RECEIVED BY DTIC NOV 6 1968

CONF-681017--6

MASTER

BNWL-SA-1826

FAST FLUX TEST FACILITY INSTRUMENTATION REQUIREMENTS ^(a)

C. D. Swanson ^(b)

Battelle Memorial Institute
Pacific Northwest Laboratory
Richland, Washington

September 1968

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Paper for presentation at the IEEE 15th Nuclear Science Symposium to be held in Montreal, Canada October 23-25, 1968.

- (a) This paper is based on work performed under United States Atomic Energy Commission Contract AT(45-1)-1830.
- (b) Manager; Instrumentation, Control and Electrical Section; FFTF Project.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

fly

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

FAST FLUX TEST FACILITY INSTRUMENTATION REQUIREMENTSABSTRACT

The Fast Flux Test Facility (FFTF) is the major irradiation test facility planned for the Liquid Metal Fast Breeder Reactor (LMFBR) Program. This paper discusses the facility instrumentation requirements, placing particular emphasis on out-of-core instrumentation. In-core instrumentation requirements are described in more general terms (due to their greater dependence on reactor concept decisions).

Out-of-reactor instrumentation requirements included in these discussions are: sodium and inert gas purity, sodium heat transport system measurements, fuel handling and examination, radiation monitoring, containment actuation and process control problem areas. Considerations leading to provisions for a multiple digital computer-based data handling and display system are described.

INTRODUCTION

The overall objective of the Fast Flux Test Facility (FFTF) is to meet the fast flux testing requirements of the U.S. AEC's Liquid Metal Fast Breeder Reactor (LMFBR) Program and, in addition, to be responsive to the requirements of other AEC development programs. In meeting these requirements the primary function of the FFTF is to provide reliable closed loop^{*} testing of fuels, materials and sensors under closely controlled and instrumented environmental conditions. Broad goals of the FFTF may be summarized as follows:

^{*} See Appendix for definition of test facility terms.

- Accommodate a wide range of test objects in terms of number, size and physical characteristics.
- Control environmental conditions for tests.
- Monitor the environment and the test objects to provide data for determination of cause and effect relationships established by the tests.
- Add to the technical base of the LMFBR program by using components prototypic of the future fast breeder power reactors.

To better assure that future testing requirements would be identified and used as a basis for design, a series of studies by potential user organizations was initiated by Battelle-Northwest. The results of these studies indicate that a broad range of test requirements exists. Some stated requirements are judged to be impractical to meet in the initial facility because of constraints affecting the number, type and capability of test facility which can be provided. These constraints include:

- Technological state-of-the-art.
- Funding availability.
- Safeguards limitations.

PNL has selected a reference set of initial test facility requirements. However, it is recognized that changes in technology and in development program objectives can greatly alter the type and emphasis of future fast flux testing requirements. Therefore, a fundamental FFTF design objective is to provide a flexible facility that can be adapted to provide many different types of testing.

An efficient test facility must provide adequate means of monitoring the test objects and their environment to provide data for the significant cause and effect relationships established by the tests. The design of the FFTF differs from the design of conventional power plants in its requirements for more extensive instrumentation and the need to consider the sensor and transmission media space and access requirements for proposed experimental activities. In addition, space and access must be provided for instrumentation which may be developed in the foreseeable future.

Present test facilities do not provide for instrumentation in fast flux environments even approaching FFTF and future LMFBR levels. Therefore, the ability to develop and test measurement systems to meet LMFBR program requirements is severely handicapped. It may not be possible to predict with confidence what the accuracy and life expectancy of initial FFTF in-core sensors will be. It can be seen that a major objective of early FFTF testing must be to establish this technology if effective fuels and materials testing is to be accomplished. That is, development of sensing and data transmission (cables, multiplexed transmission, etc.) techniques may have a priority comparable to fuels and materials development and testing. The relative ability of various reactor concepts to provide adequate access and space for instrumentation has been a major tradeoff consideration in the evaluation of proposed FFTF concepts.

At the time preparation of this paper was proposed, it was expected that the FFTF conceptual design phase would be complete and that rather definitive instrumentation requirements could be stated. However, additional reactor concept evaluation has been required. Therefore, the discussion of reactor

instrumentation requirements in this paper will be general in nature and less specific to a given concept.

Summary Description of the FFTF

The principle component of the facility is the Fast Test Reactor (FTR). It will be designed to meet the process and handling requirements of multiple closed test loops.* The core is formed by an array of subassemblies containing a combination of driver fuel*, experimental test and control rod positions. The specific core concept has not been selected. Typical of the concepts which have been considered are the skewed, vertical and split cone designs illustrated in Figures 1, 2 and 4.

The initial core has a volume of approximately 1,000 liters and has about 100 lattice positions. The core, reflector, experimental facility and various support and hydraulic components are housed in a stainless steel vessel with a cover providing access to the driver and experimental positions via reactor cover nozzles. These provide access for refueling and housing for instrument connectors. Additional design data are given below in Table I.

* See Appendix for definition of test facility terms.

Table 1. FFTF Design Data Summary

Peak Flux	$>7 \times 10^{15}$ nv
Total Power	400 MW _t
Core Outlet Temperature	1200°F (900)
Core Delta Temperature	(350)°F
Number of Closed Loops	6(5)
Number of Open Test Positions	(11)
Number of Core Lattice Positions	≈ 100
Number of Control and Safety Rods	24
Active Core Height	32-36 in.
Fuel Pin Diameter	0.23 in.
Number of Pins/Subassembly	217
Subassembly Cross Section Dimension	Hexagon, 4-5 in. across flats

() indicates initial operating conditions.

Sodium coolant transports heat from the core in several independent parallel circuits. The current concept uses three independent parallel circuits with three primary and three secondary sodium loops and associated air heat dump exchangers. Primary sodium from the reactor outlet passes from shell and tube intermediate heat exchangers and is pumped back to the inlet by free surface centrifugal pumps. The secondary sodium receives the thermal energy in the IHX's and is pumped through the forced air heat dumps and back to the IHX by the secondary pumps.

Reactor refueling operations are accomplished by a shielded refueling machine above the reactor. The fuel handling equipment in the work space, performs operations needed in preparation for refueling as well as the transfer of fuel and experiments into and out of the reactor. Fuel handling equipment is capable of maintaining required test temperatures and chemistries during transit to and from the reactor. The fuel handling equipment has special capabilities for preserving the essential test information. Figure 3 shows a conceptual arrangement of the facility.

The closed loop test facilities represent the most important capabilities of the FFTF. Each multi-megawatt facility will provide isolated recirculating coolant to a test channel in the reactor in which specimens are irradiated. The isolated coolant circuits allow variations of coolant chemistry, temperature and pressure, and the use of coolants other than sodium. Provision will be made to install packaged closed loops in reflector or core test positions for low power or materials tests.

Open test positions^{*} will use the primary sodium for cooling test specimens and testing flexibility in these positions will correspondingly be limited. Both closed and open test positions will have the capability for test specimen contact instrumentation. Short-term tests will be accommodated in peripheral core positions. These may take the form of rabbit or trail cable facilities and/or axial positioners.^{*}

Facilities will be provided for pre-irradiation inspection, flow testing, nuclear proof testing and interim and post-irradiation examination of test specimens and subassemblies. Examination capabilities will include

* See Appendix for definition of test facility terms.

visual observations, dimensional checks and radiography. Shielded inert gas-filled cells will be provided for examination of reactive fuels and materials specimens. These facilities are designed to retrieve the maximum amount of test information. Equipment is provided for transfer positioning and detailed examination of the specimens.

Facilities will be provided to purify sodium as required for the reactor, closed loops and the fuel storage containers. Sodium purification equipment will be capable of maintaining reactor or loop coolant impurities--oxygen, carbon, nitrogen and hydrogen to below the concentrations required to protect the stainless steel, refractory, super alloy or other cladding and structural materials.

During the first stage of reactor operations, the primary coolant chemistry will be controlled by cold trapping the impurities from the coolant stream. Oxygen concentrations in the sodium coolant stream may be controlled to less than 10 parts per million by diverting part of the primary coolant stream through a cold trap purification unit. Space and connections for oxygen and carbon hot traps will be provided. These hot traps will be installed as required for reactor operation or by testing needs and are expected to control carbon to less than 20 parts per million and oxygen to less than five parts per million.

A gas purification system will control reactor primary system cover gas chemistry impurities consistent with open test position coolant requirements.

The chemistry of the primary sodium in the closed loops will be controlled independently of the primary sodium in the reactor coolant loops. Coolant instrumentation and equipment will be provided for control

and/or monitoring the cooling environment for the inserted test specimens, capsules, rods or subassemblies.

Sample stations and laboratory facilities will be provided for frequent sampling and rapid chemical analyses of sodium and gas samples.

Operation of the reactor, test facilities and auxiliary support systems is under the surveillance and control of a centralized data handling and control system. The reactor system operations, including driver fuel and experimental facilities, are monitored by continuous on-line computers.

REACTOR INSTRUMENTATION REQUIREMENTS

FFTF reactor instrumentation requirements may be divided into five categories for convenience:

- Reactor core instrumentation
- Reactor bulk process instrumentation
- Reactor vessel and shield instrumentation
- Flux monitoring instrumentation
- Failed fuel element detection and location instrumentation.

Each of these categories will be discussed below. Reactor measurements are needed for three purposes: plant operation, experimental data and plant protection. Protective instrumentation must be kept sufficiently independent of control and data handling instrumentation to ensure that malfunctions tending to initiate potentially unsafe conditions will not at the same time render the protection system inoperative. A more detailed discussion of these considerations is given under Plant Protection Systems.

Reactor Core Instrumentation

Reactor core instrumentation is defined herein as instrumentation which is placed in driver, open test, closed loop test, control rod or reflector test positions.* This instrumentation is divided into two categories: core safety and core surveillance instrumentation. Core safety instrumentation must indicate conditions leading to impending loss or loss of fuel integrity and actuate reactor safety circuits. Three types of incidents which could require safety action are:

1. Coolant flow channel blockage
2. Accidental reactivity insertion
3. Fuel cladding failure.

Flow blockages may be detected by either flow or temperature monitoring. Reactivity additions cause increases in both the neutron flux levels and coolant temperatures and may be detected by either the flux monitoring system or core outlet temperature sensors. Fuel cladding failure detection may not require rapid safety action but could lead to local flow blockages and/or gross primary coolant system contamination if not detected at an early stage of development. Dynamic analyses to be made during the preliminary design period will establish the performance requirements for these protective systems.

Core surveillance instrumentation is used to obtain data in support of experiments and to increase operational efficiency. Although not directly depended upon for reactor safety, surveillance instrumentation can provide trend information to give advance warning of developing problems. Operation nearer to safeguards limits is permitted due to the availability of improved

* See Appendix for definition of test facility terms.

temperature, flow and neutron flux distribution information. Surveillance instrumentation performance requirements are, in general, less demanding than for reactor safety instrumentation.

Figure 4 indicates the types of instrumentation required.^{1,2} Outlet temperature monitoring is required for all channels in the core as a basic malfunction detection measurement. In-core flow measurements have been specified for all channels, both for data and plant protection inputs. Thermal calculations obtained by flow and differential temperature measurement would provide a means of checking and improving the accuracy of neutron flux distribution measurements. Closed loop test position flows can be measured external to the core except in the case where internal bypass flows are established in the test assembly.

All open test and closed loop test facilities must have the capabilities for installation of substantial amounts of in-core instrumentation. All instrumentation connections will be external to the reactor vessel. Connectors will accommodate electrical signal and power leads, pneumatic lines for sensing and control of test parameters, possibly sodium and/or fission gas sampling lines and traversing sensor thimbles. It is planned to provide each open test position and each closed loop test position with connectors for a minimum of:

- 20 electrical leads
- 1 pneumatic line
- 1 sodium or fission gas sampling line
- 1 traversing sensor thimble.

A fundamental guide during conceptual design has been that accessibility and connector compatibility must not preclude exploitation of any foreseeable instrumentation sensor or measurement developments. Signal lead accessibility, accessibility for disconnecting instrumentation and compatibility with fuel handling facilities, reactor cover nozzles, guide tubes and fuel channels must be provided. Figure 5 illustrates the complexity of the reactor top cover (space) caused by the need to monitor all reactor process channels.

Operational measurements which will be made on each test facility to assure reliable operation and safety of all facilities are:

- Coolant flow rate
- Coolant inlet temperature (reactor inlet temperature for open test positions)
- Coolant outlet temperature
- Power generation (computed)
- Failed fuel element detection and location
- Inlet pressure (reactor inlet pressure for open test positions)
- Coolant chemistry (same as the reactor primary loop for open test positions).

Test specimen measurement capabilities requested by potential facility users include:

- Surface temperature
- Internal temperature
- Internal pressure
- Coolant pressure drop

- Position
- Strain
- Vibration.

Sensors and instrumented assemblies capable of making these measurements will normally be provided by the experimenter. Connectors and other access and space requirements will be provided by the facility.

Measurement accuracies expected for initial operation and projected for future operations are given in Figure 6. In addition, neutron flux is expected to be known at all points in the core within approximately 10% initially.

Improved accuracies may be possible. However, the present meager knowledge of the severe environmental factors (neutron and gamma flux and fluence, temperature gradients, etc.) on sensor characteristics and transmission media, and the effects of handling during fuel charge and discharge operations lead to pessimistic predictions at this time. It is expected, however, that proper correlation of in-core measurements with the more accurate

MEASUREMENT ACCURACIES - INITIAL AND FUTURE OPERATION

	<u>Initial Operation</u>			<u>Future Operation</u>	
	<u>Reactor Coolant Facilities</u>	<u>Closed Loop Coolant Facilities</u>	<u>Ex-Reactor Measurements (Closed Loops)</u>	<u>In-Reactor Measure- ments</u>	<u>Ex-Reactor Measure- ments</u>
Coolant Flow Rate	10%	10%	5%	5%	1%
Coolant Inlet Temperature (°F)	2%	2%	1%	1%	0.5%
Coolant Outlet Temperature (°F)	2%	2%	1%	1%	0.5%
Power Generation/Channel	15%	15%	8%	8%	3%
Inlet Pressure (psig)	3%	--	2%	--	1%
Outlet Pressure (psig)	--	--	2%	--	1%
Cover Gas Pressure (psig)	2%	--	2%	--	--

-13-

NOTE: Accuracies are listed as percentages of normal operating values, e.g. 700°F ± 14°F.

FIGURE 6

out-of-core bulk measurements and with other in-core measurements will permit improved estimates of true in-core conditions as operating experience is gained.

Perhaps an even greater reason for optimism is that the installed, operating instrumentation will comprise a measurement system testing operation which will rapidly provide test data that are largely unobtainable anywhere today. This is expected to lead to the development of significant improvements in measurement techniques in the first few years of test facility operation.

(Reactor) Bulk Process Instrumentation

This category is used to describe those sensors and associated instrumentation which are located in the reactor vessel to measure bulk process conditions. It excludes driver, open test, poison rod and closed loop instrumentation and the instrumentation required for vessel and shield surveillance. The function of this instrumentation is to monitor cover gas and bulk coolant pressures and temperatures and the coolant level in the vessel. These measurements are required for the surveillance of process conditions in the reactor vessel and are used primarily for reactor control and in some cases for reactor protective purposes. Experimenters having tests in open test positions will require bulk inlet coolant temperature and pressure information. Measurements included in this group are:

- Inlet coolant supply pressure
- Low pressure plenum (if hydraulic holddown is used)
- Cover gas pressure

- Cover gas temperature
- Bulk coolant outlet temperature
- Bulk coolant inlet temperature
- Coolant level.

Inlet plenum pressures are needed to monitor the inlet plenum space for changes in hydraulic characteristics and possible malfunctions. Cover gas pressure measurement is important since excessive pressure can force coolant from the vessel and insufficient pressure can cause pump cavitation. In addition, cover gas pressure must be measured reliably if it is necessary to force reactor sodium into a fuel handling machine for cooling of irradiated subassemblies during handling operations.

It is expected that inlet plenum pressures will be sensed with a metal bellows (or diaphragm) exposed to the coolant sodium on the outside and filled with NaK on the side sealed to a capillary line. The NaK line transmits the sodium pressure to the top of the assembly where temperatures are expected to be less than 500°F. Here the pressure signals will be converted, by pressure-to-electrical signal transducers, to permit electrical transmission of the information through the nozzle connector. Since the inlet temperatures may be as high as 900°F in the ultimate condition, it does not appear practical at this time to use pressure-to-electrical transducers directly in the inlet plenum.

Temperatures will be measured with sheathed chromel/alumel thermocouples similar to those used for the driver subassemblies. Choice of location of the assemblies containing the coolant thermocouples is important to assure that they will be sensing the average temperature of the coolant. Several locations may be required to assure that all inlet and outlet streams and gas spaces will be monitored adequately.

Cover gas temperatures are required for monitoring of temperature gradients for surveillance information. In addition, sodium fires, caused by severe in-leakage of air, could be detected by gas temperature measurements. Inlet coolant temperature is required to obtain an indication of the core ΔT for calculation of total thermal power and individual tube powers and for malfunction detection. Outlet coolant temperature is a primary operations and protection system variable and is also required to calculate core power.

Reactor coolant level measurement is necessary to monitor the operation of the coolant control system and the vessel coolant inventories for safety purposes, both during reactor operation and for drain and fill operations. The present concept favors the inductance method of level measurement.³ Time domain reflectrometry methods also have promise due to their simplicity and self-checking features. However, further development is needed to prove these devices in sodium service.

Level measurement for drain and fill operations will be infrequently required. Rather than compromise the operating level detector by requiring it to operate over the entire vessel depth, it will be desirable to install, for such occasions, a special assembly in one of the removable reflector positions. After such use the assembly could be removed or left in place as a backup to the operating level instrumentation.

The location and mounting method for each sensor must be given careful attention. Coolant velocities, especially in the inlet plenum, may produce significant forces and vibrations for the thermocouples and bellows sensors.

Mechanical protection, such as cages or similar housings, may be necessary to reduce the possibility of mechanical damage. However, these housings must be designed to avoid dead-end oxides buildup, sulfur and rust

corrosion.

Environmental Conditions

Temperatures during ultimate operating conditions will be approximately as follows:

- Up to 900°F in the inlet plenums
- Up to 1200°F in the coolant outlet and cover gas
- Up to 400°F in the region above the vessel cover. In-

stead top

Expected neutron and gamma environments are given in Figure (7). These values may vary significantly with the detailed reactor design. The greatest exposure to gamma and neutrons may be for the thermo-couple and capillary leads from the sensors located in the inlet plenum since these leads may pass through the core.

Reactor Vessel and Shield Instrumentation

Provisions must be made for means of monitoring conditions which affect reactor vessel and shield life. These conditions include component motion and vibration, component temperatures and temperature gradients, shield coolant temperatures, neutron and gamma flux profiles, structural cracks or other abnormalities, leaks and earthquakes (refer to Figure (4)). These measurements, except for earthquake detection, will not generally be used in reactor protective circuits and have less stringent response and accuracy requirements than the reactor core instrumentation.

The use of thimbles installed in selected locations facilitates a flexible surveillance program. Such sensors can be moved, if necessary, to monitor areas found to be critical after initial operating tests. Vessel surveillance by ultrasonic, eddy current methods is made easier since movable probes can be inserted in the thimbles. However, because of the limited space available and the probable necessity to handle these devices remotely, there are challenging problems associated with the detailed design of these measurement systems.

Environmental temperatures for sensors in the vessel and fixed reflectors may be as high as 1200°F. Other shield temperatures will be considerably lower due to shield cooling. Estimated neutron and gamma flux environments are given in Figure 7. Other environmental conditions such as vibration or coolant velocity should not be a significant factor in the life or operation of most of these sensors.

Since the surveillance instrumentation does not, in general, provide protective circuit inputs, the reliability requirements are lower than for instrumentation required for operational safety. Therefore, redundant sensors will not be necessary if they are accessible for replacement or if they are needed only for pre-operational testing. Redundant sensors are advisable in inaccessible locations where information is needed beyond the initial reactor testing period.

Flux Monitoring Instrumentation

Flux monitoring requirements are of two general types:

- Determination of gross neutron flux for reactor power level control and plant protection.

- Determination of neutron flux spatial distribution, energy spectra and fluence.

Power level control and plant protection instrumentation performance requirements are generally similar to those of thermal reactors since the reactor dynamics are governed largely by the delayed neutron effects. However, due to the more rapid response of "fast flux" reactors to large ($> \$1.00$) positive reactivity inputs, careful analysis will be necessary to properly assess the accident potential of the chosen reactor concept and determine the corresponding plant protection system performance requirements.

The high gamma background created by the decay of the activated primary sodium coolant is a significant potential problem in all in-vessel or near vessel neutron measurements. The control and protective neutron monitoring systems are required to monitor the reactor flux level at all times when fuel is in the vessel. Gamma backgrounds of 10^5 R/hr or greater may be encountered during reactor shutdown when the neutron signal levels are approaching a minimum. Insufficient experience with neutron monitoring system performance under these conditions has necessitated a testing program to establish performance capabilities.

PNL is currently testing neutron pulse type chambers in gamma environments up to 10^6 R/hr using a Co-60 gamma source and a Pu-Be neutron source. High temperature environment capabilities are being added to this testing system.

In addition, Argonne National Laboratory is conducting tests on a mean squared voltage system in EBR-II at gamma levels up to about 4×10^4 R/hr.

Reactor testing of neutron monitoring systems at gamma levels in excess of 10^5 R/hr and temperatures up to 700°F may be required to sufficiently verify system performance.

Commercial ionization chambers capable of sustained operation in environments up to primary coolant temperatures (1000 - 1100°F) are not available today. The reference concept, which locates the operational chambers in the neutron shield external to the reactor vessel, requires cooled thimbles to reduce the chamber environmental temperature to about 600°F or below. Uncooled shield temperature could exceed 700°F with some designs.

Experimenters will want to know the nuclear environment in their test locations with the best available precision. Extensive measurements will be made in the NPTF. Measurement must be made in the FTR core for purposes of verification and correlation with the NPTF. Provision will be made in the FTR for measurement of the neutron flux, fluence, spectra and fission rates in the core and vessel using activation techniques and ion chambers. It is expected that all in-core measurements employing ionization chambers will be performed at low reactor power levels using traversing pulse type chambers.

An unforeseen, but conceivable, change in core, vessel and shielding design could reduce the leakage flux in the reference sensor locations to unacceptable low values, which might force the chambers to be located inside the reactor vessels. Ambient environments under those conditions would approach 1100°F and 10^6 R/hr gamma.

Failed Fuel Element Detection and Location

A fuel failure detection system is required because potential safety and operational hazard and fuel failures cannot be adequately prevented. It must be assumed that fuel will fail. The manner in which fuel will fail and the frequency of fuel failure cannot be accurately determined until the fuel has performed in its intended function for an extensive time period.

Experimental data and analytical techniques provide the bases for predicting failure modes, failure frequencies and the consequences of failure. These predictions, which are difficult to quantify, form the basis for foundations for fuel failure monitor requirements.

Fuel failures could range from the intermittent release of small fission gas bubbles from the fuel pin gas plenum to large releases which could significantly reduce local heat transfer capabilities, and produce clad overheating. Fuel debris releases could cause subchannel coolant blockages. The latter cases could conceivably cause adjacent pin failures, leading to propagation of subchannel blockages and eventual destruction of the fuel subassembly.

Detection sensitivity requirements are, therefore, dependent on assessments of factors such as the magnitudes of probable fission gas or fuel debris releases, their behavior in coolant streams or spaces subject to sampling, the efficiency of the sampling systems and the probability that the failure will grow to destructive proportions if not detected at a specified state of development.

Although detection of a failed fuel element is of primary importance to reactor safety and operation, a generally more difficult task is to locate the failed pin or even the fuel subassembly containing the failed pin. Plant downtime must be minimized in any practical test facility or power reactor. Therefore, rapid location of a failed element is a highly desirable feature. The cost of such a system must be evaluated against the cost of plant downtime and the problems associated with otherwise unnecessary fuel shuffling, changeout, etc.

Fuel failure detection in sodium-cooled reactors has generally been accomplished either by fission gas or delayed neutron monitoring techniques. Both the fission gas and delayed neutron monitoring methods require sampling systems which must cope with the problems of sample line plugging and efficient fission product collection and transportation to a detector. While these problems have been solved with reasonable success for bulk detection systems, the techniques required to identify the particular process channel (subassembly) containing the failed element are still in a highly developmental stage.

Other methods of failure location are being evaluated. These include:

- Isotopic Tagging. Non-fission product xenon gas is added to the fill-gas of each fuel pin. All pins in each subassembly are given a characteristic tag of mixed isotopes of xenon. Upon failure of a fuel pin, its gas tag escapes to the cover gas. Mass spectrometric analysis of the cover gas identifies the failed subassembly.

- Gettering. Fission product gettering tabs are placed in each channel. When fuel failure is indicated, the tabs are removed. High activity indicates channel containing failure.
- Grab Sampling of Fission Gas. Upon fuel failure some fission gas collects in a gas plenum at the top of the channel containing the failed pin. Each plenum is successively tapped until abnormal fission gas activity is detected.
- Fission Gas Detection by Channel Flowmeter. Each fuel channel may contain a magnetic flowmeter. Fission gas released from a failed pin would pass through the field of the flowmeter. Signal noise from the flowmeter could indicate the channel containing the failed pin.

The FFTF fuel failure detection and location concept has not yet been chosen. Conceptual design decisions are awaiting the results of several studies, including the final choice of reactor concept.

PLANT INSTRUMENTATION REQUIREMENTS

Plant instrumentation is defined herein as that group of instrumentation which is required for the heat transport systems, fuel handling and examination facilities, sodium and inert gas storage, handling and purification systems, radiation monitoring, containment and other auxiliary support systems. Most of the instrumentation requirements for these systems are similar to those of other liquid-metal cooled reactors and test facilities (see Liquid Metal Engineering Center (LMEC) reports) and will not be repeated

here. However, several features which may be of interest to the reader are presented, including a discussion of sodium "quality" measurements.

Plant availability goals must be considered in the design to ensure that maintenance and access restrictions due to heat, radiation and sodium hazards are adequately accommodated. All critical instrumentation will be designed for remote operation or be located such that access is available whenever required.

Main Heat Removal System

Flow Measurements

Measurements of primary coolant loop flows are of particular interest because of their use in the calculation of reactor thermal power level, control of reactor outlet temperature or core differential temperature and as a primary plant protection system input. An accuracy goal of $\pm 2\%$ of design-rated flow has been set for the primary flowmeters. Means of maintaining this level of accuracy and perhaps improving it by the use of more accurate secondary loop measurements and heat balance calculations, or transit-time methods⁶ using thermocouples as the sensors, are under study.

Redundant flow signals are needed for plant protection circuits. It is planned to use magnetic flowmeters with multiple signal connections to obtain this redundancy. This eliminates the need to install relatively large numbers of expensive and space-consuming magnetic flowmeters in the limited primary loop equipment space. The probability of failure of the magnetic assembly is expected to be extremely low.

In addition, the multiple primary loop configuration provides an inherent mechanical equipment redundancy which limits the consequences of flow reduction or loss of flow in a single loop.

Control System Considerations

The physical arrangement of the main heat transport system results in large transport lags in the coolant loops. The primary coolant loops are entirely within the containment sphere and have delays in the order of seconds. The secondary coolant loops span the distance from within the containment sphere to the air heat dumps. These loops have delay times of the order of tens of seconds. At low flow rates the delay times present temperature control stability problems that may require the use of dead time compensation techniques. Even at high flow rates this type of control appears to be desirable to achieve precise control of reactor and test specimen temperatures and to permit orderly changes in operating temperatures over a minimum time interval.

The use of four parallel air heat dumps on each main heat removal loop presents a complex control problem since sodium freezing, or cold trapping, is to be avoided. The heat dumps are large units, both physically and thermally. The exhaust stacks will be approximately 120 ft. high with an estimated control damper cross sectional area of 250 ft². The design thermal dissipation of each unit will be 34-40 megawatts with a secondary coolant Δt of 300°F centered around 650°F.

Heat dump characteristics which must be considered in the control of these units are:

- The units must be designed for efficient heat dissipation and at low power operation they can overcool.
- Air leakage through control dampers when they are in the closed position will impose heating requirements.
- Large changes in seasonal ambient temperatures (summer to winter) will probably require changes in operating procedures.

Procedures for control of the air dump units will include auxiliary heating of the air dumps until reactor power is high enough to provide a sufficient heat source. It is expected that one unit per loop will be brought on-line in sequence to accommodate increasing heat load. As each unit approaches its design load, another unit will be brought on-line. This procedure will continue until all units on all loops are operational. When the reactor power is reduced for shutdown, the sequences for removing heat dump units will generally be in the reverse order to that followed for the startup sequence. A possible 140°F difference in ambient temperature from winter to summer may affect both the sequence and the operating points. The startup and shutdown sequence will have to be defined for separate ranges of ambient temperature.

Operation of the heat dump units following a reactor scram may have to be preprogrammed and automatically initiated. It is planned to develop these operational control requirements using a comprehensive dynamic simulation of the process as a major analytical tool.

Sodium pipe heating will be necessary to get from a shutdown or refueling condition to the desired reactor startup conditions. A large number of zone heaters and temperatures must be monitored and controlled to bring the system temperature up uniformly and to keep temperature gradients within permissible limits.

Computer-based monitoring of the pipe heating systems provides for high speed scanning of all points and rapid display of all deviations.

The operator is relieved of the tedium of manually selecting all points but is still informed of the up to the minute status of the entire system. Computer control of the auxiliary sodium heating and cooling systems appears to be advantageous from both an economic and equipment protection standpoint.

Closed Loop Heat Removal Systems

Operation of the closed test loop heat removal loops involves essentially the same problems described for the main heat removal system. The control methods used in the main heat removal loops are expected to be applicable to the closed test loops. The actual operating conditions for the closed test loops will be defined on the basis of their physical characteristics and upon the requirements of the tests to be run.

Sodium Quality Measurement

Design requirements for the quality of sodium in the FTR are:

- Below 1000°F, O in Na shall be controlled to less than 10 ppm.
- Above 1000°F, O in Na shall be controlled to less than 5 ppm and C in Na shall be controlled to less than 20 ppm.

Control will be provided by the operation of conventional cold traps and hot traps.

Measurement will be made by on-line monitors supported by "wet chemical" analyses of sodium samples taken from sidearm loops. "Wet chemical" analyses may be by the mercury amalgamation technique or the vacuum distillation technique. Suitable on-line monitors have not yet been selected, but will probably be one or more of the types described below. The ultimate choice depends on the success of development efforts over the next few years.

Plugging Meters

oxygen (O) in sodium (Na).

Plugging meters do not measure ~~oxygen~~. They measure the temperature at which the precipitation of impurities in sodium begins to impede flow. If the first impurity to precipitate in a sodium system is Na_2O , and if the plugging meter is calibrated to translate plugging temperature to Na_2O solubility, then the plugging meter can be said to measure O in Na.

However, in view of the uncertainties associated with these two premises, it is a questionable practice to translate plugging temperature to "equivalent oxide" concentration. If the plugging meter is chosen as the on-line monitor, it will not, of itself, satisfy the requirement ~~for~~ for measuring O in Na stated at the beginning of this section. It can only be used as an indicator of the trend of sodium quality between the intervals of "wet chemical" measurements. Even so, the plugging meter is the best on-line instrument available for monitoring sodium quality at this time.

There are two types of plugging meters, the Intermittent Plugging Meter and the Continuous Plugging Meter. Of the two, the Intermittent

Plugging Meter is the best known and the most generally used, but measurements can be obtained only at intervals of 30 to 60 minutes. The continuous Plugging Meter gives essentially continuous measurements and is intrinsically more useful in relatively pure systems where O in Na is $\approx$ 5 to 10 ppm.

UNC Oxygen Meter

Another on-line process instrument being considered for measuring O in Na at the FFTF is United Nuclear Corporation's Electrochemical Oxygen Meter. This meter is essentially a galvanic cell consisting of a Cu:Cu₂O half-cell and a Na:Na₂O half-cell separated by a solid ThO₂-Y₂O₃ electrolyte. The Cu:Cu₂O is a reference half-cell. The sodium system being monitored comprises the variable concentration Na:Na₂O half-cell. The meter readout is a voltage which is proportional to the negative log of the oxygen activity (not concentration) in sodium. This meter has two inherently important advantages:

- a. Corrosion rates are more closely related to oxygen activity (which it measures) rather than to concentration.
- b. Because of the logarithmic character of its signals, this meter is capable of measuring very low (few ppm) O in Na levels.

Field tests at a number of laboratories over the past few years discovered a number of problems which must be resolved before this instrument can become a useful process instrument, however. Among these are:

- Signal hysteresis with changing O in Na. Cause is not known, but worsens with time. May be due to attack of impurities in the electrolyte.
- Time dependent calibration shifts. May be due to sodium interactions also.
- Vibration effects. These disturb the stability of the meter, sometimes for extended periods.
- Thermal shock. Severe thermal shock causes electrolyte to crack and fail.

If these problems can be overcome or minimized, the UNC oxygen meter may support or even replace the plugging meters. At the present time, the electrochemical oxygen meter could be used for information purposes, but not for control.

UNC Carbon Meter

The only on-line instrument capable of monitoring carbon in sodium at this time is United Nuclear Corporation's Carbon Meter. Three meters have been fabricated for the USAEC of which one has been assigned to Mine Safety Appliance for testing. Preliminary information indicates a highly satisfactory performance. Barring subsequent unfavorable reports, it seems probable that the UNC carbon meter will be used at the FFTF.

In the operation of this meter, a sodium stream, heated to 1400°F, is passed across a 0.010 inch thick iron membrane. Carbon in the sodium diffuses through the iron and is converted to CO at the opposite face by a decarburizing sweep gas of (Ar, H₂ and H₂O). The sweep

gas travels over a Ru catalyst (600°F) where CO is quantitatively converted to CH_4 . Finally, the sweep gas passed through a flame ionization detector where the CH_4 burns, the resulting ions are collected and current flow is a measure of the concentration of CH_4 in the gas stream. A step-change of C in Na is reported by the ionization detector with 45 seconds.

The carbon meter measures "carb^uridization potential" rather than total carbon in sodium. "Carburization potential," for the moment an undefined term, is related to carburization rate. Carburization rate, rather than total carbon in sodium is the parameter which needs to be controlled. It is expected that "carb^uridization potential" will eventually be translated into carburization rates for various steels and other reactor materials. Tests with the carbon meter at UNC have shown that H in Na tends to decrease the "carb^uridization potential" while O in Na tends to increase the "carb^uridization potential." These effects are confirmed by carburization of sample tabs.

Hydrogen Meters

Design criteria for the FFTF do not explicitly require the measurement of H in Na. However, it is recognized that the Carbon Meter, the Electrochemical Oxygen Meter and the Plugging Meters all respond to the presence of H in Na. It is, therefore, concluded that some independent means for monitoring H in Na, preferably a continuous means, would be helpful for interpreting the readings from the carbon, oxygen and plugging meters. This is especially important when the sodium in the reactor is relatively fresh and has not been heated sufficiently to remove the hydrides and hydroxides from it.

At present, there are no continuous, on-line process instruments for monitoring hydrogen in sodium. There are, however, two instruments in various stages of development. These are the APDA Diffusion Type Hydrogen Meter and the UNC Electrochemical Hydrogen Meter. The operating principal of the APDA meter is the diffusion of H in Na through a membrane permeable to hydrogen. The diffused hydrogen is then catalytically converted to water vapor and is carried in a sweep gas to a moisture monitor which continuously measures the concentration of water vapor in the gas stream. The principal of the UNC meter is to measure the potential between a reference half-cell (typically a mixture of Ca and CaH_2) and the variable concentration half-cell of Na and NaH. An electrolyte (typically LiCl-KCl eutectic + LiH) separates the two half-cells.

Of the two types of hydrogen meters, the APDA concept is the more advanced in development at this time but the UNC concept is being actively pursued.

Fuel Handling and Examination

The reactor refueling machine proposed for the reference dispersed head reactor concept has the general shape of a cylinder approximately 65 feet high. It is shielded and is expected to weigh at least 150 tons. This machine requires instrumentation and controls which will remotely index and align it in a vertical or tilted (up to approximately 10°) position over any one of the approximately 100 process channels. Instrumentation must be provided to continuously indicate the position of fuel handling grapple and control the push and pull forces exerted by the machine. Temperature control must also

be provided for the heat exchange system which is required to remove decay heat loads of up to 35 kilowatts.

The reactor refueling machine operates in a containment area. This area must be maintained at a pressure which is negative with respect to the surrounding work areas but slightly positive with respect to the reactor cover gas. The pressure differential between the reactor cover gas and the containment area must also be closely controlled near zero differential to prevent airborne contamination from entering the containment area.

The inert gas examination facility provides a controlled environment to:

- Retrieve post irradiation data from fuels, materials and components irradiated in FTR.
- Service defective core components irradiated in FTR.

Examinations will be conducted remotely using the following techniques:

- Visual inspection and measurement
- Photograph (infrared and normal)
- Neutron radiography
- Gamma autoradiography
- Gamma scanning
- Nondestructive testing.

Argon will be used as the inert atmosphere for the examination cells.

Nitrogen, oxygen, sodium vapor and water vapor concentrations in these cells must be monitored and maintained at low levels.

Containment System Instrumentation

The FTR will be provided with a Reactor Containment Vessel (~ 190 feet diameter) which will function as the final containment barrier in the event

of nuclear accidents, or sodium fires, to limit the release of radionuclides to meet requirements of AEC Site Criteria, 10CRF100.

Instrumentation will be provided to monitor continuously the pressure, temperature and radioactivity in the individual cells, and containment vessel effluent gases. A reactor shutdown and ventilation containment will be actuated upon indication of an increase in radionuclide concentration in the effluent air stream above preset limits. Automatic closure of all containment valves, except those essential for reactor cooling, will be initiated upon an increase in pressure following ventilation closures.

The chief problem area, as far as containment instrumentation is concerned, is that of deciding on criteria for, and methods of, monitoring the radionuclide release rate. In order to determine radionuclide concentrations in the containment vessel and effluent air for all conditions, a very wide range of air flows and radionuclide concentrations must be measured.

Even though containment incidents should seldom if ever occur, ~~the~~ design of the monitor system must provide a high degree of reliability and must remain operable even during and following a Design Basis Accident. Protection must be provided against flying missiles and means must be provided for frequent testing.

No very unusual control or instrumentation problems are foreseen for the ventilation system. Interfacing and interlocks with the containment and fire control systems will be provided.

Air particulate and halogen monitors will be provided downstream of the exhaust filters which will provide alarms on above normal activity levels.

Radiation Monitoring

Radiation monitors will also be utilized for sodium leak detection. These leak detectors will be located in vent ducts which are considered to be low background areas. When a sodium leak occurs, vaporized radioactive sodium will enter the duct and be detected. These detectors as well as other airborne activity monitors can be utilized to isolate individual cells in the building.

Continuous monitors utilized for personnel protection to detect alpha airborne contamination must automatically separate signals due to natural radon and thoron activity from hazardous activity deposited on a collector. A detection goal of 1 MPC of Pu^{239} in two hours, in the presence of radon and thoron gases which would have an alpha activity equivalent to a 60 MPC airborne plutonium atmosphere, has been established.

Radiation monitors will be provided with great enough range to provide usable data during and after at DBA. Capability for reading these monitors from the perimeter of the building will also be provided.

Criticality monitors will be neutron sensitive. Under-moderated BF_3 tubes are the presently preferred sensors.

CONTROL AND DATA ACQUISITION SYSTEM REQUIREMENTS

The FFTF is a test facility. Its primary objective is to provide a controlled environment for the testing of fuels and materials and to provide documentation of test conditions and results. These data will be used to further the development of reliable and economical fast breeder reactor components. Therefore, the FFTF data acquisition systems must be designed to provide reliable and complete data acquisition, manipulation and storage.

FFTF control requirements are also different from commercial power reactor control requirements in that good experimental data must include an accurate record of test environments. Each experiment should be accurately repeatable to permit verification of small but significant variations in test results. The central control and data acquisition systems must provide precise control of individual experiments and at the same time, provide for orderly, safe and reliable operation of the total facility.

Due to the relatively large number and variety of experiments which can be run simultaneously in the FFTF and because of their significant interactions (both physical and operational), the reactor plant operators must have convenient access to pertinent operational data and assistance from automatic control systems. Although it is expected that many of the process variables, including the reactor power level, will be controlled manually during early operating periods, maximum facility efficiency will be attained only by use of a relatively high degree of automatic-control.

Control, data handling and display systems which use modern digital computers as the basic functional elements offer many features which are especially attractive to the FFTF system designer. This type of system is receiving rapid acceptance in the electrical power generating industry, as well as in other industries. Battelle-Northwest proposed at the outset of the FFTF Project that this kind of system be thoroughly investigated for its proper applications to the project. It was recognized that computer control experience on nuclear plants was severely limited at that time (1965); also, that there would be opposition to reliance on such a system for initial FFTF operation since the physical reactor plant design itself represented a significant technical challenge and many people felt that conventional control hardware would be capable of providing satisfactory performance. However,

because of the many potential advantages to be gained from the provision for digital computer control capability, Battelle requested and received USAEC approval to assess this potential and recommend an appropriate conceptual design. Space does not permit a detailed discussion of this topic. However, a summary of the status of the conceptual design will be given.

Early agreement was obtained on the need for a digital computer-based data acquisition and display system for the FFTF. The inherent advantages of improved flexibility and accuracy and improved data display, storage and manipulation capabilities, combined with data authenticity and self-testing capabilities, to name a few, are acknowledged. However, because of the lack of specific computer-based reactor control experience of a comparable magnitude, a generally conservative design approach and the relatively tight schedule set for FFTF construction and startup it has been decided to provide in the initial design those analog controls and displays of critical variables which are necessary to permit reactor startup without a fully operational computer-based control and data handling system. Provision for efficient expansion to digital control as the facility requirements dictate, and as confidence is gained in the computer-based system, will be a primary design objective.

A typical conceptual arrangement of the proposed system is shown in Figure 9. A multiple computer configuration is found to be necessary to meet the data acquisition reliability and plant availability requirements. Design studies completed to date by independent workers indicate a range of two to four central processing units (CPU's) to be optimum, depending upon

the types of CPU's and their inherent design features and limitations. A brief summary of the tentative requirements imposed on this system is given below:

- The plant protection system shall be functionally independent of the control and data handling system.
- No single failure of either a computer or peripheral equipment shall result in a loss of total digital system capability.
- Computer-directed displays shall be provided at the reactor operator's console, the process and experiment operator's console, and on a general purpose console in the control room.
- A failure in the digital system or its displays shall not require a plant shutdown. Critical data which are required to continue the operation of the facility shall be available to the operator from analog displays.
- The system shall include the capability to expand to digital control, and this expansion will require only software additions to the digital system for a specific number of control functions yet to be defined.
- A software check for reasonableness of all acquired data shall be provided. The system design shall include the capability to cross check redundant sensor inputs.
- The system shall have the capability to store complete operating histories of the facility experimental data and fuels and materials inventories. The data shall be recorded in a format which is compatible with commercial data processing equipment

3

(IBM-compatible). Operating histories shall include significant operator actions and responses.

- The system shall include sufficient redundancy of bulk storage devices so that no single failure will cause a loss of previously stored data or make the recently acquired data inaccessible for display purposes.
- System software shall include the provision for analysis and calculational aids in support of operations. As a minimum, these will include the following:
 - Trends toward malfunction of critical plant components or processes.
 - Thermal power calculation.
 - Supervisory controls and displays.
 - Heat exchanger performance.
- The system shall include the hardware and software capability needed to implement fuel inventory and accounting activities in support of fuel handling and loading operations. Reactivity status and core analysis capability shall be included.
- The facility shall provide data terminals in experimenter's data handling rooms. These data terminals shall include CRT display equipment and a typewriter.
- The system shall include the capability to interface with signal transmission equipment which may be installed in the future for high-speed data transmission to an EDP facility or to an offsite data terminal.

Plant Protection

A comprehensive discussion of the plant protection system requirements is beyond the scope of this paper. However, one aspect of its relationship to the control and data handling system is discussed briefly.

Reactor plants must be provided with plant monitoring and control systems which can adequately detect abnormal and potentially unsafe events or plant malfunctions and initiate proper corrective actions. These systems, usually called "Plant Protection Systems," require careful analysis by licensing bodies and must be described clearly in safety analysis reports. Plant protection systems should be functionally independent of plant control systems to prevent the occurrence of equipment malfunctions or failures which could initiate unsafe conditions through control actions, while simultaneously disabling the protective system.⁴ Methods or systems designed to monitor or test the integrity of the Plant Protection System must be designed in such a way that they do not compromise plant protection.

The above requirements can lead to a tendency on the part of protective system engineers to impose severe restrictions on the degree to, and methods by, which plant protection systems can be monitored. These restrictions, improperly applied, can result in substantially increased plant costs, loss of important operating and test data and conceivably, in reduced plant protection.

The following illustrates some of these factors. The question of whether the FFTF data acquisition system should be used to monitor Plant Protection System signals has been raised. The principle objective appears to stem from a fear that some malfunction of the data acquisition system

could simultaneously disable all channels of a redundant plant protection function. This would imply either a design with inadequate circuit isolation or multiple undetected isolation circuit failures prior to the incident. It is expected that application of good design principles can satisfactorily protect the system against both of these possibilities.

Much valuable data that are not normally available from control or data acquisition instrumentation channels will be generated by Plant Protection System instruments. An example of this is the FFTF fuel subassembly outlet temperature measurements which are used to insure the integrity of the individual fuel subassemblies. Four thermocouples (in a 2 out of 4 logic arrangement) are deemed necessary to provide adequate safety and plant availability in view of the hostile environments in which these sensors are placed. A fifth thermocouple may be provided for normal data acquisition purposes or as an operational spare.

Due to the large number of pins used in the FFTF fuel subassemblies and the unequal temperature differences which will be generated in each of these coolant subchannels, it is not currently known what the mixed outlet temperature will look like as a function of position in the coolant duct and time.

Multiple measurements in the duct can provide more detailed information on the behavior of these phenomena than can be obtained by a single measurement. If data from the protection instrumentation cannot be made available to the ~~(FFTF:digital)~~ data acquisition system, a serious loss of important data will occur since it will not be practical, due to space, access and cost considerations, to duplicate the protective sensors in each subassembly.

Studies are now in progress to assure that adequate means of isolating the data acquisition system monitoring circuits can be provided. Basic requirements for these studies have been stated as follows:

- No single failure in the monitoring system will affect more than one instrumentation channel.
- The monitoring system shall not significantly reduce the independence of redundant instrumentation channels.
- Failures in the protective/monitoring isolation system shall automatically be brought to the attention of the control room operator.

In addition to the need to make maximum use of available data, the digital data handling system could be used effectively to aid in the determination of protection system integrity. Erratic operation of a single channel could be effectively identified. Calibration and test operations could be performed on-line at preprogrammed intervals or as an operator desires.

These aspects of plant protection system/data acquisition system interfaces are currently in a design study status and are mentioned here in an attempt to stimulate interest in this aspect of protective system monitoring.

REFERENCES

1. R. E. Peterson. Technical Basis for FFTF Driver Fuel Instrumentation, BNWL-555, September 1967.
2. J. J. Regimbal. Technical Bases for FFTF Open Test Position and Closed-Loop Instrumentation, BNWL-779, April 1968.
3. H. W. Slocomb. Liquid Metal Level Measurement (Sodium) State-of-the-Art Study, LMEC, NAA-SR-MEMO 12582, p. 27, November 15, 1967.
4. "Proposed Criteria for Nuclear Power Plant Protection Systems," IEEE No. 279 (in preparation).

ACKNOWLEDGMENTS

The author wishes to acknowledge with thanks the help of the FFTF staff who assisted in the preparation of this paper. Particular thanks are due to C. R. F. Smith, who contributed the section on Sodium Quality Measurements, assisted with editing and preparation of illustrations and helped in many other ways; to R. R. Cone, who contributed to the sections on Reactor Instrumentation; and to G. G. Thieme, who contributed the section on heat transport system control considerations and directed the preparation of many of the illustrations. Other contributors to the text were M. O. Rankin, J. K. Flickinger and R. J. Hennig.

APPENDIX

A GLOSSARY OF TEST FACILITY TERMS

Driver Fuel - The driver fuel is the bulk of the core fissile material configured in a dense array to form the major segment of the critical mass and, at the same time, create a fast flux environment.

Closed Loops - A closed loop is an in-reactor loop whose recirculating coolant is isolated from the reactor coolant and other test facilities. Each closed loop has its own equipment cells, heat exchangers, pumps, coolant control equipment, secondary coolant loop and air-blast heat exchanger. A closed loop may be used for testing fuel and materials specimens under conditions other than those provided or permitted by the reactor coolant. A closed loop facility has its own independent coolant chemistry control and heat removal capabilities designed to special testing requirements. The FFTF closed loops are designed for sodium coolant, but coolants other than sodium may ultimately be permitted. Coolant temperatures are limited to 1400°F maximum with bulk temperatures not to exceed 1200°F. Closed loop systems are designed to accept defected fuel, vented fuel and fuel clad failures short of fuel meltdown during testing.

In-Core Open Test Position - In-core lattice positions which ordinarily contain driver assemblies may become in-core open test positions by replacement of the driver fuel assembly with an in-core open test assembly. The test assembly is directly cooled by the reactor primary coolant.

Reflector Open Test Position - Reflector positions immediately adjacent to the reactor core may become reflector open test positions by replacement of the reflector with an open test assembly. Test assemblies in these positions

are directly cooled by the reactor coolant. Coolant flow and temperature control are made possible by orificing, auxiliary heating and by provision for an internal flow channel. Since all first row reflector positions are provided with full instrumentation capability, any position is available for testing.

Short-Term Test Facilities - These facilities are designed for the irradiation of test specimens over intervals shorter than one reactor cycle. One or both of the following types of short-term test facilities may be installed in first row reflector positions in the initial core:

- A rabbit type (pneumatic or hydraulic carrier)
- A trail cable type (mechanically driven carrier).

The FFTF will be designed for installation of additional short-term test facilities in selected core and first row reflector positions. These additional facilities will be added as required.

Capsules and Package Loops - Within mechanical interference, criticality and safeguard constraints, self-contained capsules and package loops of any type may be installed in any core lattice position during any reactor refueling interval. Capsules and package loops are constrained to the envelope dimensions of the core components that they displace.

Axial Positioner - An axial positioner is a mechanical device which provides dynamic control of the axial position of the test specimen in a test facility, thus providing capability for precisely controlling flux or simulating transients in the test during the irradiation period. The characteristics and features of axial positioners are essentially similar to those for control rod drives.

FAST FLUX TEST FACILITY

Figure 1. Typical Reactor Concept

BATTELLE-NORTHWEST
PHOTOGRAPHY UNIT

ORDER NO. 41120-1

PNL

PLEASE CREDIT

PACIFIC NORTHWEST LABORATORY
OPERATED BY
BATTELLE MEMORIAL INSTITUTE
FOR THE
U. S. ATOMIC ENERGY COMMISSION

CORE GEOMETRY

VERTICAL vs DISPERSED

Figure 2. Typical Core Arrangements

PLEASE CREDIT

PACIFIC NORTHWEST LABORATORY
OPERATED BY
BATTELLE MEMORIAL INSTITUTE
FOR THE
U. S. ATOMIC ENERGY COMMISSION

BATTELLE-NORTHWEST
PHOTOGRAPHY UNIT - 300 AREA

ORDER NO. PN0690423-51

-8-

Figure 3. Conceptual Facility Arrangement

BATTELLE-NORTHWEST.
PHOTOGRAPHY UNIT - 700 AREA

PM 46676-1

PLEASE ORDER
PACIFIC NORTHWEST LABORATORY
OPERATED BY
BATTELLE MEMORIAL INSTITUTE
FOR THE
U. S. ATOMIC ENERGY COMMISSION

Figure 4. Reactor instrumentation schematic.

-50-

Figure 5. Typical reactor vessel cover area.

FIG. 7A
 NEUTRON FLUX ABOVE CENTERPOINT OF
 FTR CORE (AT 400 MW POWER)
 REFERENCE: COMMUNICATION FROM
 C.A. MANSIUS (PNL) JULY 68

NEUTRON ENERGIES

- THERMAL = < 0.076 EV
- .-.-.- INTERMEDIATE = 0.076 EV TO 0.067 MEV
- FAST = > 0.067 MEV

BATTELLE-NORTHWEST

PHOTOGRAPHY UNIT

700 AREA

ORDER NO.

PM 48528-1

PLEASE CREDIT
PACIFIC NORTHWEST LABORATORY
OPERATED BY
BATTELLE MEMORIAL INSTITUTE
FOR THE
U. S. ATOMIC ENERGY COMMISSION

FIG. 7b
 GAMMA RAY INTENSITY ABOVE CENTERPOINT
 OF FTR CORE AT 400 MW POWER
 REFERENCE: COMMUNICATION FROM
 C.A. MANSIUS (PNL) JULY 68

BATTELLE-NORTHWEST
PHOTOGRAPHY UNIT
700 AREA

ORDER NO. PM 8528-2

PLEASE CREDIT
PACIFIC NORTHWEST LABORATORY
OPERATED BY
BATTELLE MEMORIAL INSTITUTE
FOR THE
U. S. ATOMIC ENERGY COMMISSION

Figure 8
HEAT TRANSPORT SYSTEM ARRANGEMENT

NOTE: 31 INDICATES SYSTEM NUMBER

TENTATIVE ARRANGEMENT OF INITIAL CORE

- 31 REFLECTOR SHIM ROD (8)
- 32 SHIM SAFETY ROD (6)
- 33 SAFETY ROD (4)
- 34 CLOSED LOOP (5)
- 35 NEUTRON SOURCE (1)
- 36 OSCILLATOR (1)
- 37 REFLECTOR OPEN TEST POSITIONS (39)
- 38 DRIVER FUEL (77)
- 39 INCORE OPEN TEST POSITIONS (1)
- 40 SHORT-TERM TEST (1)

FUEL EXAMINATION DATA SCANNING AND STORAGE LOCATED NEAR EACH CELL COMPLEX

Figure 9. Digital data handling system inputs.