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FAST FLUX TEST FACILITY

PLANT OPERATION AND CONTROL

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FFTF PLANT OPERATION AND CONTROL

ABSTRACT

A discussion of plant operation and a concept for control and protection of the FFTF are presented. The first section of the document discusses plant operation including operation of tests, the reactor, and heat removal systems during startup, steady-state operation, and normal shutdown. The second section presents a concept for plant control including nuclear systems, main heat removal systems, and closed loop heat removal systems. The third section discusses abnormal and emergency plant control, including a concept for controlled power reduction and plant protection.

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INTRODUCTION

In order to establish the proper functional and design criteria for each control system of the Fast Flux Test Facility (FFTF), a systematic plan for the overall plant operation and control is needed. The desired approach to plant operation may be described as a "plant operational philosophy," and outlines the operation of the reactor and main heat transport systems and the various test facilities. From the plant operational philosophy one may then develop a concept for overall control and also for each separate control system.

Accordingly, within the limitations of FFTF conceptual design, the purpose of this document is to present the current approaches to: (1) normal operation of the test facilities, (2) normal operation of the Fast Test Reactor (FTR) and its heat transport systems, (3) control of the reactor, heat transport and closed test loop systems, and (4) protection of the plant against malfunctions and failures. Interactions between protection systems and control systems are also considered (e.g., heat transport system control response to reactor scram).

This document provides support information for the Conceptual System Design Descriptions (CSDD) for Central Control and Data Handling System No. 91 and Plant Protection System No. 99. Specifically, it serves to illustrate (1) control and operation coordination and integration, and (2) safety coordination.¹ The instrumentation and control CSDD's,²

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1. Refer to References, Appendix A, Item 1, See Figure i.
 2. Refer to References, Appendix A, Items 1, 2, 3, 4, 5, 6, and 7.

in addition to this report, provide a sufficiently complete view of plant operation and control for the Plant Designer to move from conceptual design into preliminary design of control for the reactor and heat transport systems. Auxiliary systems (e.g., fuel handling) are not considered at this time.

Where possible, estimates are provided for numerical values which may prove useful during design. It is understood that such values require updating as the design progresses.

SUMMARY AND CONCLUSIONS

In developing an FFTF plant control concept for the conceptual design, the plant operational philosophy is first considered. The reactor and its heat transport systems plus the test facilities are of primary interest; auxiliary and support system operation and control are deferred for later study. Testing objectives and operating desires lead to the following general conclusions about plant control:

1. Control of closed test loops should be versatile in order to meet differing test objectives. For example, close control of inlet temperature may be required for one test, while close control of test section ΔT may be required for another. Control system parameters (gain, reset) should be variable to accommodate the different tests.
2. Control of the reactor and main heat transport loops should be based on simplicity and safety. Safety of startup operation is enhanced when a minimum number of operating parameters are changing (e.g., constant flow startup), freeing the operators to concentrate on safety-related information.

At present, direct digital computer control is not proposed for initial operation. However, as the FFTF becomes more established in its operation, increased confidence in the plant will allow the computer to perform more complicated startup and shutdown routines.

The plant control concept has the following features:

1. Reactor control based on neutron flux level with continuous calibration against reactor thermal power. The flux level set point will be set manually, with the option of direct manual control of the rods.

2. Heat transport system control based on control of primary coolant temperature by air flow at the DHX. The set point for control of air flow may be derived from IHX primary outlet temperature and reactor power. Control of sodium flows is accomplished by equalizing primary loop flows with a common manual set point and by matching secondary flows to their respective primary flows.
3. Closed test loop control based on control of test inlet and outlet temperatures by sodium and air flows. Flow set points will be provided from the test coolant temperatures by an analog control circuit which will be easily reprogrammed to meet test objectives (constant test ΔT , constant inlet temperature, etc.).

Corrective action will be based on two approaches: the full emergency shutdown (scram) and a controlled power reduction (CPR). The scram will be the fastest shutdown possible of the reactor power, with the primary purpose of protecting the FTR and the tests, and will be initiated by the Plant Protective System. All safety rods will be inserted, and heat transport systems will respond by reducing sodium flows to minimize thermal transients. CPR action will be designed into the normal control system and will take two forms:

(1) setback, in which only the power is reduced by the reactor flux control system in a manner corresponding to the incident (e.g., a percentage power reduction for the DHX module or a continued control rod insertion until a reactor overpower condition is corrected, and (2) programmed shutdown, in which power and flow are reduced together to the decay heat range, in order to reduce thermal transients below those due a scram. These corrective actions will be sufficient to protect the FFTF reactor and associated systems for the full spectrum of anticipated incidents.

SECTION 1.0 PLANT OPERATIONAL PHILOSOPHY

The following section of this document is organized as follows: (1) the objectives for each type of test are stated, (2) the operational aspects of each type of test facility are described from startup through shutdown, with a discussion of the parameters which will require monitoring and control, and (3) operation of the reactor and heat transport systems is discussed.

1.1 TEST OBJECTIVES

The mission of the FFTF is to provide experimenters with the desired controlled environments for the testing of fuels and materials for future liquid metal fast breeder reactors (LMFBR). Specific objectives are high neutron flux in the high-energy spectrum, elevated coolant temperatures (up to 1400 °F sodium in closed test loops), and controlled coolant chemistry. Operation of the FTR and the tests themselves must, therefore, be matched to these objectives.

Several different testing facilities are being planned for use in the FTR and are fully described elsewhere.¹ Although these facilities may change as the plant design progresses beyond the current conceptual stage, it is possible to state at this time the general operation of the tests, based upon the projected desires of experimenters. Conceptual design of control systems from the standpoint of testing needs may then be developed from the desired test operation. A summary of planned FFTF testing capability is shown in Table 1-I.

1. Refer to References, Appendix A, Item 8.

TABLE 1-I. FFTF Testing Capabilities and Objectives

Facility	Tests	Objectives
Closed Loops	Prototype or partial fuel elements; materials tests	Burnup, breeding parameters; thermal-hydraulic characteristics; clad corrosion and mass transfer; fuel failure characteristics; vented fuel performance; fission product deposition; materials damage and property changes.
Open Test Positions (Core and Reflector)	Prototype or partial fuel elements; single pin tests; materials irradiation; instrument irradiation	Burnup, breeding parameters; thermal-hydraulic characteristics; fission gas release; materials damage and property changes; instrument integrity and response.
Axial Positioners	Tests in closed loops or open test positions	Fuel and cladding characteristics as functions of flux or thermal cycling; instrument response and integrity as functions of flux or thermal cycling.
Package Loops (Not supplied as part of the FFTF.)	Partial fuel elements; single pin tests; fuel and materials capsule irradiation (Placed in open test positions.)	Burnup, breeding parameters; fission gas release; materials damage and property changes; clad corrosion and mass transfer.
Short-Term Facilities (Trail Cable)	Fuel and materials capsule irradiation; single and multiple pin tests	Fuel and material damage ("Screening")

As shown in Table 1-I, testing facilities may serve the needs of experiments with different test objectives. Therefore, operation of the FTR must consider the operation of each test facility and its needs, which may then require versatile control systems to serve different purposes.

1.2 OPERATING THE TESTS

Each test facility will have its own operational needs and problems based both on its design and on the enclosed test at any particular time. For example, at-power operation of a closed loop will require precise control of coolant purity and inlet temperature for corrosion and mass transfer studies, whereas for prototype fuel cluster performance tests, coolant purity may receive less emphasis than will precise flux control. In order to develop a concept for control of the reactor, the operating philosophy must be outlined for each test facility and its expected tests for all levels of operation, from startup through shutdown.

1.2.1 Closed Loops

Both fuel and materials may be tested in closed loops, in which isolation from the main heat removal system sodium is achieved.¹ Flow, temperature, and coolant chemistry will be independent of the main loop sodium. The type of test will influence some aspects of operation, as noted.

Prestart

Preheating of the sodium supply and loops will proceed in the same manner as for the main loops until suitable temperatures for filling are established. Normal electrical

1. Refer to References, Appendix A, Item 9.

preheating rates will be limited by component designs (estimated to be about 5 to 10 °F/hr for dry preheat--empty of sodium, and about 50 to 100 °F/hr for wet preheat). Automatic control of preheating is preferred in order to provide close control and conserve manpower during startup. This approach may be accomplished through proper programming of the digital data-logging computer. Manual control of preheating will also be available. The loops will then be filled to a level sufficiently above pump bearings for adequate lubrication, but limited in order to allow for thermal expansion. Filling of the loops may be done manually, since the loops have a relatively small sodium inventory.

Flow will be established using both redundant pumps in the primary and secondary of each closed-loop system as a function check for each pump. Preheating of the loops (by electrical pipe heaters, test inlet heaters, pump energy and shim heating mounted in the test package, if available) to the desired isothermal coolant temperature for startup will continue through the startup stage, in conjunction with the main system preheating. Test assembly design may influence heatup rates, though it is more likely that the tests will be required to conform to predetermined maximum heatup rates based on closed-loop design.

Sodium purification will be established and impurities controlled to the desired startup levels. It should be noted that some tests may require a particular impurity level before the test is inserted in the loop. The capability for purification of the closed loop sodium, independent of the reactor test section (i.e., from the fill tanks), is also desirable so that purification may continue (probably at a reduced sodium flow) during test handling or primary system

maintenance. Cold trapping will be used for removal of oxides¹ while hot trapping will be used for removal of carbon and oxygen. Oxygen content will be reduced to a level such that the plugging temperature (oxide precipitation temperature) is at least 100 °F below the startup temperature. Allowable carbon in sodium is dependent on the expected test surface temperature, and carburizing effects are monitored by analyzing materials specimens exposed to the coolant.

Startup

Preheating will continue until the desired startup temperature is achieved. In general, the preheat goal for the closed loop will be the operating cold-leg temperature for the loop. However, special tests may require as high a bulk temperature as possible throughout the testing period (including shutdown-restart). Thus, the preheat goal could be the desired hot-leg temperature and the test ΔT "spread" downward from the higher temperature as the FTR is brought to operating power. The maximum hot-leg temperature will be 1200 °F.

Flow will be established at a value consistent with protection against a startup incident, high enough to ensure good thermocouple response and adequate heat removal in the event of a startup reactivity incident (estimated minimum of 20 to 50% full flow). Closed-loop temperature and flow instrumentation will be observed during the approach to criticality of the FTR. Both pumps in each loop (primary and secondary) will be operating.

1. Cold trapping may also be an important means for fission product removal following tests to rupture. However, flushing the system is expected to be more effective.

To Power

In general, closed-loop flow and reactor power will be increased from the minimum flow in such a manner as to minimize thermal transients on the test section and closed-loop system. The extent of possible rise-in-power sequences is shown with the following two examples. The actual sequence used will reflect the needs of the test and provision may be required for a variety of sequences to satisfy different tests.

1. With full flow and the closed test loop preheated to the desired cold-leg temperature, the test ΔT is then increased by allowing outlet temperature to rise as the FTR power rises. The closed test loop cold-leg temperature is maintained by air flow control at the loop DHX.
2. With the closed loop preheated to the desired hot-leg operating temperature, the test ΔT is then increased by allowing cold-leg temperature to fall as FTR power rises, through heat losses at the DHX and by decreasing electrical heat.

The first approach is preferred as it is simplest from an operations standpoint, and parallels operation of the FTR. The latter approach may be desirable for tests which require continuing high temperatures, even during shutdown (e.g., materials tests for corrosion and mass transfer). Thus, versatility is needed in closed-loop approaches to power.

As temperature increases, sodium purification and impurity monitoring may continue since additional second-phase (solid) impurities would go into solution. Coolant purity will be established at the level desired for at-power operation. Oxide plugging temperatures should be below 300 °F in order

to ensure continued flow even at low temperatures (particularly in DHX tubes). Corrosion and mass transfer studies as well as fission product deposition studies may require addition of impurities. Such additions could be made before startup, or during operation with the aid of suitable remotely-operated devices.

At Power

Inlet and outlet coolant temperatures will be controlled within the limits specified by the test. As an example, mass transfer test data may require inlet and outlet temperatures to be controlled within ± 10 °F. Control system parameters will be adjustable in order to serve the purposes of different experiments. It is expected that inlet temperature will be controlled by DHX heat dissipation and inlet heater adjustment, and outlet temperature by primary flow adjustments in response to power changes. However, simulation studies (similar to those reported in Appendix B) will determine the best method of control, particularly from the standpoint of stability.

Both of the two coolant pumps in each heat transport loop will have control capability so that should one begin to fail, it may be shut down while the other picks up the total load. At steady-state the pumps will be operated at equal load.

Test ΔT and bulk coolant temperatures may be operated at different values from the FTR driver fuel, for both fuels and materials tests. Test section outlet temperature may exceed design temperatures of the closed-loop process systems (up to 1400 °F) with the use of bypass streams around the test section. Electrical heating, both in the test section

and in the process system, may be required for elevated temperatures (particularly in materials tests), in order to make up for the limited heat generation in the test specimen.

Coolant purity will be maintained at the desired levels for each loop by periodic or continuous monitoring and operation of purification systems, as required. Impurity levels required by the tests will be superseded only by the requirement for an oxide plugging temperature at least 100 °F lower than the coldest coolant temperature in the heat transport circuits.

Capability for the analysis of closed-loop cover gas is provided for monitoring vented fuel tests and tests to failure. Flow and temperatures will be monitored at all times.

Shutdown

Normal shutdown will proceed with minimal thermal transients as required by the tests, decreasing sodium coolant and air flows along with reactor power. The turndown limit of a single pump is expected to be at about 15% to 20% of full flow. Thus, one primary pump gradually will be taken off-line with the other pump providing decay heat removal.

Emergency shutdown (scram) of the FTR will be accompanied by a fast turndown of the test loop pumps in order to minimize thermal transients on the fuel test section. Should pump design preclude a fast turndown to the desired speed, the pumps will be fully shut down and the emergency electromagnetic pump will then provide decay heat removal. If the test is for nonfuel materials, flow may be maintained at full value, as there is little coolant ΔT to collapse.

Following scram, flow will be adjusted to maintain a ΔT on the fuel test coolant as close to the operating ΔT as possible, in preparation for a restart. For a total shutdown between operating cycles, the test coolant ΔT may be collapsed for the purpose of allowing higher, more controllable flow during shutdown. The philosophy of emergency (protective) actions, for both the tests and the reactor, is more fully discussed in Section 3.0 of this report.

Electrical heating and pump energy will be used to maintain the desirable isothermal coolant temperature for shutdown and the next startup. The desired shutdown temperature for closed-loop tests may be any value between cold- and hot-leg temperatures (500 to 1200 °F). For example, tests investigating the effect of irradiation or coolant ΔT on mass transfer may require a low temperature with the reactor shut down, in order to avoid diffusion effects during shutdown. Conversely, material strength properties tests may require high temperatures during shutdown in order to avoid metallurgical changes due to thermal quenching.

Handling operations during shutdown will prevent mixing of the loop coolant with the main coolant through proper use of sodium level control systems for both main and closed-loop sodium. Closed-loop flow will be reduced or interrupted for loop refueling operations. Overheating of the test specimen (i.e., test specimen surface temperature above in-reactor operating temperature) will be prevented during handling operations by providing adequate cooling during all phases of fuel handling.

If design requires closed-loop cover gas in the reactor section, the gas must be purged or purified to an acceptable impurity level (with particular regard to fission products) before the loop may be opened.

1.2.2 Open Test Positions

Fuel, materials, and instruments may be tested in the open positions, both in the core and reflector regions. Coolant chemistry is entirely dependent on the main sodium system, while temperature and flow may be variable with respect to the main system by the use of electrical heating or variable orifices in the test section (such devices are dependent on successful development programs). Driver fuel elements will be used in in-core positions where no tests are inserted. The following paragraphs consider the testing aspects of operation; in all other respects, the open test positions may be considered as a normal part of the FTR (i.e., as a driver element).

Prestart

Variable flow mechanisms will be thoroughly function-checked. Such flow control devices will be under the direct control of plant operations, rather than the experimenter, in spite of their incorporation into the test assembly. Test instrumentation and electrical heating in the test section will also be function-checked.

Testing desires may influence the main system coolant chemistry, although tests with stringent coolant purity requirements (e.g., less than 5 ppm oxide) will be placed in closed test loops. Test design may influence the system preheat rates though it is more likely that tests will be designed to withstand the expected transients.

Startup

Flow will be available in each test channel at its maximum value with respect to the main system (i.e., variable orifices will be at full flow conditions), consistent with the protective philosophy for guarding against startup accidents.

The presence of tests in open positions will influence the approach to critical only in calculations for predicting the inverse multiplication curve. The procedure for the approach to critical will remain unchanged.

To Power

Variable flow mechanisms may be used to minimize thermal transients on the test. This may be accomplished by restricting the test flow once criticality is achieved and then increasing test flow (apart from reactor flow) as reactor power is increased. It is recommended, however, that approach-to-power operation be simplified by limiting flow changes. Thus, the open positions will follow the driver core in terms of spreading ΔT for full power operation.

At Power

Inlet and outlet temperatures will be controlled to the desired values within limits specified by the tests (estimated to be within ± 10 °F). Extra electrical heating may be built into the test assembly to boost the inlet temperature to the test section itself.

Flow variations with respect to the main primary flow may be made to achieve test objectives (for those tests with built-in flow restrictors) within thermal stress limits of the open position reactor tubes. Primary users for such a capability may be materials testers, as corrosion and mass transfer tests are affected by coolant velocity. Maintaining a relatively constant velocity, however, may be performed manually since quick response to main loop flow changes is not necessary. Limits on the amount of flow restriction will be built-in to avoid starving the channel of coolant.

Flow and temperatures will be monitored at all times. Control elements adjacent to tests may be moved to change flux level and to examine test instrument response.

Shutdown

Before normal shutdown, test flow deviations with respect to the main primary will be returned to normal, depending on test desires. Open test positions will otherwise follow main primary flow and temperature changes, including scram response. Flow restrictors will not be operated during the shutdown period itself, in order to simplify shutdown activity of the operators. Elevated temperatures may be provided during shutdown by extra electrical heating in or below the test section, primarily for materials tests.

1.2.3 Axial Positioners

Axial positioners will be designed to raise and lower tests in a single closed loop and a single open test position.¹ The basic objective is to control the level and rate-of-change of temperature and neutron flux in the test section independent of the reactor. Thermal and flux cycles may be performed at a greater frequency than provided by the normal operating routine. Except for the following items, operation will be the same as for the closed loop or open position. Tests will be fuels, materials, or instruments.

Prestart

Flow and temperature instrumentation and the positioning mechanism will be thoroughly function-checked.

1. Refer to References, Appendix A, Item 8.

Startup

The following interlock for rod withdrawal will be established: positioner inoperative.

To Power

The axial positioner will remain inoperative until steady-state power is achieved.

At Power

Test motions will be performed only at steady-state power operation. Many startup and shutdown thermal cycles and flux cycles may be simulated with proper positioner speed. The rate of motion may be variable to accomplish differing experimental objectives such as a slow rate (on the order of 1 ft/hr) for simulating operational thermal transients, and a fast rate (on the order of several inches per minute) for simulating many operating flux cycles and testing instrument response. Motion of the tests depends on their reactivity worth and will be restricted to speeds producing reactivity changes easily handled by the nuclear control system (on the order of cents per minute as contrasted to cents per second). However, reactivity variations will be minimized by design of followers.

Shutdown

During normal shutdown, the positioner will be inoperative in order to assist in an orderly shutdown. In the event of a scram, the test will remain at its prescram position. Motion of the test may resume after full shutdown is achieved, in order to position the test for a restart.

1.2.4 Package Loops

Package loops (not provided by the FFTF) will be designed to be compatible with FTR open test positions. Fuel and materials specimens with small cooling requirements may be tested independently of main system coolant chemistry.

Prestart

Instrumentation will be function-checked. If a pumping unit is built into the test package, flow will be established. Sufficient main coolant flow through the open test position for heat removal from the package will be verified.

Startup

The test will not be treated in any special manner.

To Power

Temperatures will be monitored to check for adequate heat removal from the package. Internal flow may be adjusted to achieve the desired startup thermal transient on the test section. Flow control may be achieved by a built-in pumping unit or control valve for natural circulation, and will be operated independent of the main control system. Experiment design will preclude excessive temperatures (e.g., fuel melting) in the event of pumping unit or valve failure, by adequate natural circulation.

At Power

Internal flow may be adjusted to achieve the desired testing temperatures. Open position channel flow (the secondary coolant for the package loop) will most likely not be variable with respect to main primary flow. Neutron flux at the test may be adjusted by repositioning the control rod adjacent to the test. Flow and temperature will be monitored at all times.

Shutdown

Internal flow and/or electrical heating may be adjusted to keep test temperatures elevated during shutdown. Decay heating from the tests will probably be small. Internal flow control will not be related to the plant protective system and will not respond to a scram signal.

1.2.5 Short-Term Facility

The short-term facility will be of the "trail cable" type, and will be used for fuel and materials capsule testing at times less than an operating cycle.¹

Prestart

The short-term test facility will be function-checked for operability of instrumentation, test transport mechanism, and cooling system.

Startup; To Power

Test specimens utilizing the trail cable facility may be in or out of the core region. Tests will not be moved until steady-state power operation of the FTR is achieved, with one exception: foils for power-flux calibration may be moved at intermediate power levels.

At Power

After steady-state operation of the FTR is achieved, short-term facility tests are transported to the core region. Allowable speed of motion is dependent on reactivity worth of the specimen and the response characteristics of the

1. Refer to References, Appendix A, Item 10.

nuclear control system, which will compensate for the test insertion and withdrawal (current estimates of user needs in terms of fast insertion of tests give a maximum expected reactivity change rate of 1-2¢/sec).

The trail cable facility will be cooled by FTR sodium. Interaction between the trail cable channel and the remaining FTR channels (e.g., flow diversion with the test specimen recovered) is expected to be negligible. The test will be transported from the core region at the end of its required irradiation period (generally prior to shutdown) either automatically or manually. Manual control will have priority, i.e., the operator has the option of taking control away from the automatic control system.

Shutdown

Test specimens will not be moved during normal shutdown so that an orderly power decrease may be performed. In the event of a scram, cooling to the test will be decreased as reactor cooling is decreased in response to the scram.

1.2.6 Capsule Irradiation Positions

Additional capsules may be irradiated in the nosepieces of the FTR control and safety rods and in the axial reflector regions of the driver fuel.¹ Requirements for instrumentation or cooling for these capsules are not expected to affect overall control. The only area of influence of these tests on FTR operation will occur at power with capsules located in the tips of control rods. It may become desirable to

1. Refer to References, Appendix A, Item 8.

reposition the rods vertically, in order to maintain the flux at a prescribed level for the test. However, control requirements as to rod positioning will take precedence over test objectives (e.g., positioning rod at the point of maximum differential reactivity worth for controllability reasons). In addition, specimen design will have no influence on speed-of-control requirements. Rather, tests will be designed as to reactivity worth in order to meet requirements already set by speed-of-control.

1.3 PLANT OPERATION

The following discussions are based on the assumption that all equipment and facilities are operating normally and that no serious malfunctions are present. Responses to equipment failures or to corrective control actions, such as scram or controlled shutdown of the reactor, are discussed in a later section of this report.

Current approaches to plant operation could change somewhat as the design evolves. For example, the determination of the normal mode of operation of the FTR (e.g., automatic or manual control at power) and its associated heat dissipation systems must consider the following:

- Coolant transport time in the primary and secondary systems.
- Interactions between heat transport systems and the reactor.
- Thermal stress limitations imposed by design and safeguards.

However, testing objectives and conceptual design are sufficiently firm for defining an operating philosophy with confidence.

Figure 1-1 shows the relationship of the process variables of the reactor and the heat transport systems. Only a single heat transport circuit is shown, as each circuit should respond to plant control in an identical manner. The basic problem of control design then becomes defining what control loops are needed between the operational objectives and the controlled process variables. In the following discussions, reactor and heat transport system operations are considered in order to define operational objectives. Operation of auxiliary systems (such as sodium and gas purification) and support systems (such as fuel handling) will not be emphasized except where they affect operation of the reactor and the heat removal systems.

1.3.1 Reactor Operation¹

Prestart

During shutdown operation, the prime efforts will be refueling, maintenance, and plant modifications. Fuel handling will require most of the shutdown cycle to complete the required changeout and rotation of driver fuel, as well as removing and installing test assemblies. The refueling sequence will proceed at its own pace and other activities will generally defer to the fuel handling needs. During this period, the following conditions will be maintained:

- The primary flux instrument at this time will be a sub-critical neutron monitor (fission chamber) that will have an on scale reading at all times.

1. Refer to References, Appendix A, Item 11, for reactor concept description.

FIGURE 1-1. Schematic Diagram of Reactor and Heat Transport System Relations

- The period circuit will provide an alarm if the period exceeds the shutdown period set point.
- The flux monitoring circuit will provide an alarm if the flux level exceeds the shutdown flux set point.
- The subcritical reactivity will be calculated from the subcritical neutron monitoring signal, and will be displayed for continuous observation by the operator.
- The safety circuit will be tripped (the state for initiating a reactor scram).
- All nuclear control rods (safety, shim, shim-safety, and regulatory) will be fully inserted.
- The main heat transport systems will be adjusted to maintain the primary and secondary coolant temperatures constant between 350 and 400 °F.

It is expected that a number of control elements will be withdrawn occasionally during the refueling cycle to calibrate the subcritical reactivity monitor and to assure that the monitor is operational. This will require the capability and circuitry to withdraw individual poison rods during this interval of time. A desirable alternative to this procedure is to use a direct, on-line monitor of the shutdown reactivity, allowing poison rods to remain inserted throughout shutdown. The feasibility of obtaining such a device before initial plant startup is small.

The reactor will be preheated by electrical heating on piping and pumping power in the primary loops, in addition to vessel heating by circulating hot gas exterior of the vessel. Electrical immersion heaters may also be employed. Heatup rates for the core will be limited (estimated to be 50 to 100 °F/hr) to minimize thermal stresses on fuel and core components. Preheating will continue to the desired

startup temperature. This temperature will be the reactor inlet temperature for the next test run (as high as 900 °F) in order that a maximum amount of thermal expansion of reactor and systems may take place before control rods are withdrawn.

Startup

Before the approach to criticality is made, the values of the important process variables are checked to assure that equipment and instruments are in good operating condition. During this period the plant protection systems are reset to satisfy all interlocks for withdrawing control rods.

Most of the status checks for interlocks and process variables will be accomplished using computer-programmed sequences to determine that all equipment is in proper condition for operation. Means for performing these checks manually will also be provided.

As the checks proceed, discrepancies will be alarmed and require either operator acknowledgement or correction before the checkout procedure continues. When the entire check has been completed, the check procedure may be rerun, possibly on a limited basis, to assure that no changes have occurred on critical parameters or devices. With all systems information being available in the control room, the digital data handling equipment will provide this service.

The approach to critical will follow the sequence below:

1. The safety rods will be withdrawn singly at a predetermined rate of motion.
2. The rod which will be used during normal operation as the regulating rod will be withdrawn at a predetermined speed to its predetermined initial operating position.

3. The shim-safety rods will be individually withdrawn to their predetermined initial operating positions.
4. The proper number of peripheral shim rods will be individually withdrawn in a predetermined sequence designed to result in a balanced core flux.
5. During the above rod withdrawals, the rod positions and subcritical reactivity (derived from the neutron flux level) will be correlated by a computer program. A significant lack of correlation will be annunciated and the sequence will be halted.
6. The withdrawal of the peripheral shims will continue until the desired true period is obtained. This period will be on the order of 30 to 100 sec and will be maintained until the reactor power level is approximately 1/2% of full power (approximately 2 MW).
7. At this point, a peripheral shim rod will be inserted and the power will be stabilized at about 1% of full power (approximately 4 MW). A full systems checkout will then be made to determine the readiness for power increase.

During startup, the controlling variables to be observed will be the low level (subcritical) flux and the apparent reactor period until the reactor goes critical. After the reactor goes critical, the intermediate range flux period and level will become the controlling parameters. At the time of transfer from the low level instrumentation to the intermediate range instruments, both instrument sets must have on-scale readings and must agree with each other. At 1% of full power, the linear power range instruments must also be on-scale and in agreement.

Ascent to Power

Following the full systems checkout, the regulating rod is withdrawn to establish a reactor period between 50 and 200 sec. As reactor power increases, the heat transport systems for main and closed test loops will respond to the increased heat by changes in DHX air flow rate in order to dissipate the heat and maintain proper temperatures. As reactor power approaches full power, the period will be increased to limit the magnitude of the power change with time. All reactivity adjustments will be made with the selected regulating rod.

The ascent to power from roughly 1 to 100% of operating power requires a reactor power increase that does not exceed the limit of either the minimum acceptable period (estimated operational limit at 30 sec) or the maximum acceptable operational thermal transient (estimated at 100 °F/hr for bulk sodium temperature). Within the overall bulk thermal transient, a maximum rate of change will also be specified (estimated at ± 30 °F/min for the fuel coolant outlet temperature of the initial core), based on the large number of startup-shutdown transients expected over the plant life (estimated at 2000). Period and thermal transient limits will be reevaluated as reactor design progresses.

In order to limit the rate of temperature increase, it may be necessary to have holding points (e.g., at 25% increments of full power). For example, with three heat transport circuits in operation at full flow, the full power conditions for the initial FTR core are 400 MW_t and 300 °F ΔT . If full flow is maintained during the rise to power, 3 hr are needed to spread the ΔT to meet a thermal transient limit of

100 °F/hr at the core outlet. Since the power would be increased from 4 MW_t to 400 MW_t in 15 min while on a 200-sec period, holding points appear to be needed unless thermal transient limits are relaxed.

Initially, the approach to power will be performed manually. However, provision should be made for automatic rise to power based on preprogrammed power and flow control.

At-Power Operation

While at power, a single regulating rod will continually be positioned to maintain the reactor at the desired power level. Neutron flux will be controlled to within a specified deadband (estimated at ±2% maximum) about the mean level, to achieve testing flux goals, and to provide ease of control for the entire heat transport system. Larger variations will not be desirable because of the resulting temperature variations and their effects on coolant flow control. Linear power range flux instrumentation, entirely apart from the protective instrumentation, will provide the feedback for this control loop. Automatic flux control will be provided as well as manual control, with manual control having priority (i.e., the operator may switch to manual at any time).

Neutron flux for any test loop will not be allowed to vary more than ±15% from the testing goal during the operating cycle. These are long-term local variations, due to burnup and the resulting control repositioning, as contrasted to oscillatory behavior of the average reactor flux about a mean value. Neutron flux may be changed at a test location by multiple control rod position adjustments. Although such rod positionings have a limited effect on the flux at

adjacent lattice positions (estimated at 5 to 10% change for a rod full-in to full-out for the FTR), the effect is sufficiently strong to be considered in the design stage. Since the regulating rod should not be fully withdrawn, the resultant need is for regulating ability on each shim control rod, as opposed to choosing a single rod for exclusive regulating duty.

As fuel burns up, the regulating rod must be further withdrawn to maintain criticality. When the regulating rod approaches its upper operating elevation limit, a shim rod will be raised in order to allow the regulating rod to be returned to the lower elevation part of its operating range. The desired operating range for the regulating rod is estimated to be between 20 and 80% withdrawn, based on limits at one-half the peak differential rod worth. Any of the in-core shim rods or regulating rods on the periphery of the core may be used as the regulating rod.

As an additional operator aid, reactivity changes with respect to a reference core condition will be continuously computed from the appropriate temperatures, flows, and power, and compared to that computed from control rod positions with respect to reference rod positions. Deviations between the two values will be alarmed so that the operator may be made aware of reactivity anomalies (e.g., shifting of fuel).

Normal Shutdown

The reactor is shut down to replace fuel, to reposition fuel, to replace or check experiments, to replace poison rods, or to perform scheduled or emergency maintenance. Normally, the power level is reduced to about 1% of full neutron power. The reactor may then be scrammed as a check on the safety system, with minimal thermal transients.

The shutdown sequence will be roughly the reverse of the rise-to-power procedure. Holding points in the power descent may again be prescribed to ease thermal stresses. The shutdown sequence will normally proceed according to a programmed plan and any variations from expected limits will be alarmed and displayed. Flux instrumentation will be phased from the power range to the intermediate range monitors.

1.3.2 Heat Removal System Operation

The heat transport systems for the reactor driver core consist of three complete circuits, each having a primary coolant loop, a secondary coolant loop, and a sodium-air tertiary heat dump (DHX). Each DHX will have four modules operating in parallel, each with its own air blower.¹

Precise control configurations for the control of individual flows or temperatures in the heat removal system are not well defined at this time because of the early design stage for FFTF. It is possible, however, to discuss desired operating techniques and point out special problem areas. Conceptual control configurations may then be developed (see Section 2.0).

Prestart

All heat transport systems and components will be function-checked for operability and control (e.g., check of pumps over expected flow range). Preheating of the systems will proceed up to the desired operating cold-leg temperature,

1. Refer to References, Appendix A, Item 12.

in order to achieve as much of the expected thermal expansion as possible before starting the plant. Preheating will be automatically controlled, utilizing the digital data logging system, in order to conserve operator time and effort during the busy startup period. Electrical pipe heaters and coolant pump energy will both be used. During this period, DHX air flow will be restricted to conserve power, speed preheating, and minimize DHX sodium cold spots. System preheat rates will be limited (estimated at 100 °F/hr).

Startup

All of the primary loops that are to be utilized during the run are on-line and flows and temperatures are stabilized by adjusting air flow and electrical heating throughout the system.

Minimum flow through the core (suggested as 50% full flow) will be established and maintained throughout reactor startup, consistent with protective requirements for a startup reactivity incident and with the capabilities of FTR individual channel flowmeters.¹ Each reactor channel will be monitored for adequate flow before reactor startup. All interlocks pertaining to the heat transport systems will be satisfied: minimum flows, component operability.

The heat transport systems will remain on manual control throughout the reactor startup and up to 1% of full power. At this power level, the heat transport systems may be placed on automatic control for the approach to full power.

1. Refer to References, Appendix A, Item 2, p. 35.

Ascent to Power

The ascent to power from roughly 1 to 100% of operating power requires that the heat transport system respond to the reactor power increase within the limits of the acceptable thermal transients (present estimate at 100 °F/hr). This applies to the closed test loop heat transport systems as well as the main systems. Sodium and air flows may be controlled to achieve full power conditions in a variety of ways. Several will be considered here.

Figures 1-2 through 1-4 show three ascent-to-power approaches which might be used to meet the system startup goals. The goal of any of these combinations is to start at the conditions of coolant flow and temperature which exist at very low nuclear power levels and obtain the required coolant and flow conditions at full nuclear power with minimum time and thermal stress and maximum ease of operation and safety.

Figure 1-2 shows a sequence in which the core ΔT is developed at a relatively low power level. Initially, the primary coolant flow is at about 50% of rated flow and by means of pipe heating the inlet temperature is adjusted to the desired cold-leg operating temperature. As reactor power is increased, the core ΔT increases proportionally with power. Core inlet temperature is held constant by DHX air flow, and the increase in core ΔT raises outlet temperature. When outlet temperature is at the desired operating value, then the coolant flow is increased in proportion to power to hold the outlet temperature constant.

An advantage of such an approach is that all core thermal expansion would occur at a low power level. A disadvantage is that power and flow must be increased simultaneously,

FIGURE 1-2. Heat Transport System Startup Response, Spreading Core ΔT at Low Power.

FIGURE 1-3. Heat Transport System Startup Response,
Holding Core Center Temperature Constant.

FIGURE 1-4. Heat Transport System Startup Response with Constant Flow and Inlet Temperature.

increasing control complexity. Outlet temperature changes will be limited (estimated at 100 °F/hr), requiring a relatively long time period (3 to 4 hr) to increase power to 50% of full value.

Figure 1-3 shows a sequence basically the same as described in Figure 1-2. The goal of this sequence is to hold the core coolant average temperature constant. As reactor power increases, inlet temperature has to be controlled downward as outlet temperature increases, by increasing DHX air flow. Again, the initial sodium flow would not be changed until the core ΔT is established.

This sequence subjects the core outlet to thermal transients that are less than the transients of the sequence of Figure 1-2 for a comparable rate of power increase. Therefore, for the estimated limit of 100 °F/hr, the approach to full ΔT may be performed in half the time. Core inlet temperature, however, will experience a thermal transient equivalent to that at the outlet. The reactivity effects due to heating in the upper core are partially offset by reactivity changes due to cooling in the lower portion of the core. The chief disadvantage is that one must control two variables (inlet and outlet temperature) as a function of power level from approximately 1 to 50% of power level.

The sequence of Figure 1-4 provides for rated coolant flow at the beginning of the approach to full power. The core ΔT is allowed to increase in proportion to reactor power. The desired core ΔT is not attained until the desired reactor power is achieved.

The advantage of this sequence is the simplicity of control, as the core inlet temperature is maintained constant by the DHX and the flow is held constant. From the standpoint of

ease of operation, this sequence is preferred as the operators are free of routine matters in order to be more vigilant of safety concerns. It is assumed that the secondary coolant loop has a similar sequence with constant secondary flow at full value. Therefore, the coolant transport delay time between the dump heat exchanger and the primary cold-leg temperature is reduced to a minimum. Control stability of the reactor inlet temperature is a function of this transport delay time, and is greatly enhanced.

The recommended approach to power for the FTR is that of spreading ΔT with power, while maintaining constant full sodium flows and constant inlet temperature (Figure 1-4). Ease of operation and control stability are the chief advantages of this scheme. Should testing desires dictate, a different approach (e.g., that of Figure 1-3) may be employed for the closed test loop heat transport systems.

The most difficult problem in approaching full power operation is getting the Na-air heat dumps on-line. At the start of the approach to full power, DHX air flow is limited (fans off, fan outlet dampers closed, stack outlet dampers closed) in order to minimize heat losses and reach the desired operating temperatures. When the desired cold-leg temperature is achieved, the heat dumps must be put in operation to maintain that temperature.

Current DHX design calls for constant-speed fans; therefore, startup of a DHX fan must precede opening of dampers. Control will be effected by changing fan inlet vane angle. The best approach to power is by starting all four fans of the DHX, with full sodium flow through each module, with all dampers closed and with inlet vanes in the lowest flow position. Stack outlet dampers will be opened first on all

modules. Next, blower outlet dampers will be opened gradually in response to heat load (up to an estimated heat load of 5 to 10% full load). Finally, inlet vanes will be adjusted as heat load increases to full power. In this way, all DHX modules would be brought on-line simultaneously. Should such an approach prove to be unsuitable because of coarse control steps, one module at a time could be brought on-line by employing the block valves on the sodium side. As each succeeding module is started, however, care must be taken in starting sodium flow through the module in order to avoid thermal shock at the inlet header. At the desired operating power, the modules will be adjusted to carry an equal portion of the heat load.

Normal Operation at Power

After steady-state operating conditions have been achieved, the control and data handling system will begin a set of routine checks and calculations designed to assure the desired plant status. Included in these calculations are heat balances at key points in the heat removal system (vessel, IHX, DHX), and checking each test environment to see that it falls within the range required by the individual experiment. Secondary flow of each heat transport circuit will be nearly equal to its corresponding primary flow in order to minimize IHX tube sheet thermal stresses. At the IHX, secondary sodium pressure must be greater than primary sodium pressure under all circumstances of operation.

Normal at-power operation of the heat transport systems (both main and closed test loops) is expected to proceed with the following objectives: constant power, constant core inlet temperature, and constant outlet temperature.

User requirements for open test position temperatures are estimated to be on the order of ± 10 °F as a maximum deviation from test specifications. The integrated process system controls will meet at least this requirement for all reasonable perturbations expected during operation (excluding those produced by incidents such as equipment failure). Example perturbations to be analyzed during preliminary design are as follows: (1) DHX air inlet temperature change of ± 20 °F in 2 min, (2) DHX fan voltage changes of $\pm 2\%$ at 1 to 5 Hz, and (3) coolant pump voltage fluctuations of $\pm 2\%$ at 1 to 5 Hz. This list will be expanded as the FFTF system design progresses.

The control system will allow the operator to put on manual control any control loop or system that requires special attention due to nontypical operations. All such systems would continue to be monitored by the control and data system with all alarms and diagnostics in service. During this period of manual operation, the operator would have all such data displayed to him but all decisions for action remain with the operator. Operator actions will be recorded as permanent record.

Special transients or variations in plant thermal-hydraulic conditions, either short-term (hours) or long-term (days or weeks) will be accommodated as a special programmed operating sequence. Such special experiments, however, will mostly be restricted to the closed test loop systems. It is expected that the main systems will generally maintain constant system parameters (temperature, flow) over the operating cycle, except in response to desired reactor power changes.

Shutdown

Shutdown of the systems will follow roughly the reverse of an ascent-to-power sequence. The typical shutdown of reactor power will involve reducing the nuclear power of the reactor and bringing coolant temperatures and flows to the conditions compatible with the shutdown operations scheduled. Figures 1-5 and 1-6 show example sequences for variable during shutdown.

Figure 1-5 shows the outlet temperature decreasing as a function of power until the power level is between 1 and 10%. At this point, the reactor may be scrammed with full flow, as the resultant transient is relatively minor (10% full ΔT produces about -30°F in 10 sec at the core outlet for the initial core). The reactor ΔT is then effectively collapsed, and bulk temperature is maintained by electrical heating and pump energy. This shutdown sequence will require the reactor to experience thermal transients while the reactor is still at appreciable power levels, but is simple and stable from the control standpoint.

Figure 1-6 shows the coolant flow decreasing as a function of reactor power in order to maintain the core ΔT at a constant value. Under these conditions the core is not subjected to thermal transients until the reactor is at a very low nuclear power, although two variables must be controlled.

The recommended shutdown from full power is that of collapsing the core ΔT with power (Figure 1-5), essentially the inverse of the approach to power. The same justifications of operational simplicity and control stability hold true.

FIGURE 1-5. Heat Transport System Shutdown Response with Constant Flow and Inlet Temperature.

FIGURE 1-6. Heat Transport System Shutdown Response by Holding Core ΔT to Low Power.

SECTION 2.0 PLANT CONTROL SYSTEMS

Section 2.0 of this document draws on the information presented in Section 1.0 to present a preferred concept for overall plant control and for individual system control. The various levels of control are defined first. This is followed by a simplified diagram of the complete plant control scheme. Additional discussions and diagrams are then presented on control of individual systems including reactor flux, main heat transport primary heat removal, secondary heat removal, tertiary heat removal, and closed test loop control systems.

2.1 GENERAL CONTROL ORGANIZATION

Overall control of the FFTF will make use of multiple-level control coordinated in such a way that the functions of each level of control are clearly described. Such a control hierarchy is shown in Figure 2-1 where three levels of control are defined.

Parameter control defines the requirements and configurations to control a given process variable within operational limits for that variable. The measured variable and the process control actuator for each variable will generally be determined at this level. System control defines configurations and procedures related to a given process system that is not provided when individual parameters are considered by themselves. Interactions between system variables and methods for using them to best advantage would be defined and recommended at this level of control coordination. Plant control, in turn, provides control coordination between the various process systems and defines the plans

FIGURE 2-1. Plant and Process Control Hierarchy; Functions of Each Level.

or procedures to be used by the facility to achieve operating and testing goals. Examples of each control level may be outlined as follows.

In parameter control, the individual parameters are sensed and compared to set points for those parameters. These comparisons provide error signals which are used to position process control actuators so as to reduce the errors between the parameters and their set points. The adjustment of DHX dampers so that air flow matches the flow set point is an example of parameter control.

System control considers all the individual parameter control loops within a given process system and determines how each loop shall operate to attain the needs of the process system. For example, the heat transport primary system temperature measurement is the input to a controller whose output is the set point of an air dump temperature controller. The feedback control loop on the air heat dump provides a relatively fast response to local disturbances. The primary system temperature controller, which adjusts the air heat dump set point, then assures that the tertiary cooling responds rapidly to changes in the plant heat removal requirements as indicated by the primary system temperature measurement. This is an example of a multiple-level (cascade) control confined to a single process system or system group. In its simplest form, system control would be accomplished by direct operator action. In a more sophisticated form, system control would be automated to some degree.

At the plant control level, system controls and, therefore, the parameter controls of the various systems will be

coordinated as required to achieve plant objectives. For example, the main heat removal rates must respond to changes in the nuclear power level so that temperature changes which would damage plant components or disturb plant operation do not occur. The desired flow and temperature set points of the main heat removal systems will probably be functions of reactor power level. These set point variations may be made manually by plant operators using operating charts or tables, to determine required set point changes, or calculations made by computers. They may also be made automatically by computer-based or other special program controllers. In like manner, the nuclear flux level must be constrained to given rates of rise or periods because of safety, operating and testing requirements and, possibly, because of temperature rise rate limitations. While the individual parameter control loops and system controls act to maintain selected measured variables at their set point values (i.e., to move rods to adjust flux or change pump speed to control flow and/or temperature), the set points to these local controls will be determined at the plant control level as a function of the current facility testing requirements and the corresponding required operating conditions.

Thus, plant control is defined by the testing and operating philosophy and is implemented by integrating the various control systems, either by operator action or by computer. An overall functional control picture is developed in the next section. Succeeding sections examine the individual control systems.

2.2 OVERALL PLANT CONTROL

The overall plant control system is functionally diagrammed in Figure 2-2. Neutron flux level of the reactor is adjusted to meet the requirements of a particular testing cycle by control rod manipulation. Main primary and secondary loop flows and levels are regulated to be within design restrictions. Tertiary (air) flows are maintained at proper values to provide the required heat dissipation. Closed test loops are controlled in much the same manner, with the ability to achieve different temperature and coolant velocity objectives than for the driver core. Other test facilities are controlled for their special requirements (e.g., required speed of insertion for short-term specimens).

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 transport delays in the order of seconds. The secondary coolant loops span the distance from within the containment sphere to the air heat dumps. This loop has 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 and oxide plugging are to be avoided. Heat dump characteristics which must be considered in the control of these units are:

- The units must be designed for efficient heat dissipation and, therefore, 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 heat dump units must include auxiliary heating of the air dumps until reactor power is high enough to provide a sufficient heat source. When the reactor power is reduced for shutdown, the sequence for removing heat dump units will generally be in the reverse order to that followed for the startup sequence.

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.

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 checkpoints and 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.

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 or operational tasks will be designed for remote operation or be located such that access is available whenever required.

The plant control configuration of Figure 2-2 is examined in greater detail at the system control level in the following sections. It must be stressed that control designs recommended at the current stage of conceptual design are likely to be changed as design progresses. Such configurations are presented here to show some prospective solutions to the design problem of how to move from an operating philosophy to a control design. Initial analysis of overall plant control is presented in Appendix B. In this study (utilizing a "hybrid" computer model of the FTR and heat transport systems), several process control schemes were evaluated for response to expected operational transients. The best scheme is presented in Appendix A and represents a possible approach for uniting the control loops of Figure 2-2 into a unified plant control; further simulation studies will be performed as FFTF design progresses.

2.3 REACTOR NUCLEAR POWER CONTROL

Nuclear power control¹ is achieved by sensing flux level and moving control rods to maintain power level at the set point value. The control console equipment consists of flux parameter displays (reactivity, power level, period) and manual and automatic control capabilities. The flux sensors cover the nuclear flux from the low level range (10^{-8} to $10^{-3}\%$) through intermediate range (10^{-5} to 100%) to the linear full power range (1 to 300%). The intermediate range and linear power range channels are separate and independent of similar sensors used in the protective system. The rod control logic provides automatic sequencing of rod drives during controlled power reductions and interlock functions to meet reactivity insertion rates during normal operation. The rod drive is the powered unit that positions the control element. Rod position transducers are considered to be part of the rod and drive assembly. Figure 2-3 shows schematically the major hardware features of the nuclear power level control.

In manual control, the operator observes on the displays the power level and period and adjusts control rod positions to achieve and maintain the desired power level. The operator has direct access through the rod control logic to the rod drive actuators when in manual control. In automatic control, the operator selects the feedback loop to control the regulating rod position. When controlling with flux as a feedback signal, it is possible to calibrate the steady state flux signal against calculated thermal power to give

1. Refer to References, Appendix A, Item 5.

the system the response capability of flux control and the accuracy of thermal power control. The rod to be driven and its speed are selected by the operator by means of the rod control logic the same as in manual control except that under certain conditions, a change of rod or rod speed or a transfer from auto to manual might be required automatically. When the designated control rod reaches the end of its defined operating zone and a backup rod has been designated, transfer of the control function to the backup rod can be automatic. Rod drive speed may be one value if the rod is moved in (i.e., reactivity decreased), and a different speed if the rod is to move out (i.e., reactivity increase). In the event of a nonmovement of a rod in response to a control signal, an alarm will sound and the system will transfer to manual mode for operator action.

Power setback is a controlled reduction in power level and is not considered to be accompanied by programmed changes in other process variables such as flow. The other process variables respond to the power change in such a manner that they maintain the established set points. Power setback will have two modes: percent reduction of power, and continued power reduction until the initiating condition clears. The first mode is operational during periods of automatic neutron power control, and is initiated in response to a process failure such as the loss of a DHX module. It is accomplished by automatically setting the neutron level set point to a predetermined lower value corresponding to the incident. The reactor control system then seeks the new level automatically. The second mode (continued power reduction) is initiated in response to reactor overpower conditions, and continues driving control rod(s) in until the overpower condition is corrected.

The programmed shutdown is a timed and sequenced event that is intended to completely shut down the reactor and associated processes. This is performed by the central control equipment and takes precedence over any normal control effort of either an automatic or a manual nature. The controlled shutdown is intended to provide a slower shutdown with reduced thermal transients in lieu of a reactor scram. During a controlled shutdown, the protective system can still initiate a reactor scram and override the controlled shutdown. The power setback and the programmed shutdown are briefly described here to provide a more complete picture of the reactor flux control. These two modes of operation are more thoroughly discussed in Section 3.0.

2.4 PRIMARY HEAT REMOVAL CONTROL SYSTEM

2.4.1 Primary Coolant Flow Control

The coolant flow control for the main heat removal system primary loop¹ is designed to provide the necessary coolant flow to the reactor and, at the same time, assure flow balance among the loops.

Figure 2-4 shows a controller arrangement whereby the total coolant flow rate is adjusted as a function of power level or in response to other inputs such as core ΔT or operator judgment. The flow control for each individual loop receives its set point from the total flow controller and provides an increase or decrease in flow together with the other loops. Balance between the individual loops is achieved by fine adjustment of set point bias controls on each loop controller.

1. Refer to References, Appendix A, Item 12.

FIGURE 2-4. Main Heat Transport System Primary Coolant Flow Control.

The flow control bias inputs have manual capability or automatic response to calculated system heat balance. The total flow controller set point is manually or automatically adjusted in response to operators' requests or computer supervisory control. The actual relationship of this set point is not defined at this time, but could be a function of system temperature, reactor power, etc.

The coolant flow response to emergency conditions is provided either by transfer of control set point to an emergency set point source or by transfer of the pump speed controller input to a separately generated controller signal. A programmed shutdown would be accomplished by transfer of the total flow set point input to the programmed shutdown sequences. A scram would use a direct input to the pump speed controller to reduce the flow so that failures of the normal control hardware would not prevent a flow reduction.

2.4.2 Primary Level and Pressure Control

Figure 2-5 shows the main heat loop primary coolant level and cover gas pressure control configuration. Level control in the reactor vessel is accomplished by means of an overflow in the reactor vessel and continuous Na fill for the system to the reactor vessel. The sodium inventory is maintained by spilling any increase in volume due to temperature increase during startup and providing sodium-fill flow to compensate for volume contraction during a normal plant shutdown.

The sodium level in the primary coolant pumps will change as a function of primary coolant flow. This change of static head in the pump is an offset to the changing

FIGURE 2-5. Heat Transport System Primary Coolant Level Control.

flow-induced pressure drop between the reactor vessel and the coolant pump. The primary coolant pump is basically a Fermi type with the pump barrel on the suction side of the pump.

The sodium level change in the pump barrel due to rated loop flow is operationally acceptable. In the event of pipe rupture in the primary loop the sodium level in the pump barrel will drop far enough to prevent siphoning the sodium from the reactor vessel.

The cover gas pressure for the pump and the reactor vessel are equalized by means of a gas header connecting the two volumes. The cover gas pressure will be controlled using gas inlet and vent. If the gas pressure is to be maintained at or near zero psig, the gas vent must be to a vacuum to facilitate adequate gas flow.

2.5 SECONDARY HEAT REMOVAL CONTROL SYSTEM

2.5.1 Secondary Coolant Flow Control

The secondary heat removal loops on the main heat transport system¹ are separate from each other except for the sodium purification system. Each secondary loop consists of the secondary side of the intermediate heat exchanger, the primary side of the sodium-air heat exchanger, the secondary coolant pump, and interconnecting piping.

The independent secondary coolant loops normally will be balanced so that each loop is transporting the same amount of heat. The heat balance will be checked using power

1. Refer to References, Appendix A, Item 12.

calculations and system temperature profiles. The control of the individual secondary loops, however, must be so arranged that each secondary loop can respond individually to transients in its own primary or tertiary coolant loops.

Figure 2-6 presents schematically the secondary heat removal loop and the flow control configuration. Included also on this diagram are the sodium level control features. Flow control is obtained by sensing the loop flow and referencing this to the flow set point. Pump speed is adjusted by the flow controller to achieve the desired flow rate. The set points for the individual secondary flow controllers will be generally equal in value but independent from each other.

The set point for the secondary loop flow control will be proportional to the coolant flow set point on the primary loop with which the secondary loop is associated. Such a configuration of flow control would provide for a consistent ΔT throughout the heat removal system. The source of the secondary loop flow set points can be the primary loop master flow set point or the individual primary coolant flows.

The coolant flow response to a controlled shutdown will be programmed by transfer of the flow set point from the normal source to the programmed shutdown sequences. Scram response will be provided by means of a direct input from the scram circuitry to the pump speed controller.

2.5.2 Secondary Level and Pressure Control

The secondary coolant system has only one free sodium surface, and this is in the coolant pump surge tank. The IHX and the sodium-air heat dumps will be filled containers;

FIGURE 2-6. Heat Transport System Secondary Coolant Flow and Level Control.

however, they may have gas return lines (not shown on diagram) to the surge vessel to prevent gas spaces developing in these components.

Figure 2-6 includes the level control configuration. The secondary loop surge tank accommodates sodium volume changes when the system temperature changes. The system pressure and the pump suction pressure is controlled by means of the pump surge tank cover gas pressure. The pressure is sensed and gas admitted or vented from the cover gas space to maintain the desired system pressure. If cover gas pressures near zero psi gage are required, the cover gas vent must be to a vacuum system to assure venting of gas for pressure control.

2.6 TERTIARY HEAT REMOVAL CONTROL (SODIUM-AIR HEAT DUMPS)

The tertiary system consists of finned sodium piping, a blower which forces air across the heat dissipation fins, and the housing and exhaust stack that confines and directs the air flow.¹

Figure 2-7 is a hardware and control configuration of the tertiary air dumps. The variable vanes on the air blower are controlled to maintain the desired air flow to the dump heat exchangers. The actual air flow is determined as a function of DHX sodium outlet temperature so air flow is adjusted to maintain the desired outlet temperature. The reactor vessel outlet or IHX primary outlet temperature will provide the DHX outlet sodium temperature set point during automatic control.

1. Refer to References, Appendix A, Item 12.

FIGURE 2-7. Tertiary Na-Air Heat Dump Control.

A fire or a sodium leak in the tertiary air dump would initiate the closing of both the upper and lower dampers, and the heat transfer space would be filled with inert gas. Valves on the secondary sodium piping provide flow balance between the four parallel air dumps, and in the event of a sodium leak, will provide isolation capability. During reactor shutdown, the dampers will be closed and the space heated to prevent freezing of sodium in the secondary piping.

Control or operation of the sodium-air heat dumps in response to scram or programmed shutdown will be sequenced in a predetermined manner by transferring set points or actuator inputs from the normal source to a special circuit or sequence for the particular corrective action. Definitions of startup and shutdown or scram sequences are dependent upon DHX design and are not covered at this time. Possible requirements for determining the sequencing are developed in Section 3.0.

2.7 CLOSED TEST LOOP CONTROL

A closed test loop consists of an in-reactor test loop, a parallel out-of-reactor test chamber and the pumps, piping, and heat exchangers needed to provide a single heat transport system with a primary and a secondary coolant loop.¹ Each coolant loop uses two parallel pumps to assure coolant flow in the event of a single pump failure. The sodium-air heat dump uses parallel air blowers to a single stack for heat dump backup. The closed test loop control concept as shown in Figure 2-8 is to treat the test section ΔT and the

1. Refer to References, Appendix A, Item 9.

test inlet temperature as the basic controlled variables. The method is to control inlet temperature constant by DHX air flow and then control test ΔT by variations in Na coolant flow. For any given test assembly at a constant heat generation rate, once a desired ΔT is specified the coolant flow rate is also determined.

The inlet temperature to the test section is controlled by adjusting the heat dissipation rate (air flow rate) in the sodium-air heat dump. In addition a separate electrical heater is used on the inlet to the test section for fine temperature control and for additional heat input if the desired inlet temperature is higher than can be tolerated by the primary loop component design.

The test outlet temperature is controlled by using the outlet temperature to provide a primary coolant flow set point. Such an arrangement of ΔT control by adjusting coolant flow rate requires a convenient means of scaling the controller and signal ranges to accommodate the various ΔT coolant flow combinations for tests of various heat generation capabilities.

The out-of-reactor test chamber is intended to permit a second test assembly to be run at the same time with coolant flow and ΔT the same as the in-reactor test. The two assemblies can be evaluated to separate irradiation effects from temperature and flow-induced changes.

Control for the out-of-reactor test chamber consists of a flow sensor and control valve that are independent of the coolant pumps and an electrical heater in the test chamber to produce the desired test ΔT . Control interactions between the out-of-reactor test chamber and the closed test loop primary system are not considered to be significant, as the out-of-reactor flow is about 10% of the total.

The parallel primary coolant pumps operate together from a common control signal which uses loop flow as the control variable. In the event of a primary pump failure a check valve prevents backflow through the failed pump, and the remaining pump continues to supply coolant to the test assembly. An electromagnetic pump provides backup in the event of failure of both of the centrifugal primary coolant pumps.

The surge tanks for the two parallel primary pumps are connected with headers to provide a single cover gas volume and a single sodium volume. Primary loop pressure is maintained by controlling cover gas pressure in the pump surge tanks. Sodium inventory for the primary system is maintained during operation by a high level overflow in the pump surge tanks, and by make-up sodium injected into the surge tanks. Sodium fill is also available during shutdown on the discharge side of the pumps near the inlet to the in-reactor test section. This sodium makeup line also serves as emergency cooling in the event of a loss of all primary pumping capability or a primary pipe rupture.

The secondary has a pump and surge tank arrangement similar to the primary system. System pressure is controlled by adjusting cover gas pressure. Sodium inventory is maintained by permitting volume contraction or expansion to change the level of the sodium in the surge tank. Sodium makeup capability is provided to the surge tank but this would be used primarily to replace the coolant which is drained from the system for purification.

Secondary flow is sensed and this signal used to control the two parallel pumps. The two pumps assure coolant flow in the event of a single pump failure. The secondary coolant

flow can be adjusted as a function of primary coolant flow. This capability provides for more uniform temperature increase or decrease throughout the system.

An electromagnetic pump provides a minimum coolant capability in the event of failure of both centrifugal pumps. The electromagnetic pump would also provide low flows for decay heat removal.

The sodium-air heat dump has two parallel air blowers feeding a single stack, to assure cooling. Each blower has a butterfly valve on the inlet and outlet of the blower. The valves on the blower outlets are on-off valves to provide isolation and permit maintenance on the isolated blower. The valves on the blower inlet, along with adjustable vane pitch, provide for continuous coolant flow control.

The heat dump sodium outlet temperature is sensed and compared to a set point derived from the test loop inlet temperature and this error signal used to provide a control signal to the blower inlet air control valves.

During reactor scram or programmed shutdown, position of the air control valves will be determined by a special scram or shutdown sequence. For periods of very low heat removal, dampers above the cooling fins would be closed and an electrical heater operated to maintain sodium outlet temperature from the heat dump. This may be required to prevent freezing or cold trapping due to overcooling in the air dump.

SECTION 3.0 ABNORMAL AND EMERGENCY PLANT CONTROL

When the reactor and process systems are performing normally at power, a wide class of system perturbations (e.g., air inlet temperature, pump voltages) will be handled adequately by the normal reactor and plant control systems as discussed in the previous sections. Those incidents involving malfunction or loss of components, however, will require more drastic action to correct the abnormality. Two basic levels of correction are proposed for the FFTF: (1) Controlled Power Reduction, which will reduce reactor power in response to selected abnormal conditions when the normal control systems are unable to react, and (2) Plant Protection, which will scram the reactor in response to any serious abnormal condition or in the event that the normal control or Controlled Power Reduction Instrumentation fails to act.

The following section of this document is organized as follows:

(1) Controlled Power Reduction Instrumentation is discussed with respect to items such as the need for this instrumentation, the results of preliminary analysis, and the preferred concepts proposed for conceptual design, (2) Plant Protection Instrumentation is defined and a concept is presented outlining the scram trips needed for reactor protection, and (3) Engineered Safeguards Control is discussed.

3.1 CONTROLLED POWER REDUCTION INSTRUMENTATION

Instrumentation must be provided to detect abnormal situations which might arise during operation of the facility and which may lead to damage if left uncorrected. The standard method of providing this detection is by the use of a Protection System which initiates a reactor scram. In most power reactors, reactor scram is the only form of correction applied to

abnormal operation other than the normal control restraints. The FFTF will, of course, have a protection system which will initiate a reactor scram. This system is covered in Section 3.2 of this document.

In addition to a protective system which initiates a reactor scram, the FFTF should have an instrumentation system which initiates corrective action somewhat less severe than the scram. The action taken could include items such as:

(1) individual rod scrams, (2) all rod insertions at a predetermined rate and/or (3) regulating rod insertion at its normal rate of travel. The instrumentation provided for this purpose will be called Controlled Power Reduction (CPR) Instrumentation. The philosophy governing the use of CPR instrumentation is outlined as follows:

- A. The CPR Instrumentation is to be separate from the Protective Instrumentation. Every effort will be made to provide separate and distinct channels (from the sensor to the actuation device input terminals) for the Protective Instrumentation. The CPR Instrumentation will use sensors, amplifiers, etc., provided by the Control Instrumentation or those provided exclusively for CPR use.
- B. The CPR Instrumentation will be designed and used only where it is backed up by Protective Instrumentation. If any CPR channel should fail to function, the reactor must still be protected by reactor scram. It is expected that in many cases the backup will be provided by a related variable. For example: the CPR Instrumentation could provide power setback on low flow in a secondary loop with the Protective Instrumentation providing backup by reactor scram upon abnormally high temperature in the primary cold leg in addition to high core outlet temperature.

3.1.1 Need and Requirements for CPR Instrumentation

CPR Instrumentation is desirable for the following reasons:

- A. Since the FTR is a test reactor, the events which will require some form of corrective action will be more frequent than for a nontest reactor. With the all-rod scram as the only form of correction, this could lead to a large number of scram trips. As shown in Columns 1 and 2 of Table 3-I, for example, SEFOR and EBR-II have approximately 24 and 31 trips, respectively, which cause a reactor scram during full power operation. If equivalent protection is applied to the FTR, excluding individual channel temperatures, the number of scram trips would be greater than 100, as shown in Columns 3 and 4 of Table 3.I. A large number of these trips (70 for the FTR-SEFOR extrapolated case and 95 for the FTR-EBR-II extrapolated case) are due to the abnormal behavior of process variables (flow, level, and temperature) in the closed loops. The selected use of CPR instrumentation in areas where analysis shows it to be adequate (secondary and heat dump systems) will provide a significant reduction in the number of scram trips.
- B. The characteristics of the tests irradiated in the FTR over its design life will be varied and the detailed needs of the tests will be unknown during design and initial operation. With a CPR system, future experimenters have a selection of automatic corrections available for use as their needs warrant. When CPR is not available, one is forced into a decision of scrambling the reactor because of one experiment with the resultant consequences of scram forced on all other experiments in the reactor.

TABLE 3-I. Full Power Scram Trip Comparison

<u>Trip Source</u>	<u>SEFOR¹</u>	<u>EBR-II²</u>	<u>FTR-SEFOR³</u>	<u>FTR-EBR-II⁴</u>
Neutron Flux	2	4	4	4
Reactor Primary Sodium	9	19	21 (3 loops)	39 (3 loops)
Closed Loop Primary Sodium	0	0	45 (5 loops)	95 (5 loops)
Secondary Sodium	6	0	15 (3 loops)	0
Closed Loop Secondary Sodium	0	0	25 (5 loops)	0
Other (main electrical power, radiation, miscellaneous pressure, etc.)	13	6	5	5
Individual Channel Temperature	0	2	76 (driver and open test positions)	76
Total Scrams Excluding Individual Channel Temperature	24	29	115	143
Total Scrams	24	31	191	219

1. Refer to References, Appendix A, Item 14.
2. ANL Drawing ID-2C-11142, EBR-II Reactor Shutdown Schematic-Operate Mode, February 1967.
3. Protection equivalent to that used in SEFOR applied to FTR heat transport systems.
4. Protection equivalent to that used at EBR-II applied to FTR heat transport systems.

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- C. The FTR, being a test reactor, will operate in areas where the technology and the consequent approach to unsafe conditions are relatively unknown. This is especially true for the closed test loops. If an all-rod reactor scram is the only form of correction available, this can lead to an abundance of spurious scrams due to operational drifts, minor instrument noise, etc. If CPR correction can be imposed around the operational band, plant availability can be significantly improved.
- D. The thermal transients imposed on the reactor internals and the vessel after a reactor scram are expected to be severe because of the large temperature rise through the core. The use of a less severe form of shutdown, when time is available, will reduce the magnitude of the transients and provide additional conservatism to the core design.
- E. The thermal reactors (MTR, ETR, ATR)¹ located at the National Reactor Testing Station (NRTS) in Idaho, make extensive use of CPR Instrumentation. The series of corrections which they apply for abnormal operation (slow setback, fast setback, all-rod rundown, junior scram, slow scram, and fast scram) has proved to be very flexible in allowing different types of power reductions according to the needs of the particular experiments in the reactor.
- F. The use of CPR Instrumentation should be considered to take advantage of the transport delay times in the Main Heat Transport and Closed Loop Heat Transport Systems. Upon loss of secondary flow with the present design, for example, the time required for the increased primary

1. Refer to References, Appendix A, Item 15.

cold-leg temperature to reach the reactor core is approximately 13 sec. Action can be taken by the CPR Instrumentation to reduce reactor power at a predetermined rate, thereby taking full advantage of this time. The transport delay time for abnormal operation in the tertiary system to propagate to the reactor core is expected to be about 25 to 30 sec. Here again, some form of CPR action seems to be in order.

- G. CPR Instrumentation may be used to good advantage to protect plant equipment other than the reactor core such as heat exchangers and piping in the secondary and tertiary systems. The use of CPR Instrumentation for this purpose prevents unnecessary scrams and at the same time allows CPR Instrumentation to be considered nonprotective.

If CPR Instrumentation is to provide a valuable contribution to control of the reactor, it must be able to meet some general requirements. The complication that this instrumentation adds to the control system must be justified by less severe thermal transients, and increased plant availability. The general requirements are stated below.

- A. CPR Instrumentation must provide a significant decrease in the thermal transients resulting from its use when compared to a reactor scram. Reductions in power by use of CPR Instrumentation will be slower, allowing more time for the flow control instrumentation to adjust the flow to a given power level. In the ideal situation, flow will be reduced to hold the core ΔT and thereby provide the capability of a hot startup and return to power.

- B. CPR Instrumentation must provide a varied response to individual abnormal situations. One of the disadvantages of Protective Instrumentation is that it is designed to reduce reactor power as rapidly as possible from the operating trip point to decay heat level with no intermediate stopping point. CPR Instrumentation will be designed to provide slower power reductions in power to intermediate levels sufficient to alleviate the particular abnormal situation.
- C. CPR Instrumentation must provide increased plant availability over and above that of a system which employs only annunciators, normal control loops, manual operator control, and protective action.
- D. CPR Instrumentation must significantly reduce the number of scrams by preventing selected process variables from reaching scram trip points.

3.1.2 Controlled Power Reduction Analysis and Concept

Preliminary analysis has been completed to determine the reactor response to various reactivity reductions. These results are included as part of Appendix C. The first general class of reactivity reductions considered was for single and partial rod scrams of 50¢ to 15\$ inserted over 1 sec, with no primary flow reduction. The second class of reactivity reductions considered was for continuous reactivity reductions with rates between 1 and 75¢/sec and with the possibility of primary flow reduction.

The studies of single and partial rod scrams indicate that fast reactivity reductions of less than \$1.50 are required to hold the rate-of-change of tube outlet temperature below 45 °F/sec. For this range of reactivity reductions (below

\$1.50), it would be difficult to predict the proper rod insertion required without the use of extensive digital control. It was concluded that this form of reactivity reduction should not be considered further at this time.

The second class of reactivity reductions, continuous rod insertions, provides a much more flexible method of reducing power. Rod insertions up to 10¢/sec could be used with no reduction in primary flow with a maximum rate-of-change of tube outlet temperature of about 16 °F/sec. Rod insertions greater than 10¢/sec could be combined with flow reductions to minimize outlet temperature transients. Further analysis is required to define the optimum method of reducing primary flow and to coordinate the response of the secondary and tertiary portions of the main heat transport system.

Two categories of CPR action have been selected for further study and analysis. These actions are called Programmed Shutdown and Power Setback. Programmed Shutdown is accomplished by an all-rod insertion with primary flow adjustment to limit thermal transients. The exact rate of shutdown has not been determined, but is expected to be greater than 10¢/sec negative reactivity insertion which corresponds to a power reduction from 100% to 5% power in 60 sec as shown in Figure 3-1. Conditions to be analyzed for initiating programmed shutdown are listed as follows:

A. High Radionuclide Concentration-Containment Exhaust.

Upon detection of excess activity in the containment exhaust, the reactor will be shut down until the source of activity can be located and identified. The shutdown does not have to be rapid and thermal transients should be minimized.

FIGURE 3-1. Automatic FFTF Power Reductions

- B. Sustained Loss of One of the Two Main Sources of Electrical Power. Upon detection of loss of one of the two main sources of outside power, the reactor will be shut down until power is restored. Thermal transients should be minimized during shutdown. The shutdown rate must be sufficient to prevent sodium boiling due to loss of power to the affected main heat transport loop(s).
- C. Low Flow - Each Closed Test Loop Secondary Sodium Circuit. Upon detection of this condition, the reactor power should be reduced to avoid the occurrence of a reactor scram upon detection of high sodium temperature in the closed test loop primary cold leg. The rate of power reduction should be fast enough to avoid a reactor scram if one of the two secondary closed loop sodium pumps is lost.
- D. High Temperature in Closed Test Loop Secondary Cold Leg. This condition indicates the loss or malfunction of the heat dump system for a closed loop. The reactor power is reduced before the temperature unbalance can propagate back to the primary system, thereby taking advantage of transport time delays.

The second action selected will be called Power Setback. Power Setback is accomplished by a single rod insertion to reduce reactor power to a predetermined level. Setback is also used to reduce reactor power until the condition which initiated the setback has cleared. Primary flow reduction would not be required for all initiating events. The rate of power reduction is expected to be less than that for the negative 10¢/sec which corresponds to a rate-of-change of core outlet temperature of -16 °F/sec with no primary flow reduction. Conditions to be considered for initiation of Power Setback and listed below.

- A. Low Flow - Each Main Heat Transport Secondary Loop.
Reactor power will be reduced in order to avoid a reactor scram upon detection of high temperature in the primary cold leg. Detection of this condition could also initiate a program to take the affected loop out of service automatically while operation continued at a reduced reactor power.
- B. High Temperature - Main Heat Transport Secondary Cold Leg.
This condition indicates a malfunction of the heat dump system for one of the main heat transport loops. The reactor power should be reduced before the effect of this disturbance can propagate back to the primary system causing a reactor scram. The loop could also be taken off-line while the reactor continues to operate at a reduced reactor power, using the unaffected loops.
- C. High Neutron Flux - Power Range Control Channels. This condition would be detected by the power range control channels and would initiate a power reduction to avoid the occurrence of a reactor scram. The setpoint for this power reduction, for example, might be set at 105% power with the scram setpoint at 110% power on the protective channels. The power reduction will correct moderately fast changes in power level before the scram setpoint is reached.
- D. High Temperature - Core Outlet. Upon detection of high temperature at the core outlet, reactor power would be reduced to avoid a temperature increase to the scram setpoint. Control channels would monitor temperature and cause a power reduction, for example, at 815 °F, and protective channels would monitor temperature and initiate

a reactor scram at 830 °F. For the power reduction, it will not be desirable to reduce primary flow as this would only compound the overtemperature problem.

- E. High Temperature - Individual Driver and Open Test Position Channels. Action in response to this condition is now planned as protective, requiring a reactor scram. Protective action would have the advantage of assuring that the reactor is shut down in the fastest possible manner, but with 352 protective channels (4/reactor channel) in the system, plant availability could be affected. The design should include the option of changing from protection to power setback, if future analysis and developmental results so indicate. Power setback would have the advantage of reducing power to an intermediate level while holding flow at a constant level. The instrumentation for individual channel over-temperature detection should be designed to meet all protective requirements.¹ The final trip output could then be used for either Protection or Power Setback as required.

3.2 PLANT PROTECTION INSTRUMENTATION

The FFTF Plant Protection Instrumentation is defined as all electrical and mechanical devices and circuitry (from, and including sensors to actuation device input terminals) involved in generating those trip signals associated with the protective function. Included are those signals which actuate reactor scram and which, in the event of a serious reactor accident, actuate engineered safeguards.

1. Refer to References, Appendix A, Item 16, pages 31-16.

The discussion here will be divided into three sections. The first section will discuss protective instrumentation to be used to scram the reactor in the event of abnormal operation in the nuclear and process systems. The second section will discuss response of process systems to scram action. Finally, the actuation and operation of engineered safeguards are discussed.

3.2.1 Concept For Scram Trips

In order to define the instrumentation needed for FFTF plant protection a list of abnormal nuclear and process events was compiled with the aid of the FTR fault tree.¹ This list is included in Appendix D of this document. A concept for scram protection of the FTR is outlined below.

- A. Positive Reactivity Insertions. Period trips are used to guard against the effects of positive reactivity insertions from the all-rods-inserted condition to approximately 1% power, by low range and intermediate range channels. High level trips are also provided every five decades in power level as a backup to period protection. Protection against overpower in the range between approximately 1% and 150% power is provided by power range nuclear channels which will scram the reactor on both high level and high rate-of-change of level. Two additional trips may be provided to guard against selected positive reactivity insertions; namely, hydraulic holddown ΔP to prevent reactivity increases due to fuel movement, and sodium level in all primary pumps to prevent voiding of the core by gas entrainment

1. Refer to References, Appendix A, Item 17.

in the sodium. A backup to the above scram trips is provided by high bulk core outlet temperature.

- B. Negative Reactivity Insertions. Protection against large unexpected decreases in power is provided by a negative rate-of-change scram. This scram protects against negative reactivity insertions due to sodium boiling in the outer core regions and fuel meltdown where fuel moves to a region of lower reactivity worth.
- C. Loss of Primary Heat Transport Systems - Main and Closed Loop. Protection is initiated for loss of level in the FTR vessel as a result of sodium leakage from the primary system. Low level in each primary pump will also scram the reactor to protect against void insertions. Protection of the reactor in the event of loss of primary flow will be provided by a high/low flow trip in each main heat transport system primary loop, a single trip in response to low sodium inlet plenum pressure, and a single power/flow comparator for the main primary system. The power/flow comparator utilizes the signal from the power range flux channels and a total primary flow signal to guard against loss of flow during transient periods when flow is changing. Each closed test loop will also have a power/flow trip.

Although not included in this concept, one has the option of using the individual channel flowmeters as protection against loss of flow. These flowmeters have some advantages over loop flowmeters:

- Their response would not depend on the location of a pipe break with respect to the flowmeter location.
- They would be sensitive to operation of check valves to prevent reverse flow in the event of a pipe break.

- They would provide increased redundancy since 73 driver channels are provided.

During operation with all flowmeters in service, reactor scram would occur if 4 of the 73 driver channels indicate low flow. If one or more flowmeters became inoperative, the logic would be changed by use of individual bypass switches, to $4/n$ where $n < 73$. Administrative control would be used to assure that n was greater than a number such as 12. These low-flow trips would be backed up by trips for high core outlet temperature and high closed loop outlet temperature.

- D. Loss of Heat Removal - Individual Driver and Open Test Position Channels. Protection against loss of flow will be provided by provision for reactor scram upon detection of high temperature in each individual driver and open test position channel (see Section 3.2.2, Item E). Flowmeters are provided in each channel. It will only be possible to include one flowmeter in each channel, however, because of space limitations. Since the reliability for a single sensor in this environment is questionable, and dependent on a successful development program, the use of individual channel flowmeters to initiate a reactor scram on loss of flow from a single channel is not contemplated. The flowmeters will still be valuable in providing assurance of adequate flow in all channels prior to reactor startup.
- E. Loss of Secondary and Tertiary Heat Transport Systems - Main and Closed Loop. Protection against loss of secondary and tertiary systems will be provided by detection of high temperature in the primary cold leg of each

closed test and main loop. Backup protection is provided by high temperature at the core outlet and the closed loop hot leg. Individual malfunctions in the secondary and tertiary systems such as low flow or high temperature in individual secondary loops, will be provided by Controlled Power Reduction Instrumentation. (See Section 3.2.)

- F. Other Scram Trips. Additional scram trips are provided for loss of electrical power from both outside sources, high seismic activity level, high radiation level in the containment exhaust coincident with high containment pressure, and for special closed loop experimental needs such as clad temperature, fuel temperature, etc., as might be required by individual experiments.

Table 3-II is a compilation of all reactor trips in response to abnormal operation, including both Protective and Controlled Power Reduction. A total of 198 protective channels and 129 CPR channels are provided. In addition, 304 channels are provided to detect overtemperature in individual driver and open test position channels with the option of using these for either Protection or CPR. Instrument ranges and trip points shown in Table 3-II are only estimations at this stage of design, and will be redefined as design progresses.

3.2.2 Response to Scram Trips

A reactor scram occurs when the plant protective system acts to release the primary safety rods into the core. Because of the speed of such action, the responses of the reactor and heat transport systems to the scram must be automatic. Operating personnel cannot be expected to perform all necessary post-scram functions.

TABLE 3-II. Protective and Controlled Power Reduction Trips¹

		Plant Control				Plant Protection
	Total Channels Required	Startup Interlock	Annunciate	Power Setback	Programmed Shutdown	Reactor Scram
<u>NUCLEAR</u>						
Low Level Range (1 to 10 ⁶ CPS)	3					
Short period			1/3			2/3
Overlap with intermediate range ²			1/3			
High level (10 ⁵ CPS)			1/3			2/3
Not on scale (10 CPS)		Yes	1/3			
Intermediate Level Range (10 ⁵ CPS - 1% power)	3					
Short period			1/3			2/3
High level (0.1% power)			1/3			2/3
Overlap with high range channels ²			1/3			
High Range - Linear (0.1% to 150% power)	4*-3					
Fast rate-of-change (positive)			1/3	2/3		
*Very fast rate-of-change (positive and negative)			1/4			2/4
High level (105% power)			1/3	2/3		
*Very high level (110% power)			1/4			2/4
*Very high flux/flow ratio (1.10)			1/4			2/4

1. Based on 3 main heat transport loops, 6 closed loops, 73 driver channels, and 3 open test positions.

2. Provide alarm or corrective action if less than two nuclear channels are in service.

* Protective Channels

TABLE 3-II. (Contd)

	Total Channels Required	Plant Control			Plant Protec- tion
		Startup Interlock	Annunciate	Power Setback	Programmed Shutdown
TEMPERATURE ¹					
Bulk Sodium					
High core outlet temperature (815 °F)	3		1/3	2/3	
Very high core outlet temperature (830 °F)	4		1/4		2/4
High closed loop primary hot-leg temperature	18		1/3	2/3	
Very high closed loop primary hot-leg temperature	24		1/4		2/4
High main HTS primary cold-leg temperature (515 °F)	9		1/3	2/3	
Very high main HTS primary cold- leg temperature (530 °F)	12		1/4		2/4
High closed loop primary cold-leg temperature	18		1/3	2/3	
Very high closed loop primary cold-leg temperature	24		1/4		2/4
High HTS secondary cold-leg temperature	9		1/3	2/3	
High closed loop secondary cold- leg temperature	18		1/3		2/3
Point Sodium ²					
High individual driver outlet temperature (825 °F)	292		1/4	2/4 ²	2/4 ²
High individual open test position outlet temperature (825 °F)	12		1/4	2/4 ²	2/4 ²
FLOW					
Bulk Sodium					
High/Low primary flow - individual HTS loops	12	Yes	1/4		2/4
High Flux/Flow - Main HTS	4	Yes	1/4		2/4
High Flux/Flow - Individual Closed Loops	24	Yes	1/4		2/4
Low secondary flow - individual HTS loops	9	Yes	1/3		2/3
Low secondary flow - individual closed loops	18	Yes	1/3		2/3

1. Trip points noted are representative of initial operation.

2. Could be used for power setback or reactor scram
dependent on further analysis and development of thermocouples.

* Protective Channels

TABLE 3-II. (Contd)

		Plant Control				Plant Protection	
		Total Channels Required	Startup Interlock	Annunciate	Power Setback	Programmed Shutdown	Reactor Scram
FLOW (contd)							
Point Sodium Flow ¹							
Low flow - individual driver elements		73	Yes	1/1			
Low flow - individual open test positions		3	Yes	1/1			
Low flow - closed loop test sections		6	Yes	1/1			
*PRESSURE							
Low core inlet plenum pressure		4*	Yes	1/4			2/4
Low hydraulic holddown ΔP		4*	Yes	1/4			2/4
High containment building pressure coincident with high radiation level in containment exhaust		4*		1/4			2/4
*LEVEL							
Low reactor vessel level		4*	Yes	1/4			2/4
Low HTS primary pump level		12*	Yes	1/4			2/4
Low closed loop primary pump level		24*	Yes	1/4			2/4
MISCELLANEOUS							
Sustained loss of Electrical Power (>1/2 sec)							
Loss of one main bus		3	Yes	1/3		2/3	
Loss of both main buses		4		1/4			2/4
Radiation Level							
High containment exhaust level		3	Yes	1/3		2/3	
Seismic Intensity							
Low trip (instrument malfunction)				1/4			
High level		4		1/4			2/4
Closed Loop Experimental							
Special experimental needs		24*-18			2/3	2/3	2/4

1. Not to be used in automatic power reduction circuitry in this concept (see Discussion under 3.3.1).

* Protective Channels

TABLE 3-II. (Contd)

	Total Channels Required	Plant Control				Plant Protec- tion
		Startup Interlock	Annunciate	Power Setback	Programmed Shutdown	Reactor Scram
Total Protective Channels ¹	198					
Total Protective Channels Including						
Individual Open Test Position and Driver						
Overtemperature Protection	502					
Total CPR	129					
Total CPR Including Individual Open Test Position and Driver Overtemperature	433					

-
1. Channel includes the sensor, amplifier, trip unit, etc.

- A. Reactor Response. A safety circuit trip signal or loss of safety circuit continuity causes all scrammable rods to be released into the core simultaneously and the remaining rods are driven into the core. In addition, the drives of the scrammable rods then automatically drive down with maximum speed to assure that the rod reaches its fully inserted position. When a safety circuit function is actuated, it is impossible to delay or cancel the resulting scram effect.
- B. Heat Transport System Response. A scram response is not required of the heat transport systems at any time when the ΔT across the core is less than 10% of the operational full value. Based on the initial oxide core, a reactor scram with full primary flow at a core ΔT of 30 °F (10% of initial ΔT) will produce a core outlet transient of about -5 °F/sec for 3 to 4 sec. The resulting thermal stresses are well within expected design values for the core. Since the main purpose for having any sort of a heat transport system scram response is to limit the severity of thermal shock, primary flow reduction is not required.

The heat transport system will respond in an identical manner to each scram, regardless of its cause. Response of the process systems (once the core ΔT is above 10% full value) will be as follows:

1. Primary sodium flow will be automatically decreased to a preset value. A short time delay (on the order of 1/2 to 1 sec) will elapse between safety rod drop and the beginning of flow decay, in order to assure power shutdown and remove stored heat in the fuel. The post-scram flow may depend on the type of pump

selected. A wound-rotor, motor-driven pump may have a post-scram flow as high as 35% full value, tending to collapse the core ΔT relatively rapidly. Thus, to preserve core ΔT for scram recovery, it would be necessary to allow constant speed pony motors to provide decay heat removal (at about 10 to 15% of full flow) by removing power from the wound-rotor motors. A constant-speed induction motor with an electromagnetic clutch (the alternative method) would maintain a post-scram flow between 5 and 10% of full value. Core ΔT would be maintained. While the latter pump drive approach is preferable from the thermal transient point of view, the former approach may be dictated by other process system concerns.

2. Secondary system flow will behave as does the primary.
3. DHX air flow will cut back to maintain cold-leg temperatures at a constant level. Expected response is a stopping of all fans followed by control on the DHX stack louvers to limit the amount of natural air draft cooling. Should it be necessary from the standpoint of controllability, several modules of the DHX may be taken off-line by valving off sodium flow and by total closure of the stack louvers.
4. Heat transport systems for the closed test loops will respond in the same manner as for the main loops, when fueled tests are in-loop. For materials tests, flow response to scram is not critical because of the relatively small test ΔT .

C. Scram Recovery. A complete integration of the FTR and closed loop primary, secondary, and tertiary coolant system is required to effect a temperature-balanced scram

recovery startup. A common control point is required to correlate the combined responses of these systems. That common variable should be the core primary bulk outlet temperature.

The response of the systems should be automatic and should result in all systems remaining balanced during the recovery to full power conditions. At full power conditions, the system should automatically control on primary outlet temperature.

The most complex system, in terms of the number of programmed actions required, is the tertiary system. Initially, two or more modules in each system will be completely shut down and one or possibly two others will be in a minimum cooling status. As the scram recovery progresses and nuclear heat is again generated in the FTR, the dampers on the operating tertiary modules will gradually open. This will continue until at some predetermined condition the need for the automatic start of one of the shutdown units will be required. The shutdown unit will automatically start and assume its portion of the cooling load as required. This sequence will continue as the reactor power level is increased, until all modules are back on the line.

3.2.3 Engineered Safeguards

- A. Introduction Engineered safeguards are provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems.¹ Their use for the prevention or

1. Refer to References, Appendix A, Item 18.

reduction of fission product release to the environment may be accomplished by three general methods:¹

- Preventing or minimizing, by emergency cooling or otherwise, the overheating of the fuel materials.
- Removing the fission products from the containment atmosphere by filtering, scrubbing, storage, etc.
- Constructing two or more barriers around the primary system so that the probability that a large quantity of fission product activity may leak out is negligible, or at least greatly reduced.

Engineered Safeguards for the FFTF will include the containment and emergency core cooling systems. The conceptual control scheme for these systems is described below.

- B. Containment. A method of providing containment isolation is shown in Figure 3-2.

Two degrees of isolation are provided.

1. Upon the occurrence of high radionuclide concentration in the plant exhaust air, the ventilation supply and exhaust valves close. At the same time, a programmed reactor shutdown is automatically initiated. The instrumentation provided for the purpose will not be considered protective.
2. Upon the occurrence of high radionuclide concentration in the exhaust air coincident with high pressure in the outer containment volume a reactor scram is initiated and all lines penetrating the containment not

1. Refer to References, Appendix A, Item 19.

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FIGURE 3-2. Containment Actuation Logic Diagram Plant Protection System

essential to safe reactor shutdown are isolated.

The valves that are closed will include those in the ventilation supply and exhaust lines and may include lines such as the inert gas supply and pneumatic air supply depending on the pressure rating of the portions of these systems outside the containment building. The secondary sodium lines which penetrate the containment building will be designed with a double barrier with a pressure rating above that of the containment and will, therefore, not require isolation.

- C. Emergency Cooling. The concept for emergency cooling for the driver core has not been selected at this time.

Since control of this emergency cooling system is concept dependent it is not possible to provide a control scheme. For example, if a natural circulation system is used the control required will be quite limited. If, however, a system is used which requires the starting of pumps and the manipulation of valves the control could become quite complex. Regardless of the concept used it will be considered an engineered safeguard and its control will be initiated and controlled by Protective Instrumentation.

The concept for emergency cooling of each closed test loop employs an e-m pump in parallel with the two centrifugal pumps in both the primary and secondary portions of the circuit. Each e-m pump is rated at 15% of the primary pump capacity to assure forced circulation flow through the test assembly in the event that both centrifugal pumps fail. The e-m pumps are not normally in service, but are automatically started upon the occurrence of low flow.

Each closed test loop is also provided with a backup emergency coolant supply in the primary portion of the system. In the event of a primary coolant piping failure or leak the level will drop in the pump tank. This condition is sensed by the protective instrumentation which closes the valve providing recirculating flow and opens a supply valve leading directly to the in-reactor tube. The Fill and Recirculating pump which is used for this operation is normally in operation. In this manner coolant is supplied to the test even in the event of catastrophic primary coolant piping failure.

APPENDIX A

REFERENCES

APPENDIX A

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APPENDIX B

EVALUATION OF PLANT CONTROL WITH HYBRID SIMULATION

APPENDIX B

EVALUATION OF PLANT CONTROL WITH HYBRID SIMULATION

The analog-digital ("hybrid") simulation of the FFTF reactor and heat transport systems¹ was used to evaluate several possible plant control schemes. A plant control scheme was selected based on simplicity and on control response to expected operating transient conditions. The results are presented in this appendix, with the clear understanding that they are preliminary and that control system evaluation will continue as the FFTF design progresses.

Figure B-1 shows the schematic of the reactor and heat transport system with the reference control scheme included. A single heat transport circuit was simulated, thus assuming that all heat transport loops behave in the same manner. Future analysis will include the behavior of process control with multiple coolant circuit interactions. The hybrid simulation includes the following features pertinent to process control:

1. Primary and secondary pumps are driven by a constant-speed synchronous motor, with pump speed controlled by excitation current on an electromagnetic clutch.
2. DHX air flow is provided by constant speed fans, with flow control based on inlet valve angle.
3. Neutron flux control is provided by adjustment of control rod position (at a constant reactivity rate) whenever the flux is outside a predetermined deadband ($\pm 1\%$).

1. Refer to References, Appendix A, Item 20.

FIGURE B-1. Schematic of Process Control Simulation

4. Sodium mixing plena are included at the inlet and outlet of the IHX and DHX, as well as at the inlet and outlet of the reactor vessel.

Future analysis will also include other design possibilities, such as sodium pumps driven by wound-rotor motors.

The control scheme shown in Figure B-1 represents a configuration which meets the test criteria known at this time. Flow control for the main heat transport circuits is for constant flow (auto-control on flow signals, with manual setpoints), and FTR inlet temperature is controlled by the DHX air flow, using an anticipatory control technique. Air flow is adjusted to produce the desired DHX sodium outlet temperature, in response to power changes, such that the FTR inlet is maintained at the defined temperature, ± 10 °F. A control algorithm, based on reactor power and IHX performance, provides the setpoint for a P&I (gain-plus-reset) controller for this control loop. In the simulation, gain and reset rate for the controller were arbitrarily selected; no attempt was made to maximize controller performance.

The initial conditions for studying the control scheme were as follows: (a) Power = 400 MW_t, (b) Reactor inlet temperature = 600 °F, (c) Reactor Outlet Temperature = 900 °F, (d) Number of heat transport circuits operating = 3, (e) IHX LMTD $\approx$ 110 °F, and (f) Air inlet to the DHX = 100 °F. The simulation was made to respond to the following transients:

1. Power setback from 400 MW to 300 MW by changing flux setpoint, allowing control to seek new power level at 1¢/sec; full flow is maintained.
2. Power setback from 300 MW to 200 MW at 1¢/sec; full flow.

3. Power setback from 200 MW to 100 MW at 1¢/sec; 50% flow.

4. Power setback from 100 MW to 50 MW at 1¢/sec; 50% flow.

Results are shown in Figures B-2 through B-5. For the extreme power changes which were forced on the process system, at high and low flows, the resulting core inlet temperature changes were within the desired goal of ± 10 °F. In each case, the desired steady-state inlet temperature was re-established without extremely long settling times and without excessive DHX control action.

FIGURE B-2. System Response to Power Ramp, 400-300 MW_t

FIGURE B-3. System Response to Power Ramp, 300-200 MW_t

FIGURE B-4. System Response to Power Ramp, 200-100 MW_t, 50% Flow

FIGURE B-5. System Response to Power Ramp, 100-50 MW_t, 50% Flow

APPENDIX C

PRELIMINARY ANALYSIS OF CONTROLLED POWER REDUCTION

APPENDIX C

PRELIMINARY ANALYSIS OF CONTROLLED POWER REDUCTION

Some preliminary analysis has been completed with the objective of determining what forms of CPR action would be feasible for the FTR. Numerical results presented here are useful for the comparison purposes of this study. However, changing core design has since altered the magnitude of the thermal transients presented here. The three areas covered by the study were: single or partial rod scram insertions, all-rod run-in insertions, and single rod run-in insertions.

SINGLE OR PARTIAL ROD SCRAMS

The use of a single or partial rod scram was explored for use as an intermediate step between a "setback" (rod run-in mode) and a full reactor scram. The basic purpose of this action would be to provide a definite shutdown in the event of an incident while at the same time maintaining a sufficient power level to minimize the time needed for re-start.

The simulation studies assumed that the primary flow rate remained at full flow during each scram and examined the effect of scram reactivity on power and core coolant outlet temperature. An oxide core was assumed and the different scram reactivities were inserted as ramps over a 1 sec interval. Initial conditions were 400 MW_t and 300 °F ΔT. The assumption of full flow appears to be reasonable for this type of partial shutdown for the following reasons: (1) a "junior scram" should offset incipient reactivity incidents without the risk of undercooling, and (2) flow coastdown following a scram will be initiated only after it is certain that rod(s) have been inserted. Results of the brief study are shown in Figures C-1 through C-4. There is little

FIGURE C-1. Effect of Scram Reactivity on Power Level After 10 Seconds

FIGURE C-2. Effect of Scram Reactivity on Power Level After 5 Minutes

FIGURE C-3. Full Flow Scram: Effect of Scram Reactivity on Tube Outlet Temperature (Maximum rate of change)

FIGURE C-4. Full Flow Scram: Effect of Scram Reactivity on Tube Outlet Temperature (Temperature change in 10 sec)

difference in the effects of scram reactivity for magnitudes larger than \$3 to \$5. For lower scram reactivities, within the single and partial rod category, the effects of reactivity on power and core outlet thermal transients are more pronounced. However, the utility of a junior scram appears to be limited precisely because maximizing the post-scram power and minimizing the thermal transients requires that reactivity be accurate in the region of $\sim 50\text{¢}$ to $\sim \$1.50$. It will be difficult to provide reactivity reductions of this small magnitude without making extensive use of digital control over the complete range of operating conditions. It was concluded, therefore, that the use of single or partial rod scram insertions would not be considered further, at this time, as a form of Controlled Power Reduction.

ALL-ROD RUN-IN INSERTIONS - PROGRAMMED SHUTDOWN

Reactivity insertions of the magnitude representative of several or all rods were explored. Primary flowrate was controlled to hold core outlet temperature constant with minimum flow set at 20% of full flow. The results are shown in Figure C-5.

SINGLE ROD RUN-IN INSERTIONS - SETBACK

The use of reactivity reductions of a magnitude representative of a regulating rod at its normal speed of insertion was explored. The basic purpose of this action would be to reduce reactor power in response to selected abnormal conditions to turn the situation around before the scram trip point was reached.

The simulation studies again assumed that the primary flowrate remained at full flow during the setback and examined the

FIGURE C-5. Programmed Shutdown Insertion Rates Effect on Reactor Power Tube Outlet Temperature Held Constant by Reducing Primary Flow (Min-20%)

effect of reactivity reduction on power and core coolant outlet temperature. Initial conditions were 400 MW_t and 300 °F ΔT. Results of the study are shown in Figures C-6 through C-8. The results show, as expected, that various rates of core outlet temperature and reactor power reduction can be obtained depending on the rate of reactivity reduction.

FIGURE C-6. Rod Insertion Rate Versus Maximum Rate-of-Change Tube Outlet Temperature Full Primary Flow

FIGURE C-7. Effect of Rod Insertion Rate on Tube Outlet Temperature

FIGURE C-8. Effect of Rod Insertion Rate on Reactor Power

APPENDIX D

EVENTS REQUIRING PROTECTIVE ACTION
AND/OR CONTROLLED POWER REDUCTION

APPENDIX D

EVENTS REQUIRING PROTECTIVE ACTION
AND/OR CONTROLLED POWER REDUCTION

In order to define the instrumentation needed for FFTF protection, a list of abnormal process and nuclear events which could occur is presented. This list will also define the complexity and extent of the instrumentation needed. The general categories covered are as follows:

1. Positive Reactivity Insertion-Precritical Operation
2. Positive Reactivity Insertion-Critical to Low (1%) Power
3. Positive Reactivity Insertion-Power Operation
4. Loss of Primary Heat Transport (Main and Closed Loop)
5. Loss of Secondary Heat Transport (Main and Closed Loop)
6. Loss of Tertiary Heat Dump (Main and Closed Loop)
7. Loss of Heat Removal from Individual Driver and Open Test Position Channels
8. Sodium Leakage
9. Loss of Electrical Power
10. Seismic Activity.

With the aid of the FTR fault tree,¹ the above general categories are broken down into subcategories, and the following tabulations are made:

1. A list of the events guarded against (possible causes)
2. A list of the results if protective action is not taken (possible results)

1. Refer to References, Appendix A, Item 17.

3. A list of the variables or signals that could be used for the detection of abnormal operation (possible monitored variables)
 4. A list of possible action that could be taken to correct the abnormality (possible action).
1. POSITIVE REACTIVITY INSERTION-PRECRITICAL OPERATION
 - A. Possible Causes
 - Fuel element dropped into position
 - Loading error
 - Uncontrolled rod withdrawal
 - Rod expulsion
 - Removal of material within control or safety rod clad
 - Maintenance error
 - Loss of fuel holddown
 - B. Possible Results
 - Short reactor period
 - Fuel clad failure and fuel meltdown
 - C. Possible Monitored Variables
 - Reactor period-low range channels
 - Shutdown reactivity monitor
 - ΔP indication of hydraulic holddown
 - High count rate-low range channels
 - D. Possible Action
 - Reactor scram
 - Cocked rod scram
 2. POSITIVE REACTIVITY INSERTION-CRITICAL TO LOW (1%) POWER
 - A. Possible Causes
 - Neutron spectrum shift
 - Dynamic instability
 - Positive void reactivity effect

- Positive coolant temperature effect
- Uncontrolled rod withdrawal
- Rod expulsion
- Loss of fuel holddown
- Removal of material within control or safety rod clad

B. Possible Results

- Short reactor period leading to fuel melting and/or sodium boiling

C. Possible Monitored Variables

- Reactor period-low range channels
- Reactor period-intermediate range channels
- ΔP indication of positive hydraulic holddown
- High level-intermediate range channels
- Primary sodium temperature

D. Possible Action

- Reactor scram

3. POSITIVE REACTIVITY INSERTION-POWER OPERATION

A. Possible Causes

- Same as in 2A above
- Probability of fuel or control rod melting greater due to higher temperature

B. Possible Results

- Power excursion leading to fuel meltdown or sodium boiling

C. Possible Monitored Variables

- Overpower-power range channels
- High rate-of-change-power range channels
- Overtemperature in primary sodium
- ΔP indication of positive hydraulic holddown
- Sodium boiling detection

- Flux-flow ratio
- Low primary pump level
- D. Possible Action
 - Reactor scram or setback
 - Flow increase
- 4. NEGATIVE REACTIVITY INSERTIONS - POWER OPERATION
 - A. Possible Causes
 - Sodium voiding near the upper axial face of the core due to loss of flow and consequent boiling
 - Complete subassembly voiding in outer radial regions of the core where leakage is high and void worth is negative
 - Passage of bubbles through the core resulting in a reactivity decrease as bubble enters the bottom axial face of the core.
 - B. Possible Results
 - Propagation of voiding to central region of the core resulting in a positive reactivity insertion.
 - C. Possible Monitored Variables
 - Low level -- power range channels
 - Negative rate-of-change power range channels
 - Low flow individual channels
 - Low flow HTS loops
 - Sodium boiling detection
 - D. Possible Action
 - Reactor scram
- 5. LOSS OF PRIMARY HEAT TRANSPORT (MAIN AND CLOSED LOOP)
 - A. Possible Causes
 - Loss or increase of sodium due to:
 - Failure of sodium purification and makeup system
 - Operator error-sodium drained from loop

- Coolant leakage from reactor vessel
- Coolant leakage from primary loop
- Overpressure of cover gas
- Loss of sodium flow due to:
 - Very low sodium level (pumps lose suction)
 - Blockage of flow through core
 - Blockage of flow through IHX
 - Check valve malfunction
 - Rupture of primary coolant boundary
 - Flow bypassing core (reverse flow)
 - Mechanical or electrical failure or primary pump
- Other possible causes
 - Failure of primary heat sink (secondary loop)
 - Failure of secondary heat sink (heat dump)
 - Carbonaceous material on fuel heat transfer surface
 - Gas collection on fuel surface

B. Possible Results

- Inadequate heat removal from core
- Sodium boiling and/or fuel meltdown
 - Overpressure of primary system
 - Loss of level in primary pump leading to gas entrainment in sodium

C. Possible Monitored Variables

- For loss or increase of sodium level
 - Reactor vessel sodium level
 - Primary pump sodium level
 - Sodium leak detectors
 - Cover gas pressure (vessel and/or pump)
- For loss of sodium flow
 - Flow monitors in primary loop
 - ΔP across primary pump

- ΔP across core
- ΔP across IHX
- Electrical supply to primary pump
- Pump shaft speed
- Temperature increase in primary hotleg
- Sodium boiling detector
- Low temperature secondary hotleg
- Individual flow meters
- Inlet plenum pressure
- For other possible causes
 - Temperature increase in primary coldleg
 - ΔP across core
 - Sodium boiling detector

D. Possible Action

- For loss of sodium level
 - Replenish sodium
 - Take loop out of service
 - Reactor setback, shutdown, or scram
 - Increase flow in other loops
 - Return cover gas pressure to normal
- For loss of sodium flow
 - Increase pump speed
 - Decrease pressure drop through loop
 - Take loop out of service
 - Reactor setback, shutdown, or scram
- For other causes
 - Restore secondary or tertiary loops
 - Reactor setback, shutdown, or scram

6. LOSS OF SECONDARY HEAT TRANSPORT SYSTEM (MAIN AND CLOSED LOOP)

A. Possible Causes

- Loss of sodium level due to:
 - Failure of secondary Na purification and makeup system
 - Operator error-sodium drained from loop
 - Secondary system rupture or leakage
- Loss of flow due to:
 - Loss of sodium level
 - Blockage of flow in IHX or DHX
 - Inadvertent valve closure
 - Electrical or mechanical failure of secondary pump
- Other causes
 - Failure of tertiary heat dump
 - Loss of heat transfer to tertiary heat dump-oxide or carbonaceous deposits on tubing

B. Possible Results

- Loss of primary heat sink
- Inadequate coolant heat removal from core
- Fuel melting and/or sodium boiling

C. Possible Monitored Variables

- For loss of sodium level
 - Expansion tank level
 - Secondary pump level
 - Sodium leak detectors
- For loss of flow
 - Flow monitors in secondary loop
 - ΔP across pump
 - ΔP across IHX
 - ΔP across DHX
 - Electrical supply to secondary pump
 - Pump shaft speed
 - High temperature in hotleg of secondary loop

- For other causes

- High temperature in coldleg of loop

D. Possible Action

- For loss of sodium level

- Replenish sodium
 - Reactor setback, shutdown or scram
 - Take loop out of service

- For loss of flow

- Increase pump speed
 - Take loop out of service
 - Reactor setback, shutdown or scram

- For other causes

- Reactor setback, shutdown or scram

7. LOSS OF TERTIARY HEAT DUMP (MAIN AND CLOSED LOOP)

A. Possible Causes

- Air flow blockage
- Blower or motor failure
- Loss of heat transfer in DHX

B. Possible Results

- Increase in temperature in secondary loop
- Failure of heat removal capability in one complete heat transport loop
- Power/heat removal imbalance in reactor

C. Possible Monitored Variables

- Air flow monitor in heat dump
- Shaft speed on blower
- Loss of power to blower
- Blower air temperature

D. Possible Action

- Remove loop from service
- Reactor setback, shutdown, or scram

8. LOSS OF HEAT REMOVAL FROM INDIVIDUAL DRIVER AND OPEN LOOP CHANNELS

A. Possible Causes

- Flow blockage by carbonaceous material
- Flow blockage by foreign material
- Flow blockage by fuel swelling
- Gas collection around fuel pins

B. Possible Results

- Individual channel boiling
- Individual channel fuel meltdown

C. Possible Monitored Variables

- Sodium boiling detector
- Individual channel thermocouples
- Individual channel flowmeters
- Individual channel clad temperature

D. Possible Action

- Reactor scram, shutdown, or setback

9. SODIUM LEAKAGE

A. Possible Causes

- Loss of reactor vessel barrier
- Break in primary piping (main heat transport and closed loop systems)
- Break in secondary sodium barrier (main heat transport and closed loop systems)
- Sodium instrument line break

B. Possible Results

- Loss of flow if break is large
- Loss of heat transport due to loss of flow
- Sodium fire or smoke

C. Possible Monitored Variables

- Spark plug leak detectors
- Airborne activity for radioactive sodium
- Smoke detection
- Loss of flow
- Loss of instrument indication
- Level

D. Possible Action

- Reactor shutdown or scram
- Drain sodium from area of leakage

10. LOSS OF ELECTRICAL POWER

A. Possible Causes

- Transformer failure
- Switchgear failure
- Circuit breaker failure
- Feeder conductor failure

B. Possible Results

- Loss of electrical power to critical equipment or instrumentation

C. Possible Monitored Variables

- Undervoltage on 13.8 kV system
- Undervoltage on 2.4 kV system
- Undervoltage on 480 Vac system
- Undervoltage on zero time outage bus

D. Possible Action

- Reactor shutdown or scram

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APPENDIX E

GLOSSARY

APPENDIX E

GLOSSARY

FFTF	Fast Flux Test Facility
FTR.	Fast Test Reactor (reactor portion of the FFTF)
HTS.	Heat Transport System
IHX.	Intermediate Heat Exchanger
DHX.	Dump Heat Exchanger (air-cooled)
Protective Function	Action taken by the Plant Protection System to initiate a reactor scram or to actuate Engineered Safeguards
Protection Instrumentation	All electrical and mechanical devices and circuitry (from, and including, sensors to actuation device input termi- nals) involved in generating those trip signals associated with the protective function.
Protection Channel.	An arrangement of components and modules as required to generate a single protective trip.
Trip	The operation of a bistable device to initiate action by the protection instrumentation.
Scram	Rapid reactor shutdown by in- serting all safety rods into the reactor.

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