**TRANSIENT EFFECTS OF LOSS OF ONE COOLANT
CIRCUIT ON THE ENRICO FERMI REACTOR AND
ASSOCIATED HEAT TRANSPORT SYSTEM**

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ABSTRACT

This report presents results obtained from a single-circuit shutdown test performed on the Enrico Fermi reactor at a thermal power level of 20 Mw. The test, commencing with deactivation of one coolant circuit during three-circuit operation, was conducted to ascertain the transients in the principal plant parameters, to observe plant stabilization characteristics, and to verify system controls and interlocks.

Recordings obtained in the test have provided values for the negative rate-of-change of reactor power, steady-state power level, rates-of-change of the principal temperature parameters, and the transport lags of the responses.

FOREWORD

This report is one of a series of reports on the low-power (up to 1 Mwt) and high-power (up to 200 Mwt) nuclear testing of the Enrico Fermi reactor. The Nuclear Test Program is planned, directed and evaluated by Atomic Power Development Associates, Inc. (APDA). The tests are conducted by Power Reactor Development Company (PRDC). The reactor proper is owned and operated by PRDC. The steam generators and electrical generating facilities are owned by The Detroit Edison Company (DECO).

Many individuals have contributed to the nuclear testing of the Fermi reactor. Listed below are those, exclusive of the authors, who have made significant contributions to the work described in this report.

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TABLE OF CONTENTS

	<u>Title</u>	<u>Page</u>
	LIST OF ILLUSTRATIONS	ix
	LIST OF TABLES	ix
	SUMMARY	xi
I.	PURPOSE OF TEST	1
II.	GENERAL DESCRIPTION OF THE ENRICO FERMI REACTOR	3
III.	EXPERIMENTAL APPARATUS AND EQUIPMENT	9
	A. DETECTORS	9
	1. Reactor Outlet Temperature (TE-201 and TE-217) ...	9
	2. Core Outlet Temperature (TE-115)	12
	3. Steam Generator Sodium Inlet Temperature (TE-308) .	12
	4. Steam Generator Sodium Outlet Temperature (TE-301)	13
	5. Reactor Inlet Temperature (TE-209)	13
	6. Water-Steam Temperature (TE-601)	13
	7. Neutron Flux	13
	8. Sodium Flowmeters	14
	B. SIGNAL CONDITIONING CIRCUITS AND ADJUNCTIVE APPARATUS	14
	C. COOLANT SYSTEM	17
	1. Primary Loop	17
	2. Secondary Loop	17
	3. Single-Circuit Shutdown Interlocks	18
IV.	EXPERIMENTAL PROCEDURE AND PLANT CONDITIONS...	19
V.	EXPERIMENTAL RESULTS AND ANALYSIS	21
	A. RESULTS	21
	1. Reactor in Manual Control	21
	a. Primary Loop Sodium Flow	21
	b. Neutron Flux	21
	c. Neutron Flux Rate-of-Change, dn/dt	24
	d. Core Outlet Temperature (TE-115).....	24
	e. Reactor Outlet Temperature (TE-201)	24
	f. Steam Generator Sodium Inlet Temperature (TE-308)	24

TABLE OF CONTENTS (Cont'd.)

	<u>Title</u>	<u>Page</u>
g.	Reactor Inlet Sodium Temperature (TE-209)	25
h.	Steam Generator Sodium Outlet Temperature (TE-301)	25
i.	Steam Generator Steam and Water Temperature (TE-601)	25
j.	Regulator Rod	25
k.	Regulator Rod Velocity Demand	25
2.	Reactor in Automatic Control	26
a.	Primary Loop Sodium Flow	26
b.	Neutron Flux	26
c.	Neutron Flux Rate of Change, dn/dt	26
d.	Core Outlet Temperature (TE-115)	26
e.	Reactor Outlet Temperature (TE-201)	27
f.	Steam Generator Sodium Inlet Temperature (TE-308)	27
g.	Reactor Inlet Sodium Temperature (TE-209)	27
h.	Steam Generator Sodium Outlet Temperature (TE-301)	27
i.	Steam Generator Steam and Water Temperature (TE-601)	28
j.	Regulator Rod	28
k.	Regulator Rod Velocity Demand	28
B.	ANALYSIS	28
VI.	CONCLUSIONS	33
	REFERENCES	35

LIST OF ILLUSTRATIONS

<u>Fig.</u>		<u>Page</u>
1.	Perspective View of Reactor	4
2.	Reactor Cross Section	5
3.	Heat Removal Systems	7
4.	Schematic Diagram of Resistance Temperature Detector Locations in the Fermi Reactor Heat Transport System	11
5.	Bridge Circuit for Resistance Thermometer Detectors	15
6.	Temperature Detector Signal Circuits	16
7.	Parameter Responses Following Single-Circuit Shutdown During Manual Control	22
8.	Parameter Responses Following Single-Circuit Shutdown During Automatic Control	23

LIST OF TABLES

<u>Table</u>		<u>Page</u>
1.	Recorder Scale Spans Used During Single-Circuit Shutdown Test at 20 Mwt	10
2.	Maximum Thermal Transients of Principal Plant Parameters Observed in Data Recorded During Single-Circuit Shutdown Test at 20 Mwt	29
3.	Transport Time Delays of Principal Plant Parameters Relative to Instant of Pump Trip in Single-Circuit Shutdown Test at 20 Mwt	30
4.	Sodium Flow Rates in the Primary and Secondary Heat Transport Loops During the Single-Circuit Shutdown Test at 20 Mwt	31

SUMMARY

This report describes a single-circuit shutdown test of the Enrico Fermi reactor at a power level of 20 Mwt. The test consisted of two single-circuit shutdowns, the first being conducted with the reactor and the feed-water flow control systems in manual control, and the second with the reactor system in automatic control.

Results in the first part of the test show a maximum negative dn/dt (rate-of-change of neutron flux) of 0.75 Mwt/min. due to the core temperature coefficient with stabilization at 13.4 Mwt, representing 67 per cent of the initial power level. Power stability was reached in a nonoscillatory decay in approximately 12 min. The temperature of the sodium outlet of the reactor exhibited an increase of approximately 4 F, with similar transients occurring in the temperatures of the sodium inlet and outlet of the steam generator and the sodium inlets of the loops remaining in operation.

Temperature transients observed during the second part of the test were substantially like those of the first part. In the automatic mode, the negative dn/dt due to the core temperature coefficient was 1.2 Mw/min, with the power level stabilizing at 16.2 Mwt, representing 81 per cent of the initial power level. Power stability was reached by a slightly oscillatory decay in approximately 17 min.

Inasmuch as the test power level was low, the specific rates-of-change of power and the thermal transients which were experienced at 20 Mwt were well within established safety limits. Operation of the reactor control system, the feedwater control system, and single-circuit control interlocks were found to be satisfactory.

I. PURPOSE OF TEST

The purpose of the single-circuit shutdown test was to determine the dynamic behavior of the Fermi reactor when one coolant circuit is lost through deactivation by the single-circuit shutdown interlock provisions and to observe the responses of the reactor and the feedwater flow control systems to the perturbations introduced.

II. GENERAL DESCRIPTION OF THE ENRICO FERMI REACTOR¹

The Fermi reactor and its associated structures are shown in perspective in Fig. 1. The reactor is contained in a stainless steel reactor vessel sealed at the top by a rotating shield plug which supports the control mechanism, the fuel subassembly holddown mechanism, and the fuel subassembly handling mechanism. The reactor vessel is surrounded by neutron shielding material of borated and plain graphite which is contained inside the primary shield tank. The reactor is of the fast-breeder type, cooled by sodium, and it is operated at essentially atmospheric pressure. The maximum reactor power obtainable with the first core loading (Core A) is 200 Mwt.

The core and blanket are located in the lower reactor vessel and consist of 2.65-in. -square subassemblies containing the fuel pins and blanket rods. The core and blanket subassemblies are arranged to approximate a cylinder about 80 in. in diameter and 70 in. high. The core, which is contained in the central portion of the core subassemblies, approximates a right cylinder 31 in. in diameter and 31 in. high; it is axially and radially surrounded by the breeder blankets. The fuel in the first core loading consists of zirconium-clad pins containing U-10 w/o molybdenum alloy with the uranium enriched to 25.6 w/o U-235. The blanket is depleted U-3 w/o molybdenum alloy.

The reactor cross section, shown in Fig. 2, indicates the placement of individual components within the lower reactor vessel. There is a total of 149 central lattice positions used for core and inner radial blanket subassemblies and operating control and safety rods, all of which are supplied with sodium coolant flowing upwards from a plenum which is fed directly from the discharge of the primary sodium pumps. Owing to the large pressure drop across this region, through which flows about 90 per cent of the sodium, the subassemblies located in it require downward mechanical restraint in addition to their own weight, to prevent ejection from the lower support plates. This restraint is supplied by the holddown mechanism.

The lattice positions surrounding the inner radial blanket comprise the outer radial blanket area and, when filled with outer radial blanket subassemblies, form an annular region whose top and bottom are at the same elevation as the top and bottom of the inner radial blanket. Surrounding the outer radial blanket are lattice positions used for steel subassemblies which provide thermal and neutron shielding. The lattice positions occupied by outer radial blanket and shielding subassemblies are supplied with sodium coolant from a low-pressure plenum which is fed from the primary sodium pumps through throttle valves. The force produced by the pressure drop

FIG. 1 PERSPECTIVE VIEW OF REACTOR

FIG. 2 REACTOR CROSS SECTION

acting on these subassemblies is less than their weight so that no holddown is required.

Control of the reactor is accomplished by two operating control rods and seven safety rods. All the rods are of the poison type, containing boron carbide (B_4C) in which the boron is enriched in B-10. One operating control rod is for regulating purposes and the other is for shimmiing; they have maximum reactivity insertion rates of about 2/3 cent/sec and 1 cent/min, respectively. Together, they provide a total of 92 cents of reactivity to compensate for temperature override, burnup, and operating flexibility. The seven safety rods, which are located around the core about 7 in. from its vertical centerline, provide a shutdown reactivity of over 8 dollars. During reactor operation, the safety rods are poised just above the axial blanket section of the core so that they can be rapidly inserted into the core if necessary for scram, or driven in at the rate of 10 ft/min. The safety rods provide the shutdown during refueling operations.

The plant is designed for manual approach to criticality and low-power operation, and for either manual or automatic plant loading in the power range. Plant loading is determined by the reactor, not by system demand.

Heat produced in the reactor is removed by three primary coolant loops contained within the reactor building. Flow in the primary coolant system is upwards through the core and blanket into a large plenum region designated the upper reactor pool. Because the primary sodium is radioactive, its heat is transferred by three intermediate heat exchangers within the reactor building to three nonradioactive secondary sodium loops, and, finally, to three once-through steam generators in the steam generator building. Here steam is produced to operate the turbine in the adjacent turbine house. The loops of the heat removal circuits are shown in Fig. 3.

FIG. 3 HEAT REMOVAL SYSTEMS

III. EXPERIMENTAL APPARATUS AND EQUIPMENT

An understanding of reactor control and overall system operation requires knowledge of plant instrumentation employed for the principal plant parameters. To secure accurate test data, it was necessary to properly utilize the circuit of a detector, and in some cases even provide special auxiliary circuitry to prevent loading, which would alter the signal voltage and upset normal operation of the circuit. In recognition of these facts, this section presents information on the detectors of main interest in this test, their main function, and a brief account of the signal-handling circuitry employed to obtain the signal recording. The signal parameters recorded during this test are listed in Table 1 which shows the scale spans used for the strip-chart recorders.

A. DETECTORS

Brief descriptions of the detectors of concern in this test and their physical locations and plant functions are presented below. Locations of the temperature detectors in the heat transport system are illustrated schematically in Fig. 4.

1. Reactor Outlet Temperature (TE-201 and TE-217)

The temperature detector used to sense reactor outlet temperature is a resistance thermometer having a time constant of 6.07 sec (Ref. 2). One such detector is provided in each of the three 30-in. reactor outlet pipes; mounting is in a well-type arrangement. These detectors are given the designation TE-201 followed by a number representing the coolant circuit number. The TE-201 detectors are used in the reactor control system, reactor setback system, and the feedwater flow control system. To avoid impairing the normal control functions of the detector, connection of the strip chart-recorder was made at the output of the demodulator. The demodulator provided a low source impedance and thereby prevented introduction of reading errors due to loading by the input circuit of the recorder.

The detectors designated TE-217 (followed by a number representing the coolant circuit number) are sheathed thermocouples located in thermowells in the 30-in. reactor outlet pipe. The time constant of the basic detector is about 1 sec. These detectors have been installed as back-up for the TE-201 detectors, and are specifically intended to be replaceable. The TE-217 detectors are located nearer the reactor proper than are the TE-201 detectors, and are inside the primary shield tank, while the TE-201 detectors are outside the primary shield tank but inside the secondary shield wall.

TABLE 1 - RECORDER SCALE SPANS USED DURING
SINGLE-CIRCUIT SHUTDOWN TEST AT 20 Mwt

<u>Parameter</u>	<u>Manual Control Mode</u>	<u>Automatic Control Mode</u>
TE-115-9	500-600 F	500-600 F
TE-201-3	500-550 F	500-550 F
TE-209-3	500-550 F	500-550 F
TE-301-1C	480-530 F	480-530 F
TE-301-3C	480-530 F	480-530 F
TE-308-1	500-550 F	500-550 F
TE-308-3	500-550 F	500-550 F
Neutron Flux	$0 - 1 \times 10^{-5}$ amp	$0 - 1 \times 10^{-5}$ amp
TE-601-3	500-550 F	500-550 F
FE-201-1	-1×10^6 to 5×10^6 lb/hr	-1×10^6 to 5×10^6 lb/hr
FE-201-3	-1×10^6 to 5×10^6 lb/hr	-1×10^6 to 5×10^6 lb/hr
dn/dt	-50 to +50 Mw/min	-10 to +10 Mw/min
Velocity Demand	± 15 in.	± 15 in.
Regulator Rod Position	0 - 9.5 in.	0 - 9.5 in.

**FIG. 4 SCHEMATIC DIAGRAM OF RESISTANCE TEMPERATURE DETECTOR LOCATIONS
IN THE FERMI REACTOR HEAT TRANSPORT SYSTEM**

A stainless steel guide tube (3/8-in. OD) connects the thermowell and the thermocouple installation port. The guide tubes for TE-217-1 and TE-217-3 are about 12 ft long, and have two 60-deg bends; the offset in this line also is about 2-1/2 ft. It was evident from the response of TE-217-2 during the 20-Mwt and 67-Mwt scram tests that the thermocouple was not properly seated in the thermowell.³ The thermal lag of TE-217-2 has been found to be about four times that of TE-217-3. The seating problem is probably the result of an accumulation of dirt in the thermowell.

2. Core Outlet Temperature (TE-115)

Fifty-five thermocouples are provided in the Fermi reactor to monitor the coolant outlet temperatures of the core subassemblies. These thermocouples have junctions grounded to the sheath (contact only, not welded) for a fast response of about 1 sec. The junctions have become ungrounded in several of the TE-115 thermocouples in the course of normal operations because of a difference of coefficients of linear expansion of the sheath and conductor materials. The response of an ungrounded thermocouple could be of the order of 5 sec. Mounting inside the holddown finger is such that the thermocouple is close to the conical cavity which seats on the subassembly handling head. These sensors are given the designation of TE-115.

The TE-115 detectors are used in the reactor safety system and core temperature automatic scanner and alarm. Core outlet temperature signals from four TE-115 thermocouples control bistable multivibrators which feed the 2-out-of-4 AND gate logic circuit in the safety system. The AND gate is connected in a manner that two "unsafe" signals must be present before an "unsafe" signal is applied to the scram logic OR gates, scrambling the reactor.

3. Steam Generator Sodium Inlet Temperature (TE-308)

The TE-308 is a resistance-type temperature detector mounted in a thermowell in the 18-in. pipe of the inlet piping system to the steam generator. More specifically, the detector location is in the region of the "T" connection in the 18-in. pipe which splits the steam generator inlet sodium flow to the two 12-in. inlet nozzles of the steam generator. The time constant of this sensor is about 25 sec. The detectors are given the designation of TE-308.

The TE-308 detectors have no control functions, but provide information to an indicator in the control room. A pneumatic retransmitter is used to drive the indicator. A separate bridge circuit was provided to obtain a signal from the resistance thermometer detector (RTD). A suitable scale factor was set up for the recorder by means of the recorder gain adjustment.

4. Steam Generator Sodium Outlet Temperature (TE-301)

The steam generator outlet sodium temperature is used as one of the basic signals in the feedwater control system to establish a demand signal for feedwater flow. Emergency feedwater flow also is controlled by the steam generator sodium outlet temperature. The TE-301 detector as originally provided in the outlet piping of the steam generator is a resistance thermometer located about 1-1/2 ft from the bottom of the steam generator vessel. Response of the detector is estimated to be 2.5 sec. Early in the reactor test program it was observed that flow-channeling of relatively hot sodium along the walls of the sodium outlet piping of the steam generator was producing erratic and high readings of TE-301 because of poor mixing at this point. To obtain a better temperature signal, thermocouples were installed in thermowells about 75 equivalent pipe diameters downstream from the original TE-301 detector location. Three thermocouples have been provided in each loop and carry the designations D, E and F. The response of the basic thermocouple is about 1.5 sec.

5. Reactor Inlet Temperature (TE-209)

The reactor inlet temperature detector designated TE-209 is located in the 6-in. standpipe. The TE-209 temperature detector is a resistance thermometer identical to the TE-201 detector which has a time constant of 6.07 sec.

6. Water-Steam Temperature (TE-601)

The TE-601 is a resistance-type temperature detector thermowell, mounted in the outlet pipe of the steam generator. The TE-601 is the same type as the TE-308, having a time constant of 25 sec. The water-steam temperature signal is used in the reactor program controller and plant recorders and indicators.

7. Neutron Flux

The neutron flux detector for measuring and monitoring reactor power was provided as separate apparatus, independent of the reactor safety system and control instrumentation. The detector utilized was NE-116, a B-10 lined compensated ion chamber (Reuter-Stokes, Inc.). Installation of the detector is in neutron-counter tube No. 5, the location of which is illustrated in Fig. 3 of Reference 4. Output of the ion chamber is read on an electrometer (Keithley Instruments Inc. Model 415) located in the control room. A coaxial cable approximately 250 ft long interconnects the ion chamber and the electrometer. The time constant of the neutron flux measuring instrumentation is of the order of 10 msec.

8. Sodium Flowmeters

Flow in the primary and secondary sodium loops is measured by permanent magnet flowmeters. Two flowmeters are in each primary loop and one is in each secondary loop. One of the flowmeters of the primary loop is in the 14-in. pipe which feeds the core and inner radial blanket, and the second flowmeter is in the 6-in. pipe which feeds the outer radial blanket. The flowmeters in the 6-in. pipe drive only indicators, and have no control functions. The 14-in. flowmeters, however, supply signals to the scram circuits, single-circuit shutdown interlocks, recorders, console indicators and flow alarms. The designation for the 14-in. and 6-in. flowmeters are FE-201 and FE-202, respectively, followed by the circuit number. For example, the 14-in. and 6-in. flowmeters in the primary loop of circuit No. 1 are designated FE-201-1 and FE-202-1. The secondary loop flowmeters are in the 18-in. pipe. The flowmeters of the secondary loop are designated FE-301, followed by the circuit number.

B. SIGNAL CONDITIONING CIRCUITS AND ADJUNCTIVE APPARATUS

Since a readily available signal was not existent in some cases to record a plant parameter, a special bridge circuit had to be provided for some RTD's to enable their operation and calibration; the basic circuit is shown in Fig. 5. To simulate a temperature signal, the variable resistance is set to a value which corresponds to the RTD resistance. Special care had to be exercised with some detectors to prevent upsetting the normal operation of the circuit by recorder loading or by an accidental short circuit which could impair or otherwise alter the detector output signal. Inasmuch as the output of the TE-201 detectors has important control functions, the output of each TE-201 detector recorded was taken at the output of the demodulator where the circuit impedance is low. An available spare TE-301 detector already in the system was placed into use for the tests. The TE-209 detector utilized was temporarily diverted from its normal function in the safety rod withdrawal interlock circuit. A signal from the TE-601 detector was obtained from the retransmitting slidewire of the plant recorder into which the detector normally works. The TE-301 signal was obtained at the output of a load-isolating (low output impedance) amplifier.

Figures 6a, 6b, and 6c show, in block diagram form, the circuits of the resistance thermometer detectors, and Fig. 6d shows the circuit for the TE-301 thermocouples.

Adjunctive apparatus and equipment employed in the single-circuit shutdown tests consisted of the following major items:

FIG. 5 BRIDGE CIRCUIT FOR RESISTANCE THERMOMETER DETECTORS

FIG. 6 TEMPERATURE DETECTOR SIGNAL CIRCUITS

- Eight-channel strip-chart recorders having a minimum response of 50 cps, and containing suppression and span calibrating provisions -- used for recording the system parameters such as temperature, regulator rod position, and neutron flux.
- Millivolt box -- used for simulating the output of a thermocouple.
- Four-arm resistance bridge circuits, equipped with a battery power supply for excitation of the bridge -- used for simulating an RTD during recorder calibration, and to provide the required bridge circuitry for operation of RTD's TE-209 and TE-308, apart from their respective plant recorders.

C. COOLANT SYSTEM

1. Primary Loop

Primary sodium flows by gravity through three 30-in. pipes from the upper reactor vessel to the shell side of the intermediate heat exchanger (IHX) and then to the pump tank. Sodium is pumped from the pump tank first into a 16-in. header and then through a Tee into a 6-in. and 14-in. pipe. The 14-in. pipe delivers approximately 87 per cent of the flow to the plenum serving the reactor core and approximately 13 per cent flows to the plenum serving the radial blanket through the 6-in. pipe. The flow of coolant to the blanket plenum can be adjusted by means of a partially closing throttle valve.

The three primary pumps are vertical-shaft, single-stage, centrifugal pumps mounted in tanks. Each pump tank is connected to the IHX with a 30-in. -diam pipe. The design point for each of the three primary pumps is 11,800 gpm at total dynamic head of 310 ft.

The main motor drives for the primary pumps are 3-phase, 60-cycle, wound rotor motors rated at 1000 hp, 900 rpm, and 4800 volts. The sodium flow rate in each loop may be adjusted by varying the speeds of the pump drive motors. This speed change is accomplished by varying the external electrical resistance of the rotor circuit. The liquid rheostats provide stepless resistance changes. The liquid used is a solution of sodium carbonate in water.

2. Secondary Loop

The secondary coolant loop is the intermediate link that transfers heat from the primary coolant loop to the steam generator. There are three secondary loops, each consisting of the tube side of the IHX, the shell side of the steam generator, the secondary coolant pump, and connecting piping. The pressure of the secondary coolant loop is maintained

slightly higher than that of the primary loop to prevent leakage of radioactive primary coolant into unshielded areas. Each secondary sodium loop is independent.

The three secondary pumps are vertically-mounted, single-stage, centrifugal pumps, with their single-suction and single-discharge connections welded directly in the 18-in. piping. The design point for each of the secondary pumps is 13,000 gpm at total dynamic head of 100 ft.

The main motor drives for the secondary pumps are 3-phase, 60-cycle, induction motors rated at 350 hp, 900 rpm, and 4800 volts. Speed control of the secondary pumps is accomplished by means of an electronic circuit which controls the field excitation of an eddy current coupling.

3. Single-Circuit Shutdown Interlocks

The purpose of the single-circuit shutdown interlock is to remove a coolant circuit from the operating system to protect against damage from thermal shock in the event of a loop malfunction. A circuit is defined as one primary sodium loop, its secondary sodium loop, and steam generator feedwater loop. The single-circuit shutdown interlock deactivates both the primary and secondary sodium pumps, and closes feedwater valves in the affected circuit when any one or a combination of the abnormal conditions listed below exists:

- a. Primary sodium flow < 25% of normal flow
- b. Secondary sodium flow < 60% of normal flow
- c. Feedwater flow > 115% of normal flow
- d. Feedwater flow < 85% of normal flow
- e. Steam generator sodium outlet temperature > 550 F
- f. Steam generator sodium outlet temperature < 485 F
- g. Steam generator sodium outlet temperature > 575 F
- h. Steam generator sodium outlet temperature < 425 F

Shutdown conditions "c" and "d" are negated when feedwater flow control is manual. Shutdown conditions "e" and "f" are locked out or negated when a scram occurs, leaving the wider-tolerance limits "g" and "h" to be in effect. The high limit of 550 F and the low limit of 485 F comprise the allowable parameter extremes for normal operation.

IV. EXPERIMENTAL PROCEDURE AND PLANT CONDITIONS

The single-circuit shutdown test basically consisted of (1) the de-activation of one of the three coolant circuits in the reactor heat transport system while the reactor was in operation at the test power level and, (2) the observation of the plant parameters for transient effects and the controls for correct performance.² The plant safety system and interlock provisions allow continuation of plant operation on two coolant circuits, provided that the plant parameters are within set normal limits. Shutdown of two of three coolant circuits will execute a reactor scram; therefore, all three coolant circuits must be operational to perform a single-circuit shutdown test. The test performed consisted of two parts -- the first with the reactor and feedwater system in manual control, and the second with the reactor in automatic control but the feedwater system in manual control. All safety devices and interlocks were required to be operable, and no abnormal system conditions were needed to perform the tests. The execution of the single-circuit shutdown test entailed: (1) setting the reactor test power level and required sodium flow, (2) establishment of stable operating conditions, (3) reading the steady-state values of plant parameters, (4) setting up and starting of the strip-chart recorders for reading the transient-observed parameters, (5) tripping off the pumps of the circuit shutdown, (6) recording the parameters until level-off of the transients, (7) readjusting the sodium flow in the two operating primary loops according to established operating procedures until the loop flows are 100 per cent, and (8) recording the plant parameters after stabilization.

When a single-circuit shutdown occurs, the obtained total sodium flow rate of the two operating primary loops rises from the 66.7 per cent value to approximately 83 per cent of the original three-loop flow, i. e., the contribution of each of the operating loops increases from 33.3 to about 41.5 per cent. Flows in the secondary loops, however, remain at the 33.3 per cent value because they are isolated from each other. To re-establish operating conditions after a single-circuit shutdown, the operating procedure calls for adjustment of the speed of the operating primary pumps to secure the original flow rate per loop.

V. EXPERIMENTAL RESULTS AND ANALYSIS

A. RESULTS

The single-circuit shutdown test at 20 Mwt, consisting of two parts, was successfully completed. The parameter responses resulting from the single-circuit shutdown are shown in Figs. 7 and 8. The curves are re-plotted on a contracted time scale and an expanded temperature scale to convey clearly the absolute and relative changes which occurred in the parameter values. The zero point on the time scale represents the instant of pump trip. Presented in the following are descriptive summaries of all the data recorded on the strip charts. These summaries are included for completeness, and are essential especially for describing parameters such as the regulator rod velocity demand signal, sodium flows, noisy temperature signals, and others which are too complex or too fast in response to be accurately replotted.

1. Reactor in Manual Control

a. Primary Loop Sodium Flow

The single-circuit shutdown was executed by shutdown of coolant circuit No. 1 (see Fig. 3). Up to the instant of trip, the primary sodium flow rate was approximately 3.12×10^6 lb/hr in each loop. After pump trip, the flow drop-off of loop No. 1 was fast, with level-off at reverse flow of 0.4×10^6 lb/hr being reached in about 5 sec. Flow in the active loops indicated an increase of about 24 per cent, bringing the total flow of the two operating loops from 67 per cent of original flow up to 83 per cent. The recording of the signal from FE-201-3 (the sodium flow in the 14-in. pipe of loop No. 3) contained perceptible fluctuations, of the order of 0.1×10^{-6} lb/hr, whereas the flow trace of loop No. 1 (FE-201-1) was smooth and free of perceptible fluctuations.

The cause of the noise in the flow signal from FE-201-3 is attributed to flow turbulence in the 14-in. pipe on which the flowmeter is mounted. The flowmeter is located a distance of only about eight pipe diameters from the "T" connection to the 16-in. outlet pipe of the sodium pump.

b. Neutron Flux

Drop-off of neutron flux began simultaneously with start of the drop-off of sodium flow in the loop which was shut down. Up to the instant of shutdown, the electrometer reading was 0.66×10^{-5} amp,

FIG. 7 PARAMETER RESPONSES FOLLOWING SINGLE-CIRCUIT SHUTDOWN DURING MANUAL CONTROL

FIG. 8 PARAMETER RESPONSES FOLLOWING SINGLE-CIRCUIT SHUTDOWN DURING AUTOMATIC CONTROL

corresponding to a reactor power level of 19.7 Mwt. After the shutdown, drop-off of the electrometer reading was of the general form of an exponential decay curve, with level-off in 480 sec to the value of 0.46×10^{-5} amp, corresponding to a reactor power level of about 13.7 Mwt. From the rate-of-change observed, the value of the gross power derivative for the power change which occurred was $6.0 \text{ Mw}/8 \text{ min} = 0.75 \text{ Mw}/\text{min}$.

c. Neutron Flux Rate-of-Change, dn/dt

A readable recording of dn/dt was not obtained because a recorder scale factor of 50 Mw/min (rather than 10 Mw/min) was inadvertently used, relegating the neutron flux change of 0.75 Mw/min to an imperceptible level.

d. Core Outlet Temperature (TE-115)

At the same instant that the neutron flux decay appeared, the core outlet temperature measured by thermocouple TE-115-9 started to increase and reached the maximum value in approximately 2 sec. Up to the instant of the start of the increase, the reading of TE-115-9 was 551 F. The reading increased to 555 F as a result of the single-circuit shutdown, and dropped off smoothly to 546 F in approximately 590 sec.

e. Reactor Outlet Temperature (TE-201)

The recording of the reactor outlet temperature showed a temperature of 543 F up to the instant where the temperature began to rise as a result of the single-circuit shutdown. The transport lag was approximately 12 sec. The TE-201-3 temperature rose to a broad peak at 547 F in about 250 sec and then dropped to the level-off value of 542 F in about 700 sec.

f. Steam Generator Sodium Inlet Temperature (TE-308)

After the transport delay period of approximately 34 sec, the temperature of TE-308-3 rose from the before-pump trip value of 538 F to a broad peak at 542 F in 360 sec, dropping off thereafter to the level-off value of 539 F in about 700 sec.

In the loop which was shut down, the TE-308-1 temperature, after the transport delay period of approximately 40 sec, dropped off in a somewhat nonuniform decay curve. The response included a dip from 537 F to 517 F at 540 sec followed by a rise to the level-off value of 523 F in about 690 sec.

g. Reactor Inlet Sodium Temperature (TE-209)

The reactor inlet sodium temperature as recorded (TE-209-3) was 516 F up to the instant of the shutdown. Following the shutdown, a transport delay of 10 sec occurred, after which the temperature rose smoothly to a broad peak of 522 F about 330 sec after the shutdown, and then leveled off at 518 F after about 700 sec.

h. Steam Generator Sodium Outlet Temperature (TE-301)

Recordings were obtained of the steam generator sodium outlet temperature of the loop shut down (TE-301-1C), as well as one of the two loops operating (TE-301-3C). In the circuit which was shut down, the recorded trace exhibited pronounced random fluctuations as high as 6 F in peak-to-peak amplitude, up to the instant of loop shutdown. Almost immediately following circuit shutdown, TE-301-1C exhibited a relatively sharp dip from about 515 F to 486 F which occurred at 150 sec, rising again to 511 F at 450 sec, and then dropping off at the level-off value of 505 F after about 700 sec. About 30 sec after shutdown of the circuits, the random fluctuations of TE-301-1C reached a maximum peak-to-peak value of about 10 F, and thereafter diminished to an imperceptible level by the time the level-off value was reached.

The recording of TE-301-3C was relatively smooth and free of random fluctuations. Following shutdown, and after a short transport delay period, TE-301-3C exhibited an increase from the 522 F pre-shutdown level, rising to a broad peak of 522 F at about 400 sec, and slowly dropping thereafter to the level-off value of 514 F after about 700 sec.

i. Steam Generator Steam and Water Temperature (TE-601)

The recorded steam generator outlet temperature was that of circuit No. 3, detector TE-601-3 -- one of the two circuits remaining in operation. The outlet temperature exhibited no change from the 531 F value for 147 sec after shutdown of loop No. 1; TE-601-3 had a slow decrease for about 447 sec, dropping to 529 F, and then slowly rising back to 531 F, stabilizing beyond about 700 sec.

j. Regulator Rod

Inasmuch as the reactor was in manual control during this test, no motion of the regulator rod occurred as a result of the single-circuit shutdown.

k. Regulator Rod Velocity Demand

The velocity demand signal indicated an anomalous behavior immediately following loop shutdown. A down command existed for about 2

sec, after which the polarity changed to an up command for about 70 sec. During the 70-sec period a sharp, transient down command occurred for a period of about 4 sec. Further information about the transient is unavailable except that noise bursts have previously been observed in this control and the cause is being investigated for corrective action.

2. Reactor in Automatic Control

a. Primary Loop Sodium Flow

The single-circuit shutdown was executed, as in the first test, by shutdown of coolant circuit No. 1. Up to the instant of shutdown, the primary sodium flow rate was approximately 3.1×10^6 lb/hr. After pump trip, the flow drop-off of loop No. 1 was steep, with level-off at reverse flow of approximately 0.4×10^6 lb/hr being reached in 4-5 sec. The percentage of flow increase and the noise observed in the flow signal from flow-meter FE-201-3 were as described under Section V-A-1-a.

b. Neutron Flux

As in the first test, drop-off of neutron flux occurred simultaneously with start of the drop-off of sodium flow in loop No. 1 which was shut down. Up to the instant of shutdown, the electrometer reading was 0.6×10^{-5} amp, corresponding to a reactor power level of 20 Mwt. Upon trip of loop No. 1, the electrometer signal dropped in the manner of an exponential decay curve to a reading of 0.41×10^{-5} amp or 13.4 Mwt in 450 sec, for a gross power derivative of 1.2 Mw/min. The signal then rose smoothly to 0.40×10^{-5} amp or 16.2 Mwt and leveled off in about 1000 sec.

c. Neutron Flux Rate-of-change, dn/dt

At the instant of shutdown of loop No. 1, the neutron flux rate-of-change signal appeared with a negative polarity and a peak value of about 2 Mw/min. The duration of the signal was about 3 sec. About 60 sec after the shutdown, a positive polarity rate-of-change signal of about 1 Mw/min appeared, followed by a negative polarity signal of the same magnitude at 75 sec. After 75 sec, essentially no signals were present.

d. Core Outlet Temperature (TE-115)

Up to the instant of the single-circuit shutdown, the core outlet temperature as detected by TE-115-9 was 538 F. Following shutdown, the reading increased to 543 F within about 2 sec, remained at this level for about 90 sec, and thereafter slowly dropped to 532 F at 575 sec. Beyond the 575-sec point, the temperature again rose, and leveled off at 536 F after about 850 sec from the instant of circuit shutdown.

e. Reactor Outlet Temperature (TE-201)

Up to the instant of circuit No. 1 shutdown, the TE-201-3 temperature was constant at 541 F. After shutdown and a transport delay of approximately 12 sec, the temperature rose monotonically to a peak of 544 F in 190 sec. Following the peak, the temperature dropped to 539 F in 610 sec, rose again, and leveled off at 539 F in 1000 sec.

f. Steam Generator Sodium Inlet Temperature (TE-308)

The steam generator inlet temperature of one of the circuits not shut down was recorded in addition to that of the circuit which was shut down. In the latter, the reading of TE-308-1 up to the instant of pump trip was 535 F. After the transport delay period of approximately 40 sec, TE-308-1 dropped monotonically to the level-off value of 529 F in 300 sec.

In circuit No. 3, not shut down, TE-308-3 measured 535 F up to the instant of pump trip. After the transport delay period of approximately 38 sec, the TE-308-3 reading rose to a broad peak at 540 F in 290 sec then dropped to 536 F in 720 sec and rose, finally, to the level-off value of 539 F in 1200 sec.

g. Reactor Inlet Sodium Temperature (TE-209)

Up to the instant of pump trip the reactor inlet sodium temperature TE-209-3 read 514 F. After the 14-sec transport delay, the temperature rose to a broad peak at 518 F in 250 sec, then slowly dropped to 515 F in 850 sec, and finally leveled off at 517 F in about 1200 sec.

h. Steam Generator Sodium Outlet Temperature (TE-301)

The steam generator sodium outlet temperature was measured in the circuit shut down by detector TE-301-1C, and in circuit No. 3 by detector TE-301-3C. Random fluctuations of a peak-to-peak value of 5 F were exhibited by TE-301-1C, with an average reading of about 513 F up to the instant of pump trip. After circuit shutdown, and following the transport delay period of about 4 sec, TE-301-1C decayed to 488 F in about 200 sec, with several random fluctuations being evident as severe as 10 F, peak-to-peak. Following the low point of 488 F, the temperature rose slowly to 493 F in about 1000 sec, with level-off beyond the end of the recording. After about the 200-sec point, the TE-301-1C trace was smooth and free of significant fluctuations.

In circuit No. 3, TE-301-3C read 511 F up to the instant of pump trip. Following the trip, the temperature rose to a peak of 514 F in 310 sec, then dropped slowly to 510 F in 680 sec; thereafter, it rose slowly

to 512 F, without apparent level-off, in 1200 sec. This was the limit of the recording.

i. Steam Generator Steam and Water Temperature (TE-601)

The recorded steam and water temperature of the steam generator was that of TE-601-3 in circuit No. 3. Up to the instant of shutdown of circuit No. 1, as well as during the 140-sec transport delay period, TE-601-3 was essentially constant at 526 F. Following the transport delay period, the reading of TE-601-3 dropped only to 525 F in 300 sec after the shutdown, and rose again to 526 F, at which it leveled off in 590 sec.

j. Regulator Rod

Almost immediately after shutdown of circuit No. 1, the regulator rod position was observed to undergo a 0.4-in. outward movement for a duration of about 6 sec. About 60 sec after the shutdown, the regulator rod was observed to undergo an inward and outward movement transient of about 0.4 - 0.6-in. amplitude, lasting for about 6 sec. Following the second transient, the regulator rod withdrew about 0.3 in. and remained in the higher position up to about 460 sec after the shutdown. Beyond 460 sec, the regulator rod twice shifted position stepwise, each time a distance of about 0.2 in.

k. Regulator Rod Velocity Demand

The polarity of the velocity demand signal was consistent with the direction of travel of the regulator rod.

B. ANALYSIS

The areas of main interest in the results obtained were the magnitude of the negative rate-of-change of power due to the reactor power coefficient, the response of the reactor outlet temperature of the loops left operating, the steam generator sodium outlet temperature of the loop shut down, and the performance of the single-circuit shutdown interlocks. In the test of the reactor and feedwater controls under the manual mode, the observed rate-of-change of power was -0.75 Mw/min. The response of TE-201-3, in an operating circuit, exhibited a rise followed by a perceptible oscillation, continuing for one full cycle. The period of the cycle was approximately 830 sec. The oscillatory characteristic of TE-201-3 resulted in similar responses in TE-308-3, TE-301-3C and TE-209-3, but in each case the rate of change in temperature was well within the design limits of the plant because of the low reactor power involved.

In the test in which the reactor control was in the automatic mode, the observed rate-of-change of power was -1.2 Mw/min. As in the test for the

manual mode, the characteristic of TE-201-3 was oscillatory, and highly damped, resulting in similar characteristics in TE-308-3, TE-301-3C and TE-209-3; but, in each case the amplitude of the transient was well within the design limits of the plant.

In Table 2 are listed the maximum rates-of-change observed in the recorded principal plant temperatures, and the approximate duration of the change for both tests. Table 3 lists the transport time delays as read from the strip chart. Table 4 lists the sodium flow rates in the primary and secondary heat transport systems.

Performances of the single-circuit shutdown and other system interlocks were found satisfactory in both tests at 20 Mwt.

TABLE 2 - MAXIMUM THERMAL TRANSIENTS OF PRINCIPAL PLANT PARAMETERS OBSERVED IN DATA RECORDED DURING SINGLE-CIRCUIT SHUTDOWN TEST AT 20 Mwt *

<u>Parameter</u>	<u>Rate-of-Change, F/sec</u>	<u>Duration, sec</u>
TE-115-9	+ 3.75	2
TE-201-3	+ 0.98	30
TE-308-1	- 0.13	30
TE-308-3	+ 0.03	70***
TE-301-1C **	+10	2
TE-301-3C	+ 0.02	100
TE-209-3	+ 0.06	30
TE-601-3	- 0.02	80

* Transients were essentially the same for both manual and automatic control of the reactor.

** Random fluctuations.

*** Long duration relative to TE-201-3 due to large time constant of TE-308-3 detector.

TABLE 3 - TRANSPORT TIME DELAYS OF PRINCIPAL PLANT
PARAMETERS RELATIVE TO INSTANT OF PUMP TRIP*
IN SINGLE-CIRCUIT SHUTDOWN TEST AT 20 Mwt

<u>Parameter**</u>	<u>Transport Time, sec</u>	
	<u>Manual Control Mode</u>	<u>Automatic Control Mode</u>
TE-115-9	< 1	< 1
TE-201-3	12	12
TE-209-3	11	14
TE-301-1C	≈ 4	≈ 4
TE-301-3C	30	33
TE-308-1	40	40
TE-308-3	34	38
Neutron Flux	< 1	< 1
TE-601-3	127	140
FE-201-3	≈ 0	≈ 0
dn/dt	< 1	< 1
Velocity Demand	< 1	< 1
Regulator Rod Position	(No response Manual Control)	4

* Circuit No. 1 shut down.

** The last digit in the parameter (detector) designation represents the number of the coolant circuit, i. e., TE-201-3 designates the reactor sodium outlet temperature detector in circuit No. 3; TE-308-1 is the sodium inlet temperature of the steam generator in circuit No. 1, etc.

TABLE 4 - SODIUM FLOW RATES IN THE PRIMARY AND SECONDARY HEAT TRANSPORT
LOOPS DURING THE SINGLE-CIRCUIT SHUTDOWN TEST AT 20 Mwt

<u>Test</u>	<u>Primary Loop Sodium Flow, lb/hr x 10⁶</u>					
	<u>Circuit No. 1</u>		<u>Circuit No. 2</u>		<u>Circuit No. 3</u>	
	<u>14-in. Pipe</u>	<u>6-in. Pipe</u>	<u>14-in. Pipe</u>	<u>6-in. Pipe</u>	<u>14-in. Pipe</u>	<u>6-in. Pipe</u>
Manual Control Mode	2.72	0.40	2.60	0.41	2.6	0.40
Automatic Control Mode	2.70	0.40	2.70	0.41	2.5	0.41

	<u>18-in. Secondary Loop Sodium Flow, lb/hr x 10⁶</u>		
	<u>Circuit No. 1</u>	<u>Circuit No. 2</u>	<u>Circuit No. 3</u>
Manual Control Mode	2.92	2.91	2.91
Automatic Control Mode	2.90	2.92	2.91

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VI. CONCLUSIONS

The single-circuit shutdown test conducted at 20 Mwt for the conditions of manual control and automatic control of the reactor has been completed with satisfactory results. As would be expected for the low-power level of the test, the thermal transients which resulted from loss of a coolant circuit were insignificant. Operation of the plant was stable before and after the loop shutdown. Power decrease due to the core temperature coefficient exhibited the characteristics of an exponential decay curve. Performances of the reactor and feedwater controls, plant instrumentation, and safety interlocks were satisfactory in each case.

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