ANL-LEO-TR-76-2

ARGONNE FACILITIES

LIQUID METAL FAST BREEDER REACTOR PROGRAM

ARGONNE NATIONAL LABORATORY FACILITY PROFILES ANL-EAST ARGONNE, ILLINOIS ALPHA-GAMMA HOT CE
FACILITY APPARATUS FOR MONITORING AND PURIFYING SODIUM ARGONNE THERMAL SOURCE REACTOR CORE
COMPONENT TEST LOOP FAST NEUTRON GENERATOR 500-GPM WATER LOOP FLOW-INDUCED VIBRATION TEST FA
HIGH-TEMPERATURE LIQUID-METAL MHD FACILITY LMFBR HEAT TRANSFER SIMULATION LOOP MIXING COMPONE
TEST FACILITY OUTLET PLENUM MIXING TEST FACILITY OUT-OF-PILE EXPULSION AND REENTRY APPARATUS
PLASMA TORCH HEATED FURNACE SODIUM FLOWMETER CALIBRATION FACILITY STEAM GENERATOR TEST FACILI
STRUCTURAL DYNAMICS TEST LOOP ZERO POWER REACTOR-6 ZERO POWER REACTOR-9 ARGONNE NATIONAL LAB
FACILITY PROFILES ANL-WEST IDAHO FALLS, IDAHO ARGONNE FAST SOURCE REACTOR EXPERIMENTAL BREE
REACTOR NUMBER TWO EBR-II IN-CORE INSTRUMENT TEST FACILITY EBR-II INSTRUMENTED SUBASSEMBLY TE
FACILITY EBR-II NUCLEAR INSTRUMENT TEST FACILITY EBR-II RADIOACTIVE SODIUM CHEMISTRY LOOP HO
FUEL EXAMINATION FACILITY SODIUM LOOP SAFETY FACILITY TRANSIENT REACTOR TEST FACILITY TREAT
REACTOR TEST APPARATUS (R-SERIES) TREAT MARK-IIC INTEGRAL LOOP ZERO POWER PLUTONIUM REACTOR

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Argonne, Illinois 60439

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Compiled by
Sophie V. Stephens
Breeder Reactor Evaluation Office

September 1976

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PREFACE

The development of the U. S. Liquid Metal Fast Breeder Reactor Program requires the use of a large number and wide variety of experimental and test facilities ranging in complexity from small test rigs to large multi-purpose reactor installations. The Argonne National Laboratory plays a major role in support of this program.

In December 1975, the Division of Reactor Development and Demonstration, U. S. Energy Research and Development Administration published ERDA-68 "LMFBR Program Facility Profiles," which presented data on the national facilities used in support of the LMFBR program. The document was compiled by the Argonne National Laboratory Breeder Reactor Evaluation Office.

This volume is devoted solely to major experimental facilities at the Argonne National Laboratory involved in the conduct of the national LMFBR program. It brings together facility profiles that have already appeared in ERDA-68 as well as new LMFBR facilities that were not included in ERDA-68.

Each profile presents brief descriptions of the overall facility, its test area, and data relating to its experimental and testing capability. It is not intended that this compilation serve as an experimental manual, but rather as an introduction to the versatility of the facilities operated by the Argonne National Laboratory in support of the Energy Research and Development Administration's Fast Reactor Program.

S. V. Stephens
Breeder Reactor
Evaluation Office

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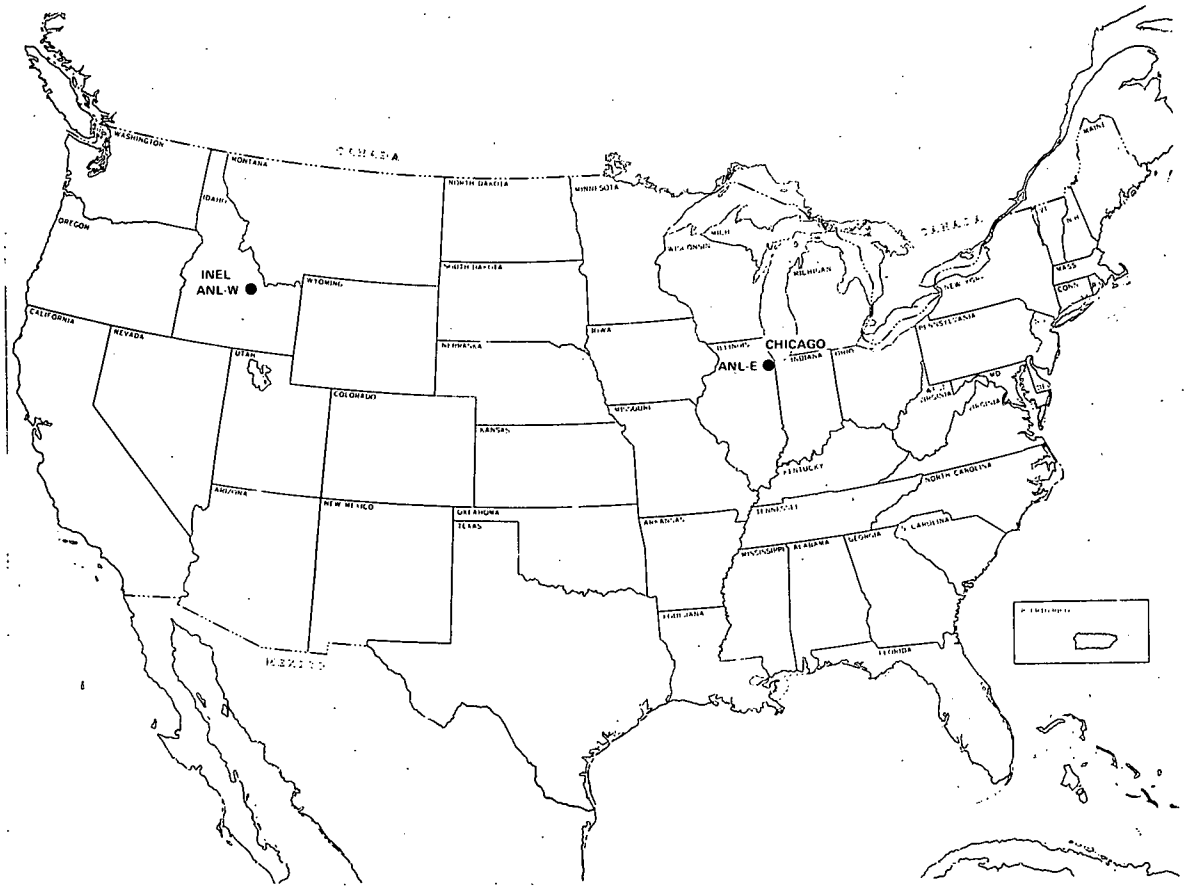
INTRODUCTION

The objective of this document is to present in one volume an overview of the Argonne National Laboratory test facilities involved in the conduct of the national LMFBR research and development program.

Existing facilities and those under construction or authorized as of September 1976 are described. Each profile presents brief descriptions of the overall facility and its test area and data relating to its experimental and testing capability.



The volume is divided into two sections: Argonne-East and Argonne-West. Introductory material for each section includes site and facility maps. The profiles are arranged alphabetically by title according to their respective locations at Argonne-East or Argonne-West.

A glossary of acronyms and letter designations in common usage to describe organizations, reactor and test facilities, components, etc. involved in the LMFBR program is appended.

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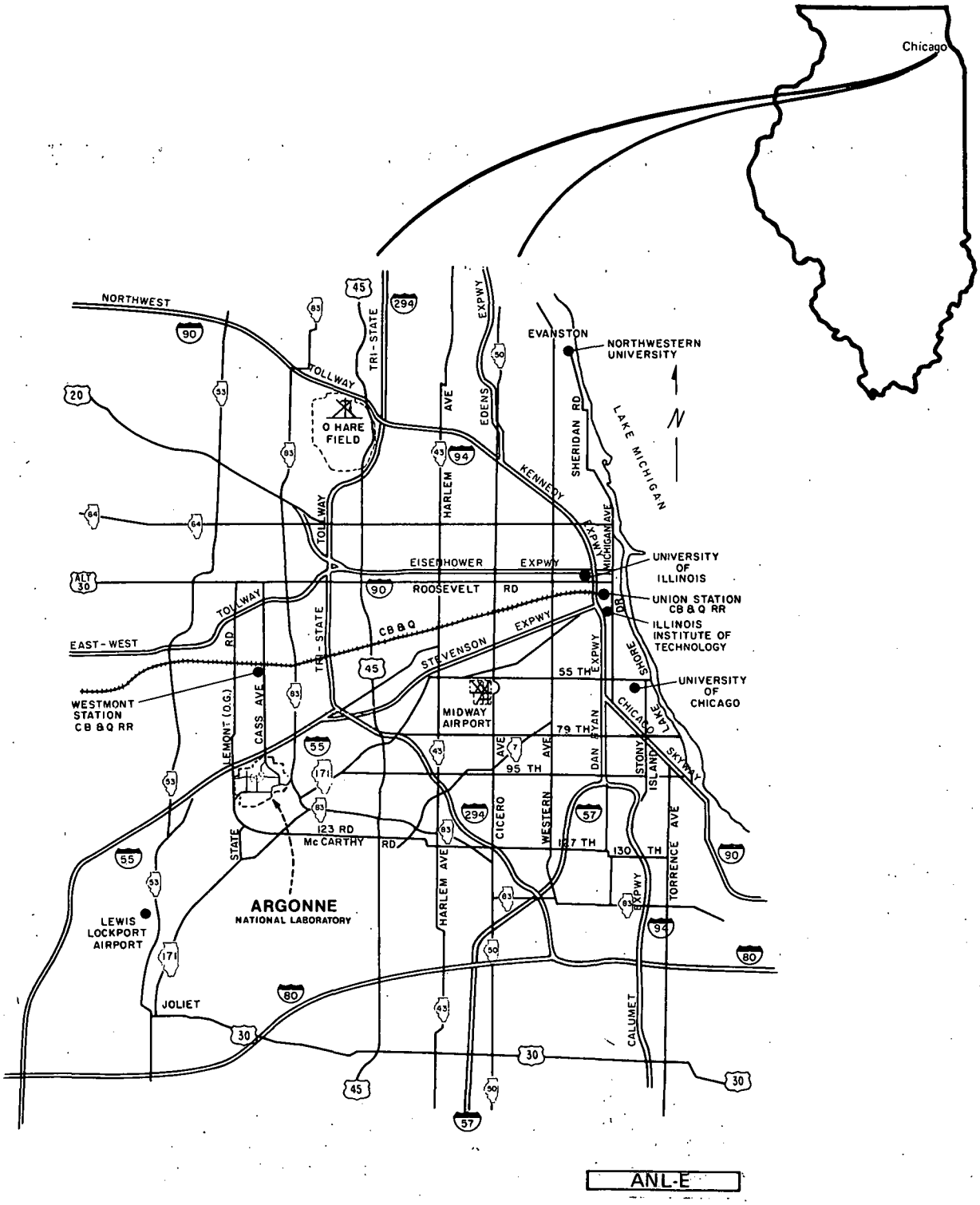
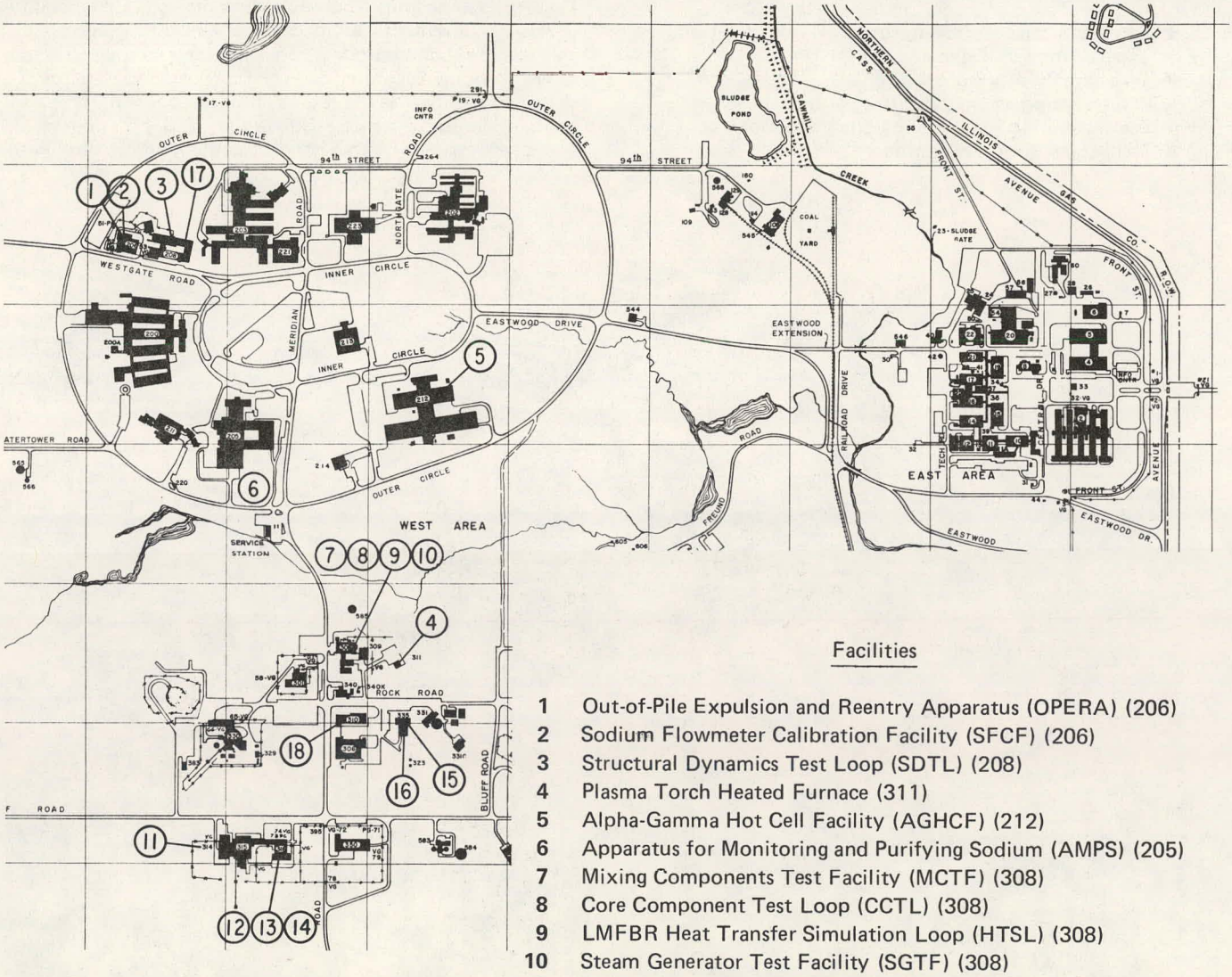
ARGONNE NATIONAL LABORATORY — ILLINOIS

(ANL-EAST)



Argonne National Laboratory - Illinois (ANL-E) Site Location



Facilities

1 Out-of-Pile Expulsion and Reentry Apparatus (OPERA) (206)
2 Sodium Flowmeter Calibration Facility (SFCF) (206)
3 Structural Dynamics Test Loop (SDTL) (208)
4 Plasma Torch Heated Furnace (311)
5 Alpha-Gamma Hot Cell Facility (AGHCF) (212)
6 Apparatus for Monitoring and Purifying Sodium (AMPS) (205)
7 Mixing Components Test Facility (MCTF) (308)
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16 500-GPM Water Loop (335)
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18 High-Temperature Liquid-Metal MHD Facility (HT-LMMHD) (310)

Argonne National Laboratory - Illinois (ANL-E) Facilities Map

ALPHA-GAMMA HOT CELL FACILITY (AGHCF)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses.

- Nondestructive and destructive examination of experimental fuel elements.
- Disassembly of irradiation vehicles.
- Preirradiation assembly of irradiation vehicles.
- Postirradiation fuels and materials experiments.
- Mechanical property testing.
- Postirradiation examination of electronic and mechanical irradiation gear.
- Preparation of fuels and materials specimens for subsequent testing or examination.

Facility Description. The Alpha-Gamma Hot Cell Facility consists of a multicurie hot cell, two shielded glove boxes, and ancillary laboratories. The hot cell is divided into interior rooms separated by shield doors and "air locks" that provide two levels of atmosphere control: one area containing nitrogen with <50 ppm O_2 and ~50 ppm H_2O, and a second area of three rooms containing nitrogen with

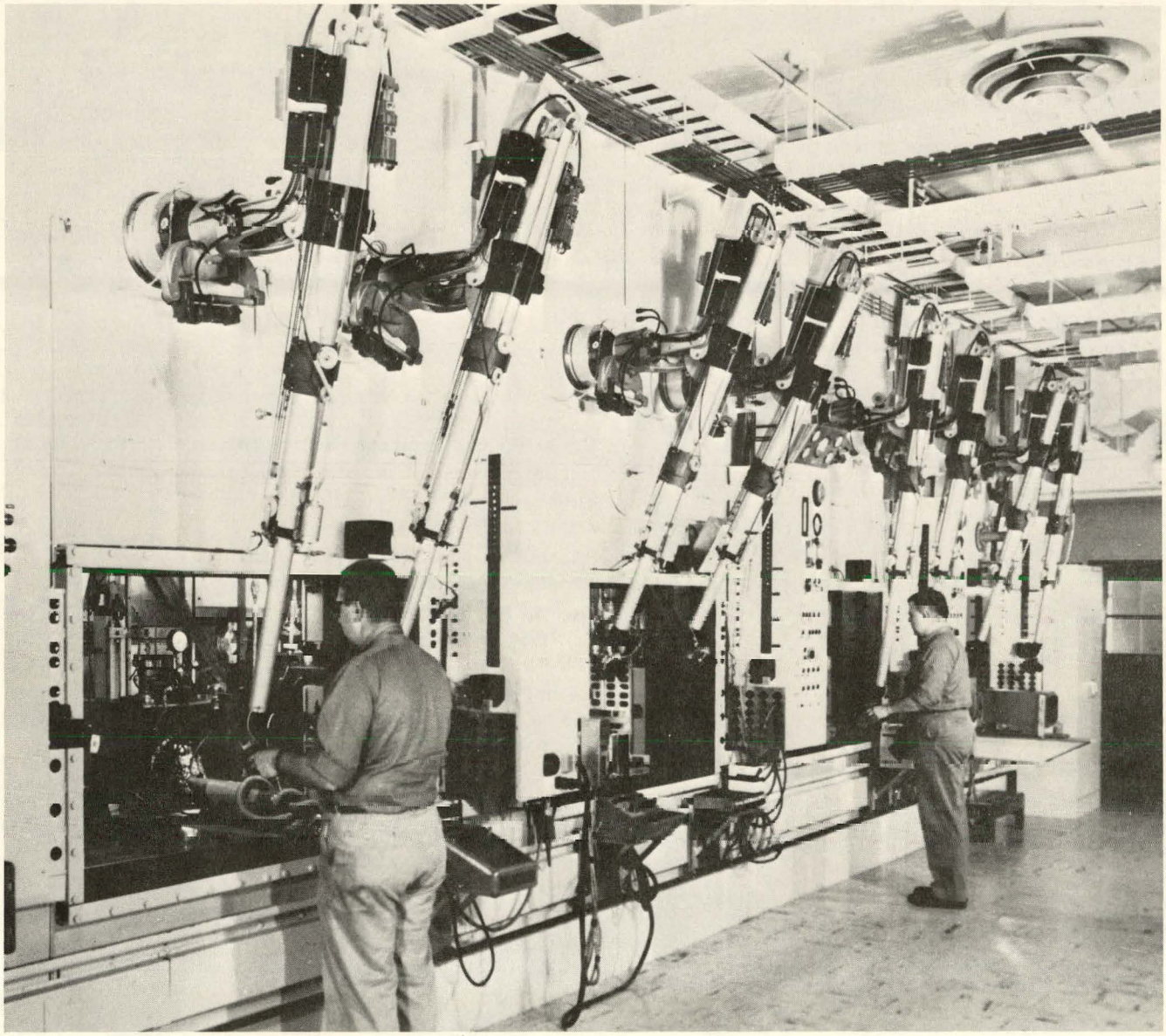

Figure 1. Work Stations of the High-Purity Nitrogen Side of the Alpha-Gamma Hot Cell Facility.

~0.75 wt % O_2. The high-purity area is used for disassembly of irradiation vehicles and destructive examination of fuel elements; sodium and NaK are routinely handled here, the area containing a high-purity flowing sodium loop. The less pure area contains nondestructive examination facilities, maintenance facilities, and the main transfer area into and out of the cell.

The facility has a total of ten work stations, each consisting of a viewing window equipped with master-slave manipulators. Each station has from one to three job functions, most of which are semi-permanently set up at the station. Specimens being examined proceed from one work station to another as prescribed on routing cards that follow the specimens along outside the cell.

A shielded microprobe is connected to the cell by a pneumatic transfer system and contains nitrogen with ~1 wt % O_2. The shielded glove boxes are used to prepare small specimens of fuel for electron microscopy or other tests, and for density measurements on claddings.

Test Capability. The Alpha-Gamma Hot Cell is capable of handling most fuel elements and special irradiation vehicles being used in the LMFBR steady-state and transient testing programs. Unloading arrangements are compatible with most approved horizontal shipping containers weighing up to 10 tons. The cell is capable of handling fuel elements up to 6 ft nominal length, although longer lengths can be accommodated under special circumstances.

The hot cell is equipped for in-depth postirradiation examinations of fuel elements, fuel cladding and structural materials, and small test components.

Auxiliary Equipment.

- Disassembly equipment: vertical and horizontal abrasive cut-off machines, power hacksaw, tubing cutters, longitudinal slitting device.
- Gamma scanner: 4096-channel pulse-height analyzer, Ge-Li detector, magnetic tape readout.
- Macro-cameras (through window and through wall) for magnifications of from 1X to 20X.
- Contact profilometer with strip-chart readout and computerized data analysis, storage, and printout.
- Contact micrometers, dial-gage micrometers, and optical micrometers.
- Eddy-current inspection equipment for cladding defects.
- High-temperature furnace and pneumatic pressurization equipment for mechanical testing of cladding.
- Equipment for metallographic specimen preparation and vacuum impregnation.
- Bausch and Lomb and Leitz metallographs; microhardness indenter.
- Fission gas sampling and collection system.
- Leak-detection systems.
- Micro-sampling equipment.
- Density apparatus.
- Sodium loop facility.
- Quantimet analyzer.
- Shielded electron microprobe.
- Alpha-gamma scanning electron microscope.
- Transmission electron microscope with alpha capability.
- Remote computer station.
- Direct electrical heating apparatus for oxide fuels.

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APPARATUS FOR MONITORING AND PURIFYING SODIUM (AMPS)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses. Development and performance testing of:

- Instruments for on-line measurement of tritium, hydrogen, and oxygen in sodium and cover gas.
- Equipment and procedures for removing impurities from sodium.

Facility Description. AMPS is a 100-gal pumped sodium loop with provisions for adding and removing impurities at controlled rates and with instrumentation for direct on-line measurement of the effectiveness of purification processes. It is located in a 12 X 12 X 18-ft steel cell and is capable of continuous unattended operation at sodium temperatures up to 1200°F. On-line purity instrumentation includes:

In Sodium	In Cover Gas
1 Westinghouse oxygen meter	2 Tritium meters
1 ANL hydrogen meter	2 Hydrogen meters
2 ANL tritium meters	
1 Multipurpose sampler	
1 Specimen equilibration device	

Test Capability. AMPS has been used to develop and test in-sodium tritium meters and a prototype multipurpose sampler. It is presently being used to develop cover-gas hydrogen and tritium meters and methods of removing tritium from sodium. Tritium can be added and removed as needed and is currently at a level simulating EBR-II secondary sodium. Prototype removal devices such as cold traps and gettering traps can be added to the system, performance tested and removed for examination. Techniques are being developed for continuous on-line measurement of the concentration drop produced by such devices, a previously unavailable capability.

Parameters

Circulating sodium inventory	100 gal
Maximum operating temperature[a]	1200°F
Maximum operating pressure	50 psig
Typical piping size	1-in. Sched.-40
Material	Type 304 SS
Sodium flow, main loop	20 gpm
Sodium flow, purification loop	4 gpm
Maximum flow, experimental cold trap	2 gpm
Blower size, experimental cold trap	5 HP, 1200 CFM

[a]Local sodium temperature is 1382°F (750°C) during specimen equilibration in the multipurpose sampler and the specimen equilibration device.

Auxiliary Equipment. Supporting services are available on-site for analysis of sodium and cover gas, metallurgical examination of materials and nondestructive testing. Neutron radiography is especially effective in location of hydrogen-rich deposits. Computer center resources are used in developing analytical models for predicting tritium transport and release in LMFBR plants.

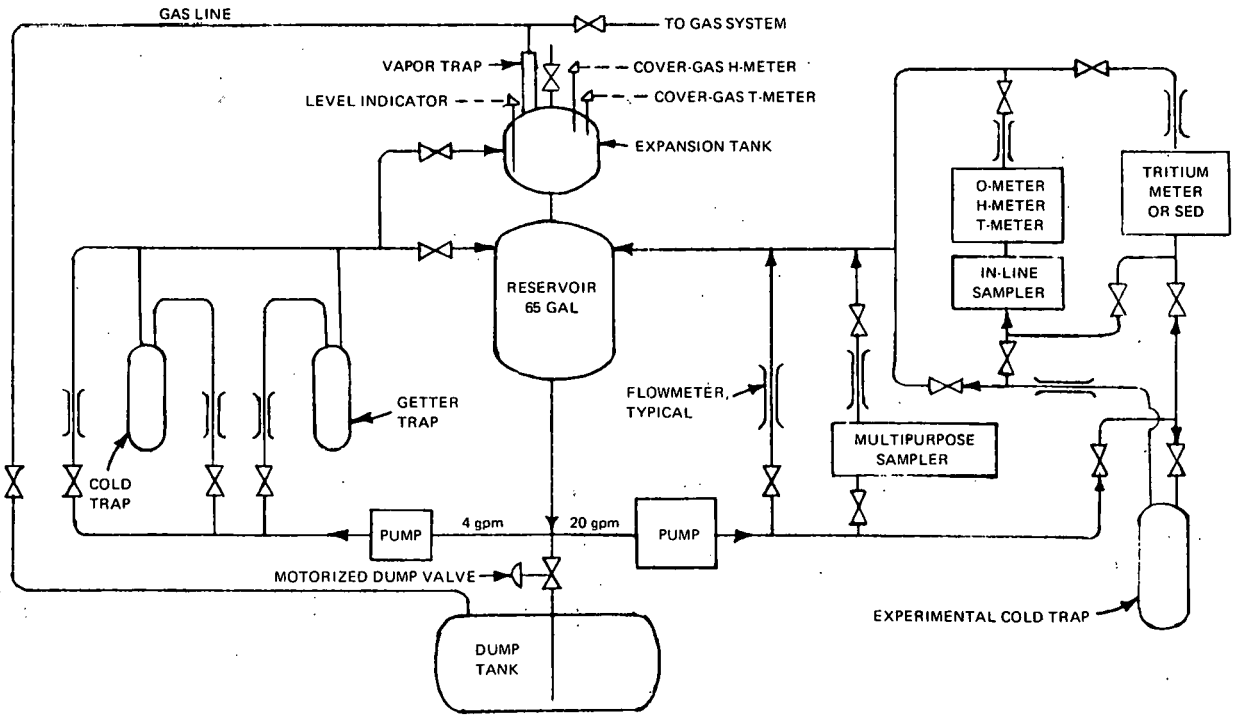

Figure 1. AMPS Schematic Diagram.

ARGONNE THERMAL SOURCE REACTOR (ATSR)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses. This facility provides thermal neutrons for detector calibration purposes and experimental development in support of the Zero Power Reactors.

Facility Description. The ATSR is a highly enriched, light-water-moderated thermal reactor. Movable shielding allows complete access to one face (leakage face) of the core. There is ready access to the central and two peripheral core locations, and it is possible to measure samples with small reactivity effects in these locations. During normal operation, a large graphite thermal column or a cube of depleted uranium one meter on a side (snell block) is placed against the leakage face. The reactor has been approved for operation at 10 kW. Figure 1 is a schematic diagram of the ATSR. Figure 2 is a schematic side view of the facility.

Test Capability. A rabbit tube allows a small sample to be pneumatically driven from the control room into the center of the core, where thermal neutron fluxes of up to 6.5×10^{11} n/cm^2-sec are available. Fluxes of up to 4×10^{11} n/cm^2-sec are available in the graphite thermal column with cadmium ratios for 0.002 in. gold foils from about 6 to 4000. Fast-reactor type neutron fluxes are available in the snell block providing ^{235}U fission rates of up to 10^7 fissions/gram-sec.

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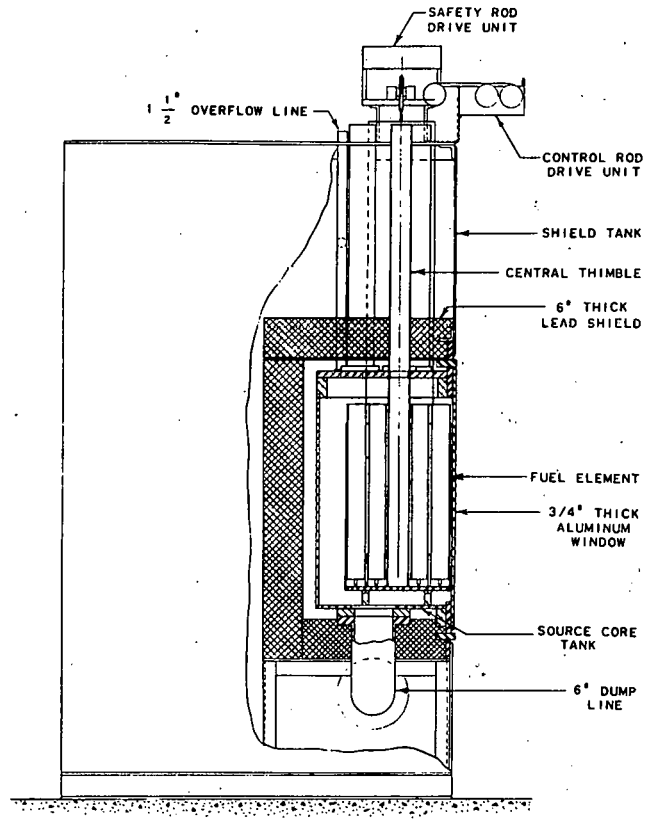
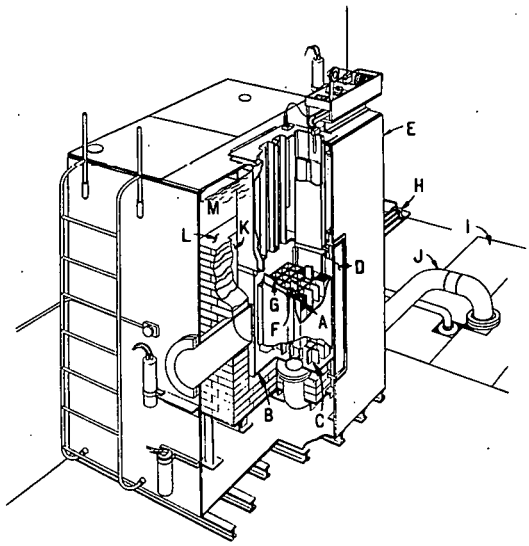


Figure 2. Side View Schematic of ATSR.



A-BORAX FUEL PLATES
B-CORE TANK
C-EGG CRATE ASSEMBLY
D-FRONT LEAKAGE FACE
E-SHIELD TANK
F-CONTROL ROD GUIDE
G-SAFETY ROD GUIDE
H-SOURCE DRIVE
I-DUMP PIT
J-DUMP LINE
K-INNER WATER SHIELD
L-LEAD SHIELD
M-OUTER WATER SHIELD

Figure 1. Schematic Diagram of ATSR.

CORE COMPONENT TEST LOOP (CCTL)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses.

- Hydraulically test LMFBR core components; e.g., fuel assemblies, control and safety rods.
- Demonstrate mechanical integrity of LMFBR core component designs and construction.
- Test in-core nonnuclear instrumentation; e.g., flowmeters, pressure transducers, accelerometer, and void-fraction indicators.
- Test noncore related LMFBR components; e.g., sodium-quality instrumentation, steam generator leak detectors, acoustic flowmeters, cold traps, hot traps, valves, ultrasonic nondestructive testing equipment, and vapor traps.

Facility Description. The major feature of the CCTL is a large (3 ft dia, 25 ft high) vessel into which components are placed for testing. This feature together with the external piping, pump and sodium quality instrumentation constitutes a small nonnuclear model of a loop-type LMFBR primary system less the intermediate heat exchanger. The loop sodium flowrate is controlled by pump speed, and the isothermal temperature (to 1200°F) is attained by electrical immersion and pipe trace heating. The sodium-containing parts of the loop are located below floor level in a concrete and steel-lined pit (20 X 25 X 20 ft deep) which is enclosed during operation, and accessible by elevator at other times. The total sodium inventory (920 gal) can be dumped to a storage tank at the bottom of this pit.

Main loop piping is 4 or 5 in. Schedule 10, Type 304 stainless steel; auxiliary piping varies from 3 to 1/2 in. dia; and all piping is designed for 1200°F, 150 psig. Sodium-quality instrumentation includes electrochemical oxygen meters, diffusion-type hydrogen meters, vanadium wire oxygen equilibration and sodium-sampling equipment. A free surface mechanical-centrifugal pump provides up to 800 gpm at 110 psi head. An auxiliary EM pump provides flow to 2 cold traps. Permanent loop instrumentation includes liquid-level measurement in vessels and pump, loop temperature, flowrate by electromagnetic and flowtube measurement, and pump pressure.

Experimental Parameters

Pump flowrate	0–800 gpm
Sodium temp.	to 1200°F
Sodium inventory	920 gal
Cold trap flowrate	7 gpm
Pump head	110 psi
Facility design temp.	1200°F
Facility design pressure	150 psig
Test component size limit (in-vessel)	3 ft dia., 25 ft high
Minimum cold trap temp.	250°F
Pump flow and pressure control	Continuous

Test Capability. Although many test components (e.g., valves and flowmeters) can be located outside the test vessel, the preferred location is inside the test vessel. This arrangement allows sodium leakage and lower pressure design of the article, since the test vessel is the ultimate outer container. The CCTL is particularly well suited for

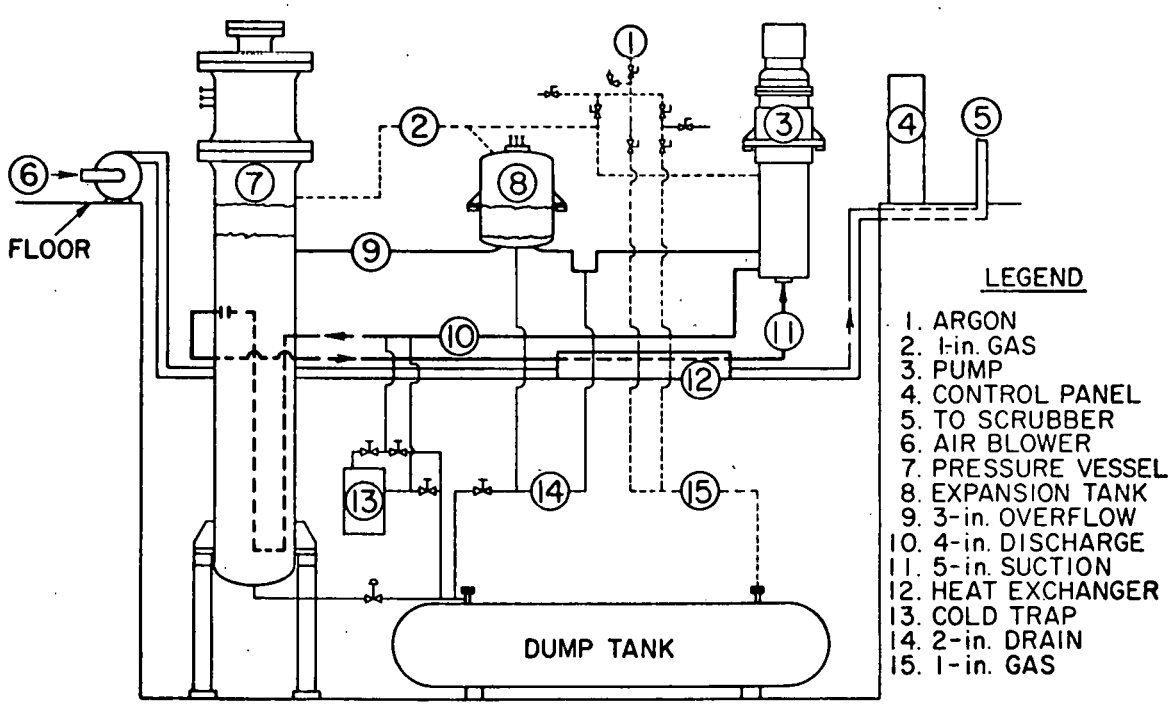


Figure 1. Schematic of CCTL System and Components.

full-scale fuel assembly hydraulic tests, since the reactor grid, hydraulic holddown, and upper discharge plenum can be perfectly simulated. In this case, the main pump draws suction from the test vessel exactly as in a loop-type LMFBR primary system. Access for component removal and instrumentation is achieved at the top of the test vessel through either a full 36-in. flange, or through 14-in., 8-in., and many smaller flanges. These flanges will accommodate other components as evidenced by current testing of steam generator leak detectors in conjunction with a hydraulic model of a steam generator housed inside the test vessel. In this case, the pressure capability of the test vessel allows reduced pressure and leak-tight requirements of the model. Steam injections will be made directly into the model and the reaction products detected at various locations using various prototype detectors such as in-sodium hydrogen diffusion, cover gas hydrogen diffusion, and electrochemical oxygen meters. Acoustic detection of the reaction noise also will be conducted.

Auxiliary Equipment.

- 20-ton and 5-ton bridge cranes.
- Remote Automatic Data Systems (RADS) interconnected with Laboratory computers.
- Dymec Model 2010G Data Acquisition System (200 channels).
- Hewlett-Packard Model 9601B Data Acquisition System with mini-computer, multi-plexer (128 channels) and crossbar scanner (200 channels).
- Small satellite machine shop.
- Sodium oxide removal scrubber.
- Instrument repair and calibration shop.

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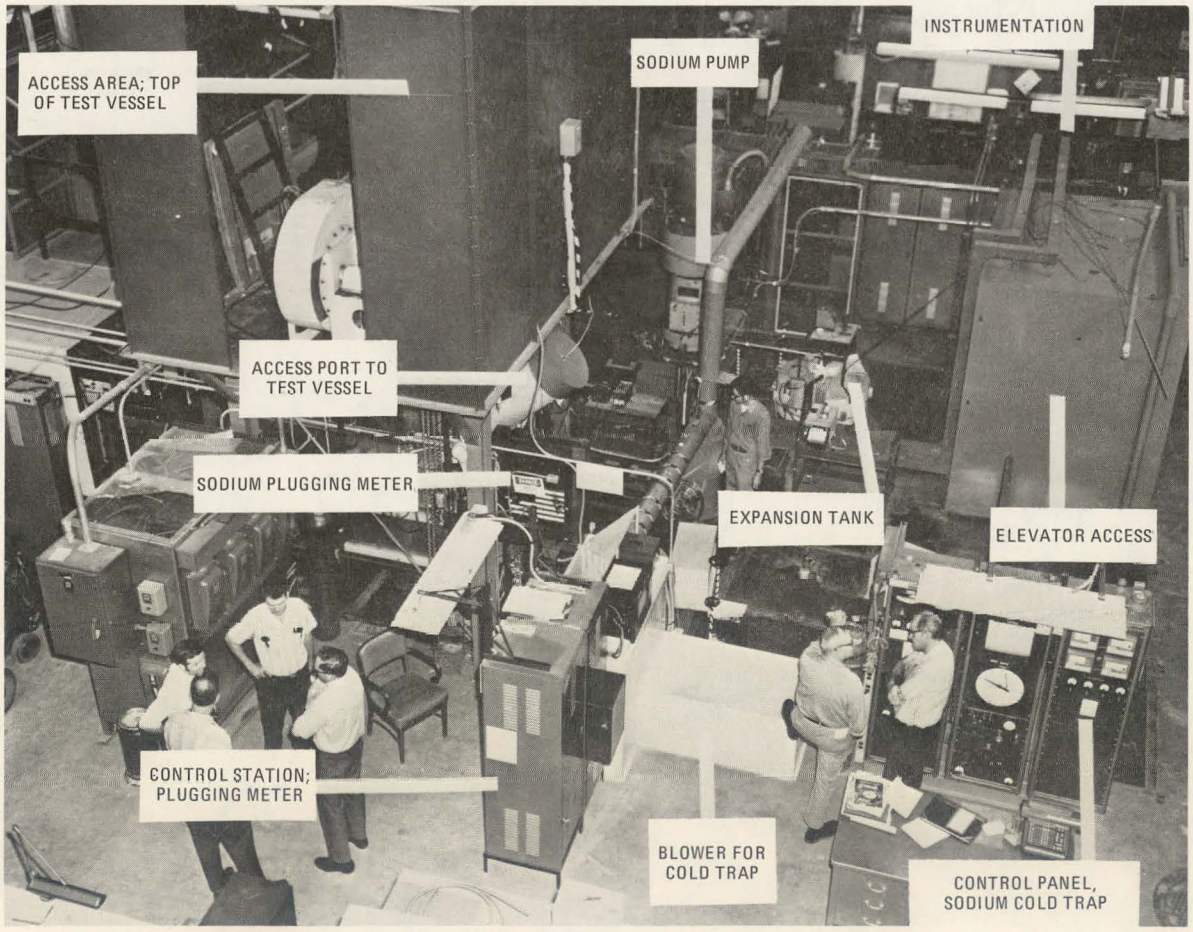

Figure 2. Plan View of Operating Floor Area.

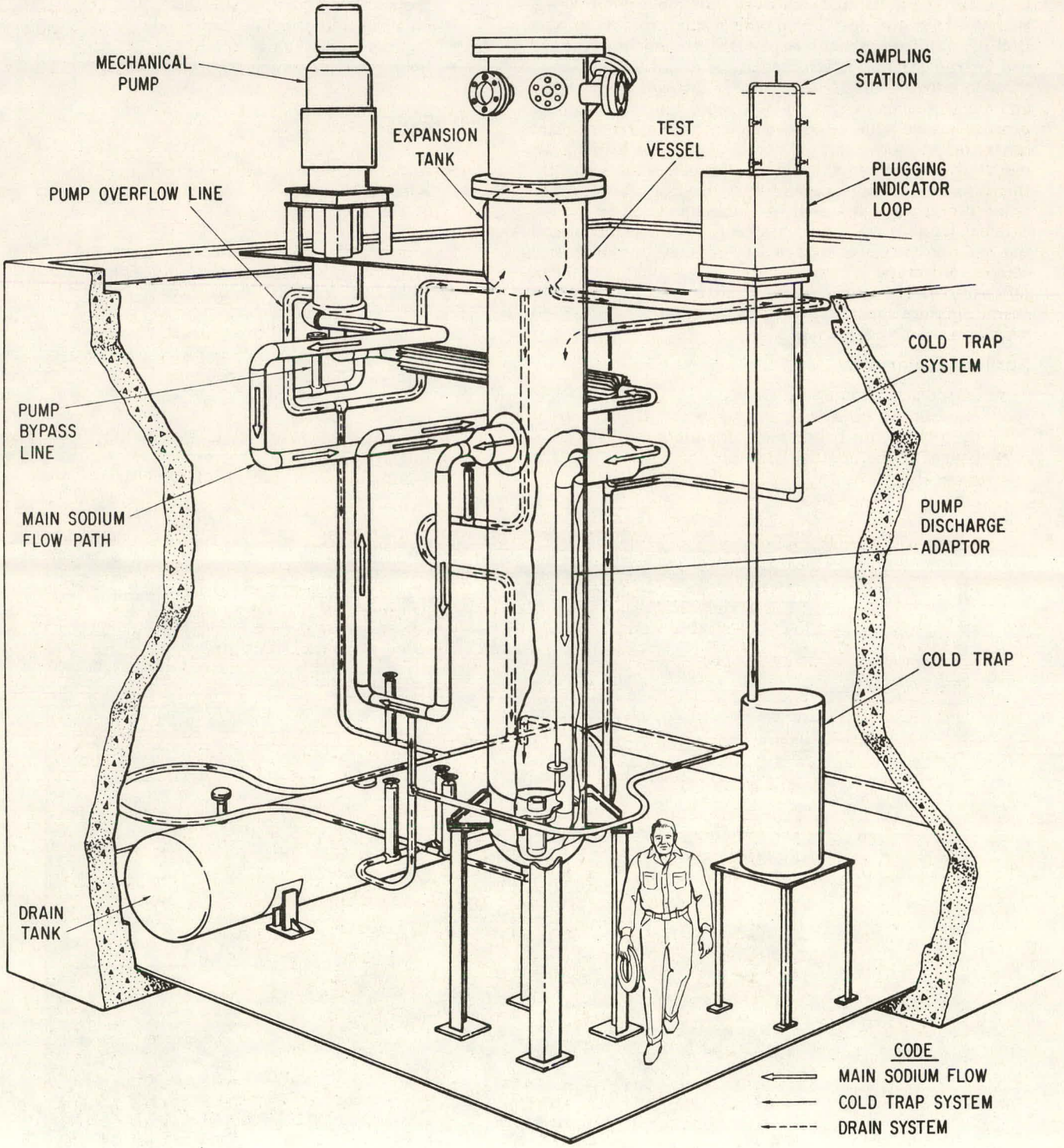

Figure 3. Isometric of CCTL Test Pit.

FAST NEUTRON GENERATOR (FNG)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Use. Determination of fast neutron cross sections and microscopic nuclear properties of fissile, fertile, structural and other materials requisite to economical and safe design of LMFBR systems.

Facility Description. The FNG facility is located in a special purpose building adjacent to the ZPR-6 critical cell with provision for providing positive ion beams in the FNG target area for nuclear data studies and within the ZPR cell for studies of fast critical assemblies. The physical plan of the facility is shown in Figure 1.

Basic characteristics, capability and major components of the FNG facility are:

- Fast Neutron Generator. The FNG is a high-intensity tandem positive ion accelerator operating on the Dynamitron principle. Operation is in both pulsed and steady-state modes with the production of mono-energetic, pseduowhite and white neutron sources. Contemporary operating characteristics are:
  - Beam Energy. Either protons or deuterons are accelerated to energies up to 8 MeV. The equivalent neutron source energies with different targets are 0.03–8.5 MeV and 14–20 MeV with high intensity and 0.01–25 MeV with reduced intensity.
  - Beam Current (tandem model). DC (steady state) beams of up to 200 μamps are available. Nanosecond peak pulsed (rate <2 MHz) intensities are 1–15 milliamps. Peak micro- and millisecond pulsed intensities are $\lesssim$ 200 μamps.
  - Beam Energy Stability. Ion beam energy stability is $\approx$1 keV throughout the operational range.
  - Neutron Producing Targets. Five neutron-producing targets are in routine operation with a potential for expansion to 9. Neutron source reactions are primarily ^{7}Li(p,n), ^{3}H(p,n) and D(d,n).
- Digital Data Acquisition and Reduction and Process Control. An extensive complement of digital computing equipment is employed in data acquisition and treatment and in experimental control including:
  - Two SEL-840 MP disc-based systems each with 16k of 24-bit words.
  - Three CDC-160A tape-based systems each with 8k of 12-bit words.
  - One CDC-1700 disc-based system of 20k of 18-bit words.

Auxiliary Equipment.. Special experimental systems dedicated to the program include:

- 10-Angle Nanosecond Time-of-Flight System. This is a unique system for velocity-spectra measurements of emitted neutrons resulting from scattering, fission and other processes. Flight paths extend to $\approx$7 meters and reaction (detection) angles to $\pm 160^{\circ}$. The

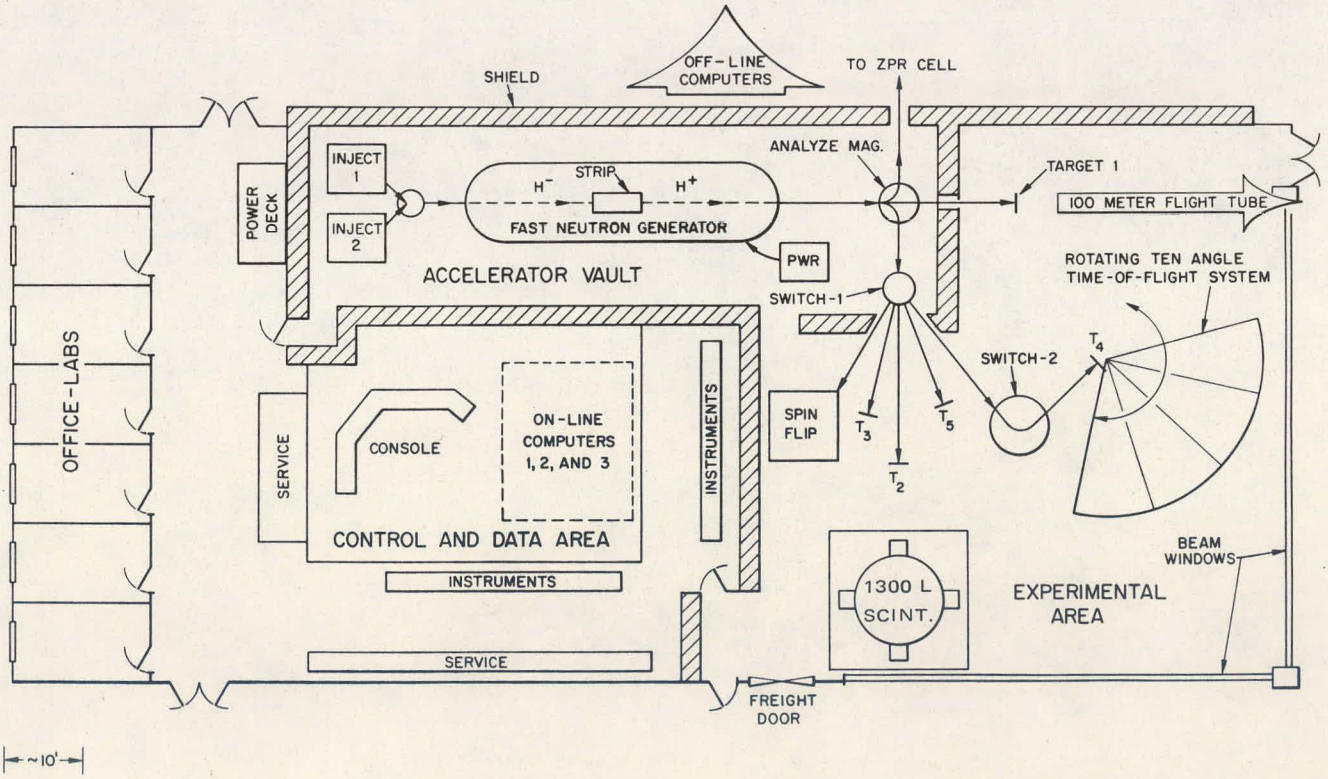

Figure 1. Schematic Fast Neutron Generator Facility.

concept and magnitude of this system is indicated in Figure 2.

- Large Liquid Scintillator. A 1300-liter liquid scintillator system inclusive of static (lead) and dynamic (anti-coincidence system) shielding for very low backgrounds is employed in neutron capture studies.
- 100-Meter Flight Path. A 100-meter evacuated flight path is employed in white-source and monoenergetic-source neutron studies including total and partial cross sections.
- Spin-Flip Solenoid. A magnetic system is applied to studies of spin-parameters in neutron processes.
- Delayed Neutron Measurements System. Beam pulsing, detection, and timing equipment is utilized in delayed fission neutron measurements.
- System for (n, n'γ) and Gamma-Ray Production Measurements. A special system employing GeLi detection is used for neutron-induced gamma-ray measurements with emphasis on high precision in both magnitude and energy resolution.

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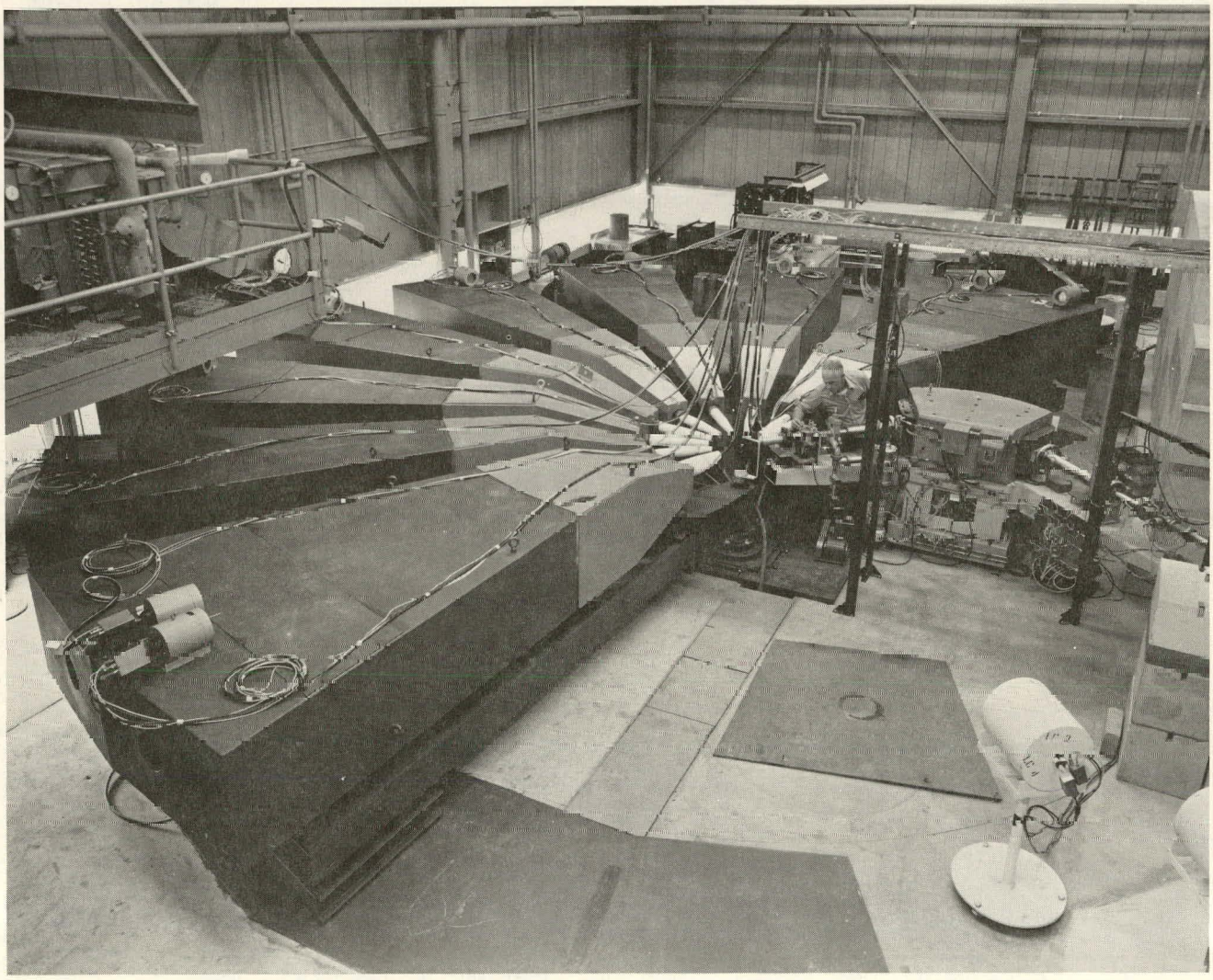

Figure 2. View of the 10-Angle Fast Neutron Time-of-Flight System.

500-GPM WATER LOOP

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses.

- Studies of vibration of flexible rods in axial flow.
- Scale-modeling of flow-induced phenomena.
- Investigate component behavior in selected flow fields.

Facility Description. The loop (Figure 1) features one pump which supplies a maximum of 500 gpm of domestic water at room temperature at 150 psig from an open-top tank to a distribution header at the bottom of the 20-ft test section area. The header can service three test legs. Massive supports and loop-piping vibration isolators are employed to minimize structural-borne vibration of the test section.

Test Capability. The test section area extends 20 ft from the mezzanine to the grade level of a high-bay building (see Figure 2). Ample space is available for either vertically- or horizontally-oriented test sections, and for local visual monitoring/recording equipment. The loop control console and the data acquisition and reduction equipment are located in a centralized control room which services all test facilities in the building (see Figure 1). The facility is completely operated by remote control.

Auxiliary Equipment.

- 10-ton overhead crane.
- Hewlett-Packard 5451/71A FFT (Fast Fourier Transform) System for data acquisition.
- Two 14-channel FM 1-in. magnetic tape recorders.
- Signal-conditioning equipment and instrumentation to measure flow conditions and structural vibrations.

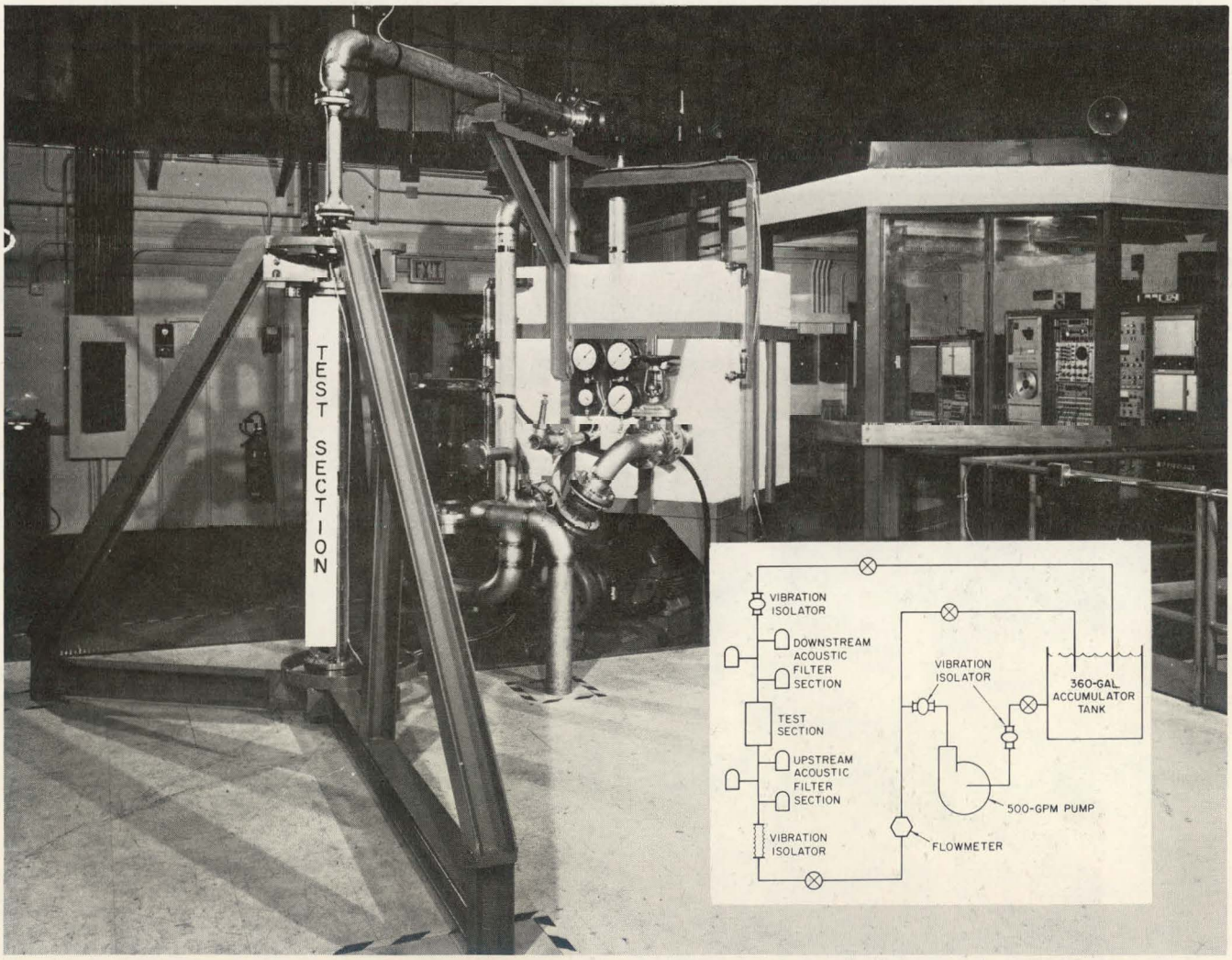

Figure 1. 500-GPM Water Loop Centralized Control Room with Data Acquisition and Reduction Equipment Partially Visible to the Right and Rear of the Accumulator Tank.

Bibliography.

1. "Flow-Induced Vibration Scale Model Tests of CRBR-UIS," Argonne National Laboratory, ANL-CT-76-4, July 1975.

2. "Flow-Induced Vibration Scale Model Test of the CRBR Instrument Post," Argonne National Laboratory, ANL-CT-76-26, February 1976.

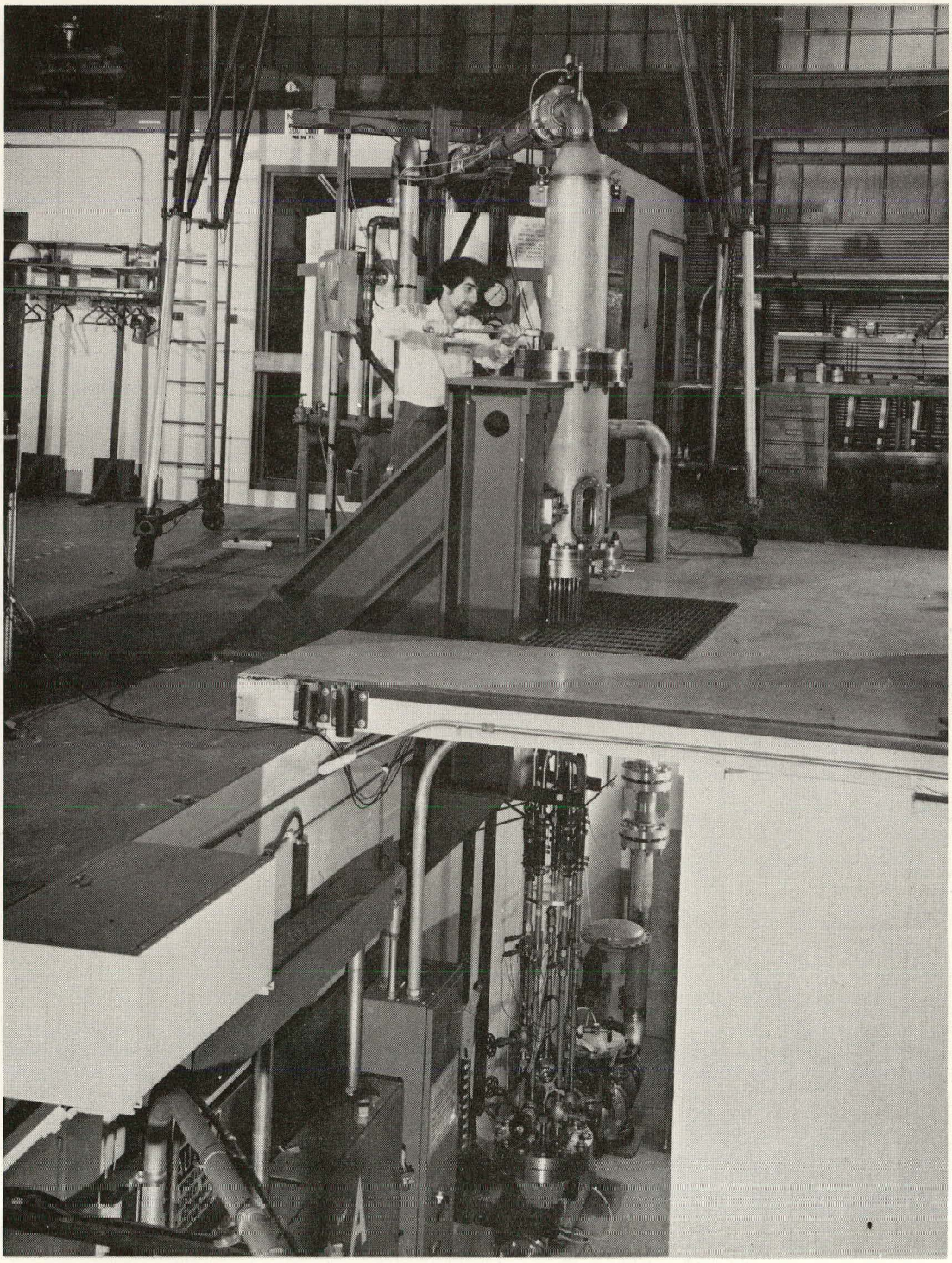

Figure 2. Technician Installing CRBR Upper Internal Structure Components for Flow-Induced Vibration Test.

FLOW-INDUCED VIBRATION TEST FACILITY (FIVTF)

Location ANL-E, Argonne, Ill.

Status Phase I (1500 gpm) Operational
Phase II (8000 gpm) July 1977

Principal Uses.

- Flow-induced vibration tests of nuclear and non-nuclear systems and plant components including valves, reactor upper plenum components, instrumentation wells, thermal liners, and flow-directing baffles.
- Experimental studies of tube bundles in crossflow, single cylinders in skewed flow, and cylinders subject to inflow turbulence and wake buffeting.
- Performance of tests, both scale-model and prototypic, to evaluate specific LMFBR component designs from the standpoint of flow-induced vibrations.

Facility Description. The FIVTF has four pumps with flow rates of 500, 1000, 2500, and 4000 gpm at 150 psig discharge pressure. The pumps discharge individually through their own control and by-pass valves, flowmeters and piping (including pressure and temperature indicators) to an accumulator. Thus, the flow can be varied to provide discrete and stable water flow from 50 gpm to a maximum of 8000 gpm by operating combinations of the pumps and valving. To allow ease of operation by the experimenter, all pumps and valves are operated remotely from the existing data acquisition facility via a control console.

To provide for "quiet flow," various design considerations have been employed. Piping with a minimum number of bends (all long radius), and of the largest diameter practical, is used to reduce turbulence. Structural-borne vibrations are minimized by using vibration attenuation pads at the pipe

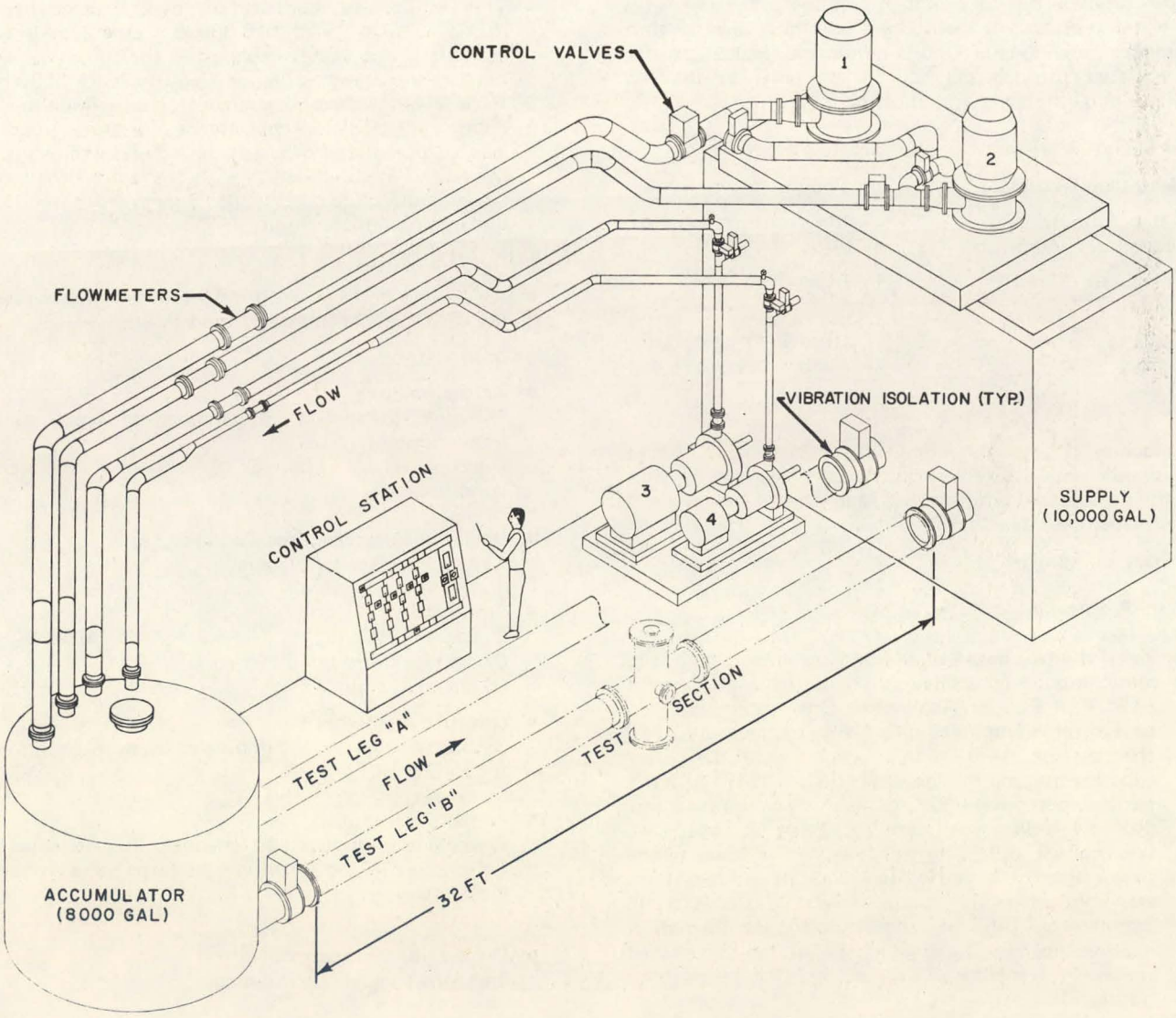

Figure 1. Isometric View of Flow-Induced Vibration Test Facility (FIVTF).

support locations, which are close together (for short span length). Flexible piping sections are used for connection to the pumps and at the entrance and exit to the test sections.

The 8000-gal accumulator at the entrance to the test legs has a maximum turnover rate of once per minute at the highest flow rate. This turnover rate, along with internal baffling and the small height-to-diameter ratio, tends to isolate the pump supply from the test item. Three large diameter nozzles (two horizontal and one vertical), for attachment of test items to the accumulator, are provided with isolation valves, so that one test can be performed while another test is being assembled.

The pump supply tank is constructed of concrete to reduce vibration and has a large volume (~10,000 gal) to accommodate baffling and water-conditioning equipment. The conditioning equipment is used to maintain the quality of the water so that the transducers for the data acquisition system will not be adversely affected by variable water conditions. For example, during an experiment, water temperature will not vary enough to affect water viscosity, and water resistance is maintained at a high level so that transducer shorting-to-ground is minimized. Make-up water is circulated through the appropriate units so that its quality will be the same as that of the loop.

Water Quality Variables	Control Methods
Mean temperature	Cooling tower
Water resistance and chemical content	Ion exchange column
Particulate matter (up to 7 μ)	Filters
Bacteria content	Ultraviolet sterilization unit

The facility has been divided into two phases: Phase I includes the entire system except for the two larger pumps and their associated piping, valves, and flowmeters; Phase II will provide these items.

Auxiliary Equipment.

Data Processing

- Two digital data reduction systems which are minicomputer-controlled systems (Hewlett-Packard 5451 Fast Fourier Analyzers). One system (5451B) has a minicomputer with 16K of memory, two channels of A–D and one channel of D–A for experimental input. The other system (5451A) has a minicomputer with 32K of memory and a magnetic disk for additional memory. Both systems have two-channel capability for analysis of data where time reference is needed (correlation). Both systems are controlled via special keyboards (similar to calculators). All of the standard mathematical relationships for random data can be determined (Fourier Transform, inverse Fourier Transform, transfer functions, etc.). The systems automatically retain amplitude calibration and provide (via digital display) the appropriate scale factors and units for all analyses performed.
- Two 14-channel 1-in. magnetic tape recorders for Wideband FM Record/Reproduce (Honeywell 7600 and 5600).
- An analog servo system (Spectral Dynamics equipment SD-1056 amplitude servo, SD-101B dynamic analyzer, SD-101BS slave) used for swept sine control of force generators for vibration testing and transducer calibration.
- A real-time spectrum analyzer (Spectral Dynamics model SD-330A) capable of 150-line spectrum analysis with cursor-controlled digital readout of frequency and amplitude values. The analyzer has built-in spectrum averaging and X-Y plotting of data simultaneous with viewing on a built-in CRT.
- Associated electronics (filters, amplifiers, scopes, plotters, and calibration equipment).

Vibration Exciters

- This equipment consists of two electrodynamic shaker systems with 10,000-lb (Line 275A) and 1,200-lb force vector capabilities for sinusoidal and random vibration testing at frequencies to 5000 Hz. It provides excitation sources for determination of component vibration characteristics (natural frequencies, damping, and mode shapes). These exciters have been used in combination with the water flow loops to provide controlled force inputs to test items undergoing flow testing.

Lifting Apparatus

- An overhead crane, with a 10-ton capacity, services the entire area occupied by the FIVTF.

Instrumentation

- Accelerometers
  Piezoelectric—approx. 30 channels of signal conditioning (high sensitivity)
  Piezoresistive—approx. 6 channels of signal conditioning
- Optical displacement measuring systems
  4 channels—sensitivity variable
- Strain gauges
  6 channels dynamic
  6 channels static and dynamic
  64 channels static
- Hot-film anemometry
  3 channels signal conditioning calibration equipment for H_2O or air
- Pressure-transducers
  Piezoelectric—approx. 30 channels (high sensitivity)
  Calibration: static (dead weight tester) and dynamic via comparison
- Displacement transducers
  2 channels signal conditioning
- Reflective photostress equipment
  Includes strobe lights

HIGH-TEMPERATURE LIQUID-METAL MHD FACILITY (HT-LMMHD)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses. The High-Temperature Liquid-Metal MHD Facility is used to evaluate sodium-argon two-phase liquid-metal MHD generator performance as a function of temperature and other parameters.

Facility Description. The facility provides known flows of liquid (sodium) and gas (argon) for testing the two-phase liquid-metal MHD generator. It consists of a sodium loop and an argon loop with a common mixer, test section, and separator/sodium dump tank. The sodium is recirculated, while the argon, which is supplied from a tank, is exhausted to the atmosphere from the separator through a scrubber to remove all traces of sodium. The argon exhaust line is equipped with a control valve to fix the generator exit pressure. The sodium loop includes an electromagnetic pump, a flowmeter, control valves, and pressure transducers. The argon loop comprises a heater and control system to heat the gas to the liquid temperature, a flowmeter, a flow-rate controller, and pressure transducers. Nitrogen is supplied from a tank for accessory cooling, and a dc iron-cored magnet provides the magnetic field.

Test Capability. The test section fits between Graylok flanges located about 6 ft (2 m) apart. The generator is inserted into the 11-in. (0.28 m)-long magnet air gap. Instrumentation is provided to measure and control gas and liquid flow rates, pressures, and temperatures and to measure (dc) generator voltage and current, pressure along the generator channel, magnetic field strength, and void fraction distribution in the generator.

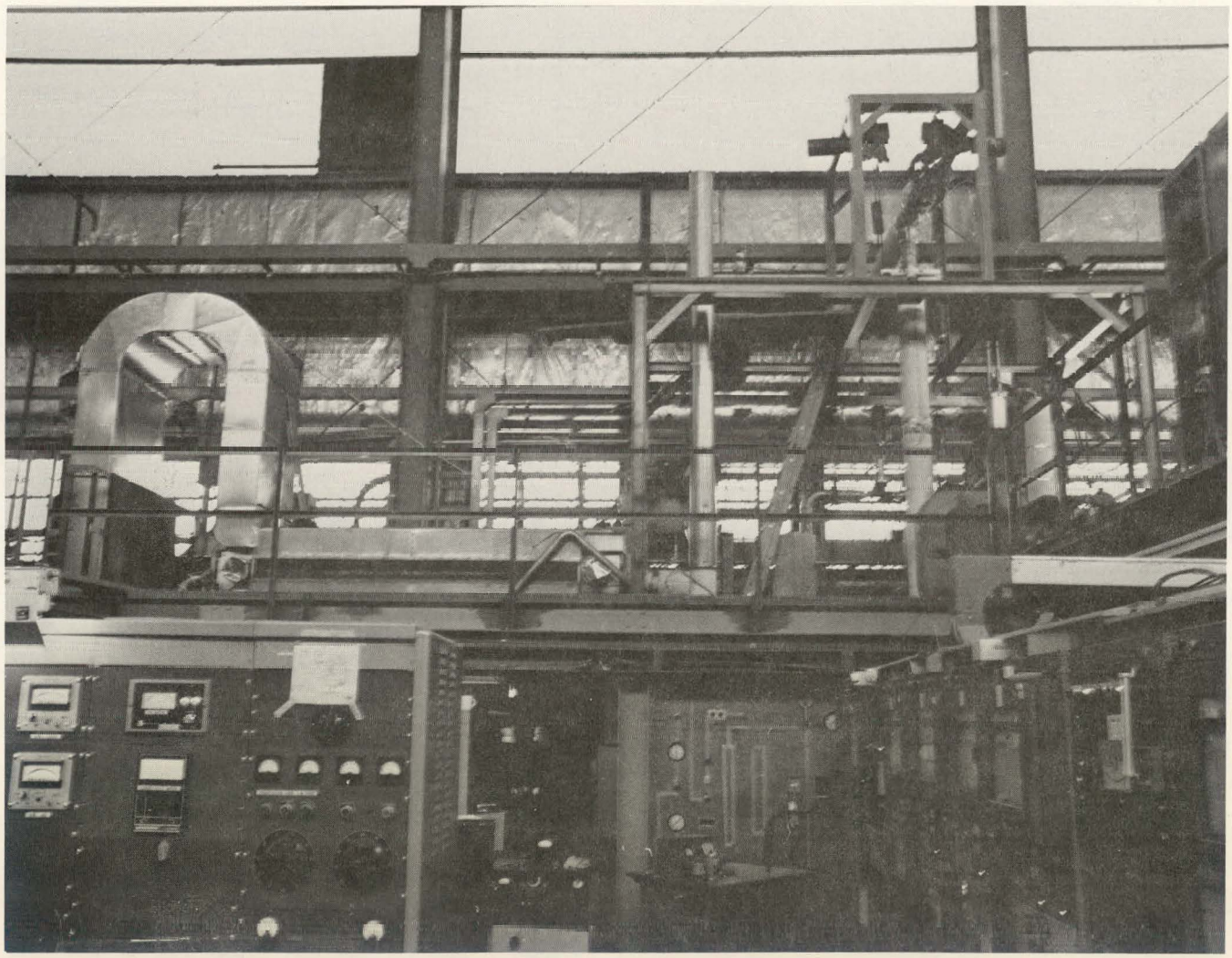

Figure 1. Overall View of High-Temperature Liquid-Metal MHD Facility (HT-LMMHD).

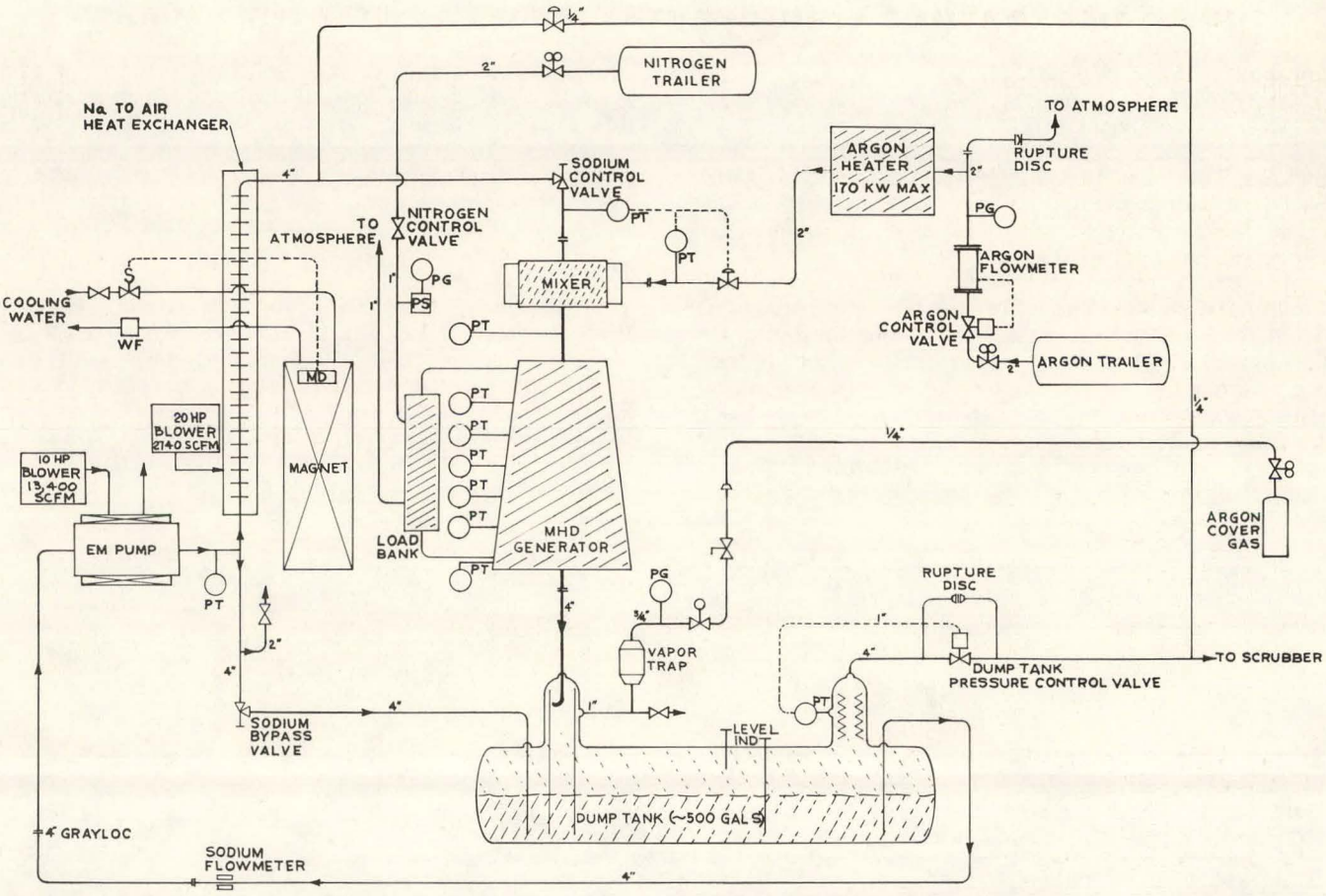

Figure 2. High-Temperature Liquid-Metal MHD Facility (HT-LMMHD) Flow Diagram.

General Parameters

Working fluids	Sodium and argon
Maximum flow rates	500 gpm (26 kg/s) sodium 324 scfm (0.25 kg/s) argon
Maximum pressures	100 psia (6.9×10^5 Pa) sodium 145 psia (1.0×10^6 Pa) argon
Maximum temperature	1000°F (811°K)

Auxiliary Equipment.

- Data acquisition system – up to 96 channels; temperature or voltage input; punched tape output.
- dc magnet, 0 to ~1.0 T - working volume: 22 in. (0.56 m) by 12 in. (0.30 m) by 11 in. (0.28 m) gap.
- Void fraction measuring system, 3 channels.

Bibliography.

1. "HT-LMMHD System Design Description," Argonne National Laboratory, to be published.

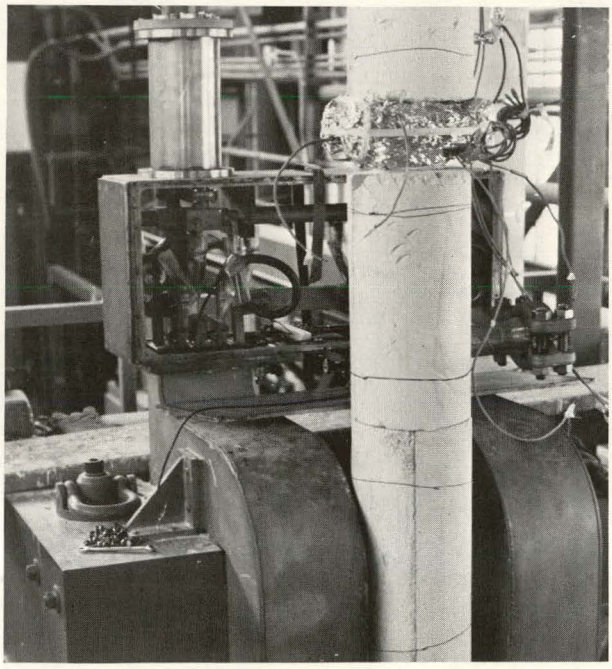

Figure 3. Magnet with Dummy Test Section.

LMFBR HEAT TRANSFER SIMULATION LOOP (HTSL)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses. The LMFBR Heat Transfer Simulation Loop is used to generate heat transfer, coolant mixing, and hydraulics data in support of the FFTF, Demo Plant, and other LMFBR facilities. These data include sodium superheat, boiling sodium stability, sodium burnout limitations, sodium mixing in 1/15-scale LMFBR outlet plenums, and mixing of sodium streams at two different temperatures in reactor components.

Facility Description. The HTSL has a circulating volume of 33 gal sodium. The main loop piping is 1-1/2 in. Schedule 40 Type 304 stainless steel. Key components include a helical induction pump, a sodium-to-air heat exchanger, preheater, an independently heated test section, plenum test vessel, fill and drain tank, and an oscillating plugging meter. Pump power is controlled by Variac transformers. Sodium flow rate can be adjusted by several flow control valves. The flow rate can be increased at a lower pressure by modifying the pump duct. Air flow to the heat exchanger can be varied. Trace heating is employed to maintain minimum loop operating temperature. In addition to standard loop instrumentation and safety interlocks, the loop is filled with sodium under vacuum, the plenum is blanketed with argon, plenum sodium level is observed with inductance-type level detectors, and all piping, valves, and components are monitored with sodium leak detectors. Finally, the entire loop is enclosed in a steel containment shell, which can be filled with inert nitrogen in the event of sodium leakage.

Experimental Parameters

		Pressure, psia	Temp., °F
Plenum outlet to heat exchanger outlet		100	1200
Heat exchanger outlet to pump inlet		100	1000
Pump inlet to plenum		500	1000
Sodium flow rate	30 gpm		
Power	200 kW		
Heat exchanger capacity	100 kW at 1000°F inlet		
Pump characteristics—helical induction	500 psi at cut-off 100 psi at 30 gpm		

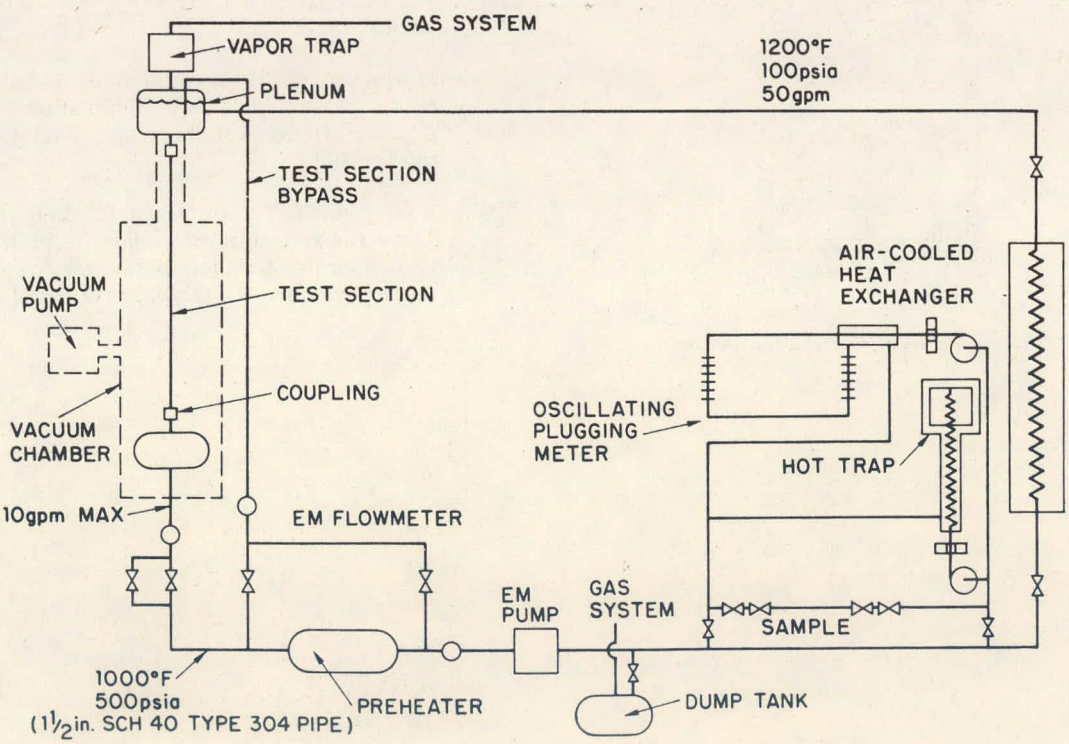

Figure 1. Schematic of HTSL System and Components.

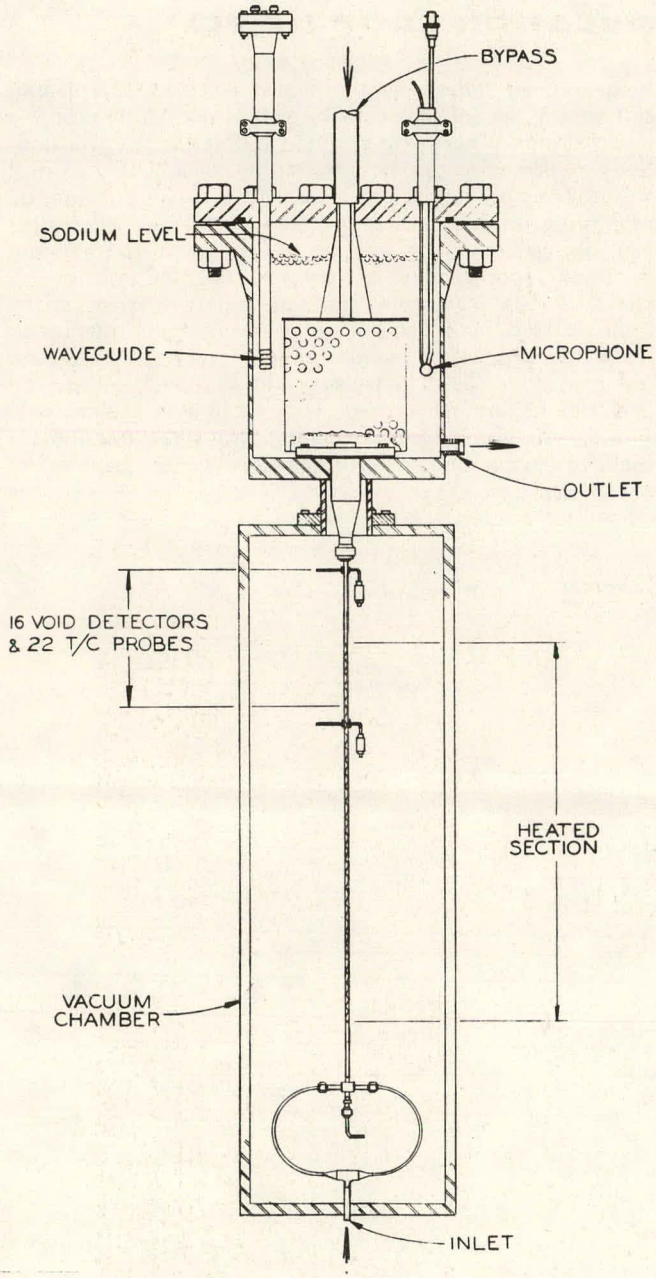

Figure 2. Typical Test Section: Plenum Arrangement.

Test Capability. The heated test area is enclosed in a stainless steel vacuum chamber (18 X 18 X 60 in.), and is accessible through two side doors of the chamber. Sodium in this area can be heated by either electrical-resistance or electron bombardment.

The sodium plenum test section is a stainless steel vessel (20 in. dia X 20 in. high) with a removable top flange closure. Liners and core outlet structures can be inserted to provide 1/15-scale LMFBR outlet plenums for sodium-mixing studies.

Auxiliary Equipment.

- Mini computer-controlled data acquisition system (16k of 16-bit words).
  - Disc, magnetic tape deck, line printer, plotter, 32-channel high-speed A/D, 200-channel crossbar scanner, and digital voltmeter.

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3. Anderson, T. T., Bobis, J. P., and Gavin, A. P., "Acoustic Detection of Sodium Voids in the LMFBR Heat Transfer Simulation Loop," Argonne National Laboratory, ANL-CT-75-32, February 1975.

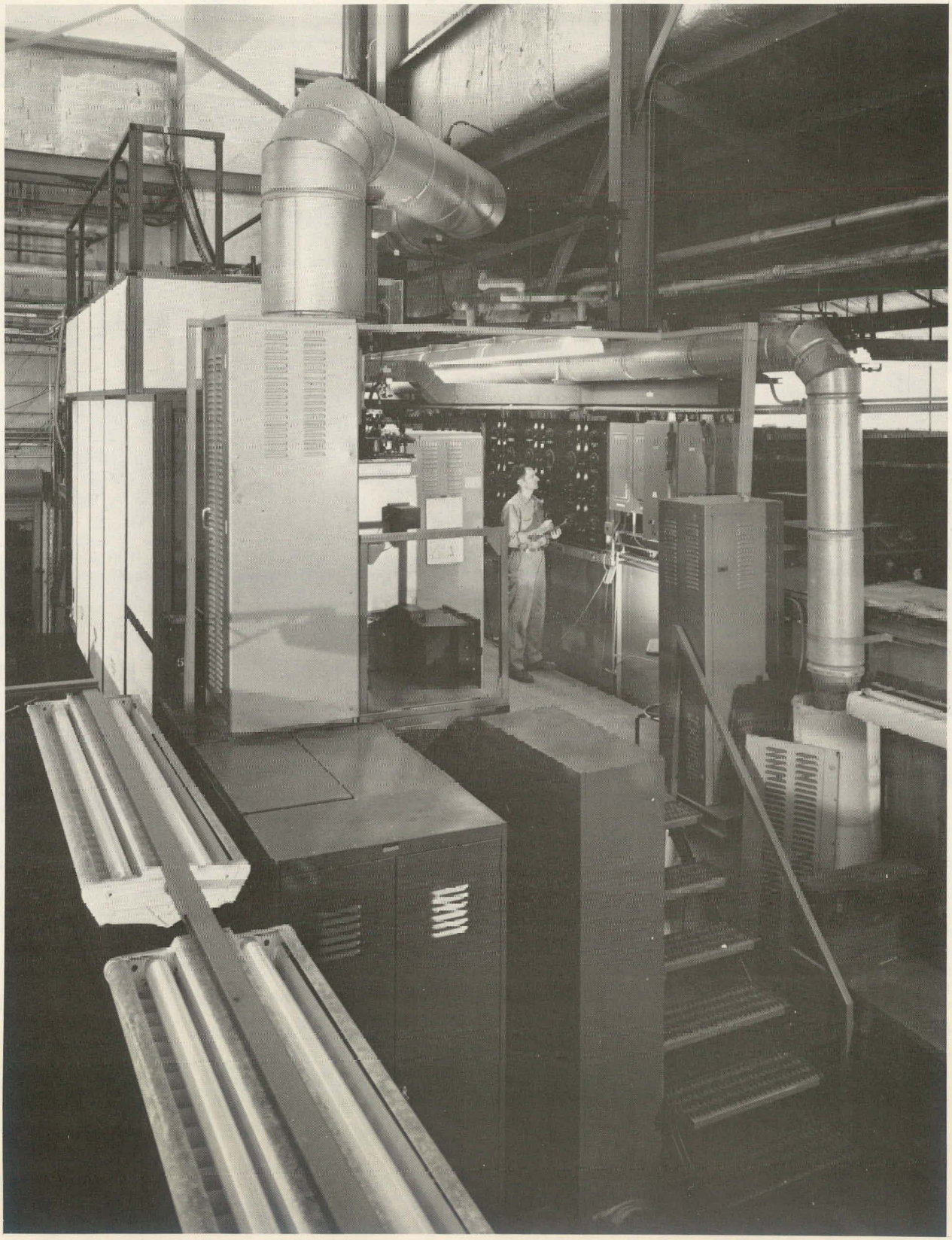

Figure 3. LMFBR Heat Transfer Simulation Loop.

MIXING COMPONENTS TEST FACILITY (MCTF)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses. The facility is used to accumulate experimental data pertaining to thermal-hydraulic phenomena that will aid in the design of compact, thermal-fatigue-free mixers and related piping components for LMFBR plant service. These data include: flow patterns, pressure losses, temperature fluctuations and other thermal parameters associated with mixing of different-temperature streams; size-scaling laws relating mixing characteristics of laboratory models to prototype components; and the effects of nozzle reactions, entrapped gas, and thermal stratification on the resistance of candidate mixing components to thermal/mechanical fatigue.

Facility Description. The MCTF is a steady-state low-temperature low-pressure closed-loop system designed to circulate two streams of water at different temperatures to an instrumented test section (pipe tee or other component) wherein both streams are mixed. Loop piping (4 in. dia) and valves are fabricated primarily of copper, brass, and bronze. As shown in Figure 1, the combined effluent from the test section flows to the suction side of the pump. The pump output is diverted into two streams, one to the heater and the other to the cooler, where the respective initial temperatures are restored. The streams then flow back to the test section to repeat the cycle. Flow rates of both streams are manually adjusted with globe valves. The loop is instrumented with orifice flowmeters, thermocouples, and pressure transducers for continuous monitoring of flow conditions.

Experimental and General Parameters

Pump flowrate	0–400 gpm
Circulating water inventory	600 gal
Water temperature	70–200°F
Cold inlet temperature	70–120°F
Nominal cold inlet temperature	100°F
Hot inlet temperature	70–200°F
Nominal hot inlet temperature	150°F
Inlet temperature difference	0–100°F
Nominal inlet temperature difference	50°F
Cold inlet flow rate	0–200 gpm
Hot inlet flow rate	0–200 gpm
Nominal system pressure	50 psi
Pump head at 300 gpm	32 psi
Temperature control	Continuous
Pressure and flow rate control	Manual

Test Capability. The MCTF is particularly well-suited for testing laboratory models which utilize 4-in.-dia pipe to simulate large prototype mixing components. The basic test section area (40 X 46 in. cross section) is readily accessible and can accommodate vertically aligned components with lengths in excess of 7 ft. Inlet and effluent connections to the test section are effected by 4-in. flanges. Test sections can be fabricated from stainless steel, plexiglas (for flow-visualization studies), or any material compatible with the prevailing hot or cold streams. The piping is designed to accommodate tests involving intersection angles of 180° and 90° between the hot and cold streams, as shown in Figures 1 and 2.

Auxiliary Equipment.

- 50-channel, copper-constantan, 0°C, thermocouple reference junction.
- Honeywell Visicorder Model 1858 with signal-conditioning amplifiers (9 channels).
- Honeywell Model 96 Magnetic Tape Recorder/Reproducer System (7 channels FM record, 4 channels reproduce).
- Dual-channel, 400-volt (10 amp max), variable pulse rate, power supply for experiments using the hydrogen-bubble flow visualization technique.
- Integrating DISA Type 55D31 dc Digital Voltmeter, and DISA Type 55D35 True RMS Voltmeter.
- Spectral Dynamics Model SD330A Real-Time Analyzer for frequency analyses of temperature-fluctuation signals.
- Mobile carts, with height adjustments, for support and positioning of test components.
- Overhead bridge crane.
- Nearby access to machine shop and instrumentation shop.

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1. "System Design Description: Mixing Components Test Facility," Argonne National Laboratory, T0025-0007-SA-00, January 3, 1974.

2. "Mixing Components Test Facility (MCTF): Operating Procedure," Argonne National Laboratory, T0025-0027-SE-00, April 8, 1974.

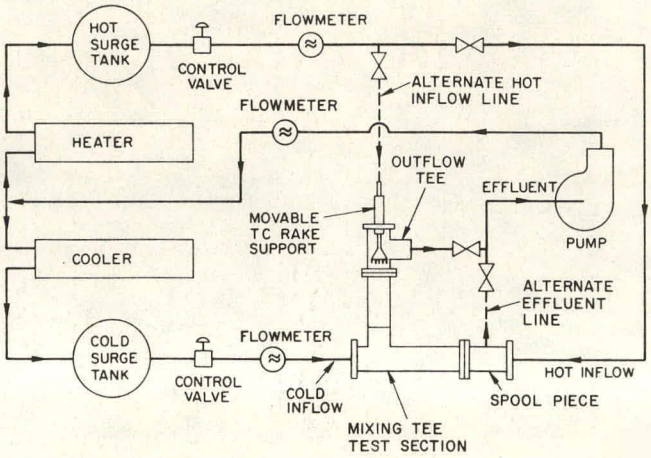

Figure 1. Schematic of MCTF Loop and Components.

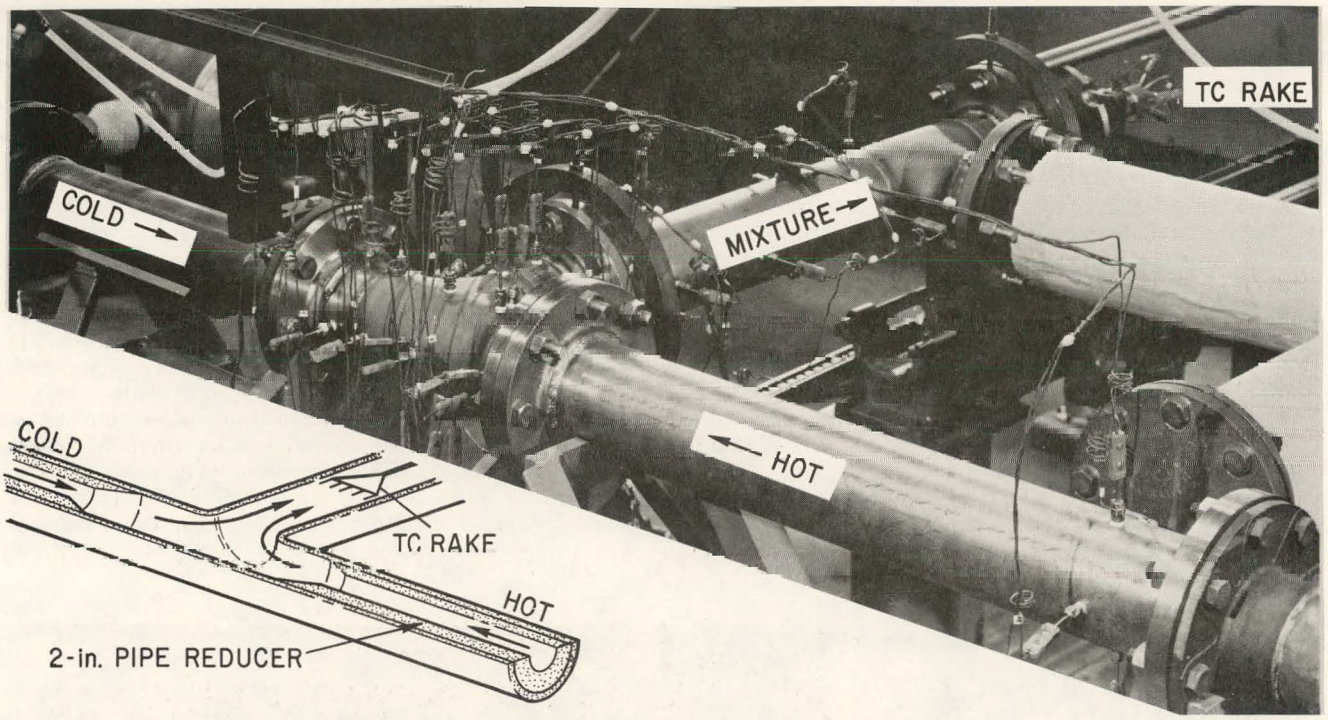

Figure 2. MCTF Test Section Area.

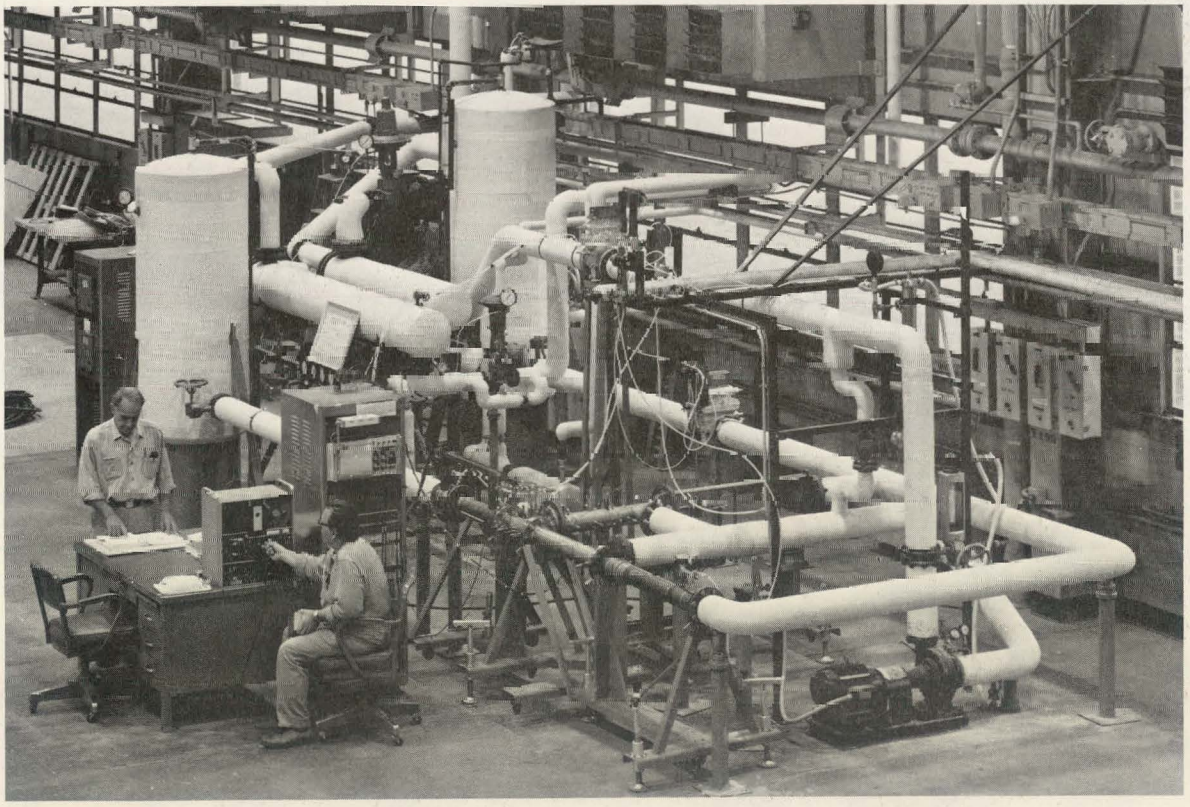

Figure 3. View of Mixing Components Test Facility.

OUTLET PLENUM MIXING TEST FACILITY

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses. The facility can be operated in either of two modes: steady-state or transient. In the steady-state mode:

- Dye injection is employed to enhance flow-visualization studies; e.g., flow distribution in fuel and blanket regions.
- Thermal striping in the outlet plenum is investigated.
- Steady-state temperature field is measured.

In the transient mode, normal scrams or constant flow scrams can be simulated to:

- Observe phenomena of flow stratification.
- Measure thermal oscillations at the hot flow-cold flow interface.
- Measure the exit nozzle temperature transient.
- Obtain data for comparison with mixing-code predictions, and/or code verification and improvement.

Facility Description. The facility (Figure 1) is designed to simulate temperature and flow fields in an LMFBR outlet plenum under either steady-state or transient conditions. During steady-state operation, hot water from the 2000-gal vessel enters the bottom of the pressure chamber and flows up through a 1/10-scale simulated reactor core into the outlet plenum. Simultaneously, cold brine from the 320-gal tank enters the pressure chamber and flows up through a simulated blanket region. A simplified flow diagram is shown in Figure 2. The temperature difference allows the use of thermocouples to measure concentrations of flow through the fuel and blanket regions. A transient is simulated by first achieving steady-state with hot water flow, and then interrupting this flow to simulate reactor shutdown. The flow and temperature transition from steady-state to shutdown is predetermined and computer-controlled. Various 1/10-scale LMFBR geometries can be accommodated in the pressure chamber (see Figure 3).

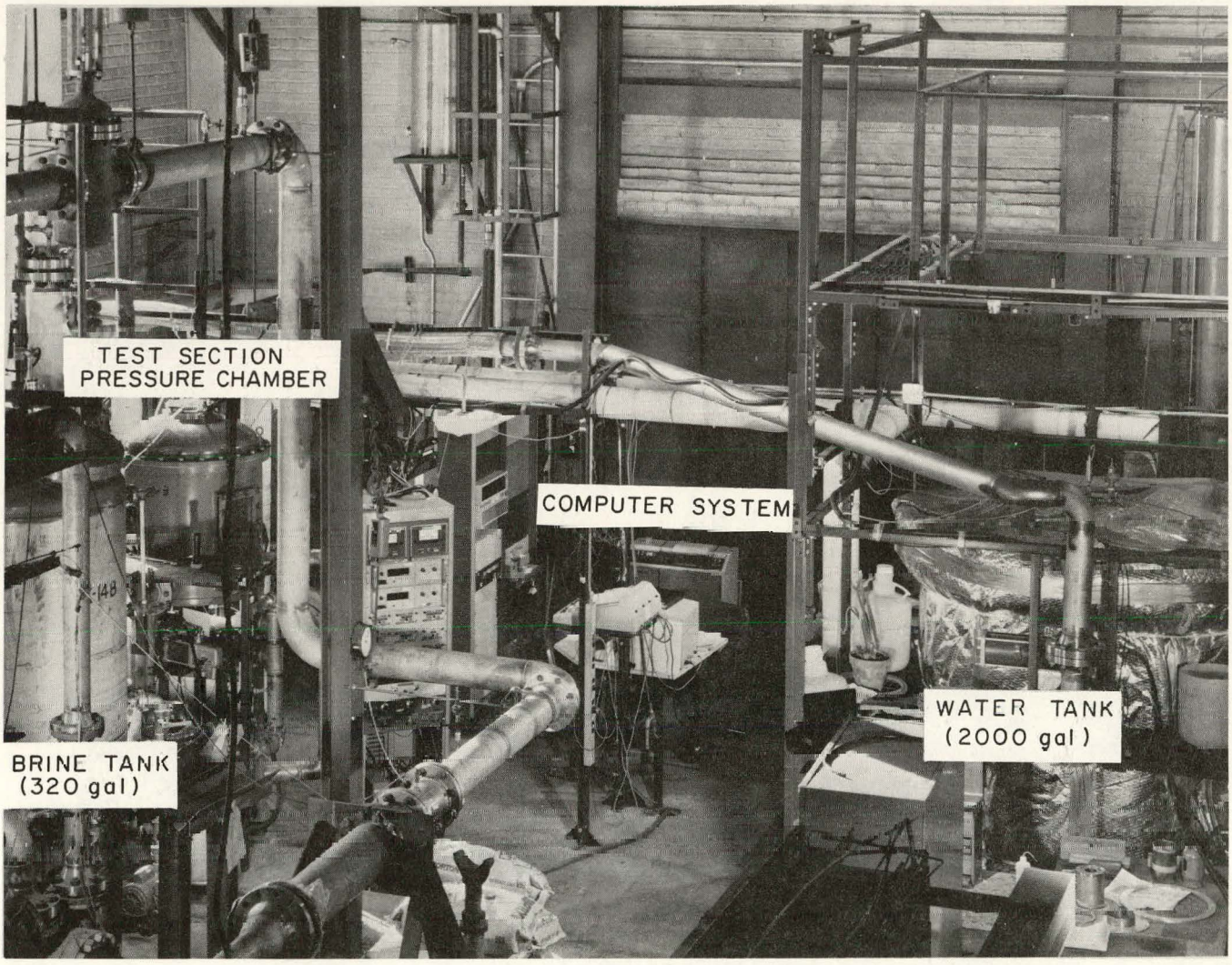

Figure 1. Outlet Plenum Mixing Test Facility.

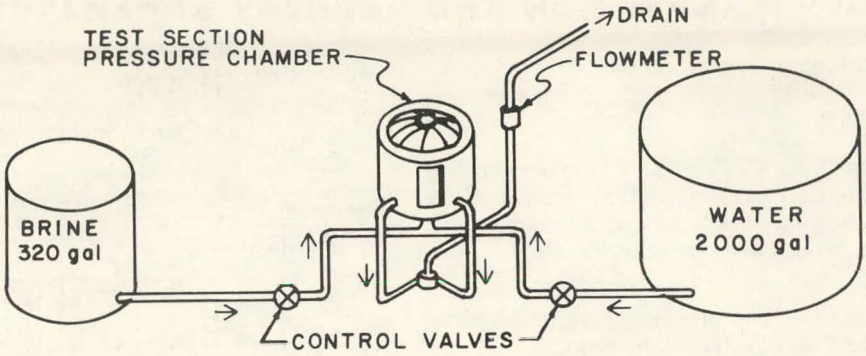

Figure 2. Simplified Flow Diagram.

Design and Operating Parameters

Flow rate	14–400 gpm
System pressure	0–30 psig
Temperature	50–200° F
Piping	4-in. copper and stainless steel
Outlet plenum model	
Diameter	24 in. (max)
Height	27 in. (max)
Scale	nominally 1/10th

Auxiliary Equipment.

- Hewlett-Packard 21MX computer with 32K 16-bit words to control flow and temperature transients. Other computer functions include printing, plotting, and disk storage of data.
- A high-speed (9500 samples/s) multiplexor of 64 channels to record temperature fields and flow rates.

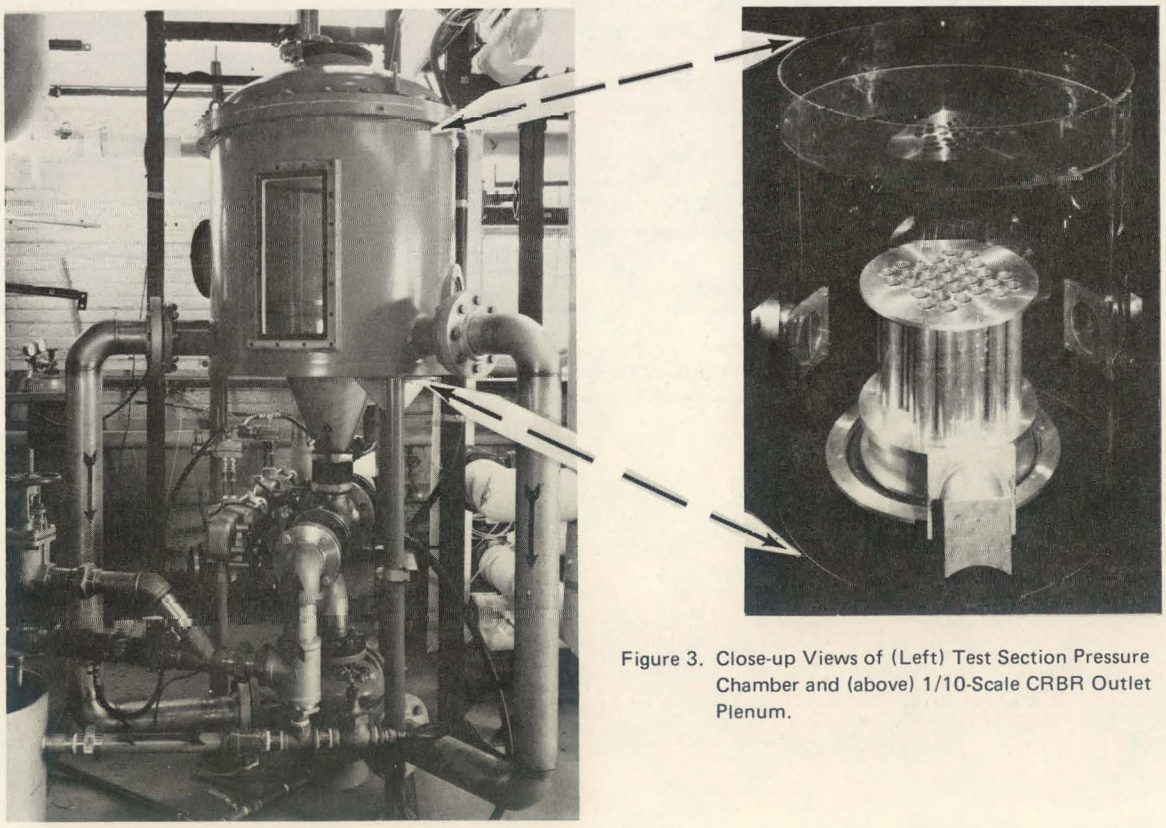

Figure 3. Close-up Views of (Left) Test Section Pressure Chamber and (above) 1/10-Scale CRBR Outlet Plenum.

OUT-OF-PILE EXPULSION AND REENTRY APPARATUS (OPERA)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses. OPERA is a versatile sodium facility constructed primarily to perform prototypic studies of the coolant behavior following postulated coolant transients in an LMFBR. Current experiment plans call for:

- Coolant expulsion and reentry tests for pump coastdown and flow blockage transients.
- Tests determining the sensitivity of coolant monitors, which may be incorporated in the plant protection system, to various postulated flow transients.

Facility Description. OPERA is a once-through (blowdown) facility with the test section in a vertical configuration between the blowdown and receiver tanks. It is designed to run transient tests in a 7-pin bundle with prototypic LMFBR length fuel pins (the fuel pins are electrical heaters). The two large vessels provide an excellent simulation of the upper and lower reactor plenum regions.

Flow is initiated by pressurizing the blowdown vessel to a value larger than that of the receiver vessel and opening the gas operated control valve at the bottom of the blowdown vessel. A complete inlet blockage is simulated by closing the control valve, whereas a pump coastdown is generated by venting the gas in the blowdown vessel.

Test Capability. The entire facility is contained in a stainless steel box 3 X 4 X 18 ft which is placed in a concrete pit for running the experiments. The entire assembly can be removed from the pit and secured on the main floor level when the test section is being changed. In this position, all four sides of the box can be removed and the system can be worked on from all angles.

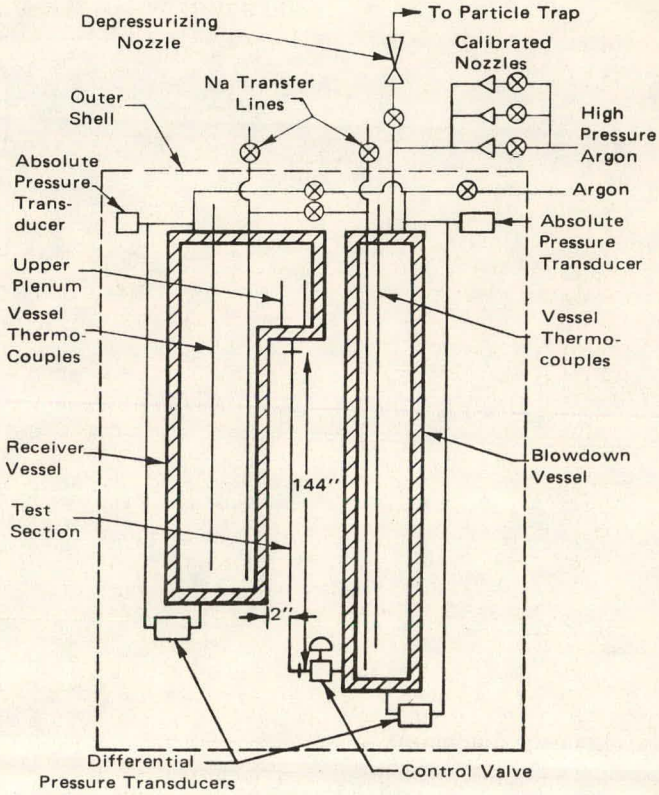

Figure 1. OPERA Loop.

The control and instrument racks as well as the gas supply systems are remoted from the system to provide protection for the operating personnel and to alleviate the tight quarters around the pit area.

TABLE I. GENERAL PARAMETERS OF FACILITY

DESIGN REQUIREMENTS:	
SODIUM INVENTORY	70 GAL MAX.
BLOWDOWN VESSEL DESIGN	200 PSIG AT 1200°F
RECEIVER VESSEL DESIGN	20 PSIG AT 1600°F
POWER AVAILABLE	7 LINES AT 320 V AND 224 AMP (0.5 MW)
POWER CONTROL—POWERSTATS	ONE SINGLE AND TWO 3ϕ
TIME FROM ZERO TO FULL POWER	60 SEC

PLASMA TORCH HEATED FURNACE

Location ANL-E, Argonne, Ill.

Status On Standby – Available on Request

Principal Uses. The furnace will accommodate postaccident heat removal experiments in which contact can be sustained between kilogram quantities of molten UO_2 and slabs of structural materials such as steel, refractory metals, concrete, or sacrificial material. Heat fluxes equivalent to those generated by decay heating will be applied continuously to the UO_2 surface.

Facility Description. Two DC plasma torches will deliver a total of about 40 kW to flowing argon gas. As shown in Figure 1, the gas will enter tangentially into a cylindrical chamber enclosed on its sides by tungsten heat shields and on the top by tungsten screens through which the gas is exhausted. The test specimens, consisting of a layer of UO_2 above structural material, will be placed at the bottom of the heated cavity. Heat radiated from the hot gases, the radial heat shields, and the tungsten screens should maintain a heat flux of about 100 watts/sq cm to a UO_2 surface at 3000°C. The lower portion of the test specimen will be cooled.

Temperatures within the test specimen and the temperature of the UO_2 surface will be monitored. Capabilities for recording rapid temperature changes will be provided. Heat fluxes will be determined during the test.

Auxiliary Equipment. Post-test inspection—metallographic examination of test specimens.

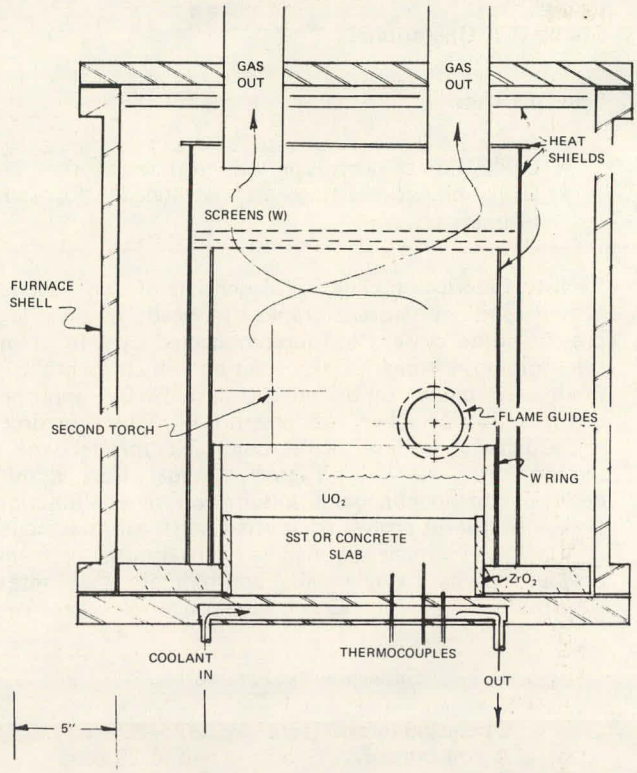

Figure 1. Diagram of Plasma Torch Heated Furnace.

TABLE I. GENERAL PARAMETERS OF FACILITY

NUMBER OF TORCHES	2
MAX. POWER (EACH POWER SUPPLY)	40 KW
ESTIMATED MAX. POWER TO GAS (EACH TORCH)	20 KW
SIZE OF HEATED CHAMBER–DIAMETER	30 CM
–HEIGHT	30 CM
MAX. TEMPERATURE	3100°C

TABLE II. EXPERIMENTAL PARAMETERS

MAX. QUANTITY OF UO_2	10 KG
MAX. DIAMETER MOLTEN UO_2 POOL	20 CM
MAX. DOWNWARD HEAT FLUX (AT 3000°C)	100 WATTS/SQ CM
MAX. SURFACE TEMPERATURE	3000°C
DURATION OF TEST	~1 HR

SODIUM FLOWMETER CALIBRATION FACILITY (SFCF)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses.

- Calibration of prototype sodium flowsensors.
- Development testing of advanced flow-sensing techniques.

Facility Description. The SFCF consists of two electrically heated sodium storage tanks serviced by an argon-pressurization system and interconnected by 2-in. stainless steel piping through a test section which contains the flowmeter being calibrated or tested. By appropriate manipulation of valves, gas pressure is applied to force up to 100 gal of sodium from one tank through the test section into the second tank. Actual flow rates are determined from changes in sodium volume as a function of time, using level probes positioned at 10-gal increments in one tank. A permanently installed conventional permanent-magnet flowsensor gives the approximate flow rate for adjustment of gas pressure- or sodium flow-control valves.

Experimental Parameters

Operating temperature	350–800°F
Argon pressure	up to 25 psig
Flow rate	1–80 gpm
Accuracy of flow rate	±1%

Test Capability. The testing area measures 65 in. between companion flanges of the loop piping. The sensor to be calibrated or tested is installed in a specially designed, electrically heated and insulated test section, which, in turn, is centrally positioned in an enclosure (24 X 24 in.) and connected to the flanges. The system includes signal monitors and instrumentation for recording dc millivolts versus time.

Auxiliary Equipment. A steam scrubber for post-test removal of residual sodium from the sensor is available.

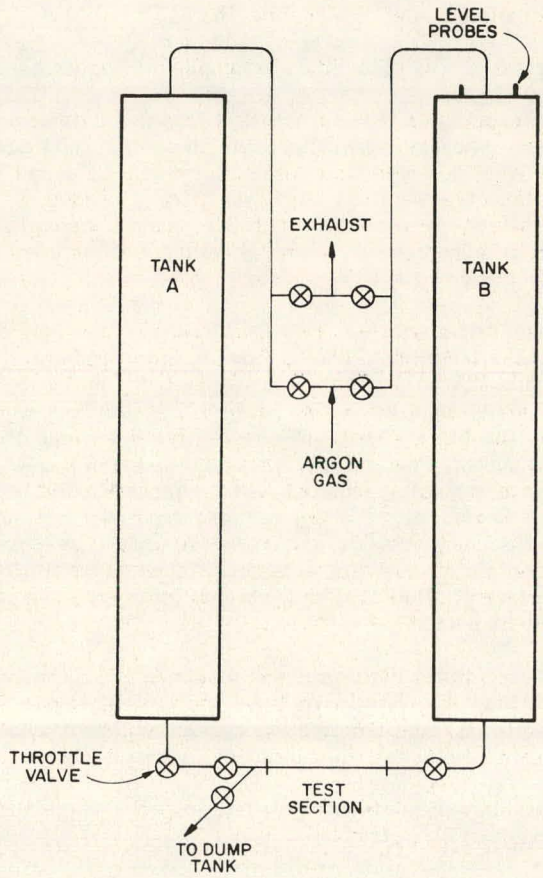

Figure 1. Schematic of SFCF.

Bibliography.

1. Glass, M. C., and Popper, G. F., "The Calibration and Stability of a 1200°F Permanent Magnet In-Core Sodium Flowmeter," IEEE Trans. Nucl. Sci. *NS-15*(1):37–40 (February 1968).

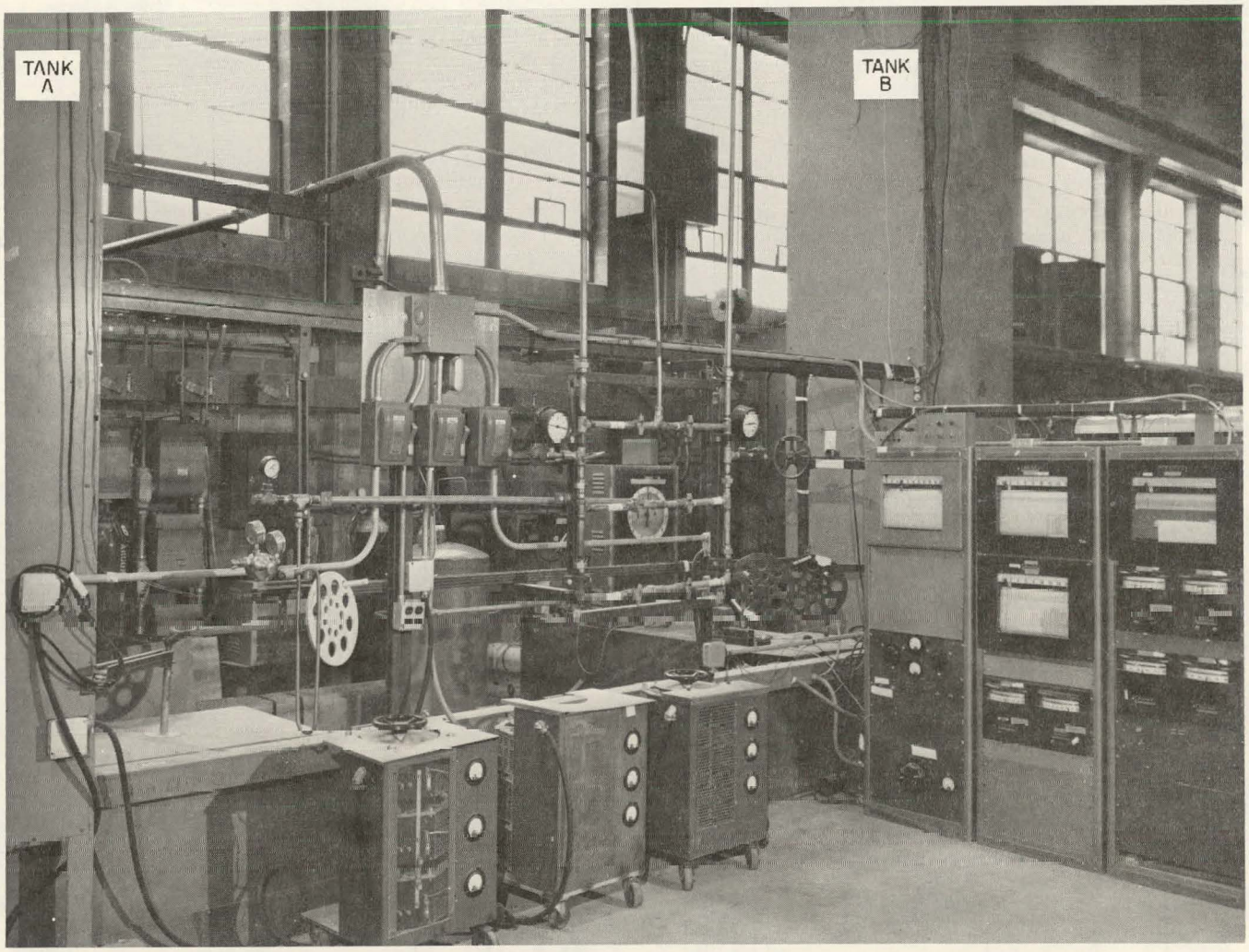

Figure 2. Sodium Flowmeter Calibration Facility.

STEAM GENERATOR TEST FACILITY (SGTF)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses.

- Thermal performance of full-scale tubes.
- Liquid metal-heated boiling water heat transfer.
- Hydrodynamic stability.
- Natural convection, with application to decay heat removal.
- Transient system response characteristics.
- Flow distribution related to inlets, exits, baffles, spacers, and other internals.

Facility Description. The SGTF is a sodium/water system designed to accommodate testing of full-scale LMFBR steam generator tube designs (straight- or bayonet-type) or other vertically oriented components with lengths typically of the order of 60 ft. The system may be operated in the recirculation mode for testing steam generator evaporators or in the once-through mode for integrated units.

Maximum Operating Parameters

Sodium system	
Heat input	550 kW
Temperature	1200°F
Flow rate	70 gpm
Pressure	70 psig
Water system	
Temperature (excluding test section)	650°F
Flow rate (test section plus bypass)	130 gpm
Pressure	2400 psig
Pressure drop	130 psi

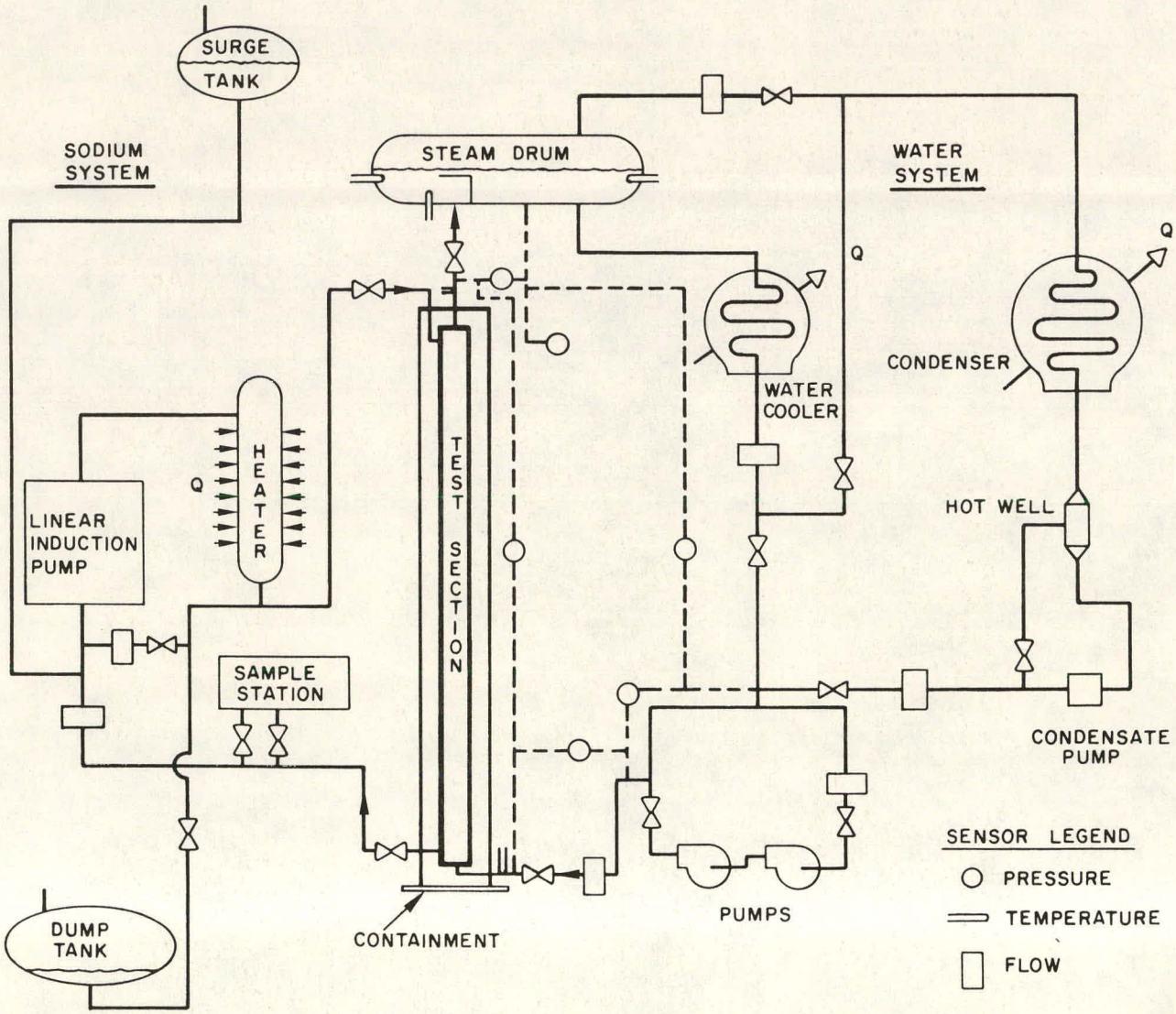

Figure 1. Schematic of Steam Generator Test Facility System and Components.

Test Capability. The test section area inside the building measures 150 ft^2 in cross section and extends 35 ft above and 30 ft below the main floor. The test section, support, and containment are removable; an access port is provided in the building roof for installation and removal of test sections. Longer test sections may be accommodated by adding containment above the roof. SGTF operating parameters are continuously monitored and displayed by a computerized data acquisition system. Operator control and monitoring consoles are located on the main floor adjacent to the test section support tower.

Auxiliary Equipment.

- Computerized data acquisition system for reduction of digital data.
- Analog tape-recording systems and spectrum-analyses equipment for processing analog tapes.
- Instrument calibration standards.

Bibliography.

1. "System Design Description for Steam Generator Test Facility, AI Test Section," Argonne National Laboratory, G0033-0004-SA-00, April 1975.

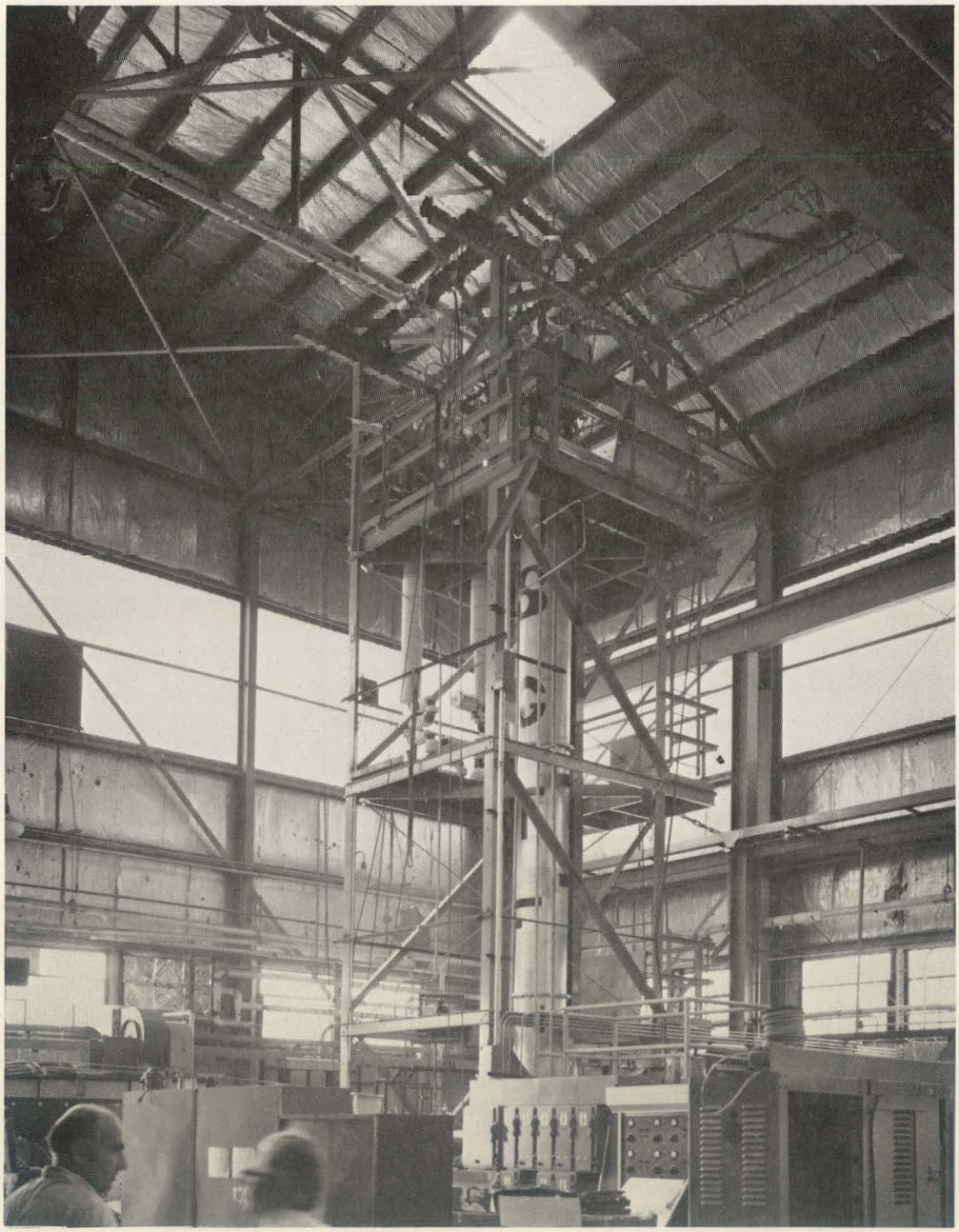

Figure 2. Steam Generator Test Facility.

STRUCTURAL DYNAMICS TEST LOOP (SDTL)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses.

- Experiments in support of base technology studies of flow-induced vibration of in-vessel or ex-vessel components; e.g., crossflow-induced vibration of heat exchanger tubes.
- Development testing of specific LMFBR component designs; e.g., FFTF thermowell assemblies.

Facility Description. Constructed for the most part from stainless steel, the SDTL is designed to circulate demineralized water at flow rates up to 3200 gpm. As shown in Figures 1 and 3, the 3 major loop components are: 2 pumps in parallel which service the main circuit containing the test section, a flow bypass, and a branch with heat exchangers. Flow and pressure in the test section are regulated by 3 valves remotely adjusted from the control panel: the main control and suction-side valves installed upstream and downstream, respectively, of the test section, and the bypass valve connected in parallel with the test section. Suction pressure for the pumps is provided by a surge tank pressurized by compressed air. The heat exchangers have ample capability to remove heat generated by the conversion of mechanical energy by the pumps. Cooling of the heat exchangers, pump seals, and bearings is accomplished by a separate water system. Water flow through the test section is measured by a turbine flowmeter. The SDTL is complemented by extensive instrumentation and data-processing capabilities tailored, but not limited, to the determination of dynamic structural and flow behavior.

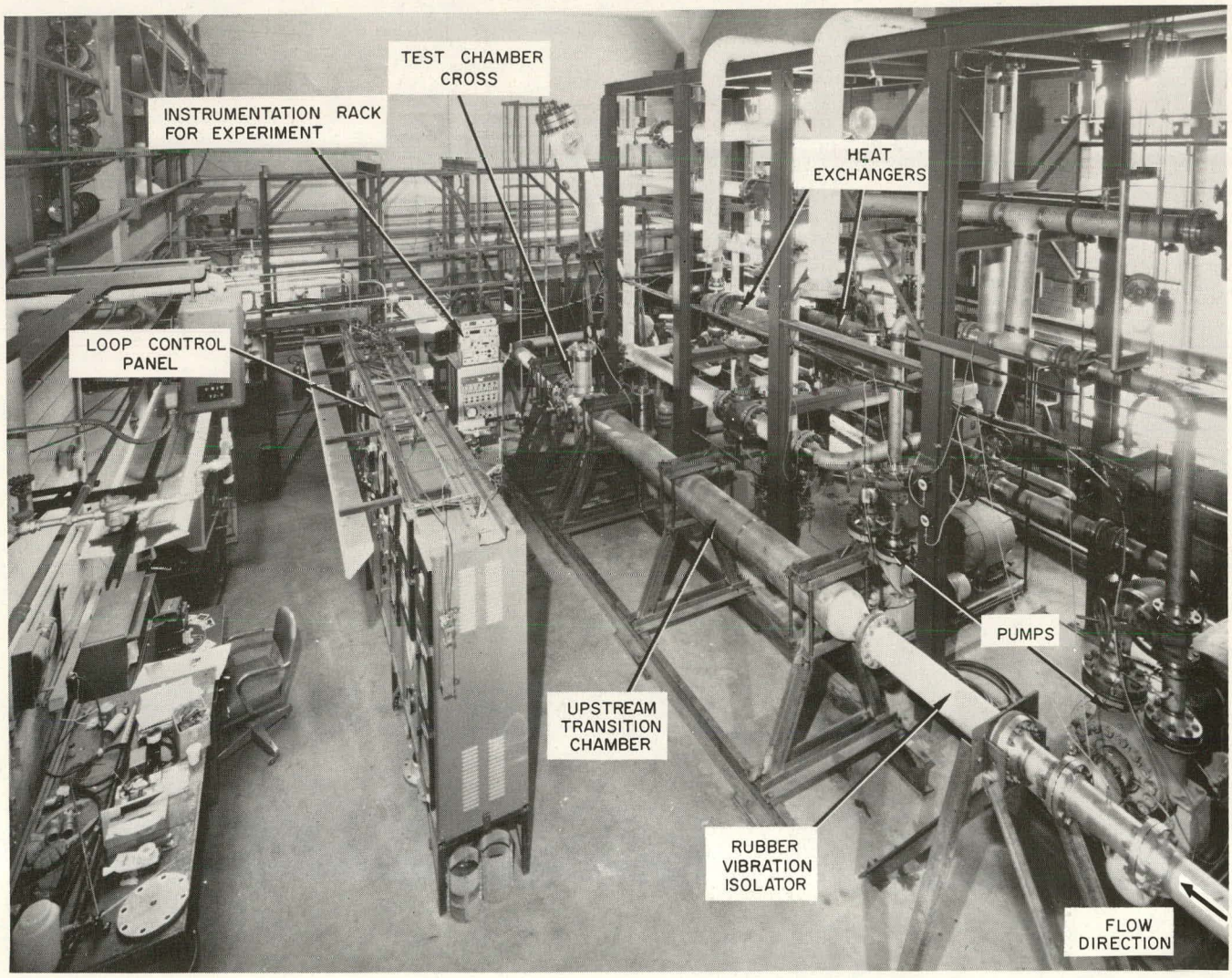

Figure 1. View of Structural Dynamics Test Loop.

General and Experimental Parameters	
Water flow rates	
Maximum with straight pipe in test section and 150 psig total system pressure drop	3200 gpm
Experimental with 50 psig pressure drop in test section	2600 gpm
Flow control	80–3200 gpm
Maximum pump pressure at zero flow	275 psig
Demineralized water	
Maximum temperature	140°F
Temperature regulation	Available
Maximum test section length (along horizontal plane)	
With vibration isolators	20 ft
Without vibration isolators	26 ft

Test Capability. In the test area, the 6-in.-dia loop piping is oriented horizontally about 52 in. above the floor level. This provides space to install and firmly support large test sections with sizeable appendages at a convenient working height. An overhead crane is available. In the flow direction, there is sufficient distance to insert test sections up to 20 ft long, in addition to the two 3-ft-long rubber vibration absorbers that have been used to improve isolation of flow-induced vibration experiments. An aisle adjacent to the test section permits ready access for instrumentation installation, and working space for equipment such as optical displacement trackers. The loop water temperature can be regulated with an automatic control system. There is a presently undetermined excess cooling capacity to remove externally introduced heat.

Auxiliary Equipment.

- Three hot-film anemometer systems.
- Numerous channels of accelerometer/amplifier instrumentation.
- Two optical displacement trackers.
- Twelve strain gage amplifiers (dynamic and bridge type, six each).
- Recording oscillograph.
- Portable FM Magnetic Tape Recorder, 14-channel capacity.
- Back-up data processing facility (not on loop site) with separate FM Magnetic Tape Recorder and Hewlett-Packard Model 5451A Fast Fourier Transform Spectrum Analyzer.

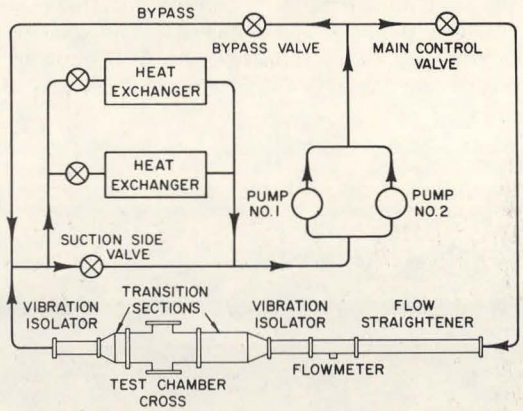

Figure 2. Schematic of Structural Dynamics Test Loop.

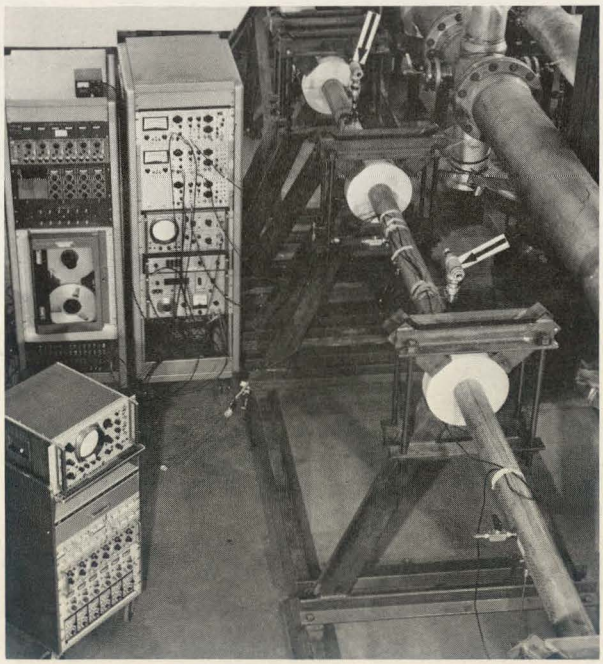

Figure 3. Test Chamber Cross and Transition Sections (upper right) Removed to Accommodate Test Section for Cross Flow-Induced Vibration of Prototype LMFBR Thermowells (all arrows).

ZERO POWER REACTOR-6 (ZPR-6)

Location ANL-E, Argonne, Ill.

Status Operational; In Standby

Principal Uses. The facility may be used to obtain fast reactor physics integral parameters from a wide variety of fast reactor cores, including benchmark, engineering mock-up, or special purpose experiments.

Facility Description. The ZPR-6 is a split-table-design critical facility very similar to ZPR-9, which is described in another profile. ZPR-6 has been modified to include the Variable Temperature Rodded Zone (VTRZ). The VTRZ, which is capable of temperatures up to 550°C, was designed for use in studying temperature and rod-plate heterogeneity effects. It was designed for installation and removal from ZPR-6. With the VTRZ removed, the ZPR-6 matrix is returned to its original pre-VTRZ condition.

A pulsed neutron source and a 100-m time-of-flight (TOF) path give ZPR-6 the capability to measure the neutron spectrum in the low energy range. The facility was placed on standby after the BTRZ and TOF modifications were essentially complete, but, prior to their actual use in experimental programs.

The present matrix size permits studies of the Demonstration Plants (~3000 liters). The reactor tables have sufficient dimensional and weight capabilities to each accommodate a 12 X 12 ft matrix. This "stretch capability" will permit the study of much larger LMFBR assemblies. Design work on this modification is completed and authorization is being requested for FY 1976

Test Capability. Fast reactor cores up to about 3000-liter volume may be built on ZPR-6. Critical mass, coolant-void and Doppler coefficients, power distributions, control rod

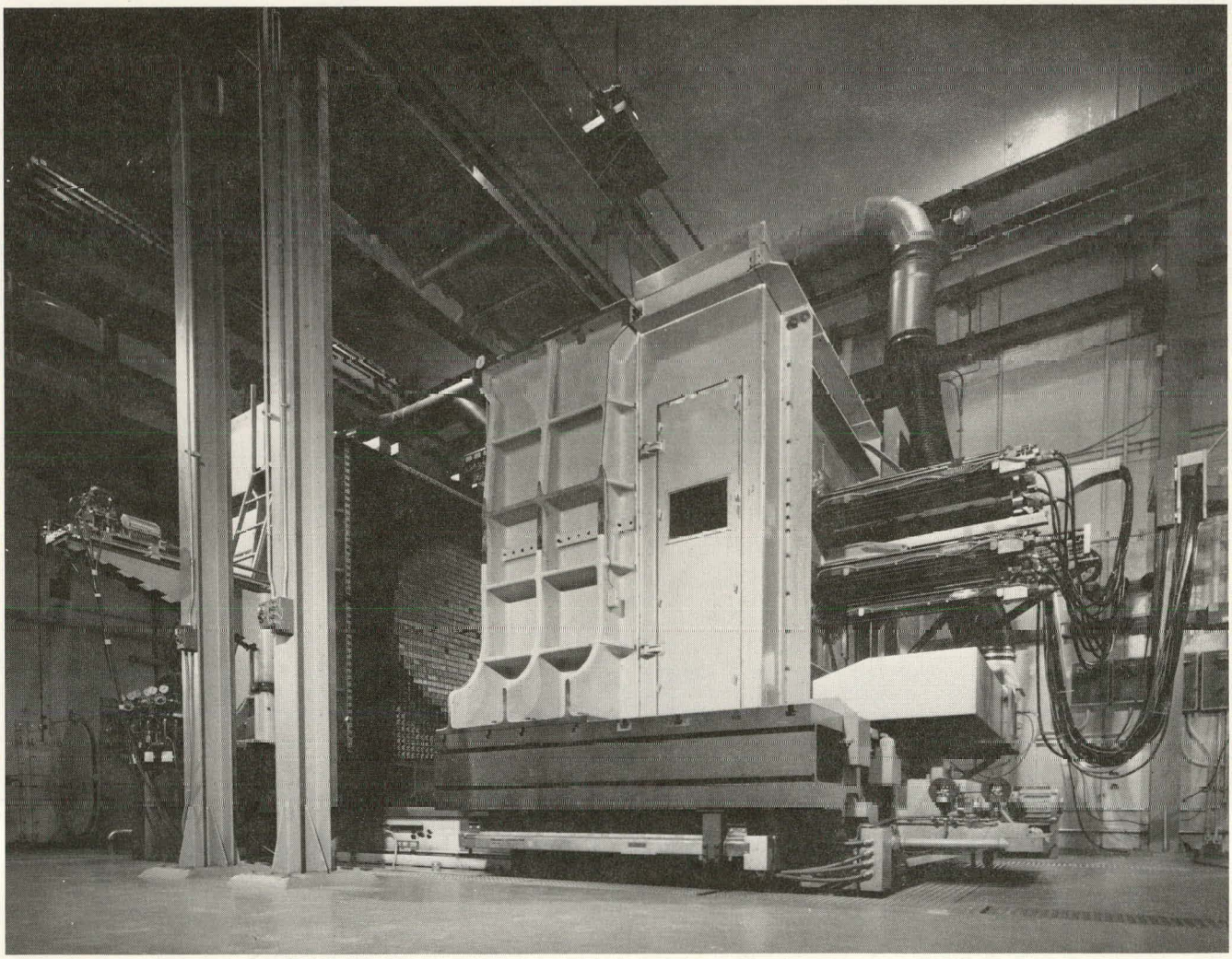

Figure 1. View of ZPR-6.

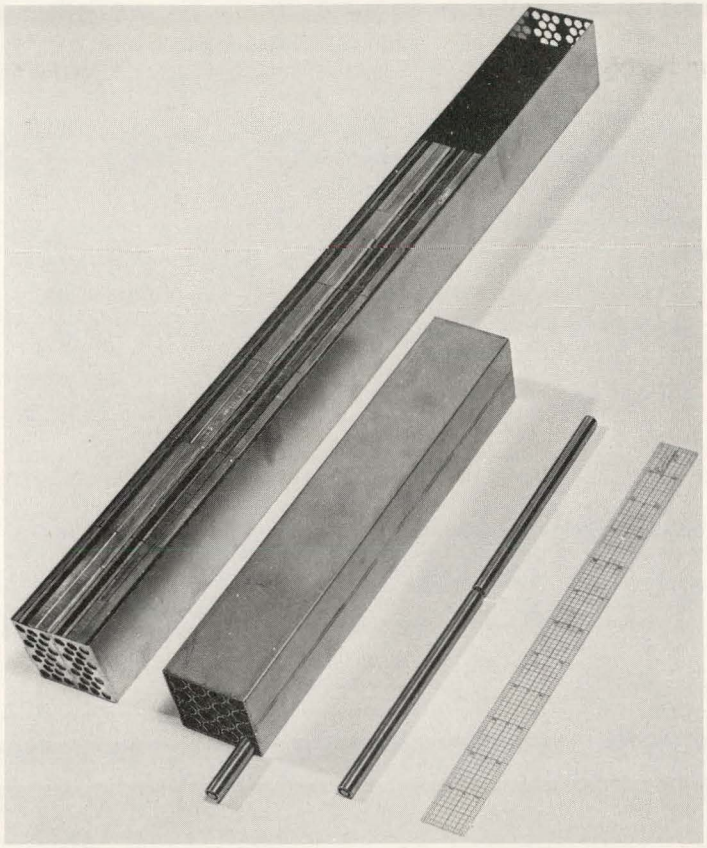

Figure 2. ZPR Loaded Drawers and Calandria.

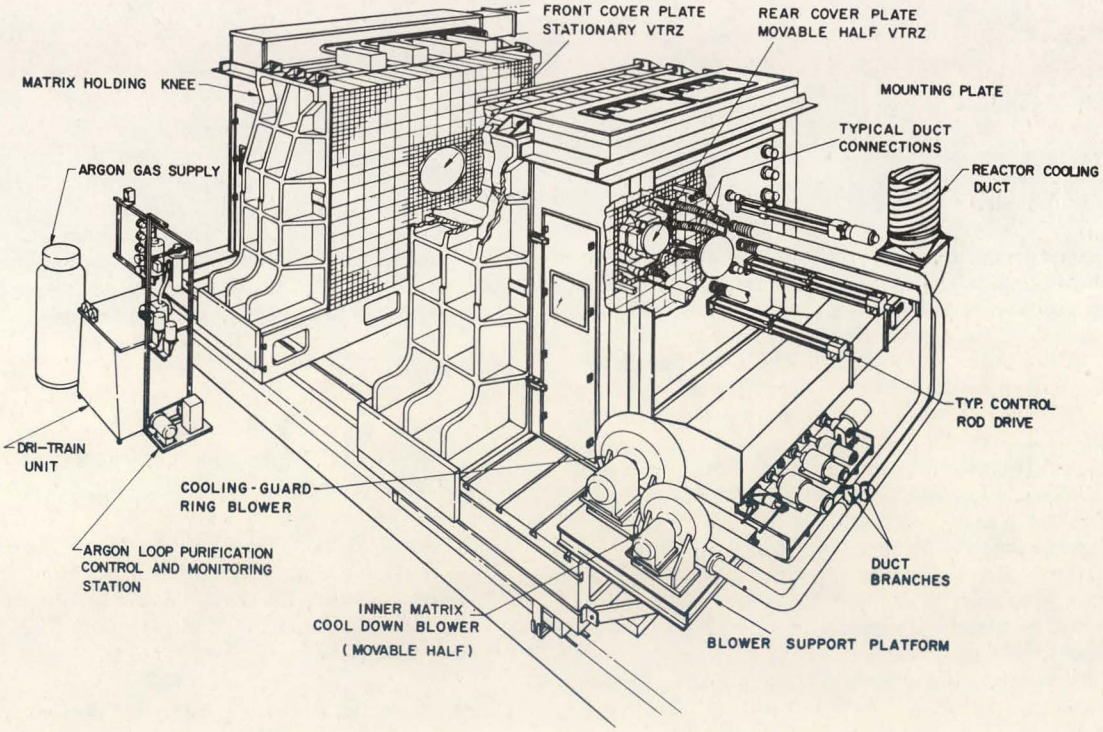

Figure 3. Schematic of ZPR-6 with VTRZ.

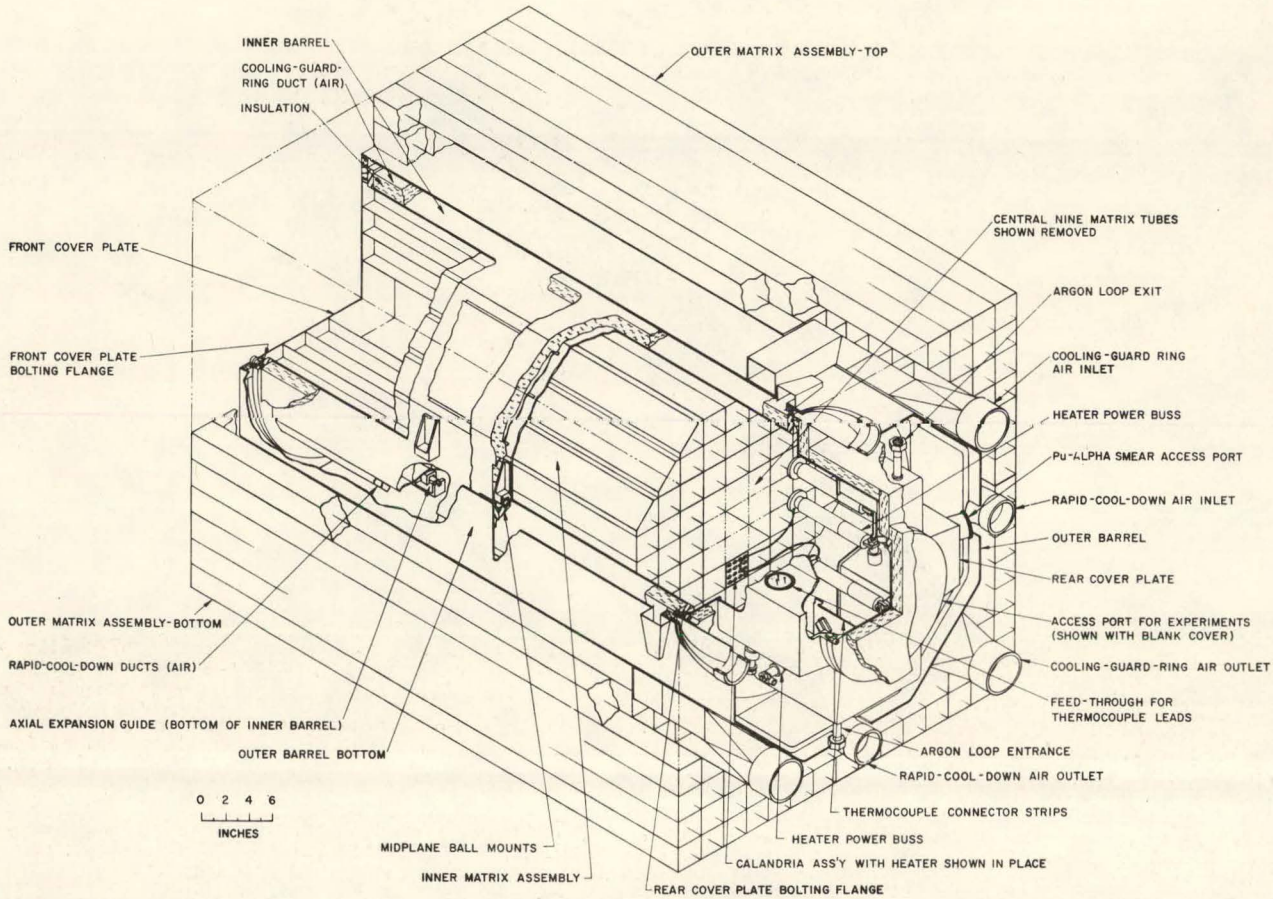

Figure 4. Schematic of VTRZ.

worths, reaction rates and ratios, reactivity coefficient, neutron and gamma spectra, gamma heating rates, and kinetics parameters, are measured in the critical experiments.

Auxiliary Equipment.

- Sufficient quantities of plutonium, and enriched uranium coolant and structural materials to construct metal, oxide or carbide cores in excess of 6000 liters are available. Fuel is available in plate or rod form and in various plutonium isotopic compositions. Uranium composition can be varied by changing the ratio of enriched and normal or depleted uranium plates.
- Remote-positioning equipment for measuring material worths or reaction rates as a function of core position both axially and radially is available.
- A mechanism for oscillating heated uranium and plutonium samples to permit investigation of the Doppler coefficients of these large cores is available.
- A variable speed, sinusoidal oscillator is used to measure kinetics parameters.
- An automated, high-volume, foil-counting facility, employing primarily high-resolution gamma-ray spectroscopy, where activated foils are counted to determine fission and capture distributions throughout the reactor, is available.
- Fission rates are also measured using fission counters and by fission track recorder techniques.
- Gamma heating measurements are made using thermoluminescent dosimeters (TLD's).
- An SEL-840-MP computer provides the capability for immediate on-line data acquisition and analysis.
- Programs exist for reactivity measurements utilizing noise analysis and reactor kinetics to provide data on the reactivity of a static system, control rod worth measurements, and small sample worths.

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1. Kato, W. Y., et al., "Final Safety Analysis Report on the Use of Plutonium in ZPR-6 and -9," Argonne National Laboratory, ANL-7442, February 1970.

2. Lewis, R. A., Dance, K. D., Meyer, J. F., and Bohn, E. M., "Variable Temperature Rodded Zone Project, Final Safety Analysis Report," Argonne National Laboratory, ANL-7638, January 1972. (Published March 1974.)

3. Lewis, R. A., et al., "Variable Temperature Roddec Zone Project, Design Summary Report," Argonne National Laboratory, ANL-7639, October 1972. (Errata November 26, 1973.)

ZERO POWER REACTOR-9 (ZPR-9)

Location ANL-E, Argonne, Ill.

Status Operational

Principal Uses. The facility is used to obtain fast reactor physics integral parameters from benchmark, engineering mockup, or special purpose experiments.

Facility Description. ZPR-9 is a split-table-design critical facility, consisting of matrix assemblies positioned on two large tables; one table is movable and the other is stationary. On each table are mounted approximately 2000 two by two in. square stainless steel matrix tubes which form an 8 X 8 X 4 ft matrix assembly. Figure 1 shows ZPR-9 with the tables separated for loading. Stainless steel drawers loaded with core materials (fuel, structural, and coolant) are loaded into the matrix assembly. The two tables each form one-half of an 8 X 8 X 8 ft reactor assembly. Criticality is attained by bringing the two tables together, inserting fuel containing control rods, and removing the boron absorber rods.

Test Capability. Fast reactors of the Demonstration Plant reactor size (cores up to ~3000-liter volume) may be built on ZPR-9. Integral parameters such as critical mass, coolant-void and Doppler coefficients, power distributions, control rod worths, reaction rates and ratios, reactivity coefficients, neutron and gamma spectra, gamma heating rates, and kinetics parameters, are measured in the critical experiments.

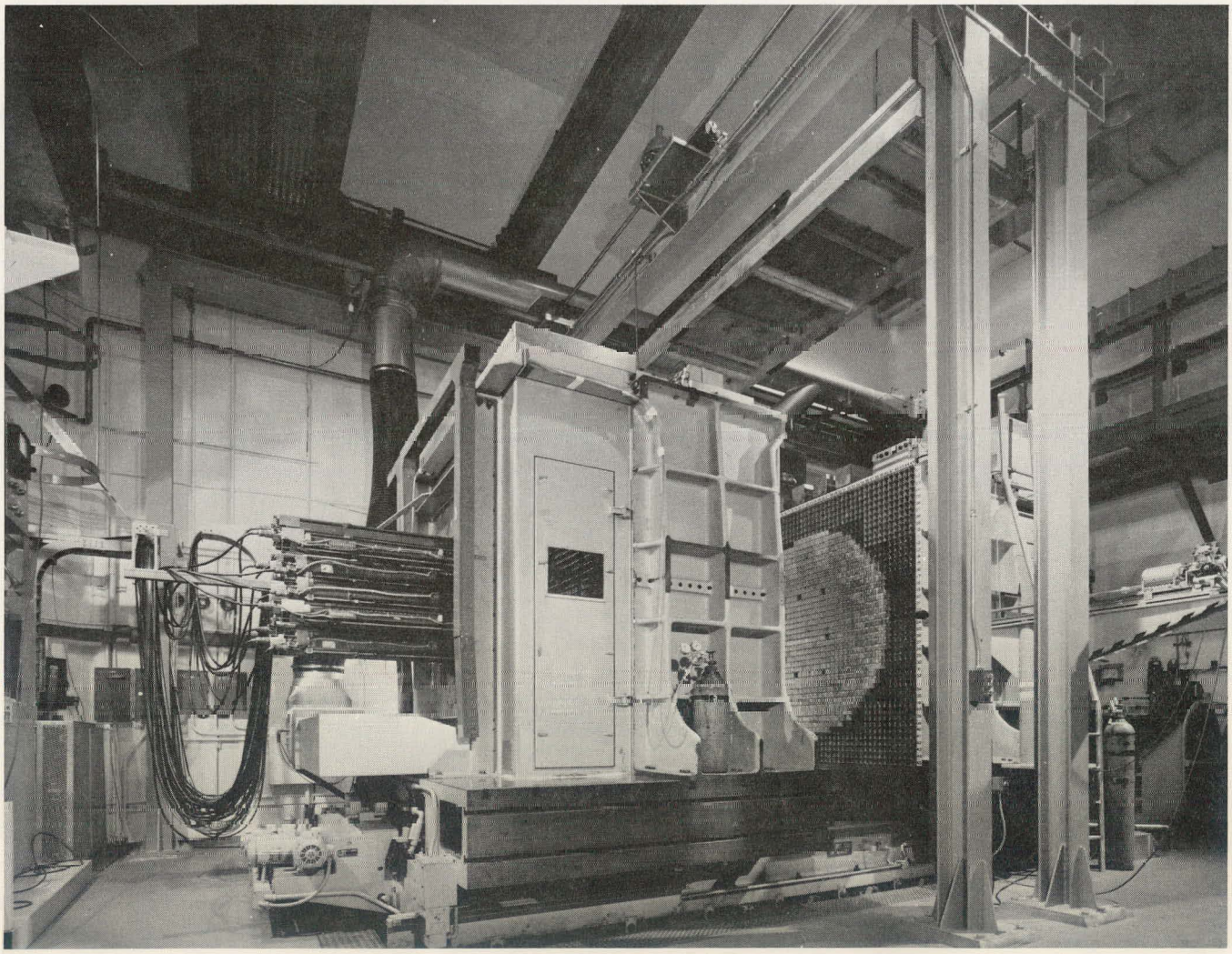


Figure 1. View of ZPR-9.

Auxiliary Equipment.

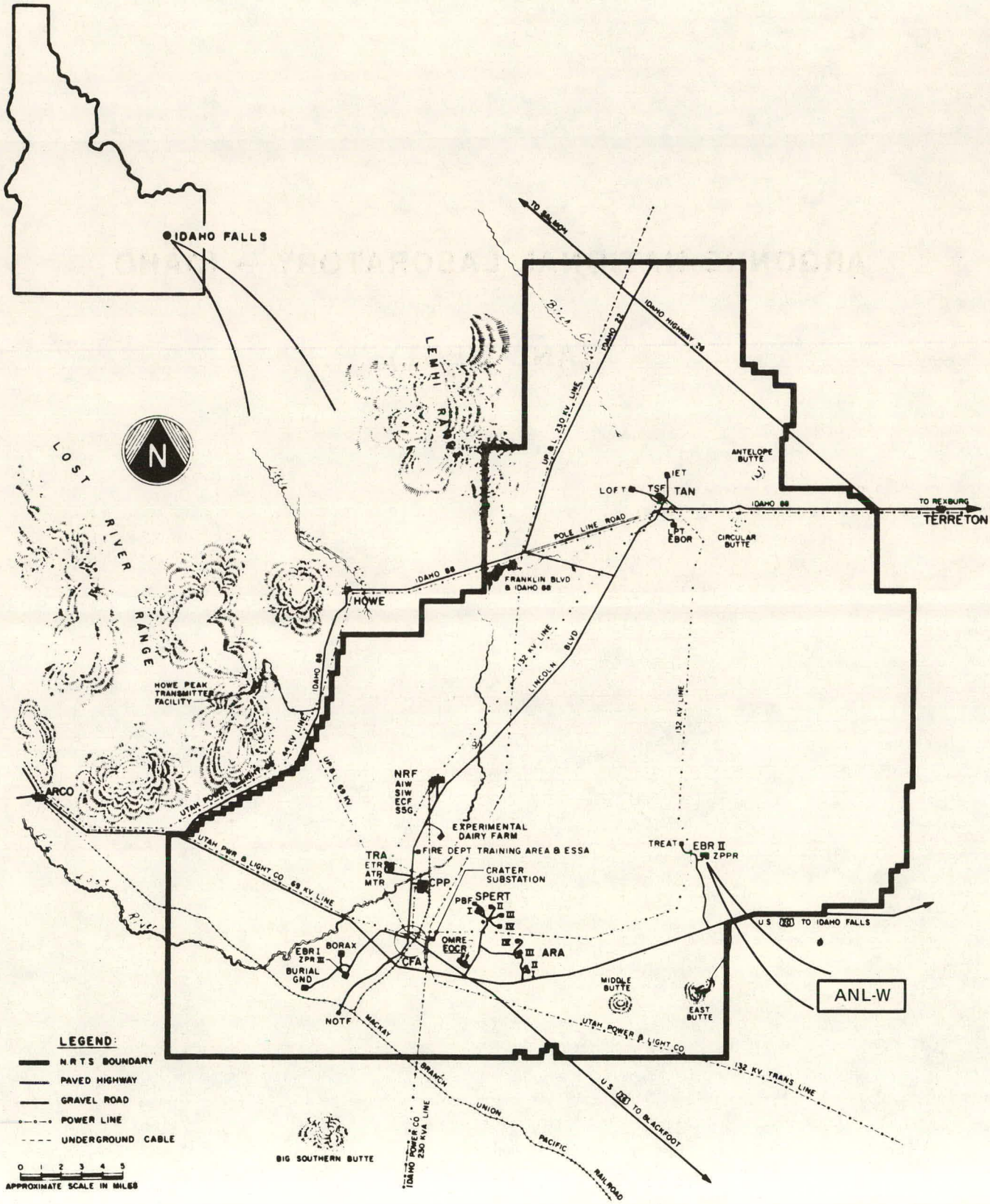
- Sufficient quantity of plutonium, coolant and structural materials to construct metal, oxide or carbide cores in excess of 6000 liters is available. Fuel is available in plate or rod form and in various plutonium isotopic compositions.
- Radial and axial remote-positioning equipment for measuring material worths or reaction rates as a function of core position is available.
- A mechanism for oscillating heated uranium and plutonium samples to permit investigation of the Doppler coefficients of these large cores is available.
- A variable speed sinusoidal oscillator is used to measure kinetics parameters.
- An automated, high-volume, foil-counting facility, employing primarily high-resolution gamma-ray spectroscopy, where activated foils are counted to determine fission and capture distributions throughout the reactor is available.
- Fission rates are also measured using fission counters and by fission track recorder techniques.
- Gamma heating measurements are made using thermoluminescent dosimeters (TLD's).
- An SEL-840-MP computer provides capability for immediate on-line data acquisition and analysis.

Bibliography.

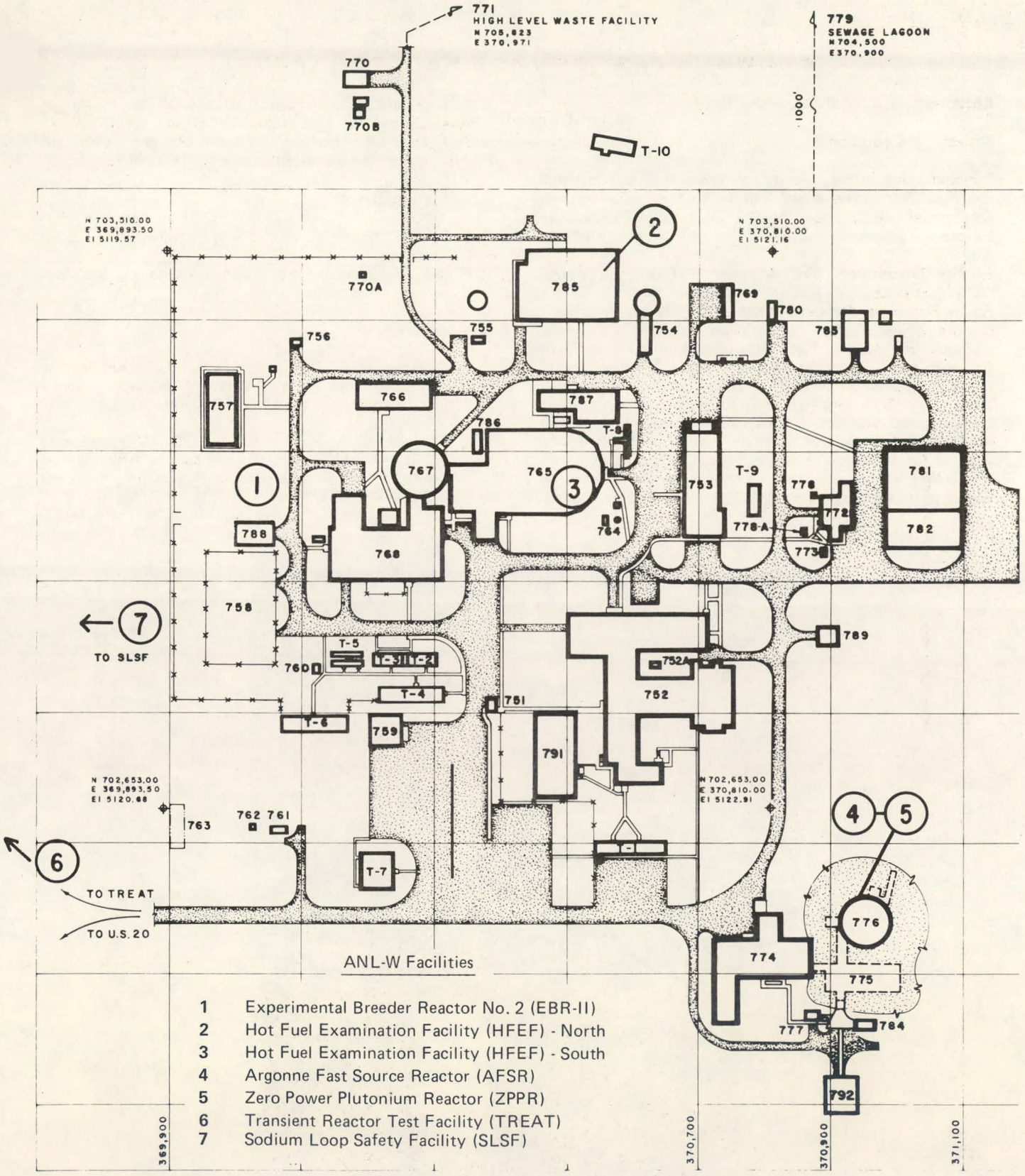
1. Kato, W. Y., et al., "Final Safety Analysis Report on the Use of Plutonium in ZPR-6 and -9," Argonne National Laboratory, ANL-7442, February 1970.

ARGONNE NATIONAL LABORATORY – IDAHO

(ANL-WEST)



Idaho National Engineering Laboratory (INEL)
Argonne National Laboratory - Idaho (ANL-W)
Site Location



ANL-W Facilities

1 Experimental Breeder Reactor No. 2 (EBR-II)
2 Hot Fuel Examination Facility (HFEF) - North
3 Hot Fuel Examination Facility (HFEF) - South
4 Argonne Fast Source Reactor (AFSR)
5 Zero Power Plutonium Reactor (ZPPR)
6 Transient Reactor Test Facility (TREAT)
7 Sodium Loop Safety Facility (SLSF)

Argonne National Laboratory - Idaho (ANL-W) Facilities Map

ARGONNE FAST SOURCE REACTOR (AFSR)

Location ANL-W, INEL, Idaho

Status Operational

Principal Uses. This facility provides fast and thermal neutrons for calibration purposes and experimental development in support of the Zero Power Plutonium Reactor.

Facility Description. The Argonne Fast Source Reactor (AFSR) is housed in a room adjacent to the ZPPR (Zero Power Plutonium Reactor) at the INEL. The reactor core is a right cylinder of 93% enriched uranium 4.51 in. in diameter and 4.68 in. high. The core is split with 55% of the core fixed in the blanket and 45% of the core on a movable ram. When the core is assembled, it is surrounded by an 8-in. thick depleted uranium blanket. The core and blanket assemblies are in turn surrounded by a nominally 4.5 ft thick high-density concrete biological shield. A top view of the facility is shown in Figure 2. The penetrations for access to the neutron fluences are also shown in Figure 2. Maximum power is 1 kW.

Test Capability. Fast neutron fluxes of the order of 5×10^{11} n/cm^2-sec are obtainable at the center of the core. Thermal neutron fluxes of the order of 2×10^{8} n/cm^2-sec are obtainable in the thermal column.

Auxiliary Equipment.

- Special graphite holders for foils and fission chambers.
- ND 180 analyzer and signal lines to an analyzer-computer laboratory.

Bibliography.

1. Brunson, G. S., "Design and Hazards Report for the Argonne Fast Source Reactor (AFSR)," Argonne National Laboratory, ANL-6024, June 1959.

2. Matlock, R. G., and Vosburgh, R. O., "Addendum to the Design and Hazards Report for the Argonne Fast Source Reactor (ANL-6024)," Argonne National Laboratory.

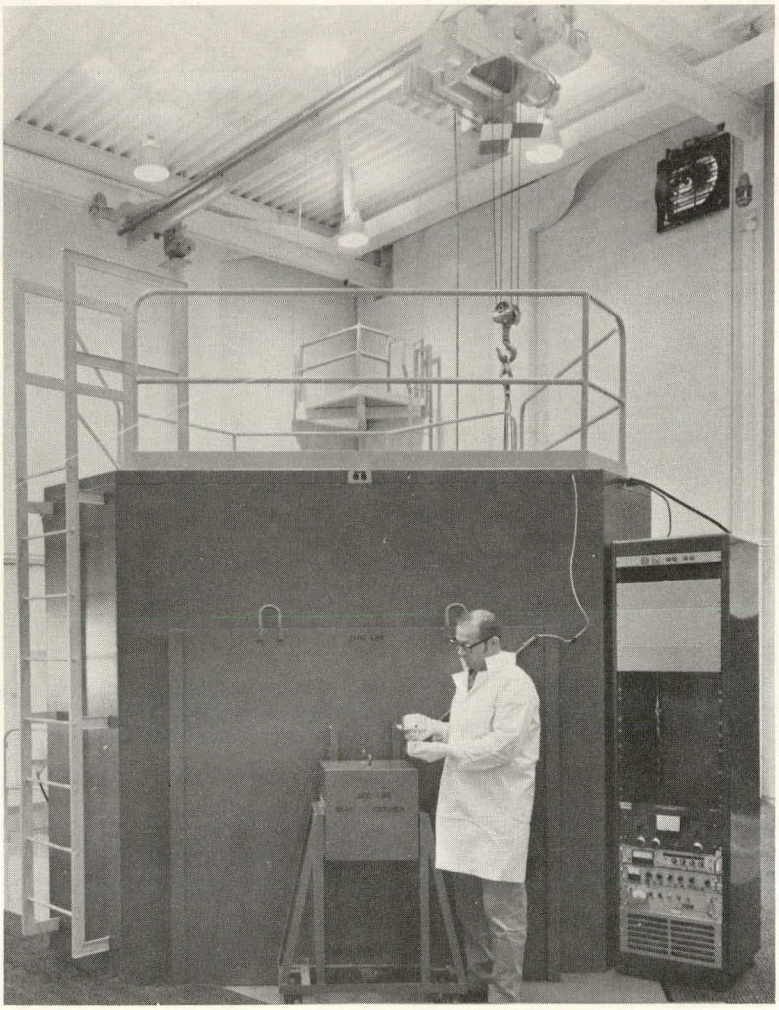


Figure 1. Argonne Fast Source Reactor (AFSR).

TABLE I. FACILITY PARAMETERS

CRITICAL MASS	21.25 KG
CORE MATERIAL	ENRICHED U
CORE SIZE	4.51 IN. DIA X 4.68 IN. HIGH
BLANKET MATERIAL	DEPLETED U
BLANKET THICKNESS	8 IN.
COOLING	AIR AT 78 CFM
MAX. POWER	1 KW
MAX. CORE TEMPERATURE	120°C
GLORY HOLE DIAMETER	0.514 IN.
GRAZING HOLE DIAMETER	1.078 IN.
BEAM HOLE DIAMETER	2.15 IN.
THERMAL COLUMN	4 FT H X 4 FT W X 4 FT 9 IN. L
MAX. GLORY HOLE FLUX	5×10^{11} N/CM^2-SEC
MAX. GRAZING HOLE FLUX	1.5×10^{11} N/CM^2-SEC
MAX. BEAM HOLE FLUX	8×10^{10} N/CM^2-SEC
MAX. GRAPHITE HOLE #1 FLUX	1.5×10^{9} N/CM^2-SEC
MAX. GRAPHITE HOLE #2 FLUX	1.9×10^{8} N/CM^2-SEC

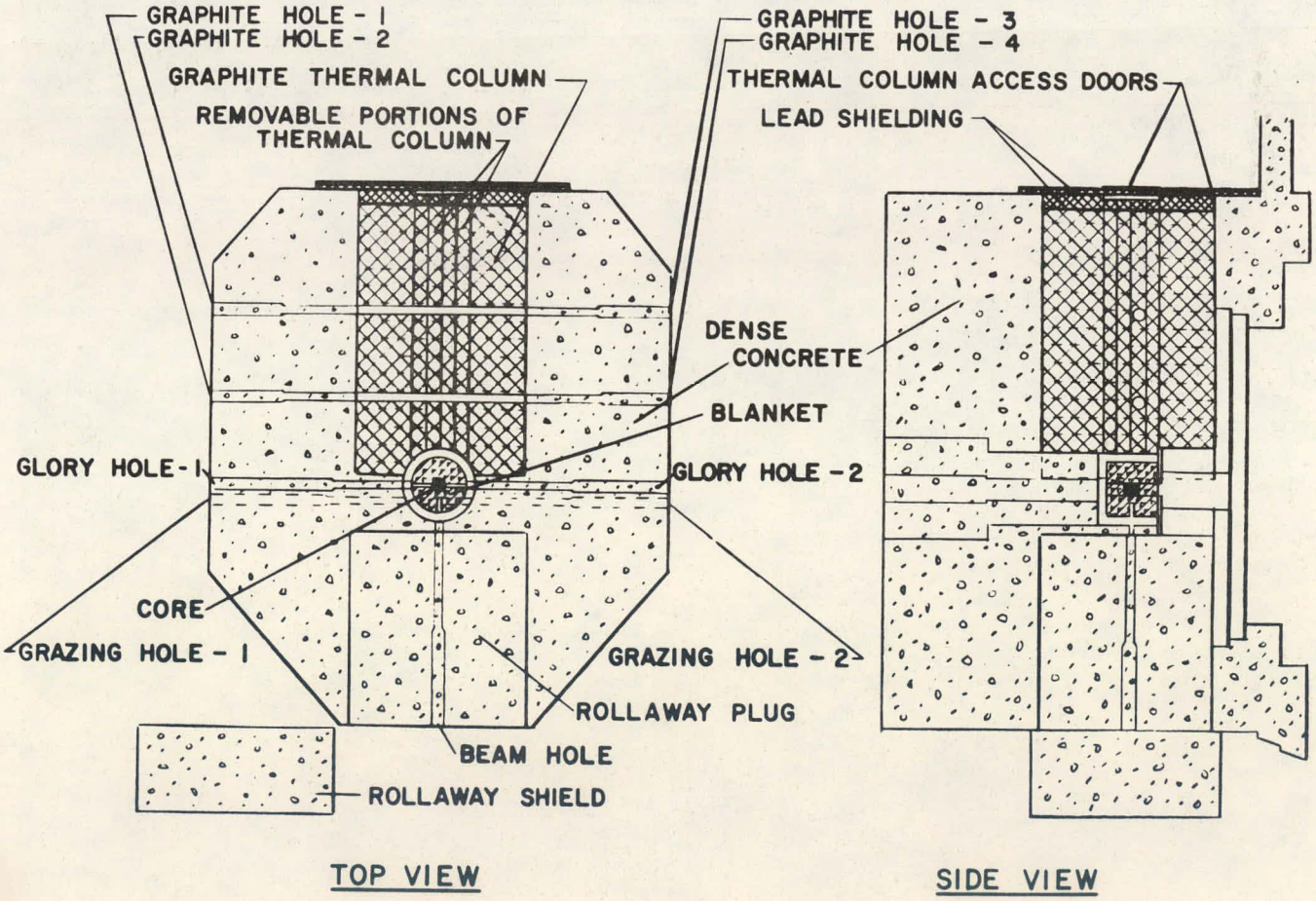


Figure 2. Experimental Facilities AFSR.

EXPERIMENTAL BREEDER REACTOR-II (EBR-II)

Location ANL-W, INEL, Idaho

Status Operational

Principal Uses. Originally built to:

- Demonstrate the feasibility of fast reactors for central station power plant applications.
- Prove that a breeding ratio greater than unity could be obtained in a power-producing reactor.
- Provide on-site capability for remote reprocessing and refabrication of fuel for return to the reactor.

In 1965 the purpose of EBR-II was redirected to provide a fast flux fuels and materials irradiation testing facility for the LMFBR program and, until the Fast Flux Test Facility

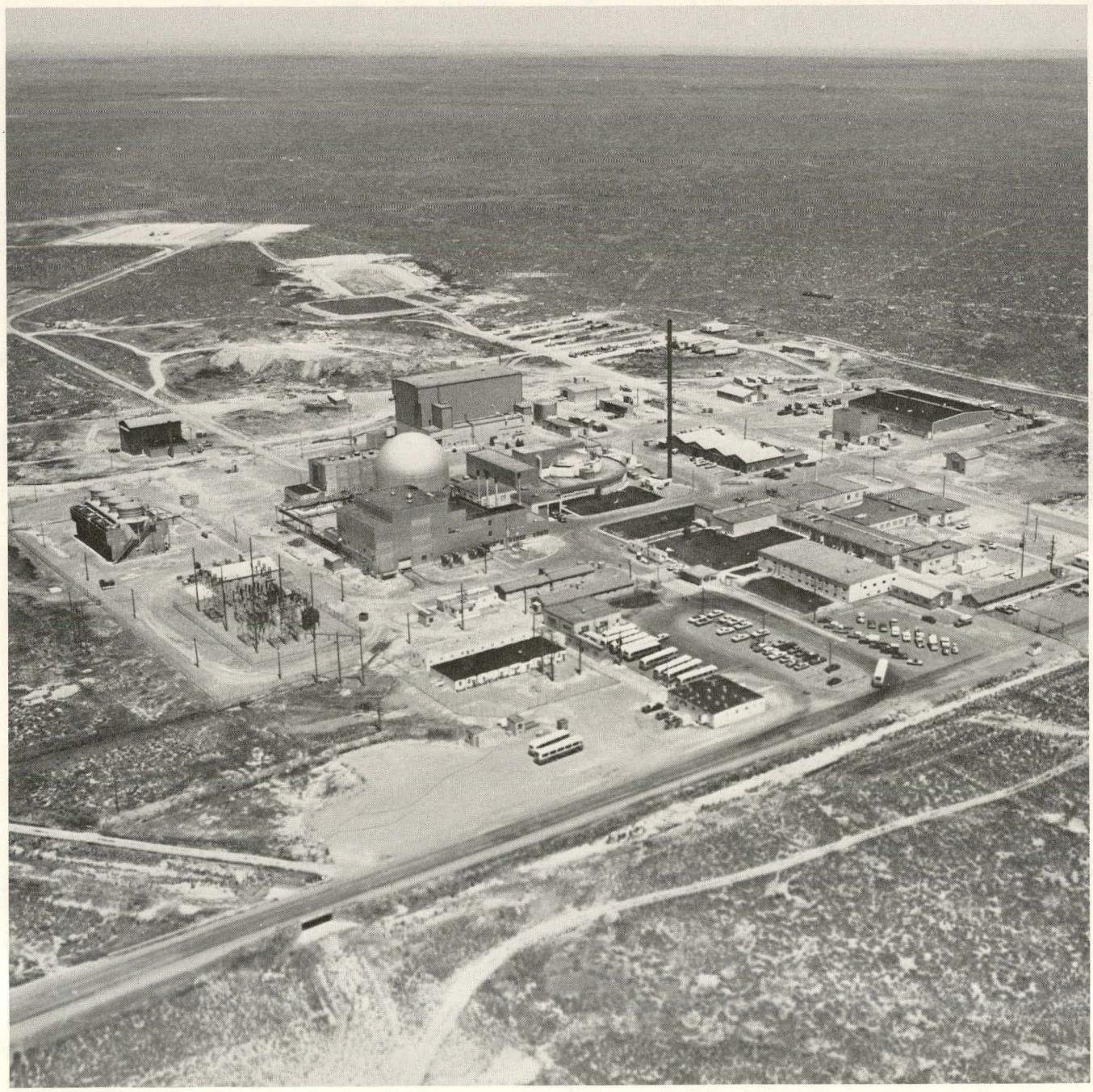


Figure 1. EBR-II Reactor Plant Site.

(FFTF) becomes operational, EBR-II will continue in that capacity. Following operation of FFTF, EBR-II will be used as:

- Irradiation facility for fuels and materials.
- Test facility for in-reactor instruments.
- System test bed for sodium components and systems.
- Operating facility to develop experience with LMFBR's.

Facility Description. EBR-II is a sodium-cooled pool-type reactor with a nominal reactor output of 62.5 MWt. It is a complete electrical power plant generating up to 20 MWe. EBR-II contains complete instrumentation and control systems which measure and control the basic parameters of fast power reactor operation. Operating variables are measured and controlled under conditions and in an environment which are similar to that which will exist in a large commercial station. Although EBR-II continues to develop electric power, thus providing relevant experience as an experimental power station, its main function is as an irradiation facility. Pending completion of the FFTF, only EBR-II can provide a combined sodium, fast neutron flux and gamma environment.

Test Capability. EBR-II has seven rows of subassembly positions that can be used for experimental irradiations (the first six rows are designed for normal use with driver fuel

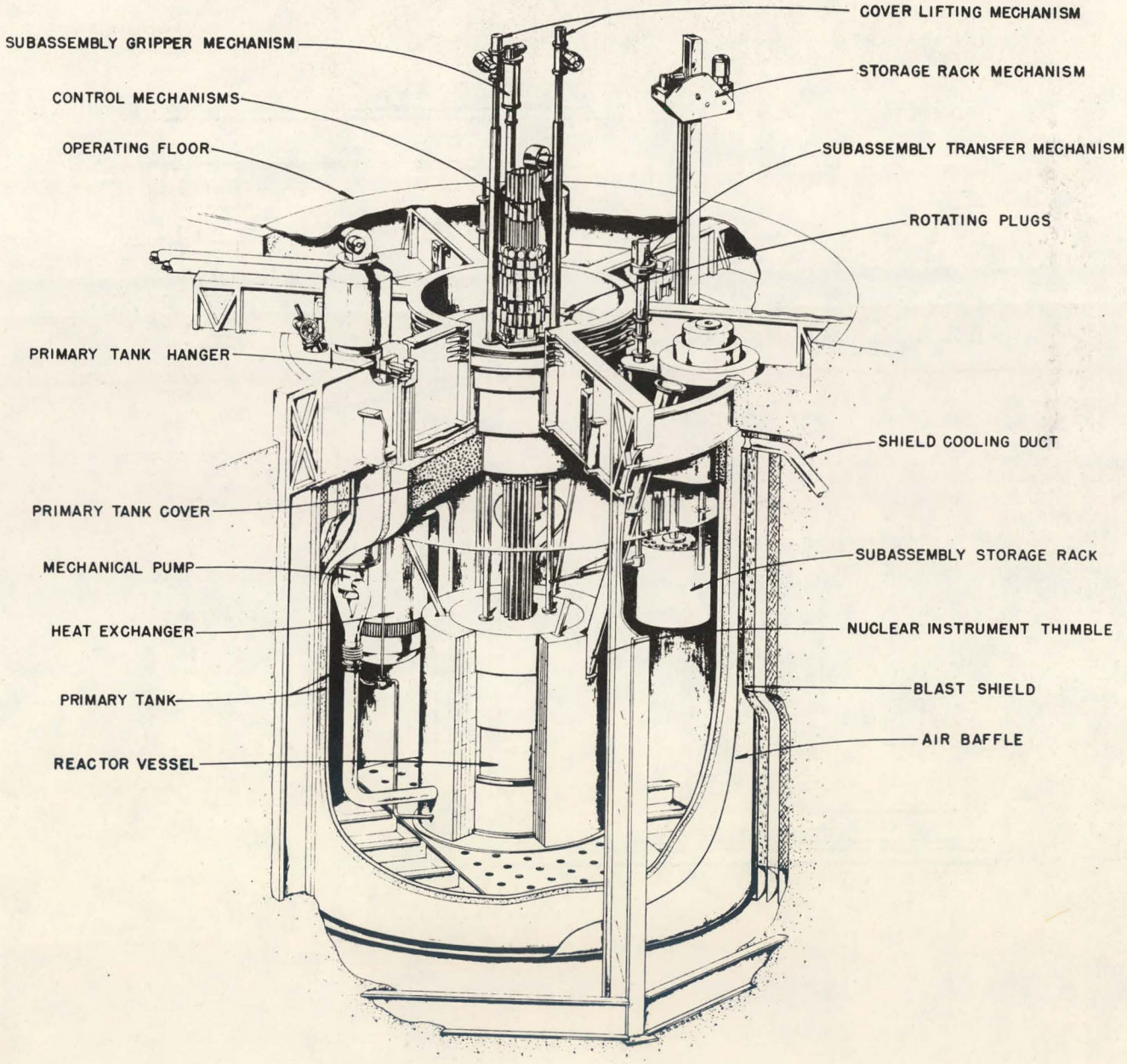


Figure 2. Cutaway View of the EBR-II Primary System.

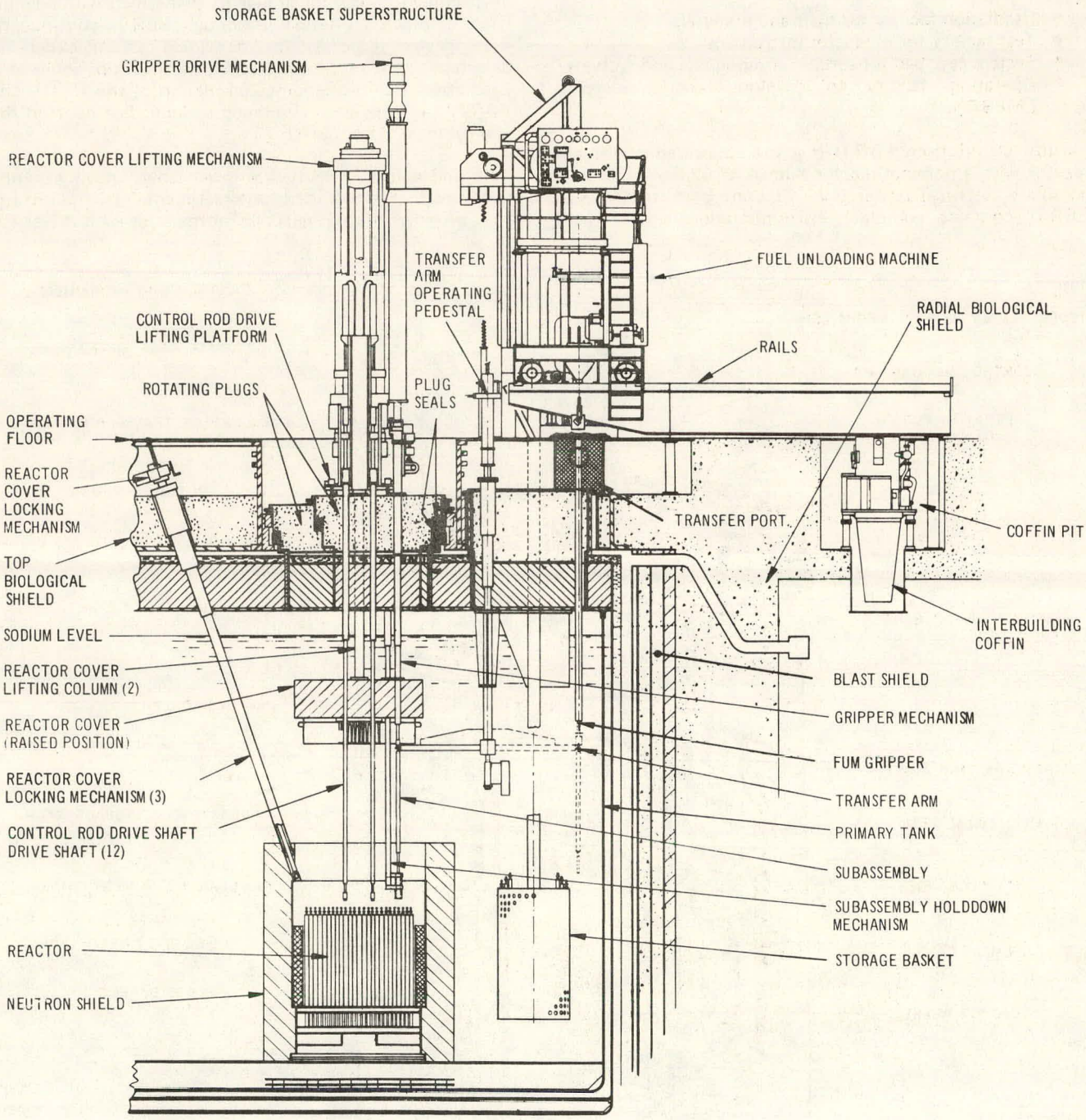


Figure 3. EBR-II Fuel-Handling System.

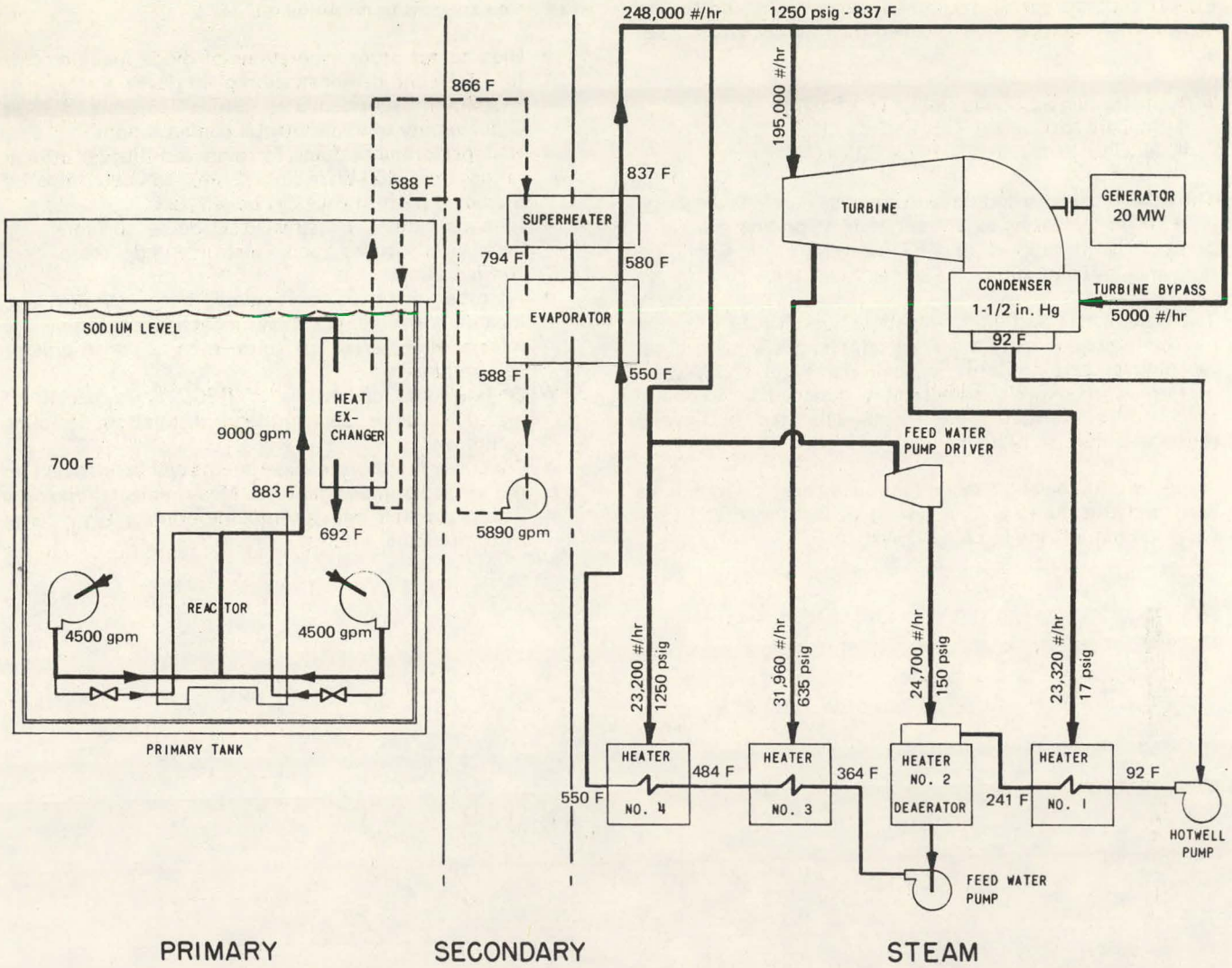


Figure 4. Simplified Flow Diagram of EBR-II.

and the seventh row is in the reflector). Of the 127 possible lattice positions, 113 are available and the remaining 14 positions are in the core and are occupied by control and safety rods. Various types of subassemblies have been designed or are under development to permit irradiations of materials in EBR-II. The subassembly may contain a bundle of capsules which contain the materials or experimental fuel elements to be irradiated, or a bundle of unencapsulated fuel elements.

Initially the principal non-oxide fuel irradiations were on uranium-plutonium alloy, with fewer experiments on uranium-plutonium carbide elements. Metallic fuel irradiations have been phased out and the number of carbide irradiations have increased. Highest burnup achieved is 10.6 at. % (100,000 MWD/Te) on vibrator-compacted fuel. Most current irradiations are on sodium-bonded pellets. Some sodium-bonded nitride elements have been irradiated to a maximum burnup of 4.6 at. % (43,000 MWD/Te).

The number of structural subassemblies in EBR-II has remained relatively constant for the past three years. There are several contributing reasons, including the availability of high-fluence stainless steel from EBR-II core components and the trend of the experimenters to perform more detailed examinations on a fewer number of specimens.

Emphasis is on the irradiation of the FFTF reference cladding material - Type 316 stainless steel with 20% cold work. Increased emphasis is also being placed on stressed in-reactor tests, since it is becoming increasingly evident that postirradiation tests are of limited usefulness in predicting in-reactor cladding behavior.

Irradiation of absorber materials for LMFBR control rod development started in 1968 and has continued primarily in support of control rod development for FFTF. Emphasis is on boron carbide, with some additional irradiations on tantalum, tantalum-rich alloys, and europium oxide. Planning is under way for a vented boron carbide experiment and for an instrumented boron carbide irradiation in INCOT.

EBR-II contains special facilities for irradiating and testing instrumented components and test instrument sensors such as:

- Instrumented Subassembly Test Facility (INSAT).
- In-Core Instrument Test Facility (INCOT).
- Nuclear Instrument Test Facilities (NITFs).

The EBR-II primary sodium and cover gas systems are being used to an increasing extent for tests of on-line analyzers, particularly in support of FFTF development. Corrosion tests are also being run in the primary coolant.

The Radioactive Sodium Chemistry Loop (RSCL) is testing oxygen-hydrogen meters and vanadium wire equilibration techniques, and In-Core Sodium Corrosion Tests were performed in a special subassembly designed for exposing specimens of various types of stainless steel in flowing primary sodium at 1250°F.

A special building and a second pumped cover gas loop have been constructed to enable testing of instrumentation for cover gas monitoring in LMFBR systems.

Tests are expected to continue on:

- High temperature irradiations of oxide fuel elements for FFTF and demonstration plant development.
- Irradiation subassemblies which accommodate a wider variety of experimental configurations.
- High-performance fuels (carbide and nitride) at heat ratings of 40 kW/ft, or more, to determine if predicted performance can be realized.
- In-reactor tests of stressed cladding specimens in pressurized tubes and instrumented creep test configuration.
- Absorber materials, particularly boron carbide, will include instrumented experiments to determine the effects of materials variables, such as stoichiometry and impurities.
- On-line tests of sensors in RSCL are expected to include sodium sparging and distillation sampling equipment.
- Cover-gas monitoring experiments will be installed on the second pumped gas loop. Measurements following fission product releases into the primary system will be of particular interest.

EBR-II IN-CORE INSTRUMENT TEST FACILITY (INCOT)

Location ANL–W, INEL, Idaho

Status Operational

Principal Uses.

- Provides the ability to test materials, instrument sensors, sensor bundles and other non-fissionable components in the EBR-II reactor core under typical LMFBR operating conditions.
- Enables the monitoring of their responses during irradiation at an available peak fast neutron flux of 2×10^{15}.
- Allows for their removal into a special handling container for transfer to a post-irradiation inspection station.

Facility Description. The In-Core Instrument Test Facility (Figure 1) consists of the following major components:

- Thimble assembly.
- Elevation system.
- Handling system.
- Terminal box assembly.
- Sensor assembly.
- Sensor data transmission system.

The thimble assembly (Figure 2), installed in grid position 5A3 (this position was formerly occupied by the No. 8 oscillator rod and drive), is the irradiation vehicle which encloses and supports the experiment. The thimble assembly extends from the core upwards through the reactor vessel cover, into the bulk sodium of the primary tank, through the small rotating plug and into the reactor building.

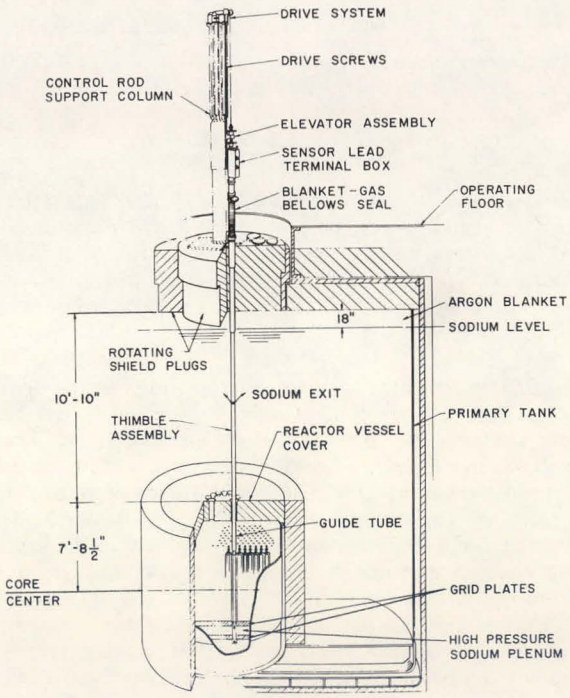


Figure 1. INCOT Facility.

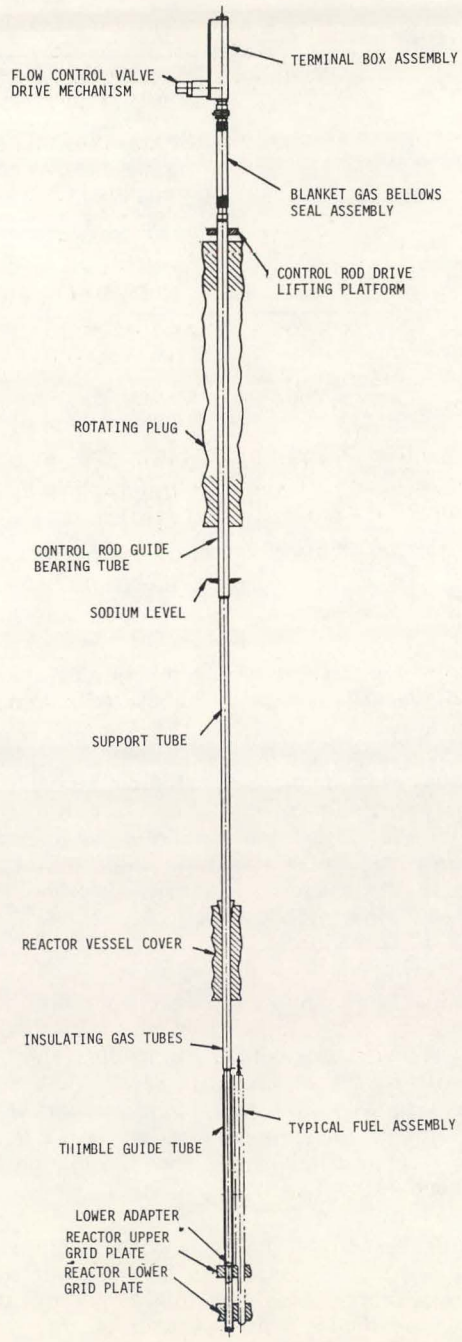


Figure 2. Thimble Assembly.

The elevation system raises the thimble assembly 80 in. from the core to clear the rotating shield plugs during fuel handling. It maintains a precise alignment over the centerline of the No. 8 opening, provides an exact elevation of the

TABLE I. IN-CORE INSTRUMENT TEST FACILITY TESTS

TEST NO.	EQUIPMENT OR MATERIAL TEST
YY01	PERFORMANCE AND RELIABILITY OF EDDY CURRENT FLOW SENSORS– TWO FLOW SENSORS WITH VOLTAGE AND TEMPERATURE READOUT
YY02	EVALUATION OF BORON CARBIDE ABSORBER MATERIALS AT VARIOUS TEMPERATURE AND FLUX LEVELS – TWENTY CAPSULES WITH PRESSURE AND TEMPERATURE READOUT
YY03	PERFORMANCE AND RELIABILITY OF SELF-POWERED IN-CORE NEUTRON AND FLUX SENSORS AND PERFORMANCE OF GAS-POWERED IN-CORE NEUTRON FLUX SENSORS
YY05	ACOUSTIC ENERGY MEASUREMENTS WITH PIEZOELECTRIC TRANSDUCERS
YY06	EVALUATE IRRADIATION EFFECTS ON BORON CARBIDE, EUROPIA, AND GAS VENTS FOR IMPROVED FBR CONTROL ELEMENTS
YY07, 9	EVALUATE TEMPERATURE, STRESS, AND FLUENCE DEPENDENCE OF IN-REACTOR CREEP BEHAVIOR ON FUEL CLADDING AND CORE STRUCTURAL MATERIAL
YY08	EVALUATE UNIAXIAL CREEP OF IN-REACTOR MATERIALS UTILIZING A LASER DATA TRANSMISSION SYSTEM

thimble in the reactor during operation, and prevents leakage of blanket cover gas to the reactor building.

The handling system removes and installs individual sensors, other experiments or the entire thimble assembly from the reactor. It provides special shielded coffins from the thimble or the sensors and transfer casks for moving the experiments to a post-irradiation inspection station. In addition to the shielding internally incorporated in the coffins, additional mobile biological shielding is provided for operator protection.

The terminal box assembly mounted on top of the thimble and sensor assemblies provides connections between the sensors and the sensor data transmission system, contains components of the flow-control valve drive mechanism, forms a secondary closure for the radioactive primary sodium, contains the inert gas at or above the sensors, and serves as a connecting link between the thimble assembly and the handling container.

The sensor assembly is mounted in the thimble assembly and positions the test sensors, or other experiments, in the thimble assembly. Based on the requirements of the experiment, different sensor assemblies are used. Three types of sensor assemblies are available. In model 1 (Figure 3a and b) the experiments are contained in a gas environment within individual sensor thimbles; in model 2 (Figure 4a and b) the experiments are contained in one gas-filled container; in model 3 (Figure 5a and b) the experiments are exposed to the primary sodium coolant through an open-ended thimble. One creep capsule test will replace the thimble assembly for a very long-term test.

The sensor data transmission system carries sensor signals from the terminal box assembly to data logging and readout equipment. It consists of cables routed from connectors in the terminal box assembly, under the operating floor deck plates through conduits and cable trays to the test instrument room.

Test Capability. Instruments and capsules can be tested in INCOT under the following environments:

	Environment*	
	Argon	Sodium
Min. Sensor Temp.	750°F	700°F
Max. Sensor Temp.	1400°F	1200°F
Max. Environment Temp.	1200°F	1200°F

*Test sensor or capsule exterior.

Based on the sensor assembly model, the sensor environment is either argon, helium, or sodium. Reactor coolant sodium enters the thimble at its base, flows through and around the sensor, and discharges from the thimble above the reactor vessel cover to rejoin the bulk sodium in the primary tank.

Sensor temperatures are controlled during reactor operation by a coolant flow control valve which varies the coolant flow entering the thimble and sensor assemblies. The flow control valve lower orifice plate is positioned in the thimble assembly lower adapter. The upper orifice plate of the valve is part of the sensor assembly. Flow is controlled by rotating the sensor assembly within the thimble assembly. The drive to accomplish rotation is part of the terminal box assembly. Within the thimble assembly, insulating gas tubes provide a thermal barrier between adjacent subassemblies and the sensor assembly to permit adjustment of the coolant temperature of the experiment by the flow control valve. The insulating gas tubes permit sodium temperature to increase to a maximum of 1200°F through gamma heating. These tubes extend into the reactor vessel cover

where they become thermal barriers to reduce stresses that would result from large radial thermal gradients through the tubes. The maximum sodium coolant temperature of 1200°F within the experiment area is cooled to within 100°F of the bulk sodium temperature when the sodium exits to the primary tank bulk sodium.

Auxiliary Equipment.

- A heater facility for long EBR-II components, installed in one of the existing storage holes in the reactor building, provides for preheating and post-irradiation heating of INCOT experiments and for borescope viewing of an experiment.
- The irradiated experiments may be handled and examined in the Hot Fuel Examination Facility at the EBR-II site.

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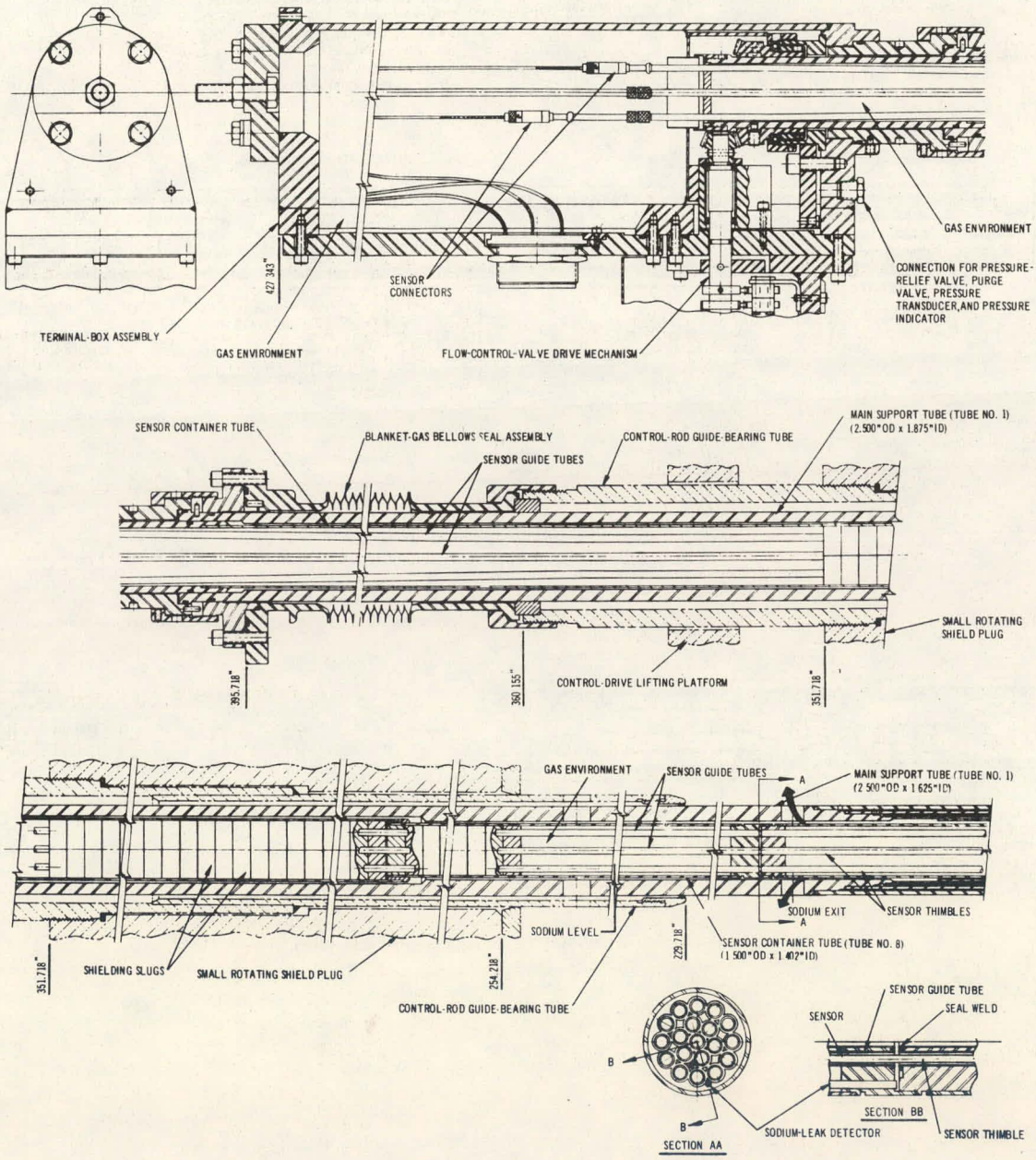

Figure 3a. Thimble Assembly Containing Model 1 Sensor Assembly (Upper Half).

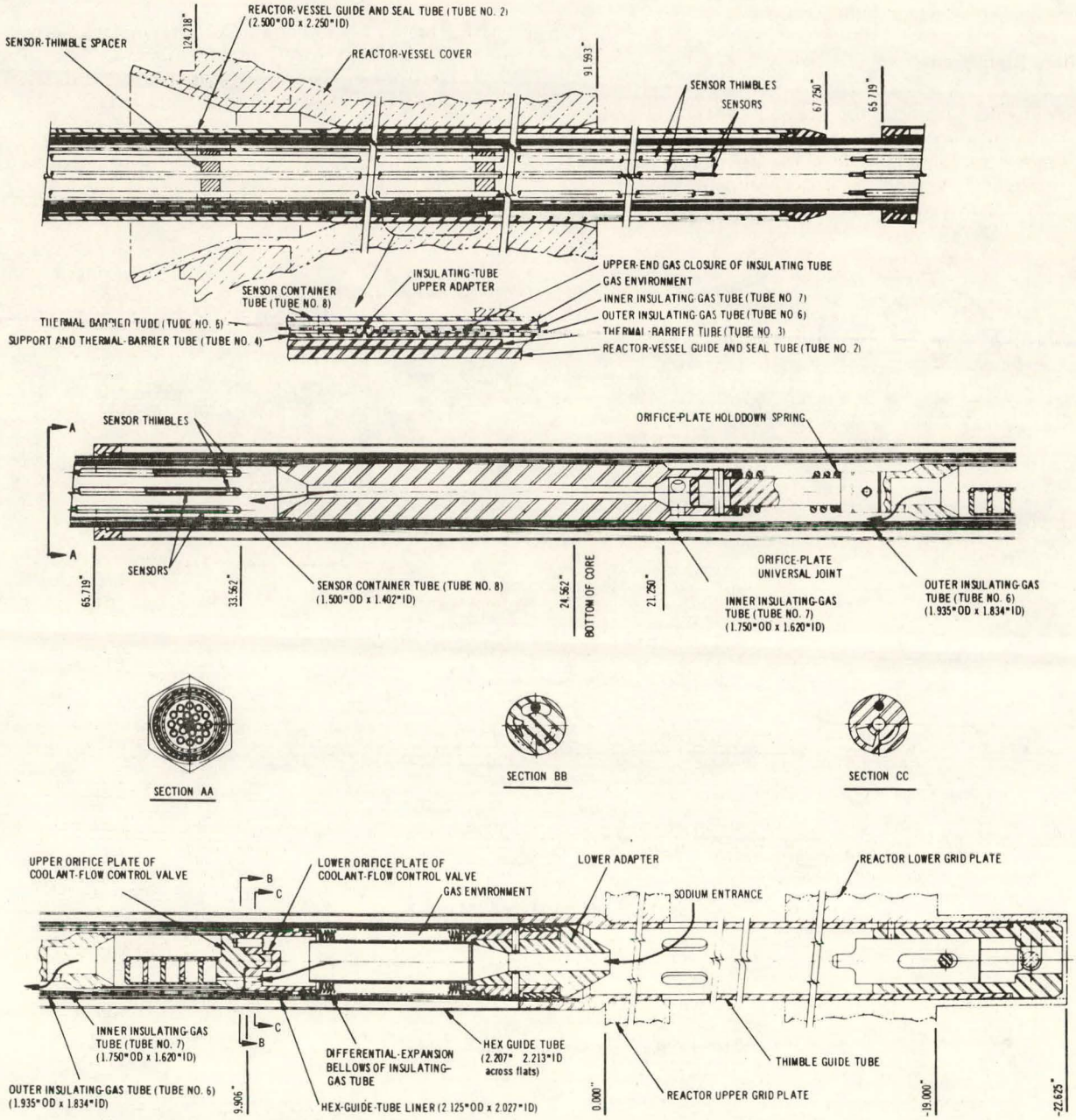

Figure 3b. Thimble Assembly Containing Model 1 Sensor Assembly (Lower Half).

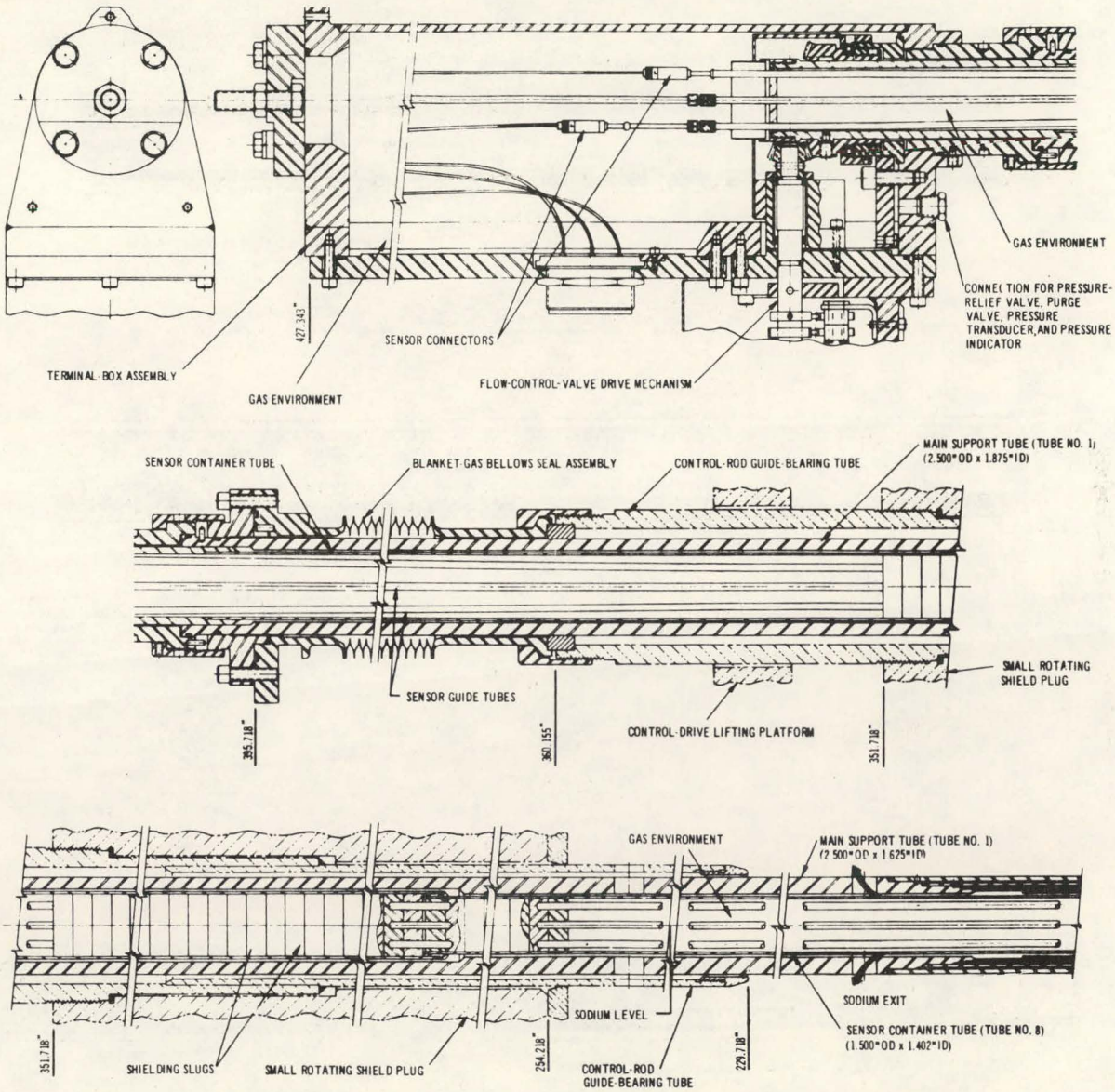

Figure 4a. Thimble Assembly Containing Model 2 Sensor Assembly (Upper Half).

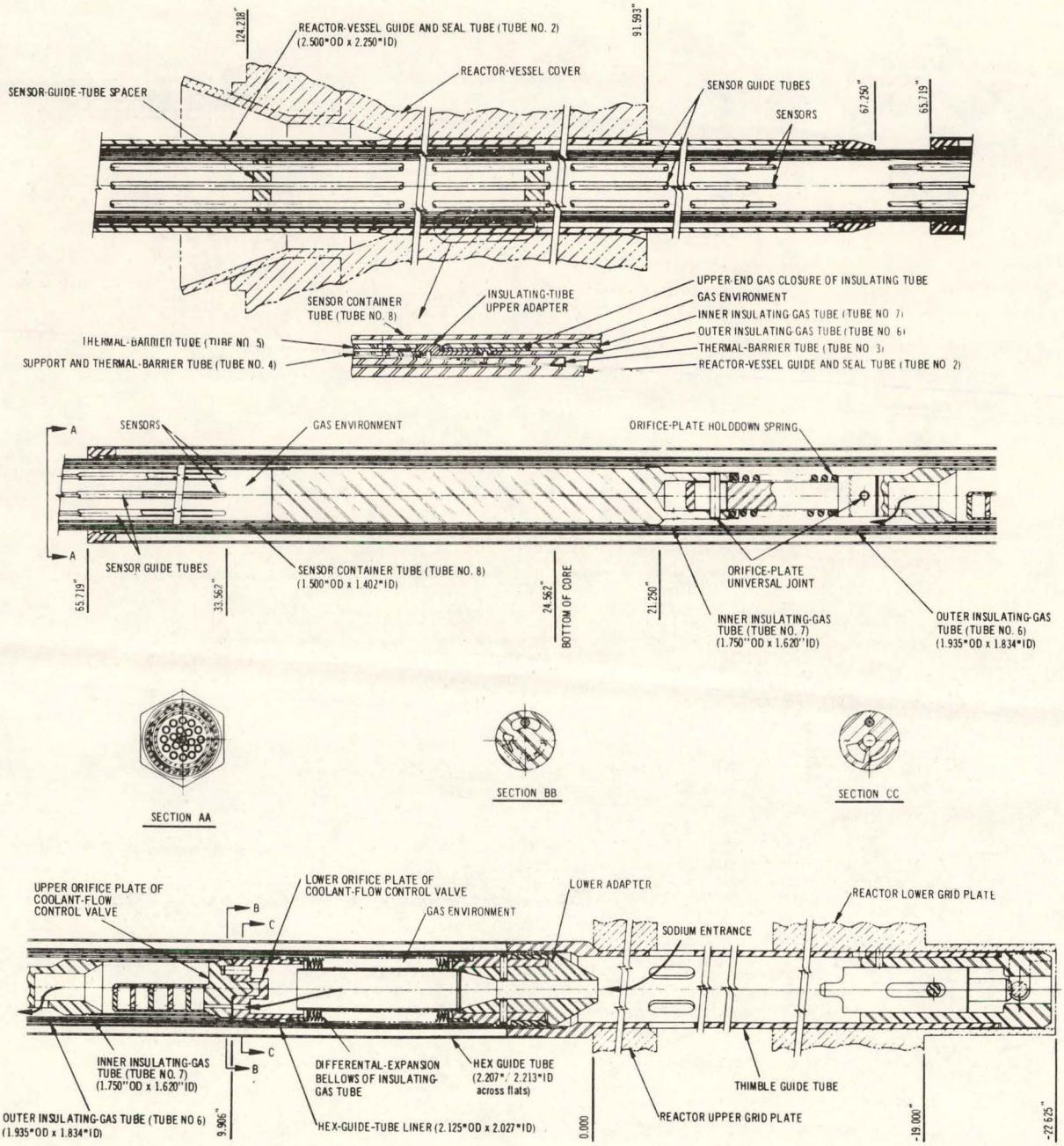

Figure 4b. Thimble Assembly Containing Model 2 Sensor Assembly (Lower Half).

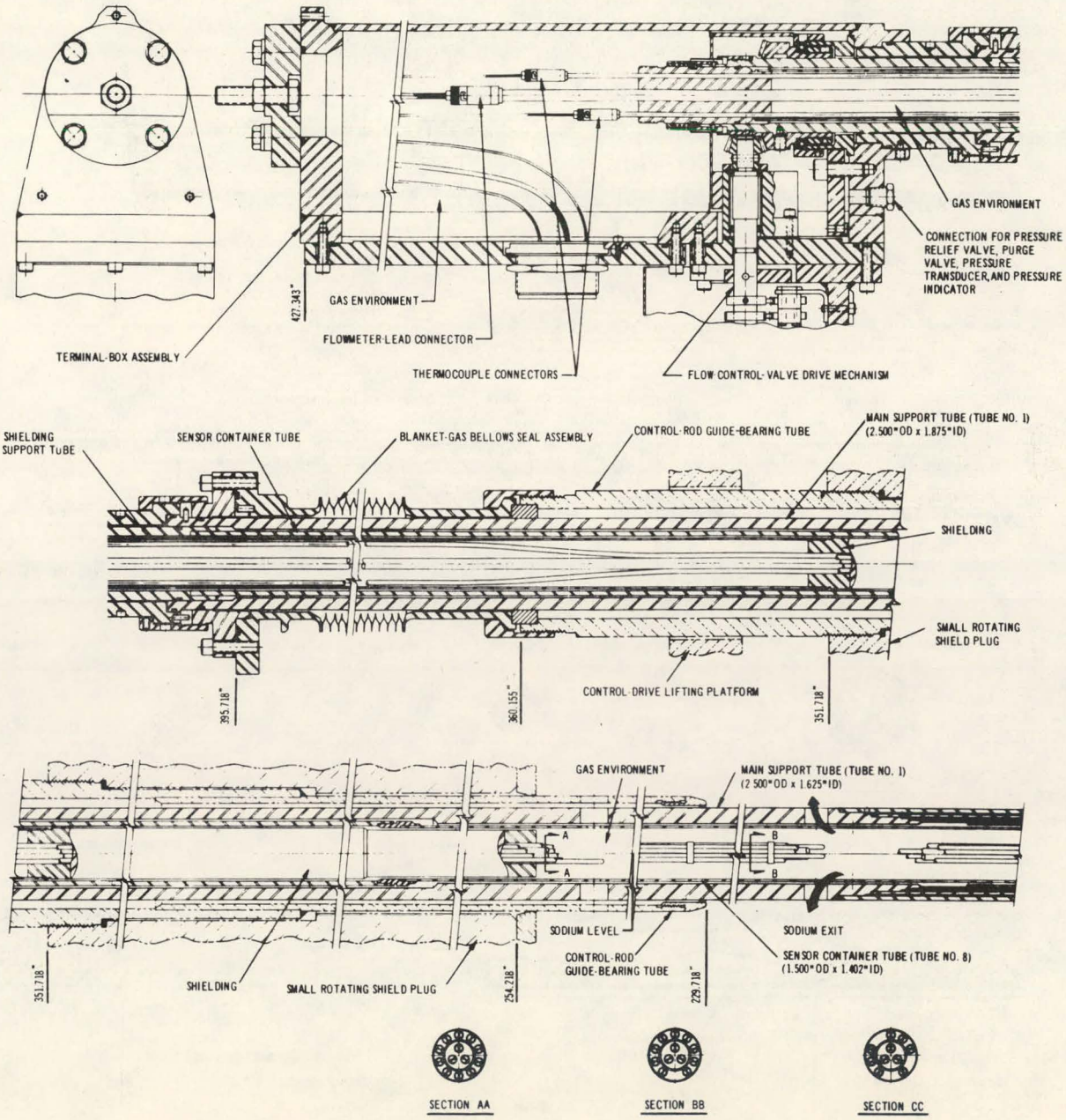

Figure 5a. Thimble Assembly Containing Model 3 Sensor Assembly (Upper Half).

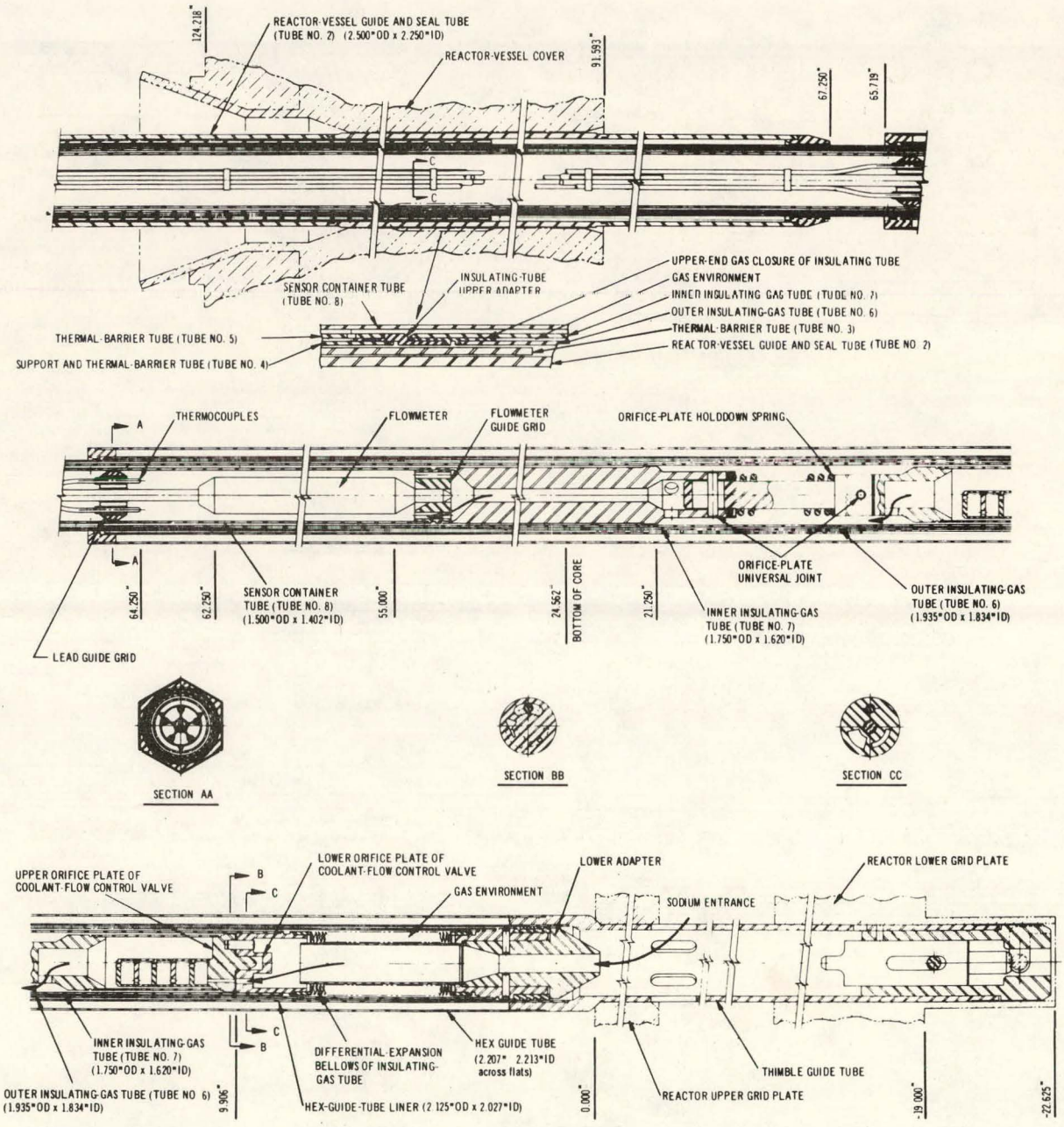

Figure 5b. Thimble Assembly Containing Model 3 Sensor Assembly (Lower Half).

EBR-II INSTRUMENTED SUBASSEMBLY TEST FACILITY (INSAT)

Location ANL-W, INEL, Idaho

Status Available

Principal Uses. INSAT enables an experimenter to directly monitor experimental irradiations of fuel elements and material specimens in the EBR-II core under typical LMFBR operating conditions in a sodium environment and a peak fast neutron flux of 2×10^{15}.

Facility Description. The system, as shown in Figure 1, consists of an instrumented subassembly, an extension tube with seals and connections for the instrument lead wires, a drive system which is electrically interlocked with the fuel handling control system, and recording and data logging equipment.

Outwardly, the configuration of the instrumented subassembly is identical to that of the standard control rod that it replaces. It is comprised of a lower adapter and an upper adapter welded to a hexagonal tube, which can accommodate bundles of fuel elements or capsules of experimental materials as well as a variety of instrument sensors (Figure 2).

The subassembly extension, a 29-ft long concentric tube with a gripper at its bottom end and an electrical terminal box at the top (Figure 3), provides a rigid connecting link between the instrumented subassembly and the drive system. The lower end of the tube fits over the subassembly upper adapter and has six holes for coolant discharge. A sodium-free dry-well is provided between the oval inner tube and the outer extension tube for routing the instrument leads to the terminal box (Figure 4). Sheathed leads, as well as tubes from the various sensors in the subassembly, pass through the sodium outlet region in the lower part of the extension tube and enter the dry-well through a bulkhead (Figure 4). The leads are sealed in the bulkhead by a high-temperature brazing process using nickel braze material as a filler.

The terminal box is flanged at the bottom for sealing to the mating flange of the extension tube and the top is bolted to the flange of the elevator assembly. An argon pressure of 14 psig is maintained in the terminal box, which is higher than the pressure of the gas and sodium surrounding the extension tube. Thus, a leak in the extension tube or dry-well would be indicated by a drop in terminal box pressure. A pressure-sensing switch is installed in the terminal box to detect a drop in pressure.

Test Capability. INSAT provides direct access of experimental sensor leads to irradiation experiments in the reactor core, as shown in Figure 1. Experiments irradiated in instrumented subassemblies include encapsulated fuel pins and structural specimens, unencapsulated fuel pins, and creep specimens stressed both biaxially and uniaxially. Sensors operated successfully include (a) thermocouples located in oxide fuel, on fuel pin cladding, in structural specimens, and in the coolant channels, (b) pressure transducers in the plenums of oxide fuel pins, (c) subassembly coolant flowmeters, (d) self-powered flux monitors, and (e) extensometers on uniaxial creep specimens with auxiliary capsule electrical heating. The maximum number of leads accommodated in a single experiment is 23.

TABLE I. REFERENCE DESIGN CRITERIA FOR INSTRUMENTED SUBASSEMBLY FACILITY (INSAT)

SUBASSEMBLY	
CONFIGURATION	HEX TUBE ~40 IN. LONG
MATERIAL	TYPE 340 SS
DIMENSIONS	
ACROSS FLATS	1.902/1.912 IN.
INSIDE FLATS	1.824/1.830 IN.
OVERALL LENGTH (LESS LEAD LENGTH)	93.6 IN.
COOLANT TEMPERATURE AT INLET	700°F
ALLOWABLE COOLANT TEMPERATURE AT OUTLET	848–1048°F
COOLANT FLOW	60 GPM
VERTICAL TRAVEL FOR FUEL HANDLING	96 IN.
SPEED OF TRAVEL FOR FUEL HANDLING	16.1 IN./MIN
EXTENSION TUBE	
MATERIAL	TYPE 304 SS
DIAMETER	2.5 IN. (NOMINAL)
WALL THICKNESS	0.120 IN. (NOMINAL)
OVERALL LENGTH (SUBASSEMBLY, EXTENSION TUBE, TERMINAL BOX)	36 FT (NOMINAL)
TOTAL WEIGHT	APPROX. 350 LB
SHEATHED LEAD SIZES	MIN. 0.0625 IN./MAX 0.1875 IN. (NOMINAL)

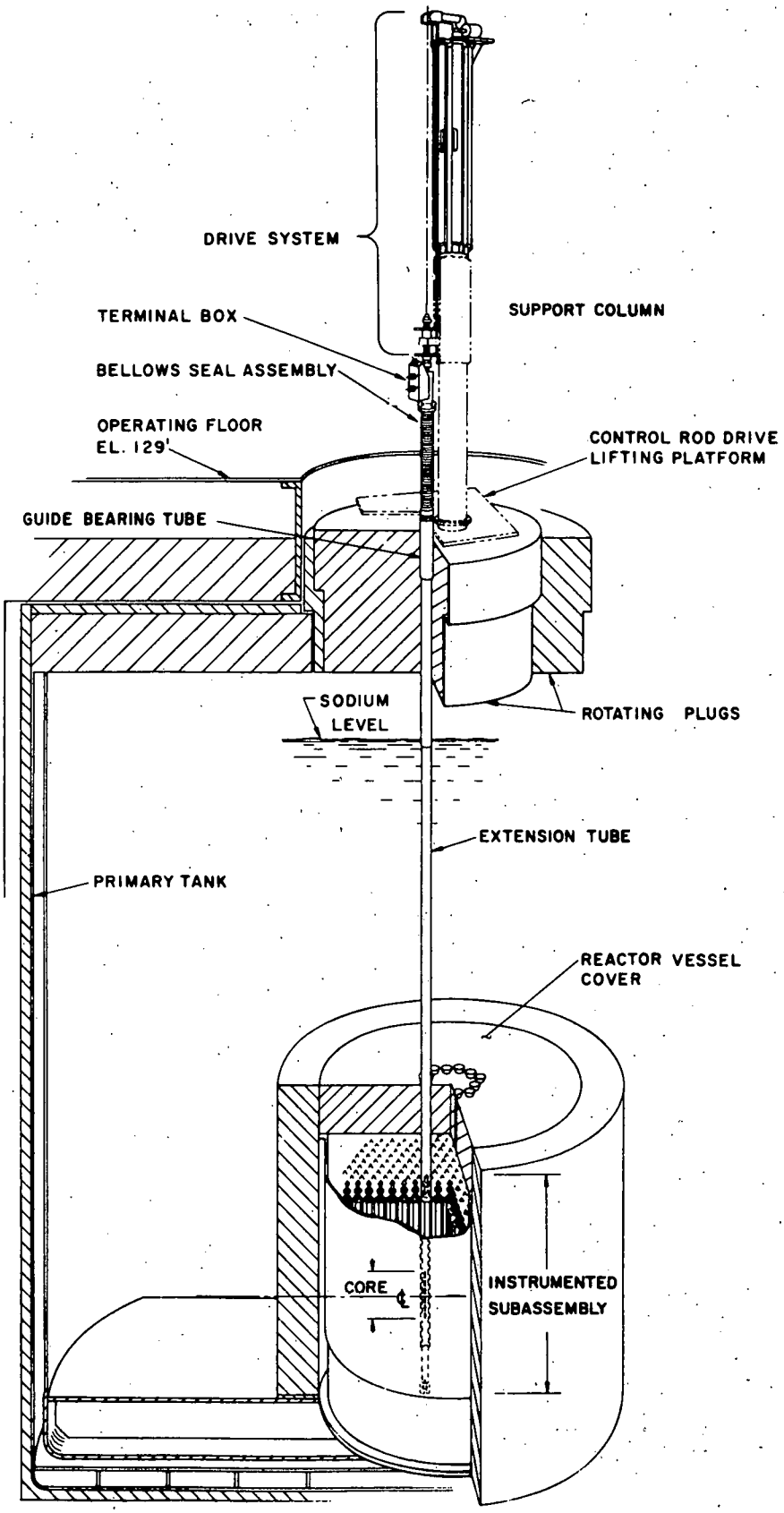

Figure 1. INSAT Installation.

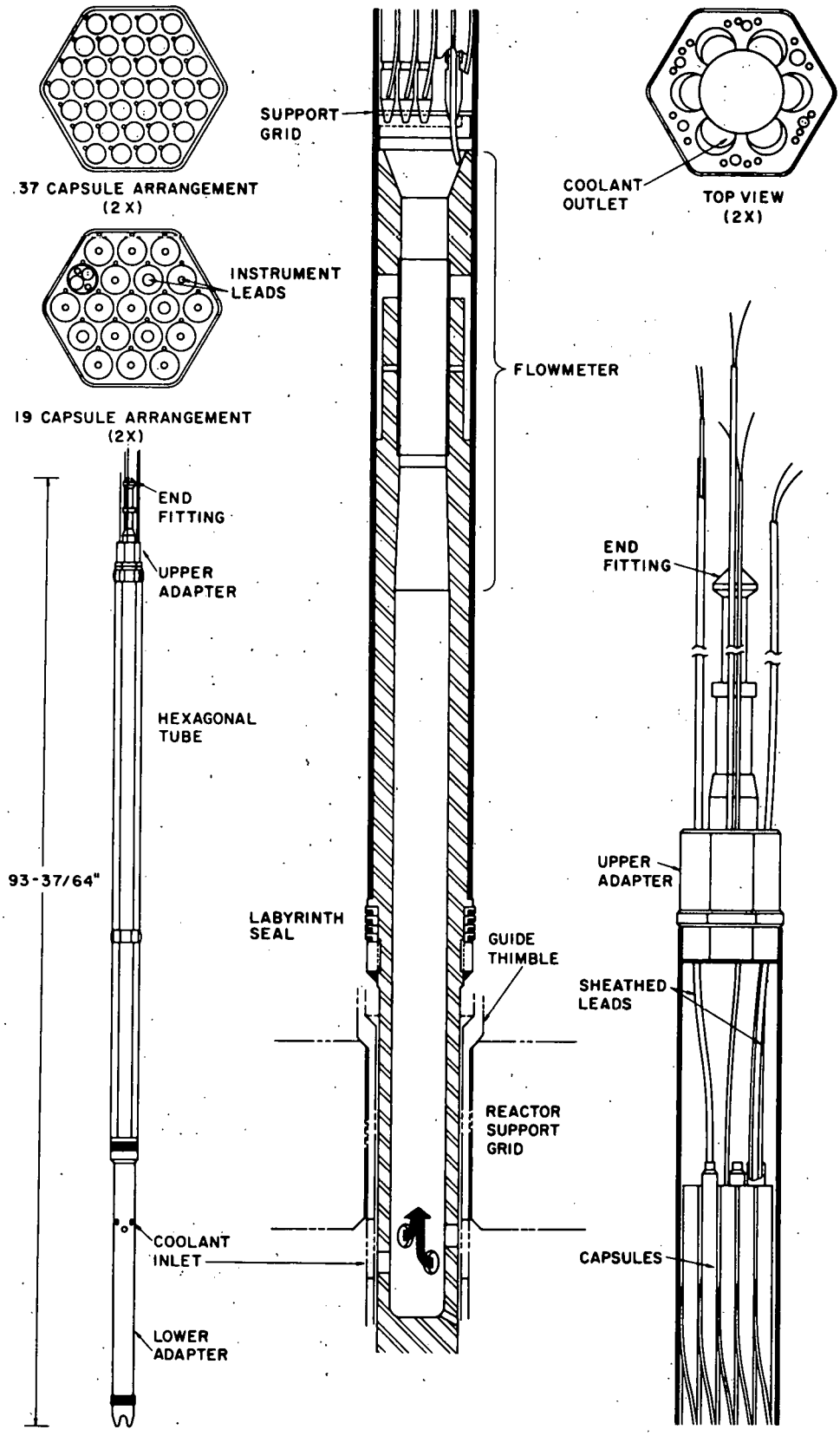

Figure 2. Typical INSAT Experiment.

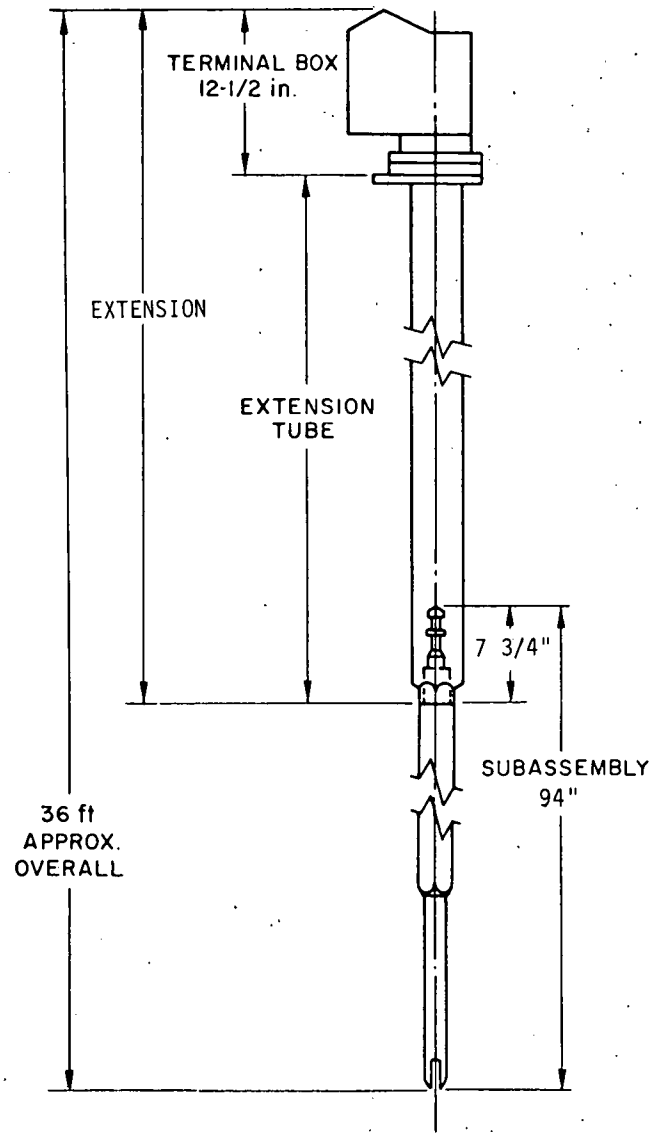

Figure 3. Subassembly and Extension Tube.

TABLE II. EBR-II INSTRUMENTED SUBASSEMBLY TESTS

TEST NO.	IDENTIFICATION	OBJECTIVE
XX01	PROTOTYPE SUBASSEMBLY	DEMONSTRATION OF SYSTEM IN EBR-II
XX02	PNL-17 MIXED OXIDE	OPERATING CHARACTERISTICS OF 37-PIN ASSEMBLY
XX03	CREEP CAPSULE MARK I	CREEP OF LMFBR REFERENCE MATERIAL
XX04	PNL-17A MIXED OXIDE	FUEL AND COOLANT TEMPERATURE MEASUREMENTS
XX05	CREEP CAPSULE MARK II	EXTENSION OF CREEP MEASUREMENTS OF XX03
XX07	PNL-17B MIXED OXIDE	FISSION GAS RELEASE AND COOLANT TEMPERATURE MEASUREMENTS

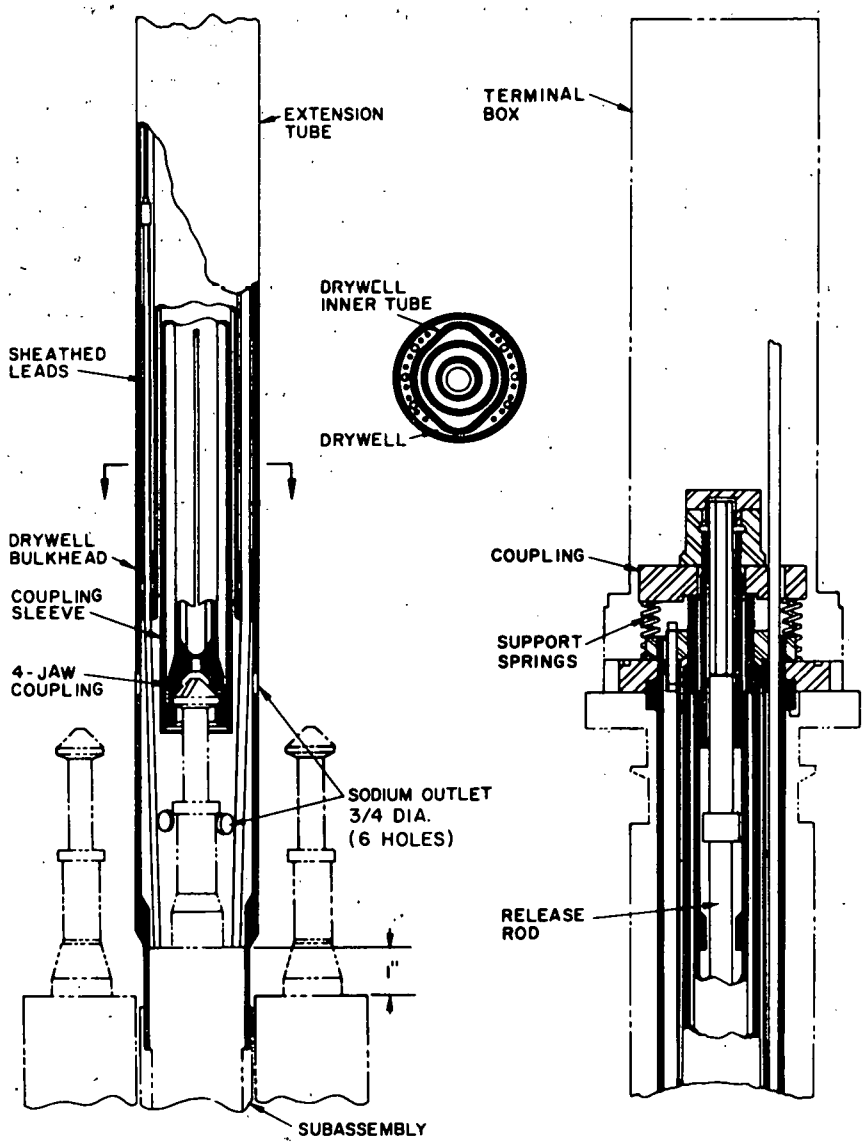

Figure 4. Extension Tube.

The INSAT system has been designed to minimize changes to the reactor system and interference with standard reactor operations. Only a small part of the drive system has to be removed to allow insertion of a new instrumented subassembly and extension tube. When a new assembly is installed, no radiation hazard exists.

Prior to insertion in the reactor, the lower 8 ft of the subassembly and extension tube is preheated to 600°F in a special heater. This preheating is done to lessen the thermal shock while the subassembly is inserted into the primary tank sodium and assures that any sodium bond in the capsules is first melted at the top and gradually melted toward the bottom. After preheating, the subassembly is transferred rapidly from the heater to the top of the reactor shield plug and lowered into the primary tank sodium through the control drive penetration.

When fuel handling is required during the irradiation of the instrumented subassembly, the drive raises the subassembly 8 ft, allowing it to clear all reactor internals so that standard fuel handling operations can take place.

After the INSAT experiment has been completed, the instrument leads are severed by a special cutting tool and the subassembly is disconnected from the extension tube. The subassembly is then removed from the reactor by the standard EBR-II fuel handling equipment.

Bibliography.

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EBR-II NUCLEAR INSTRUMENT TEST FACILITY (NITF)

Location ANL-W, INEL, Idaho

Status Operational

Principal Uses. Two instrument thimbles are available for proof-testing neutron sensors, cables, thermocouples, resistance thermometers, insulation or other types of sensors and equipment in a neutron and gamma environment at elevated temperatures. The facilities are dependent upon the reactor to establish test conditions, so experiments must be coordinated with all other EBR-II in-core and out-of-core experimental programs.

Facility Description. The NITF consists of facilities in the O-1 and J-2 instrument thimbles, which are located as shown in Figure 1. The O-1 thimble is placed vertically outside the neutron shield. The J-2 thimble is slanted 11° from the vertical and penetrates into the neutron shield. Both thimbles are immersed in the sodium coolant in the reactor primary tank and have an ambient temperature of about 700°F with the cooling air and electrical heaters turned off. Readout equipment for temperature and other sensors is located in the test instrument room (TIR).

In the O-1 facility, the cooling-air supply is controlled automatically and the temperature of the thimble can be varied by controlling the amount of cooling air supplied to the thimble. Thimble temperatures are determined by the experimenter and are part of the experiment.

The J-2 facility contains a 19.5-ft-long oven to provide for temperatures up to 1050°F. The temperature within the oven is controlled by the use of heaters and special cooling air. Four independently controlled heater zones provide a nearly uniform temperature throughout the oven length. Direct-current heaters are used so as to reduce electrical noise. In addition to temperature sensors that come with an experiment, there are 14 thermocouples within and throughout the oven length for control, alarm, indication, and emergency shutdown functions.

Test Capability. The NITF thimbles are each 28 ft long. All space below the 8-ft-long shield plug at the top is available for testing. Experiments are held in position either by a carriage connected to the bottom of the shield plug or by suspension from wire cables running through the shield plug. Two conduits through the shield plug, one 1-1/2 in. ID and the other 2 in. ID, are available for experiment cables. The conduits are also used for carrying cooling air.

The inner dimensions of the oven of the J-2 facility are 7 in. ID X 234 in. long. Experiments are held in a carrier at the bottom of the oven. The carrier is supported by two

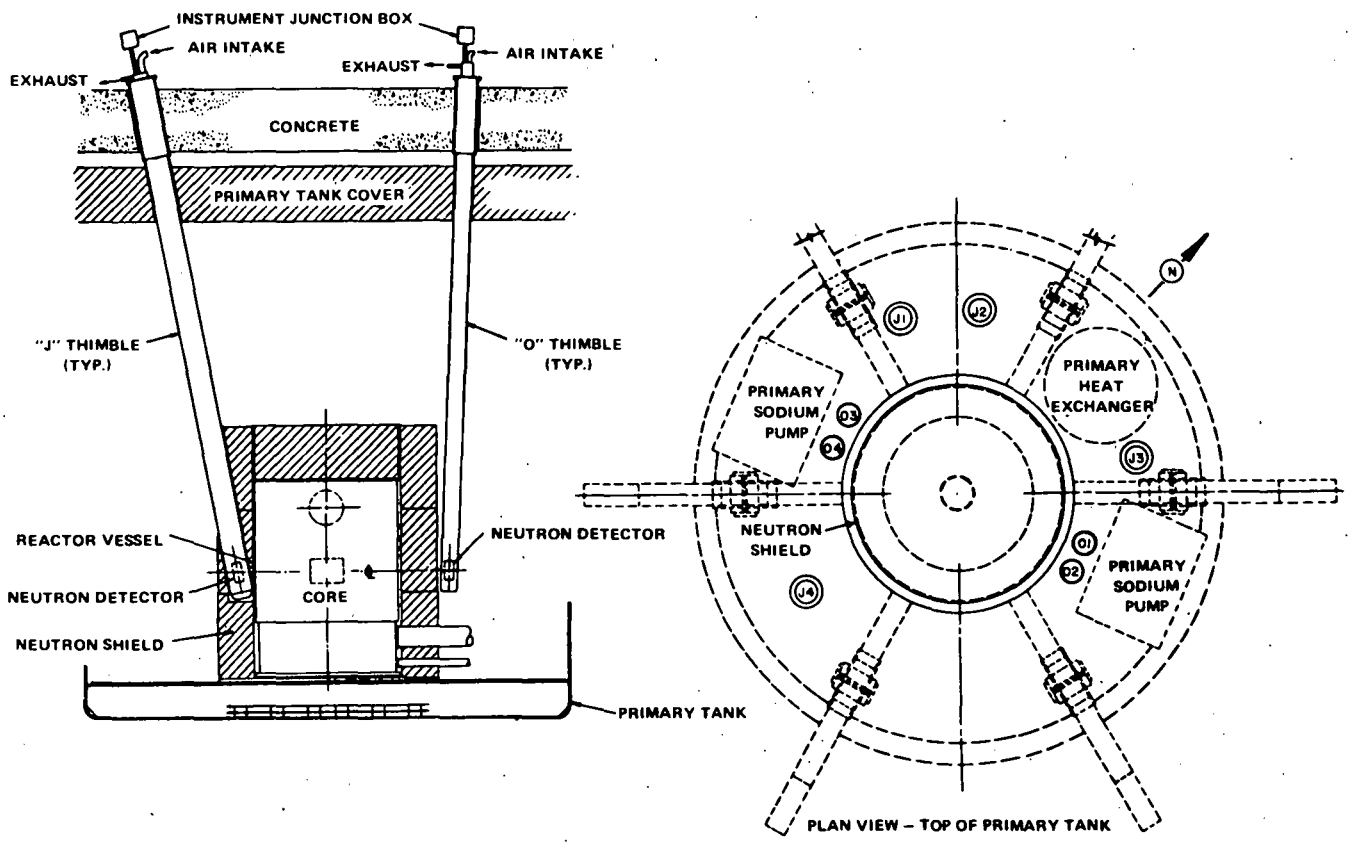

Figure 1. Location of Nuclear Instrument Thimbles in EBR-II.

1-in.-OD thermocouple conduits and three 2-in.-OD X 0.035-in.-wall conduits for experiment cables. The detector carrier in Figure 2 shows how three experiments, each about 3-1/2 in. OD X 24 in. long, can be housed.

Experiments are installed and removed by the EBR-II operations group. The detector carriers for the O-1 facility can be constructed of aluminum, whose radioactivity decays rapidly after shutdown. Detector carriers for the J-2 thimble are constructed of stainless steel. The activation of stainless steel makes remote handling necessary.

Table I lists the general experimental parameters for the NITF facilities. These values are for a six-row core with a depleted-uranium blanket. At the present time the core contains, primarily, fuel in the seventh row and a stainless steel reflector in rows 8-11. Calculations and preliminary data from operating detectors indicate that the levels of both the neutron flux and the gamma flux have increased somewhat.

Table II gives the approximate accuracy of measuring and controlling the experimental parameters.

The only information available on the neutron- or gamma-energy spectrum are neutron spectral indices from foil measurements in both the O-1 and J-2 facilities. These are listed in Table III as approximate reaction rates.

Auxiliary Equipment.

- A remote-handling station with shield window and manipulators in a reactor-building storage pit is used for loading, unloading, and other handling of highly radioactive experiments.
- A digital data acquisition system (DAS) is available for data logging, recording, and computer processing of experimenters' data.
- Some electronic equipment can be supplied for test and calibration of experimental circuits.
- Limited space is available in a controlled-temperature test instrument room (TIR) for experimental instrumentation and control equipment.
- The Hot Fuel Examination Facility (HFEF) is available for remote disassembly and examination of highly radioactive experimental components.

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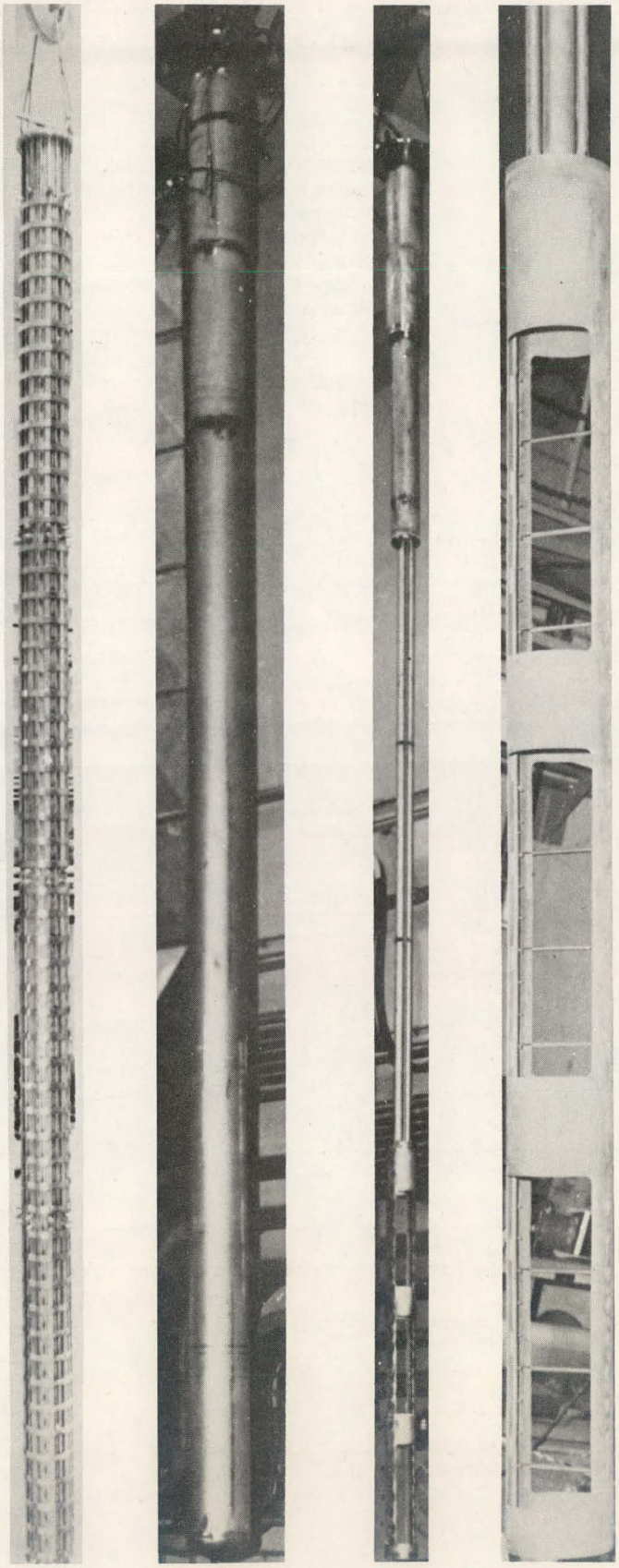

(a) (b) (c) (d)

Figure 2. NITF J-2 Facility Features.
(a) Oven Tube with Heaters Installed.
(b) Complete Oven Assembly.
(c) Shield Plug and Detector Carrier Assembly.
(d) Detector Carrier.

TABLE I. NITF OPERATING CHARACTERISTICS
(62.5 MWT, CORE MIDPLANE, SIX-ROW CORE, DEPLETED-URANIUM BLANKET)

CHARACTERISTICS	O-1 THIMBLE	J-2 THIMBLE
NEUTRON FLUX, N/CM^2-SEC	6.4×10^8	8.0×10^{10}
GAMMA ACTIVITY, R/HR	5.5×10^4	1.2×10^6
GAMMA HEAT, W/CU CM OF SS	~0.001	~0.02
TEMPERATURE, °F	125–700	700–1050
PRESSURE, PSIA	12.2	12.2
COOLANT FLUID	AIR	AIR
HEATING POWER		
ELECTRIC (STARTUP), KW	NONE	13.2
(NORMAL), KW	NONE	6.6
GAMMA, W/FT	~5	~50
TEST SPACE, IN.	4.3 ID × 240 LONG	6.25 ID × 84 LONG

TABLE II. ACCURACY OF CONTROL AND MEASUREMENT
OF NITF OPERATING PARAMETERS

PARAMETER	APPROXIMATE ACCURACY	
	CONTROL	MEASUREMENT
TEMPERATURE	± 20°F	± 12°F
NEUTRON-FLUX LEVEL	± 1%	± 10%
GAMMA-FLUX LEVEL	± 1%	± 10%
NEUTRON SPECTRAL INDICES	-	± 20%

TABLE III. APPROXIMATE REACTION RATES (REACTIONS/SEC-MG) AS MEASURED BY FOIL IRRADIATIONS IN THE O-1 AND J-2 FACILITIES
(SIX-ROW CORE, 62.5 MWT, DEPLETED-URANIUM BLANKET)

O-1 FACILITY

DISTANCE ABOVE MIDPLANE IN.	DETECTOR						
	Au ($\times 10^{-5}$)	Au(Cd) ($\times 10^{-5}$)	^{235}U[a] ($\times 10^{-5}$)	$^{235}U(Cd)$[b] ($\times 10^{-5}$)	Cu ($\times 10^{-3}$)	Co ($\times 10^{-4}$)	Ni ($\times 10^{0}$)
0	5.31	4.43	8.12	2.14	18.62	22.12	5.53
12	5.37	4.46	7.80	2.14	18.00	21.50	2.71
24	5.31	4.37	6.84	2.14	16.12	19.62	2.53
36	4.87	3.87	4.81	1.92	9.28	12.87	1.08
48	2.41	2.25	2.35	1.07	3.15	4.97	0.58
60	1.20	1.20	0.85	0.49	1.27	2.11	0.20
72	0.61	0.61	0.31	0.21	0.60	1.00	0.03

J-2 FACILITY

DISTANCE ABOVE MIDPLANE IN.	DETECTOR								
	Au ($\times 10^{-7}$)	Au(Cd) ($\times 10^{-7}$)	^{235}U[a] ($\times 10^{-7}$)	$^{235}U(Cd)$[b] ($\times 10^{-7}$)	Cu ($\times 10^{-6}$)	Fe ($\times 10^{-5}$)	Ni ($\times 10^{-3}$)	Co ($\times 10^{-7}$)	^{238}U ($\times 10^{-8}$)
0	16.00	14.25	8.55	6.41	2.85	14.00	2.22	3.76	4.85
12	11.12	9.75	6.20	4.49	2.26	11.75	1.85	3.23	4.21
24	6.75	6.12	3.48	2.29	1.66	7.37	0.82	2.12	3.11
36	3.25	3.00	1.58	1.15	0.96	4.00	0.50	1.22	1.56
48	1.25	1.25	0.63	0.45	0.37	1.87	0.18	0.57	0.78

[a]REACTION RATES ARE FOR FISSION.

[b]REACTION RATES ARE FOR FISSION WITH 0.020-IN.-THICK CADMIUM COVER.

EBR-II RADIOACTIVE SODIUM CHEMISTRY LOOP (RSCL)

Location ANL-W, INEL, Idaho

Status Operational

Principal Uses.

- Testing and demonstration of impurity meters and related devices for monitoring sodium (e.g., oxygen, hydrogen, and carbon meters, and wire equilibration devices).
- Testing and demonstration of on-line instrumentation for detecting and measuring fission products in sodium.
- Testing and demonstration of equipment for sampling radioactive sodium.
- Exposure of materials in the EBR-II out-of-core sodium environment to obtain data on corrosion rates and changes in materials properties for comparison with analogous information from in-core materials exposures.
- Demonstration of equipment for measuring deposition of fission products and radioactive corrosion products from primary sodium.

Facility Description. The RSCL is located in the basement of the EBR-II reactor building. It is adjacent to the primary purification cell, which contains the EBR-II primary cold trap, and is shielded independently of the purification cell. The loop is connected to the influent and effluent piping of the cold-trap system (Figure 1).

The main loop is a 2-in. IPS, Type 304 stainless steel circuit with a flow capability of 30 gpm at a dynamic head of 14 psig. Five pairs of supply and return lines are provided from the main loop to furnish continuously flowing sodium to each of five subsystems comprised of the experimental equipment and associated equipment for adjusting the temperature and flow rate of the sodium to desired values. Three of the subsystems are housed in a complex of three lead-shielded cells (designated A, B, and C, Figure 2). These cells are shielded independently of each other and of the pipe tunnel containing the main loop piping. They are large enough to permit personnel entry. The other two cells (D and E) will be shielded as needed by installing lead bricks around the experiments to accommodate the shape and size of the installed equipment.

General Parameters

Design pressure	100 psig[a]
Design temperature	750°F[a]
Pump shutoff head	30 psig
Operating temperature (main loop)	750°F
Design flow	50 gpm
Volume of main-loop piping	Approx 15 gal

[a]Conforms to the 150-psi rating of ANSI B16.5.

Test Capability. Experimental equipment is installed in the lead-shielded cells. The three major cells (A, B, and C) have internal floor areas of 31, 21, and 27 sq ft, respectively, and are about 7-1/2 ft high inside. The piping to carry the sodium supply and return streams between the main loop and the cells penetrates the main loop shield in such a manner that excessive radiation streaming cannot occur. Access to an individual cell is possible while the main loop and other cells remain in operation, after the system in that cell has been shut down and the ^{24}Na activity has decayed. Access to the cells is provided by vertical-lift shield doors and by shielded roof hatches.

An experimental system must be provided with electrical heaters and temperature-monitoring and control equipment and leak detectors, integrated into the overall electrical heating and leak-detection system of the RSCL facility. Instruments associated with in-cell experiments are installed in panels adjacent to the RSCL.

TABLE I. QUALITY OF EBR-II PRIMARY SODIUM COOLANT

TYPE OF CONTAMINANT	CONTAMINANT	AMOUNT
NONMETALS	CARBON	<2.0 PPM
	OXYGEN	<1.0 PPM
	HYDROGEN	<1.0 PPM
METALS	TIN	23 PPM
	LEAD	10 PPM
	BISMUTH	2.0 PPM
	IRON	0.1 TO 0.5 PPM
	TOTAL OF ALL OTHER METALS (EXCEPT ALKALI METALS)	<1.0 PPM
RADIONUCLIDES[A]	^{24}Na	1400 μCI/G (@62.5 MWT)
	^{22}Na	0.05 μCI/G
	^{137}Cs	0.013 μCI/G
	^{117}Sn	TRACE

[A]THE RADIONUCLIDES LISTED ARE FOUND IN RADIOMETRIC ANALYSES OF BULK SODIUM SAMPLES. TRACE AMOUNTS OF THE FOLLOWING ARE FOUND IN DISTILLATION RESIDUES (IN ADDITION TO ^{117}Sn): ^{54}Mn, ^{125}Sb, ^{110m}Ag, ^{113m}In.

The following general design guidelines shall apply to test systems:

- The temperature of the sample stream returning from a test system to the main loop shall not exceed 750°F.
- The flow through a test system containing an analytical device (e.g., oxygen, hydrogen, or carbon meter) shall not exceed 2 gpm.
- The heat dissipation from a test system to the RSCL ventilation system shall not exceed 1.5 kW.

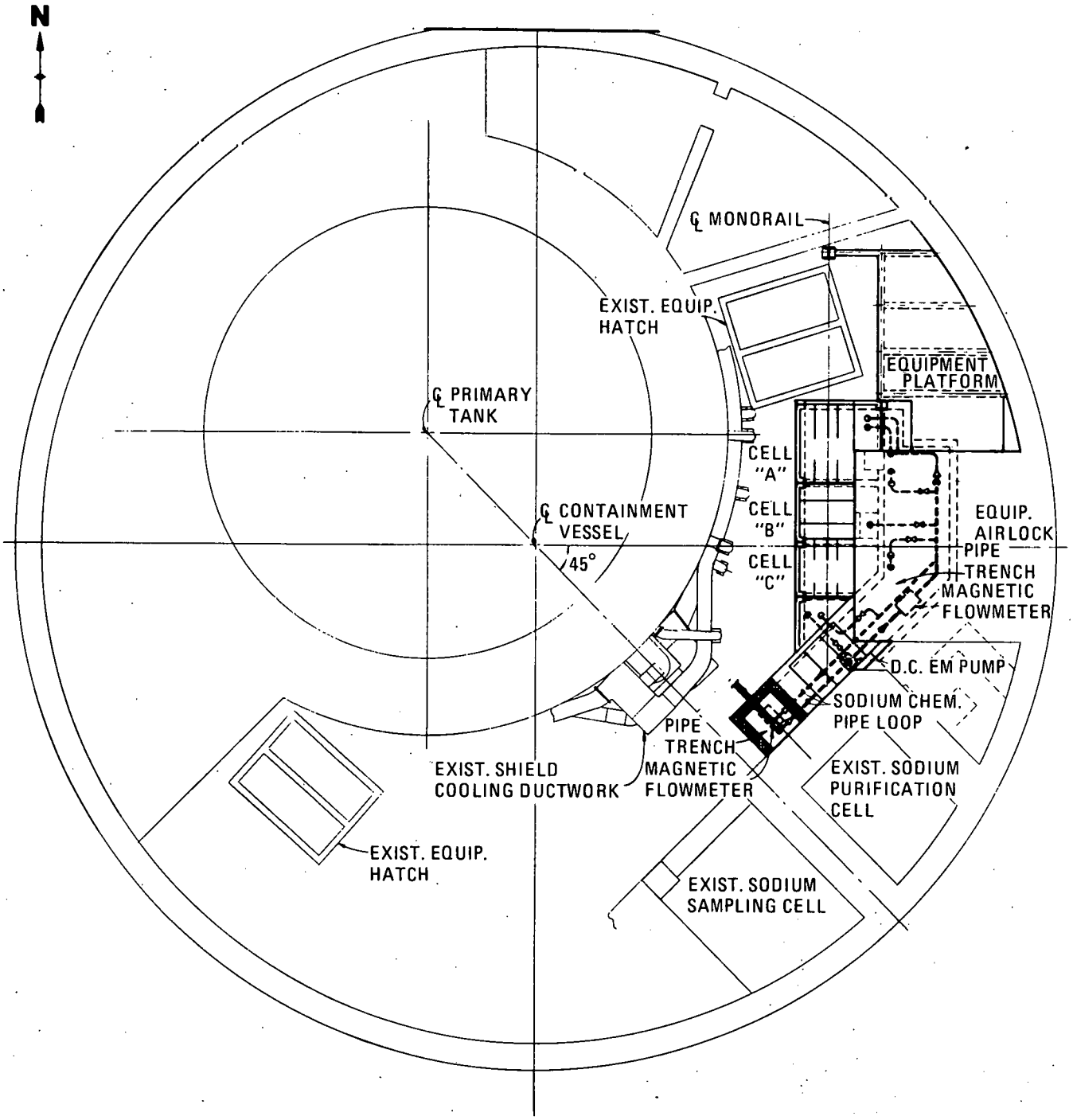

Figure 1. Location of RSCL in EBR-II Reactor Building.

Depending upon the status of test systems in the RSCL, it may be possible to modify the guidelines for a particular test if necessary. Test requirements will be considered during the assessment of test feasibility in relation to RSCL capabilities and current status of other tests.

Auxiliary Equipment. Signals from experimental equipment can be fed into the EBR-II digital Data Acquisition System (DAS) for recording and processing.

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1. "Reactor Development Program Progress Report, April–May 1971," Argonne National Laboratory, ANL-7825, pp. 1.16–1.19.

2. "Radioactive Sodium Chemistry Loop (RSCL)," from Guide for Irradiation Experiments in EBR-II, Rev. 4, Appendix J, Argonne National Laboratory, February 1971.

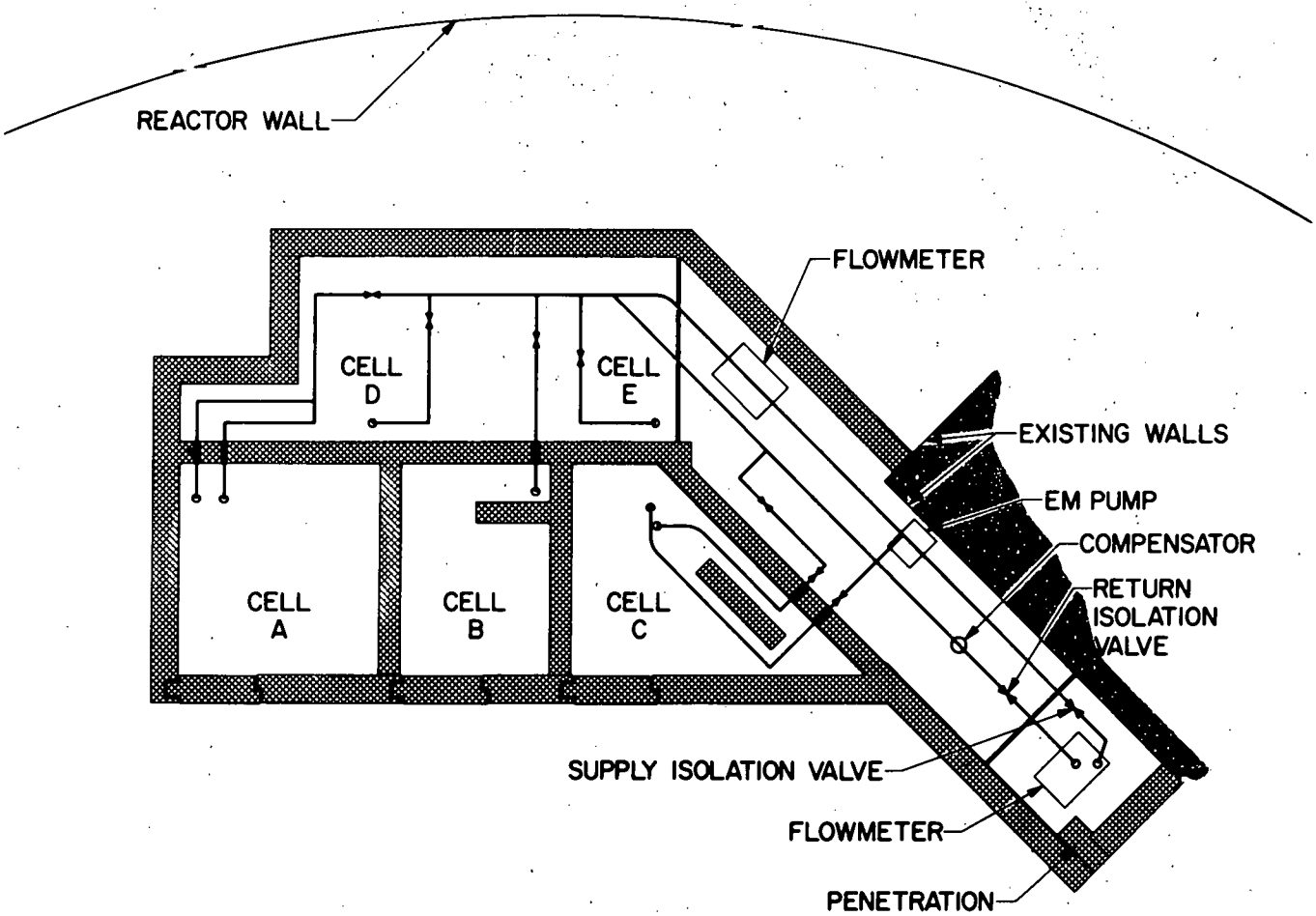

Figure 2. RSCL Test Cell Configuration.

HOT FUEL EXAMINATION FACILITY (HFEF)

Location ANL-W, INEL, Idaho

Status Operational

Principal Uses. Postirradiation handling, examination, and analysis of fuels and reactor materials from LMFBR programs, including:

- Nondestructive (interim) examinations of experimental capsules and elements—subassembly dismantling and reassembly, visual and photographic examinations, gamma scanning and weight and dimensional measurements.
- Destructive (terminal) examinations of fuel, cladding, and structural materials—cutting, fission-gas sampling, and preparation and examination of metallographic samples.
- Handling, disassembly, examination, and assembly of TREAT and SLSF loops.
- Off-site shipment of irradiated components for examinations elsewhere.
- EBR-II reactor support services—surveillance of driver-fuel and handling, examination, and disposition of spent driver-fuel, blanket and reflector subassemblies and of irradiated reactor components.
- In-cell neutron radiography (future—planned for 1977).

Facility Description. HFEF comprises two adjacent hot cell complexes—HFEF/North (HFEF/N) and HFEF/South (HFEF/S). Each complex contains two heavily shielded hot cells, one with an inert atmosphere (argon). Each complex includes unshielded repair areas, test laboratories, equipment rooms, storage areas, offices and other facilities.

HFEF/N was designed and constructed to provide highly reliable alpha-gamma containment and protection, and to accommodate large irradiated loops and components, so that SLSF, TREAT, and other LMFBR irradiations programs can be efficiently serviced. The HFEF/N cell complex is in a three-story structure that includes a large high-bay area over the cells. A 50-ton high-bay crane transports bottom-opening casks to sealed penetrations in the ceiling of the main cell for inserting large loops. Top-opening casks are lowered into a tunnel beneath the cells and sealed to transfer ports in the floor, to introduce other materials into the cells and to remove solid wastes from the cells.

HFEF/S provides complementary facilities and capabilities, but is limited to handling smaller components and fuel assemblies that do not expose transuranic elements. HFEF/S is contiguous with the EBR-II reactor building. A rail-mounted cart and a special gas-tight top-opening cask facilitate inter-building transfers.

Test Capability. The HFEF/N decon cell is designed primarily for decontamination and maintenance of equipment and for material transfers. Items to 6 ft diameter and 12 ft-10 in. long can be transferred into the main cell. Loops to 24 in. diameter and 30 ft long can be lowered directly into either HFEF/N cell from bottom-opening casks.

The HFEF/S air cell provides a remotely-operated, radiation-shielded area for EBR-II subassembly disassembly and reassembly, and for examination of fuel elements and capsules in an air atmosphere. It also serves as a terminal for gas-locks into the contiguous argon cell.

The HFEF/S argon cell accommodates experiments and operations that cannot be exposed to air atmosphere. The cell interior is a 22 ft deep annulus formed by inner and outer 16-sided regular polygons.

Auxiliary Equipment.

- Remotely operable wet-decontamination facilities.
- ANL-W Analytical Laboratory (nearby, but not part of HFEF) for wet chemical, radioisotope, fission gas, burnup, spectroscopy, and gas chromatography analyses.
- Scanning electron microscope for low-level specimens.
- Metallographic sample preparation containment box containing an independent inert atmosphere to permit in-cell metallographic sample preparation that would otherwise "poison" or "dirty" the HFEF/N main cell atmosphere.
- Bag-out system for packaging contaminated materials for safe removal from the cells, and for cask transfers.
- Gloveboxes and glovewalls permitting limited contact-maintenance operations on contaminated items.
- On-site neutron radiography (provided by nearby TREAT reactor facility operated by ANL but not part of HFEF).
- Data acquisition and process control system for computer control of selected in-cell equipment, and for data collection and reduction.

Modifications Authorized. After the newly-activated HFEF/N facility is thoroughly debugged (late in 1975), the HFEF/S facility will be shut down for extensive refurbishment. The cells will be entered and decontaminated, the cell windows and lighting will be modified, and the remote handling equipment will be improved. In-cell examination equipment will also be repaired, modified, or replaced as appropriate. In 1977, HFEF/S will be reactivated to complement HFEF/N for interim nondestructive examinations of EBR-II experimental irradiations, and to provide a standby backup hot cell facility for LMFBR program activities.

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1. Sales, W. L., and DeCoria, K., "Radioactive Materials Shipment at the Fuel Cycle Facility," Argonne National Laboratory, May 1969.

2. "Guide for Irradiation Experiments in EBR-II, EBR-II Project," Argonne National Laboratory, latest revision.

3. "HFEF/N Basic Facility System Design Descriptio (BFSDD)," Argonne National Laboratory, Apr 1974.

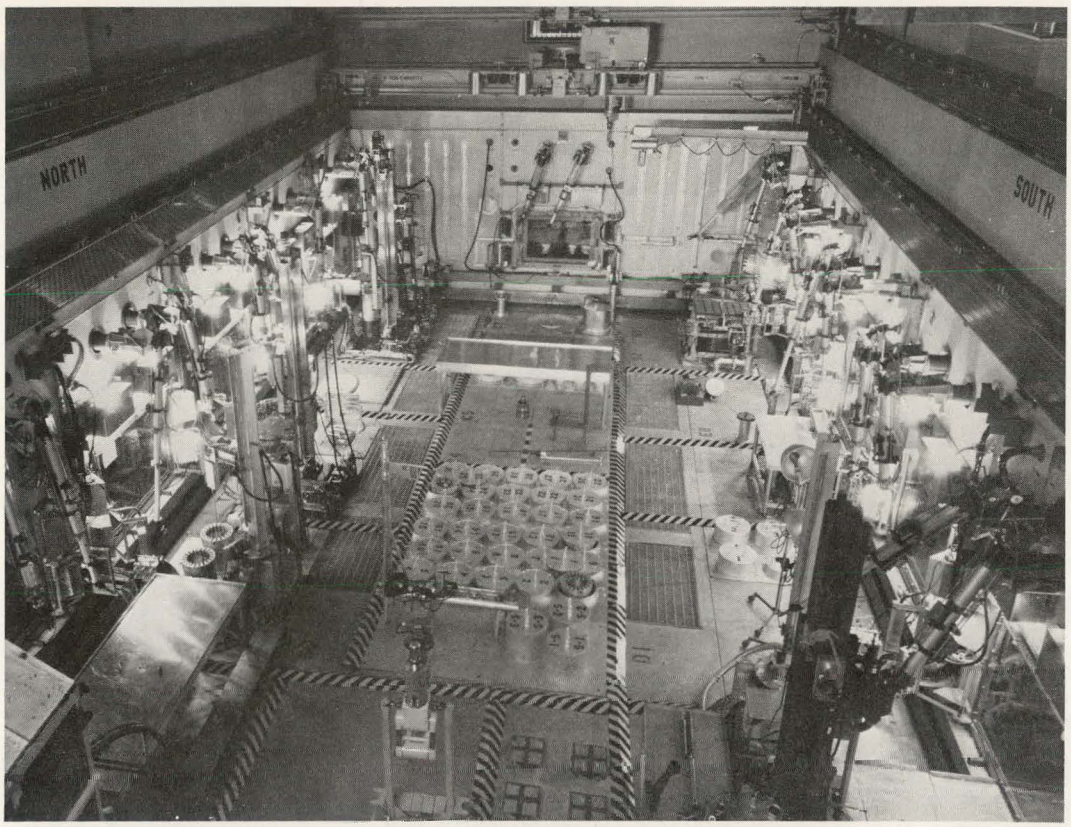

Figure 1. Interior View of HFEF/N Main Cell.

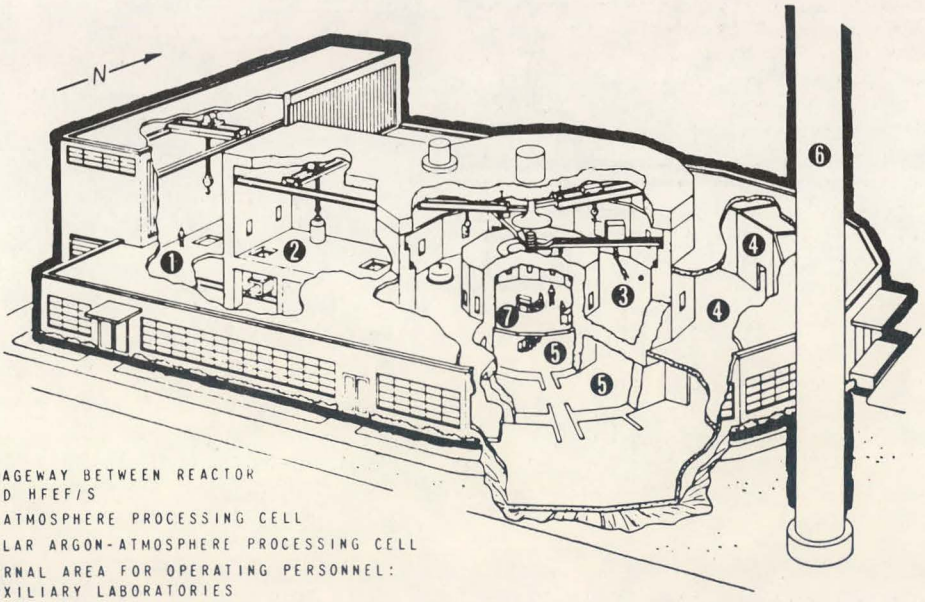

Figure 2. Cutaway View of HFEF/S.

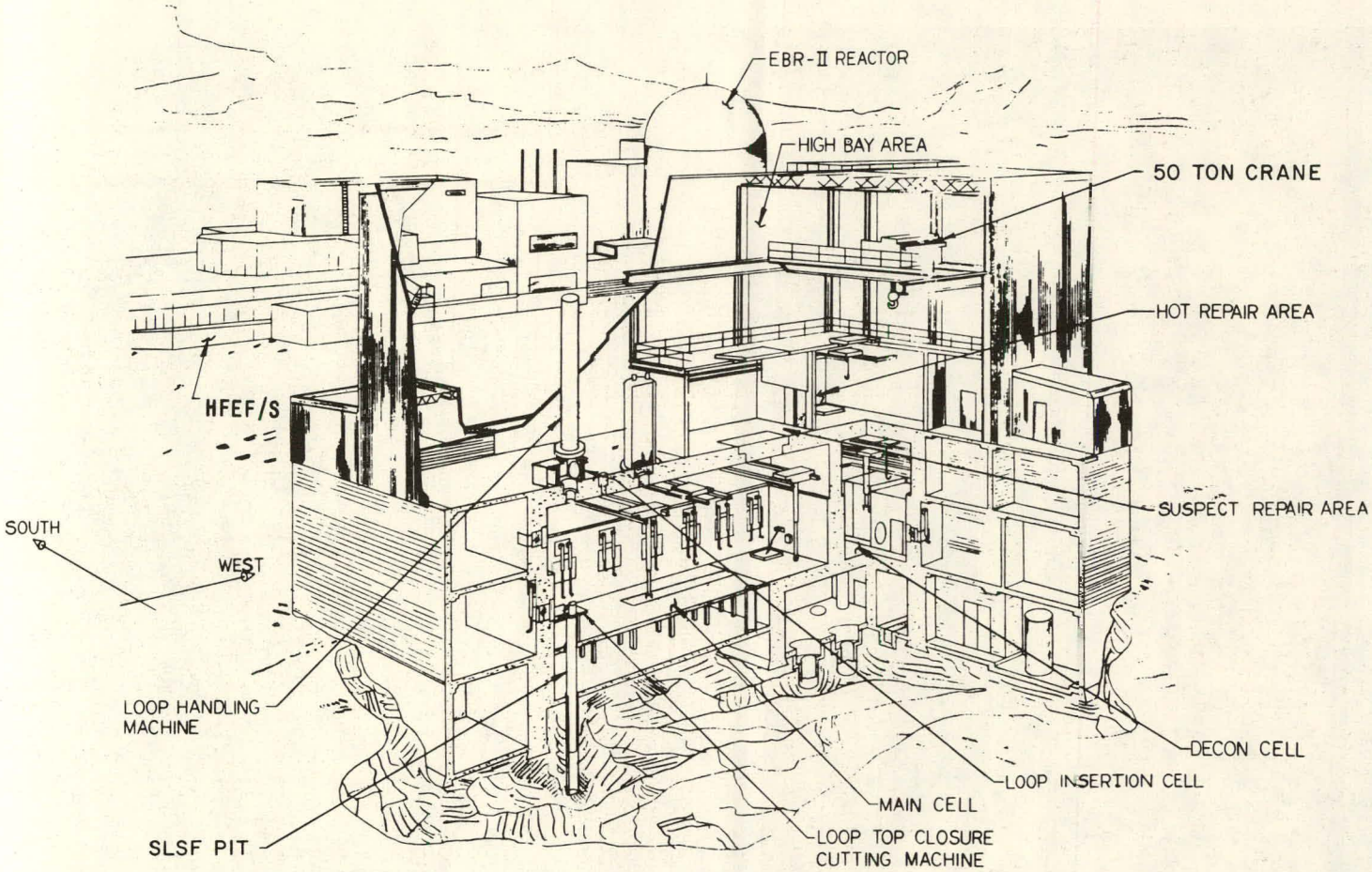

Figure 3. Cutaway View of HFEF/N.

SODIUM LOOP SAFETY FACILITY (SLSF)

Location ANC, INEL, Idaho

Status Operational

Principal Uses. The SLSF, designed by the Argonne National Laboratory, is used to investigate the following:

- Consequences of a local malfunction within an LMFBR core.
- Circumstances under which such malfunctions could cause failure propagation and thus involve a larger segment of the core.
- Methods for detection of and protection against those phenomena that could cause failure propagation.
- Selected phenomena associated with whole core accidents.

The SLSF should be useful in a wide range of in-pile tests such as:

- Loss of flow tests.
- Fuel clad failure and fuel-coolant interactions.
- Failure propagation resulting from continued operation with failed fuel elements.
- Gas release experiments.
- Experiments involving overenriched pins.
- Experiments which involve the potential for the release of large amounts of fuel.
- Experiments involving pre-irradiated fuel.

Facility Description. The in-pile loop is a doubly-contained, packaged sodium loop designed for use in the Engineering Test Reactor (ETR) located at the Idaho National Engineering Laboratory (INEL).

The loop, 27 ft long, consists of the primary containment tube, secondary containment tube, top closure, test train, and instrumentation.

The test train, a separate subsystem within the loop, is 26 ft long and includes the flow divider, fuel subassembly tungsten meltdown cup, and associated instrumentation. The test train and loop are sized to accommodate a subassembly of up to 61 fuel elements.

TABLE I. STEADY STATE LOOP PARAMETERS

TYPE	OPERATING RANGE
FLOW RATES	
TOTAL LOOP SODIUM	15–150 GPM
TEST SECTION INLET	10–100 GPM
TEST SECTION BYPASS	10–100 GPM
TEMPERATURES	
TEST SECTION INLET	400–950°F
TEST SECTION OUTLET	400–1300°F
HX INLET	400–1100°F
HX OUTLET	400–850°F
PRESSURES	
TEST SECTION INLET	20–200 PSIA
TEST SECTION OUTLET	20–200 PSIA
LOOP GAS PLENUM	10–60 PSIA

MAX. TRANSIENT PRESSURE, 1800 PSIA.
MAX LOOP POWER, 1.5 MWT.

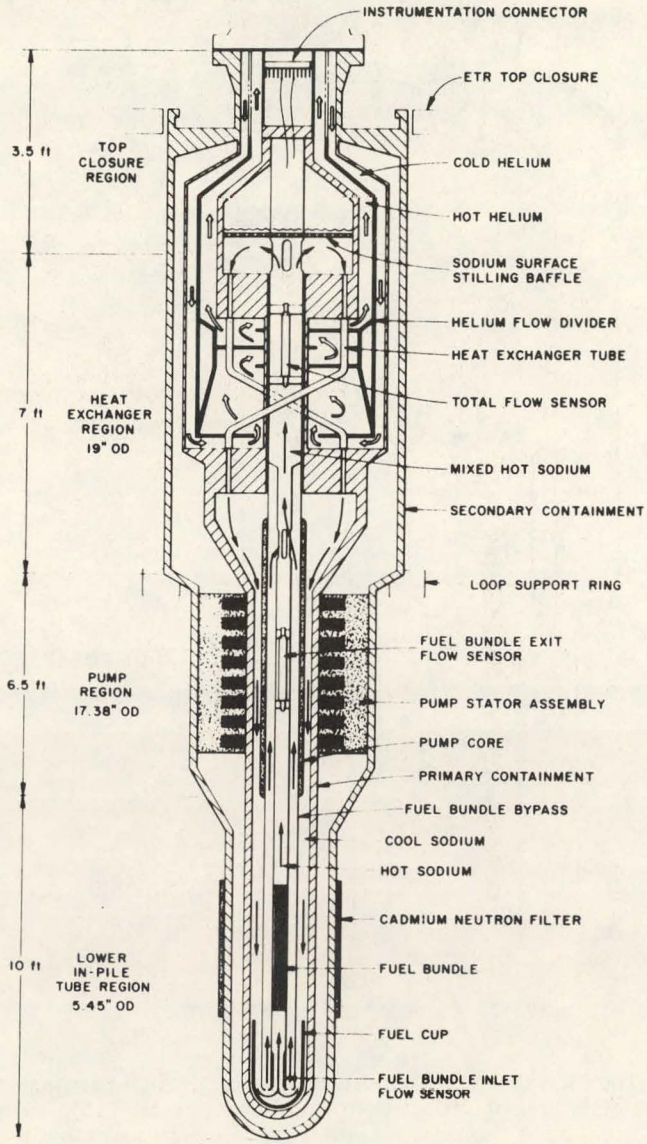

Figure 1. SLSF Loop Schematic.

In addition to the in-pile loop and test train the complete SLSF includes:

- Data acquisition system.
- Loop control system.
- Safety system.
- Instrument integration system.
- Loop secondary coolant system.
- Loop handling system (LHS).
- Post-test examination system.
- Loop filling, storage, and remelt (FSR) system.
- ETR integration system.

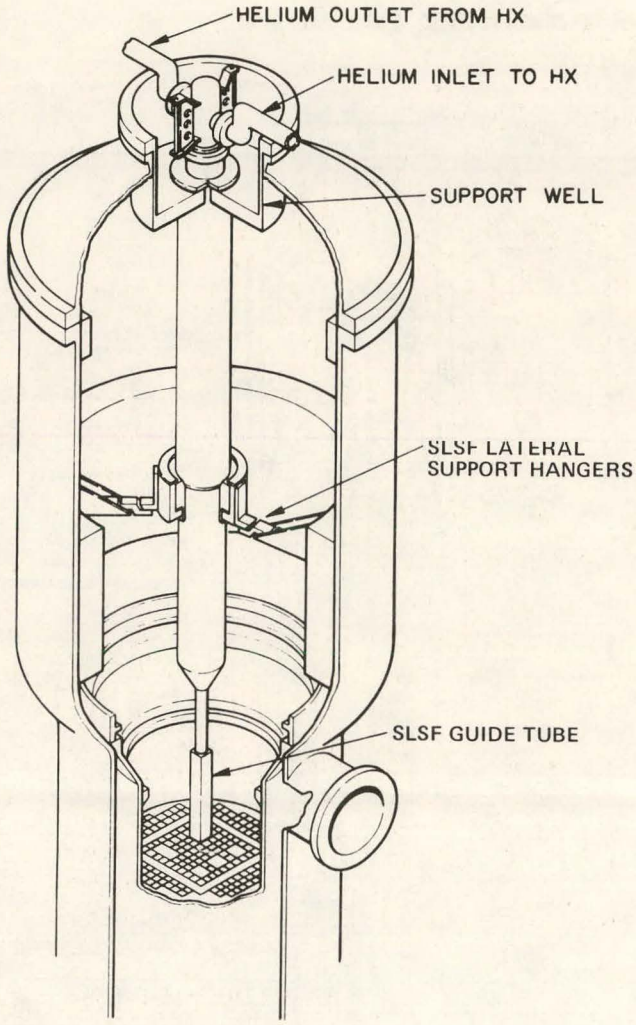

Figure 2. SLSF Loop in ETR.

The loop design is intended to provide the best practicable simulation of the desired conditions in ETR for each experiment, consistent with constraints imposed by the design and flux source. A neutron filter is provided to harden the neutron spectrum which in turn allows a temperature profile within a fuel element close to that found in a fast reactor.

Test Capability. The SLSF is assembled, charged with sodium, and checked out in the FSR System, located in the southeast corner of the ETR main room. The SLSF is removed from the FSR System and placed in the LHS. A 50-ton crane at ETR provides for the transport of the LHS and positioning of the loop assembly in the ETR core.

Heat is removed from the sodium loop by a secondary helium coolant system and released to the ETR coolant.

The test train is provided with sensors for measuring sodium flow rate, sodium temperature, and sodium pressure, under both static and transient conditions. Special pressure sensors on selected fuel pins measure the release of pressure from a fuel pin due to failure. Sodium flow through the fuel bundle and total flow are measured. Temperature and pressure sensors are provided for the loop containment. Sensor signals are suitable conditioned and integrated with a Data Acquisition System, Loop Control System, and Safety System.

The Data Acquisition System consists of a PDP-15-20 computer and associated peripherals to provide a system where real-time data acquisition and control tasks are combined with program development and testing. A complementary analogue system records broad-band experimental data, offers limited recording redundancy for digital recorded data, and retains data signals that were recorded ten seconds before a planned or premature experimental event.

Upon completion of active testing, the SLSF Loop is removed from the ETR in the Loop Handling Machine (LHM) and transferred to the Hot Fuel Examination Facilities (HFEF) for post-irradiation examination.

Auxiliary Equipment. Post-irradiation examinations, which include disassembly of the loop and detailed examination of the components, are performed in the HFEF. The remote handling cells provide:

- Clean inert atmosphere.
- Contamination control areas.
- Experiment support devices.
- Contamination control structures.
- Cutting machines.
- Sampling devices.
- Disposal systems.
- Nondestructive and destructive testing devices.
- The capability of transferring the fuel elements to a neutron radiography facility.
- The capability to assemble test trains containing preirradiated fuel.
- Equipment to obtain fission gas samples from the loop plenum and from fuel element plenums.
- Capability of collecting dispersed fuel.

TABLE II. STEADY STATE OPERATING CONDITIONS

PARAMETER	P1	P2
TEST SECTION INLET TEMP.	792°F*	792°F*
TEST SECTION SODIUM FLOW	4.74 LB/SEC	4.51 LB/SEC
BYPASS SODIUM FLOW	8.30 LB/SEC	9.09 LB/SEC
TOTAL LOOP POWER	1080 KW	1011 KW
TEST SECTION POWER	675 KW	730 KW

*TEMPERATURE CONTROL ±30°F.
FLOW CONTROL ±10%.

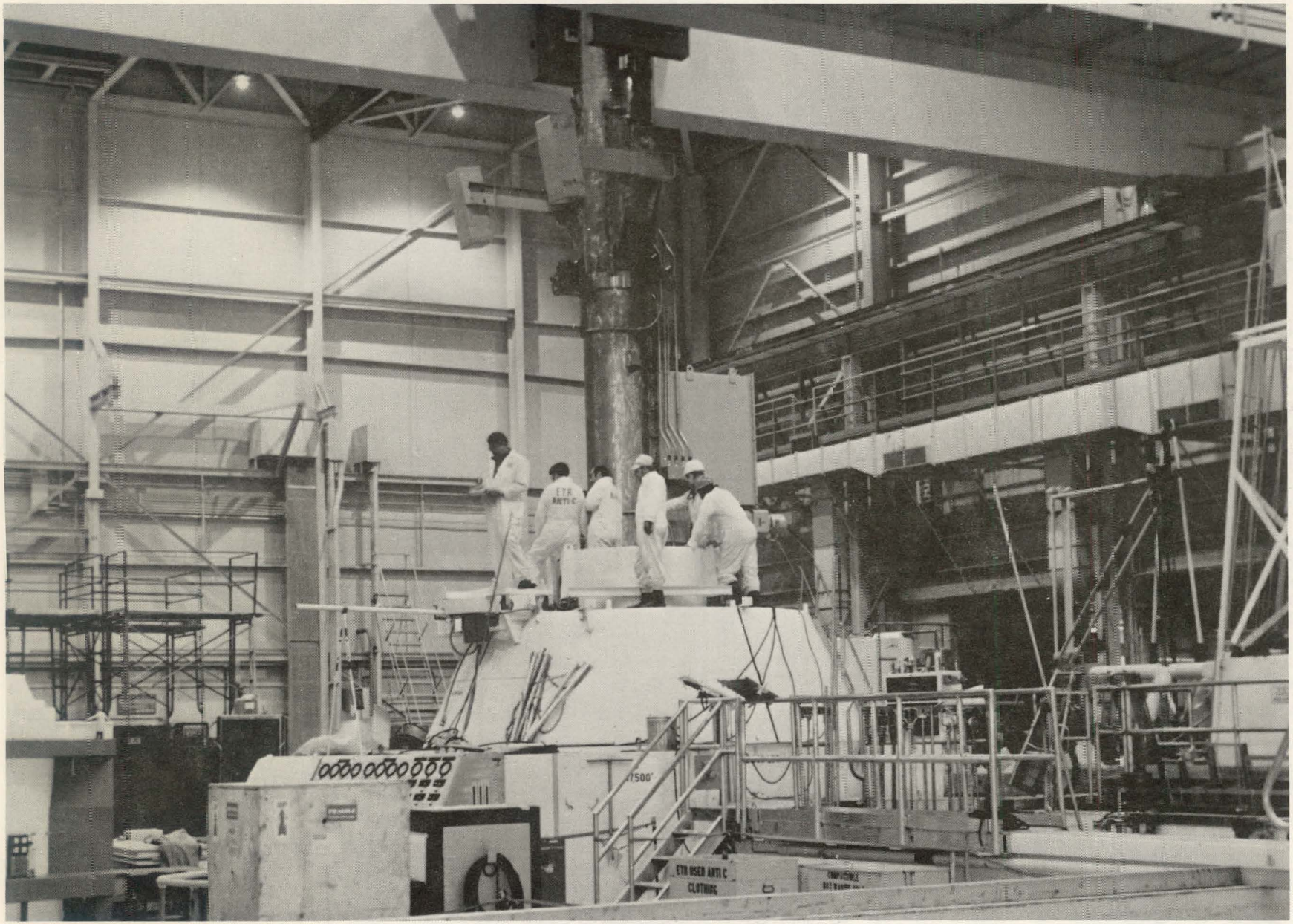

Figure 3. The Sodium Loop Safety Facility (SLSF) Is Lowered into the ETR for Irradiation.

TRANSIENT REACTOR TEST FACILITY (TREAT)

Location ANL–W, INEL, Idaho

Status Operational

Principal Uses. TREAT is an air-cooled, thermal, heterogeneous test facility designed to evaluate reactor fuels under conditions simulating various types of nuclear excursions and transient undercooling situations in a nuclear reactor.

The following types of experiments can be performed:

- Fuel meltdowns.
- Fuel-coolant interactions.
- Fuel-failure thresholds.
- Coolant dynamics during overpower or undercooling transients.
- Molten material relocation.
- Fuel restructuring during sub-failure transients.

In its steady-state operation TREAT can be used as a neutron-radiography facility.

Facility Description. The TREAT fuel assemblies are 4-in. square by 9-ft long consisting of a 4-ft central section of fuel with a 2-ft graphite reflector section on each end. There is a 19 X 19 grid for fuel assemblies or 8-ft graphite reflector assemblies. The assemblies are surrounded by 2 ft of permanent graphite reflector. The fuel is UO_2 dispersed in graphite. The uranium is fully enriched, and the carbon-^{235}U atomic ratio is 10,000:1. The size of the core can be adjusted to provide the amount of excess reactivity required to perform a particular experiment. During a transient the fission energy is absorbed by the graphite moderator which is then air-cooled following the transient.

There are six electromechanical control-rod drives, each of which operate two safety shutdown rods. Two hydraulically actuated drives each operate two transient control rods. The transient rods can be operated in two different modes. In one mode, a step removal of the poison sections results in a power burst which is limited by the reactor's negative temperature coefficient. In the second mode, preprogrammed power shapes are generated with a closed-loop-control system in which a computer uses the reactor linear-power, period, and log-power instrumentation signals as feedback parameters. A cutaway view of the TREAT reactor is shown in Figure 1.

TABLE I. GENERAL PARAMETERS

TRANSIENT CONTROL		
UNSHAPED TRANSIENTS	– STEP REACTIVITY ADDITION AND TEMPERATURE LIMITED	
SHAPED TRANSIENTS	– COMPUTER CONTROLLED TRANSIENT SHAPE	
MAXIMUM REACTOR OPERATING CONDITIONS		
TRANSIENT OPERATION		
	UNSHAPED	SHAPED
INTEGRATED POWER	2100 MW-SEC	1400 MW-SEC
PEAK POWER	16,000 MW	1000 MW
FUEL TEMPERATURE	600°C	600°C
REACTOR PERIOD	0.023 SEC	0.100 SEC
K_{EX}	4.7%	6.2%
STEADY STATE OPERATION		
POWER LEVEL	120 KW	
FLUX DATA IN CENTRAL VOID		
NEUTRON	5.6×10^6 FISSIONS/G ^{235}U-W-SEC 5.0×10^6 NVT/W-SEC (σ_f^{235} = 440 BARNS)	
GAMMA	2.2 R/HR/W 6.0×10^3 R/MW-SEC	

TABLE II. EXPERIMENTAL PARAMETERS*

TYPE OF EXPERIMENTS	– CAPSULES, LOOPS, AND PHOTOGRAPHIC CAPSULES.
TYPE OF FUEL	– METAL, OXIDE, CARBIDE, NITRIDE (U OR PU)
NUMBER OF FUEL PINS	– UP TO 19 PINS, UNIRRADIATED OR IRRADIATED.
COOLANT	– STATIC OR FLOWING SODIUM, NAK, WATER, STEAM, AIR, OR INERT GAS.
MEASURED EXPERIMENT PARAMETERS	– TEMPERATURE, PRESSURE, FLOW, FUEL MOVEMENT, ACOUSTIC EMISSION, LINEAR DISPLACEMENT.

*LIMITATIONS UPON THESE PARAMETERS GENERALLY ARE A FUNCTION OF THE EXPERIMENT DESIGN. A WIDE VARIETY OF DESIGNS CAN BE ACCOMMODATED WITHIN THE CONSTRAINTS OF THE PARAMETERS LISTED IN TABLE I AND OVERALL GEOMETRIC COMPATIBILITY.

Test Capability. The test element for a transient experiment will normally replace either one or two normal TREAT fuel assemblies at the center of the core. (Figure 2 gives the dimensions of a TREAT fuel assembly.) Either single or double containment, depending upon the test specimen, is required so that the containment will retain its integrity during all planned tests and credible accident conditions. Any experiment containing plutonium is limited to a maximum of 150 g plutonium per experiment.

Test elements may also be inserted radially through slotted fuel assemblies. This capability is normally used with test assemblies which have a window in their outer faces. High-speed photographs can be taken of fueled test specimens as they are irradiated and melted.

A mechanical hold-down on the test element must be provided to counteract any forces generated during the test that could lift the test element enough to cause a significant addition of reactivity. All instrument, electrical and coolant leads to the test element must be designed so they can be easily connected and disconnected when the experiment is in the reactor.

Auxiliary Equipment.

- Neutron radiography for pretransient and post-transient nondestructive examination of experiments.
- Neutron hodoscope for monitoring encapsulated fuel movement during irradiation.
- Analytical chemistry and radiochemistry services.
- IBM 360/75 computer services.
- The Hot Fuels Examination Facilities at EBR-II for loading and unloading experiments and for destructive examination.

Modifications Authorized. A new neutron hodoscope is being installed which will increase the experiment viewing slot from 21 in. to 36 in.

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1. "Guide for Irradiation Experiments in TREAT," Argonne National Laboratory, May 1971. (Copies available at TREAT Facility, Argonne National Laboratory, P. O. Box 2528, Idaho Falls, Idaho 83401.)

2. "TREAT Baseline Description Document," Argonne National Laboratory, ANL/RAS-72-23, June 1975.

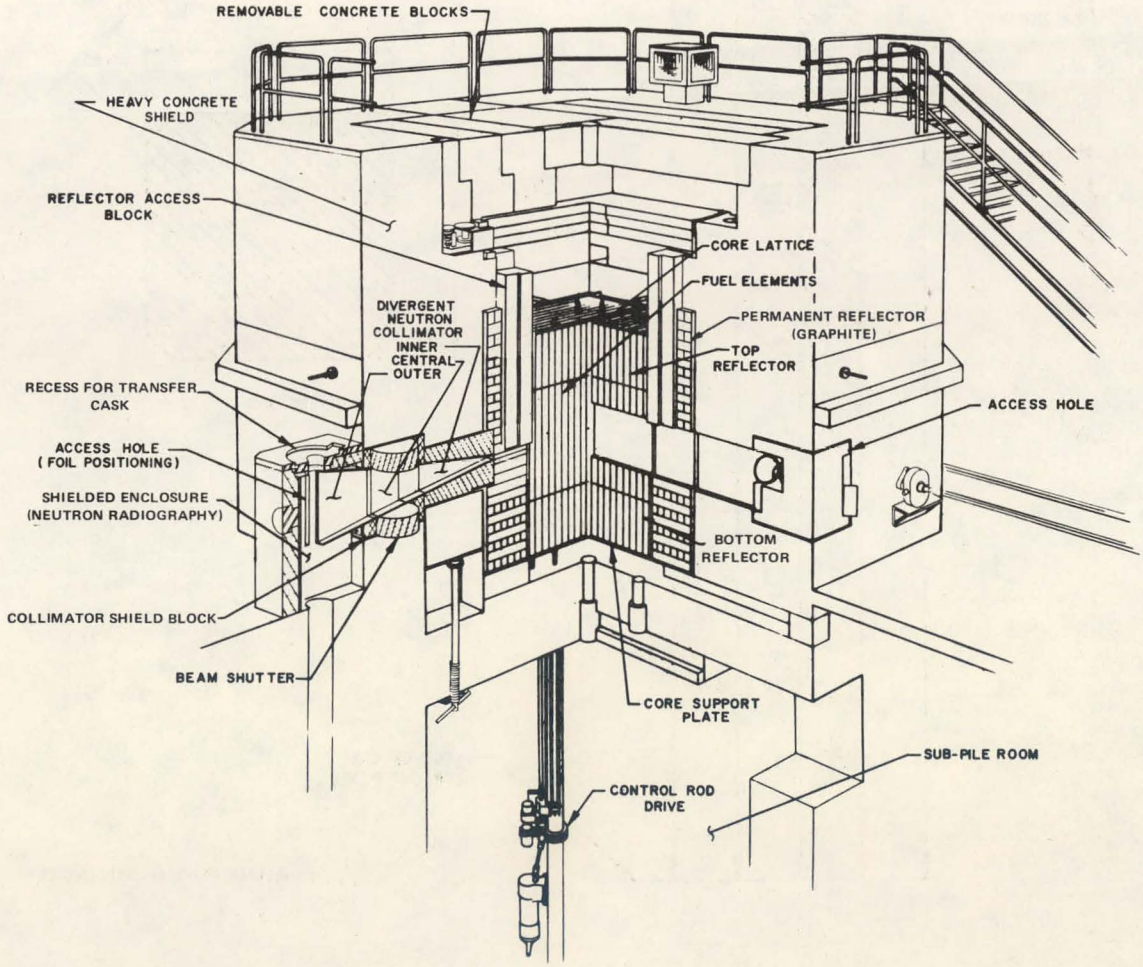

Figure 1. Cutaway View of the TREAT Reactor.

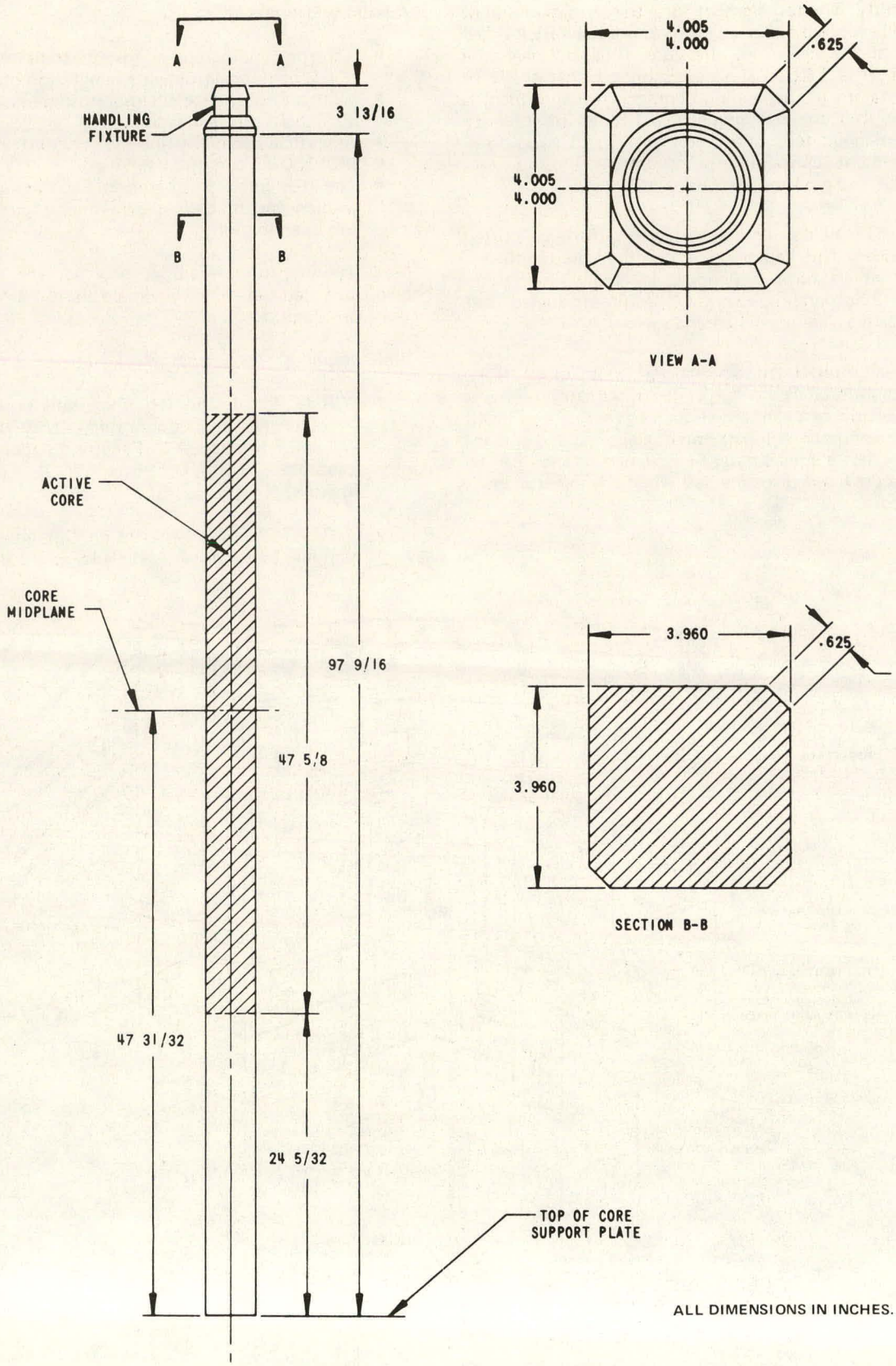

Figure 2. TREAT Fuel Assembly.

TREAT IN-REACTOR TEST APPARATUS (R-SERIES)

Location ANL-W, INEL, Idaho

Status Operational

Principal Uses. The R-Series In-Reactor Test Apparatus is used to conduct simulated loss-of-flow tests with sodium in the TREAT reactor. These are transient tests in which fresh oxide-fuel undergoes transients resulting from the intentional reduction of sodium coolant flow while maintaining the test fuel at constant power. Full-length LMFBR-type fuel pins can be employed in this facility. Among the loss-of-flow tests which can be conducted are:

- Pump coastdown tests.
- Sudden complete inlet blockage tests.
- Pipe rupture tests.
- Partial inlet blockage tests.

These tests can be conducted for various sodium inlet temperatures, average pin powers, and numbers of pins.

Facility Description. The basic design limitations are imposed upon the R-Series In-Reactor Test Apparatus by the TREAT reactor. The reactor is a thermal neutron transient test reactor limited to less than 26 sec of operation at the power level required for this experiment series. The thermal neutron characteristics, the basic size of the core penetration for test apparatus, the available reactor integrated power, and safety precautions limit the practical size of the test fuel to 7 pins of typical FFTF dimensions.

The test apparatus consists of a once-through primary sodium piping system made up of two tanks and connecting piping. The tanks are positioned atop the TREAT reactor with a single piping penetration into the reactor core region. Up to 7 test fuel pins of current FFTF design can be accommodated in an integral test bundle within the piping assembly penetrating the reactor core. The apparatus arrangement is such that neither the tanks nor the test fuel piping penetration become a permanent or integral part of the reactor itself. Therefore, the test apparatus does not affect reactor operation or control once routine provisions are made to compensate for the removal of two reactor fuel elements.

Sodium flow within the system can be driven and controlled by regulating the cover-gas pressure in either of the sodium tanks. The sodium piping contains a single valve for isolation purposes. System auxiliaries include gas supply and regulation equipment, pipe and tank heating, electrical control, instrumentation, and a secondary housing.

Test Capability. Space and size limitations require assembly of the test vehicle atop the TREAT reactor with the downcomer and test section penetration into the 4 X 8 in. core slot. Assembly and disassembly operations require the use of the crane (14.5 ft maximum height) over the reactor. Sodium piping includes filters, downcomer, U-tube test section, plenum simulator, and discharge piping. These piping sections are held together by mechanical connectors, and the entire primary sodium system is positioned inside a secondary housing which serves as a sodium drip container and helps maintain an inert atmosphere for the primary sodium piping.

The instrumentation provides for measurement and readout of sodium flow rates, pressures, temperatures, liquid sodium levels, and the location of boiling inception. Also provided are interlocks and interface controls with the TREAT reactor. An electrical system provides connection and power for the instrumentation as well as component heating.

Auxiliary Equipment. The following auxiliary equipment is available:

- Two Ampex 1800 L, 14 track, tape decks are used to gather the data. This data are then printed out by employing the IBM 360 computer.
- After removal from TREAT, the sodium system is placed in a storage pit in the TREAT reactor building. After radiation levels decay, cleanup of the salvageable portion is effected at the EBR-II facility. The test section and U-tube are cut up and shipped to Argonne-East for posttest examination.
- The neutron radiography facilities, located at TREAT, are employed to examine the apparatus before and after the experiment.
- The Hodoscope is used to examine fuel motion during the experiment.

TABLE I. GENERAL PARAMETERS

FUEL	UO_2
FUEL PIN GEOMETRY	CURRENT FFTF-TYPE PINS
NO. FUEL PINS	UP TO 7 PINS
COOLANT	SODIUM
VOLUME OF COOLANT	UP TO 7 CU FT
STRUCTURAL MATERIAL	TYPE 316 SS
GAS IN TANKS	HE
GAS IN SECONDARY HOUSING	N_2
THERMAL HYDRAULICS	CAPABLE OF SIMULATING FFTF STEADY-STATE AND FLOW-COASTDOWN CONDITIONS

TABLE II. EXPERIMENTAL PARAMETERS
(FROM FIRST IN-PILE EXPERIMENT)

7-PIN BUNDLE	
LENGTH OF HEXAGONAL CAN	8 FT X 11-5/16 IN.
WALL THICKNESS OF HEXAGONAL CAN	0.020 IN.
HEXAGONAL CROSS SECTION ACROSS INSIDE FLATS	0.796 IN.
HEXAGONAL CROSS SECTION ACROSS INSIDE SHARPS	0.922 IN.
TOTAL CROSS SECTION AREA (INSIDE CAN)	0.543 SQ IN.
TOTAL CROSS SECTION AREA (FUEL PINS)	0.291 SQ IN.
TOTAL CROSS SECTION AREA (WIRE WRAP)	0.0071 SQ IN.
TOTAL CROSS SECTION AREA (FILLER WIRE)	0.0276 SQ IN.
COOLANT FLOW AREA	0.217 SQ IN.
WETTED PERIMETER–CAN	2.766 IN.
WETTED PERIMETER–PINS	4.336 IN.
WETTED PERIMETER–WIRE WRAP	0.773 IN.
WETTED PERIMETER–FILLER WIRE	1.169 IN.
HYDRAULIC DIAMETER	0.127 IN.
COOLANT/BUNDLE AREA RATIO	0.400
COOLANT/PIN AREA RATIO	0.746
COOLANT/(COOLANT AND PIN) AREA RATIO	0.427
FUEL-COOLANT MASS RATIO (MASS FUEL)/(MASS COOLANT)	11.8
FUEL-COOLANT VOLUME RATIO (VOLUME FUEL)/ VOLUME COOLANT)	0.95
THERMAL HYDRAULICS (AT STEADY STATE)	
AVERAGE LINEAR POWER IN FUEL PINS	10.0 KW/FT
MAXIMUM LINEAR POWER IN FUEL PINS	11.4 KW/FT
MAXIMUM LINEAR POWER IN FUEL PINS WITH O.P.F. = 1.2	13.7 KW/FT
AXIAL MAXIMUM/AVERAGE	1.14
RADIAL MAXIMUM/AVERAGE	1.10
SODIUM INLET TEMPERATURE	600°F
SODIUM OUTLET TEMPERATURE	900°F
7–PIN BUNDLE ΔT	300°F
SODIUM MASS FLOW RATE	2 LB/SEC
AVERAGE SODIUM VELOCITY	25 FPS
TOTAL SUBASSEMBLY PRESSURE DROP	110 PSI
DOWNCOMER PIPING PRESSURE DROP	50 PSI
PRESSURE DROP ACROSS FUEL PINS	55 PSI
DISCHARGE PIPING PRESSURE DROP	5 PSI
PRESSURE IN DRIVER TANK	125 PSIA
PRESSURE IN RECEIVER TANK	15 PSIA

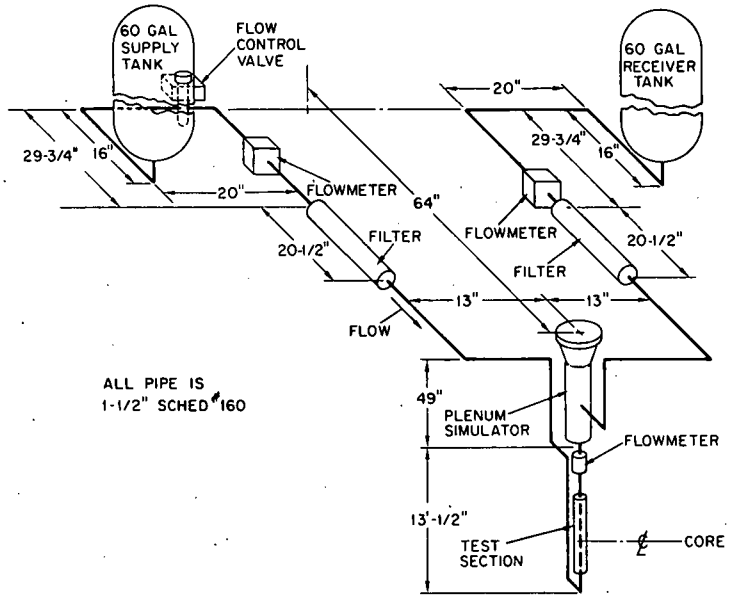

Figure 1. Diagram of Test Area.

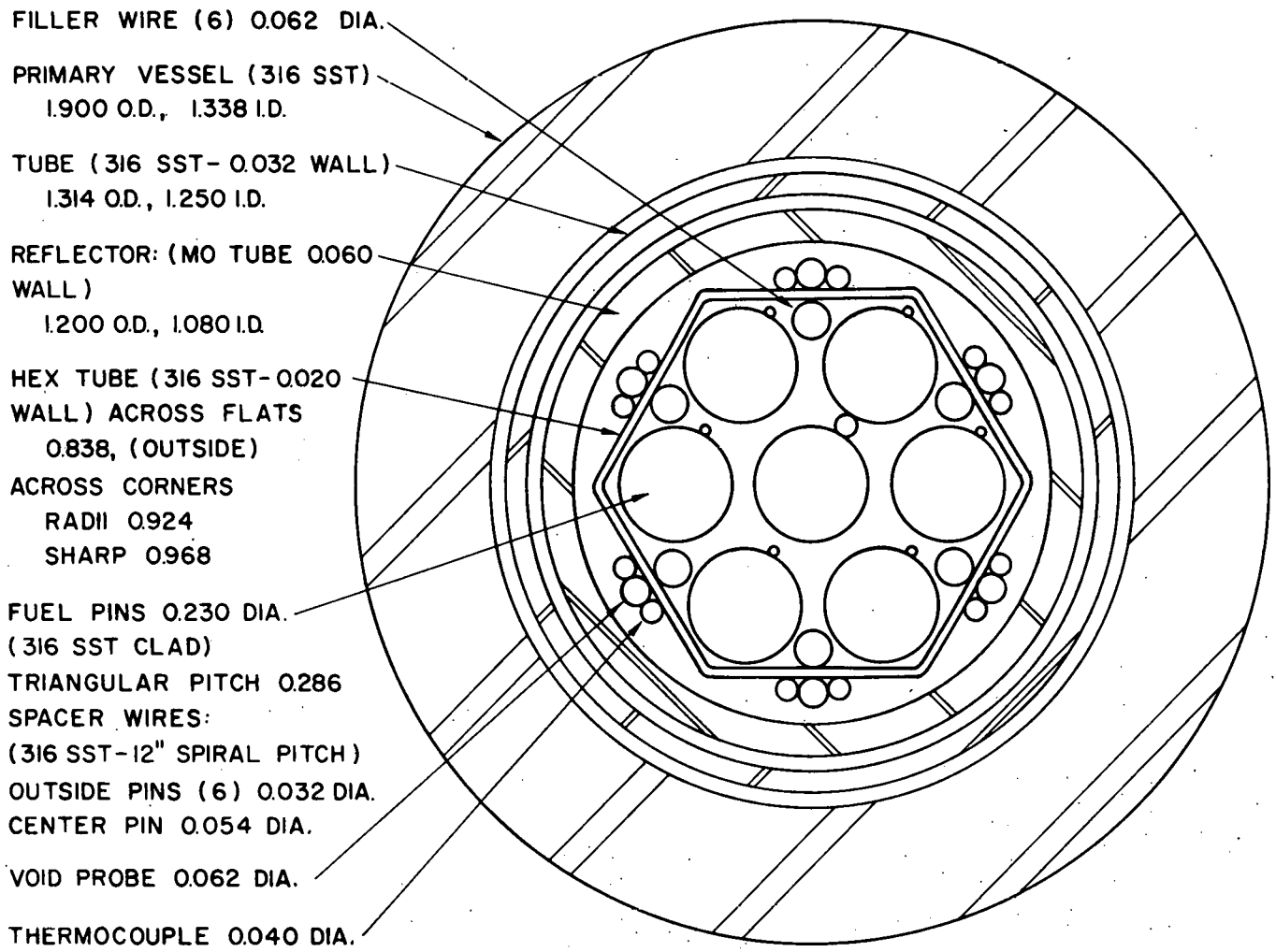

Figure 2. Schematic of R-Series Facility.

TREAT MARK-IIC INTEGRAL LOOP

Location ANL-W, INEL, Idaho

Status Operational

Principal Uses. The Mark-IIC Integral Loop is used for testing LMFBR-type oxide-fuel elements in the TREAT Facility under transient conditions of coolant flow, coolant temperature, and fuel element power represenative of those that may occur in a reactor.

It is designed to contain the high-amplitude pressure pulses which might be generated by the failure of refractory fuel elements with melting points well above 2000°C. The loop in its entirety is designed to function properly under steady state conditions at the operating temperatures, coolant flow, and fuel element power typical of fast power reactors. The flow of sodium in the Mark-IIC Loop is adequate to simulate thermal hydraulic conditions in typical LMFBR oxide-fueled designs. The loop provides for:

- The establishment of liquid metal coolant (sodium) flow and temperature values representative of LMFBR conditions.
- Control of coolant flow and temperature and for variation of flow during an experiment.
- Containment of the coolant, nuclear fuel including plutonium, and fission products before, during, and after an experiment.
- Alteration and shaping of the TREAT reactor neutron flux in the vicinity of the loop such that representative power distributions are obtained in the fuel elements.
- Monitoring of flow, pressure, and temperatures of coolant and structure as required for experimental and operational purposes.
- Remote or semi-remote handling at the reactor and/or a hot cell.

Facility Description. A complete Mark-IIC Loop, illustrated in Figure 1, is composed of three major systems contained within a secondary containment:

- The primary containment - consisting of a loop body weldment (LBW) and the pump tube of the annular linear induction pumps (ALIP). The LBW consists of a test section, closure tube, and connecting piping. The circulatory portion of the primary containment loop is located in two long parallel straight flow channels connected at top and bottom by short radius bends. The sodium pump is in one leg of the elongated circuit; and the test section is located in the opposite leg, with access to the loop interior provided by an axial extension (closure tube) of the test section above the upper, or return bend of the circuit.
- The auxiliary containment - consisting of the dual purpose dump/safety tank system and piping which must contain the sodium released from the rupture disc following an over-pressure of the LBW.
- The ancillary equipment - consisting of such items as electrical heaters, a cooling oil system for the ALIP, an inert gas system, thermal insulation, and instrument outfitting (for monitoring the test reactors and loop components).

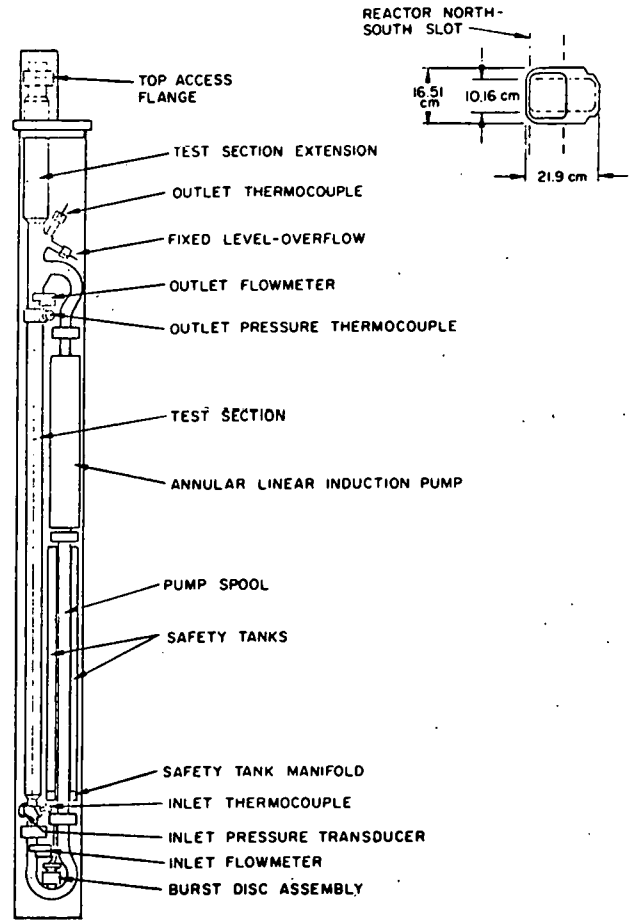

Figure 1. Mark-IIC Integral Loop.

When fully assembled and prepared for experimental use, the Mark-IIC Integral Loop presents the appearance of a long, slender stainless steel enclosure of rectangular cross-section, capped at one end by a thick steel plate which bears a smaller, approximately rectangular enclosure, a variety of quick connect valves and two multicontact electrical through-connectors. Supported by its lifting bridle in the vertical position, as for lowering into the TREAT core, the Mark—IIC Loop facility stands 257.86 cm (101.125 in.) high, from base to top plate, with the top plate and access/closure protective hood adding 25.25 cm (9.94 in.), for a total overall length of 283.11 cm (111.065 in.).

The loop is positioned within the TREAT reactor such that the test section penetrates into the reactor core region. The apparatus arrangement is such that no portion of the loop becomes a permanent or integral part of the reactor itself. Therefore, the Mark-IIC Loop does not affect reactor operation or control once routine provisions are made to compensate for the removal of two reactor fuel elements

Sodium is circulated within the loop by a miniaturize annular linear induction pump and controlled by regulating the voltage applied to the pump coils. The Mark-IIC Loop is

TABLE I. LOOP PARAMETERS

PRIMARY CONTAINMENT	
NORMAL OPERATING TEMPERATURE (BEFORE REACTOR PULSE)	800°F
NORMAL OPERATING PRESSURE (BEFORE REACTOR PULSE)	50 PSIG
ABNORMAL OPERATING TEMPERATURE (AT REACTOR PULSE)	1000°F
ABNORMAL OPERATING PRESSURE (AT REACTOR PULSE)	<5000 PSIG
DESIGN TEMPERATURE	1000°F
DESIGN PRESSURE	6250 PSIG
TEST TEMPERATURE	1000°F
TEST PRESSURE	6250 PSIG
VOLUME	
SODIUM FILL (FUEL INCL.)	APPROX. 2234 CU CM
CLOSURE TUBE, GAS	APPROX. 971 CU CM
TOTAL PRIMARY CONTAINMENT	APPROX. 3205 CU CM
MATERIAL	
LOOP BODY WELDMENT	TYPE 316 SS
ALIP PUMP TUBE	INCONEL GRADE 718
ANNULAR LINEAR INDUCTION PUMP	
FLOW RATE	20 GPM
AUXILIARY CONTAINMENT	
DESIGN TEMPERATURE	1000°F
DESIGN PRESSURE	6250 PSIG
TEST TEMPERATURE	1000°F
TEST PRESSURE	6250 PSIG
DUMP/SAFETY TANK SYSTEM	
NO. OF TANKS	3
TOTAL VOLUME OF 3 TANKS	2075 CU CM

capable of operating with liquid sodium velocities of up to 5 m/sec (~20 gpm), temperatures of 1000°F, and pressures as high as 5000 psig.

Test Capability. The Mark-IIC Loop is designed to be accommodated within the volume of two TREAT fuel elements in the reactor. The test section is capable of accommodating as many as 7 prototypical LMFBR fuel pins up to 61 in. in length with the fuel column approximately centered on the horizontal centerline of the TREAT core. The length is sufficient to accommodate the longest pins under irradiation in EBR-II. The 8-ft vertical length and 4 X 8 in. cross section of two elements accommodate an elongated miniature loop of two straight parallel legs, the downcomer with the liquid metal pump on one side and the riser test section on the other side of the shell. Access to the fuel-containing test section is provided by adding an extension beyond the return bend of the loop and on the centerline of the test section. The extension also provides a plenum for expansion volume. The loop features a simplified geometry offering the immediate advantage of providing the fast neutron hodoscope with a relatively uncluttered view of the fuel-containing test section, while the loop is positioned in the reactor with the test section leg in the hodoscope viewing slot. A direct straight-through sodium flow is provided to reduce hydraulic losses, to assure greater ease in instrumenting both the inlet and outlet of the fueled test section, and to facilitate the assembly and disassembly of successive experiments by remote handling.

n experimental use, the Mark-IIC Loop permits ease of fuel oading and unloading, whether the operations are erformed in- or out-of-cell. In-cell operations on the loop are performed in the HFEF at the EBR-II reactor site.

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ZERO POWER PLUTONIUM REACTOR (ZPPR)

Location ANL–W, INEL, Idaho

Status Operational

Principal Uses. The facility is used to obtain integral reactor physics parameters from fast spectrum critical assemblies such as benchmark or engineering mockup experiments.

Facility Description. ZPPR is a split-table-type critical facility designed for the study of the physics of power breeder systems. The ZPPR matrix is made up of two 10 X 10 X 4 ft assemblies of matrix tubes mounted on separate steel tables. The tables are separated while the matrix is loaded with drawers containing reactor simulated materials in the form of small plates and then brought together for operation.

Test Capability. ZPPR is capable of simulating fast reactor compositions characteristic of the proposed 300-500 MWe demonstration plants and 1000 MWe commercial plants. ZPPR can be used to simulate many reactor designs to cores ranging up to about 3000 liters. Experiments include studies of the power-producing, control and safety characteristics of large plutonium-breeder power-reactor compositions.

Auxiliary Equipment.

- Sufficient quantity of plutonium, coolant and structural materials to construct metal, oxide or carbide cores.
- Radial and axial, remote-positioning equipment for measuring material worths or reaction rates as a function of core position.
- A mechanism for oscillating heated uranium and plutonium samples to permit investigation of the Doppler coefficients of these large cores.
- An automated, high-volume, foil-counting facility, employing primarily high-resolution gamma-ray spectroscopy, where activated foils are counted to determine fission and capture distributions throughout the reactor.
- An on-line, SEL-840-MP computer provides capability for data acquisition and immediate analysis.

Modifications Authorized. The present matrix size permits studies of the demonstration plants (~3000 liters). The reactor tables have sufficient dimensional and weight capabilities to each accommodate a 14 X 14 ft matrix. This "stretch capability" will permit the study of cores exceeding 6000 liters in volume. The expansion, authorized in FY 1975, will be completed in FY 1978.

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Facility Parameters

Reactor matrix size	10 X 10 X 8 ft
Reactor matrix tubes	2.273 X 2.175 in.
Primary reactor inventory	Pu, U, U_3O_8, stainless steel and sodium
Cell diameter	50 ft
Cell height	~21 ft
Cell entrance	6 ft-8 in. high X 4 ft wide
Reactor cooling	Air
Reactor temperature	Room temperature ~68°F
Neutron and gamma flux	Dependent on size, composition and operating power of assembly
Neutron and gamma spectra	Dependent on size and composition of assembly
Experimental loop capability	2 in. X 2 in. X variable length
Loop external wall temperature	Approximately room temperature
Loop availability	Any matrix position (1000 core or blanket positions)

Figure 1. View of ZPPR with Reactor Halves Separated.

The following acronyms and letter designations are in common usage to describe organizations, reactor and test facilities, components, etc., involved in the LMFBR program. In certain cases, parenthetical notes have been added to the definitions to indicate the locations or operators of facilities.

ACRP	Advisory Committee on Reactor Physics
ACRS	Advisory Committee on Reactor Safeguards
A-E	Architect-Engineer
AEC	Atomic Energy Commission (now ERDA and NRC)
AECL	Atomic Energy of Canada Limited
AETR	Advanced Epithermal Thorium Reactor
AFSR	Argonne Fast Source Reactor (ANL-W/INEL)
AGC	Aerojet General Corporation
AGHCF	Alpha-Gamma Hot Cell Facility (ANL-E)
AGR	Advanced Gas-cooled Reactor
AI	Atomics International Division, Rockwell International
AIF	Atomic Industrial Forum
AMCF	Alkali Metal Cleaning Facility (HEDL)
AMPS	Apparatus for Monitoring and Purifying Sodium (ANL-E)
ANC	Aerojet Nuclear Company
ANL	Argonne National Laboratory
ANL-E	Argonne National Laboratory - Illinois (East)
ANL-W	Argonne National Laboratory - Idaho (West)
ANS	American Nuclear Society
APDA	Atomic Power Development Associates
ARD	Advanced Reactors Division, Westinghouse Electric Corporation
ARCO	Atlantic Richfield Company
ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor (ANC/INEL)
ATSR	Argonne Thermal Source Reactor (ANL)
BFS-1	Brookhaven Fast Source Reactor (BNL)
BLTC	Bottom Loading Transfer Cask
BMI	Battelle Memorial Institute
BNL	Brookhaven National Laboratory

BNWL	Battelle Northwest Laboratories
BOM	Bureau of Mines (U. S. Dept. of Interior)
BRC	Breeder Reactor Corporation
B&W	Babcock and Wilcox Company
BWR	Boiling Water Reactor
CBR	Commercial Breeder Reactor
CCTC	Core Component Transfer Cask - FFTF
CCTL	Core Component Test Loop (ANL-E)
CDA	Core Disruptive Accident
CDS	Component Disassembly Station (HEDL)
CE	Combustion Engineering, Inc.
CEA	Commissariat a l'Energie Atomique (France)
CEL	Carbon Equilibrium Loop (WARD)
CEN	Centre d'Etudes Nucleaires (France)
CERN	European Organization for Nuclear Research
CESNEF	Centro Studi Nucleari 'Enrico Fermi'
CFR	Commercial Fast Reactor (United Kingdom)
CFRMF	Coupled Fast Reactivity Measurements Facility (ANC/INEL)
CHCF	Component Handling and Cleaning Facility (LMEC)
CLEM	Closed Loop Ex-Vessel Handling Machine - FFTF
CLIRA	Closed Loop In-Reactor Assembly - FFTF
CMM	Core Mechanical Mockup (HEDL)
CNEN	Comitato Nazionale per l'Energia Nucleare
CP-5	Chicago Pile No. 5 (ANL-E)
CRBR	Clinch River Breeder Reactor
CRCTA	Composite Reactor Component Test Activity (HEDL)
CRDM	Control Rod Drive Mechanism
CRM	Core Restraint Mechanism
CRTF	Creep Ratchetting Test Facility (WARD)
CSDD	Conceptual System Design Description
DAS	Data Acquisition System - EBR-II
DBA	Design Basis Accident
DF	Decontamination Factor
DFR	Dounreay Fast Reactor (United Kingdom)

DHX	Dump Heat Exchanger
DTRS	Development Test Requirement Specification
EACRP	European-American Committee on Reactor Physics
EANDC	European-American Nuclear Data Committee
EBR-I & -II	Experimental Breeder Reactors Nos. 1 and 2 (ANL-W/INEL)
ECCS	Emergency Core Cooling System
ECEL	Epithermal Critical Experimental Laboratory
EDF	Electricite de France
EDL	Engineering Development Laboratory (HEDL)
EEI	Edison Electric Institute
EMC	Engineering Mockup Critical Experiment
ENDF	Evaluated Nuclear Data File
ENEA	European Nuclear Energy Agency
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
ERDA	Energy Research and Development Administration
ETR	Engineering Test Reactor (ANC/INEL)
EURATOM	European Atomic Energy Community
EURFNR	U. S.-EURATOM Fast Reactor Exchange Program
EVHM	Ex-Vessel Handling Machine
FBR	Fast Breeder Reactor
FCAF	Fuel Cycle Alpha Facility (ORNL)
FCF	Fuel Cycle Facility (now HFEF/S)
FCI	Fuel-Coolant Interaction
FEDAL	Failed Element Detection and Location System
FEF	Fuel Examination Facility (now HFEF/S)
FEFPL	Fuel Element Failure Propagation Loop (now SLSF)
FFM	Fuel Failure Mockup (ORNL)
FFTF	Fast Flux Test Facility (HEDL)
FIVTF	Flow-Induced Vibration Test Facility (ANL-E)
FMEF	Fuels and Materials Evaluation Facility
FNAL	Fermi National Accelerator Laboratory
FNG	Fast Neutron Generator (ANL-E)
FOTA	Fuels Open Test Assembly - FFTF

FRPP	Fuel Reprocessing Pilot Plant (formerly HPP)
FSAR	Final Safety Analysis Report
FSDD	Final System Design Description
FSF	Fuel Storage Facility (HEDL)
FTR	Fast Test Reactor (now FFTF)
FWC	Foster-Wheeler Corporation
GA	General Atomics (Gulf/Shell)
GCFR	Gas-Cooled Fast Reactor
GE-NED	General Electric Company Nuclear Engineering Division
GETR	General Electric Test Reactor (GE)
GLASS	Germanium-Lithium Argon Scanning System - EBR-II
GPL	General Purpose Loop (WARD)
GRT	Gulf Radiation Technology, Inc.
GTWA	New Designation for TIG Welding
HAZ	Heat Affected Zone
HCDA	Hypothetical Core Disruptive Accident
HCM	Hydraulic Core Mockup (HEDL)
HEDL	Hanford Engineering Development Laboratory
HFEF	Hot Fuel Examination Facility (ANL-W/INEL)
HFEF/N	HFEF-North
HFEF/S	HFEF-South (formerly FCF and FEF)
HFIR	High Flux Isotope Reactor (ORNL)
HNL	Holifield National Laboratory (now ORNL)
HNPF	Hallam Nuclear Power Facility
HPFL	High Performance Fuel Laboratory
HPP	Hot Pilot Plant (reprocessing) (now FRPP)
HRI	Houston Research Institute
HRLAL	High Radiation Level Analytical Laboratory
HRLEL	High Radiation Level Examination Laboratory (ORNL)
HTGR	High Temperature Gas-cooled Reactor
HTL-1	Heat Transfer Loop No. 1 (HEDL)
HT-LMMHD	High-Temperature Liquid-Metal MHD Facility (ANL-E)
HTS	Heat Transport System
HTSL	LMFBR Heat Transfer Simulation Loop (ANL-E)

HTSF	High Temperature Sodium Facility (HEDL)
HVEM	High Voltage Electron Microscope (HEDL)
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IEMTF	Interim Examination and Maintenance Training Facility (HEDL)
IHX	Intermediate Heat Exchanger
INC	Idaho Nuclear Corporation
INCOT	In-Core Instrument Test Facility - EBR-II
INEL	Idaho National Engineering Laboratory (formerly NRTS)
INSAT	Instrumented Subassembly Test Facility - EBR-II
INTERATOM	Internationale Atomreaktorbau gmbH
IRFM	Integral Reactor Flow Model (HEDL)
ISML	Intermediate System Mockup Loop (GE)
IT	Instrument Tree
ITF	Interstitial Transfer Facility (WARD)
ITR	Instrument Test Rig (LMEC)
IVHM	In-Vessel Handling Machine
JAERI	Japan Atomic Energy Research Institute
JAEC	Japanese Atomic Energy Commission
JAPFNR	U. S.-Japan Fast Reactor Exchange Program
JCAE	Joint Committee on Atomic Energy (U. S. Congress)
KAPL	Knolls Atomic Power Laboratory
KNK	Kompaktes Natriumgekuhlte Kernreaktoranlage (Compact Sodium-cooled Nuclear-reactor-plant - Germany)
LAMPRE	Los Alamos Molten Plutonium Reactor Experiment
LASL	Los Alamos Scientific Laboratory
LCDM	LMFBR Components Development (ANL-CTD)
LINAC	Linear Accelerator
LLTR	Large Leak Test Rig (LMEC)
LLTV	Large Leak Test Vessel
LMEC	Liquid Metals Engineering Center
LMFBR	Liquid Metal Fast Breeder Reactor

LOFT	Loss-of-Flow Test (INEL)
LRL	Lawrence Radiation Laboratory
LSFF	Large Sodium Fire Facility (HEDL)
LTSSS	Low Temperature Sodium Supply System (AI)
LTV	Large Test Vessel (AI)
LWR	Light Water Reactor
MCTF	Mixing Components Test Facility (ANL-E)
MLTF	Multiloading Test Facility (WARD)
MOTA	Materials Open Test Assembly - FFTF
MPHL	Multi-Purpose Hydraulic Loop (WARD)
MPS	Mechanical Property System (WARD)
MSAR	Mine Safety Appliances Research Corporation
MSBR	Molten Salt Breeder Reactor
MSG	Modular Steam Generator
MSRE	Molten Salt Reactor Experiment (ORNL)
MTI	Mechanical Technology, Inc.
MTL	Materials Test Loop (WARD)
MTR	Materials Testing Reactor (ANC/INEL)
MUST	Mechanisms Under Sodium Test Facility (WARD)
NACF	Sodium Cleaning Facility (WARD)
NAL	National Accelerator Laboratory (now FNAL)
NASA	National Aeronautics and Space Administration
NCBR	Near-Commercial Breeder Reactor
NCSAG	Neutron Cross-Section Advisory Group
NFS	Nuclear Fuel Services, Inc.
NITF	Nuclear Instrument Test Facility - EBR-II
NMDF	Nuclear Materials Development Facility (AI)
NRC	Nuclear Regulatory Commission
NRF	Neutron Radiography Facility (HEDL)
NRL	Naval Research Laboratory
NRTS	National Reactor Testing Station (now INEL)
NUFTIC	Nuclear Fuels Technology Information Center (ORNL)
OPERA	Out-of-Pile Expulsion and Reentry Apparatus (ANL-E)

ORELA	Oak Ridge Electron Linear Accelerator (ORNL)
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Research Reactor (ORNL)
PAL	Prototype Applications Loop (HEDL)
PBF	Power Burst Facility (ANC/INEL)
PBTF	Pump Bearing Test Facility (LMEC)
PCML	Plutonium Critical Mass Laboratory (PNL)
PCTF	Plant Component Test Facility
PEC	Prova Elementi di Combustible (Fuel Element Test Reactor - Italy)
PFR	Prototype Fast Reactor (United Kingdom)
PIOTA	Post-Irradiation Open Test Assembly - FFTF
PLBR	Prototype Large Breeder Reactor
PMC	Project Management Corporation
PMIS	Precision Mechanisms in Sodium Loop (LMEC)
PNL	Pacific Northwest Laboratories (Battelle)
PRDC	Power Reactor Development Company
PSAR	Preliminary Safety Analysis Report
PSDD	Preliminary System Design Description
PSTF	Pump Seal Test Facility (LMEC)
PWR	Pressurized Water Reactor
R, D&D	Research, Development and Demonstration
R&D	Research and Development
RAPTUS	Rapido Toria Uranio Sodiu
RAREF	Radiation and Repair Engineering Facility
RDD	Division of Reactor Development and Demonstration, USERDA
RDT	Division of Reactor Development and Technology, USAEC (now RDD)
RFP	Request for Proposal
RL	Richland Operations Office - ERDA
RM	Reactor Manufacturer
RML	Radiometallurgy Laboratory (HEDL)
RMF	Reactivity Measurement Facility
RRD	Division of Reactor Research and Development, ERDA (now RRD)
RSCL	Radioactive Sodium Chemistry Loop - EBR-II
RV	Reactor Vessel

S/A	Subassembly
SACCT	Single Assembly Cooling Cask Test (ORNL)
SAD	Safety Assurance Diagram
SAEA	Southwest Atomic Energy Associates
SAR	Safety Analysis Report
SAREF	Safety Research Experiment Facility
EBR-II SRM	Experimental Breeder Reactor-II Safety Research Modifications
SF	Support Facilities
STF	Safety Test Facility
TU	TREAT Upgrade
SCCF	Sodium Components Cleanup Facility - EBR-II
SCEL	Small Components Evaluation Loop (HEDL)
SCTF	Shutdown Component Test Facility (GE and WARD)
SCTI	Sodium Components Test Installation (LMEC)
SCTL	Small Components Test Loop (LMEC)
SDD	System Design Description
SDTL	Structural Dynamics Test Loop (ANL-E)
SEFOR	Southwest Experimental Fast Oxide Reactor
SELNI	Societe Elettronucleare Italiana
SERF	Special Environment Radiometallurgy Facility (HEDL)
SFCF	Sodium Flowmeter Calibration Facility (ANL-E)
SG	Steam Generator
SGR	Sodium-Graphite Reactor
SGTF	Steam Generator Test Facility (ANL-E)
SGTR	Steam Generator Test Rig (GE)
SIR	Submarine Intermediate Reactor
SLSF	Sodium Loop Safety Facility (formerly FEFPL) (ANL-W/INEL)
SMF	Shielded Materials Facility (HEDL)
SNPC	Struthers Nuclear and Process Company
SNR	Schnell Natriumgekuhlt Reaktor (Fast Sodium-cooled Reactor - Germany)
SOWAT	Small Leak Sodium-Water Reaction Test Rig (GE)
SPTF	Sodium Pump Test Facility (LMEC)
SRDA	Sodium Removal Development Apparatus (HEDL)
SRE	Sodium Reactor Experiment
SRI	Stanford Research Institute

STCL	Source Term Control Loop (HEDL)
STF	Safety Test Facility (see SAREF)
SV/CT	Small Valve and Component Test Facility (LMEC)
SWRI	Southwest Research Institute
SWEATER	Sodium Friction and Wear Test Rig (WARD)
SWL	Steam/Water Loop (WARD)
SWS	Self-Welding System (WARD)
TIG	Tungsten Inert Gas (welding technique - now designated GTWA)
TRFAT	Transient Reactor Test Facility (ANL-W/INEL)
TRU	Transuranium Processing Plant (ORNL)
TSF	Tower Shielding Facility (ORNL)
TTF	Transient Test Facility (LMEC)
TTL	Transient Test Loop (HEDL)
TTTF	Thermal Transient Test Facility (ORNL)
TURF	Thorium-Uranium Recycle Facility (ORNL)
TVA	Tennessee Valley Authority
UKAEA	United Kingdom Atomic Energy Authority
UNC	United Nuclear Corporation
VTRZ	Variable Temperature Rodded Zone
W	Westinghouse Electric Corporation
WARD	Westinghouse Advanced Reactors Division
WHC	Westinghouse-Hanford Company
ZPPR	Zero Power Plutonium Reactor (ANL-W/INEL)
ZPR	Zero Power Reactor