**SCRAM TESTS TO DETERMINE SHUTDOWN TRANSIENT
CHARACTERISTICS OF THE ENRICO FERMI REACTOR AND
ASSOCIATED HEAT TRANSPORT SYSTEM**

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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), who own and operate the reactor proper. The steam generators and electrical generation facilities are owned by The Detroit Edison Company (DECo).

Many people have contributed to the success of the nuclear testing of the Fermi reactor. Listed below are the names of those people, exclusive of the authors, who made a significant contribution to some phase of the work reported in this document.

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SUMMARY

Scram tests were performed at nominal reactor power levels of 9.5, 13.3, 20, and 67 thermal megawatts. During the 9.5-, 13.3-, and 67-Mwt tests, operation was on two coolant loops; whereas, during the 20 Mwt test, operation was on three coolant loops. At 67 Mwt, significant thermal transients of the nature of random fluctuations were observed in the reactor outlet temperature as measured by the temperature detectors. The fluctuations first appeared approximately 15 seconds after the instant of scram and became most pronounced after about 40 seconds. Approximately 220 seconds after the scram, the fluctuations diminished to an imperceptible level. Thermal fluctuations and transients occurring in scrams at 9.5, 13.3, and 20 Mwt were insignificant, apparently because of the low power levels involved. No significant fluctuations were observed in other system temperatures recorded.

Analysis by analog computer to correct for the detector and associated circuit time constants have produced thermal transient data representative of the actual thermal fluctuations measured by the temperature detectors. Results of the 67-megawatt test show that the most severe thermal transient was experienced at the reactor outlet in primary sodium loop No. 1, where a 44°F -per-second transient was recorded over a temperature change of 76°F . In primary sodium loop No. 2, the most severe transient experienced was 20°F per second over a 48°F temperature change. The original design specification for the Fermi reactor stipulates the worst-case temperature transient in the reactor outlet temperature to be 15.8°F per second for a change of 300°F , with a maximum of 500 transients tolerable. Although the temperature rates of change experienced were comparatively high, the total change in temperature in each instance was well below the specification value.

Transport lags of the principal plant parameters were read from the scram test data obtained on strip chart recorders. Transport lags were somewhat smaller in the 67-Mwt case than at the lower reactor power levels. The difference is attributed to a slightly higher coolant flow rate at 67 Mwt. At this power, representative transport time for the reactor outlet sodium temperature is 11 seconds; for the steam generator inlet sodium temperature, 62 seconds; for the reactor inlet sodium temperature, 70 seconds; and for steam generator water and steam temperature, 91 seconds. At the 9.5-, 13.3-, and 20-Mwt power levels, the transport lags are somewhat longer, with representative values of the above parameters being 12, 70, 100, and 139 seconds, respectively. The difference in transport lags is attributed primarily to a higher flow rate for the 67-Mwt case.

Performance of the scram interlocks and reset of the start-up interlocks also were of major interest during the scram tests. Performance of all interlock and reset circuitry, as well as operation of the reactor control system, was found to be satisfactory.

I. PURPOSE OF TEST

The main objectives of the scram tests were to determine the characteristics of transients which occur in the principal plant parameters, to measure transport lags, to obtain the plant parameter time constants, and to verify correct performance of the scram interlock circuitry and reset of the start-up interlocks. Especially important among the interlock actions is reset of the safety rod raise interlock.

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II. GENERAL DESCRIPTION OF THE ENRICO FERMI REACTOR

The Enrico Fermi reactor, with its associated structures, is shown in perspective in Figure 1. The reactor is contained in a stainless steel 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 borated and plain graphite neutron shielding material which is contained inside the primary shield tank. The reactor is of the fast-breeder type, cooled by sodium and operated at essentially atmospheric pressure. The maximum reactor power obtainable with the first core loading (Core A) is 200 Mwt.

The core and blanket region, located in the lower vessel, consists of 2.65-inch-square subassemblies containing the fuel pins and blanket rods. The core and blanket subassemblies are arranged to approximate a cylinder about 80 inches in diameter and 70 inches high. The core, which is contained in the central portion of the core subassemblies, approximates a right cylinder 31 inches in diameter and 31 inches 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 Mo alloy with the uranium enriched to 25.6 w/o U-235. Each subassembly contains 140 fuel pins having a total mass of approximately 4.75 kg of U-235. The blanket is depleted U-3 w/o Mo alloy.

The reactor cross section, shown in Figure 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 these subassemblies are supplied with sodium coolant flowing upward 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 about 90 per cent of the total sodium flows, subassemblies located in it require downward mechanical restraint in addition to their own weight in order 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

FIG. 1 PERSPECTIVE VIEW OF REACTOR

FIG. 2 REACTOR CROSS SECTION

pumps through throttle valves. The pressure drop force acting on these subassemblies is less than their weight; hence, no holddown mechanism is required.

Control of the reactor is accomplished by two operating control rods and eight safety rods. All the rods are of the poison type, made of B_4C using boron enriched in B-10. One operating control rod is for regulating purposes and the other is for shimming; they have average reactivity insertion rates of about one cent per second and one cent per minute, respectively. Together they provide a total of 92 cents of reactivity to compensate for temperature override, burnup, and operating flexibility. The eight safety rods, which are spaced uniformly around and about seven inches from the vertical centerline of the core, provide a shutdown reactivity worth of over eight 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 it becomes necessary to scram the reactor. During normal operation, the safety rods are inserted slowly into the core, where they remain during refueling to provide the necessary shutdown reactivity.

The plant is designed for manual approach to criticality and to 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, all contained within the reactor building. Flow in the primary coolant system is up 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.

A more detailed description of the Fermi reactor may be found in Reference 1.

III. EXPERIMENTAL APPARATUS AND EQUIPMENT

A. TEMPERATURE DETECTORS

A brief description of the principal resistance temperature detectors (RTD's) and their physical locations and plant functions is presented below. Locations of the RTD's in the heat transport system are illustrated schematically in Fig. 3.

1. Reactor Outlet Temperature

The temperature detector used to sense reactor outlet temperature is a resistance thermometer (TE-201) having a time constant of 6.07 seconds.^{2,3} One such detector is provided in each of the three 30-inch reactor outlet pipes; mounting is in a well-type arrangement. The TE-201 detector is used in the reactor control system, reactor setback system, and the feedwater flow control system. To avoid impairing the normal control functions of this detector, connection of the strip chart recorder was made at the output of the demodulator. The demodulator provides a low source impedance and thereby prevents introduction of reading errors due to loading by the input circuit of the recorder.

As backup for the TE-201 detectors, sheathed thermocouples (TE-217) having time constants of 1 second, are located in the 30-inch reactor outlet pipe. These detectors, specifically intended to be replaceable, are located nearer the reactor proper than are the TE-201 detectors, i. e., they are inside the primary shield tank, while the TE-201 detectors are outside the primary shield tank but inside the secondary shield wall.

A 3/8-inch OD stainless steel guide tube connects the thermowell and the thermocouple installation port. The guide tubes for TE-217-1 and TE-217-3 are about 12 feet long and have two 60-degree bends; the offset in this line is about 2-1/2 feet. It was evident from the response of TE-217-2 during the 20- and 67-Mwt scrams that the thermocouple was not properly seated in the thermowell. 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

Fifty-five thermocouples (TE-115) are provided in the Fermi reactor to monitor the coolant outlet temperatures of the core subassemblies. As manufactured, these thermocouples have junctions grounded to the sheath for fast response, which is about 1 second. In several of the

FIG. 3 SCHEMATIC DIAGRAM OF RESISTANCE TEMPERATURE DETECTOR LOCATIONS IN THE FERMI REACTOR HEAT TRANSPORT SYSTEM

thermocouples the junctions have become ungrounded 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 seconds. Mounting inside the holddown finger is such that the thermocouple is close to the conical cavity which seats on the subassembly handling head.

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 each control bistable multivibrators which feed the 2-out-of-4 AND gate logic circuit in the safety system. The AND gate is connected in such 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

The steam generator sodium inlet temperature is monitored by resistance-type temperature sensors (TE-308) mounted in thermowells in the 18-inch pipe of the inlet piping system to the steam generator. More specifically, the location of the detectors is in the region of the T-connection in the 18-inch pipe which splits the steam generator inlet sodium flow to the two 12-inch inlet nozzles of the steam generator. The time constant of this sensor is about 25 seconds.

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

4. Steam Generator Sodium Outlet Temperature

The steam generator outlet 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 detector originally provided in the steam generator outlet piping is a resistance thermometer (TE-301) located about 1-1/2 feet from the bottom of the steam generator vessel. Response of the detector is estimated to be 2.5 seconds. Early in the reactor test program it was observed that flow channeling of relatively hot sodium along the walls of the steam generator sodium outlet piping was producing high, erratic 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 location. The thermocouples in-

stalled in each loop carry the designations D, E, and F. The response of the basic thermocouple is about 1.5 seconds.

In setting up the recorder scale factors for the 67-Mwt scram test, the output of the TE-301-2D1 thermocouple was observed to have a high-level, 60-cycle extraneous signal, preventing recording of the temperature information. The difficulty was diagnosed to be a ground fault somewhere in the thermocouple extension leads.

5. Reactor Inlet Temperature

The reactor inlet temperature detector (TE-209) is located in the 6-inch standpipe. The TE-209 temperature detector is a resistance thermometer identical with the TE-201, and also has a time constant of 6.07 seconds.

6. Water-Steam Temperature

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

7. Neutron Flux

The neutron flux detector for measuring and monitoring reactor power was provided as a separate apparatus, independent of the reactor safety system and control instrumentation. The detector utilized was a B-10-lined compensated ion chamber (NE-116) installed in neutron-counter tube No. 5, shown in Figure 4. The output of the ion chamber is read on a Keithley Model 415 electrometer located in the control room. A coaxial cable approximately 250 feet long interconnects the ion chamber and the electrometer. The time constant of the neutron flux measuring instrumentation is of the order of 10 milliseconds.

B. SCRAM EXECUTION

Briefly, the reactor scram circuit functions in the following manner: The SCRAM pushbutton is manually depressed to obtain a normal scram of the reactor. Since contacts of the scram switch are in series with the solid-state relay control circuit and scram coil power unit which supplies 28 volts average d-c to the safety rod holding electromagnet, breaking of the circuit causes the holding electromagnet to be de-energized, thereby releasing the mechanical latch. The spring-loaded rod is then accelerated into the core at an average velocity of about 70 inches per second. This is equivalent to a rate of negative reactivity insertion of about \$17 per

FIG. 4 LOCATION OF NEUTRON - COUNTER TUBES AND SODIUM PIPES IN GRAPHITE SHIELD

second for eight rods dropping simultaneously. The release time from initiation of a scram signal to breakaway of the safety rods has been measured to be approximately 0.045 second.

Any scram signal, whether manually- or safety-system-introduced, energizes the fast drive motors for all the safety rods. These motors drive the safety rod extensions down into the core, requiring approximately 30 seconds to travel from the upper position to a point 6 inches above the lower position. The operating control rods are not released during a scram, but are driven down at the setback rate of about 5.35 inches per minute.

C. SIGNAL CONDITIONING CIRCUITS AND ADJUNCTIVE APPARATUS

Since a signal was not always available in all 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. With some detectors, special care had to be exercised to prevent upsetting the normal operation of the circuit by recorder loading or by accidental short circuit which could impair or otherwise alter the detector output signal. Inasmuch as the output of the TE-201 detectors has an important control function, 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) preamplifier.

Figures 6a, 6b, and 6c are block diagrams of the circuits of the RTD's, and Figure 6d shows the circuit for the TE-301 thermocouples.

Adjunctive apparatus and equipment employed in the scram tests consisted of the following major items:

- Two eight-channel strip chart recorders, having a minimum response of 50 cps and containing suppression and span calibrating provisions, were used for recording system parameters such as temperatures, regulator rod position, and neutron flux.
- A broadband, 4-channel, frequency- or amplitude-modulated signal tape recorder was used for recording the outputs of the TE-201 detectors and read-in of verbal narration.

FIG. 5 BRIDGE CIRCUIT FOR RESISTANCE THERMOMETER DETECTORS

FIG. 6 TEMPERATURE DETECTOR SIGNAL CIRCUITS

- A millivolt box was used for simulating the output of a thermocouple.
- Three four-arm resistance bridge circuits equipped with a battery power supply for excitation of the bridge were used to simulate 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.

IV. EXPERIMENTAL PROCEDURES AND PLANT CONDITIONS

The reactor operating control system provides a choice of manual or automatic control. Manual control is primarily for start-up, test, and emergency use. Reactor start-up from source level up to about 10 megawatts is a manual operation under strict administrative control. Normally, the plant loading programmer controls the plant during run-up in the power range and during power level changes. Four of the five subsystems comprising the feedwater and steam control system also are operable in either manual or automatic control. The four subsystems which have this duality of operation are the feedwater flow control, feedwater pump speed control, steam generator pressure control, and the No. 4 heater pressure control. The fifth subsystem, the emergency feedwater flow control, operates only in the automatic mode.

Reactor control at the test power level in the scram tests was in the manual mode up to the instant of scram. Steam and feedwater controls also were in the manual mode. At 9.5, 13.3, and 67 Mwt, operation was on two coolant loops; whereas, for the 20-Mwt test, operation was on three coolant loops.

The basic steps in executing a reactor scram were as follows: First, the required nominal primary and secondary sodium flow rates were established; second, reactor power was brought to the desired test level in accordance with the prescribed plant operating procedures; third, feedwater flow was adjusted manually to establish a steam generator sodium outlet temperature consistent with the reactor power schedule. No special plant conditions or limitations were imposed to perform the tests. Plant parameters were read on the panel and console instruments and logged. The reactor was manually scrammed. After the transients resulting from the scram diminished to imperceptible levels, the plant parameters were again read and logged. Data recorded on strip chart recorders were as follows:

- Reactor outlet temperature (TE-201)
- Core outlet temperature (TE-115)
- Steam generator sodium inlet temperature (TE-308)
- Steam generator sodium outlet temperature (TE-301)
- Reactor inlet sodium temperature (TE-209)
- Water-steam temperature at steam generator (TE-601)
- Feedwater flow
- Neutron flux (NE-116)
- Regulator rod position
- Reactor outlet temperature (TE-217).

The recorder scale factors used for the recordings are listed in Table I.

Data logged from the panel and console instruments were

Reactor outlet temperatures (TE-201-1, 2, 3)
Reactor inlet temperatures (TE-209-1, 2, 3)
Steam generator sodium inlet temperatures (TE-308-1, 2, 3)
Steam generator sodium outlet temperature (TE-301-1, 2, 3)
Feedwater temperature (E-50)
Steam temperature (TE-601-1C, 2C, 3C)
Primary sodium flows (FR-201-1, 2, 3 and FR-202-1, 2, 3)
Secondary sodium flows (FR-301-1, 2, 3)
Feedwater flows (FR-601-1, 2, 3)
Steam pressure (D-74)
Reactor power (NR-108)

The data obtained are on file in the reactor operations office at the Enrico Fermi plant.

TABLE I - SCALE FACTORS USED FOR SCRAM TEST RECORDINGS

Parameter	Recorder Scale Range *			
	9.5 Mwt	13.3 Mwt	20 Mwt	67 Mwt
Temperature Recording, °F				
TE-115	500-600	500-600	500-600	500-800
TE-201	500-550	500-550	500-550	500-700
TE-209	500-550	500-550	500-550	500-550
TE-217	-	500-550	500-550	500-700
TE-301	475-550	475-550	500-550	500-550
TE-308	500-550	500-550	500-550	500-700
TE-601	500-550	100-600	500-550	500-700
Steam Pressure, psig	-	-	-	500-1000
Feedwater Flow, lb/hr	-	-	-	$0-2 \times 10^5$
Neutron Flux, amps	$0-1 \times 10^{-5}$	$0-1 \times 10^{-5}$	$0-1 \times 10^{-5}$	$0-3 \times 10^{-5}$
<hr/>				
* Chart Width, cm	5	5	5	5
Chart Speed, sec/cm	10	4	10	10

V. EXPERIMENTAL RESULTS AND ANALYSIS

A. RESULTS

The plant parameters which were recorded on strip chart recorders to obtain transient and steady-state data are listed in Section IV. Parameter responses after the scram were observed for seven transient characteristics: transport delay, time constant, maximum rate of change, duration of the maximum rate of change, overshoot, oscillation, and settling time. Among these characteristics, the most important from the standpoint of plant safety and longevity are the maximum rate of change and its duration. Plant design criteria contained in Volume 1 of Reference 1 stipulates maximum values of parameter transients for various regions of the reactor system. One of the principal parameters is the temperature in the outlet nozzle region of the reactor, which is read by TE-201 detectors. Of major concern are the rate of change and duration of transients for the two-loop condition as compared to the three-loop condition at power levels of 67 Mwt and higher. At relatively low reactor power levels, however, such as 20 Mwt and less, thermal transient data are relatively insignificant because of the low heat levels involved. The tests at the low reactor power levels have their usefulness mainly in checking out the start-up provisions in the reactor and feedwater control systems and the plant interlock systems.

B. ANALYSIS

1. Transport Delay

Transport delay is the time lapse between the instant of initiation of the test and beginning of the transient response of the parameter observed. In reading the charts, the instant of initiation of the test is taken as occurring simultaneously with the beginning of decay of the neutron flux density signal following scram of the reactor. In Table II are listed the transport delays of the principal plant parameters as read from the strip chart recordings; the delays are illustrated in a time plot in Figure 7, where the transport delays of the steam generator sodium inlet temperature (TE-308), reactor inlet sodium temperature (TE-209), and steam temperature at the steam generator (TE-601), are seen to be somewhat shorter for the 67-Mwt case than for the 20-Mwt case, despite the fact that operation at 67 Mwt was on only two coolant loops, whereas in the latter case operation was on three coolant loops. The shorter transport time results from a slightly higher coolant flow rate at 67 Mwt.

The sodium flow rates in the primary and secondary coolant systems during the scram tests are given in Table III. The negative flow

TABLE II - TRANSPORT TIME DELAYS OF TEMPERATURE TRANSIENTS RELATIVE TO THE INSTANT OF NEUTRON FLUX DECAY *

Parameter	Time Delay, sec			
	9.5 Mwt	13.3 Mwt	20 Mwt**	67 Mwt
TE-115	1.5	1	1	1.5
TE-201	14	13	10	11
TE-209	106	100	97	70
TE-308	76	72	72	62
TE-301	116	112	112	***
TE-601	151	136	132	91

* See Figure 7.

** Three-loop operation; two-loop operation in other tests.

*** Unobtainable because of parameter instability.

rate noted in some cases is due to the back pressure of the core inlet plenum (high pressure) and is indicated for the 14-inch line in each case where the loop is not operative during the test. The relatively high forward flow in the 6-inch line exists due to the pressure in the 14-inch line. In the secondary system, where the three coolant loops are isolated from each other, no flows are indicated for those loops shut down because for all practical purposes, the flow is zero with the exception of convective circulation.

2. Time Constants

The time constants of principal plant parameters obtained during the 9.5-, 13.3-, 20-, and 67-Mwt scram tests are presented in Table IV. The value given for each parameter is that of the major time constant. Time constants of higher order are also present, but their effects on the overall response are small by comparison with the effect of the major time constant. For example, the recorded TE-201 response includes the lag effects of the 6.07-second time constant of the resistance thermometer. Also present in the recorded TE-201 response, though of lesser significance than the effect of the resistance thermometer, are the effects of the time constants of the signal amplifier and the strip chart recorder.

The time constant of TE-601 in the 67-Mwt case was influenced by manual adjustments of the feedwater flow rate; hence, the rather substantial differences between the value at 67 Mwt and the values obtained at the lower power levels are attributed largely to the manual flow adjustments which were made following the scram.

TABLE III - SODIUM FLOW RATES IN THE PRIMARY AND SECONDARY
HEAT TRANSPORT SYSTEMS DURING SCRAM TESTS

Reactor Power, Mwt	Primary System Sodium Flow, 10^6 lb/hr					
	Loop No. 1		Loop No. 2		Loop No. 3	
	<u>14-Inch Line</u>	<u>6-Inch Line</u>	<u>14-Inch Line</u>	<u>6-Inch Line</u>	<u>14-Inch Line</u>	<u>6-Inch Line</u>
9.5	2.93	0.288	-0.440	0.240	2.44	0.288
13.3	2.99	0.300	-0.330	0.255	2.70	0.300
20	2.64	0.338	2.60	0.420	2.50	0.400
67	2.93	0.310	3.13	0.305	-0.450	0.287

Reactor Power, Mwt	Secondary System Sodium Flow, 10^6 lb/hr		
	<u>Loop No. 1</u>	<u>Loop No. 2</u>	<u>Loop No. 3</u>
9.5	2.95	-	3.10
13.3	2.98	-	2.90
20	2.91	2.95	2.92
67	2.88	3.18	-

FIG. 7 TRANSPORT TIME DELAYS AS DETERMINED FROM SCRAM TRANSIENT TESTS

TABLE IV - MAJOR TIME CONSTANT OF PRINCIPAL
TEMPERATURE PARAMETERS

Parameter	Time Constant, sec			
	9.5 Mwt	13.3 Mwt	20 Mwt [*]	67 Mwt
TE-115	1.5	1.5	1.5	1.5
TE-201	57	47	30	48
TE-209	58	67.5	77	87.5
TE-308	87	75	84	75
TE-301-2D1	63	90	87	**
TE-601	95	92	88	110

* Three-loop operation; operation on 2 loops in other tests.

** Unobtainable because of parameter instability.

A graphical method was employed in obtaining the time constants in Table IV. Basically, the procedure consists of plotting on semilog graph paper the percentage of incompleteness of the transient and determining from the plot the time required for the incompleteness to diminish to 36.8 per cent of its initial value. This technique is commonly employed in the analysis of process control systems.

Illustrated in Fig. 8 is the plot from which was obtained the major time constant of the TE-201-1 response for the 67-Mwt scram. The technique proceeds as follows: First, the percentage of incompleteness of the transient response is plotted on semilog graph paper, with incompleteness on the log axis and time on the linear axis. Second, the linear portion of the curve, designated A in Fig. 8, is extended to obtain an intercept point on the per cent incompleteness axis. The percentage at this point is seen to be about 117. Taking 36.8 per cent of this value, 43 per cent is obtained. The major time constant is read as the time to the 43 per cent point on the curve which was plotted for the percentage incompleteness; a figure of 48 seconds is thus obtained.

To obtain the first minor time constant, the percentage of difference between curve A and its linear extension to the vertical intercept is plotted. The maximum value of the difference obtained is about 20 per cent. A curve of the difference is plotted and designated as curve B on Fig. 8. Taking 36.8 per cent of the intercept of curve B, a value of approximately 10 seconds is obtained for the first minor time constant. Considering reading limitations of the charts and introduction of unavoidable small inaccuracies in the data reduction process, the value of 10 seconds compares favorably with the 6.07-second figure predicted in Reference 2 for the TE-201 resistance thermometer. A similar procedure was used to obtain all of the time constants given in Table IV.

FIG. 8 DETERMINATION OF TIME CONSTANTS FOR TE 201-1
RESPONSE IN SCRAM TEST AT 67 Mwt

3. Thermal Transients

Because of the low heat levels involved, thermal transients at reactor power levels of the order of 20 Mwt and below are essentially insignificant. The 67-Mwt test is of main interest in determining maximum thermal transients from the scram test recordings.

Recordings of reactor outlet sodium temperature (TE-201) obtained in the 67-Mwt scram with operation on two coolant loops have shown temperature fluctuations which were nonperiodic in nature and of significant amplitude and rate of change. Evaluation of this data recorded on magnetic tape was performed on an analog computer. It was found that transients as high as 44°F per second had occurred in loop 1 during a temperature fluctuation of 76°F , while in loop 2, a transient of 20°F per second occurred during a temperature fluctuation of 48°F . These transients were the predominant ones among the temperature fluctuations which were present in the main decay transients of the TE-201 temperatures. In general, the fluctuations were nonperiodic in nature and had time intervals between peaks ranging from approximately one to five seconds. The maximum design transient for TE-201 and for other key parameters are given in Table V, which was obtained from Reference 1, Appendix A.

The severity of temperature fluctuations and transients which have been encountered is largely the result of 2-loop operation, where the rate and homogeneity of coolant mixing is poorer than that which would be obtained in operation on three loops. It should be mentioned that the 30-inch reactor sodium outlet nozzle nearest the core is that of loop No. 2; whereas the nozzles for loops 1 and 3, equidistant from the core, are more distant therefrom (see Figure 4).

The TE-217-1 detector, which measures reactor outlet temperature, produced a signal which contained fluctuations much more prominent than those of the TE-201 outputs. Because of the relatively low time constant of the TE-217 detector, its response is much faster than that of TE-201. As described under Section III, the TE-217 detectors are sheathed thermocouples having a response of the order of one second for the basic, unmounted unit; however, usable output was not obtained from detector TE-217-2 because of an apparent condition of poor thermal contact between the thermocouple and the thermowell. Analog computer analysis of the signal from TE-217-1, taking into account appropriate coefficients to correct for the lower time constant of the TE-217 detector, provided transient figures of essentially the same magnitude as obtained from the TE-201-1 data. Because of the physical separation of the two detectors, as well as the random nature of the sodium mixing action in the upper reactor plenum, the transient waveforms which were recorded were substantially dissimilar.

TABLE V - MAXIMUM DESIGN TRANSIENTS

<u>Sensor</u>	<u>Location</u>	<u>Maximum Design Transient</u>			<u>Ref. I Table No.</u>
		<u>°F/sec</u>	<u>ΔT, °F</u>	<u>Cycles</u>	
TE-201	Reactor Outlet	-15.8	300	500	102.2
TE-209	Reactor Inlet	-21.4	150	5	102.2
TE-308	Steam Generator Sodium Inlet	40	350	5*	106.9
TE-301	IHX Secondary Sodium Inlet	30	350	5*	106.5
TE-601	Steam Generator Steam Outlet	40	350	5*	106.9

* The minimum number of cycles used in design was 1000 cycles — in most cases the equipment is capable of withstanding cycles up to the endurance limit.

Responses of other temperature parameters in the 67-Mwt test were analyzed using Equation (1) in Reference 3 to determine representative values for the maximum thermal transients. The expression is

$$T_s = T_t + (\tau_1 + \tau_c) d T_t/dt + \tau_1 \tau_c d^2 T_t/dt^2$$

where

T_s = actual sodium temperature, °F
 T_t = measured temperature, °F
 τ_1 = time constant of the resistance temperature detector, sec
 τ_c = time constant of the instrumentation circuit, sec
 t = time, sec

The transient values calculated for TE-115, TE-209, TE-301, TE-308, and TE-601 are given in Table VI along with the values which were obtained for TE-201-1 and TE-201-2 by computer analysis. Also given in Table VI are the detector time constant value τ_1 used in making the computations. Inasmuch as the time constant of the instrumentation equipment τ_c is small in comparison with the detector time constants, the value of τ_c was taken as zero in the calculations.

It will be noted that plus and minus values are indicated for the magnitude of change and the rate of change of the TE-201 and TE-301 parameters. The notation is used where the recorded trace was fluctuating.

Except for the values obtained for TE-201-1 and TE-201-2, the maximum transients observed in the 67-Mwt scram test are well below the stipulated maximum value. Although the rates of change of TE-201-2 and TE-201-1 are significantly in excess of the -15.8°F/sec maximum value stated, the transients are well within the 300°F figure given for the maximum magnitude of the change for TE-201.

TABLE VI - MAXIMUM THERMAL TRANSIENTS DERIVED
FROM 67-Mwt SCRAM TEST RECORDINGS

<u>Parameter</u>	<u>Detector Time Constant, sec</u>	<u>Magnitude of Change, °F</u>	<u>Rate of Change, °F/sec</u>
TE-201-1	6.07	+76	+44
TE-201-2	6.07	+48	+20
TE-209-2	6.07	-10.2	- 0.2
TE-301-2D1	1.2	+ 4.4	+ 1.75
TE-308-2	0.25	-68	- 0.85
TE-601-2	25	-42	- 0.56

Note: 1. A + before the number indicates that the recorded parameter was fluctuating.

2. Operation was on 2 coolant loops during the 67-Mwt scram test.

4. Plant Parameter Values

Listed in Table VII are the values of the principal plant parameters before and after the scram as read from the strip chart recordings. Performance of the steam system at the power levels of 20 megawatts and below was stable and of little interest for transient studies because of the low power levels involved. Consequently, recordings of steam pressure and feedwater flow were taken only during the 67-Mwt scram. The before-scram values given in Table VII were obtained from the strip chart recording traces with readings taken at a point only a few seconds from the apparent starting point of the scram-produced change. The after-scram values were read at the first region where a change in the parameter was no longer evident. This time interval, which will be referred to herein as the settling of level-off time, is in most cases within 1000 seconds, or approximately 15 minutes, for the parameters recorded.

5. Parameter Response Characteristics

To provide information on the nature of the parameter responses without reproducing the actual recorded traces, a brief descriptive commentary on the transient responses obtained is presented below. The transport delay and major time constant of the principal plant parameters were given in Tables II and IV, respectively.

a. Core Outlet Temperature

As expected, the decay of the recorded core outlet temperature (TE-115) was monotonic in all of the scram tests. Two values of settling time were observed: approximately 30 seconds in the 67- and 13.3-Mwt responses and approximately 20 seconds in the 20- and 9.5-Mwt responses.

b. Reactor Sodium Outlet Temperature

Recordings of the reactor sodium outlet temperature exhibited random fluctuations of high amplitude in both the TE-201 and the TE-217 detector outputs in the 67-megawatt (2-loop) test. The beginning of the fluctuations coincided essentially with the beginning of the decay of the recorded temperatures. The fluctuations became most pronounced about 30 seconds after the time of initial appearance and thereafter decayed so as to become indiscernible at 175 seconds in the case of TE-201, and at about 205 seconds in the case of TE-217. The settling time of the TE-201 and TE-217 responses was about 600 seconds.

In the 20-Mwt test the TE-201 and TE-217 decays, as recorded, were essentially monotonic, with only a slightly discernible trace of fluctuation. The settling times of the TE-201 and the TE-217 recordings in the 20-Mwt test were approximately 590 seconds.

**TABLE VII - VALUES OF PLANT PARAMETERS BEFORE
AND AFTER SCRAM OF THE REACTOR**

	Parameter Value	
	Before Scram	After Scram
67 Mwt 2-Loop Operation		
Temperature Recordings, °F		
TE-115-9	746	524
TE-201-1	666	526
TE-201-2	668	528
TE-209-2	542	518
TE-217-1	652	522
TE-217-2	652	540
TE-301-2D1	520	525
TE-308-2	586	512
TE-601-2	610	516
Steam Pressure Loop No. 2, psi	900	875
Feedwater Flow Loop No. 2, lb/hr	1.1×10^5	0
Neutron Flux (Electrometer), amps	2.43×10^{-5}	0
20 Mwt 3-Loop Operation		
Temperature Recordings, °F		
TE-115-9	552	500
TE-201-1	532	< 500
TE-201-2	537	500
TE-201-3	540	506
TE-209-3	533	518
TE-217-2	*	*
TE-217-3	535	500
TE-301-3C	523	< 500
TE-308-3	528.5	500
TE-601-2	508	520
Neutron Flux (Electrometer), amps	0.67×10^{-5}	0
13.3 Mwt 2-Loop Operation		
Temperature Recordings, °F		
TE-115-9	529	506
TE-201-1	536	510
TE-201-3	540	512
TE-209-3	523	517.5
TE-217-1	539	512
TE-217-3	543	513
TE-301-3C	529	510
TE-308-3	528	508
TE-601-3	525	430
Neutron Flux (Electrometer), amps	0.45×10^{-5}	0
9.5 Mwt 2-Loop Operation		
Temperature Recordings, °F		
TE-115-9	542	~ 510
TE-201-1	533	~ 512
TE-201-3	537	~ 517
TE-217-1	**	**
TE-217-3	**	**
TE-209-3	526	~ 521
TE-301-1C	541	~ 533
TE-301-3C	540	~ 533
TE-308-3	523	510
TE-601-3	630	~ 523
Neutron Flux (Electrometer), amps	0.31×10^{-5}	0

* Detector not functioning

** Not recorded

Recordings of TE-201 and TE-217 in the 13.3-Mwt test both show random fluctuations, becoming most pronounced about 30 seconds after beginning of the temperature decay and diminishing to insignificance at 75 seconds in the case of TE-201, and about 125 seconds in the case of TE-217 responses.

In the 9.5-Mwt test the TE-201-1 decay shows a perceptible degree of fluctuation, becoming most pronounced at about 30 seconds after start of the decay, whereas the TE-201-3 decay is relatively free of fluctuations. TE-217 responses were not recorded during the 9.5-Mwt test as the detectors were not yet connected for use. The fluctuations in the TE-201-1 trace diminish to an imperceptible level in about 75 seconds after start of the decay. The settling times of the TE-201 responses were somewhere beyond 500 seconds, which was the duration of the recordings taken.

c. Reactor Inlet Temperature

The decay of the reactor inlet temperature (TE-209) in the 67-Mwt test exhibited an undershoot to 514°F at about 470 seconds after start of the decay, followed by a slight upswing and downswing, with level-off at approximately 900 seconds.

In the 20-Mwt test, the decay of TE-209 was essentially monotonic, with level-off somewhere beyond 400 seconds — the limit of the recording obtained.

In the 13.3-Mwt test, the decay of TE-209 was perceptibly oscillatory, with about two oscillation cycles occurring before level-off in about 850 seconds after start of the decay.

In the 9.5-Mwt test, decay of TE-209 was perceptibly oscillatory, with one cycle of oscillation taking place well before level-off, the latter occurring somewhere beyond 500 seconds, the limit of the recording.

d. Steam Generator Sodium Inlet Temperature

Decay of the steam generator sodium inlet temperature (TE-308) was monotonic in the scram tests at all power levels. The level-off times in the 67-, 20-, 13.3- and 9.5-Mwt tests were 545, 650, 650, and beyond 500 seconds (limit of the recording), respectively.

e. Steam Generator Sodium Outlet Temperature

In the 67-Mwt test the steam generator sodium temperature recording contained a good deal of "hash", or relatively high frequency fluctuation, before the scram. The peak-to-peak level of the hash as

evident on the chart was about 5°F maximum. Some long period variation in the TE-301 recording was evident up to as well as beyond the instant of reactor scram. The parameter was unstable, precluding analysis for transport lag and time constant. A sharp drop from approximately 520 to 515°F occurred 340 seconds after the scram, followed by an upswing beginning at 495 seconds, then one slight oscillation, with level-off in about 875 seconds.

Decay of TE-301 in the 20-, 13.3- and 9.5-Mwt tests was monotonic, with the traces containing essentially no evidence of hash or high frequency fluctuations as were present in the recording during the 67-Mwt test. Level-off occurred at approximately 600 and 500 seconds in the 20- and the 13.3-Mwt tests, respectively, and after 500 seconds (duration of recording) in the 9.5-Mwt test.

f. Water-Steam Temperature

Decay of the water-steam temperature (TE-601) was monotonic at all test power levels, with no significant fluctuations apparent. Level-off times were 600, 560, 500 seconds in the 67-, 20-, and 13.3-Mwt tests, respectively, and approximately 450 seconds in the 9.5-Mwt test.

g. Feedwater Flow

This parameter was recorded only in the 67-Mwt test, with flow control being in the manual mode. The response characteristic as recorded exhibited cutback in flow in six essentially distinct steps, i.e., as opposed to a smooth curve, from the initial value of 1.1×10^5 ; 0.84×10^5 ; 0.54×10^5 ; 0.41×10^5 ; 0.34×10^5 and zero lb/hour/loop; the time at which these values were first reached following the scram were 80, 200, 385, 425, 430 and 435 seconds, respectively.

h. Regulator Rod Position

In all of the scram tests, run-down of the regulator rod performed satisfactorily. Run-down velocity was 5.35 inches per minute. The time lag between the instant of scram as defined by the neutron flux decay signal and the start of the run-down was less than one second, about the limit of time resolution on the strip chart recordings.

i. Neutron Flux

In all the scram tests the recorded trace of the neutron flux parameter exhibited essentially a step change with only a small degree of round-off, due to delayed neutrons.

VI. CONCLUSIONS

The scram tests which have been conducted at 67-, 20-, 13.3-, and 9.5-Mwt have demonstrated satisfactory performance of the reactor control system, interlock circuits, and the safety rod raise interlock. Thermal rates of change in the reactor outlet temperature (TE-201) were found to be in excess of the maximum design value in the 67-Mwt test which was conducted with operation on two coolant loops; however, the magnitude of temperature change over which these rates of change occurred was comparatively small and well within the design maximum. It is to be expected that operation on three coolant loops will significantly reduce the rate of change value of the TE-201 transients. The magnitudes of the transients in parameters other than TE-201 have been found substantially below the design maximum values.

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