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MASTER

FFTF, COMPANION TEST AND DEVELOPMENT FACILITY FOR THE LMFBR PROGRAM

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Hanford Engineering Development Laboratory
Richland, Washington operated by WADCO
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The Fast Flux Test Facility*is the reactor test complex which will be used to aid in the solution of the fuels and materials problems which presently are most critical to the development of a liquid metal cooled fast breeder reactor (LMFBR). This paper describes the FFTF and its design features, including those that are unique to this test facility, and those which are prototypic of expected future LMFBR's. The operation of the FFTF is also discussed.

The FFTF is being designed and constructed at the Hanford Engineering Development Laboratory Richland, Washington, operated by WADCO Corporation, a subsidiary of Westinghouse Electric Corporation for the United States Atomic Energy Commission under Contract No. AT(45-1)-2170.

INTRODUCTION

The steadily growing demand for electrical power, coupled with constantly increasing costs of fossil fuel, and ever more rigorous anti-pollution requirements, all combine to make nuclear power plants increasingly attractive. In fact, the consensus of those in the power industry is that future electrical needs cannot be met except with the aid of nuclear power plants. Work to date has been quite fruitful, and light-water reactor generating plants have been developed to the point where they are technically feasible and economically competitive with fossil-fuel plants. Unfortunately, light-water thermal nuclear plants utilize only about 1% of the uranium, and at that rate there will not be enough nuclear fuel to satisfy future requirements. A breeder reactor, however, permits a 60 to 80% utilization of the uranium thus extending the nuclear fuel supply manyfold. For these reasons, the breeder reactor program can be considered to be the most important power reactor program in the United States today.

The importance of developing a breeder reactor was recognized years ago, but the need was not as critical then and the attendant difficult technical problems permitted prime focus on thermal reactors. However, the time is now rapidly approaching when it will become critical. Studies conducted both in the U.S.A. and abroad indicate that the Liquid Metal-Cooled Fast Breeder Reactor (LMFBR) shows the greatest promise of the different types of breeder reactors for early and economical development.

The key program being pursued by the Hanford Engineering Development Laboratory, is the design, construction and future operation of the Fast Flux Test Facility, which is a liquid-metal-cooled, fast flux power reactor plant which will be used for the testing of materials, the development of associated components, and the gaining of operating experience. WADCO, a subsidiary of

Westinghouse Electric Corporation operates the Hanford Engineering Development Laboratory for the Atomic Energy Commission. Commissioner James T. Ramey of the Atomic Energy Commission recently stated that this laboratory was the "Commission's first laboratory dedicated solely to the engineering development of civilian nuclear energy systems." The first project of the Laboratory is the Fast Flux Test Facility. It will provide a temperature, radiation and coolant environment typical of future fast breeder power reactors. The results of testing in this facility will be vital in the future development of such reactors.

The architect-engineer for the FFTF project is the Bechtel Corporation of San Francisco, California. They are also acting as construction managers on the project. The containment vessel is to be fabricated by Chicago Bridge and Iron.

In addition to the architect-engineering work, major parts of the plant are being designed by the Westinghouse Advanced Reactors Division, which is designing the reactor and main fluid portions of the plant; Atomics International, which is designing the main fuel handling equipment, and Aerojet-General, which has the contract for the design of the equipment in the maintenance cell of the plant.

THE FAST FLUX TEST FACILITY

The FFTF is being designed to satisfy five major objectives:

1. Provide a controlled, instrumented, fast-flux environment for testing of fuel, fuel pins, fuel subassemblies, and reactor construction materials.
2. Provide capabilities to test fuel and materials up to and including failure in dynamic sodium.

3. Provide a test environment which is as close to that expected in future LMFBR's as possible.
4. Provide reliable plant performance.
5. Provide a facility to serve all U.S. fast flux requirements.

In addition, the FFTF program will contribute markedly to the following:

- The development of the competitive industrial capabilities necessary for the ultimate realization of industrial LMFBR's.
- The development of systematic methods of safe, economical plant design, construction, and maintenance.
- The development of standards and specifications for the LMFBR Program.
- The development of sodium systems and components for LMFBR's such as purification systems, valves, intermediate heat exchangers, and pumps.

It will be noted that none of these goals is concerned with the utilization of the heat produced by the reactor. Because of this, the monetary saving, and the fact that the operation of the Fast Test Reactor (FTR) will vary from low to full power and will be intermittent, the decision was made to dump the heat to the atmosphere.

An understanding of the fuel problems which the FFTF will be called upon to solve can be gained from a comparison of the relative performance criteria for LMFBR fuels and thermal reactor fuels, Table 1.

The requirements for reactor and plant materials development are comparable to those for fuel development. Also the new demands placed on reactor materials may well result in the emergence of problems which have not previously been encountered or even anticipated. Two such materials problems which have already been identified as a result of testing programs in experimental breeder reactors are stainless steel swelling and irradiation-induced creep.

TABLE 1

RELATIVE PERFORMANCE CRITERIA FOR LIQUID METAL FAST BREEDER REACTOR FUELS AND THERMAL REACTOR FUELS

Performance Criterion		Required Improvement in Fast Breeder Reactor Fuel as Compared with Thermal Reactor Fuel
Burnup	(MWd/ton)	5-10 X
Fuel Power Density	(MW/ton)	3 X
Core Power Density	(kW/l)	10 X
Total Neutron Flux	(n/cm^2-sec)	30 X
Neutron Exposure or Fluence	(nvt)	50 X
Thermal Shock	(°F/sec)	5-10 X

Since the principal purpose of the FFTF is the solution of problems which presently inhibit the construction and use of LMFBR's as power plant heat sources, the FTR has been designed, wherever possible, to be prototypic of the design expected to prevail in future power reactors. The reactor must also be large enough to reflect actual LMFBR operating conditions, particularly a high fast-neutron flux. Satisfaction of these requirements has led to the design of the 400 MWt breeder reactor shown in Figure 1. A tabulation of the design characteristics of the FFTF is given in Table 2. The facility is shown in Figure 2. The principal sub-systems include the FTR, the irradiation test facilities, the fuel handling system, the heat transport system, and the containment and structures.

TABLE 2

FFTF BASIC FACILITY DESIGN CHARACTERISTICS

Core Arrangement	Vertical, 91 Hexagonal Lattice Positions, 75 Driver Fuel Sub-Assemblies.	
Sub-Assembly Length	12 ft overall, 3 ft fuel, 4 ft Maximum Gas Plenum (Advanced Cores)	
Fuel Composition	20-30 Weight % PuO_2, 80-90 Weight % UO_2	
Fuel Target Burnup	45,000 MWd/t Average, 80,000 MWd/t Peak	
Peak Flux	Initial Flux 0.7 x 10^{16} N/cm^2-sec.	
	Future Flux 1.3 x 10^{16} n/cm^2-sec.	
Closed Test Loops		
Initial Number	4 - General Purpose 2 MWt each	
Ultimate Number	6 - 4 General Purpose (4 MW)	
	2 Special Purpose (4 MW each)	
Outlet Temperature	1400°F (Bypass flow permitted)	
Number of Cells Provided	4 Initial with space for two more later	
Open Test Loops		
Initial Number	4 - One with Proximity Instrumentation	
Future Number	3 - One with Proximity Instrumentation	
Heat Transport System (Three Primary Loops)		
	Initial Capability	Maximum Capability
Reactor Power	400 MW	400 MW
Reactor Outlet Temperature	860°F	1050°F
Core Outlet Temperature	900°F	1100°F
ΔT -- Core	300°F	400°F
ΔP -- System	500 ft of Na	500 ft of Na
Intermediate Heat Exchangers, Log Mean Temperature Difference	85°F	100°F
Dump Heat Exchanger Modules	12 @ 33 MW	12 @ 33 MW
Total Coolant Flow	43,500 gpm	43,500 gpm
Sodium System Cover Gas (argon)		
Containment Vessel		
Vessel Material	ASTM-A-516 Low Carbon	
Construction	Welded Construction, ASME Code	
Size and Shape	135 ft diam. x 179 ft high, Elliptical Heads	
Reactor Vessel		
Vessel Material	Type 304 Stainless Steel	
Construction	Welded Construction, ASME Code	
Size	20 ft diam. x 46 ft high, 2-in. wall	

FAST TEST REACTOR

The FTR is a fast neutron test reactor. The core design provides locations for experiment placement, establishes the neutron flux level and power density, and, in conjunction with the coolant and heat rejection systems, sets the temperature levels at which the system will operate. A vertical cross-section of the reactor is shown in Figure 2.

The reactor core consists of a hexagonal array of vertical elements, the positions and types of which are shown in Figure 3. The lattice spacing is constant at 4.730 in. Test positions are dispersed throughout in a Y-shaped pattern of three radial corridors extending from the center outward. Test assemblies, i.e., open tests and closed loops, may be interchanged (with the exception of one test position which can only accomodate a proximity instrumented test). The trisected design is compatible with the refueling system which utilizes three in-vessel handling machines, each of which serves a 120 degree segment of the core.

The hexagonal core rests upon a core support plate and is laterally supported by a core barrel. A radial restraint system is included to offset the effects of irradiation-induced swelling, thermal bowing, and creep. This entire assembly is contained within a stainless steel reactor vessel which is designed for a minimum service life of 20 years under the neutron fluences expected.

The neutron flux level in the FTR is controlled by vertical movement of neutron absorbing material. The control components consist of (1) primary safety rods for rapid shutdown of the reactor; and (2) control rods to regulate the power level, shut down the reactor in a controlled manner, and maintain the reactor subcritical. The locations of the safety and control rods are shown in the core map, Figure 3.

Coolant enters the reactor vessel through three 16 inch inlet nozzles, 120° apart, located near the bottom. From the nozzles, the coolant enters a large inlet plenum from which it flows to the reactor core. The flow is then in parallel up through the driver fuel assemblies, open test assemblies, and around the control rods, reflectors, and shielding, Orificing of the inlets of the various assemblies limits the flow in each and establishes proper flow distribution.

About four percent of the total flow is directed through the annulus between the thermal liner and the wall of the vessel to attenuate thermal transients. Another four percent passes between the assemblies and between other close fitting mechanical parts. A portion of this flow is used to maintain a hydraulic force balance across the inlets of the fuel assemblies.

To stay within the capability of existing pump technology, the maximum allowable pressure drop from the coolant inlet nozzles to the outlet nozzles has been set at145 psi. The pressure in the reactor vessel varies from approximately 20 psia at the outlet nozzles to 165 psia at the inlet nozzles.

The active core consists of driver fuel assemblies, test assemblies, and control and safety rods. It is surrounded radially by a reflector region which in turn is surrounded by the radial shield. All of these components are supported by the core support structure and are laterally restrained within the core barrel. In-vessel storage locations for irradiated fuel assemblies and other components are provided outside the core barrel. The instrument structures (instrument trees) are mounted on plugs in the vessel head and are laterally supported by the core barrel.

The main component of the driver fuel assembly is the flow duct region containing a close-packed hexagonal array of 217 fuel pins with a 0.230 inch diameter. The active core section of each fuel pin contains a column of PuO_2-UO_2 pellets. Helical wire spacers provide radial support for the pins and ensure space between them for coolant flow. The external structural member of the fuel assembly is the coolant flow duct. The bottom inlet nozzle mates with the core support plate to separate the high pressure inlet plenum from the low pressure plenum surrounding the fuel assemblies. The top end of the flow duct is designed to mate with the instrument tree and the in-vessel fuel handling machine grapple. The instrument tree provides a secondary support.

Load pads are provided on the outsides of the ducts at elevations above the active core and at the tops of the ducts. These load pads separate adjacent ducts and provide bearing surfaces for restraining loads and interactions between ducts. The core support structure, made of Type 304 stainless steel, provides structural support for all of the removable and semi-removable components within the reactor core. It also provides the pressure barrier between the high pressure inlet plenum and the outlet sodium pool. Vertical support with allowance for radial thermal expansion, is afforded by a conical skirt which is welded to the outer ring forging of the core support structure and to the ring forging on the reactor vessel. The central section of the core support structure consists of a removable basket which contains the removable core components. Outboard of the removable section, the core support structure provides a space for the in-vessel radial sheilding which is supported laterally by the core barrel.

IRRADIATION TEST FACILITIES

Fuel testing under breeder reactor conditions carries with it many uncertainties, especially at operation near extreme reactor conditions of temperature, flux, and fluence. Malfunctions and failures of test samples can be expected under such conditions. It is not feasible to run potentially disruptive tests in the unprotected core of the reactor where they could cause extensive damage which would result in shutting down the reactor. To obviate such an occurrence, closed test loops have been designed into the system. Initially there will be two such closed loops. Provisions are being made for four additional loops to be added in the future. The test positions are spaced throughout the core, see Figure 3.

A closed loop is an isolated circuit which permits precise control over all experimental parameters including coolant flow rate and purity, temperature, and pressure. The closed loop circuit also includes an in-core test section exposed to known, predictable neutronics. Each closed loop is a separate and distinct system. It includes an in-reactor tube; primary piping, coolant, and coolant pump; an intermediate heat exchanger; secondary piping, coolant, and coolant pump; and an air-blast heat dump.

The in-reactor tube consists of a flow tube inside a pressure tube assembly arranged to form a re-entrant flow system. Coolant enters at the top end of the pressure tube and exists from the top end of the flow tube. A test sample is located in the flow tube and is supported on the end of a hanger tube which has axial spacing fins and contains the desired instrumentation leads. Coolant flow is downward outside the flow tube and then upward through or around the test assembly.

Both the flow tube and the pressure tube are double walled with insulating material or spacers in the annuli to maintain concentricity. Both inner and outer walls are designed so that each is capable of withstanding the design temperature and pressure conditions. The assembly contains a refractory metal cup at the bottom of the pressure tube to retain molten test fuel without causing rupture of the outer pressure wall. Provisions have been made to prevent the escape of any material from a cladding failure or fuel meltdown. Backup cooling is provided by the reactor coolant pool outside the pressure tube.

The primary heat transport system includes two coolant pumps in parallel, either of which can handle the flow requirements and which are connected to both normal and auxiliary power systems. There is also an auxiliary sodium supply tank which contains enough sodium to provide sufficient cooling to preserve the structural integrity of the tube in case of a leak in the primary system smaller than the complete rupture of a pipeline. In the case of failure of the secondary heat transport system, there is sufficient mass of sodium in the auxiliary system to absorb the decay heat.

There is also a group of open test positions in the core of the FTR. There will be seven initially, with the number being reduced to three in the future as four are replaced by closed loops. The open test loops are integral parts of the reactor core and are supplied with reactor primary coolant in the same way as the driver fuel channels. These test positions provide the capability for irradiating specially instrumented assemblies. The open test loops will be restricted in power and configuration to conditions similar to those of a driver fuel assembly. Each test design will be subjected to a safety review prior to insertion into the core to make certain that no potentially disruptive tests are conducted in open test loops.

FUEL HANDLING SYSTEM

The fuel handling system is designed to handle the various components which must be put into the reactor or removed from it for experimental or maintenance purposes. It consists of two main sub-systems. The portion which lies outside the reactor vessel is called the closed-loop, ex-vessel machine (CLEM) and the portion which is inside the reactor vessel is called the in-vessel handling machine (IVHM).

The fuel handling system has been designed to satisfy a number of specific requirements including the following:

- Prevent contact of air with sodium and minimize the escape of any substance during the handling procedure.
- Provide adequate radiation shielding.
- Prevent dangerous or inadvertent release of load from an IVHM or CLEM grapple.
- Provide adequate measurable cooling for component handling and removal of decay heat.
- Prevent the occurrence of a critical array of fuel components.
- Minimize the possibility of reactor contamination.
- Permit identification of core components before loading and indicate position and orientation of components placed in the core.
- Incorporate all safety features including emergency power, manual override, and a "dead man" control to automatically return the system to a safe position in the event of a malfunction.

- Provide adequate heating and cooling to prevent sodium freezeup or thermal shock to components during handling.

The system which has been designed to satisfy the specified objectives can best be described by outlining the handling sequence to place a driver fuel assembly in the core.

The assembly is transported on a transfer dolly through the equipment air lock into the containment building (see Figure 2) and into a controlled environment cell. All traces of air and moisture are removed and the assembly is temperature conditioned. With the aid of the CLEM the assembly is then placed in a finned pot for interim storage. The finned pot with its contents is next transported by the CLEM to the in-vessel transfer station. From there the IVHM takes the assembly out of the finned pot and places it in the core.

At the completion of irradiation, the IVHM transfers the driver fuel assembly to an in-vessel storage position for decay during the next operating cycle. The assembly is next lifted out by the IVHM, replaced in a finned pot, and then moved by the CLEM to the interim decay storage. From storage, the driver is placed in a core component transfer cask. The polar bridge crane is used to place the cask and its contents in a transfer dolly which carries the cask through the equipment airlock to the outside.

The closed-loop, ex-vessel machine (CLEM) is a cooled, shielded, grapple-hoist machine that is mounted on a trolley which is supported by a gantry. Control is semiautomatic and is accomplished from a local control panel.

Cooling or heating is achieved by heat transfer to a NaK-cooled cold wall by a combination of radiation and natural connection. The CLEM has the thermal capability to handle a 10 MW open position test assembly in a finned pot, a 7 MW driver fuel assembly, or a 2 to 4 MW closed-loop.

HEAT TRANSPORT SYSTEM

The HTS carries the heat generated in the core of the reactor to the dump heat exchangers which dissipate it to the atmosphere. There are three independent, parallel heat transport circuits. A flow diagram of one circuit is shown in Figure 4. The HTS has been designed so that the facility will be capable of operating with one circuit inoperative either as a result of an accident or for maintenance.

Sodium heated in the core of the reactor flows out through the normally-open, hot-leg isolation valves to the suction sides of the centrifugal, free-surface, primary-loop, circulating pumps. Sodium from the pump discharges flows to the intermediate heat exchangers (IHX) and vertically downward through the shell, side, transferring heat to the tube-side, secondary sodium. From the IHX outlets, the primary sodium flows through check valves and the cold-leg isolation valves to the reactor vessel inlets. The rate of sodium flow is adjustable by means of variable-speed pump drives.

The three secondary loops circulate the non-radioactive sodium coolant from the intermediate heat exchangers to the dump heat exchangers. Each secondary loop has one pump, one expansion tank, and the necessary valving. Heat from the secondary sodium is transferred to ambient air by means of forced-draft, sodium-to-air heat exchangers. Each of the three heat dumps consists of multiple heat exchanger modules.

The foregoing describes the normal operation. There are also many features to ensure adequate Emergency Core Cooling (ECC). The function of Emergency Core cooling is to maintain a minimum reactor coolant flow in the event of a component failure, a breach of the primary reactor coolant system boundary, or a loss of electrical power.

In the event of a complete loss of all power, both off-site and on-site the equipment elevations and the low system hydraulic resistance provide sufficient natural convective cooling to ensure core integrity. In the event of an off-site power failure, pony motors supplied with emergency power drive the primary and secondary pumps to supply approximately 7 1/2% of normal flow. This is sufficient to remove the normal decay heat during emergency cooling. The pony motor operation will also be used to remove heat during refueling.

In the event of a leak in the sodium system or a component failure, other safety provisions become operative. The reactor vessel outlet nozzle centerlines are 14 feet below the normal sodium level in the reactor. From the nozzles, the outlet piping rises approximately 12 feet before it passes over the top of the guard vessel. The sodium volume in the reactor vessel, above the minimum safe level for ECC, is greater than 3500 ft^3. This reservoir, in combination with the other ECC features, prevents the sodium level in the vessel from falling to the top of the outlet nozzles in the event of leakage at any point in the coolant boundary, thus the siphoning action of transferring sodium to the primary pumps remains unimpaired.

A guard vessel is fitted around the reactor vessel, the free annular space between them has a volume, to the top of the guard vessel, of about 2800 ft^3. The reactor vessel outlet elbow and vertical piping are contained within the guard vessel up to the top of the guard vessel.

The reactor vessel inlet piping is installed vertically between the reactor vessel and the guard vessel and is shrouded by a guard-pipe. If a leak occurs in the inlet piping close to the reactor vessel nozzle, the sodium level rises quickly in the small clearance volume inside the guard pipe and builds up a back pressure above the leak to maintain sufficient core flow.

Guard tanks of limited clearance volume are fitted around all three IHX's and primary pumps. The vertical primary piping to these components is contained within the guard tanks. All other primary loop piping is either contained within the guard vessel and guard tanks or is elevated above the top of the guard vessels. Another important feature of the ECC system is that the shutoff head developed by the primary pumps at pony motor speed is approximately 5 feet. If a leak occurs in one of the primary loops, the sodium level in the reactor vessel will fall as the guard tanks fill up or fluid is pumped out of the elevated piping. However, as soon as a leak occurs it will be sensed by the system instrumentation and the pumps will be tripped to pony motor drive. The sodium inventory above the outlet nozzles ensures that when the pumps have coasted down to pony motor speed the sodium in the reactor will not have fallen below a safe level above the outlet nozzles. When the sodium is at the minimum level, the head developed by the pumps at pony motor speed is insufficient to lift the sodium over the top of the reactor guard vessel and the guard tanks or out of the elevated primary piping.

CONTAINMENT AND STRUCTURES

The FFTF containment and structures include the Reactor Containment Building, Reactor Service Building, Engineering Operations and Control Building, Auxiliary Equipment Building, and auxiliary structures including the electrical substation, main heat dumps, closed loop heat dumps, NaK heat dumps, inert gas storage, pumphouse, water storage tank, and cooling towers.

The reactor containment system includes an inner containment barrier consisting of steel lined reinforced concrete cells, and the outer containment vessel.

The reactor cavity houses the reactor vessel, the guard vessel, and associated shielding. Three heat transport system (HTS) cells surround the reactor cavity on three sides. A fuel decay storage area occupies the fourth side. Four closed loop cells and auxiliary equipment are also located within the containment vessel.

The main work area inside the containment vessel is the operating floor above the reactor. Facilities in this area include a gantry crane serving most of the area, heating and ventilating equipment, miscellaneous accessories, utilities, and control stations. This work area is shielded for continuous occupancy.

The containment vessel is a cylindrical, welded, carbon-steel pressure vessel. Penetrations consist of the personnel airlock, emergency airlock and the equipment transfer lock; ducts for the supply and exhaust of ventilation; and penetrations for piping and wiring. Penetrations are designed and tested for leak tightness to meet the overall leak rate requirement of the containment vessel.

The containment vessel is equipped with vacuum relief and pressure relief devices as well as a lightning rod. Buildup of potential to ground is prevented by grounding.

The reactor cavity, approximately 36 feet diameter x 48 feet high, is inside the containment vessel and is formed of reinforced concrete. A pipeway space is located between the reactor cavity and each HTS cell. The interior surfaces of the cylinder walls and bottom, and of the pipeways, are lined with steel plate to make them gas tight and also provide a barrier in the event of a sodium leak. The cavity and pipeways are capable of withstanding a short-term pressure surge of 35 psig.

The heat transport system cells are also formed of reinforced concrete lined with steel plate. They are rectangular, approximately 33 feet wide by 36 feet long. About half of each cell has a depth of 17 feet and the remaining area, nearest the reactor, has a depth of 44 feet for the primary pumps and the intermediate heat exchangers.

Shielding between the cavity and pipeways and the HTS cells permits limited personnel access to the cells. The cells are capable of sustaining a pressure of approximately 10 psig.

The reactor head access cell is the volume above the reactor vessel head and cover ring. Closure for this volume is the operating deck cover plate. The head cavity is designed to accommodate an air atmosphere during reactor operation. Low radiation levels will permit manned access during reactor operation. To provide an inerting capability, the operating deck is designed so that it can be sealed if required. Irrespective of the cavity atmosphere, the operating deck serves as a missile barrier and provides structural support for refueling and maintenance equipment.

The closed loop and other cells are isolated from the cells forming the inner containment barrier. All of the cells containing primary sodium, plus the maintenance assembly and disassembly cell, are lined with steel plate. These cells include: closed loop cells, pipeways, cold trap cells, fuel decay storage cells, and the primary sodium storage and overflow tank cell.

All penetrations through a wall in any part of the containment system are designed to maintain the integrity of the system under all normal and hypothesized conditions.

The status of the FFTF program is as follows. The plant is now in the mid-course of the design stage. Conceptual studies have been completed and designs are being released for fabrication. Long lead hardware such as pressure vessels, pumps and valves have been ordered. The preparation of the site for major construction is now underway. All plans are targeted toward a criticality date of June 1974.

Stringent quality control procedures will be applied during the fabrication and construction phases of the FFTF plant, and it is anticipated that much worthwhile experience will be gained for future liquid metal fast breeder reactor systems.

Some detail on the Fast Flux Test Facility and its present status has been presented. Of greatest importance is the progress that is being made and the ambitious program ahead.

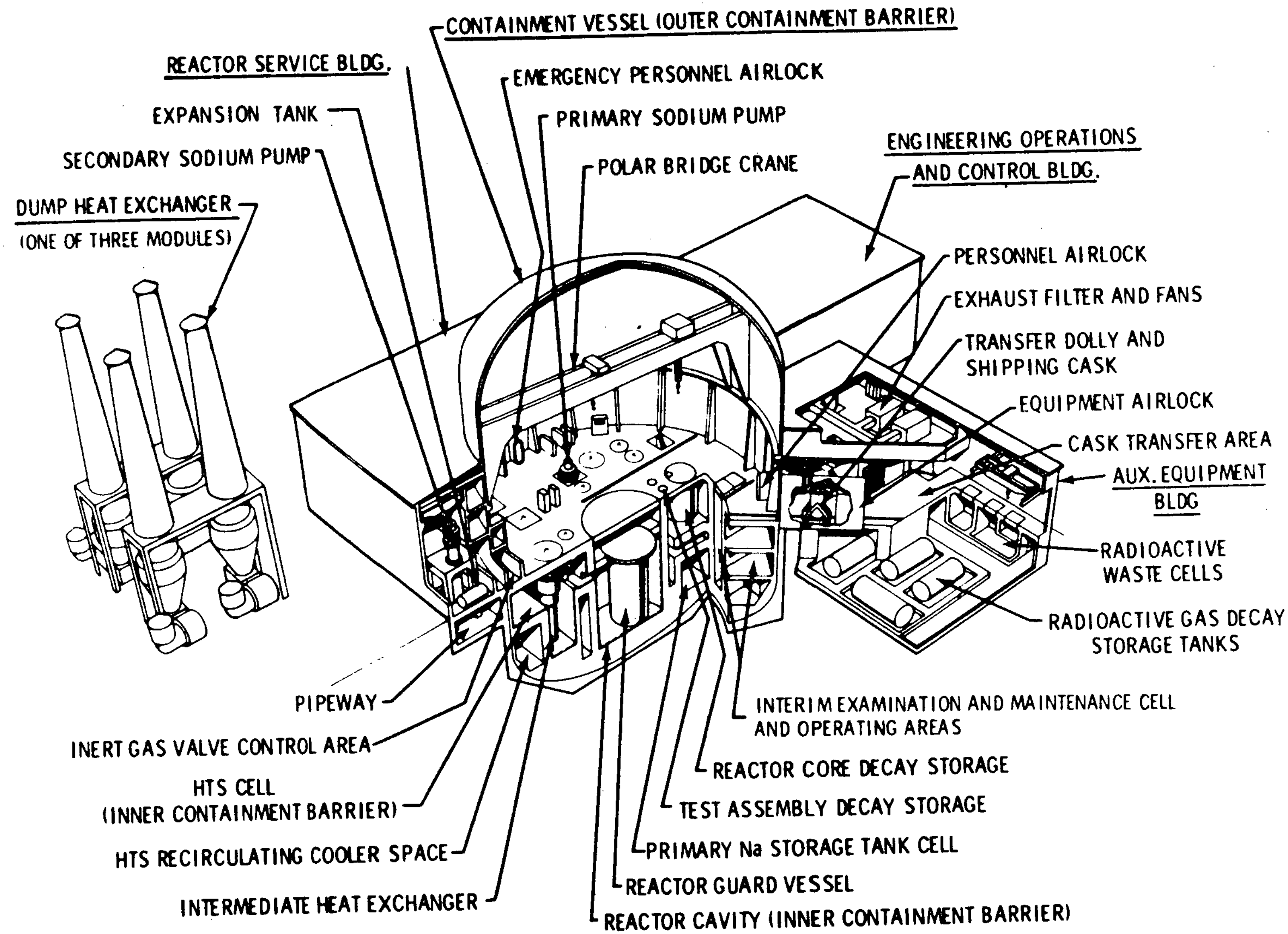


FIGURE 1. FFTF General Arrangement

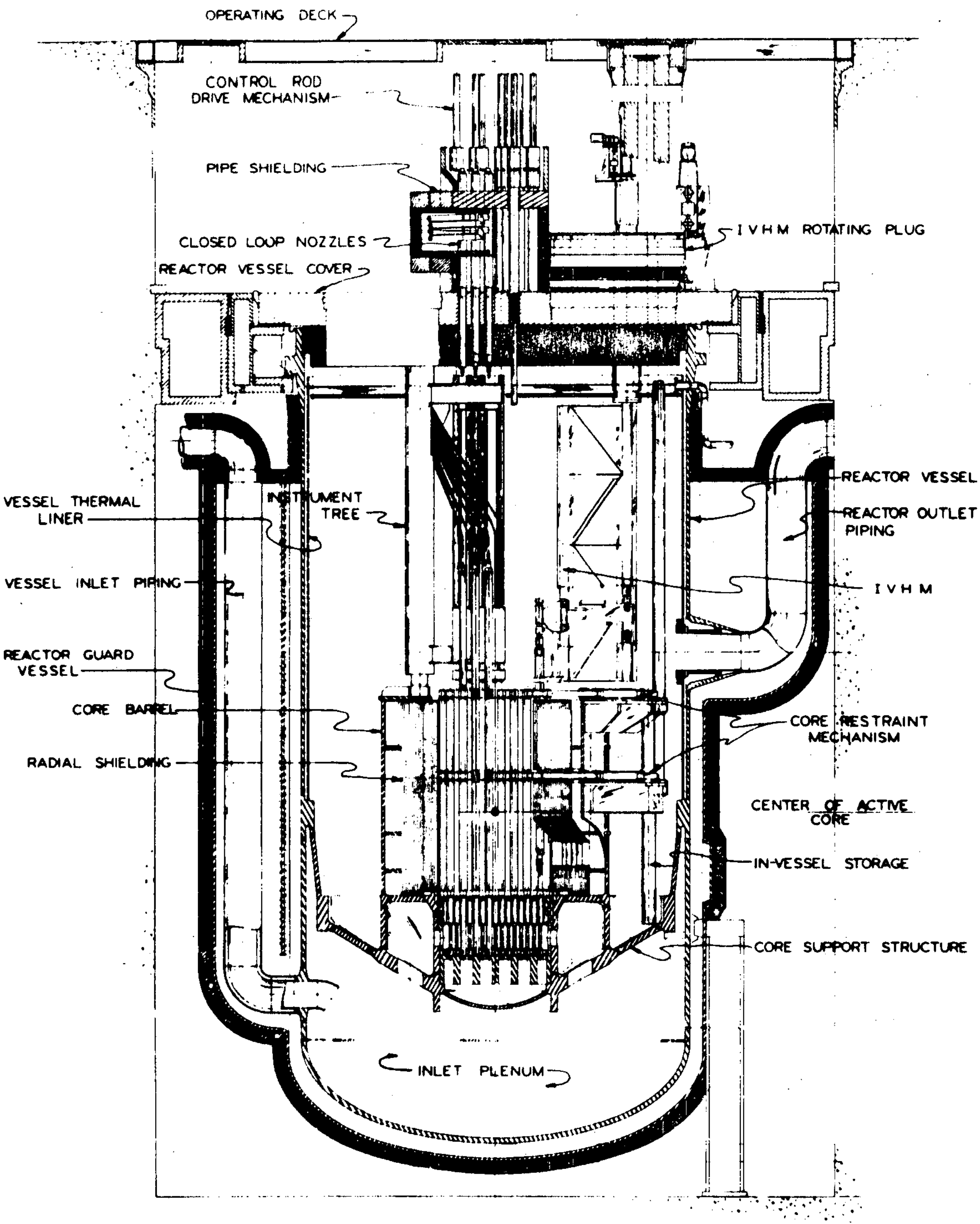
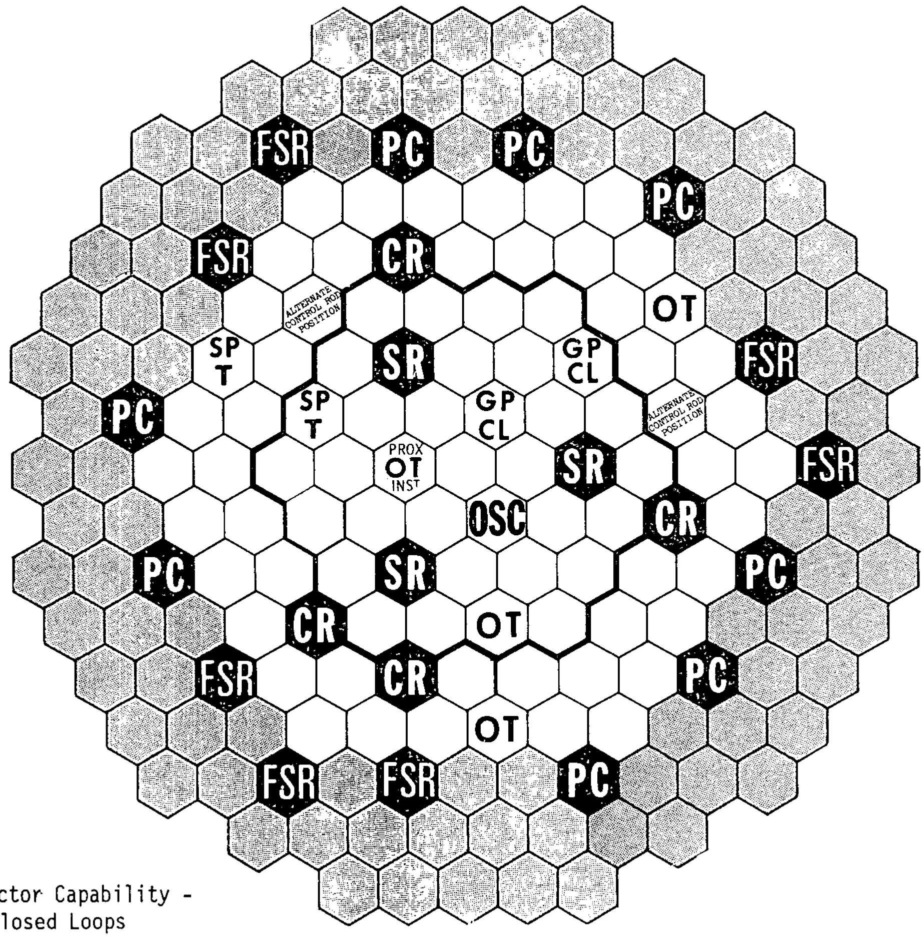


FIGURE 2. Reactor Elevation



Note: Reactor Capability -
6 Closed Loops

GP CL	GENERAL PURPOSE CLOSED LOOPS-2		DRIVERS—28 INNER (ROWS 1—4) —47 OUTER (ROWS 5—6)
SP T	SPECIAL PURPOSE TEST - 2	SR	SAFETY RODS—3
OT	OPEN TEST ASSEMBLIES— 3	CR	IN—CORE SHIM/SCRAM RODS—4
PROX OT INST	OPEN TEST ASSEMBLY WITH PROXIMITY INSTRUMENTATION—1	PC	PERIPHERAL CONTROL RODS—8
OSC	OSCILLATOR - 1	FSR	FIXED SHIM RODS-7
			REFLECTORS—93

AEC-RL RICHLAND, WASH.

FIGURE 3. Core Map (Issued November 1970)

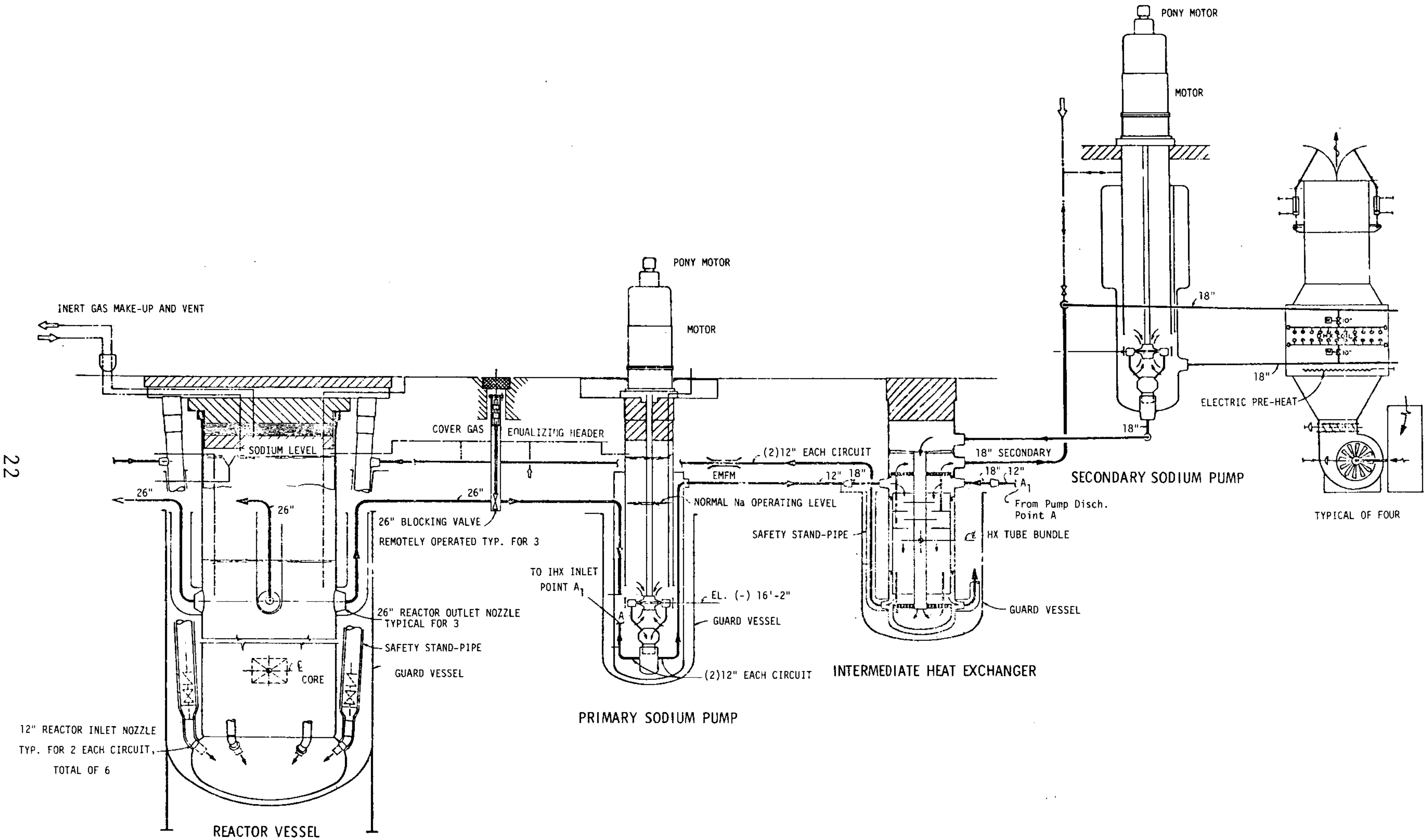


FIGURE 4. Reactor Heat Transport System No. 51 Typical Circuit