictated by the system requirements. Abnormal operation is annunciated in the control room and provides for plant shutdown in those critical applications.

2. Atmosphere Control

The reactor cell, vault and cavity cooling system is designed to maintain temperature within limits specified. The reactor vault and cell atmosphere control system is designed to:

- a. Provide an inert gas atmosphere and maintain a negative pressure within the cell and positive pressure in the vault during operation.
- b. Provide for air replacement of the vault and cell atmosphere for personnel access to these areas during shutdown.
- c. Provide for oxygen and water vapor removal when changing from air to inert gas atmosphere within the cell and vault.
- d. Analyze the cell, vault and cavity inert gas atmosphere for O_2 , moisture, and alarm abnormal levels.

3. Smoke Detection

A smoke detector system is provided to detect and annunciate the presence of combustion products within the cell, vault control room, and sodium equipment areas.

4. Air Conditioning

The heating, ventilating, and air conditioning control system is designed to:

- a. Heat, cool, ventilate, and air condition the control room and cable routing room.
- b. Heat, cool, and ventilate the remainder of the support wing.
- c. Maintain a positive pressure in the support wing relative to the reactor building and to the atmosphere.
- d. Heat, cool, ventilate, and maintain negative pressures in the reactor building.

IX. ELECTRIC POWER AND SERVICE SYSTEMS

A. Power Distribution

1. 13.8-kV System

The normal power source for FARET is the EBR-II 13.8-kV switchgear. A 13.8-kV pole line connects the FARET substation with EBR-II. This line is fed from an existing spare circuit breaker in the EBR-II switchgear.

The FARET substation consists of three outdoor metal-enclosed switches throat connected to the three power transformers next to the reactor building. Each transformer is equipped with a load break switch and current-limiting fuse, completely enclosed in a weatherproof housing.

Three transformers are located in the substation--one transformer to supply power to the 2400-volt switchgear and two transformers for the 480-volt switchgear. The transformers are askarel-filled, self-cooled with provision for future addition of fan cooling. The 480-volt transformers are each sized to supply the total 480-volt facility load within the forced air cooled rating.

2. 2400-volt System

The 2400-volt system, which is the power source for motors rated 125 horsepower and above, is resistance grounded with a neutral resistor sized to carry full ground fault current for 30 minutes. The 2400-volt switchgear consists of indoor metal-clad drawout units and is fed from the 13800-2400-volt transformer through cable in conduit. The switchgear consists of a feeder circuit breaker for each motor.

The feeder breakers are equipped with phase and ground over-current and current balance protective relaying. Bus undervoltage and phase-sequence protection is provided to protect the motors against reverse rotation and harmful heating. Each motor is equipped with resistance temperature detectors to allow loading by winding temperature. Surge capacitors are provided for each motor. A ground relay in the transformer neutral is provided to alarm a 2400-volt system fault not cleared by the feeder ground relaying.

3. 480-volt System

The 480-volt system, which furnishes power for motors 100 horsepower and below, for piping and equipment heating and for emergency loads, consists of switchgear, load centers and a diesel-engine driven emergency generator as shown on Fig. 108 and 109.

The 480-volt switchgear is indoor, metal-enclosed, with draw-out circuit breakers. The switchgear is sectionalized into two buses each fed from a separate 13800-480-volt transformer through metal-enclosed bus run from the substation. Each bus is provided with a main incoming circuit breaker and the two buses are connected by a tie breaker. Feeder breakers are provided to distribute power to the 480-volt load centers. Power is distributed at 480 volts, 3 wire, to the 480-volt emergency load center and 480/277 volt 4-wire pipe heating load center. All other feeder circuits are 480-volt, 3 wire.

Two 480-volt emergency load centers supply power for critical loads which can tolerate only a short power interruption. These load centers are fed normally through circuit breakers from the 480-volt switchgear and, in an emergency, from the diesel generator. On loss of normal power, the diesel generator starts automatically and the emergency load centers transfer automatically to the generator by operation of stepping relays. On restoration of normal power, the diesel breakers are opened by the operator and the 480-volt switchgear breakers are closed.

The 480-volt load centers are provided to distribute power to the motor, electric heating and other loads. The load centers are indoor metal-enclosed structures consisting of circuit breaker units and combination circuit-breaker motor-starter units. The units are the disconnecting type to permit isolating the feeder circuits from the 480-volt bus.

B. 120-volt a.c. Continuous Instrument and Control Power Systems

1. Continuous Power

The 120-volt a.c. continuous power system supplies power for instrumentation, control, and protective devices which require uninterrupted, noise-free power (Fig. 104). The 120-volt a.c. continuous power supply consists of a three-machine motor-generator set and a 125-volt d.c. battery. The three machines are a synchronous motor fed by the 480-volt emergency load center, a d.c. motor fed by the battery, and a 120-volt a.c. generator which feeds the 120-volt a.c. continuous-power load center.

Under normal operation the generator is driven by the 480-volt synchronous motor. The d.c. motor, acting as a generator, maintains the battery in fully charged condition. On loss of the 480-volt supply, the generator is driven by the d.c. motor fed by the battery. The battery is sized for 30 minute operation at full generator output.

Maximum variations of a.c. output voltage and frequency are as follows:

	Steady State	20% Step Load Change	Transfer from a.c. to d.c. Drive
Voltage	$\pm 1\%$	$\pm 5\%$	$\pm 5\%$
Frequency	$\pm 1\%$ (d.c. drive)	$\pm 4\%$ (d.c. drive)	$\pm 2\%$

Harmonic content shall not exceed $\pm 4\%$.

The d.c. machine, when operated as a motor, drives the unit at full a.c. load with input voltage in the range of 100 to 125-volts d.c. When operated as a generator, it delivers output over a range of 129 to 140-volts d.c. In the d.c. drive (as a motor), the speed regulation is effective over the entire input voltage range of 100 to 125-volts d.c.

The a.c. motor is adequate to supply motor-generator set losses and drive the alternator and d.c. machine when loaded up to their ratings, within the specified output limits with input voltage in the range of 440-volts, $\pm 10\%$, 3 phase, 60 cycles.

The motor-generator controls permit starting from either the a.c. feeder or the battery at the option of the operator.

With normal supply voltage available, the motor-generator set operates in the normal drive mode in which the synchronous motor drives both the d.c. generator and alternator. Automatic transfer to the d.c. drive mode is initiated for the following abnormal situations:

- (1) Severe line disturbances in which the supply voltage drops 10% below 440 volts for 0.5 seconds or more.
- (2) Severe line disturbances in which the supply frequency varies more than $\pm 3\%$ of 60 cycles for 0.5 seconds or more.

The delay time specified above is adjustable over the range of 0.5 to 5 seconds.

A reverse power or current relay in the 125-volt d.c. circuit is provided to trip the synchronous motor line contactor immediately on transition from normal to d.c. drive mode. This relay has essentially no time delay. Apparatus is provided to automatically limit the output current of the d.c. machine when operating as a generator. The current is limited to a value not exceeding 110% of rating for this mode of operation. Apparatus is also provided to automatically transfer the motor-generator set from d.c. to a.c. drive mode following restoration of the normal supply

voltage to 93% of 440 volts and 98% normal frequency. The transfer to normal drive mode is delayed and is adjustable over a range of from 1 to 10 seconds.

The 120-volt a.c. continuous power load center is a circuit-breaker panel board and contains a manual transfer switch to connect the continuous power load center to the 120-volt a.c. instrument power load center when the motor-generator is out of service.

The loads on the continuous power three-machine motor-generator set are listed in Fig. 74. The critical d.c. loads are supported by the 125 volt battery and are also shown in Fig. 74.

2. Momentarily Interrupted Instruments and Control

The 120-volt instrument power system furnishes power for instrumentation and control devices which can tolerate short power interruptions. The system is supported by a transformer fed from the 480-volt emergency load center. Distribution is accomplished from the 120-volt instrument load center, a circuit-breaker panel board.

3. Momentarily Interrupted Rotating Loads

Critical rotating loads which can tolerate a momentary power interruption are fed from a second two-machine motor-generator set. On normal power failure, power is supplied from the 125-volt d.c. battery. Critical loads connected to this generator are the fuel handling machine, fuel handling machine blower, and the fuel disassembly machine.

A separate inverter-converter, with a second battery on floating charge, serves the main primary shutdown pump. The pump is energized from the inverter, which normally is fed from the battery charger. The battery charger is energized from the 480-volt system.

C. Equipment in Inert Gas Environment

All motors operating in an argon atmosphere are totally enclosed and fan-cooled or totally enclosed and nonventilated with Class H insulation. These motors are designed for a temperature rise in air corresponding to Class A insulation.

All electrical equipment, other than motors, is derated for operation at a pressure of 23.75 inches of mercury and for the reduced cooling capability of argon. The allowable losses in argon for self-cooled equipment are limited to 65% of the allowable losses in air. The allowable losses in argon for forced-cooled equipment are limited to 71% of the allowable losses in air.

Equipment heaters have a maximum voltage of 277 volts, 60 cycles. All other equipment has a maximum voltage of 240-volts, 60 cycles. Wiring, connectors, and terminals in argon are designed for 600-volt service but limited to operation at a maximum voltage of 240-volts, 60 cycles for equipment other than heaters.

Bare conductors are provided with sufficient separation to insure a strike distance of $1/8$ inch and a creep distance of $1/4$ inch. The windings of all equipment, including motors, are impregnated by the vacuum pressure method.

D. Miscellaneous Electrical

1. Lighting

The basic lighting system consists of fluorescent fixtures supplied by a 208/120-volt 4-wire distribution system. Incandescent and mercury-vapor fixtures are used in some areas inside the buildings and for outdoor lighting. The cell interior lighting consists of mercury-vapor fixtures fed from a 480/277-volt 4-wire system. The control room has a luminous ceiling. Lighting is provided for the roads and parking lot. Convenience outlets at 120-volts are provided throughout the facility for general usage and in the parking lot for automobile engine heaters.

Battery-powered emergency lighting units are located to cover all exits and occupied areas inside the buildings including the control room. These units are of the self-contained portable type and include a battery charger. Cell emergency lighting consists of incandescent units fed from the 125-volt d.c. load center through a contactor automatically actuated by failure of normal lighting power.

Part of the cell interior lighting is supplied from the 480-volt emergency load center to insure that cell illumination will be partially maintained by the 480-volt emergency generator during an outage of the normal 480-volt system.

One lighting transformer supplies the support wing and two transformers feed the reactor building. The reactor building transformers are connected each to a separate 480-volt load center. Lighting circuits supplied by each transformer are alternated through the building to insure that no area will be blacked out by the loss of one transformer or its associated 480-volt supply.

Lighting transformers are 480-208/120-volt 3-phase units. Each transformer is provided with a distribution panel to feed lighting panels located near the areas served.

2. Communications

Three separate communication systems are provided as follows:

- (1) An automatic dial telephone and integrated public address system.
- (2) A sound-powered telephone system.
- (3) An intercommunication system.

The automatic dial telephone and public address system is an extension of the existing system at the site.

The support wing, reactor building and yard stations consist of desk-or wall-mounted handsets. The inside and outside cell stations are jack-type suitable for portable headsets. The cell stations are on a single line common to all the cell stations as well as to a two-way speaker in the control room. This common circuit permits communication between any or all of the cell stations and the control room.

A public address feature is an integral part of the automatic telephone system. The facility is divided into areas each covered by a separate speaker station. Any one of the speaker stations is available for use in public address by dialing the area number from any telephone station. An emergency number will be provided to combine all speaker stations to alert the entire facility.

A telephone cable is provided to the FARET telephone cabinet from Building 752 at the EBR-II site. The cable is supported by messenger cable on one of the 13.8-kV pole lines from EBR-II.

The sound-powered telephone system consists of three circuits with jacks for portable sound-powered equipment. These are located at various points through the plant, such as in the cell, basements, and other operating areas.

The intercommunication system is a separate multi-station system linking the control room with the cell interior and the air locks. The air locks and control-room station are each provided with a single two-way speaker and the cell station has sufficient two-way speakers to permit pickup from any point in the cell.

3. Grounding

The equipment ground system consists of a buried cable grid underlying the building foundation and the facility yard. The FARET grid is connected to the EBR-II ground system by a cable buried along the 13.8-kV pole lines to EBR-II. The grid is also connected to the well casing at the FARET facility. A similar well casing connection exists at EBR-II. Building columns and all electrical equipment and control boards are grounded to the grid. A separate ground is provided for nuclear instrumentation channels.

All exposed non-current-carrying metallic parts of electric equipment, raceway systems, building structural steel, metallic frames of major equipment, vessels, tanks and crane rails, are grounded. Ground connections are made directly to the ground mat or to the nearest building steel. Brazed or thermite process connections are employed throughout the ground grid. Structural steel columns are connected to the grid, and ground conductors are bonded to steel by the thermite process.

The nuclear instrumentation ground system is independent and insulated from the main grounding system over the full length of the connection from the control board to the FARET well. Ground cable for the main grounding system is bare, stranded, soft-drawn copper. The nuclear instrument ground cable is single conductor, stranded copper, Type TW, 600-volt.

4. Cable, Wiring and Raceways

a. Power Cables

Cables for 2400-volt and 13800-volt service have ozone-resistant butyl rubber insulation with a temperature rating not less than 85°C. Conductors are tinned, soft-annealed copper with semiconducting tape applied over the conductor. PVC jackets are provided for mechanical protection. The cables conform to the applicable provisions of IPCEA S-19-81, ASTM D1352 and ASTM D1047. Cables for 2400-volt service have an ungrounded neutral rating of 3000-volts. Cables for 13800-volt service have a 15000-volt ungrounded neutral rating, have semiconducting tape applied over the insulation and are shielded with tinned copper tape.

Power cables for 480-volt and lower voltage service, except cables exposed to high radiation, have 600-volt insulation, and are of tinned, soft annealed stranded copper with Class C stranding. Minimum size of conductors used in power circuits are No. 12 AWG. Single conductor cables have heat- and moisture-resistant rubber insulation type RHW and neoprene jacket. Multi-conductor cables have heat- and moisture-resistant rubber insulation with PVC or neoprene jacket. Power cables exposed to high radiation are mineral insulated.

b. Instrument and Control Cable

Control cables and general-purpose instrument cables, unless exposed to high radiation, are 600-volt grade, polyethylene insulated with PVC jacket. Minimum conductor size is No. 14 AWG copper. Multi-conductor cables are color coded. Cable construction conform with IPCEA S-61-402. Thermocouple cables, unless exposed to high radiation or high temperature, have PVC insulation with PVC jacket Thermo-Electric type P/ALPTwK or equal; the conductors are Chromel-Alumel twisted in pairs and metallicly shielded under the jacket. Multi-pair thermocouple cables have a minimum conductor size of No. 20 AWG. Conductors are color coded and numbered for identification. Single-pair thermocouple cables have a minimum conductor size of No. 16 AWG.

c. Mineral-Insulated Cable

Mineral-insulated cables with metal jackets are used wherever possible for wiring exposed to high radiation. Power, control and instrument cable, except for thermocouple extension cable, are type MI, 600-volt, in accordance with the NEC, and consist of solid copper conductors insulated with highly compressed magnesium oxide encased in a solid copper outer sheath. Thermocouple extension cable is single pair, No. 20 gage, Chromel-Alumel, with highly compressed magnesium-oxide insulation encased in a solid copper-nickel alloy sheath, 1/4 inch OD.

d. Raceways

Conduits are galvanized rigid steel conforming to ASA C80.1 and are normally run exposed. Conduits are installed to prevent collection of trapped condensation wherever possible. In conduit runs below ground level, couplings and conduit threads are covered with a lead-base pipe-joint compound before coupling to provide a water seal. Underground conduits are encased with a minimum of 3 inches of concrete on sides, bottom and top with minimum depth of cover of 18 inches below finished grade.

5. Lightning Protection

The steel stack is connected to the ground grid to provide lightning protection for the facility. Lightning rods are provided on the stack to supplement the shielding effect of the stack.

6. Evacuation Warning

A siren is mounted on the reactor building roof for area evacuation alarm. Horns are provided within the facility to sound in

conjunction with the siren. The evacuation warning alarm is initiated by manual operation of a control switch located in the control room and a signal is transmitted to EBR-II.

7. Pipe and Equipment Heating

Sodium piping and equipment is provided with a heating system to preheat before filling with sodium or to maintain sodium in the liquid state when no nuclear heat is available, but normal heat losses are taking place.

All sodium piping can be held at 600°F by pipe heating alone. Electrical heating is installed in process equipment to assist raising the system temperature to 600°F.

Heating of piping and equipment for sodium is accomplished by electric heaters in direct contact with the heated surface. The heaters are resistance type with tubular stainless steel sheaths and are clamped or strapped directly to the pipe or equipment before insulation is applied.

For control purposes the piping runs are divided into sections approximately 12 ft long. Each section is provided with a heater and a temperature-sensing thermocouple. Power to the heater for each section is controlled by an individual contactor located in the pipe preheat control switch cabinet. Contactor operation is controlled by the associated thermocouple in conjunction with a temperature controller located in the Temperature Monitoring and Preheat Control Board.

Heater power is supplied from a 480-volt load center and is distributed to the pipe preheat control switch cabinet at 480/277-volts, 3-phase, 4-wire. The individual heaters operate at 277-volts single phase.

Equipment heaters are controlled similarly to the piping heaters except that large volume equipment with correspondingly heavier electrical loads are fed directly from the 480-volt load center. Each feeder circuit for these heavier loads is controlled by a contactor in the load center which supplies power to a local circuit-breaker heater panel near the heated equipment. The load is divided into several sections, each supplied by a sub-feeder circuit from the local heater panel. This load subdivision permits manual heat adjustment at the local heater panels.

Spare heaters and thermocouples are provided for all inaccessible sodium piping and equipment.

X. MISCELLANEOUS SERVICE SYSTEMS

A. Argon Systems

1. General

Argon is used as a blanket gas in all systems containing sodium or NaK. In addition, argon is used as the atmosphere in the cell, cavity, and vault during normal operation.

Non-radioactive liquid metal gas blanket systems are vented through vapor traps directly to the atmosphere. Radioactive liquid metal system venting is routed through vapor traps and filters to a gas retention system for controlled release to the stack via additional filters. Venting of the containment region atmospheres is via filters directly to the stack unless high activity levels are sensed, at which time automatic switching will route these materials to the retention system.

2. Supply and Distribution

Liquid argon is supplied and stored on site. Suitable storage provides a reserve available at all times. The liquid is expanded through pressure control to an evaporator. The gas is routed by a 100 psig header to the appropriate pressure reducing stations which lead to the blanket gas and gas atmosphere systems as shown in Fig. 90.

3. Blanket Gas Systems

Gaseous argon is piped to all equipment requiring blanket gas (Fig. 91) such as the wet fuel storage tank, the reactor vessel, all drain and expansion tanks and traps of the primary, secondary and auxiliary systems. Reducing valves and pressure controllers, where required, are provided on each of these branch lines to assure maintenance of the proper pressure in each of the sodium or NaK systems.

4. Cell, Vault and Cavity Atmosphere Systems

Argon is used as the atmosphere in the cell, cell annex, and cavity and in the equipment vault (Fig. 88). Reducing valves and pressure controllers maintain the respective pressure of argon in each of the volumes. Refrigeration regulates the temperature in each volume to 100°F under normal conditions and to about 165°F during power outages. Replacement of argon with air or vice versa is accomplished by gas displacement. Impurities concentration (where argon is the desired atmosphere) is controlled by purging with high purity argon.

An independent gas cooling circuit is provided for each of the containment areas (annex, cell, cavity vault) as follows:

1. Annex (Fig. 110)

a. A standard commercially available air handling unit consisting of a centrifugal fan (1000 CFM) and a flooded type finned tube evaporator is provided and is located in the annex.

b. Gas is drawn from the vicinity of the reactor head and is filtered before being cooled and discharged near the ceiling of the annex. In this manner gas always flows down the head lifting bellows towards the potentially contaminated area around the control rod penetrations.

2. Cell (Fig. 110)

a. Two half capacity cooling units are provided for the cell. These units are located within the annex and each consists of a vane axial blower, (6500 CFM) a flooded type fin tube evaporator, a back draft damper and interconnecting ductwork.

b. Since it is possible for the cell to be operating with an argon atmosphere at the same time the annex is filled with air, it is necessary for the cell cooling units and ductwork to be of leak-tight all-welded construction.

c. Warm gas is drawn from a distribution header at the top of one long wall of the cell, is pulled through the fin tube evaporators, and then distributed through a similar header along the other long wall. These distribution headers are triangular in shape with two sides formed by the cell liner and the third by a steel plate. Grilled openings are provided in this plate to aid in cooling distribution.

d. In case of loss of normal power supply, one of the two cell cooler blower motors is automatically switched to the diesel generator emergency power supply.

3. Cavity (Figs. 111 and 112)

a. The cavity cooling system consists of a filter plenum, bag type filters, two large vane axial fans (one operating - one standby) two backdraft dampers and interconnecting ductwork. The equipment is designed for a flow rate of 51,000 CFM and is located in a pit in the southeast section of the vault.

b. The reactor cavity cooling gas flow circuit is as follows: Gas is drawn from the vault down into the area around the sodium storage tank (T-4); then ducted to the cavity directly below the reactor vessel; then distributed to the annular space between the reactor vessel insulation and the blast shield; then flows upward through this annulus; then downward through an annulus formed by the other side of the blast shield and a sheet metal baffle; then is ducted to the filter plenum; then flows through the filters, through the vane axial blower and is discharged back to the vault.

c. The wet fuel storage cavity gas flow is as follows: Vault gas (from the tank T-25 pit area) is drawn into a plenum under the wet fuel storage cavity and is distributed to the cells of the metal roof decking that lines the walls of the cavity; the cooling gas flows upward through these roof deck cells, then into the cavity proper, from where it is ducted to filter plenum as described above for reactor cavity cooling.

d. Medium range efficiency filters (bag type) have been provided for the purpose of maintaining the vault relatively free of activated particles which may be carried from the reactor cavity regions.

e. In case of electrical power failure the cavity cooler fan motor is automatically switched to the diesel generator emergency power source. Should the operating cavity cooler fan fail, for any mechanical or electrical reason, the standby fan will be automatically energized.

4. Vault Cooling System (Figs. 111 and 112)

a. The vault cooling system consists of five (5) standard commercial air handling units. Each unit is rated 30,000 CFM and consists of one flooded type ammonia finned tube evaporator coil and three centrifugal blowers. These standard units will be specially altered to the extent that each of the three blowers will be driven by its own motor rather than all the blowers being driven from a common shaft and a single motor.

B. Cell Gas Purification

1. General

The function of the cell gas purification system is to maintain a low oxygen concentration in the cell argon atmosphere. The main requirement is that gas purity must be maintained within reasonable limits to minimize oxide build-up in the primary sodium system during fuel handling periods when the sodium in the reactor vessel and wet fuel storage tank are exposed to the cell environment.

The purification system is also subjected to intermittent peak loads that depend on the frequency of personnel access to the cell. Admitting a single individual into the cell, wearing breathing apparatus that exhausts into the cell atmosphere, would cause the oxygen concentration to reach 100 ppm after ~24 min. See Fig. 113. After about 4 hrs equilibrium would be established. Under these conditions the purification system would be capable of holding the oxygen concentration at ~400 ppm. Permitting two individuals in the cell simultaneously and providing cooling air at the rate of 360 ft³/hr/individual for their rubberized protective clothing would be typical of a "peak load" situation. Under this condition the oxygen level would reach a peak value after ~5 hours with the purification system maintaining equilibrium at ~0.6% oxygen in the cell.

In making this analysis the following conditions were assumed:

1. The purification system had reduced the oxygen level in the cell to $\sim 0.73 \text{ ft}^3$ ($\sim 17 \text{ ppm}$). The purification rate was taken as $400 \text{ ft}^3/\text{min}$ and the return gas from the purifier contained about 10 ppm oxygen.
2. The air inleakage into the cell was considered to be $\sim 14 \text{ ft}^3/\text{day}$.
3. Personnel entry occurs at t_0 . The person or persons are wearing rubberized protective clothing supplied with air for cooling and breathing or for cooling alone and a Scott type Air Pak is supplied for breathing.¹ In either case a total of $\sim 400 \text{ ft}^3$ of air/hr is being supplied and simultaneously this same amount is released into the cell.
4. This volume, 400 ft^3 , plus the normal inleakage is pumped from the cell to maintain a constant pressure in the cell and purification continues under the same conditions stated in (1).
5. The change in oxygen content between the inhaled and exhaled air is neglected.

The curves in Fig. 113 show the rise in oxygen concentration in the cell as a function of time for cases when one, two, or four persons enter the cell. The data is summarized as follows:

Oxygen level in cell	Number of persons entering cell		
	1	2	4
	Time before reaching oxygen level (hrs)		
0.1%	0.6	0.28	0.1
0.5%		2.2	0.8
1.0%			2.25

A second analysis was made neglecting the $350\text{--}360 \text{ ft}^3$ of air needed for cooling a heavy rubberized suit and considering only the normal air inleakage into the cell plus the exhaled air from breathing equipment like a Scott Air Pak. These conditions simulate the case when emergency short time entry to the cell is necessary and only light weight protective clothing is worn. Under these conditions one person might occupy the cell for a period of about 24 minutes, before the oxygen level reaches 100 ppm.

¹Respiratory exchange, based on information in PB 121784 Radiological Health Handbook, is approximately $43 \text{ ft}^3/\text{hr}$ for an average man at work respiring 20 times a minute.

The atmosphere in the cell is maintained at a slight negative pressure to prevent the release of contamination. Although most penetrations are pressurized with argon, some air inleakage is expected. The purification system is provided to remove oxygen and maintain acceptable oxygen concentration levels in the cell atmosphere.

The system comprises a filter, blower, pre-reaction cooler, catalytic reaction bed (where the oxygen is reacted with hydrogen), a post-reaction cooler and a moisture removal column (Fig. 81). A continuous stream of cell gas is routed to the purification system to maintain the cell oxygen level below 1% during reactor operations and to less than 100 ppm when the reactor or wet fuel storage tank covers are removed. The system is instrumented to automatically prevent the discharge of moisture or hydrogen to the cell due to malfunction or maloperation.

2. Design

The argon purification system is rated for the following performance conditions:

Purification System Inlet

Flow	950 lbs/hr
Temperature	100°F
Pressure	11.9 psia
Oxygen Concentration	1½% by volume, maximum
Nitrogen Concentration	5% by volume

Purification System Outlet

Flow	960 lbs/hr
Temperature	100°F
Pressure	12.5 psia
Oxygen Concentration	0.001% by volume, maximum
Nitrogen Concentration	5% by volume
Moisture	-100°F dewpoint (1.5 ppm)

The argon blowers are of the heavy-duty constant-speed centrifugal type. They are dual purpose units in that they are used to supply not only the reactivation air requirements for the purification system, but also the general argon system circulation. These units are supplied with shaft housings equipped with double seals to insure no air inleakage along the shaft.

The two argon coolers for the purification system are located at the blower discharge and at the oxygen removal chamber discharge. Each is designed for full vacuum. Each cooler is equipped with a liquid

ammonia surge tank with controls to maintain flooding of ammonia at all times. Upon loss of normal power, one evaporative cooler and associated pumps and controls will automatically switch to emergency power to permit continuous cooling of critical components.

The oxygen removal chamber is specified as a "DEOXO" unit identical with the units in operation at EBR-II. The "DEOXO" vessel is designed for full vacuum and is constructed of Type 304 stainless steel. Oxygen removal in this system is accomplished by addition of hydrogen, the exact amount being adjustable to insure control of discharge gas. The unit also is equipped with automatic hydrogen cut-off for catalyst bed over-temperature protection. The purification system moisture removal chamber is a single tower containing approximately 300 lb of Type 5A molecular sieve desiccant. This tower is capable of the following performance:

Argon Flow	960 lbs/hr
Inlet Temperature	100°F
Inlet Moisture	1000 ppm + "DEOXO"
	Oxygen-Hydrogen reaction
Outlet Moisture	-100°F dewpoint (1.5 ppm)

The dryer-reactivation heater for the purification system is a heavy duty, hairpin, tubular type. It is stainless steel sheathed and rated for continuous operation at 1000°F sheath temperature. Reactivation air is supplied to this heater at 12 psia and 90°F D.B. and 70°F W.B. The heater capacity is approximately 30 kW and is controlled by a manually adjustable thermostat with high temperature cut-off protection.

C. Cooling Water System

1. General

The cooling water system provides cooling for the condenser for control room air conditioning, the condensers for the containment region refrigeration system, the instrument air compressors and after-coolers, the radioactive vent compressors and aftercoolers, and the reactor cavity biological shield cooling heat exchanger.

The water system is designed to preclude all possibility of water addition to any liquid metal equipment zone. Water is not used for cooling liquid metal handling equipment. No water piping is routed into any of the containment regions or in proximity to any liquid metal systems external to the containment. The latter systems are located in the rear cell area and in the southwest corner of the facility, whereas the water system and the equipment it supplies are in the north end of the plant and support wing basement.

Cooling of the containment region atmospheres and the cell purification system argon streams is performed with ammonia refrigeration; the water system is used to remove the heat from the ammonia. Since the ammonia pressures are normally higher than the pressures in the water system, water entry into the refrigeration system via the heat exchangers is unlikely.

Water leakage from any of the water cooling system is directed to floor sumps, which are provided with automatic starting pumps actuated by high sump levels. High level alarms are actuated in the control room if sump levels exceed the pump start level. A duplex pump system is used in the support wing basement to prevent damage to instrumentation wiring or the heating system if one pump fails or cannot carry a leakage load.

2. Equipment

The equipment used in the water cooling system consists of two full-capacity cooling water circulation pumps (one operating, one standby), one surge tank with automatic makeup, two evaporative coolers (each half-capacity), two water spray pumps (one for each cooler), and one combination spray water supply and drain tank with automatic makeup.

Water received from the various heat exchangers is routed to the evaporative coolers and, after cooling, is returned to the equipment with one of the two circulating pumps (Fig. 93). A surge tank, with automatic makeup, provides a constant suction head for the circulating pumps.

Each evaporative cooler is provided with a spray pump which receives its water from a combination supply and drain tank provided with automatic makeup. The coolers are provided also with air blowers. The air flow to the coolers is modulated with automatic dampers in order to match cooler capacity at prevailing atmospheric conditions to the load. In addition, automatic controls de-energize the water spray pumps and drain the spray water lines when the modulating dampers approach closed positions. Thus, the cooling water system is designed to operate over a wide range of load and atmospheric conditions.

D. Refrigeration System

1. General

An ammonia refrigeration system is provided to remove heat from the cell, cell annex, vault-cavity, cell purification system, and fuel handling machine.

In the containment regions (cell, cell annex, and vault-cavity) heat is released to the atmospheres as heat losses from the high temperature liquid metal systems and from operation of electrical equipment and lighting. Heat from these regions is removed by transfer from the atmospheric media to wet expansion type ammonia coils. Gas flow across the coils is provided by fans or blowers as appropriate.

Another source of heat is the argon purification system. Heat is added to the circulating argon first by compression of the argon and second by reaction of oxygen with hydrogen during the oxygen removal stage. Ammonia coolers are provided immediately downstream of each of the above operations to adjust the temperatures for subsequent processing.

An intermittent heat load is caused by irradiated fuel elements, which may generate significant quantities of decay heat for short periods of time. In fuel handling this heat must be removed to preclude damage to the fuel subassemblies. An argon coolant system is included as a part of the fuel handling system to remove this heat energy. The heated argon from this system is indirectly cooled by an immersed ammonia evaporative coil comprising part of the refrigeration system.

2. Equipment

The ammonia refrigeration system consists of two 125-ton condensing refrigerant units which operate in parallel (Fig. 102). Each of the main units includes:

- a. A suction tank accumulator with submerged subcooling coil and automatic level control
- b. A reciprocating compressor with automatic capacity control and oil separator
- c. A refrigerant condenser cooled with water and with automatic-head pressure control through water flow control
- d. An automatic purge system for removal of non-condensables (small quantities of ammonia may decompose to hydrogen and nitrogen from irradiation)
- e. A liquid refrigerant receiver with level indication and pressure relief devices

Liquid refrigerant flows from the high pressure liquid receivers to evaporator coils located at each of the regions to be cooled. At each coil is a liquid surge drum provided with automatic liquid level control. Ammonia is evaporated in each coil at a rate consistent with the heat load on the coil. The gaseous ammonia resulting from the evaporation is returned to the suction tank accumulator where it is directed to

the compressor. The compressed ammonia is liquefied by cooling in the water cooled condenser and is accumulated in the receiver for return to the evaporator coils.

Although both condensing refrigerant units are normally operating (in parallel), failure of any component in either unit will not interrupt operation of the other. Further, one complete unit will automatically function on emergency power in the event of a general power outage.

3. Design

The major components of the refrigeration system are designed to meet the following requirements.

a. Ammonia Compressors

Rating	125 tons of refrigeration minimum at design conditions
Refrigerant	Ammonia
Suction Pressure	73.32 psia
Discharge Pressure	180.6 psia
Suction Temperature	40°F
Discharge Temperature	90°F

Each compressor has both manual and automatic capacity reduction and unloaded starting control. The automatic capacity reduction will not reduce the compressor to less than 25% of design capacity. Increments of capacity reduction are 25% of design capacity for each stage reduction. The compressors have automatic pressure feed lubrication systems, and in the event of failure the compressor is automatically shut down and the failure is annunciated. Automatic cooling water controls and automatic jacket drain during shut down are provided. Each compressor is equipped with inlet and outlet ammonia pressure and temperature indicators, cooling water outlet temperature indicator, lubricating oil outlet temperature indicator and high temperature switch and alarm, low inlet pressure switch for compressor shut down, and compressor trip alarm.

b. Oil Separators

Each compressor discharge is equipped with an automatic oil separator to remove and automatically return to the compressor oil supply system at least 99.998% of the oil in the discharge gas.

c. Condensers

Condensers for the refrigeration system are water cooled and handle approximately 11,100 lb/hr at 40°F and 39.5 lb/ft³ ammonia. The combined units are designed to handle 250 tons of refrigeration.

d. Miscellaneous

The remaining components of the refrigeration system are more or less standard devices and include such hardware as a high pressure ammonia receiver, liquid ammonia pumps, low pressure ammonia suction traps, and various instrumentation and other devices.

E. Heating System

1. General

A central heating plant (Fig. 100) supplies the heating requirements for both the support wing and the reactor building.

The heating plant, which is in the basement of the support wing, and includes two half-capacity low-pressure packaged hot water boilers including fans, burners and stack, two full-capacity fuel oil heaters and pumps, two full-capacity hot water circulating pumps, and all instruments and controls for fully automatic operation. One fuel oil storage tank with capacity for 30 days normal operation is located inside the building. The boilers are rated approximately 150 hp each and burn No. 5 oil.

2. Support Wing

Support wing heating is provided by finned tube convectors around the building perimeter (Fig. 97); boiler water flow through each convector is regulated by a self-actuated control valve. Supplementary heat is provided by six hot water reheat coils in the air supply ducts for the various zones. Any heating required for the control room and cable spreading room is provided by the hot water heating coil in the control room multi-zone air handling unit.

3. Reactor Building

The reactor building is heated by hot water reheat coils in the supply air ducts; separate reheat coil is provided for the rear cell area and for each floor of the front cell area. Air is first tempered to approximately 60°F by the hot water preheat coils and then heated as required for each area. Each stairwell has a hot water unit heater.

F. Cooling System

1. General

Cooling for the support wing and reactor building is provided by direct contact spray-type air washers. Mechanical refrigeration is provided only for the control room and cable spreading room.

2. Support Wing

Support wing cooling is provided by an air washer in the supply air unit (Fig. 95). Temperature in the various zones is maintained by the reheat coils described in the heating section. To reduce boiler load, a face and bypass damper controls the temperature leaving the air washer.

A 25 ton reciprocating type refrigerant condensing unit furnishes air conditioning for the control room and cable spreading room. This unit supplies "Freon" type refrigerant to the finned tube direct expansion evaporator in the multi-zone air handling unit. Heat is rejected to the recirculated cooling water system in the refrigerant condenser.

An economizer type system incorporated into the design of the control room air conditioning unit provides outdoor air for cooling and eliminates the need to operate the refrigerant compressor during the winter.

The control room and cable spreading room are controlled at $75 \pm 2^\circ\text{F}$ with a maximum of 50% relative humidity to insure the accurate performance of instrumentation located in these areas.

3. Reactor Building

The reactor building is cooled by air washers with face and bypass dampers (Fig. 94); reheat coils provide temperature control.

G. Radioactive Vent System

1. General

The radioactive vent system is provided to handle gases released from (1) the blanketing regions of the radioactive liquid metal systems and (2) the containment system atmospheres. The reactor and wet fuel storage vessel gases can be purged before opening either vessel.

The radioactive liquid metal system blanketing regions, which include the reactor, wet fuel storage tank, NaK annulus system, and fuel transfer system, release small quantities of radioactive (or highly suspect) gases. These gases are routed through absolute filters and compressed for storage and subsequent controlled release to the stack.

The containment regions release larger quantities of gas during pressure control actions resulting from barometric and temperature changes. These gases normally contain very low activity levels and are routed via two absolute filters in series to the stack. If singly filtered activity levels rise above preset values, automatic monitoring instrumentation diverts the materials to compressed storage.

2. Equipment

The suspect gases are routed first through an absolute filter (Figs. 96 and 114) contained in a vessel rated at 30 psig. The filter discharge releases to a 100 ft³ suction tank which feeds either of two parallel installed compressors. Although both compressors can operate simultaneously, one is normally operating while the other is in standby condition. Each compressor is rated at 25 cfm and 125 psig. From the compressors, the gas is routed manually to one of three parallel 100 ft³ storage tanks. Each tank has a pressure alarm which annunciates when the pressure reaches 100 psig, at which time manual routing to one of the other tanks is performed.

Once a tank has been filled with compressed gas, its activity level will be measured by manually monitoring the outside of the vessel; manual sampling connections are also provided on each vessel. When activity is sufficiently low for release to the atmosphere, the compressed gas will be routed manually to a constant flow release valve which directs the flow through one of two parallel absolute filters, from which one of two parallel blowers exhausts the gas to the 200-ft stack. Gases from the containment region atmospheres, cell access locks, and containment gas analyzers are normally routed through the filters and blowers to the stack unless abnormal activity levels lead to automatic diversion to compressed storage. Once in the stack, the release gases are diluted in the large flow of building ventilation exhaust.

H. Radioactive Aqueous Waste System

1. General

The radioactive liquid waste handling system comprises a water collection system from contaminated (or suspect) regions. Such water is directed to either of two storage tanks. These tanks are used as temporary receivers until the suspect liquid can be sampled and pumped either to the tile field or to a truck loading station for disposal elsewhere. Any major decontamination requirements will be handled by supplemental facilities.

2. Equipment

All potentially contaminated water drains are manually routed to one of the two 1000 gal. liquid waste storage tanks (Fig. 97). When a filling tank reaches a near-full condition, an alarm sounds in the control room and flow is manually diverted to the second tank.

For mixing purposes, a full tank is circulated with one of two parallel pumps. A representative sample can be taken from the circulating

liquid and analyzed for activity. Low activity material is pumped to a tile field; above limit material is routed to a truck loading station for disposal elsewhere.

The tanks have a common vent blower which discharges directly to the stack. Inadvertent overfilling of one tank overflows to the other to avoid floor spills or backup of water into the vent system.

I. Shield Water Coolant System

The purpose of the water coolant system for the biological shield is to provide additional cooling -- over and beyond that provided by the argon shield cooling system -- at three locations where streaming and resultant higher than normal shield temperatures are anticipated. These locations are (1) in the vicinity of the nuclear instrument thimbles, (2) along the west side of the west pipe gallery and (3) along the east side of the east pipe gallery. The former is opposite a window in the neutron shield to permit a higher flux for the nuclear sensors. The latter two are opposite the principal reactor vessel coolant nozzles.

The cooling system (See Fig. 115) is comprised of a cooling water pump, two sets of cooling coils (the west cooling loop and the east cooling loop; each with a spare coil in parallel), a plant cooling water-to-shield cooling water heat exchanger, and a surge tank. Make-up water to the system passes through a demineralizer.

Cooling water is circulated through pipe coils embedded in the concrete. The coils are fabricated of one-inch diameter extra heavy carbon steel pipe, welded at the joints to reduce the probability of leakage. Since the coils are located outside the containment liner, leakage of water due to a pipe coil failure cannot enter the containment region. Instead, the water would leak to the subterranean or into the front cell area basement. Alternate sets of cooling coils are provided for continued cooling capability.

Prior to encasing the coils in concrete, the coils will be hydrostatically tested at 1.5 times the design pressure. The design pressure is several times the operating pressure.

If one coil in a loop should develop a leak, the loss of water would be detected by a change in liquid level in the surge tank. The spare loop would then be put into operation, and the leaking coil drained and plugged. The amount of water which would enter the concrete mass should be only of the order of several gallons.

The horizontal centrifugal pump has a capacity of 10 gpm at a differential head of 50 ft and is driven by a 3/4 hp 480-volt, 3 phase, 60 cycle motor.

The shell and tube type heat exchanger transfers heat (36,000 B/hr) from the shield cooling system to the plant recirculated cooling water system.

The mixed bed demineralizer will provide water with a maximum of 1 ppm total solids for the initial system fill and for subsequent water make-up, at a maximum flow rate of 25 gph. The demineralizer is provided to minimize the introduction of particulate material which could become activated and deposit on equipment and external piping.

The shield cooling expansion tank is approximately 1 ft in diameter by 3 ft long and is equipped with a liquid level controller and vent and overflow connections. Low level in the tank causes a valve to open allowing make-up water to enter the system from the building service water system (through the demineralizer), while high level causes an alarm to sound. A continued increase in system water inventory causes water to rise to the tank overflow level after which any additional increase is drained through the overflow to the liquid waste tanks. The expansion tank is vented to the suction of the liquid waste vent fan.

The chemical injection tank is approximately 4-in. in diameter by 1 ft long and is designed for the injection of anti-corrosion chemicals, such as hydrazine into the system. The coolant will be analyzed at regular intervals to determine the need for addition of anti-corrosion chemicals or reduction in solid content. Solid content can be reduced by draining a portion of the coolant (to the radioactive waste tanks), thus bringing in fresh make-up water through the demineralizer.

APPENDIX A
FARET Design Criteria

Part I. General

A. DESIGN OBJECTIVES

1.	Purpose	Versatile experimental facility to test neutronics and fuel performance of small high power density reactor.
2.	Maximum Reactor Power Level	50 MWt at 960°F min inlet and 240 ΔT.
3.	Maximum Core Outlet Temperature	1200°F (mixed mean).
4.	Core Outlet Operating Temperatures	a. 10,000 hrs @ 1100-1200°F.
b. 40,000 hrs @ 1000-1100°F.
c. 40,000 hrs @ 600-1000°F.
d. Remaining hrs less than 600F -
@ a 15 yr life.
5.	Maximum Core Size	301 subassemblies.
6.	Maximum Accessibility	Capability for removal of anything in cell or in the reactor vessel.
7.	Reactor Fuel Loading	Direct viewing (Future TV installation possible).
8.	Reactor Vessel	a. Cover removable from the top.
b. Replaceable extension piece for variation of penetration patterns.
c. Subassembly tops exposed to direct view with vessel cover removed.
d. Replaceable internals including grid.
e. Four rows of EBR-II type canned graphite shielding assemblies replaceable with internals.
f. Double wall construction with minimum temperature differential across walls.
9.	Sodium Level in Vessel	Capability to drain sodium to maintenance level during fuel handling whenever necessary.
10.	Coolant Flow thru Core	Upward.
11.	Maximum Coolant Flow	a. 2.99×10^6 lb/hr thru reactor vessel.
b. 2.374×10^6 lb/hr thru core only.
12.	Flow Control	10-100% full flow with 85 psi ΔP core @ 100% flow.

B. OPERATIONAL REQUIREMENTS

1.	Reactor Control	From control room.
2.	Fuel Handling Control	From console in front cell area.
3.	Test Instrumentation	From control room or in front of cell.
4.	Rear Cell Area Floor Elevation	104.5 ft max.
5.	Cell Interior Width	20 ft.
6.	Cell Interior Length	68 ft.
7.	Cell Interior Height	37 ft.
8.	Cell Annex	24 ft wide x 18 ft long x 15 ft high (interior).
9.	Cell Annex	Stationary crane to carry 35 ton load (head, bellows, etc.) supported from the roof of the cell annex over the centerline of the reactor.
10.	Operations Area	A minimum of 12 ft wide passageway outside the east wall of the cell at an elevation of 104.5 ft is necessary for cell operation, with a preferable width of 15 ft. There should be no steps or elevated obstructions in access routes through the 104.5 ft operation area.
11.	Pipe and Conduit	Penetrations to the cell are required.
12.	Fuel and Equipment Port	Required over the fuel transfer tunnel.
13.	Equipment Port Size	8 to 9 ft in diameter. (minimum)
14.	Fuel Port Size	14 in. in diameter. (minimum)
15.	Port Assembly	To accommodate the fuel cask adaptor unit.
16.	Seals	Double seal type arranged for testing at 30 psig and for monitoring at 1 psig or less.
17.	Seal Maintenance	Without personnel access to the cell.
18.	Equipment Transfer Unit	Flanged attachment for shield plug required for attaching equipment transfer unit.

19. Access

- 19.1. Normal access Thru one door from the support wing. All other personnel doors for emergency exit only. A locked door between the truck-loading and front cell area is required.
- 19.2. Control room a. Access doors from corridor will be double door width.
b. Emergency exit door required on the north wall at west end.
- 19.3. Cell a. 48 hours after shutdown after full power operation for one month.
b. Two personnel locks from rear cell area, one near each end of cell. Removable concrete blocks may be used for shielding. No stairs, ladders, or platforms on cell side of air locks.
c. Prohibited during operation.
d. Thru 2 personnel locks. Lock sizes: 7 ft dia with 5 ft x 2 ft 6 in. door; access to cell annex by 7 ft 0 in. dia shell with single 2 ft 6 in. dia hinged port. Shell is suitable for lock.
e. Thru 10 ft 0 in. x 24 ft 0 in. ceiling equipment hatch for large items.
f. Instrument and service leads to reactor vessel thru cell wall.
g. Fuel thru fuel ports to tunnel.
h. Equipment thru equipment port to tunnel.
- 19.4. Equipment maintenance access Removal of equipment from cell and vault required.
- 19.5. Vault a. Prohibited during operation.
b. Eight equipment hatches.
c. Thru port in equipment barrier and ceiling equipment hatches.
d. Access frequency--once per year.
- 19.6. Front cell area Unlimited thru support wing corridor.
- 19.7. Rear cell area Limited truck entrance or change room.
- 19.8. Cask handling a. Spent fuel cask port and truck door to be located such that door can be

- 19.8. (Contd.) closed during cask handling and loading operations.
- b. A plug between 8 and 9 ft diameter is required in the floor of the cell which can be lowered into the transfer tunnel by an appropriate mechanism. The tunnel floor is to be at elevation of 68 ft to provide a minimum floor to ceiling clearance of 27 ft.
- 19.9. Radioactive liquid waste tanks Access from rear cell basement areas.
- 19.10. Instrument air compressors Access from the front cell basement area.

C. OCCUPANCY

1. Number of personnel 38
2. Technical support area Approx. 7200 sq ft.

D. INTERFACILITY CONNECTIONS

1. Services
- 1.1. Domestic water Supplied from EBR-II.
- 1.2. Electrical power Supplied from EBR-II.
- 1.3. Radioactive waste disposal NRTS.
- 1.4. Construction power Supplied from EBR-II.
- 1.5. Warehouses NRTS.
- 1.6. Fuel shipping casks ANL or commercial.
2. Communications
- 2.1. Fire alarm system NRTS.
- 2.2. Telephone NRTS.
- 2.3. Intercom EBR-II.
- 2.4. Accident alarms EBR-II.

E. SITE DATA

1. Facility location About 4000 ft SE of EBR-II.

2. Design Ambient Temperatures

Minimum	-20°F
Maximum dry bulb	91°F
Maximum wet bulb	55°F
Total annual degree days	8500

3. Plant Mean Elevation 5170 ft NRTS datum.

4. Water Supply (EBR-II, well #1)

Temperature	65°F
pH	7.7
Total solids	192 ppm
Alkalinity	
Carbonate	0 ppm
Bicarbonate	149 ppm
Hardness as CaCO ₃	120 ppm
Dissolved solids	192 ppm

F. ECONOMIC FACTORS

1. Design Life 15 years.
2. Use Factor 50% at average power = 25 MWt.
3. Annual Charge Rate 5%
(for design evaluation only)
4. Average Operating Labor Cost \$3.32/manhour
5. Electrical Power Cost \$0.0063/kW hr
6. Fuel Oil Cost
 - 1) #2 oil \$0.139/gal
 - 2) #5 oil \$0.095/gal
 - 3) Diesel \$0.126/gal
7. Argon Cost \$0.05/scf

Part II. Nuclear

A. SAFETY

1. Seismic Considerations

1.1. Location designation

Zone 2 per uniform building code.

1.2. Seismic requirements

- a. Design for twice zone 2 seismic requirements. Superimposed on other applicable loads such as operating loads, life and dead loads and loads due to design and transient plant conditions.
- b. Design conventionally for zone 2 seismic requirements only if unimportant to safety.
- c. Accident loadings not to be considered simultaneously with seismic loadings except that containment will be designed to withstand temperature and pressures following maximum postulated accident (including the loss from sodium expulsion from reactor vessel) coincident with seismic design loadings.

1.3. Vibration

The primary system components including the reactor vessel, heat exchangers and pumps, will not have a fundamental period between 0.03 and 1.0 sec.

1.4. Dynamic analyses

Required for critical sodium piping to assure that deflections and stresses are appropriately considered.

1.5. Design assurance

- a. Dynamic analyses need to be determined by recognized expert after reviewing structures and equipment important to safety and required to remain functional during and following an earthquake.
- b. Criteria applicability to be evaluated by recognized expert.

2. Containment Structure

2.1. Scope

Except for sodium sampling stations, argon purification system, etc., the entire reactor primary system, together

- 2.1. (Contd.) with all auxiliary systems which may contain radioactive process fluids, will be included within containment structure.
- 2.2. Contents Containment structure to include cell, cell annex, reactor and wet fuel storage tanks and equipment located in the vault.
- 2.3. Environment To be housed within conventional building which is required to maintain sub-atmospheric pressure environment for containment structure to prevent ground release of radioactivity which may be released from containment structure.
- 2.4. Exhaust system Air from conventional building to be monitored and discharged through a 200 ft stack and filtered prior to entering stack in the event of high radioactivity.
- 2.5. Filters Highest efficiency type obtainable. Periodic testing required.
- 2.6. Blast effects Containment to accommodate missiles and blast effects resulting from the equivalent of 40 lbs TNT explosion at center of reactor core.
- 2.7. Inert atmosphere To be maintained within containment structure during reactor operation, fuel handling, or whenever forced sodium coolant circulation is required.
- 2.8. Testing
- a. Initial test, full design pressure and subsequent lower pressures.
 - b. Maximum leakage not to exceed 30 w/o of containment volume per day at 30 psig or less.
 - c. Penetrations--All penetrations not permanently installed to permit periodic testing at full design pressure to reconfirm leaktightness.
- 2.9. Monitoring Inert containment atmosphere for oxygen content.
3. Reactor Vessel
- 3.1. Construction Double wall.
- 3.2. Materials 304 stainless steel.

3.3.	Leakage monitoring	NaK annulus system level to be monitored continuously for sodium in-leakage from vessel.
3.4.	Vessel wall failure	Outer vessel wall able to contain both sodium and NaK in the event of inner vessel wall failure.
3.5.	Vessel penetrations	All nozzles and penetrations to be located above level of core.
3.6.	Inadvertent draining	Not possible by valve manipulations or malfunctions for vessel to drain below core level.
3.7.	Cover interlocks	Mechanisms and instrumentation required to insure control rods disengaged and fully inserted in core prior to removal of vessel cover.
4.	<u>Core</u>	
4.1.	Reactivity coefficient	Overall total reactivity coefficient to be negative under all conditions.
4.2.	Recriticality	Suitable deflection devices required to protect against recriticality following major accident.
5.	<u>Control Rods and Drive</u>	
5.1.	Type	Vertical adjustment type in-reactor control rods.
5.2.	Control	Only one (1) rod to be moved at any time to increase core reactivity.
5.3.	Reactivity rate	Maximum insertion rate no greater than 5¢/sec.
5.4.	Rod scram	By gravity assisted by pneumatic pressure.
5.5.	Malfunctions	System to be designed so that no single malfunction or operating accident (including effects of sodium spills) can cause movement of more than one (1) control rod in a manner to impart reactivity.
5.6.	<u>Shutdown margins</u>	
5.6.1.	Stuck control rod	System to shut down reactor from most reactive condition at that point of operating cycle with minimum shutdown

- 5.6.1. (Contd.) margin of one percent $\Delta k/k$ assuming one rod stuck in full-out position.
- 5.6.2. Minimum margin Not less than maximum worth of a single fuel subassembly.
6. Cooling System
- 6.1. Scope Two sodium systems; a main system for 50 MW(t) and an auxiliary system to remove decay heat and normally operated at low flow standby. Each system to include a primary and secondary loop separated by an intermediate heat exchanger.
- 6.2. Main primary loop capability EM pump delivering 6% of main loop rated flow and energized from battery source with sufficient life (about 15 min) such that the core will be cooled and remain covered (to prevent fuel cladding failure due to overheating).
- 6.3. Auxiliary system To serve as operational standby for removal of decay heat when main system is out of service.
- 6.4. Dependability Core to be cooled and remain covered in the event of either single wall failure of any portion of primary cooling system or loss of pumping power.
- 6.5. Backup Main system to provide decay heat removal through natural circulation if all forced cooling is lost.
- 6.6. Makeup Provide constant sodium level in the event of cooling system leaks to the degree that sodium is available in the primary dump tank.
- 6.7. Codes Portions of the reactor coolant system in contact with sodium to be Type 304 stainless steel to applicable ASME and ASA Codes except where design practice calls for a different material.
7. Control and Instrumentation
- 7.1. Safety trips Values to be established for parameters vital to reactor safety, including high reactor power level, short period, low detector voltage, low scram assist pressure, low reactor coolant flow, high rate of

7.1. (Contd.)

change of reactor coolant flow, high temperature in the reactor vessel outlet coolant, power interruption to the primary or secondary pump motors, low primary system sodium level, high reactor blanket gas pressure, and seismic activity.

7.2. Redundancy

All trip functions and safety systems derived from reactor sodium level, flow and temperature instruments to include redundancy and failure in a safe mode where feasible.

7.3. On-line testing

Failed channel or channel being tested to indicate a trip on that channel such that previous two out of three logic becomes one out of two logic.

7.4. Single component failure

Effectiveness of safety system not compromised by single component or wiring failure.

7.5. Testing

- a. System Test: Safety-system instrumentation to demonstrate proper operation at extremes of environmental conditions.
- b. Control Room Test: Safety system instrumentation located in control room to be tested in place demonstrating proper operation at extremes of anticipated operating environmental conditions. Control room temperature for this test to be less than 122°F (specified for instruments) and expected to be about 110°F.
- c. Specification Testing: Specifications to require all safety system instrumentation to operate in ambient temperatures between 5 and 50°C with maximum humidity of 90°. Supplier to furnish certified report of tests that verify proper operation of representative samples of units at these extremes of environmental conditions.

8. Auxiliary Systems

8.1. Containment

Argon cooling system, sodium service and purification system and portions of

- 8.1. (Contd.) radioactive waste gas system to be included within containment structure.
- 8.2. Compatibility of coolants Auxiliary system fluids in containment structure will be compatible with reactor coolant-system fluids wherever possible. Where incompatibility occurs, auxiliary systems will be designed and protected such that contact between incompatible fluids does not result in conditions that violate containment.
9. Fuel Handling and Storage
- 9.1. Visual observation Refueling and other principal cell operations require windows or viewing devices in cell.
- 9.2. Fuel storage tank To be located in cell and provide temporary capacity for up to 151 irradiated fuel subassemblies. Storage to be by immersion in sodium and double-wall tank required.
- 9.3. Storage tank cooling Tank annulus to be filled with NaK normally cooled by external NaK loop. Emergency cooling provided by sodium purification system (trap jacket side).
- 9.4. Storage tank Vessel cooling storage tank to be located such that integrity is not affected by any reactor accident.
- 9.5. Criticality Not possible in storage tank regardless of number and type of subassemblies in tank.
- 9.6. New fuel Storage requirements to prevent criticality under any conditions.
10. Power
- 10.1. Normal plant power Furnish a single 13.8 kV overhead transmission line from EBR-II through transformers to a single 2.4 kV bus and two separate 480 V busses.
- 10.2. Dependability Components whose operation is necessary to effect a significantly reduced consequences of major accidents will be provided with one normal and at least two backup power supplies or equivalent.

10.2. (Contd.)	Each backup to be independently capable of operating components for as long as required.
10.3 Emergency power	Single, automatically started, diesel generator, which is automatically connected to the No. 2 480-V plant-load center and capable of delivering power to critical plant loads (approximately 625 kVA) within 45 sec after plant outage. Fuel supply on hand to provide at least 24 hr service.
10.4. Critical plant loads	Uninterrupted power to safety systems and other critical plant loads by station battery if normal power supply is lost. Power to be available for selected loads in the event of emergency power (diesel generator) failure. Battery to provide 60 cell system continuously operated in full-charge condition for safety system and other critical loads, including portions of fuel-handling system and emergency lighting. Battery system to supply power for at least 30 min of operation. Separate full-charged 125-V battery system will be used to assure uninterrupted power for main primary sodium shutdown pump for at least 15 min.
10.5. Emergency exhaust system	Requires separate emergency power and will be in operation during reactor operations, fuel handling, movement of irradiated fuel or whenever forced cooling circulation is required. Fuel reserve available to insure operating time of 30 days or duration of maximum postulated accident whichever is less.

B. CONTAINMENT

1. Alpha-gamma Containment	Cell
2. Cell, Vault and Cavity Atmosphere	Argon
3. Stack Height	200 ft
4. Convection Cooling Requirements during Accident Conditions	Cooling of the core under any single fault condition (including pipe ruptures) is required without exceeding design maximum

4. (Contd.)

sodium temperature. Two operating primary coolant pumps are required during post scram operation, until adequate natural convection cooling can be established. In the event of power loss to these pumps the shutdown pump will operate.

5. Material Limits

Water and hydrocarbon systems are excluded from the cell and vault except for limited quantities of hydrocarbon where overriding necessity requires and no accidental admission of these materials in large quantities is possible.

6. Missile Impact Loading on Cell Annex Ceiling

6.1. Impact location

Anywhere in a rectangular area bounded by lines located 2.5 ft inside of and parallel to the inner faces of the cell annex walls. The impact force shall be considered as vertical irrespective of impact location. No major missile impact on cell ceiling.

6.2. Missile weight

100,000 lb.

6.3. Net kinetic energy

The impact load is 0.85×10^6 ft lb at the underside of the cell annex roof structure. This energy must be absorbed by the cell structure in bending strain.

6.4. Impact load area

Uniformly distributed over circular area two feet in diameter.

6.5. Design loading conditions and bases

- a. 30 psig internal pressure in cell; design by elastic analysis using 80 percent of allowable stresses for conventional building structures.
- b. Impact load (50 ton missile) concurrent with 30 psig pressure; design by plastic analysis.
- c. Plastic action of the cell structure limited to that portion of structure above a line 3 ft above top of cell annex access penetration.

6.5. (Contd.)

- d. Plastic design in accordance with report dated April 26, 1965, entitled, "Study to Determine Effects of Impact of Missile on FARET Facility Containment Structure (Revised)."
- e. Welds subjected to bending or shearing stresses located away from locations of plastic hinges in frame, insofar as practicable.
- f. Cell structure capable of sustaining 30 psig pressure (steady state) after impact with no further plastic deformation.
- g. Consideration of major missile impacting on cell walls not required.

7. Missile Impact Loading on Cell Floor

7.1. Impact location

Rectangular cell floor area 15 x 37.5 ft, including reactor cavity floor. This area is centered across the width of the cell floor and extends 17.5 ft north and 20 ft south of the vertical centerline of the reactor cavity.

7.2. Missile weight

50,000 lb.

7.3. Impact energy

Net kinetic energy impact load is the sum of energy gained during free fall of 25 ton weight from ceiling plus elastic rebound energy recoverable from design loading condition.

7.4. Load area

Uniformly distributed over a circular area 2 ft in diameter.

7.5. Design loading conditions and bases

- a. 30 psig steady internal pressure in cell; design by elastic analysis using 80 percent of allowable stresses, for conventional structures.
- b. Impact load (25 ton missile) concurrent with 30 psig pressure; design by plastic analysis.
- c. Cell structure to be capable of sustaining 30 psig pressure (steady state) after impact with no further plastic deformation.

8. Energy Absorption Shield

- 8.1. General Reactor vessel surrounded by energy absorption shield to absorb lateral incident energy by plastic strain.
- 8.2. Design description
- 8.2.1. Material Minimum of 8 in. reinforced ordinary concrete with not less than 43,000 lb of circumferential steel outside concrete. Up to 50% of this steel in continuous liner which can serve as concrete form. Remaining circumferential steel in form of individual hoops.
- 8.2.2. Penetrations Suitable yokes with the same elastic plastic properties as basic circumferential steel.
- 8.3. Location and expansion Not less than 1 ft 0 in. between shield and cavity wall. Not less than 1 ft 0 in. between vessel and shield. Shield free to expand laterally.
- 8.4. Thermal insulation and cooling Low density insulation in annular space between vessel and shield. Argon cooling baffle in annular space external to shield.
- 8.5. Negative pressure in cell, vault, and cavity 2.5 in. max.
- 8.6. Cell, vault and cavity leakage Maximum of 0.1% of volume per day with 1 psi differential with all plugs and windows in place. Maximum of 0.6% of contained volume per day at 30 psig internal pressure.
9. Leaktightness
- All seams in cell and vault liner plates leak tight under 2.5 psi soap bubble test for flat areas. Corners and areas not practical for vacuum box test will be "Zyglow" checked.
- 9.1. Leak test Cell, vault and cavity require integrated leak test and structural test.
- 9.2. Seals and penetration test A double seal system to permit test of all seals at 30 psig at any time.

C. SHIELDING

1. Design Data

- 1.1. Primary sodium activity 30 mc/cc.
- 1.2. Operating power, 1676 kW max per subassembly.
- 1.3. Cooling time, days 2.
- 1.4. Operating time Infinite.
- 1.5. Argon-41 activity in vault 0.2 μ Ci/cc.

2. Dose Rates

- 2.1. Maximum dose rate exterior to cell and vault during normal operation 2.5 mr/hr, except in specially controlled areas.
- 2.2. Dose rate at instrument thimbles During instrument handling a portion of the front cell basement area may have a dose rate exceeding 2.5 mr/hr.
- 2.3. Maximum dose rate from noncontaminants in cell after shutdown following normal operation 30 mr/hr 100 hr after shutdown with reactor vessel cover in place.

3. Miscellaneous Shielding

- 3.1. Cold traps To be provided for each trap. Design based on 6000 curie source.
- 3.2. Drain tank Primary sodium drain tank shall be shielded from the vault by one foot of ordinary concrete.
- 3.3. Maximum intensity and hot spots High density concrete shall normally be used wherever space limitations prevent full thickness of ordinary concrete. Hot spot shielding is limited to high density concrete.
- 3.4. Shield plugs Minimum of one step or equivalent to minimize streaming unless special considerations require more; plugs and plugliners will have approximately 1/16 in. maximum clearance per ft diameter up to 3/8 in. maximum.

Part III. Mechanical Systems

A. REACTOR VESSEL

1. General

1.1.	Maximum power	50 MWt
1.2.	Coolant	Sodium
1.3.	Coolant flow	Upward
1.4.	Material	304 SS
	a. Chromium	18-19%
	Nickel	11.5-12%
	Manganese	1-2%
	Silicon	0.5% max
	Carbon	0.04-0.06%
	b. virgin heats prohibited	
	c. ASTM grain size 4 to size 6	
	d. minimum cobalt practicable	
1.5.	Maximum integrated fast flux	3×10^{20} nvt (greater than 1 MeV).
1.6.	Guard piping	Required around 14 in. inlet nozzle and designed for 100 psig.

2. Design Pressures

2.1.	Inner vessel
(internal press)	-15 to 15 psig
2.2.	Outer vessel
(internal press)	-15 to 15 psig
2.3.	Outer vessel
(external press)	0 to 15 psig
2.4.	Downcomer pipes and inlet plenum	100 psig
2.5.	Core support cylinder	15 psig
2.6.	14 in. inlet nozzle	150 psig
2.7.	Inlet downcomer tor-roided chamber and high pressure inlet plenum	100 psig
2.8.	Low pressure inlet plenum	10

3. Design Temperatures

3.1.	Inner vessel	1200°F
3.2.	Outer vessel	1200°F

3.3.	Downcomer pipes, inlet plenums, and torroidal chamber	1050°F
3.4.	Core support cylinder	1200°F
3.5.	14-in. inlet nozzle	1200°F
3.6.	Extension piece and reactor vessel cover	1200°F
4.	<u>Codes</u>	
4.1.	Fixed vessel	ASME Boiler and Pressure Vessel Code Section III, Class A, Reactor Vessel, all applicable code cases under 1331 and 1335, all applicable case interpretation, and requirements of Section VIII for external pressure to bear appropriate code stamp.
4.2.	Vessel internals	ASME Boiler and Pressure Vessel Code Section III.
5.	Creep	Total creep strain on inner wall due to all gravity loads no greater than 0.01 percent over vessel design life. Uniform tensile stress in inner wall due to all gravity loads not to exceed 750 psi.
6.	<u>Vessel Cover</u>	
6.1.	Closure device	Provide relative ease of assembly and disassembly.
6.2.	Bolting	Not permitted for cover.
6.3.	Seal	Double gasket with argon gas space, higher pressure in gas space than in gas blanket inside of vessel.
7.	<u>Extension Piece</u>	
7.1.	Vessel joint	Attachment to vessel to be bolted, seal-welded, helium leak tested and to sustain the same inner pressure as required for the vessel.
7.2.	Replacement	Provide for grinding seal-weld and disassembly joint to remove vessel internals.
8.	Bypass flow	Approximately 20 percent of total inlet coolant to be diverted into reactor vessel annulus around support cylinder for shield and component cooling.

- 9. Vibration Natural frequencies in the range 1 to 30 cps prohibited.
- 10. Welding and Testing
 - 10.1. Radiographs All welds to be fully radiographed.
 - 10.2. Porosity Spheroidal defects and porosity not to exceed 10 percent of acceptance standards of ASME Boiler Code Section III.
 - 10.3. Tests Entire vessel to be pressure tested and helium leak tested.
- 11. Earthquake load factor 0.2 g

B. MAIN COOLING SYSTEM

- 1. General
 - 1.1. Reactor Power and ΔT limited by allowable outlet temperature.
 - 1.2. Power 50 MW with 240°F ΔT across core.
 - 1.3. Power pressure drop 40 to 85 psi at maximum flow (40 psi with low pressure drop fuel assemblies).
 - 1.4. Maximum flow 2.99×10^6 lb/hr thru vessel at 190°F ΔT (2.374×10^6 lb/hr thru core at 240°F ΔT).
 - 1.5. Minimum flow Ten percent of full flow maintaining control with 40 to 85 psi across the core.
 - 1.6. Maximum operating temperatures
 - 1.6.1. Primary system
 - a. Hot leg 1150°F
 - b. Cold leg 1050°F
 - 1.6.2. Secondary system
 - a. Hot leg 1150°F
 - b. Cold leg 1050°F
 - 1.7. Coolant Sodium for both primary and secondary systems.

1.8. Core operating parameters

Coolant Inlet Temp, °F	Mixed Mean Core Coolant Outlet, °F	Mixed Mean Core Coolant Temp Rise, °F	Power Generation, MW
600-960	*624-(1)	240 max	5-50
300-600	300-600	0	0
600-1050	600-1050	0 min	0 min
1050	1050-1200	0-150	0-26.2

*840 min with 50 MW power generation.

(1) as permitted by allowable reactor vessel outlet temperature of 1150°F.

1.9. Coolant temperature control

1.9.1. Isothermal system range 300 to 1050°F with nuclear heating above 600°F.

1.9.2. Rate of temperature change $\pm 50^\circ\text{F/hr}$ max for normal operating conditions.

2. System Transients

2.1. Dry heatup Ambient to 350°F, 50 occurrences in plant life, 150 hr max heatup time, argon in vessel and NaK in vessel annulus.

2.2. Normal heatup Stand-by 250 to 600°F, 1000 occurrences, maximum time of 24 hr, maximum rate 50°F per hr, sodium in vessel and externally heated.

2.3. Normal cool-down 1050 to 700°F, 1000 occurrences, maximum 50°F per hr, sodium in vessel cooled externally.

2.4. Refuel cooling 700 to 300°F, 1000 occurrences, maximum 50°F per hr, sodium in vessel cooled externally.

2.5. Nuclear heatup 600 to 1050°F, 250 occurrences, rate of temperature change 50°F/hr, nuclear heat source.

2.6. Temperature ramp at full power 600°F reactor inlet, 3500 occurrences, rate of temperature change 50°F/hr, final reactor inlet temperature 960°F, reactor outlet temperature 1150°F or less.

2.7.	Temperature ramp at full power	960°F reactor inlet, 3500 occurrences, rate of temperature change 50°F/hr, final reactor inlet temp 600°F, outlet temp 1150°F or less.
2.8.	Power ramp at full flow	600°F reactor inlet, 1000 occurrences, rate of outlet temperature change 50°F/hr (max), outlet temp 650°F or less to start with, full flow.
2.9.	Power ramp at full flow	960°F reactor inlet, 1000 occurrences, rate of outlet temp decrease 50°F/hr (max), outlet temp 1050°F or less to start with, full flow.
2.10.	Reactor scram due to loss of normal electrical power, from full power	960°F reactor inlet temperature 50 times.
2.11.	Reactor scram due to loss of main cooling loop from full power	(Accident case primary shutdown pump unavailable) 50 occurrences.
2.12.	Normal scram from forced convection, full power to 10 percent flow	960°F reactor inlet temperature 400 occurrences.
3.	Materials	Type 304 stainless steel. All materials in direct contact with liquid metals and liquid metal vapors.
4.	<u>Piping</u>	
4.1.	Piping	ASA B31.1 Code for Pressure Piping with Nuclear Cases, except that for temperatures above 1050°F, allowable stresses will be taken from ASME Code tables. No credit is allowed for the difference between the allowable stresses in the hot condition and the sum of longitudinal stresses due to pressure, weight and other sustained external loads in calculating the allowable stress range during steady state conditions.
4.2.	Corrosion allowance	None
4.3.	Accidental draining	Provisions to prevent accidental draining of sodium to the point where adequate core cooling cannot be provided.

4.4.	Double wall piping	To the extent necessary for parts of the system required to assure operation of the auxiliary primary cooling system.
4.5.	Leak detection	Required for reactor vessel annulus and guard pipe annulus. Detect approximately 100 gal leak in either direction.
4.6.	Siphoning	All piping and equipment arranged to prevent siphoning to the extent that reactor vessel or auxiliary system primary loop are compromised.
4.7.	Dry legs	Piping to be arranged to prevent possibility of dry legs in the natural convection cooling circuit when the main pump is stopped.
4.8.	Overflow	Direct connected pipe to reactor vessel.
4.9.	Valves	Manually operated even if located in the vault.
4.10.	Line size	All main sodium lines to have one 4 in. nominal diameter drain from each system low point to drain tank.
4.12.	<u>Remote operated valves</u>	
4.12.1.	Location	Vault.
4.12.2.	Type	Stainless steel bellows.
4.12.3.	Operation	Manual only.
4.12.4.	Penetrations	Provide cover plates and recesses where penetrations are thru rear cell area floor.
5.	<u>Pumps</u>	
5.1.	Type	Vertical centrifugal with internals removeable without disturbing casing.
5.2.	Drive	Electric motor with eddy current coupling, variable speed control.
5.3.	Ramp down time	From full speed to ten percent in 20 to 60 sec (secondary pump speed subsequently manually adjustable).
6.	<u>Primary Heat Exchanger</u>	
6.1.	Type	Shell and tube.
6.2.	Code	ASME Boiler and Pressure Vessel Code Section III, Class A.

6.3.	Tube bundle	Removable.
6.4.	Fouling factor	None.
7.	<u>Secondary Heat Exchanger</u>	
7.1.	Type	Sodium-to-air, finned tube with provision for heating externally.
7.2.	Enclosure	Shutter-type to provide virtually tight enclosure.
7.3.	Special duty	Provide heat input to the secondary and primary systems to raise these systems to temperature level desired for power operation. Sufficient heat input required in combination with other heaters to obtain 600°F from 350°F in 48 hr, or less.
8.	<u>Tanks</u>	
8.1.	Primary drain tank	Sized to contain entire capacity of the primary systems or the unloaded fuel storage tank; whichever is greater.
8.2.	Secondary drain tank	Sized to contain full capacity of main or auxiliary secondary system at maximum operating temperature except main primary heat exchanger secondary sodium which cannot be drained until vault is accessible and temperatures are reduced.
8.3.	Wet fuel cooling NaK drain tank	a. Sized to contain NaK system volume.
b. Pressurize to return NaK to system.
8.4.	Secondary expansion tank	Sized to accommodate a maximum system temperature swing of 240°F without need for fill.
8.5.	Annulus NaK drain tank	Sized to contain entire annulus system at 350°F.
8.6.	All liquid metal tanks	ASME Boiler and Pressure Vessel Code, Section III, Class C design vessels, no corrosion allowance.

C. AUXILIARY COOLING SYSTEM

1.	<u>General</u>	
1.1.	Flow direction	Reversing not allowed.
1.2.	Reactor vessel inlet temperature	250°F min for coolant.

1.3.	Reactor core outlet temperature	Limited to that which will provide 1150° max reactor vessel outlet temperature.
1.4.	Materials	All materials, for piping and equipment in direct contact with liquid metal in systems will be Type 304 stainless steel.
1.5.	Piping	Same as main cooling system.
1.6.	Pumps	Electromagnetic type.
1.7.	Primary heat exchanger	Same as main primary exchanger with 1 MW design duty.
1.8.	<u>Secondary heat exchanger</u>	
1.8.1.	Sodium-to-air	Finned tubes with natural convection air flow.
1.8.2.	Shutters	Required to provide a virtually tight enclosure to reduce heat losses to a minimum when desired. Supplementary heating will be included to insure against freeze-up when on standby.
1.8.3.	Dampers	Required for operating temperature control.
1.8.4.	Design duty	1.6 MW at 600°F reactor inlet temperature.
1.9.	<u>Tanks</u>	
1.9.1.	Secondary expansion tank	Sized to accommodate change in volume over the full temperature range from Na freeze point to 1100°F without need for filling or draining.
1.9.2.	Code	ASME Boiler and Pressure Vessel Code, Section III, Class C design, no corrosion allowance.

D. REACTOR FLANGE COOLING SYSTEM

1.	Design Flow Rate	200 gpm
2.	Design Temperature	1100°F
3.	Design Heat Load	600 kW

E. SODIUM PURIFICATION SYSTEM

1. Primary System Cold Traps

1.1.	Quantity required	One (one additional in future).
1.2.	Design pressure	Sodium side 50 psig; NaK side 40 psig.
1.3.	Design temperature	750°F
1.4.	Operating pressure drop	20 psi max, 6 psi normal, both at 13,700 lb/hr in either case.
1.5.	Operating inlet temperature	600°F max under normal operating conditions.
1.6.	Design flow	30 gpm at 350°F.
1.7.	Sodium flow range	5 to 30 gpm.
1.8.	Filler material	304 stainless steel York knitted wire mesh or equal; wire diameter 0.004 to 0.005 in., bulk density 20 lb/ft ³ .
1.9.	Vessel material	Type 304 stainless steel.
1.10.	Tests	Static pressure per applicable code and helium leak tests.
1.11.	Operating load	Normal procedure to operate with only one cold trap on line.
1.12.	General arrangement	Cold trap crystallizer vessel provided with outer NaK-filled jacket. NaK circulated through jacket for cooling purposes.
1.13.	Pipe connections	300 lb (min) bolted flange sets on sodium and NaK outlet pipes. Pipe both flanges to a point beyond inlet and outlet branch and provide welding ends.
1.14.	Drains	No drains required for sodium. NaK drainable by pressurizing outlet to force NaK out through inlet pipe. Argon gas connections provided for this purpose.
1.15.	Vents	Sodium and NaK portions self-venting to outlet connections.
1.16.	Trap changing scheme	Design for semicontrolled type operations (removal only).
1.17.	Oxide capacity	200 lb

2. Primary Purification
System Economizer

2.1.	Quantity required	One
2.2.	Design pressure	-15 +50 psig
2.3.	Design temperature	1400°F
2.4.	Nominal operating pressure	20 psi
2.5.	Maximum operating inlet temperature	1350°F (when operating with hot traps).
2.6.	Design flow	30 gpm at 350°F.
2.7.	Sodium flow range	5 to 30 gpm.
2.8.	General arrangement	a. Regenerative cooling for cold traps or regenerative heating for hot traps.
b. Either shell and tube or concentric tube type.
2.9.	Connection	Weld ends.
2.10.	Material	Type 304 stainless steel.
2.11.	Testing	Static tests and helium leak tests per code.
2.12.	<u>Heat transfer requirements</u>	
2.12.1.	Cooling service for cold trapping	Shell side receives 30 gpm sodium at 750°F and exits at 350°F. Tube side receives 30 gpm sodium at 300°F and exits at 700°F.
2.12.2.	Regenerative heating for hot trapping	At 30 gpm flow shell side inlet may vary as high as 1110°F with 1300°F exit, and tube side which enters at 1350°F will exit at approximately 1150°F.
3.	Hot Traps	Complete provisions for two units to be installed at a later date.

F. SECONDARY SYSTEM
PURIFICATION

1.	<u>Cold Traps</u>	Similar to primary system cold traps but only one required. Capability for remote removal not required.
1.1.	Economizer	Identical to primary system economizer.
1.2.	Hot trap	Identical to primary system hot trap but only one required.

G. TEMPORARY PURIFICATION SYSTEM

1. Piping Provide connection points in an accessible location to permit temporary installation of one cold trap for initial sodium system cleanup.
2. Space No extra provisions required; rear cell area will be adequate.

H. PLUGGING INDICATORS

1. Quantity Required Two. One each for primary and secondary systems. Accommodations for one future unit for primary system only.
2. Design Pressure 50 psig
3. Design Temperature 1200°F
4. Maximum Operating Temperature 1050°F
5. Sodium Flow Range 0 to 5 gpm
6. Type of Plugging Device Valve with fluted disc.
7. Heat Transfer Apparatus Economizer and sodium to argon forced convection cooler.
8. Material Type 304 stainless steel.
9. Tests Ultrasonic test all heat exchanger tubes, dye check welds, radiograph welds, helium leak test, and static pressure test per applicable code.
10. Heat Transfer Requirements Economizer to lower the temperature of 400 lb/hr of sodium from 600 to 275°F. Cooler to lower the temperature of 440 lb/hr of sodium from 275 to 250°F.
11. Heating Electrical heating capable of maintaining all parts of the plugging indicator at 350°F or above. Heating to be designed so that liquefaction of sodium occurs progressively from a liquid surface.

I. FUEL STORAGE AND HANDLING

1. New Fuel
- 1.1. Storage capacity 42 subassemblies.

1.2.	Location	Floor of rear cell area, using flush type covers.
1.3.	Shielding	Provided by concrete structure surrounding storage tubes and by removable top plugs. Minimum of 12 in. ordinary concrete spacing between subassemblies (no water to be considered in criticality calculations for spacing).
2.	<u>Wet Fuel Storage</u>	
2.1.	Type	Storage tank located in cell.
2.2.	Subassembly lattice spacing	2 $\frac{1}{4}$ in. OD tubes on 5 in. min. centers. Reactor grid orientation of subassemblies must be maintained.
2.3.	Subassembly storage capacity	151 subassemblies.
2.4.	Total heat removal capacity	80 kW
2.5.	Nuclear isolation	Minimum separation between cavity of reactor vessel and cavity of storage tank--12 in. of ordinary concrete.
2.6.	Containment	Wet fuel storage tank to be double wall construction.
2.7.	Size and location	Approximately 7-ft dia and located in cell as far northeast as possible.
2.8.	Tank coolant	NaK in jacket, sodium in tank.
2.9.	System heat exchanger	NaK to air outside.
2.10.	Draining requirements	Sodium to main primary sodium drain tank and NaK to separate NaK drain tank.
2.11.	Design pressures	External 15 psig Internal 30 psig Support structural requirement 2 psig vert. upward and 30 psig downward over OD of cavity seal.
2.12.	Control	The NaK system includes means for heat input, cooling and control to maintain sodium at 300 to 350°F.

2.13. Containment

Tank designed to accommodate free fall of a 200 lb subassembly (lower end at elevation 104 ft at beginning of fall) without damage to vessel.

J. INERT GAS SYSTEM

1. Gas Argon
2. Storage Liquid in tanks.
3. Specification Commercial grade.
4. Quantity 30 day normal demand for cell, vault and cavity as required for one complete charge and makeup thereafter.
5. Distribution Gas at 100 psig.
6. Recovery None. All gas discharged for controlling pressures, bleed and feed, or other uses will be exhausted to stack.
7. Venting Blanket gas vented to waste gas disposal system.
8. Cell Vault and Cavity
 - 8.1. General
 - a. Three separate cooling circuits:
 - 1 - Cell
 - 1 - Vault-cavity (including wet fuel storage tank cavity).
 - 1 - Cell annex.
 - b. Each circuit includes coolers, fans, piping, valves and controls.
 - c. One purification unit for cell circuit only.
 - d. Provision for purging each system.
 - e. Gas displacement system and blowers to exhaust cell, cell annex or vault to stack or radioactive gas disposal system.
 - f. Cell, annex, vault, and cavity also vented through exhaust ducts to stack.

8.2. Design temperature, °F

	<u>Normal</u>	<u>Min</u>	<u>Max</u>
Cell*	100	70	165
Vault	100	70	165
Cell Annex	100	70	165

*Maximum heat load: 75 kW.

8.3. Design pressure, in W. G.
(Referenced to bldg.)

	<u>Normal</u>	<u>Max</u>	<u>Min</u>
Cell*	-2.0	-0.5	-4.0
Vault	+1.0	+1.5	+0.5
Cell Annex	$-1\frac{1}{2}$	0	$-3\frac{1}{2}$

*Maximum heat load: 75 kW.

8.4. Cell gas purity

Condition:	(1) Reactor and Storage <u>Vessel Closed</u>	(2) Reactor and Storage <u>Vessel Open</u>	(3) <u>Personnel Occupancy</u>
Oxygen	1%	0.01%	19% Min
Nitrogen	5%	5%	
Water	1000 ppm	50 ppm	
Hydrocarbons		50 ppm	
Argon			10% Max

8.5. Cell condition changing

- Condition (1) to (2) in 12 hr.
- Condition (3) to (1) in 12 hr.
- Condition (3) to (2) in 24 hr.

8.6. Purification

Oxygen and moisture removed from cell argon atmosphere by catalytic purifier. 200 scfm through full flow filters. Nitrogen removal by bleed and feed. Space provided for future gettering type nitrogen removal system.

8.7. Cell gas displacement

Atmosphere changing by displacement technique. Displacement rate to provide interface movement of 16 to 20 ft per hr for first 6 ft of depth (approximately 15% of total cell volume) at which time the rate shall be a maximum of 1000 cfm. Argon gas removed from the cell will be vented to stack. No provisions for storing exhaust argon.

8.8. Cell ventilation

Approximately 1000 cfm of air maintaining a 2 in. H₂O negative pressure during personnel occupancy.

9. Vault-cavity Atmosphere

9.1. Purity

	(1) During Operation	(2) Personnel Occupancy
Oxygen	1%	19% Min
Nitrogen	5%	
Moisture	1000 ppm	
Argon	Balance	10% Max

9.2. Purity changes

- a. Condition (1) to (2) in 12 hr.
- b. Condition (2) to (1) in 12 hr.

9.3. Purification

Bleed and feed.

9.4. Displacement

Atmosphere changing by displacement technique. No provisions for storing argon gas. The displacement rate to provide interface movement of 16 to 20 ft per hr for the first 6 ft of depth at which time the rate shall be doubled. Argon gas removed from the vault will be vented to the stack. No provision for storing exhaust argon.

9.5. Ventilation

Approximately 1000 cfm during personnel occupancy.

K. GAS COOLING SYSTEM

1. Type

Ammonia refrigeration.

2. Capacity

Two 125 ton condensing refrigerant units operating in parallel.

3. Operating Conditions

1. Suction pressure 73.32 psia
2. Discharge pressure 180.6 psia
3. Suction temperature 40°F
4. Discharge temperature 90°F

4. Limitations

No water piping permitted inside spaces being cooled. All water piping within reactor building monitored for leaks. Provision for collection and disposal of leaks.

5. Equipment Design

Equipment installed out-of-doors adequately protected against freezing and designed for sub-zero operation. Coolers and gas piping prohibit water leakage into gas stream. Provision for monitoring gas stream for total oxygen.

6. Operating Conditions

1. Maximum cell temperature 165°F with one cooler only.
2. Normal cell temperature 100°F with both coolers operating.
3. Vault temperature 100°F at full reactor power.

L. SHIELD COOLING SYSTEM

1. Coolant

- a. Vault argon or air for general cooling.
- b. Water for spot cooling around instrument thimbles and piping penetration hot spots.

2. Temperature

Limit maximum shield concrete temperature to 200°F.

3. Reliability

- a. Use two independent piping loops for all water cooling.
- b. All water circulation equipment accessible during reactor operation.
- c. No water pipes inside of containment liner.
- d. Water system to be closed type.
- e. Forced cooling system and argon blowers to be two identical 100 hp units each capable of handling full system loads.
- f. Argon blowers to be connected to emergency power system with automatic switching to standby unit or emergency power as required.

4. Filters

- a. All gas returning to vault from cavity requires filtering.
- b. Absolute--type.
- c. Replaceable from inside of vault.

M. LIQUID WASTE

1. Purpose

Collect drainage from showers and drains from reactor building sumps and hold for disposal.

2. Decontamination

Liquids resulting from special decontamination to be handled by procedures not requiring this system for storage.

3. Retention Tanks

Two 1000 gal tanks, each fitted with pump and sparger.

4. Disposal Normally discharged to sanitary waste system effluent. Provisions for discharge to tank trucks or trailers for offsite disposal.

5. Venting Through absolute filters to stack.

N. WASTE GAS

1. Purpose Collect vents from main and auxiliary heat transfer primary systems and equipment, wet fuel storage system and equipment, primary purification system, and from spaces operated with inert gas atmosphere when required. These will be monitored and exhausted to stack through ventilation exhaust fan or compressed and stored until activity level permits discharge to stack.

2. Operation

- a. All vent and purge gas from reactor, primary sodium system and fuel storage tank will be stored and sampled prior to release to the stack. Other gaseous waste will be monitored and discharged to either the stack or storage, depending upon the activity level.
- b. All ventilation gas will be discharged to the stack for direct release to the atmosphere. A by-pass filtration system will be provided thru which the flow is automatically diverted in case of high count, even during abnormal operations such as loss of power. The by-pass flow will be filtered thru absolute, high efficiency type, commercial filters with renewable filter elements.

O. PLANT AND INSTRUMENT AIR

1. Purpose Supply dry, oil free air to instruments.

2. Design

- a. Plant air will be supplied to convenient points throughout the facility as required. No instrument air supplied to cell or vault-cavity areas. Argon will be supplied for this purpose.

2. (Contd.)

- b. All equipment will be standard commercial products of established manufacture and design.
- c. Both the plant and instrument air compressors will be oil-free type. The instrument air system will include moisture separators, dryers and filters. Spare instrument air capacity may be provided by cross-connecting to the plant air compressors if required.

P. VENTILATION

1. General

- a. Preheat coils, etc., will be installed and protected so they will not freeze during relatively short periods at -45°F outdoor air.
- b. Established safety practices for the protection of personnel wherever radioactive material may be encountered.
- c. Air from clean areas will flow inward to potentially contaminated areas by differential pressure control.
- d. Flow control will be established by direct mechanically induced ventilation and by design for proper directional infiltration.

2. Operational Requirements

	Temp, °F		Pressure (in. W.G.)
	Max	Min	
Support	80	70	-
Front cell area	80	70	-0.1
Rear cell area	90	60	-0.35
High bay area	105	60	-0.35
Change room	90	75	-0.25
Cable room	80	60	-
Control room	75	70	-

3. Cooling

Refrigeration for the control and cable rooms. Air washers (evaporative cooling) for all other spaces.

4. Heating

Hot water recirculating system with two half-capacity low temperature, low pressure, oil-fired hot water generators, expansion tank, circulation pumps, oil storage, pumping and heating.

5. Makeup Water Make-up water to the hot water generators will pass thru a suitable water softening system.
6. Support Wing
 - a. A slight positive pressure will be maintained with suitable air changes as dictated by good ventilation practice.
 - b. The air supply system will include filters, hot water air heater, evaporative air cooler, supply fan.
7. Front Cell--Rear Cell--High Bay
 - a. The supply air system will include filters, heaters, air washer and supply fan.
 - b. The exhaust air system will use two 1/2 capacity fans, discharging directly to the stack. On high reactivity level, air system inlets will close and in-leakage exhaust air will be diverted thru an absolute filter bank to the stack.
 - c. Minimum ventilation rate will be ASHVE Standard for cfm/sq ft of floor area or air cooling requirements based on heat load, whichever governs.
 - d. Emergency vents will be installed to the stack through filters to relieve positive building pressure.
 - e. Static pressure will be controlled by regulation of the supply system.
 - f. Separate reheat coils and appropriate controls will be used for each floor of the front cell area.
8. Tunnel Ventilation
 - a. The tunnel ventilation system will be designed to prevent the spread of contamination.
 - b. Air will flow from the tunnel operating area through a prefilter into the tunnel and exhaust through the absolute filter at the opposite end of the tunnel.
 - c. The cell ventilation exhaust filter fan assembly suitably valved may be used for the tunnel exhaust.

Part IV. Instrumentation and Control

A. REACTOR

1. Reactor Power (flux) Control
 - 1.1. Power range 500 kWt to 125 MWt (75 MWt over range).
 - 1.2. Method of control Movable control rods.
 - 1.3. Number of control rods 12.
 - 1.4. Rod position indication Read out in control room.
Accuracy ± 0.005 inch.
 - 1.5. Mode of control Manual or automatic.
 - 1.6. Automatic control range 500 kW to 75 MW.
 - 1.7. Automatic control flux level Maintained at manually set point.
Accuracy $\pm 1\%$ of set point.
2. Instrumentation Channels
 - 2.1. Range 0.1 W (Source level) to 150 MW.
 - 2.2. Total number of channels 11
 - 2.3. Minimum overlap between ranges 1 decade.
3. Start-up Range
(LCR--period channels)
 - 3.1. Number of channels 3
 - 3.2. Range 0.1 W to 5 kW.
 - 3.3. Minimum counting rate 10 c/s
4. Intermediate/Power Range
(Log N-Period channels)
 - 4.1. Number of channels 3
 - 4.2. Range 500 W to 150 MW.
5. Linear N. Channels
 - 5.1. Number of channels 1
 - 5.2. Range 500 W to 150 MW.
6. Power Range
 - 6.1. Type High flux protective channels.
 - 6.2. Number of channels 3

- 6.3. Range 500 kW to 75 MW.
- 7. Auto Flux Control Channels
- 7.1. Number of channels 1
- 7.2. Range 500 kW to 75 MW.
- 8. Detectors and Thimbles
- 8.1. Total number of detectors 11
- 8.2. Total number of thimbles 11
- 8.3. Maximum temperature at detectors 250°F.
- 8.4. Access to thimbles From front cell area.
- 8.5. Detector position in thimble Manually adjustable for locating detectors in optimum flux.

B. MAIN SODIUM COOLANT

- 1. General
- 1.1. Signal transmission Electrical
- 1.2. Flow measurement Magnetic flowmeter and flow nozzle.
- 1.3. Pressure measurement NaK filled system.
- 1.4. Temperature measurement Chromel-Alumel thermocouples or resistance type elements.
- 1.5. Level measurement "J" probes for continuous "I" probes for discrete.
- 2. Main Primary Coolant Flow Control
- 2.1. Method of control Variable speed pump drive.
- 2.2. Mode of control Manual or automatic.
- 2.3. Automatic control range 10 to 100% of full flow.
- 2.4. Automatic control flow level Obtained at a manually set point.
- 3. Main Secondary Coolant Flow Control
- 3.1. Method of control Variable speed pump drive.
- 3.2. Mode of control Manual or automatic.
- 3.3. Automatic control range 10 to 100% of full flow.
- 3.4. Automatic control flux level Maintained at a preset ratio of the primary flow.

4. Main Primary Coolant Temperature Control

- 4.1. Mode of control Automatic.
- 4.2. Method of control Temperature is maintained at a preset level by varying the secondary coolant temperature.

5. Main Secondary Coolant Temperature Control

- 5.1. Mode of control Automatic.
- 5.2. Method of control a. Variable speed fan drive.
b. Dampers position control.
- 5.3. Automatic control temperature lead To match a demand signal from the primary coolant temperature.

C. RADIATION MONITORING

1. Area Radiation Monitoring

- 1.1. Number of general area monitoring channels 15 (incl. 4 spare).
- 1.2. Range of general channels 1 mR/hr to 100 R/hr.
- 1.3. Number of controlled access area monitoring channels 10 (incl. 2 spare).
- 1.4. Range of controlled area channels 100 mR/hr to 10^4 R/hr.

2. Process Gas Monitoring

- 2.1. Stack gas monitoring a. Number of particulate activity monitoring channels--1.
b. Number of gaseous activity monitoring channels--1.
- 2.2. Number of building vent exhaust gaseous activity monitoring channels 1

3. Waste Gas Monitoring

- 3.1. Number of blanket waste gas monitoring channels 1
- 3.2. Number of cell, cell annex and vault waste gas monitoring channels 1

4. Miscellaneous

- 4.1. Cold and hot traps and certain major equipment Provided as required.

- 4.2. Reactor wing Portable gas monitor.
- 4.3. Personnel Available.
- 4.4. Cooling water and liquid waste Radiochemical analysis.

D. MISCELLANEOUS SYSTEMS

- 1. Na and NaK Pipe and Equipment Heating System
 - a. Control system function:
 - To raise or lower the temperature at a rate of 50°F/hr.
 - b. To maintain the temperature at a preset level.
 - c. Temperature control range, 300-600°F.
- 2. Cell, Cell Annex and Vault Gas Systems
 - 2.1. Controls are provided to
 - a. Maintain the gas temperature within specified limits.
 - b. Maintain the gas pressure within specified limits.
 - c. Operate gas purification system.
 - d. Operate atmosphere changing equipment for replacing inert gas with air and vice versa.
 - 2.2. Instruments are provided for measurement of
 - a. Percent oxygen.
 - b. Oxygen traces.
 - c. Moisture traces.
- 3. Heating, Ventilation and Air Conditioning
 - a. Maintain temperature in various parts of the facility.
 - b. Maintain pressures in the various parts of the facility.
- 4. Failed Fuel Element Detection System "FERD" type.
- 5. Smoke Detector System Will be provided in all areas containing liquid metal systems including vault.
- 6. Control Room
 - 6.1. Location of controls and switches Most controls and switches essential for reactor, main coolant and decay heat operation and/or shutdown located on the central reactor control board.
 - 6.2. Panel board Empty space provided in the control room for a 15 ft long by 2 ft 6 in. deep

6.2. (Contd.)

panel board for ANL Experimental and Test Equipment.

6.3. Control room floor

Approximately 4 sq ft provided for panel plug-in control stations for the following equipment:

- a. Fuel handling machine.
- b. Control rod drive grippers.
- c. Fuel storage tank cover.
- d. In-cell TV cameras.
- e. Manipulators.
- f. Cell crane.

Part V. Reactor and In-cell Equipment

A. CONTROL ROD DRIVES

1. Space Limitations and Capacities
 - a. Confined to the area above the reactor vessel cover.
 - b. Maximum height ~12 ft.
 - c. Control rod weight ~100 lb.
 - d. Twelve control rods spaced 6.14 in. apart around the perimeter of a nearly circular configuration. Alternate radial distances from core center are 11.60 in. and 12.05 in.
2. Seals
 - a. Metal bellows type between drive shaft and vessel cover.
 - b. Devices employed for sealing shaft penetrations in the reactor vessel cover positively gastight without restricting axial motion of shafts.
3. Material

All components exposed to vessel interior compatible with 1150°F sodium.
4. Motors, Drives, and Power
 - a. Driven by motorized lead-screw and ball-nut mechanism.
 - b. Gripper actuation by motor driven mechanism.
5. Operational Requirements
 - a. Scram from any position of operating stroke.
 - b. Uniform power stroke about 6 in./min (not to exceed 5¢/sec reactivity rate) over 24 in. stroke.
 - c. Scram--1 to 2 g within upper 16 in. of stroke with pneumatic (argon) assist.
 - d. Deceleration within last 8 in. of scram stroke.
 - e. 0.020 sec lapse between electromagnet trip and start of scram stroke.
6. Interlocks
 - a. Electric-interlock of lifting platform to allow movement only during reactor shutdown.
 - b. Mechanical stops to limit lifting platform movement when gripper jaws are closed.

6. (Contd.)

- c. Disengagement of control rods possible only when in "down" position.
- d. Electrical and mechanical safety devices to prevent simultaneous raising of all control rods any significant distance above their operating "Down" elevation.
- e. Each control rod drive capable of positive engagement or disengagement with its control rod.

7. Controls and Instrumentation

- a. Remote control of engagement and disengagement of control rods.
- b. Remote operated sensing device to indicate engagement and disengagement of control rods.

B. COVER HANDLING SYSTEM

1. Space Limitations and Capacities

- Bellows will be octagonal and:
- a. 11 ft maximum across flats outside.
 - b. 9 ft minimum across flats inside.
 - c. 45 ft maximum extended length.
 - d. 4 ft minimum retracted length.
 - e. Nylon tapes limit travel of convolution to 16 in.
 - f. Nylon tapes capable of supporting weight of the extended bellows.

2. Material

Nylon fabric sandwiched between a layer of hypolon on one side and neoprene on the other. 0.018 in. nominal thickness of the material.

3. Motors, Drives, and Power

Stationary crane to lift 35 ton load supported from roof of cell annex.

4. Operational Requirements

- a. Shielding provided to allow personnel access to annex when reactor cover is in "up" position.
- b. Reactor cover raised vertically.
- c. Cover separated from reactor at speed of 1/2 ft/min for a distance of 6 ft.
- d. Cover speed (after 6 ft separation from reactor) 1 ft/min for a distance of 23 ft.
- e. When cover enters cylindrical sleeve in annex floor speed decreased to 1/2 ft/min for a distance of ~11 ft.

4. (Contd.)
 - f. Minor repairs must be possible in place.
 - g. Cable take-up tank will conform to cell containment requirements.
 - h. Electrical and pneumatic lines and connectors to conform to similar services within the cell.
 - i. Bellows will provide gastight flexible connection between reactor cover and annex floor cylindrical sleeve.
5. Interlocks
 - a. Cover hoist interlocked so cover can be raised only when control drive platform and cover closure mechanisms fully unlocked.
 - b. Shutter and in-cell equipment interlocked when cover is in place to prevent shutter movement and in-cell equipment travel in area occupied by reactor cover and bellows.
 - c. Shutter and all in-cell movable equipment allowed full in cell travel only when cover is in "up" position.
6. Environment

Cell argon and intermittent exposure to air or argon diluted with air.
7. Accessibility
 - a. Personnel access thru annex hatch.
 - b. Bellows replacement thru cell ceiling equipment hatch.
8. Controls and Instrumentation
 - a. Annex contains argon atmosphere during reactor operation and fuel handling.
 - b. Argon maintained in cell and annex during movement of cover.
 - c. Argon press, differential across bellows 0.5 in. W.G. min to 6 in. W.G. max.

C. FUEL HANDLING MACHINE

1. Space Limitations and Capacity
 - a. Gripper force upward max--1500 lb.
 - b. Gripper force downward max--200 lb.
 - c. Gripper chain speed--2 in./sec.
 - d. Cooling--3 kilowatts max.
 - e. Coolant--argon.
 - f. Coolant flow--60 scfm max.
 - g. Coolant at inlet--225°F.
 - h. Coolant at outlet--450°F max.

2.	Seals	Lubricated carbon face seals.
3.	Materials	Conventional.
4.	Motors, Drives, and Power	a. 60-cps A.C. torque motors and variable-speed eddy-current drives. b. Emergency power supplied to machine and cooling system.
5.	Operational Requirements	a. Overload protection. b. Stuck subassemblies released by disengaging barrel and lowering into pit using in-cell crane. c. Special gripper provided on fuel-handling machine to permit control rod pickup.
6.	Interlocks	a. Gripper mechanically locked as soon as engagement completed. b. Interlocked gripper will not permit subassembly to disengage. c. Gripper will not release while handling subassembly.
7.	Environment	Argon atmosphere, sodium.
8.	Accessibility	a. Complete removal through cell hatch. b. Machine barrel removal through cell equipment hatch.
9.	Control and Instrumentation	a. Positioning of subassemblies by manually operated controls from outside of cell. b. Position read-out provided outside of cell.

D. FUEL STORAGE TANK COVER

1.	Space Limitations and Capacity	a. About 13 ft in diameter of cell floor space required. b. Two cover access ports.
2.	Seals	a. Argon inflatable. b. Double for pressurization between.
3.	Material	a. Cover--steel. b. Access port plugs--steel. c. Structural components--steel. d. Seals--high-strength fabric coated with silicone.

4. Motor, Drives, and Power
 - a. Reversible ratio electric motor with integral braking.
 - b. Output shaft of motor will contain a torque limiting clutch.
 - c. Maximum cover rotation $1/4$ rpm.
5. Operational Requirements
 - a. Blanket gas pressure reduced slightly when cover access port plug is removed.
 - b. Cover seal inflated and tested for leak tightness after preselected fuel position is obtained.
 - c. Cover seal deflated before cover is rotated.
6. Interlocks
 - a. Cover motor drive locked out during movement of fuel handling machine over wet fuel storage tank and machine barrel and subassembly gripper are not at their top elevation.
 - b. Cover motor drive locked out when seal is inflated.
7. Environment

Cell atmosphere and tank blanket gas.
8. Accessibility
 - a. Replacement of port access plugs remotely by the in-cell crane.
 - b. Replacement of the cover, support bearing, drive unit, and seals requires manual assistance.
9. Control and Instrumentation
 - a. Remote operation includes:
 - 1) Inflation and deflation of seals.
 - 2) Rotation of cover to preselected fuel storage location.
 - 3) Performance of seal leak tightness test by pressurization between the two inflatable seals.
 - b. Remote instrumentation includes:
 - 1) Seal inflation and deflation pressures.
 - 2) Cover position indication.
 - 3) Seal leak tightness pressure drop indication.
 - 4) Cell-tank differential pressure indication.
 - 5) Interlocks indication.
 - c. Redundancy in cover position indication by providing a stationary pointer and a graduated index ring on the cover, viewed through the cell wall window.

E. PENETRATIONS

1. Removable Type Penetrations

1.1.	Space limitations and capacities	Allow space around penetrations to permit testing, service, and internal component removal.
1.2.	Seals	Neoprene, liquid silicone rubbers, plastic resins.
1.3.	Material	Mild steels.
1.4.	Operational requirements	a. Double seals on all penetrations capable of leak testing individually at any time without pressurizing cell. b. Penetration service permitted only when reactor is secured or when control is not compromised by service procedures. c. Penetration seal internal volume to be connected to a continuous argon supply to minimize air in-leakage to cell. d. Penetrations are an extension of containment liner and meet same pressure, leak tightness, and shielding criteria.
1.5	<u>Environment</u>	a. Inside-cell atmosphere, pressure, temperature, and radiation levels. b. Outside standard atmospheric conditions.
1.6	Accessibility	a. Seals replaceable from outside of cell. b. Individual penetration leak testing performed from outside of cell.

2. Permanent Type Penetrations

2.1	Space limitations	Allow space around penetrations to permit testing and service.
2.2	Seals	All welded, brazed or soldered construction.
2.3	Material	Mild steel, copper and ceramics (non-organics).

- 2.4 Operational requirements Initial pressure and leak testing accomplished with cell proof tests.
- 2.5 Environment
 - a. Inside-cell atmosphere pressure, temperature, and radiation levels.
 - b. Outside-standard atmospheric conditions.

F. IN-CELL CRANE

- 1. Space Limitations and Capacity
 - a. 25 ton max load.
 - b. 5 ton or less.
 - c. Hoist speed 2-10 ft/min.
- 2. Material Conventional.
- 3. Motors, Drives, and Power Motor power a.c. 60 cycle with emergency power supplied.
- 4. Operational Requirements Conventional crane type characteristics.
- 5. Interlocks To prevent damage to control rod drives, fuel handling machine, etc.
- 6. Environment Air or argon atmosphere.
- 7. Accessibility
 - a. Bridge removed or installed only through top cell opening; trolley fits equipment hatch and tunnel.
 - b. Maintenance and adjustment performed manually.
- 8. Control and Instrumentation
 - a. Operation from outside of cell.
 - b. Observation through windows.
 - c. Control outlets for operation by personnel inside cell are provided.

G. FUEL AND EQUIPMENT TRANSFER SYSTEM

- 1. Space Limitation and Capacity
 - a. About 10 ft in diameter of cell floor space.
 - b. About 10 ft x 10 ft hatch in floor of rear cell area.
 - c. All of the designated fuel transfer tunnel.
 - d. Transferable loads will be limited by the in-cell crane rated at 25 ton for cell-to-tunnel transfers and by the building crane rated at 40 ton for tunnel-to-rear cell area transfers.

2. Seals
 - a. Fuel port plug seals and port-to-cask seals standard 'O' rings.
 - b. Equipment port cover seals and shield plug seals special 'O' rings.
 - c. Double seals provided in order to test leak tightness.
 - d. Sealing pressure will be 30 psi.
3. Material
 - a. Fuel port plugs and equipment port cover--steel.
 - b. Equipment port shield plug--concrete and steel.
 - c. Structural components--steel.
 - d. Seals
 - 1) Equipment port cover, shield plug, and fuel port plugs--neoprene.
 - 2) Fuel port-to-cask--teflon-coated adiprene.
4. Motor, Drives, and Power
 - a. Electric motor drive with integral braking on cask transfer carriage.
 - b. Hydraulic hoist for cask lifting.
 - c. Electric motor drive with torque limiting clutch on screw jacks support.
5. Operational Requirements
 - a. Cell containment not breached during transfers.
 - b. Seals tested at 30 psig after each re-engagement.
 - c. Casks and containers purged of air with argon.
 - d. Each closure seal pressurized to 12-in. W.G. when in sealing position.
6. Interlocks
 - a. Cell crane locked out over equipment and fuel port area when:
 - 1) Fuel port secondary plug or cask or container is not in place.
 - 2) Equipment port shield plug or equipment container is not in place.
 - 3) Purging gas analysis indicates excessive air content.
 - b. Hydraulic hoist will be locked out when:
 - 1) Cell crane is in operation over equipment and fuel port area.
 - 2) Screw jack supports are not in DOWN position.

6. (Contd.)

- 3) Equipment and fuel port seals indicate an unsatisfactory seal has been made at 30 psig.
- 4) Cask gate is open.
- 5) Equipment or fuel port is open.
- c. Screw jack support is locked out when:
 - 1) Cell crane is in operation over equipment and fuel port area.
 - 2) Equipment and fuel port seals indicate an unsatisfactory seal has been made at 30 psig.
 - 3) Cask gate is open.
 - 4) Equipment or fuel port is open.
- d. Transfer carriage is locked out when under equipment and when fuel transfer port and transfers are being made.

7. Accessibility

- a. Replacement of equipment port cover and fuel port plug remotely by the in-cell crane.
- b. Replacement of the equipment port cover and fuel port plug seals by replacing cover and plugs.
- c. All other items are accessible by personnel from tunnel.

8. Control and Instrumentation

- a. All operations remotely controlled from control room:
 - 1) 30 psig pressure testing of seals.
 - 2) 12-in. W.G. monitoring of seals.
 - 3) Purging of cask and containers with argon or air as selected.
 - 4) Electric operation and position indication of carriage, hoist, screw jacks, and cask gate.
- b. Radiation monitoring.

9. Miscellaneous

- a. Decontamination with water permissible.
- b. Control room enclosure maintained clean.
- c. Transfers from tunnel to rear cell area not accomplished concurrent with transfers to cell.
- d. Tunnel area can be used as radioactive equipment maintenance area.

H. FUEL CUTOFF MACHINE

1. Space Limitations and Capacities
 - a. Hexagonal subassemblies 2.2 in. across flats, 0.040-in.-thick type 304 stainless steel.
 - b. Cutoff point moveable over entire length of fuel subassembly with $\pm 1/16$ -in. tolerance.
 - c. Cutoff sections to fit scrap containers 12 in. ID by 72 in. long.
 - d. Cooling capacity 1 kW.
 - e. Cutting of instrumented subassemblies.
 - f. Integral manipulator with handling capacity over approximately 4-ft radius.
2. Seals

Pneumatic cylinder piston rod seals of silver-impregnated graphite labyrinth.
3. Materials
 - a. Metal tubing for all fixed lines (steel or copper) convoluted steel or bronze metal hose for all flexible lines.
 - b. Materials to be resistant to sodium and sodium compounds, radiation dose of 1.4×10^9 R minimum.
 - c. Avoid aluminum and organic materials including lubricants and paints.
 - d. Appropriate anodic plated finishes or phosphate treatment of materials.
4. Motor, Drives, and Power

Fractional horsepower, 3-phase, 220-V, A.C. induction motors with integral mounted gear boxes where necessary. (MI cable for all fixed wiring. Projection cable for all flexible wiring.)
5. Operational Requirements

Suitably placed hoods provided to trap and remove all particulate matter produced during cutting. A collecting and disposal system for adhering sodium in excess to that which will be drained before transferring to cutoff machine.
6. Interlocks

Cooling system interlocked with fuel-handling machine to prevent shutoff of cooling of one machine before other takes over.

7. Environment
 - a. Dry argon or air.
 - b. Maximum ambient temperature 165°F.
 - c. Total radiation dose 1.4×10^9 R, gamma photon energy 1 MeV average.
8. Accessibility
 - a. Inaccessible except during periods when cell is open for occupancy.
 - b. Equipment to be remotely removeable from cell for repairs outside cell.
9. Control and Instrumentation
 - a. All operations to be initiated from control station outside the cell and in full view of the operator through shielding window and/or other auxiliary devices.
 - b. Completion of cutting and handling operation to be carried out within two hours.

I. MATERIALS AND SCRAP HANDLING

1. Space Limitations and Capacities
 - a. Max. length of cut core subassembly sections 65 in.
 - b. Diameter of scrap containers 12 in.
 - c. Max. weight of scrap containers and contents--500 lb.
 - d. Dimensions of storage pits 14 in. dia. by 8 or 15 ft deep.
2. Seals

Scrap can covers sealed by gaskets and crimped on by special tool.
3. Materials

Conventional mild steel for containers.
4. Motors, Drives, and Power

None required. Handling will be by cell crane. A special motor actuated crane hook design is available if needed.
5. Operational Requirements

Administrative control, adequate marking of containers.
6. Environment
 - a. Shield covers of storage pits have gamma shielding equal to 12 in. of steel.
 - b. Equipment or scrap removed from the cell sealed in containers.
 - c. Transfer does not breach containment and adheres to the 30 psi cell pressure criteria.

6. (Contd.)

- d. All operations, including hooking and releasing of material visible to operator by direct viewing or by means such as TV monitor, mirrors, and periscopes.
- e. Sufficient lighting, supplementing the regular cell lighting fixtures, required for all operations.

7. Accessibility

- a. Personnel entry to the cell not permitted during R/A scrap, spent fuel, and source rod transfer operations, or when radiation level is above permissible levels.
- b. Regular in-cell scheduled inspection and cleaning procedures will be carried out.

8. Control and Instrumentation

- a. Suitable operating controls will be provided outside of the cell.
- b. Suitable control plugs for connecting portable control boxes will be provided at various locations inside the cell.
- c. No pendant wiring which may interfere with crane bridge and Fuel Handling Machine.
- d. Each handling motion will be controlled independently at speeds from 0 to 20 ft per minute.
- e. Load limit cut-out switches will be provided to prevent overloads.
- f. A load indicating device will be used with and will be viewed by the operator.

Part VI. Electrical

A. SERVICE

1.	Source	13.8-kV line from existing EBR-II spare switchgear.
2.	Voltage	13.8 kV (nominal).
3.	Frequency	60 cycles (nominal).
4.	Phase	3
5.	Fault Capacity	332,000 kVA (max. symmetrical available at EBR-II 13.8-kV Bus).
6.	Voltage Variation	Regulated to $\pm 10\%$ with $5/8\%$ steps by automatic tap changer on transformers (at EBR-II 13.8-kV Bus).
7.	Frequency Variation	± 0.16 to ± 0.33 cycles short time variations (max. at EBR-II 13.8-kV Bus).
8.	Service Terminal	At terminals of EBR-II 13.8-kV circuit breaker. Unit No. 14.

B. POWER TRANSFORMERS

1.	Ratings	a. 13,800-2,400 volts, delta-wye.
b. 13,800-480 volts, delta-wye.
2.	Type	Askarel-immersed, outdoor substation; self-cooled with provision for future forced air cooling; 13,800-2,400 V transformer, 13,800-480 V transformer, self-cooled/forced air cooled.
3.	Phase	3
4.	Capacity	a. 1-2,000 kVA 2,400-V transformer.
b. 2-1,500 kVA 480-V transformers each unit capable of carrying 100% of total 480-V load at forced air rating.

C. SUBSTATION

Type	15-kV outdoor metal enclosed switch and fuse units. Integral with transformers.

D. 2400-VOLT SWITCHGEAR

1.	Type	Indoor, metal clad, drawout.
2.	Bus Arrangement	Single, radial feed.
3.	Circuit Switching	Electrically operated, air breakers rated 1,200 A with interrupting capacity of 75,000 kVA.
4.	Control Voltage	125 volts d.c.
5.	Secondary Backup Protection	Transformer primary fuses.

E. 480-VOLT SWITCHGEAR

1.	Type	Indoor, metal enclosed drawout.
2.	Bus Arrangement	Double ended load center with 2-main breakers and 1-normally open tie breaker.
3.	Circuit Breakers	Electrically operated air breakers.
4.	Interlocks	Breakers interlocked to prevent paralleling 2-transformers.
5.	Bus Transfer	Manual.
6.	Control Voltage	125 volts d.c.

F. 240-VOLT A.C. POWER

- a. Load normally supplied from 240-volt "emergency" system. M-G runs on standby at no load. On loss of normal source, switch will transfer load to M-G set. Transfer will be open circuit and motors will be started in sequence.
- b. The motor-generator set will consist of a.c. generator and a d.c. motor. A.C. generator supplies critical motor loads on loss of normal source and is driven by the d.c. motor connected to the 125-volt d.c. battery.
- c. Single 125-volt d.c. lead acid battery provides emergency power to M-G set for 30 min. Same battery supplies control power, cell emergency lighting and 120-volt a.c. continuous power system M-G set.

F. (Contd.)

d. Steady state allowable variations:

Voltage $\pm 5\%$ Frequency $\pm 3\%$

e. Transient allowable variations:

Voltage $\pm 20\%$ Frequency $\pm 10\%$ G. 120-VOLT A.C. CONTINUOUS POWER1. Critical Instrument and
Control Loads

- a. Normally supplied from 2 and 3 machine M-G sets for continuous power. During outage of M-G sets supplied from unregulated, full capacity reserve transformer connected to 480 volt emergency system.
- b. Three machine M-G set consists of a.c. generator, synchronous motor and d.c. motor. Generator normally driven by synchronous motor connected to 480 volt emergency system. D.C. motor acts as generator to float charge battery under normal operation.
- c. A.C. generator driven by d.c. motor connected to battery during emergency operation.
- d. Single 125 volt d.c. lead acid battery provided to power M-G set for 30 min in emergency. Battery also supplies power for control, cell emergency lighting and the 240 volt a.c. continuous power system M-G set.

2. Continuous Power Load
Center

- a. Steady state
 - Voltage $\pm 1\%$
 - Frequency $\pm 1\%$ (d.c. drive)
- b. Transient voltage response
 - Step load change
 - a.c. or d.c. drive 20%
 - Variation $\pm 5\%$
 - Transfer to d.c. drive variation $\pm 5\%$
- c. Transient frequency response
 - Step load change 20%
 - a.c. drive, variation--None; phase shift only; frequency locked to line frequency

2.	(Contd.)	d.c. drive, variation 4%
		Transfer to d.c. drive, variation 2%
3.	Wiring	Feeder circuit wiring from continuous power load center is No. 10 AWG minimum.

H. EMERGENCY POWER

1.	Load Centers	Main, cell and vault emergency load centers (240 volts) fed from diesel generator (emergency) or from 480-240 V transformers fed from switchgear (normal).
2.	Starting	Diesel generator on the line within 10 sec following start signal.
3.	Availability	Full-time with fuel storage for 48-hr operation at full load.
4.	Manual Transfer	Required for diesel shutdown and returned to normal after emergency operation.
5.	Automatic Transfer	Required for system startup and transfer to emergency operation.
6.	Load Limitations	Provisions to carry only those loads required by sustained operation.

I. PROTECTIVE RELAYING

1.	Selective Tripping	All circuit breakers.
2.	Under-frequency	Separation from EBR-II power at approximately 57.5 cycles.
3.	Power Transformers	13.8-kV current limiting fuses; neutral resistor with overcurrent relay.
4.	EBR-II 13.8-kV Feeders	Auto-reclosing and phase and ground overcurrent relays.
5.	Feeder Circuit Breakers	Phase and ground overcurrent relays.
6.	2300-V Motors	Current phase unbalance relays and undervoltage and phase reversal.

J. MOTORS

1. General

1.1.	1/3 hp and below	120 volt single-phase fed from lighting system and/or 440 volt, 3-phase when available (except those in cell and vault).
1.2.	1/2 to 125 hp	440 volt, 3-phase, except those in cell and vault which are 220 volts, 3-phase.
1.3.	125 hp and above	2,300 volts, 3-phase.
1.4.	Lockout	Stop pushbuttons provided at certain motors and all starters and breakers require padlocking in open position.
1.5.	Disconnect switches	Hp rated and provided at motors 50 hp and below (external to vault) if out of sight of starter.
1.6.	Insulation	NEMA Class A for general service, Class H in cell and vault.
1.7.	Enclosures	Drip-proof for general service. Totally enclosed with drain plugs in end shields for cell, vault, cavity, and outdoors.
1.8.	Service factor	Service factor of one for determination of all motor hp ratings.
1.9.	Altitude derating	All motors for general service derated for operation at 5,100 ft above sea level.
1.10.	Temperature detectors	a. All 2,300 volt motors equipped with Stator temperature detectors and surge capacitors. b. All 2,300 volt motor Stators equipped with temperature detectors and bearings equipped with temperature detectors.
1.11.	Bearings	a. Sleeve bearings on horizontal motors if available at no extra cost. b. Anti-friction bearings for vertical motors.
1.12.	Valve operators	All motors 440 volt, 3-phase.

2. Argon Service

- 2.1. Enclosures Totally enclosed fan cooled or totally enclosed nonventilated.
- 2.2. Insulation Use Class H insulation. Temperature rise based on continuous operation at full load in air not to exceed rated temperature rise for Class A insulation.
- 2.3 Derating No derating for altitude of 5,100 ft.

K. 480 AND 240 VOLT MOTOR CONTROL AND LOAD CENTERS

- 1. Type Indoor metal enclosed.
- 2. Motor Starters Combination circuit breakers.
- 3. Circuit Breakers Molded case, 25,000 amperes min interrupting capacity.
- 4. Wiring NEMA Type B

L. CONTROL AND INDICATION

- 1. Control Room All major equipment controlled from control room.
- 2. Local Stations Minor equipment not controlled from control room will be provided with local control pushbutton switches.
- 3. Control Switches Required for all equipment operated and controlled from control room.

- 4. Pilot Lights

<u>Color</u>	<u>Indication</u>
Red	In service
Green	Out of service
Amber	On standby
White	Power available

For valve position, red will indicate open, green will indicate closed, both lights will be on during transition.

- 5. Annunciation
 - a. Normal manual control opening of electrically operated circuit breakers or magnetic starters will not annunciate.
 - b. Annunciation of abnormal opening of breakers or starters equipped with annunciator circuits is required.

M. LIGHTING

1. Transformers 480-208/120 fed from 480 volt switch-gear. Certain cell lighting may be 480/277 volt.

2. Lighting Level and Fixtures

Area	Foot Candles (Maintained)	Type
Control room	80-90	Fluorescent-luminous ceiling
Cell	400-120	Mercury vapor 277 volt
Operating floors	60	Fluorescent
Cable routing room	40	Incandescent
Offices	75	Fluorescent
Corridors	15-20	Fluorescent
Washrooms	12-25	Fluorescent
Roads and parking	0.5-1.0	Mercury vapor

3. Emergency Lighting
 - a. Required in cell and control room to permit safe interruption of operation following lighting system failure. Cell uses 480 volt emergency power system partly backed up by 125 volt d.c. control power.
 - b. Areas other than cell utilize battery powered self-contained portable units.
4. Cell Lighting Provided by modified EBR-II Fuel Cycle Facility fixture.

N. PIPE HEATING

1. Type Resistance heaters
2. Material Tubular Incoloy for operation at 1500°F.
3. Power Rating Low wattage 5-7 watts/in. per unit surface area heaters fed from 480-volt bus and operated at 277 volts except for hot trap heaters which are standard high wattage type.
4. Control Automatic for main and auxiliary primary, secondary, and service piping.
5. Operating Temperature 600°F (with liquid metal in system).

6.	Heat Requirements	80°F and 350°F prior to filling with sodium not to exceed 50°F/hr or 48 hr total heating time.
7.	Spare Heaters	For all piping inaccessible for replacement.

O. COMMUNICATIONS

1.	Telephone Service	Through existing switchboard located at EBR-II.
2.	Terminal Points	EBR-II switchboard for all new lines.
3.	Conduit and Terminal Boxes	Required in design.
4.	Wiring Equipment	Furnished and installed by telephone company.
5.	Public Address System	Telephone system interconnected equipment furnished and installed by telephone company.
6.	Sound Power System	Three circuit provided for communication between control room and various building areas.
7.	Special Communication	Separate system provided between control room and cell annex and cell personnel locks.

P. MINERAL INSULATED CABLE

1.	Thermocouple Extensions	Highly compressed aluminum oxide.
2.	Containment Penetrations	Sealed by brazing cable penetration to containment liner.

Q. ARGON ENVIRONMENT SERVICE

1.	Equipment	a. Derated for operation at 23.76 in. of mercury and for reduced cooling capability of argon. b. Allowable losses in argon for self-cooled equipment limited to 65% of corresponding losses in air. c. Allowable losses in argon and forced cooling in equipment limited to 71% of corresponding losses in air.

2.	Voltage	All equipment other than heating and lighting shall have maximum voltage of 220 volts, 60 cycles. All wiring connectors and terminals rated at 600-volt service.
3.	Bare Conductors	Sufficient separation to insure strike distance of 1/8 in. and distance of 1/4-in. min spacing.
4.	Control Circuits	Limited to 120 volts.
5.	Shop Tests	Demonstrate satisfactory operation in argon for motors, electromagnetic pumps and flow meters to be used in argon.

R. FUEL TRANSFER TUNNEL

1.	A.C. Outlet	440 volt, 3 ph, 60 cycle conveniently located outside tunnel
2.	Lighting	Provide minimum illumination of 20 foot candles 2 ft 6 in. above floor level and controlled from operating and viewing area of transfer tunnel.
3.	Service Connection	120 volt, single-phase, 60-cycle a.c. connection in transfer tunnel.
4.	32 V d.c. Power Supply	For fuel transfer cask blower.
5.	240 V, 3 ph, 60-cycle a.c. Supply	For fuel transfer cask door, furnished with transfer carrier.
6.	Electrical Wiring	Motors, etc., watertight where necessary for protection during washdown.

S. MISCELLANEOUS

1.	Fire Alarm System	Extended from EBR-II.
2.	Smoke Detector System	Provided in reactor wing, control room and cable routing room.
3.	Annunciators	Provided on appropriate control board to signal abnormal operating conditions.
4.	Flood Protection	None required.
5.	Obstruction Lighting	None required.
6.	Site Monitoring System	None required.

7.	Circuit Routing	Routing of circuits in conduits, cables, raceways, etc., to provide segregation of power, control and instrumentation as required should prevent noise or interference in the instrument and control circuits.
8.	Evacuation Warning System	Extension from EBR-II
9.	Grounding	Required for all electrical equipment. Separate system required for nuclear instrumentation systems.
10.	Main Sodium Shutdown Pump	Power from separate 125 volt d.c. lead acid battery and battery charger normally connected to charger floating on line.
11.	Building Exit Lights	Provided in support wing and reactor building.
12.	Parking Lot Heaters	Outlets required for car and bus heaters in parking lot.

Part VII. Civil-Structural

A. BUILDING GENERAL FEATURES

1. Reactor Building

- 1.1. Permissible leakage Sufficiently leaktight to allow heating and ventilating system to maintain negative pressure between 0.1 to 0.35 in. of water.
- 1.2. Windows None.
- 1.3. Substructure Reinforced concrete, outer surface waterproofed with one bituminous coating.
- 1.4. Superstructure Structural steel frame.
- 1.5. Ventilation exhaust stack Guyed, painted steel--no obstruction lights or obstruction painting.
- 1.6. Future expansion
Accommodations a. 40-ft extension of cell southward.
b. 40-ft extension of reactor building south of column line A.
c. No excavation for future expansion.
d. Building may be X-braced, but design will permit moving the bracing from initial south-end wall to future wall location.
e. Utilities and other outside features to allow extension of reactor building, except road may require relocation.

1.7. Cranes

- 1.7.1. Type Moderate service, traveling bridge, with pendant control station.
- 1.7.2. Capacity One 40 ton main hoist and one 10 ton auxiliary hoist mounted on the same trolley.
- 1.7.3. Drives Separate motors for each drive. Wound rotor induction type designed for crane service.

1.7.4. Speeds

<u>Function</u>	<u>Hook Load</u>		<u>Direction</u>	<u>Feet/Minute</u>
	<u>Tons</u>			
40 ton hook	40		Hoist	15/.5
40 ton hook	40		Lower	20/2.5
40 ton hook	8		Hoist	17/2.5
40 ton hook	8		Lower	18/.5

1.7.4. (Contd.)

<u>Function</u>	<u>Hook Load</u>		<u>Direction</u>	<u>Feet/Minute</u>
	<u>Tons</u>			
10 ton hook	10		Hoist	30/1
10 ton hook	10		Lower	40/5
10 ton hook	2		Hoist	34/5
10 ton hook	2		Lower	36/1
Bridge	40		Horizontal	50/6.5
Bridge	2		Horizontal	54/9
Trolley	40		Horizontal	40/5.2
Trolley	2		Horizontal	43/7.2

1.7.5. Ambient conditions

Indoor service for extended periods of operation with ambient air temperatures up to 120°F.

1.7.6. Safety factors

Determined under static full rated load stresses and based on the ultimate strength of the material used.

<u>Material</u>	<u>Factor of Safety</u>
Cast iron	12
Cast steel	8
Structural steel	5
Forged steel	5
Cables	5
Welds	5
	Based on the ultimate strength of weld metal.

1.7.7. Live loads

Determined in accordance with Specification No. 61 of the Electric Overhead Crane Institute, Inc., for Class "C" moderate service.

1.7.8. Miscellaneous

Operation at 5,100 ft altitude and in Seismic Zone II, as defined in Uniform Building Code.

1.8. Framing

Building framing on top of cell acceptable.

1.9. Pump pit

A pump assembly pit, approximately 10 ft x 10 ft x 30 ft shielded by one foot of concrete is required on the rear cell area.

2. Support Wing

2.1.	Office space	Six office spaces and three secretaries.
2.2.	Safe	No safe required.
2.3.	Lunchroom	One sink with low cabinet and space for vending machines.
2.4.	Lounge area	Provided in women's toilet.
2.5.	All windows	Fixed and calked.
2.6.	Low partitions	Separate secretarial area from corridor.
2.7.	Basement	No lockers, showers, toilets or janitor space.
2.8.	Control room	30 ft x 50 ft minimum.
2.9.	Locker room	18-20 lockers, one water closet, one urinal, two showers, and one lavatory.
2.10.	Roof decking	Long span steel.
2.11.	Roof	Insulation required.
2.12.	Fixed partitions	Concrete block.
2.13.	Control room ceiling	Luminous egg crate type.
2.14.	Doors	Hollow metal with pressed steel frames.
2.15.	Toilet, shower floors, wainscot	Glazed and ceramic mosaic tile.
2.16.	Office floors	Vinyl asbestos tile.
2.17.	Glass and glazing	"B" grade, clear.
2.18.	Toilet and shower stalls	Metal.
2.19.	Painting	Color scheme to be approved by ANL.
2.20.	Traps	Individual traps on each floor drain not required.

3. Building Loads and Design

3.1.	Building design loading conditions	a. D.L. + L.L. + Snow + Crane + Impact b. D.L. + L.L. + Wind + 1/2 Snow Load c. D.L. + L.L. + Wind d. D.L. + L.L. + Seismic e. D.L. + Minimum L.L. + Wind
(For uplift)
3.2.	Crane loads	Not combined with wind or seismic loading.

3.3. Building design loads

3.3.1. Vertical loads

Dead Loads (D.L.) include weight of framing, roof, floor, walls, platforms and all permanent equipment and material.

Live Loads (L.L.)--all load except D.L. For allowable reductions see U.B.C. 1961.

- a. Except for those in the administration area roof, all beams and girders will be subjected to a concentrated load of 2,000 lb applied at mid-span. This load will not be added to column loading.

b. General

	<u>L.L.</u>	<u>plus Added Allowance</u>
Roof Load	30 psf	15 psf (hanging lds)
Offices	80 psf	15 psf
Stairs & Corridors	100 psf	15 psf

c. Reactor Building

	<u>L.L.</u>
Basement	1,000 psf min
First Floor	1,000 psf min
Cell	1,000 psf min
Rear Cell Area	1,000 psf min
First Floor	1,000 psf min
Front Cell Area	1,000 psf min
Second Floor	250 psf
Front Cell	250 psf

- d. Support Wing 200 psf min.

- e. Major Equipment, use actual loads.

3.3.2. Lateral loads

Wind loads shall be in accordance with Idaho Operations Office Engineering Standards, ranging from 30 lb/ft² to 45 lb/ft².

Seismic loading will be in accordance with Uniform Building Code requirements for Zone 3.

B. CONTAINMENT CONSTRUCTION

1. Cell

1.1. Cell walls and roof

Carbon steel shell, filled with ordinary concrete. Walls are 5 ft thick up to elevation 121 ft 7 in. and not less than 3 ft 0 in. thick above this level. Roof is 4 ft thick plus the 1 ft thick missile shield at the 136 ft 8 in. level. Cell floor is ordinary concrete except for 3 ft thick areas over the pipeways which are high density concrete (200 lb/cu ft).

1.2. Cell floor

Liner plate 1 in. thick (mild steel) is to be provided over the impact area. Remainder of cell floor plate liner 1/2 in. thickness or greater, as dictated by other design considerations.

1.3. Reactor cavity bottom

1 ft 0 in. minimum of concrete covered by steel plate 1/2 in. thick.

1.4. Cell inner liner

1 in. plate for north wall, and ceiling, wall and floor between north wall and a line 40 ft south of north wall. Remaining portions of inner liner for cell (from the south wall to 28 ft north of south wall) 1/2 in. minimum plate.

1.5. Energy absorbing false ceiling for cell

1.5.1. Installation area

Directly under cell roof structure.

1.5.2. Cell annex roof

No false ceiling provided.

1.5.3. Design description

12 in. thickness of lightweight concrete faced with a 1/2 in. steel ceiling plate.

1.5.4. Lightweight concrete

100 lb/cu ft. density. Minimum 28 day compress strength of 2,500 psi.

1.5.5. Design considerations

- a. Good surface contact between filler of false ceiling plate and cell ceiling liner plate is required.
- b. False ceiling bending or shear resistance not additive to cell structure strength.

- 1.6. Localized inner cell liner attachments Attachment carefully detailed to avoid damage to liner plate in the event of impact loading. Minimize welding and provide adequate ties in concrete.
- 1.7. Annex floor A reactor head support cylinder and work platform at elevation 146 ft 8 in. is to be provided. Framing to be structural steel and carry 35 tons L.L. (head, bellows, etc.) distributed around support cylinder, 100 psf. L.L. uniformly distributed on the platform and one ton point load anywhere on platform.
- 1.8. Annex subfloor Framing to be structural steel. Live load on the floor is 325 psf plus a downward pull equal to 7 in. of water.
- 1.9. Cell and cell annex interior finish Zinc coating sprayed on and then shot peened. Air lock doors and fuel and equipment port coated with phenolic coating, carboline or equal.
- 2. Vault
- 2.1. Wall thickness and materials 6 ft 3 in. ordinary concrete. High density concrete wherever necessary to maintain equivalent thickness.
- 2.2. Vault liner 5/16 in. stiffened carbon steel plate for walls, floor and ceiling. Wall and ceiling liner used as form.
- 2.3. Vault design conditions 80 percent of allowable stresses for conventional building construction (e.g., ACI & AISC) at an internal static pressure of 30 psig.
- 3. Cavity
- 3.1. Wet fuel storage tank cavity Steel lined concrete construction similar to vault, with baffling system for argon coolant control.
- 3.2. Reactor cavity liner No consideration of explosive action, or missile protection on the cavity liner. Only for incident static pressure conditions.

3.3.	Reactor cavity and pipe galleries	Steel lined with 1/2 in. plate backed by concrete similar to vault.
4.	<u>Cell Windows</u>	
4.1.	Number required	Six with provision for six future windows.
4.2.	Window sleeve liners	Required.
4.3.	Hot side size	45 in. x 38 in. glass.
4.4.	Cold side size	28 in. x 32 in. glass.
4.5.	Field of vision	Downward 66° Upward 44° To each side 53°
4.6.	Design pressure	30 psig at normal stresses.
5.	<u>Air Locks</u>	
5.1.	Number required.	Two for cell.
5.2.	Description	7 ft 0 in. diameter shells 10 ft long with doors opening toward the cell at each end 2 ft 6 in. x 5 ft 0 in. clear opening.
5.3.	Door primary seals	Inflatable type using argon gas.
5.4.	Backup seals	Solid type. Leakage not to exceed 12 standard cubic feet per day at 30 psig with design temperature of 350°F in cell.
5.5.	Doors	Electrically operated with mechanical override. Electric operation arranged so both doors may be operated from each outside face of lock or from inside the lock. Mechanical operation of remote door not required from each station. Cell end doors to be provided with special apparatus to permit pressurizing each to 30 psi.
5.6.	Door interlocks	Mechanically and electrically to prevent breaking containment.
5.7.	Equalization	Toward the cell for the outer bulkhead and either way for the inner bulkhead.
5.8.	Internal pressure	Manually controlled through purging system to permit pressure equalization.

5.9. Removable concrete shielding blocks	Used outside the air locks as required. No shielding blocks or protection required for cell annex access port.
5.10. North personnel air lock	1 in. minimum steel plate for in-cell door and cell side end wall of lock.
5.11. Southeast personnel air lock	1/2 in. minimum steel plate for in-cell door and cell side end wall of lock.

C. CODES, STANDARDS, REGULATIONS AND REFERENCES

1. General	Normally, the most stringent requirements will govern in case of conflict.
a. AEC Manual of Design Criteria, Idaho Operations Office Engineering Standards.
b. IDO Bulletin "Standard Health and Safety Requirements"
c. IDO "Safety and Fire Protection Design Criteria Manual"
d. Uniform Building Code, 1964 Edition.
e. Minimum Design Load in Buildings and Other Structures ASA A58.1.
f. Standard Specification for Highway Construction--State of Idaho, Department of Highways.
g. The design of the building will satisfy the portions of the ICBO Uniform Building Code relating to this type of structure for earthquake Zone No. 3.
h. Structures and equipment will conform to the Engineering Standards, Idaho Operations Office, U.S.A.E.C., revised March, 1963, and to the codes listed therein.
2. Fire Protection	a. NFBU Bulletin No. 22, except as modified in paragraph 3.9.3 below.
b. For yard fire system see description under paragraph 3.9.6 below.
3. Foundations	Soil boring data per Dames and Moore report, dated November 15, 1957, for "proposed ARBOR Project," and supplemental report dated October, 1963.

4. Concrete and Masonry
 - a. ACI Building Code Requirements for Reinforced Concrete (ACI-318-63).
 - b. American Standard Building Code for Masonry.
5. Structural Steel

AISC "Specifications for the Design, Fabrication and Erection of Structural Steel for Buildings," 1963.
6. Welding of Structural Steel

AWS "Code for Arc and Gas Welding in Building Construction."
7. Liquid Metals
 - a. Density, viscosity, liquid heat and thermal conductivity for sodium and NaK to be determined from curves-- Figs. 116-123.
 - b. Supplementary material available in liquid metals handbook (Sodium-NaK Supplement) TID-5277, July, 1955.
8. Hydraulic Data

To be taken from technical paper No. 410 "Flow of Fluids through Valves, Fittings and Pipe" by Engineering and Research Division of Crane Company.
9. Other Materials
 - a. Physical properties to be determined (ANL-5914, "Selected Values of the Physical Properties of Various Materials" by F. J. Tebo (1958).
 - b. Ammonia properties to be selected from Bureau of Standards Circular 142 "Properties of Ammonia."

D. ALLOWABLE STRESSES

1. Foundations

In accordance with current recommendations of Dames and Moore.
2. Reinforced Concrete
 - a. Concrete: Unless otherwise specified in ACI, $f'_c = 3,000$ psi.
 - b. Reinforcing steel: Intermediate grade (use stresses as specified by ACI).
3. Structural Steel

ASTM A-36; stresses per AISC.
4. Lateral Loads Allowance

Members subject to stresses produced by wind or earthquake forces combined with other loads may be proportioned for unit stresses $33\frac{1}{3}$ percent greater than those specified above, provided that the section thus required is not less than that required for the combination of dead and live load plus impact, if any.

E. YARD WORK

1. Roads and Parking Per IDO and State Standards with 32 ft roadway and 24 ft blacktop. Parking for 12 cars. Pipe culverts will be used under roadways where drainage is required.
2. Frost Line 5 ft.
3. Water Supply System Not less than 5 ft of cover. In shallow rocky areas where site conditions permit earth cover may consist of a built-up dike with berms as required.
4. Prime Water Source 8 in. line from EBR-II site connected underground adjacent to existing pump-house in north-south run. Alignment on EBR-II site arranged to accommodate future expansion of facility.
5. Secondary Supply Develop existing test well on site using existing EBR-II construction and test pump modified to suit deeper well. Provide 2,000 gallons of pressurized (air) storage on FARET site. Pump capacity--150 gpm at 745 ft. Well is 675 ft deep with 10 in. casing and was tested at 400 gpm with no apparent drawdown. Pump to have ten additional bowls.
6. Fire Protection Requirements 400 gpm.
7. Sewer System
 - a. Basis 50 gpd/person.
 - b. Population of 38 people (approx.).
 - c. Treatment--Septic tank (2,000 gal) with chlorination.
 - d. Disposal--Leaching field.
8. Drainage
 - a. Site drainage--surface.
 - b. Culverts--as required.
 - c. Roof drainage--to surface system.

9. Fire Protection

- a. Two hydrants attached to the service water system will be provided, details per IDO Standards. Loop is not required.
- b. Assume EBR-II fire truck will also service FARET.
- c. Hose houses not required.

10. Fuel Oil Storage

Boiler fuel oil storage will be two underground tanks, of 12,000 gallon capacity each, Underwriter's Standard.

1. BUILDING CRANE
2. STACK
3. VESSEL COVER HOIST
4. VESSEL COVER AND BELLOWS ASSEMBLY
5. SECONDARY SODIUM PUMP
6. HEATING AND VENTILATING EXHAUST
7. HEATING AND VENTILATING SUPPLY
8. CELL CRANE
9. CONTROL ROD DRIVES
10. FUEL HANDLING MACHINE
11. PRIMARY SODIUM PUMP
12. SODIUM TO AIR HEAT EXCHANGER
13. FUEL AND EQUIPMENT PORT
14. WET FUEL STORAGE TANK

15. REACTOR VESSEL
16. FUEL TRANSFER CASK
17. PERSONNEL ACCESS LOCK
18. PRIMARY HEAT EXCHANGER

Fig. 1. Cutaway Pictorial of Reactor Building and Components

Fig. 2. Plan View of Zoned Core

Fig. 3. Core Subassembly

Fig. 4. Possible FARET In-reactor Loop Locations in 61-assembly Core

Fig. 5. Cross Sections Typical Test Fuel Subassemblies FARET In-reactor Loops

Fig. 6. In-core Loop System

Fuel-element Data

Fuel Material	UO ₂ -15 w/o PuO ₂	UC-15 w/o PuC	Metal ^b
Elements per subassembly	37	37	61
Fuel height, in.	18	18	18
Clad outside diam, in.	0.290	0.290	0.220
Clad thickness, in.	0.020	0.020 ^a	0.016
Spacer wire diam, in.	0.057	0.057	0.052
Axial blanket height (each end), in.	6	6	6
Gas reservoir, in.	25	25	25
Element height, in.	55	55	55
Effective fuel density, %	80	80	70
Bond material	Helium	Helium	Sodium
Fuel U/Pu atom ratio	6/1	6/1	5/1
Fuel details	Coprecipitated; O/M 1.97 - 2.00	Single Phase Solid Solution	-

^a0.015-in. C4 subassemblies.

^bU-15 w/o Pu-6.5 w/o Ti; U-15 w/o Pu-10 w/o Zr (see Table V)

Subassembly Data

Fuel Material	Subassembly Designation	Fuel Type	Clad Material	No. of Subassemblies	No. of Instrumented Subassemblies	Maximum Linear Heat Flux (Design), kW/ft	Maximum Clad Temp, °F (°C)	Maximum Fuel Temp, °F (°C)	Burnups* at Removal, a/o	Maximum Clad Thermal Stress, psi	Coolant Velocity, ft/sec
Oxide	O1	Pellets	304	4	1	15	1250 (675)	3220 (1770)	5, 9, 12, 14	13,300	16
	O2	Pellets	304	4	1	22	1290 (700)	4220 (2330)	5, 8, 10, 12	19,400	23
	O3	Vipak	304	4	1	15	1250 (675)	3220 (1770)	5, 9, 12, 14	13,300	16
	O4	Vipak	304	4	1	22	1290 (700)	4220 (2330)	5, 8, 10, 12	19,400	23
	O5	Pellets	HSA	4	1	22	1340 (725)	4250 (2340)	5, 9, 12, 15	-	20
	O6	Vipak	HSA	4	1	22	1340 (725)	4250 (2340)	5, 9, 12, 15	-	20
Carbide	C1	Pellets	304	4	1	15	1250 (675)	2030 (1110)	5, 9, 12, 14	13,300	16
	C2	Vipak	304	4	1	22	1290 (700)	2520 (1380)	5, 9, 12, 14	19,400	23
	C3	Pellets	HSA	4	1	29	1340 (725)	3050 (1680)	5, 9, 12, 15	-	30
	C4	Vipak	HSA	4	1	36	1340 (725)	3390 (1870)	5, 9, 12, 15	-	39
Metal	M1	U-Pu-Zr	HSA	4	1	14	1250 (675)	1580 (860)	5, 9, 12, 14	-	20
	M2	U-Pu-Ti	V-Ti	4	1	14	1250 (675)	1580 (860)	5, 9, 12, 14	-	20
Advanced	C4S	-	-	3	3	-	-	-	-	-	-
				51	15						

NOTES: 1. Vipak means "vibratory compacted;" HSA means "high-strength alloy," not necessarily the same for oxide as for carbide, etc.
2. Coolant inlet temperature is 810°F. Average outlet temperature from each subassembly is 1050°F. (Exception: outlet temperature from O5 and O6 subassemblies is 1085.) Axial peaking factor = 1.2.
3. Maximum clad temperature include both a 1.33 factor for flow and power maldistribution effects on coolant temperature rise, and a 1.10 uncertainty factor. Maximum fuel temperatures include the 1.33 factor but no uncertainty factor.
*Approximate burnup intervals.

Fig. 7. Reactor Core Data

Fig. 8. Power Density and Burnup

Fig. 9. NRTS and East Area Site Plan

LEGEND

- 1. LIQUID WASTE TANKS
- 2. GAS DECAY TANKS
- 3. VENT COMPRESSOR SUCTION TANK
- 4. EMERGENCY EXIT
- 5. LIQUID WASTE TANKS
- 6. SUMP PUMP
- 7. PUMP MOTORS
- 8. IN-CORE TEST LOOP EQUIPMENT AREA
- 9. IN-CORE TEST LOOP HEAT EXCHANGERS
- 10. MAIN PRIMARY SHUTDOWN PUMP
- 11. REACTOR FLANGE COOLING PUMPS
- 12. VAULT COOLERS
- 13. AUX. INTERMEDIATE HEAT EXCHANGER
- 14. ALLY PRIMARY PUMP
- 15. REACTOR FLANGE COOLER
- 16. REACTOR
- 17. PRIMARY PUMP
- 18. MAIN PRIMARY PUMP HEAT EXCHANGER
- 19. REACTOR CAVITY
- 20. PRIMARY PUMP LOOP SEAL
- 21. WET FUEL STORAGE TANK AND CAVITY
- 22. REACTOR ANNULUS HEATER
- 23. PRIMARY COLD TRAP PURIF. ECONOMIZER
- 24. FUEL TRANSFER HYDRAULIC HOIST PLATFORM
- 25. WET FUEL STORAGE TANK HEATER
- 26. PUMP STORAGE AND DECONTAMINATION PIT
- 27. NaK DRAIN TANK
- 28. SECONDARY DRAIN TANK
- 29. TRANSFER PORT SHIELD PLUG STORAGE
- 30. SECONDARY RECIRCULATING PUMP
- 31. EMERGENCY PUMPS
- 32. DRAIN TANKS AND PUMPS
- 33. IN-CORE TEST LOOP HEAT EXCHANGERS
- 34. PRIMARY DRAIN TANK
- 35. REACTOR CAVITY COOLING INLET
- 36. REACTOR FLANGE COOLING PUMP
- 37. REACTOR CAVITY COOLING OUTLET
- 38. CAVITY COOLING FILTERS AND FANS
- 39. VAULT COOLERS
- 40. REACTOR ANNULUS CIRCULATING PUMP
- 41. REACTOR ANNULUS DRAIN TANK

Fig. 11. Lower Basement Plan

Fig. 12. Upper Basement Plan

LEGEND

1. BUILDING VENTILATION STACK
2. EMERGENCY DIESEL GENERATOR
3. COOLING WATER SURGE TANK
4. ARGON GAS REFRIGERATION EQUIPMENT
5. ARGON GAS STORAGE
6. ARGON GAS EVAPORATOR
7. ACCESS CONTROL CENTER LOCKER ROOM
8. REAR CELL ACCESS
9. FREIGHT DOOR
10. EMERGENCY SHOWER
11. NEW FUEL STORAGE
12. PERSONNEL ACCESS LOCK
13. AUXILIARY SECONDARY PUMP
14. MAIN SECONDARY PUMP
15. MAIN SECONDARY AIR-COOLED HEAT EXCHANGER
16. VIEWING WINDOWS
17. REACTOR VESSEL AND CAVITY
18. MAIN PRIMARY PUMP
19. PUMP CONTROLS
20. WET FUEL STORAGE TANK AND CAVITY
21. PRIMARY COLD TRAP HATCHES
22. AUXILIARY SECONDARY AIR-COOLED HEAT
23. FUEL AND EQUIPMENT TRANSFER PORT
24. PERSONNEL ACCESS LOCK
25. PUMP STORAGE

Fig. 13. Main Floor Plan

LEGEND

1. BUILDING VENTILATION STACK
2. BUILDING EXHAUST AND FILTER EQUIPMENT
3. ELECTRICAL LOAD CENTER
4. PERSONNEL ACCESS LOCK
5. MAIN SECONDARY EXPANSION TANK
6. CELL ANNEX ROOF
7. MAIN SECONDARY PUMP
8. MAIN SECONDARY AIR-COOLED HEAT EXCHANGER
9. HEATING AND VENTILATING EQUIPMENT
10. CELL COOLER
11. CONTROL CABINET
12. CELL ANNEX ACCESS HATCH
13. AUXILIARY SECONDARY AIR-COOLED HEAT EXCHANGER
14. AUXILIARY SECONDARY EXPANSION TANK
15. WET FUEL STORAGE EXPANSION TANK
16. PERSONNEL ACCESS LOCK
17. WET FUEL STORAGE AIR-COOLED HEAT EXCHANGER

Fig. 14. Second Floor Plan

Fig. 16. Containment Liner and Radial Energy Absorber

Fig. 17. Penetration Locations

Fig. 18. Penetration Locations

Fig. 19. Penetration Locations

Fig. 20. Cell Service Penetrations

Fig. 21. Typical Service Penetration in Vault and Cell Annex

(G) EQUIPMENT PENETRATION

(H) SHAFT PENETRATION

(J) CELL PIPE PENETRATION

(K) VAULT PIPE PENETRATION

(L) TOOL PENETRATION

Fig. 22. Mechanical and Pipe Penetrations

Fig. 23. Equipment Penetrations

Fig. 24. Personnel and Window Penetrations

Fig. 25. Fuel and Equipment Transfer Port

Fig. 26. Cell Shielding Window

Fig. 27. Lower Basement Zones

Fig. 28. Upper Basement Zones

Fig. 29. Main Floor Zones

Fig. 30. Second Floor Zones

Fig 32. Reactor Building Heating and Ventilating Schematic

Fig. 33. Reactor Vessel and Cavity Assembly

Fig. 34. Reactor Post Scram Operating Conditions

Fig. 35. Reactor Post Scram Operating Conditions

Fig. 36. Reactor Post Scram Transient Conditions

Fig. 37. Neutron Shield

Fig. 38. Radial Neutron Flux thru Shield

Fig. 39. Axial Neutron Flux below Core

Fig. 40. Radial Gamma Flux Heating Rates thru Shield

Fig. 41.. Radial Temperature Distribution thru Shield

Fig. 42. Instrumented Core Subassembly

Fig. 43. Control Rod Drive.

Fig. 44. Control Rod Drive Gripper

Fig. 45. Accelerator and Shock Absorber

Fig. 46. Lifting Platform

Fig. 47. Cover and Rod Drive Assembly

Fig. 48. Reactor Vessel Cover Handling System

Fig. 49. Fuel Handling Operations

Fig. 50. Fuel Handling Machine Pictorial

Fig. 51. Fuel Handling Machine Assembly

Fig. 52. Fuel Handling Machine Gripper

Fig. 53. Wet Fuel Storage Tank Coolant System

SECTION B-B

SECTION A-A

Fig. 54. Cell Plan and Storage Tank Cover

Fig. 55. FARET Subassembly Cut-off Machine and Scrap Disposal Facilities

Fig. 56. Fuel and Equipment Transfer System

Fig. 57. Reactor Coolant Systems

Fig. 58: Liquid Metal Coolant Flow Diagram

Fig. 59. Double Containment System--Static Pressures

Fig. 60. Main Intermediate Heat Exchanger Post Scram Transient Conditions (X-1)

Fig. 61. Main Air-cooled Heat Exchanger Post Scram Transient Conditions (X-2)

Fig. 62. Main Air-cooled Heat Exchanger Post Scram Operating Conditions (X-2)

Fig. 63. Main Air-cooled Heat Exchanger Post Scram Transient Conditions (X-2)

Fig. 64. Auxiliary IHX Post Scram Transient Conditions (X-3)

Fig. 65. Auxiliary Air-cooled Heat Exchanger Post Scram Operating Conditions (X-4)

Fig. 66. Reactor Flange-cooling Heat Exchanger Post Scram Operating Conditions (X-43)

Fig. 67. Main Sodium Pump Post Scram Transient Conditions (P-1, 2, & 35)

Fig. 68. Main Sodium Pumps Post Scram Transient Conditions (P-1 & 2)

Fig. 69. Main Sodium Pump Post Scram Transient Conditions (P-35)

PLAN
CONTROL ROOM ARRANGEMENT

- REF. DWGS:
- M-271E NUCLEAR INSTR. & PLANT PROT. SYS. CONTROL BOARD C-02 ARRANGEMENT
 - M-272E REACTOR CONTROL BOARD C-04 ARRANGEMENT
 - M-273E LIQUID METAL PROCESS CONTROL BOARD C-05 ARRANGEMENT
 - M-274E RADIATION MONITORING CONTROL BOARD C-06 ARRANGEMENT
 - M-275E GENERAL SERVICES PROCESS CONTROL BOARD C-07 ARRANGEMENT
 - M-276E TEMPERATURE MONITORING & PREHEAT CONTROL BOARD C-08 ARRANGEMENT
 - M-277E AUXILIARY ELECTRICAL CONTROL BOARD C-09 ARRANGEMENT
 - M-268E CONTROL ROOM CONTROL BOARD CONSTRUCTION DETAILS
 - S-140E SUPPORT WING SUBSTRUCTURE GROUND FLOOR PLAN & DETAILS
 - M-278E RELAY & AUXILIARY EQUIPMENT RACK C-10 ARRGT.

Fig. 70

REVISION APPROVALS		ARGONNE NATIONAL LABORATORY OPERATED BY THE UNIVERSITY OF CHICAGO FOR THE U.S. ATOMIC ENERGY COMMISSION	
DWG. NO.	M-267E	DATE	4-7-54
DESIGNED BY	H.E. JORDAN	CHECKED BY	POPEOFF
DRAWN BY	POPEOFF	APPROVED BY	POPEOFF
APPROVED BY	POPEOFF	DATE	5-6-54
DRAWING APPROVED		FAST REACTOR TEST FACILITY	
ARGONNE NATIONAL LABORATORY		CONTROL ROOM ARRANGEMENT	
U.S.A.E.C. CHICAGO OFFICE		SCALE NONE	
JOB NO. 4563		DWG. NO. M-267E

REF. DWGS. (CONTINUED)

14. PAGE NUCLEAR CHANNELS NO. 1. 2. 4. 3. COMPONENT ARRGT. & CIRCUIT DIA.

M-247E NUCLEAR CHANNELS NO. 4, 5 & 6 - COMPONENT ARRGT. & CIRCUIT DIA.

FOR USE - AIRBORNE COMMUNICATIONS WASHINGTON COMPONENT APPROX 4 CIRCUIT DIA

M-249E NUCLEAR CHANNELS NO. 10411 (CONTINUED) ARMS 7-13-68 (1) 11-11-68

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ESTIMATED APPROXIMATE DWG. NO. M-227E			ARGONNE NATIONAL LABORATORY OFFICE OF THE DIRECTOR OF CHARGES FOR THE U.S. ATOMIC ENERGY COMMISSION		
ISSUED BY W. J. B. 5-24-61 CHECKED BY W. J. B. 5-24-61 APPROVED BY W. J. B. 5-24-61			BECHTEL CORPORATION SAN FRANCISCO		
DRAWING APPROVED ARGONNE NATIONAL LABORATORY DATE 5-24-61			NUCLEAR INSTRUMENTATION SYSTEM		
U.S.A.E.C. CHICAGO OFFICE DATE 5-24-61					
TITLE U-235	DRAWING NO. 7-3	DRAWING REVISION NO. PE		SHEET	
SCALE NONE	JOB NO. 4563	DWG. NO. M-227E

TYPICAL CHANNEL ARRANGEMENT
(CHANNEL 1 SHOWN)
(USE AC PWR I AND DC PWR "A" AS NEEDED)

- NOTES:
1. MOMENTARY CONTACTS ARE UTILIZED TO APPLY TRIP TEST SIGNALS.
 2. ALL OTHER SIGNALS ARE REMOVED WHEN CALIBRATION SIGNAL IS APPLIED. (NOTE THAT THE CHANNEL IS MANUALLY TRIPPED WITH SWITCHES IN CALIBRATE POSITION).
 3. EACH TRIP CIRCUIT INCLUDES TRIP INDICATING LIGHT ON CHASSIS AND IS EQUIPPED WITH MANUAL RESET.
 4. CHANNEL RANGE IS $1 - 10^6$ c/s.
 5. ALL CHANNEL TRIP RELAYS TO BE LOCATED IN RACK C-10 AND TO BE OF TYPE INDICATED ON DWG. M-278E.
 6. SEE DWG. RE-2-43912-E FOR SCRAM CIRCUIT. (M-229E).

Fig. 72

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REVISION APPROVALS		ARGONNE NATIONAL LABORATORY	
DWG. NO. M-246E	DATE 12-1-65	OPERATED BY THE UNIVERSITY OF CHICAGO FOR THE U.S. ATOMIC ENERGY COMMISSION	
DESIGNED BY A.N.L.	12-1-65	BECHTEL CORPORATION SAN FRANCISCO	
DRAWN BY J.H.	12-1-65		
CHECKED BY R.E.A.	12-1-65		
APPROVED BY J.E.P.	12-1-65	FAST REACTOR TEST FACILITY NUCLEAR CHANNELS 1, 2 & 3 COMPONENT ARRANGEMENT AND CIRCUIT DIAGRAM	
DRAWING APPROVED		BY ARGONNE NATIONAL LABORATORY	
U.S.A.C. CHICAGO OFFICE		BY	
TITLE II	SCALE NONE	BLDG. 731	PE 4563
JOB NO. 4563		DWG. NO. M-246E	
REV. 0

- > 1. MINORITY CONTACTS ARE UTILIZED TO APPLY THIS TYPE SIGNALS
- > 2. ALL OTHER SIGNALS ARE REMOVED WHEN CALIBRATION SIGNAL IS APPLIED. (NOTE THAT THIS CHANGE IS IN MANUALLY TRIPPED WITH SWITCHES IN CALIBRATE POSITION).
- > 3. EACH TRIP CIRCUIT INCLUDES TRIP INDICATING LIGHT ON CHASSIS AND IS EQUIPPED WITH MANUAL RESET.
- > 4. CRANK RANGE 10^{-10} -- 10^{-3} .
- > 5. ALL CHARGES THIS RELAYS (INCLUDING STIFFNESS CHARGE #1) TO BE LOCATED IN BACK C-10 AND TO BE OF TYPE INDICATED ON SMC. N-278 R.
- > 6. SEE. REF. 2-13810-R FOR SCRAM CIRCUIT. (N-223)

Fig 71

REVISION APPROVALS DWG. NO. M-247E DATE _____ DESIGNED BY ASL 1-11-66 DRAWN BY HEM 1-11-66 CHECKED BY HEM 1-11-66 APPROVED BY ASL 1-11-66 DESIGNED BY ASL 1-11-66			ARGONNE NATIONAL LABORATORY PREPARED BY THE UNIVERSITY OF CHICAGO FOR THE U.S. ATOMIC ENERGY COMMISSION
BECHTEL CORPORATION SAN FRANCISCO					
DRAWING APPROVED ARGONNE NATIONAL LABORATORY BY ASL U.S.A.E.C. CHICAGO OFFICE OFFICE BY ASL			FAST FACTORY TEST FACILITY NUCLEAR CHANNELS 4.5 x 6 COMPONENT ARRANGEMENT AND CIRCUIT DIAGRAM		
TITLE	STATUS	ENGINEER DESIG.	WORK PREPARING DRAWING NO.	SHEET	
SCALE	JOB NO.	DWG. NO.	REV.		
NONE	4563	M-247E			

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- NOTES:
1. ALL CHANNEL TRIP RELAYS TO BE LOCATED IN RACK C-10 AND TO BE OF TYPE INDICATED ON Dwg. M-278 E
 2. EACH TRIP CIRCUIT INCLUDES TRIP INDICATING LIGHT ON CHASSIS AND IS EQUIPPED WITH MANUAL RESET
 3. SEE Dwg. RS-2-13912-E FOR SCRAM CIRCUIT (M-229E)

Fig. 74

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REVISION APPROVALS		ARGONNE NATIONAL LABORATORY OPERATED BY THE UNIVERSITY OF CHICAGO FOR THE U.S. ATOMIC ENERGY COMMISSION	
DWG. NO.	M-248E	DATE	12-6-65
DESIGNED BY	A.N.L.	CHECKED BY	W.E.L.
DRAWN BY	CHL	APPROVED BY	W.E.L.
DRAWING APPROVED		FAST REACTOR TEST FACILITY NUCLEAR CHANNELS N-748 COMPONENT ARRANGEMENT AND CIRCUIT DIAGRAM	
ARGONNE NATIONAL LABORATORY		SAN FRANCISCO	
U.S.A.E.C. CHICAGO OPER. OFFICE		REV. O	
TYPE	II	SCALE	NONE
JOB NO.	4563	DWG. NO.	M-248E

NOTES:

1. SEE DWG. M-2728 FOR RACK COA.
2. SEE DWG. E-1375 FOR AMPLIFIATORS PPGA

REFERENCE:
A.N.L. DWG. RE-2-43912-E

Fig. 77

REVISION APPROVALS SPECIAL <u>M-229-E</u> DATE <u>8-16-78</u> DESIGNED BY <u>W. J. BROWN</u> <u>10-15-78</u> DRAWN BY <u>W. J. BROWN</u> <u>10-15-78</u> CHECKED BY <u>W. J. BROWN</u> <u>10-15-78</u> APPROVED BY <u>W. J. BROWN</u> <u>10-15-78</u> AUTHORITY <u>W. J. BROWN</u>		ARGONNIE NATIONAL LABORATORY UNIVERSITY OF CHICAGO FOR THE U.S. ATOMIC ENERGY COMMISSION BETTEL BETTEL CORPORATION SAN FRANCISCO	
REVISIONS APPROVED ARGONNIE NATIONAL LABORATORY DATE <u>8-16-78</u>		PASJ REACTOR TEST FACILITY PLANT PROTECTIVE SYSTEM DOCK PROTECTION SHEET 2	
REVISIONS APPROVED ARGONNIE NATIONAL LABORATORY DATE <u>8-16-78</u>		YOUR SUBMITTAL INCLUDE THE PE	
TITLE <u>SCALE</u>	DRAWING <u>SCALE</u>	BLDG. <u>731</u> FOR NO. <u>4563</u>	OVER. NO. <u>M-229-E</u>
SCALE <u>NONE</u>		DATE <u>8-16-78</u>	

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COMPARATORS COULD BE
CEC # 19-501 & #221888-02 MODULES

COMPARATORS F _p XN & F _p XL-479-1, -2, -3					
$\frac{F_{pri}}{F_{sec}}$	$\frac{A_1}{B_1}$	$\frac{A_2}{B_2}$	C ₁	C ₂	C ₃
< 0.9	< 1.0	> 1.0	CLOSED	OPEN	OPEN
0.9	< 1.0	1.0	CLOSED	CLOSED	CLOSED
1.1	1.0	< 1.0	CLOSED	CLOSED	CLOSED
> 1.1	> 1.0	< 1.0	OPEN	CLOSED	OPEN
INDETERMINANT BECAUSE P=F=0 (NO FLOW, POWER ON)	SPECIFY COMPARATORS SO THAT WHEN A=0, B MUST BE SOME MINIMAL VALUE (FOR EXAMPLE: 10 MV) IN ORDER TO SWITCH COMPA- RATOR.		OPEN	OPEN	OPEN
NO POWER	SPECIFY COMPARATORS FOR THIS NO-POWER CONDITION		OPEN	OPEN	OPEN

- NOTE 1. THIS EQUIPMENT TYPIFIES CEC FUNCTION MODULES. HOWEVER, THIS DRAWING DOES NOT CONSTITUTE OR INTEND A PREFERENCE FOR ANY SPECIFIC BRAND OF ELECTRONICS.
- NOTE 2. A, B, C REFER TO SEPARATE D.C. POWER PACK INPUTS.
- NOTE 3. THIS DRAWING SUPERSEDES RE-8-43558-C.
- NOTE 4. VOLTAGE VALUES ARE ILLUSTRATIVE ONLY, AND DO NOT CONSTITUTE A REQUIREMENT.

Fig. 78

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REVISION/APPENDICES		DATE	
DWG. NO. M-250E		DATE	
DESIGNED BY: A.M.L.		11-7-65	
DRAWN BY: J.H.		12-2-65	
CHECKED BY: J.H.		12-2-65	
APPROVED BY: J.H.		12-2-65	
DRAWING APPROVED		DATE	
ARGONNE NATIONAL LABORATORY		DATE	
US.A.E.C. CHICAGO OFFICE		DATE	
ARGONNE NATIONAL LABORATORY FOR THE U.S. ATOMIC ENERGY COMMISSION		DATE	
BECHTEL CORPORATION SAN FRANCISCO		DATE	
FAST REACTOR TEST FACILITY		DATE	
RATIO TRIPS		DATE	
PROPOSED ARRANGEMENT		DATE	
TITLE	STATUS	DESIGNED BY	DATE
SCALE	NONE	B.D.G. 731	PE
JOB NO.	4563	DWG. NO.	M-250E
REV.	0		

X-A OUTLET TEMPERATURE RATE OF CHANGE

X-2 SODIUM OUTLET TEMPERATURE RATE OF CHANGE

REACTOR OUTLET TEMPERATURE

NOTES:

1. SEE DWD. M-121E FOR THE ARRANGEMENT AND LOCATION OF ANY CHASSEL COMPARTMENTS.
2. SEE DWD. M-229E FOR PLANT PROTECTIVE SYSTEM SUBRA CIRCUIT.
3. SEE DWD. E-137E FOR IDENTIFICATION DESIGNATION K01 (OCCAN GROUP) AND (OCCAN) AND PLANT PROTECTIVE SYSTEM GROUP.
4. SEE DWD. M-227E FOR AC POWER SUPPLIES 1, IT & IIT.
5. ALL CHASSEL TRIP RELAYS TO BE LOCATED IN RACK C-10 AND TO BE OF TYPE INDICATED ON DWD. M-275E.

Fig. 79

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[illegible]

NOTES:

1. SEE DWG. M-121E FOR PROCESS CHANNEL ARRANGEMENT.
2. SEE DWG. M-229E FOR PLANT PROTECTIVE SYSTEM SCRAM CIRCUIT.
3. SEE DWG. E-137E FOR ANNUNCIATORS SCHEMES K&I (SCRAM GROUP) AND K&2 (NUCLEAR AND PLANT PROTECTIVE SYSTEM GROUP).
4. SEE DWG. M-227E FOR AC POWER SUPPLIES I, II & III.
5. ALL CHANNEL TRIP RELAYS TO BE LOCATED IN RACK C-10 AND TO BE OF TYPE INDICATED ON DWG. M-278E.

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REVISION APPROVALS DWG. NO. M-25E			DATE		
DESIGNED BY ANL CHECKED BY WLN APPROVED BY WLN APPROVED BY WLN	12-2-57 12-2-57 12-2-57 12-2-57	ARGONNE NATIONAL LABORATORY DIVISION OF PHYSICS & CHEMISTRY FOR THE U.S. ATOMIC ENERGY COMMISSION
BECHTEL CORPORATION SAN FRANCISCO			
DRAWING APPROVED ARGONNE NATIONAL LABORATORY BY		FAST REACTOR TEST FACILITY PLANT PROTECTIVE SYSTEM CIRCUIT DIAGRAM SHEET 3			
U.S.A.E.C. CHICAGO OFFICE BY		DATE			
TITLE SCALE	STATUS REVISIONS DATE	BLDG. 731		PE. DWG. NO.	SHEET
NONE		4563		M-25E	
					REV. O

Fig. 83

Fig. 84

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REVISION APPROVALS		ARGONNE NATIONAL LABORATORY FOR THE U.S. ATOMIC ENERGY COMMISSION	
DWG. NO. M-120E	DATE 2-5-66	BECHTEL CORPORATION SAN FRANCISCO	
DESIGNED BY: [Signature]	CHECKED BY: [Signature]	FAST REACTOR TEST FACILITY	
APPROVED BY: [Signature]	DATE 2-5-66	PIPING & INSTRUMENT DIAGRAM LEGEND & NOTES	
DRAWING APPROVED		FAST REACTOR TEST FACILITY	
APPROVED NATIONAL LABORATORY		FAST REACTOR TEST FACILITY	
APPROVED OFFICE		FAST REACTOR TEST FACILITY	
SCALE NONE	JOB NO. 4563	DWG. NO. M-120E	REV. 1

Fig. 85

REVISION APPROVALS		DATE	
DESIGNED BY: HENDERSON	DATE: 12-1-69	ARGONNE NATIONAL LABORATORY	
DRAWN BY: VON KRAKAL	DATE: 12-18-69	FOR THE U.S. ATOMIC ENERGY COMMISSION	
CHECKED BY: J. L. HARRIS	DATE: 1-3-70	BECHTEL CORPORATION	
APPROVED BY: J. L. HARRIS	DATE: 1-3-70	SAN FRANCISCO	
DRAWING APPROVED		FAST REACTOR TEST FACILITY	
ARGONNE NATIONAL LABORATORY		PIPING & INSTRUMENT DIAGRAM	
LABORATORY CHIEF: J. L. HARRIS		ARGON PURIFICATION	
SCALE: NONE		PLANT INSTRUMENTATION NO. 4563	
FOR NO. 4563		DWG. NO. M-119-E	
REV. 1

Fig. 86

DESIGN NO.	M-121E	DATE	12-1-63
REVISION	1	DATE	12-1-63
DESIGNED BY	J. J. J.	CHECKED BY	J. J. J.
APPROVED BY	J. J. J.	DATE	12-1-63
SCALE	NONE	BLDG. 731	PP
SCALE	NONE	JOB NO.	4563
		DWG. NO.	M-121E
		REV.	0

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ARGONNE NATIONAL LABORATORY
FOR THE U.S. ATOMIC ENERGY COMMISSION
BECHTEL CORPORATION
SAN FRANCISCO
EAST REACTOR TEST FACILITY
PIPING & INSTRUMENT DIAGRAM
REACTOR MAIN AND
AUXILIARY COOLING

[illegible]

Fig. 92,

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REVISION APPROVALS		DATE	
DESIGNED BY: M-127E	DESIGNED BY: M-127E	DATE: 12-1-65	DATE: 12-1-65
CHECKED BY: M-127E	CHECKED BY: M-127E	DATE: 12-1-65	DATE: 12-1-65
APPROVED BY: M-127E	APPROVED BY: M-127E	DATE: 12-1-65	DATE: 12-1-65
DRAWING APPROVED		DATE	
ARGONNE NATIONAL LABORATORY		DATE	
ILLUSTRATION OFFICE		DATE	
SCALE: NONE		JOB NO. 4563	
DWG. NO. M-127E		REV. 0	

ARGONNE NATIONAL LABORATORY
OPERATED BY THE UNIVERSITY OF CHICAGO
FOR THE U.S. ATOMIC ENERGY COMMISSION

BECHTEL BECHTEL CORPORATION
SAN FRANCISCO

FAST REACTOR TEST FACILITY
PIPING & INSTRUMENT DIAGRAM
ARGON CELL, CELL ANNEX
& VAULT ATMOSPHERE

Fig. 93

REVISION APPROVALS		DATE		ARGONNE NATIONAL LABORATORY OPERATED BY THE UNIVERSITY OF CHICAGO FOR THE U.S. ATOMIC ENERGY COMMISSION	
NO.	M-126E	DATE	12-15-65	BECHTEL CORPORATION SAN FRANCISCO FAST REACTOR TEST FACILITY PIPING & INSTRUMENT DIAGRAM COOLING WATER	
DESIGNED BY	C. S. FOWLER	DATE	1-10-65		
DRAWN BY	J. CHOD	DATE	1-15-65		
APPROVED BY	J. CHOD	DATE	1-15-65		
DRAWING APPROVED		DATE		U.S.A.E.C. CHICAGO OFFICE	
BY		DATE		BY	
TITLE		STATE	DESIGN	BLD G.	PE
SCALE		NONE	JOB NO.	4563	DWG. NO.
					M-126E
					REV. 0

DOMESTIC HOT
WATER CIRC. PUMP
P-45

EMERGENCY WATER
STORAGE TANK
T-31

DOMESTIC HOT
WATER HEATER
Z-II

Fig. 94

REVISION APPROVALS		ARGONNE NATIONAL LABORATORY FOR THE U.S. ATOMIC ENERGY COMMISSION	
DWG. NO. M-129E	DATE	BECHTEL CORPORATION SAN FRANCISCO	
DESIGNED BY G. S. J.	DATE 12/24/54	FAST REACTOR TEST FACILITY	
DRAWN BY J. L. B. J.	DATE 12/24/54	PIPING & INSTRUMENT DIAGRAM SERVICE WATER	
CHECKED BY J. L. B. J.	DATE 12/24/54		
APPROVED BY J. L. B. J.	DATE 12/24/54		
DRAWING APPROVED			
ARGONNE NATIONAL LABORATORY			
U.S.A.E.C. CHICAGO OFFICE			
TITLE II	STATUS CONST.	BLDG. 731	PE.
SCALE NONE	JOB NO. 4563	DWG. NO. M-129E	REV. O.

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WELL PUMP P-9 & CONTROLS
(EXISTING)

Fig. 95

REVISION APPROVALS DATE M-130E		ARGONNE NATIONAL LABORATORY DIVISION OF PHYSICS U.S. ATOMIC ENERGY COMMISSION	
DESIGNED BY M-130E CHECKED BY M-130E APPROVED BY M-130E APPROVED BY M-130E	DATE 1-25-64 DATE 1-25-64 DATE 1-25-64 DATE 1-25-64	BETTEL CORPORATION SAN FRANCISCO FAST REACTOR TEST FACILITY	
DRAWING APPROVED ARGONNE NATIONAL LABORATORY DATE M-130E CLASSIC CHARGED OPER. OFFICE			
PIPING & INSTRUMENT DIAGRAM PLANT & INSTRUMENT AIR		SHEET	
TITLE M-130E (LAST)	Dwg. No. 731 FOR REF. 4563	Dwg. Numbering in Negative No. DATE M-130E	REV. Q
SCALE NONE

Fig. 96

REVISION APPROVALS				ARGONNE NATIONAL LABORATORY			
DATE	BY	DATE	BY	FOR THE U.S. ATOMIC ENERGY COMMISSION			
DESIGNED BY	DATE	DESIGNED BY	DATE	BECHTEL CORPORATION SAN FRANCISCO			
CHECKED BY	DATE	CHECKED BY	DATE				
APPROVED BY	DATE	APPROVED BY	DATE	FAST REACTOR TEST FACILITY PIPING & INSTRUMENT DIAGRAM WASTE GAS			
DRAWING APPROVED				DRAWING NO. M-131E SCALE NONE FOR NO. 4563 DWG. NO. M-131E

Fig. 97

DIVISION APPROVALS DATE <u>M-132E</u>		ARGONNE NATIONAL LABORATORY REACTOR DIVISION FOR THE U.S. ATOMIC ENERGY COMMISSION	
APPROVED BY: <u>W. J. G. [Signature]</u> APPROVED DATE: <u>11-25-54</u> APPROVED FOR: <u>REACTOR DIV.</u> APPROVED BY: <u>W. J. G. [Signature]</u> APPROVED DATE: <u>11-25-54</u> APPROVED FOR: <u>REACTOR DIV.</u>	BECHTEL CORPORATION SAN FRANCISCO	FAST REACTOR TEST FACILITY PIPING & INSTRUMENT DIAGRAM LIQUID WASTE	
APPROVED: <u>ARGONNE NATIONAL LABORATORY</u> APPROVED DATE: <u>11-25-54</u> APPROVED FOR: <u>REACTOR DIV.</u>		SCALE: <u>1" = 10'</u> DATE: <u>11-25-54</u>	
STATE: <u>ILLINOIS</u> COUNTY: <u>COOK</u> CITY: <u>CHICAGO</u> DISTRICT: <u>1</u>	DIVISION: <u>BLDG 31</u> ROOM: <u>100</u> SCALE: <u>NONE</u>	PROJECT: <u>REACTOR DIV.</u> DRAWING NO.: <u>M-132E</u>	SHEET: <u>REV. 0</u>

Fig. 98

REVISION APPROVALS		ARGONNE NATIONAL LABORATORY FOR THE U.S. ATOMIC ENERGY COMMISSION	
NO. M-133E	DATE	BECHTEL CORPORATION SAN FRANCISCO	
DESIGNED BY: J. J. KELLEY	12-21-54	FAST REACTOR TEST FACILITY PIPING & INSTRUMENT DIAGRAM REACTOR BUILDING VENTILATION	
CHECKED BY: J. J. KELLEY	1-1-55		
APPROVED BY: J. J. KELLEY	1-1-55	DRAWING APPROVED	
ASSIGNED NATIONAL LABORATORY		DATE	
SCALE: NONE		JOB NO. 4563	
SCALE: NONE		DATE: M-133E	

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Fig. 100

Fig. 101

REVISION APPROVALS		ARGONNE NATIONAL LABORATORY	
DATE	NO.	DATE	NO.
DESIGNED BY: M-136E		DESIGNED BY: M-136E	
CHECKED BY: M-136E		CHECKED BY: M-136E	
APPROVED BY: M-136E		APPROVED BY: M-136E	
DRAWING APPROVED		FAST REACTOR TEST FACILITY	
FOR THE U.S. ATOMIC ENERGY COMMISSION		PIPING & INSTRUMENT DIAGRAM	
HEATING DISTRIBUTION		M-136E	
SCALE: NONE		JOB NO. 4563	

FINAL ISSUE
PROJECT TERMINATED
DEC 1965

Fig. 102

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 DEC 1965

REVISION APPROVALS		ARGONNE NATIONAL LABORATORY	
NO.	DATE	BY	DATE
1	1-1-66	J. J. J.	1-1-66
2	1-1-66	J. J. J.	1-1-66
3	1-1-66	J. J. J.	1-1-66
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95	1-1-66	J. J. J.	1-1-66
96	1-1-66	J. J. J.	1-1-66
97	1-1-66	J. J. J.	1-1-66
98	1-1-66	J. J. J.	1-1-66
99	1-1-66	J. J. J.	1-1-66
100	1-1-66	J. J. J.	1-1-66

Fig. 103

REVISION APPROVALS DWG. NO. <u>M-138E</u> DATE _____ BY <u>W. J. [Signature]</u> DATE <u>5-1-66</u> CHECKED BY <u>W. J. [Signature]</u> DATE <u>5-1-66</u> APPROVED BY <u>W. J. [Signature]</u> DATE <u>5-1-66</u> SPECIAL AGENT <u>W. J. [Signature]</u> DATE <u>5-1-66</u>		ARKONITE NATIONAL LABORATORY 10000 W. 10TH AVE. - DENVER 10, COLORADO FOR THE U.S. ATOMIC ENERGY COMMISSION	
DRAWING APPROVAL ARKONITE NATIONAL LABORATORY <u>W. J. [Signature]</u> DATE <u>5-1-66</u> SPECIAL AGENT <u>W. J. [Signature]</u> DATE <u>5-1-66</u>		BETHEL CORPORATION SAN FRANCISCO	
FAST REACTOR TEST FACILITY PRIMARY SYSTEM HYDRAULIC PROFILE		DRAWING NO. <u>M-138E</u>	
TITLE <u>HYDRAULIC PROFILE</u>	SCALE <u>1" = 10'</u>	DATE <u>7-31-66</u>	REVISIONS DWG. NO. <u>M-138E</u>
VERTICAL SCALE: 1" = 10' HORIZONTAL SCALE: 1" = 10'	FOR <u>4563</u>	REVISIONS DWG. NO. <u>M-138E</u>	REVISIONS DWG. NO. <u>M-138E</u>

NOTE:
1. FOR SODIUM PIPING REFER TO DWG NO. M-121E.

Fig. 105

REVISION APPROVALS		ARGONNE NATIONAL LABORATORY FOR THE U.S. ATOMIC ENERGY COMMISSION	
DWG. NO.	M-140E	DATE	
DESIGNED BY	J. J. KELLEY	DATE	7/25/54
DRAWN BY	W. L. KELLEY	DATE	7/25/54
CHECKED BY	W. L. KELLEY	DATE	7/25/54
APPROVED BY	W. L. KELLEY	DATE	7/25/54
DRAWING APPROVED		FAST REACTOR TEST FACILITY	
ARGONNE NATIONAL LABORATORY		PIPING & INSTRUMENT DIAGRAM	
U.S.A.C. CHICAGO DIST. OFFICE		REACTOR ANNULUS & GUARD PIPING NaK	
SCALE NONE		BLDG. 751	PE.
FOR NO. 4563		M-140E	

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DEC 1965

Fig. 106

SHIELD COOLING
DEMINERALIZER
Z-34

SHIELD COOLING
PUMP
P-47
CHEMICAL
INJECTION TANK
T-51

SHIELD COOLING
HEAT EXCHANGER
X-50

SHIELD COOLING
CIRCUITS

FINAL ISSUE
PROJECT TERMINATED
DEC 1965

REVISION APPROVALS		ARGONNE NATIONAL LABORATORY OPERATED BY THE UNIVERSITY OF CHICAGO FOR THE U.S. ATOMIC ENERGY COMMISSION	
DRAWN BY: M-141-E	DATE: 1-27-66	BECHTEL CORPORATION SAN FRANCISCO	
DESIGNED BY: R. EVELAND	DATE: 5-22-65		
CHECKED BY: R. EVELAND	DATE: 4-2-66		
APPROVED BY: J. E. KELLEY	DATE: 4-2-66		
DRAWING APPROVED		FAST REACTOR TEST FACILITY	
ARGONNE NATIONAL LABORATORY		PIPING INSTRUMENT DIAGRAM	
U.S.A.E.C. CHICAGO OFFICE		REACTOR SHIELD COOLING	
TITLE	STATUS	BLDG	PLANT ENGINEERING DEPARTMENT NO.
SCALE	JOB NO.	DWG. NO.	SHEET
NONE	4563	M-141-E	REV. 0

REFERENCE DRAWINGS

M-11775-BUILDING (HUBER) DIAG. ARCHON CELL, VAULT &
CAVITY ATMOSPHERE
M-1316 " " " " WASTE GAS
M-1336 " " " " REACTOR BUILDING VENTILATION
M-1714-RADIATION MONITORING CONTROL BOARD CO-6 ARRANG.
M-2250 " " " " CO-7 "
M-401E
THRU } INSTRUMENT INDEX
M-411E }

Fig. 107

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[illegible]

Fig. 110. Cell and Annex Cooling System

Fig. 111. Cavities and Vault Cooling System Schematic

Fig. 112. Cavities Cooling System and Vault Cooling Units Pictorial

Fig. 113. Cell Gas Degradation

Fig. 114. Radioactive Vent System

Fig. 115. Biological Shield Water Coolant System

Fig. 116. Sodium--Specific Heat

Fig. 117. Sodium--Thermal Conductivity

Fig. 118. Viscosity of Sodium

Fig. 119. Sodium Density

Fig. 120. Thermal Conductivity of NaK (78)

Fig. 121. NaK (78) Viscosity

Fig. 122. NaK(22%Na-78%K)--Density

Fig. 123. NaK(22%Na-78%K)--Specific Heat

REFERENCES--Volume I

1. *Recommended Earthquake Criteria*--John A. Blume & Assoc., *Earthquake Analysis: Critical Sodium Piping*--John A. Blume & Assoc.
2. *Fast Reactor Test Facility (FARET)--Preliminary Feasibility and Cost Study*, AI-8040.
3. *Fast Reactor Test Facility (FARET) and Initial Experimental Program* (Presented at the ANS National Topical Meeting on Fast Reactor Technology, Detroit, Michigan, April 26-28, 1965.)
4. ANL-71106--*FARET Core I Fuel Irradiation and Testing Program*, P. J. Persiani, T. R. Bump, and W. J. Kann, October 1965.
5. ANL-71115--*Reactor Development Program Progress Report*, October 1965.
6. ANL-71168 (Volume II)--*Fast Reactor Test Facility (FARET), Summary of Preliminary Safety Analysis*, J. D. Geier, ed., April 1966.