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PLANT INHERENT CONTROL TESTING IN EBR-II

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ABSTRACT

Recent tests in the Experimental Breeder Reactor II (EBR-II) have demonstrated that reactor feedbacks can passively reduce power and thus effectively mitigate reactor undercooling caused by equipment failures. A follow-on testing program is being designed to investigate the use of these feedbacks along other liquid metal reactor (LMR) characteristics to routinely control reactor power during plant maneuvers and fuel burnup. compensation, and to limit the possibility and consequences of over-power accidents. In all of the tests described in the present paper, the control rods will not be used as the plant is maneuvered over the power range between 40 and 100%. The plant variables (forcing functions) employed in the power control include the primary flow, the secondary flow, and the turbine admission position. The pretest predictions for the tests are presented and a preliminary analysis on the effects of controller failures is discussed. This paper provides concepts in reactor power control which may lead to fundamental changes in design and safety consideration of metal fueled LMRs.

INTRODUCTION

On April 3, 1986 an international audience of liquid metal reactor (LMR) experts were present to witness two very significant tests which were performed on the Experimental Breeder Reactor II (EBR-II) plant. For the first test, the reactor was initially at its full 60 MW power level when the two main coolant pumps were tripped. Prior to the test the reactor safety system was configured so that the loss-of-flow transient which ensued did not allow the reactor to automatically shut down (scram). The reactor core temperatures were observed to rise substantially, reach a peak in about 90 seconds, and then recede to much lower values as the fission power monotonically declined toward zero. What had been demonstrated was that the reactivity

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feedbacks in the metal-fueled EBR-II were very passively effectively so that they alone, without the scram system, can shut the reactor down during a loss-of-flow accident.

The the second test was conducted from full power, again without activating the scram system.[2] This time the secondary pump was tripped, and the intermediate-loop sodium flow was essentially stopped within about three minutes. A loss of sink without scram was thus accomplished since heat transfer from the primary system to the steam system which generates electricity was stopped. As the reactor inlet temperature rose, the reactivity feedback caused the reactor power and outlet temperatures to drop. There was no observed overshoot in the core outlet temperatures and the fission power approached zero as the reactor approached a quenching temperature of only 45°C above the initial reactor inlet temperature.

These tests validate U.S. efforts to design a LMR which would optimize the use of passive features to accomplish the safety functions. Future work is required to demonstrate the ability to passively limit the effects of transient over-power accidents, especially those due to rod withdrawal. One approach is to utilize the feedback characteristics in the metal fuel to minimize the need of control rods for power control and burnup compensation. The obvious solution is to minimize the amount of reactivity available in control rods so that they can not cause the power to rise to an unsafe level.

The above tests demonstrated the concept of using the reactivity feedbacks to control the reactor power. Furthermore, a series of primary flow and reactor inlet temperature perturbation tests had previously been conducted in EBR-II to study the reactivity feedback characteristics of the plant[3,4]. Data obtained from those tests along with analytical predictions also suggested that the control rods need not be used for normal reactor power control over a wide power range. If the reactor power can be passively controlled without using control rods,then the reactivity addition accident due to control rod runout can also be either eliminated or greatly reduced in severity. The potential fuel element damage associated with loss-of-flow, loss-of-heat-sink and control rods runout accidents can thus be presented by inherently safe characteristics of the plant rather than the complicated reactor safety shutdown (RSS) system, and it can be considered as the first line of safety defense. The RSS system can then be considered as second line of safety defense. As a result, a number of redundant scram functions in the RSS system of a liquid metal reactor can be significantly reduced, and the RSS system can become simpler and less costly.

The purpose of the present plant inherent control testing program is to investigate and verify the passive control concepts explored in the previous reactor tests described above. The experiments considered in this paper are the of three groups of tests which attempt to verify steady-state profiles over a power range of 40 to 100%. The power in this group will be controlled by primary flow, reactor inlet temperature,and turbine inlet pressure. These variable in turn, are regulated by the primary pump speed, the secondary flow, and the turbine admission valve position. Once the initial conditions of the test are set, the control rod drive mechanism will be deenergized, and the control rods will not be used in manouvering the plant to new steady state conditions. Information obtained from the present tests will provide important plant and component data that will be used to designed a reliable safe control system and to develop the continuing test programs. Plans for the second test group include testing alternative control systems dynamically. Finally the capabilities of the plant to safely tolerate a wide range of faults and equipment failures will be tested in the third group. In this test window, we plan to inject faults into the system and test the fault tolerance of the final proposed control scheme. We thus plan to demonstrate the reliability and safety of innovative control scheme for future LMRs.

EBR-II PLANT DESCRIPTION AND CONTROL

The EBR-II is a sodium-cooled,pool-type fast reactor designed to operate at a thermal power level of 62.5 MWt and a net electrical power output of about 20 MWe. Rated sodium flows of the primary and the secondary systems are 485 and 320 kg/s, respectively. The plant is located in Idaho and is operated by Argonne National Laboratory for the U.S.Department of Energy. EBR-II has been in operation since 1964 and has served primarily as a fast-flux irradiation facility. Recently it also served as a safety testing facility.

The primary system of EBR-II includes a double-walled tank,as shown in Fig. 1. The tank contains about 340 m^3 of sodium at 371°C under normal operating condition. Since the reactor is basically of the pool design, essentially all primary components are submerged in a large volume of sodium within the primary tank. The primary coolant is well-mixed in the common outlet plenum of the reactor vessel which is connected to the reactor outlet pipe that transports the coolant to the intermediate heat exchanger (IHX). Flow exiting from the IHX mixes with the bulk sodium in the primary tank before it enters the two primary pumps on its way to the reactor inlet pipes and into the reactor vessel.

The reactor consists of 16 rows of hexagonal subassemblies, which form the core region (rows 1-6), reflector region (rows 7-10), and blanket region (rows 11-16). The typical EBR-II driver subassembly contains 91 pins of sodium-bonded metallic fuel. About 84% of the flow enters high pressure plenum to cool the core, and the remaining 16% enters to the low pressure plenum to cool the blanket and reflector subassemblies.

A schematic of the entire EBR-II plant and controllers for plant operation is given in Fig. 2, in which C denotes controller, T denotes temperature, F denotes flow, P denotes pressure and POW denotes power. The IHX transfers heat generated in the reactor to the secondary sodium system and ultimately to the steam generator and the turbine generator. The secondary sodium, driven by an electromagnetic pump, return to the IHX after passing through the secondary piping system, two parallel superheaters and seven parallel evaporators. During normal steady state operation, the reactor power is regulated by control rods (C_1 in Fig. 2); primary flow (F_1) is controlled by pump speed to a desired flow rate (C_2); secondary flow (F_2) is controlled to maintain a constant reactor temperature (T_1) by a secondary flow/tank temperature controller (C_3); and the steam header pressure (P_1) is controlled by the throttle valve (C_4) using an Initial Pressure Regulator (IPR) and/or by the steam bypass valve (C_5). Of the 13 controllers are currently available for plant operation, five of them are shown in Fig.2. In the present test series, controller C_1 will be disabled, and a new secondary flow/steam pressure will be installed to regulate the steam pressure via secondary flow. In addition, a new controller will be installed to regulate generator output power via turbine inlet valve position.

TEST DESCRIPTION AND ANALYSIS

For nominal reactor operation, the reactor outlet and the reactor inlet temperatures are 473 and 371°C, respectively. In all of the tests, the reactor outlet temperature is to remain within the nominal envelope and the reactor inlet temperature varies from 360 to 385°C. The turbine will be on line generating electricity and the reactor power will be controlled between 40 and 100%. The control rod drive mechanism will be deenergized during the test, so that inherent reactivity feedbacks are the only mechanism to control the reactor power.

The computer program NATDEMO was used to predict plant responses for the tests. NATDEMO is a thermal-hydraulic system analysis codes, which was specifically designed for the EBR-II plant. A description of NATDEMO is presented in Ref. 4. The NATDEMO code models all 16 rows of subassemblies by dividing them into three basic regions; the driver fuel, the stainless steel reflector, and the depleted uranium blanket regions (channels). In the model, each of these regions has a separate power generation function containing both prompt fission and delayed power components, and each has separated thermal-hydraulics while all three as treated as parallel channels. One of the important features of the code is its detailed modeling of the reactivity feedbacks, which is essential for the present tests. A description of the NATDEMO reactivity feedback model was given in Ref. 3. In the present analysis, a thermal bowing reactivity component of +0.025$ is assumed while all other components are negative.

Based on the control functions employed in each test, the present test series can be represented by four individual tests, referred as tests No. 1 to 4. Test descriptions and pretest predictions of these tests are given below:

1. Test No.1 - The purpose of the test is to investigate the power controllability using primary flow. The test will be initiated by reducing the primary flow from an initial flow of

115% to a final flow of 40% at 1% per minute in three steps, and then increase the flow to 115% in the similar manner. The intermediate steps are 60 and 80% flow conditions. The reactor inlet temperature will be maintained constant at 371°C throughout the test by adjusting the secondary flow.

During the test the control rods (C_1 in Fig.2) will be deenergized. The primary flow will be controlled with the primary pumps (C_2), the secondary flow controller will be operating in its automatic mode to vary flow and thereby maintain a constant reactor inlet temperature (T_1).The throttle valve IPR (C_4) and the pressure regulating valve controller (C_5) will be operating in their normal lineup to maintain a constant steam header pressure (P_1). When the primary flow is reduced, the reactor temperature temporarily rises due to reduced of coolant flow, and the reactor power decreases as a result of negative reactivity feedback. Because the transient is slow, transient effects are small and the reactor temperature rise remains almost proportional to the power/flow ratio (P/F). As both the power and the flow are reduced, the secondary flow will be decreased automatically to maintain a constant reactor inlet temperature by a secondary flow/tank temperature controller (C_3). The normalized power and primary flow responses for the test are given in Fig. 3, in which 100% power denotes 60 MWt reactor power, 100% flow is the reactor flow that produces a reactor temperature rise of 84 °C. The secondary flow and reactor outlet temperature responses are illustrated in Figs. 4 and 5, respectively, in which 100% secondary flow denotes the flow required to maintain a 371°C primary-tank sodium temperature at a 100% power and flow condition. The predictions indicate that the reactor power follows the flow closely because at least 90 of the EBR-II reactivity feedback is proportional to P/F. It is noted that during the flow change, the steady-state P/F varies from an initial value of 0.87 at 115% flow to about 0.96 at 40% condition because of the portion of the feedback proportional to power rather than P/F. This load-following behavior can be further explained by considering changes in reactivity ($\delta\phi$), power (δP), power/flow ratio($\delta(P/F)$), and reactor inlet temperature (δT).[5] A quasi-static approximation for the reactivity perturbation can be expressed as:

$$\delta\phi = A\delta P + B\delta(P/F) + C\delta T_1,$$

in which A is the power coefficient including reactivity feedbacks proportional to power change, B is the coefficient including the reactivity feedbacks proportional to the P/F change, and C is the coefficient including reactivity feedbacks related to reactor inlet temperature variation. A detailed description of A, B, and C are given in Ref. 3 and 5.

For test No. 1, the reactor inlet temperature will be held constant throughout the test, i.e.,$\delta T_1 = 0$. Since the control rods will be locked during the test, the net reactivity change from one steady state to another steady state should be zero ($\delta\phi = 0$). By substituting $\delta T_1 = 0$ and $\delta\phi = 0$ into Equ. (1), the relationship of P/F between two equilibrium states can be obtained, i.e.

$$\frac{(P/F)_2}{(P/F)_1} = \frac{1 + (A/B)F_1}{1 + (A/B)F_2}, \qquad (2)$$

where subscripts 1 and 2 denote the steady-state conditions 1 and 2, respectively. The A/B in EBR-II is estimated to be between 0.1 and 0.25 depending on reactor P/F and the bowing components. These values are consistent with measurements made in primary flow perturbation tests, and are also consistent with the present prediction.

2. Test No.2 - This test investigates the power controllability that can be obtained by varying the reactor inlet temperature via secondary flow. The controller lineups for this test include deenergized the control rods (C_1), manual control of the primary pump speed (C_2) for a constant flow (F_1), automatic control of the secondary flow (C_3) to change the specified tank temperature (T_1), and automatic control of turbine throttle valve with IPR (C_4) to maintain constant steam header pressure. When the primary flow is held a constant, the reactor inlet temperature increases with the decrease of secondary flow. The test will be initiated from 100% power and flow condition with a primary-tank sodium temperature of 360°C and the power will be controlled from 100 to 40% in three steps. The test will be conducted by increasing primary tank sodium temperature setpoints. Higher reactor inlet temperature will cause reactor power to decrease due to negative reactivity feedback. NATDEMO analysis was performed to determine the required reactor inlet temperatures corresponding to 80, 60 and 40% reactor power conditions. The normalized power and secondary flow responses of the test are illustrated in Fig. 6, while the reactor inlet and outlet temperature variations are given in Fig 7. In the present test, the primary flow remains 100% throughout the test, i.e., $F_1 = F_2 = 1$ and $\delta\phi = 0$ because there will be no control rod movement. When these relationships are substituted into Equ. (1), we find

$$P_2 - P_1 = C \times (T_{11} - T_{12}) / (A + B), \qquad (3)$$

where (A+B) is the power reactivity decrement (PRD), i.e., the reactivity addition required to raise power from zero power hot critical to 100% power at 100% flow. The PRD in EBR-II is 0.275\$ for the loading condition considered in the analysis, and C is about 0.007\$ per °C based on data gathered from reactor inlet temperature perturbation tests and loss-of-heat-sink without scram tests in EBR-II.

3. Test No.3 - In this test the generator power is controlled automatically by turbine throttle valve . In addition, the controller lineups include deenergizing the control rods (C_1), controlling the primary flow to a constant value

(C_8) and regulating secondary flow (C_3) to maintain constant turbine header pressure (P_1). As the throttle valve opens, the steam flow increases and the steam pressure decreases. In order to maintain the initial steam pressure, more secondary flow is required. The increase of secondary flow causes the reactor inlet temperature to drop, and consequently raise the reactor power due to negative reactivity feedback as derived in Equ. (3). To automatically regulate the steam pressure using secondary flow, the secondary flow/turbine pressure controller will be used. Thus, the secondary flow/tank temperature controller will be deenergized and the reactor inlet temperature will not be controlled. The initial condition of this test will be the final conditions of test No,2 where the flow is at 100%, the power is about 40%, and the reactor inlet temperature is about 384°C. The test will be initiated in three steps by opening the turbine throttle valve corresponding to 60, 80 and 100% reactor power. The predicted power, secondary flow and temperature responses of the test are illustrated in Figs.8 and 9. These responses are approximately mirror images of those in test No.2.

4. Test No. 4 - In this test the generator power output is determined by turbine pressure which in turn is controlled by secondary flow. The purpose of the test is to study the turbine and steam plant performances as a function of steam pressure and reactor power level. For this test the control rods (C_1) will be deenergized, the primary flow will be controlled to a constant value (C_2, P_1), the secondary flow will be controlled to vary steam header pressure over a prescribed profile (C_3, P_1), and the turbine throttle valve and the pressure regulating valve (C_4, C_8) will be controlled at fixed positions. The test will be initiated from nominal reactor operating condition except that the reactor inlet temperature will be 360°C instead of the nominal 371°C. The test will be initiated by reducing the turbine pressure setting of 1 MPa (150 psi) per step, in which three steps are required for the desired pressure range. When the turbine pressure setpoint is reduced, the secondary flow will be decreased by the secondary flow/turbine pressure controller to satisfy the pressure demand. This will cause a net increase of reactor inlet temperature and a decrease in reactor power. On the other hand, when the turbine pressure alone is reduced, the steam saturation temperature will drop and cause a decrease in IHX secondary sodium inlet temperature. This effect alone would cause IHX primary coolant outlet temperature to decrease and consequently increase the reactor power. The reduction of turbine pressure thus has two opposing effects on the reactor power when the C_3 controller is used. The secondary flow reduction causes the reactor inlet temperature to increase and the pressure decrease tends to reduce the reactor inlet temperature. The net effect will be a decrease in reactor power. The normalized power, secondary flow and steam pressure responses of the test are given in Fig. 10, in which a 100% pressure denotes 8.62 MPa (1250 psig).

ABNORMAL TRANSIENTS RESULTING FROM CONTROLLER FAILURE

Extensive test data and analytical predictions indicate that power of a metal fueled LMR can be controlled using primary flow, secondary flow, and turbine pressure over a wide power range with little or no use of control rods. The main reason for this controllability is the F/P dependency of the feedbacks in metal-fueled core. This advantage could greatly reduce requirements for high worth control rods and thereby eliminate the consideration of core disruption resulting from rod runout as a design basis event. Of all the potential controller failures examined, the most severe in EBR-II appears to be one causing a secondary flow runout. It was noted in test No. 1 that when a secondary flow runout event occurs at low power and flow conditions, the reactor temperature could be excessive if this event is combined with failure of the SOT trip function in EBR-II even this event is very unlikely and the transient is slow enough to have operator intervention. Work is being done to prevent such a controller failure by either software or hardware design.

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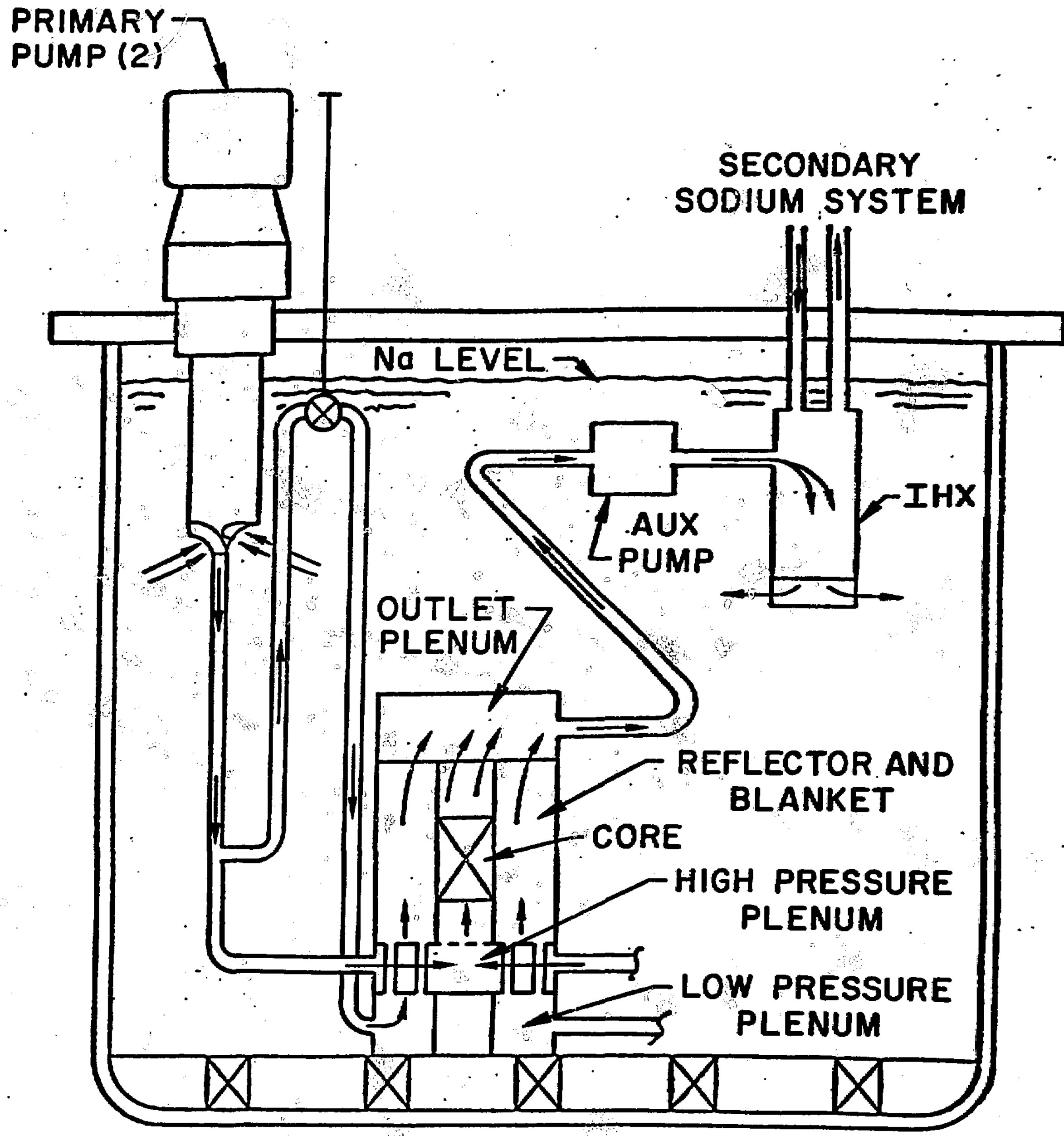

Figure 1 - EBR-II primary system

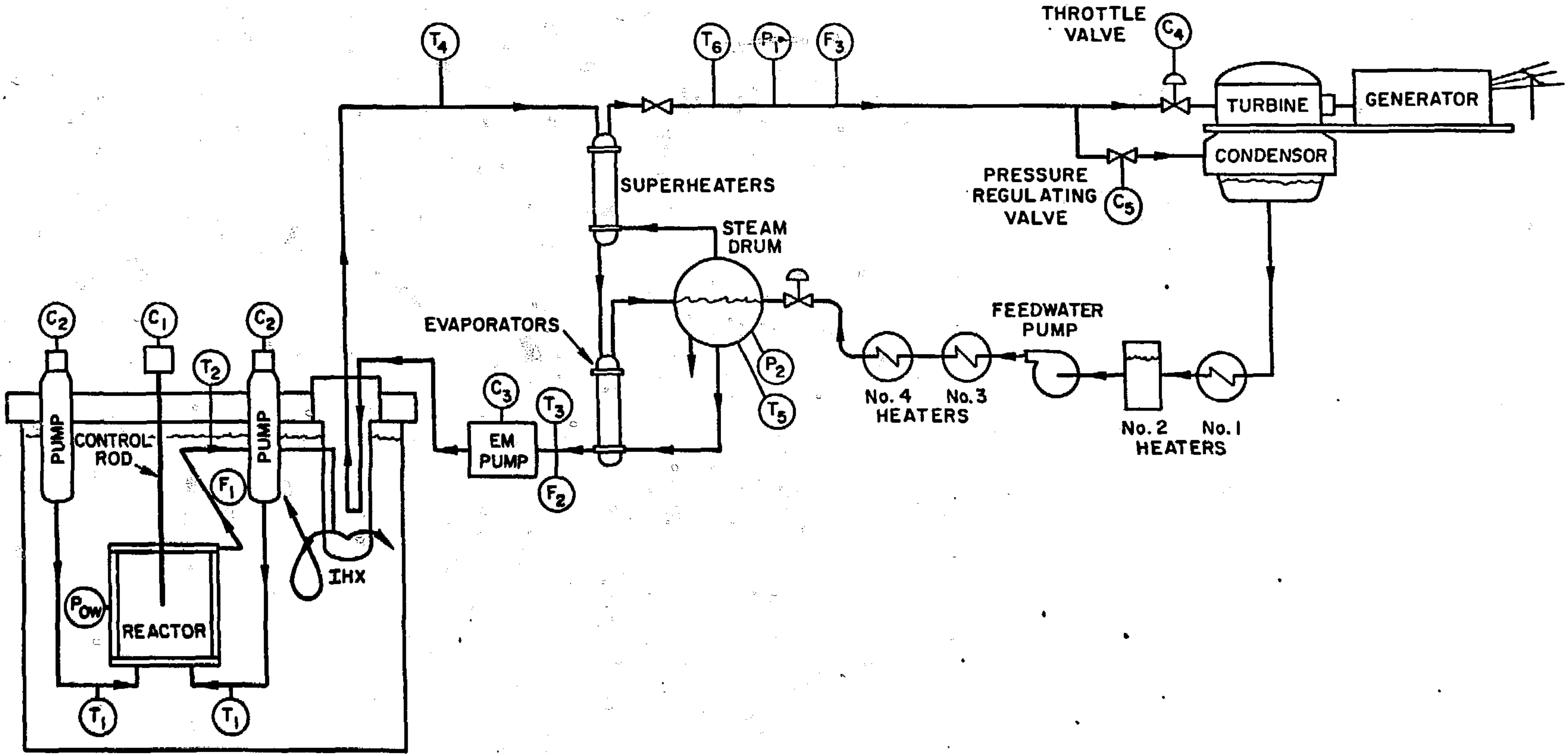

Figure 2 - Schematic Representation of EBR-II plant and Controller output Elements used in the tests

Normalized Power & Flow/Test No. 1

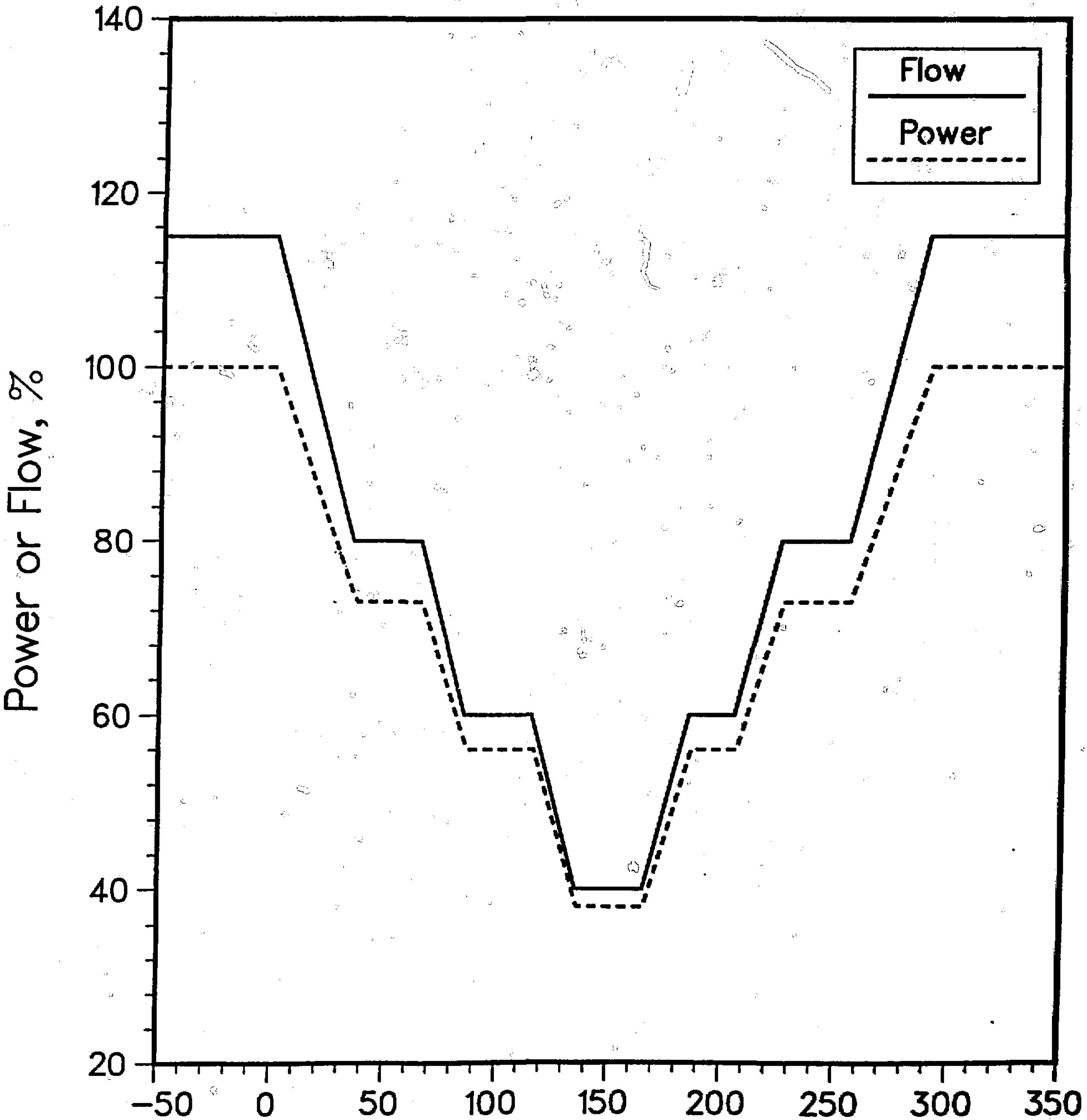

Figure 3 - Power and Flow of Test NO. 1

Normalized Secondary Flow/Test No. 1

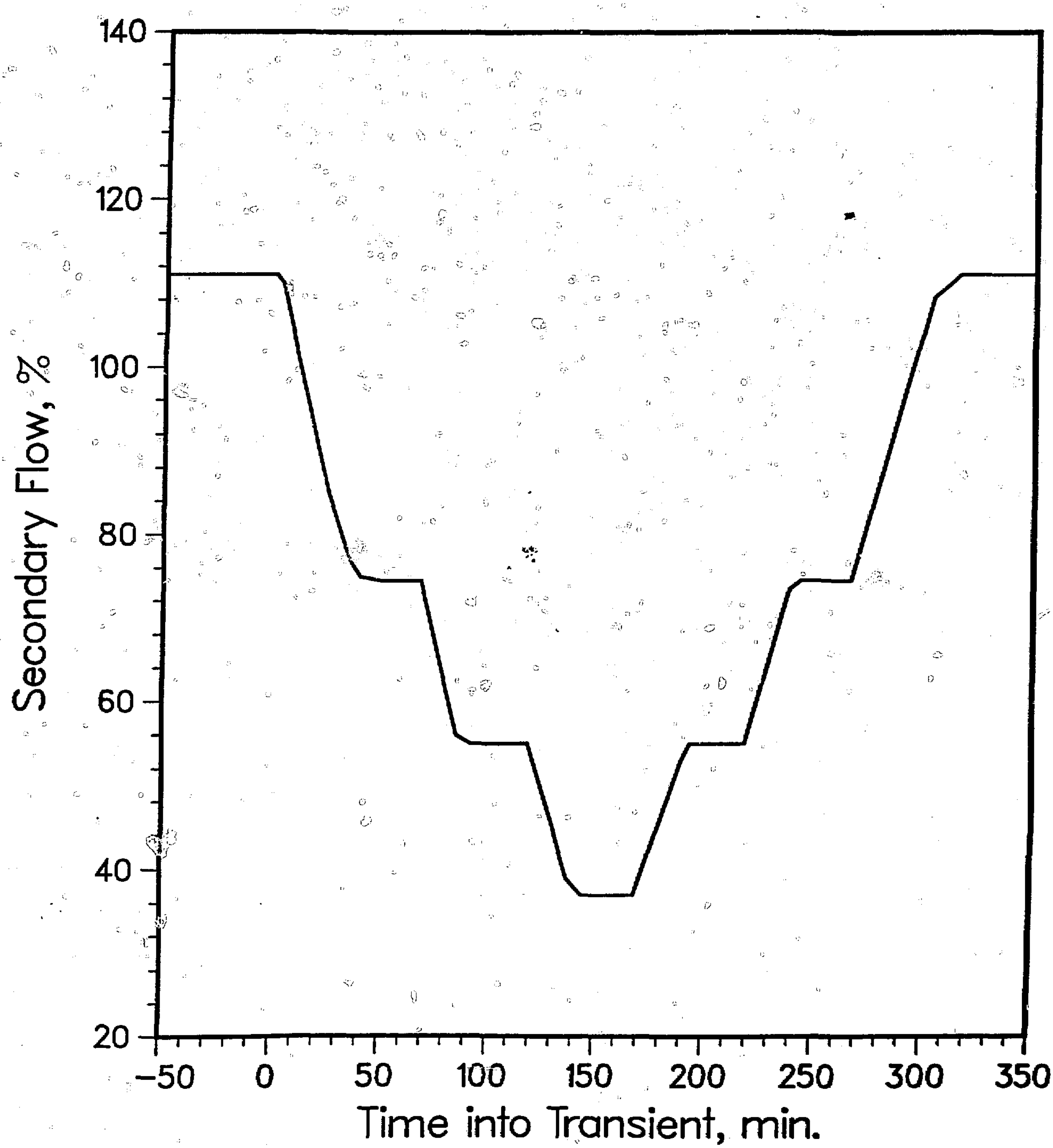

Figure 4 - Secondary Flow Response of Test NO. 1

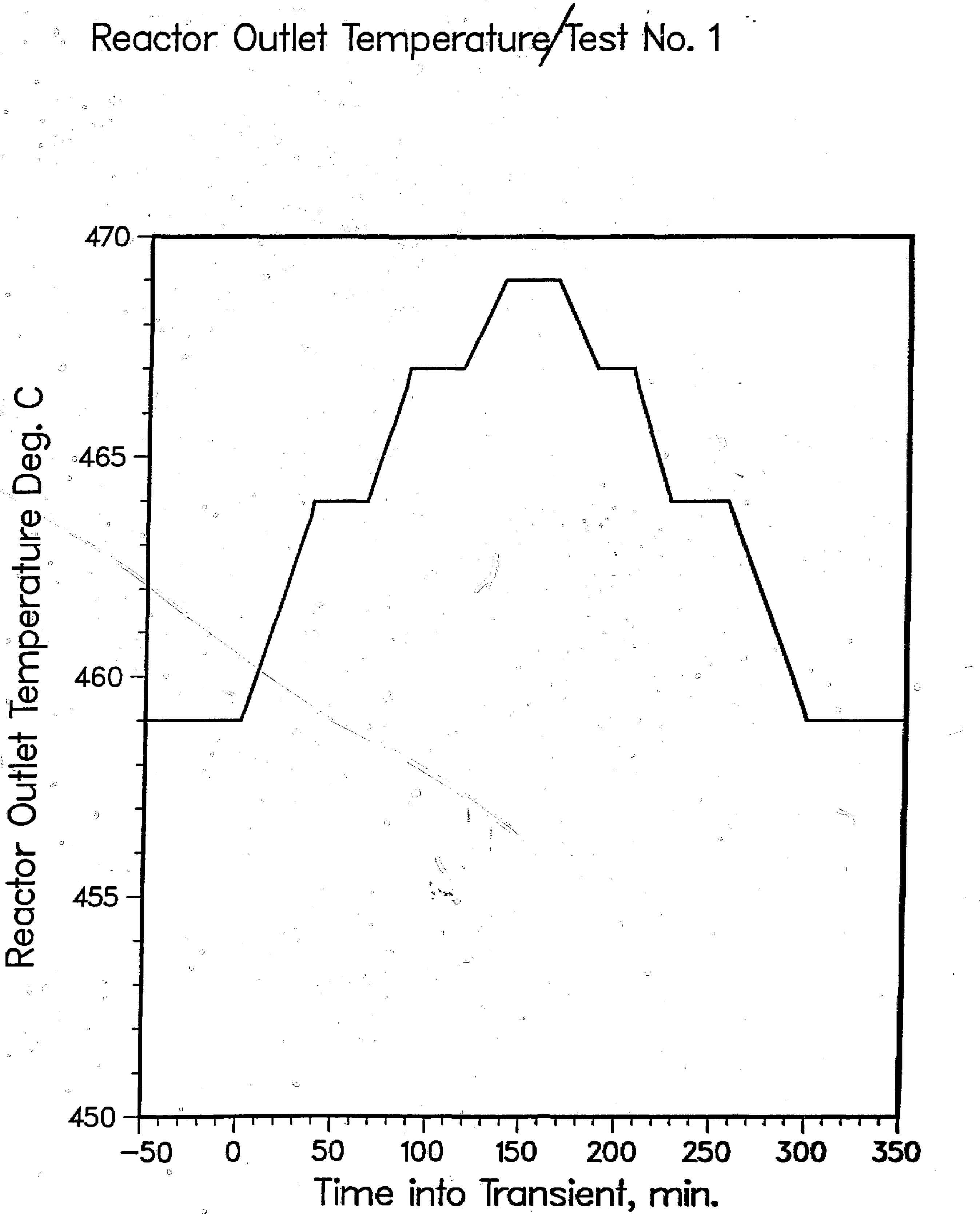

Figure 5 – Reactor Outlet Temperature of Test No.

Normalized Power & Secondary Flow/Test No.2

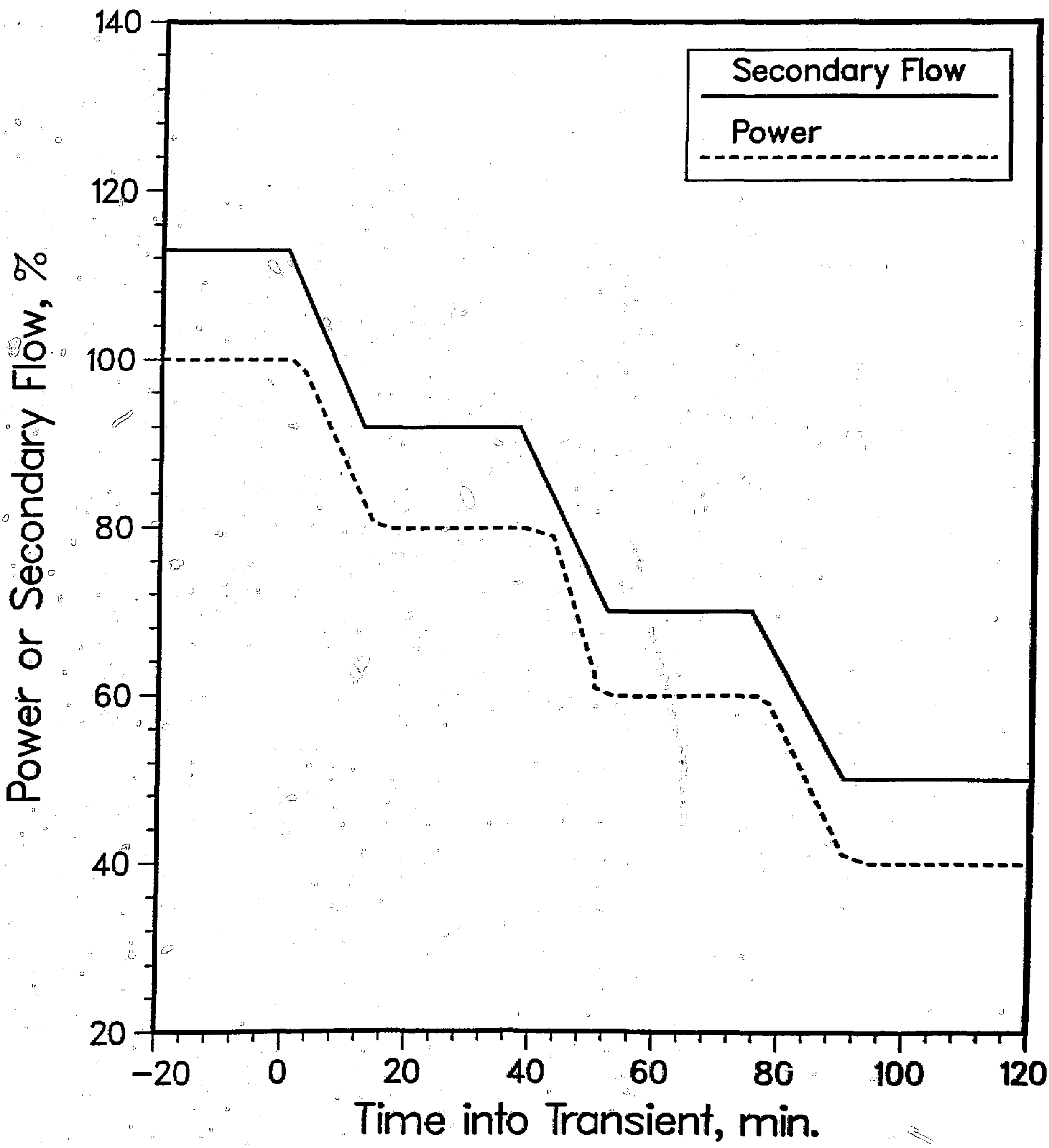

Figure 6 – Power and secondary flow of Test No. 2

Reactor Temperature/Test No.2

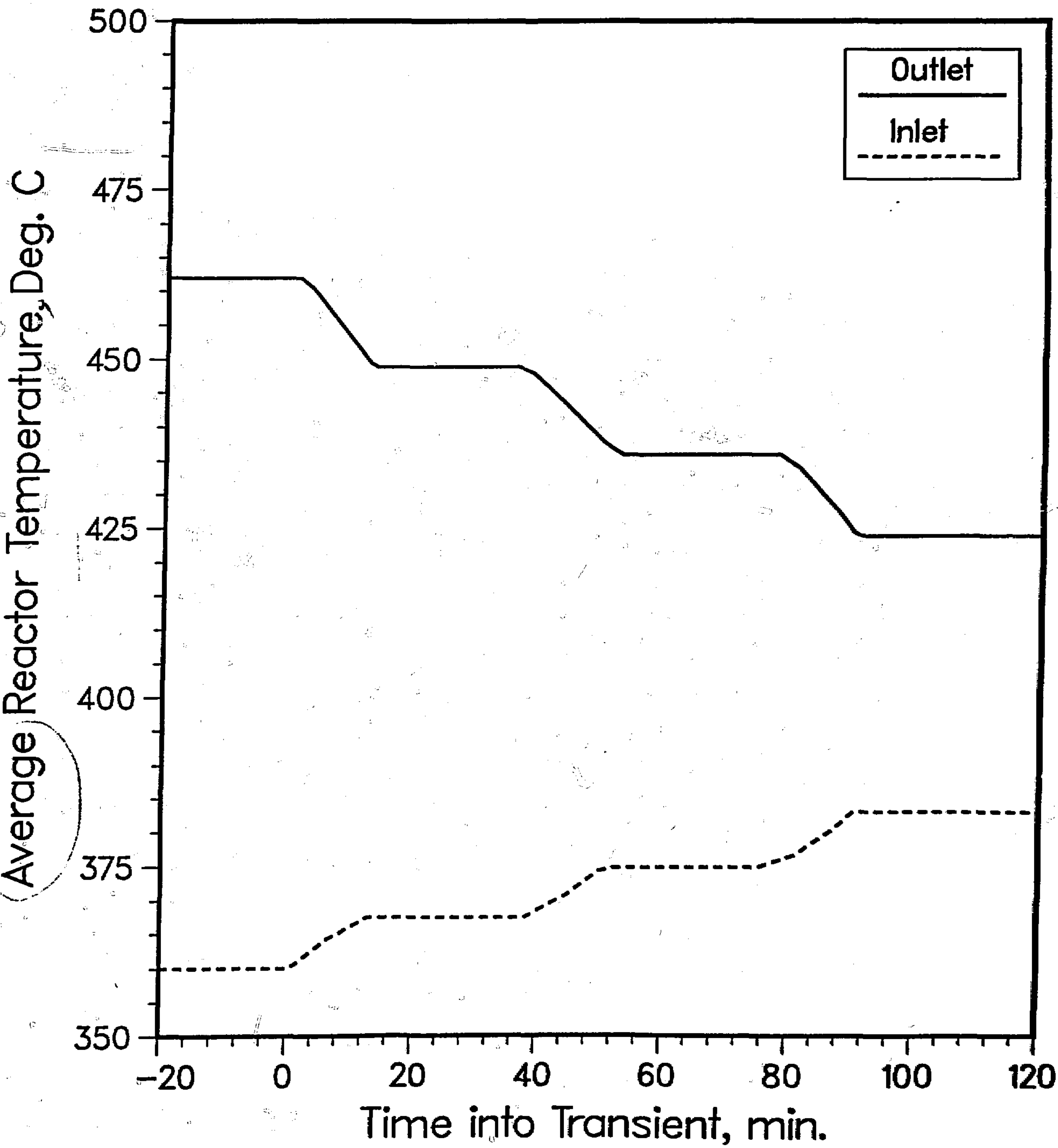

Figure 7 – Reactor inlet and outlet temperatures of Test No. 2

Power and Secondary Flow/Test No.3

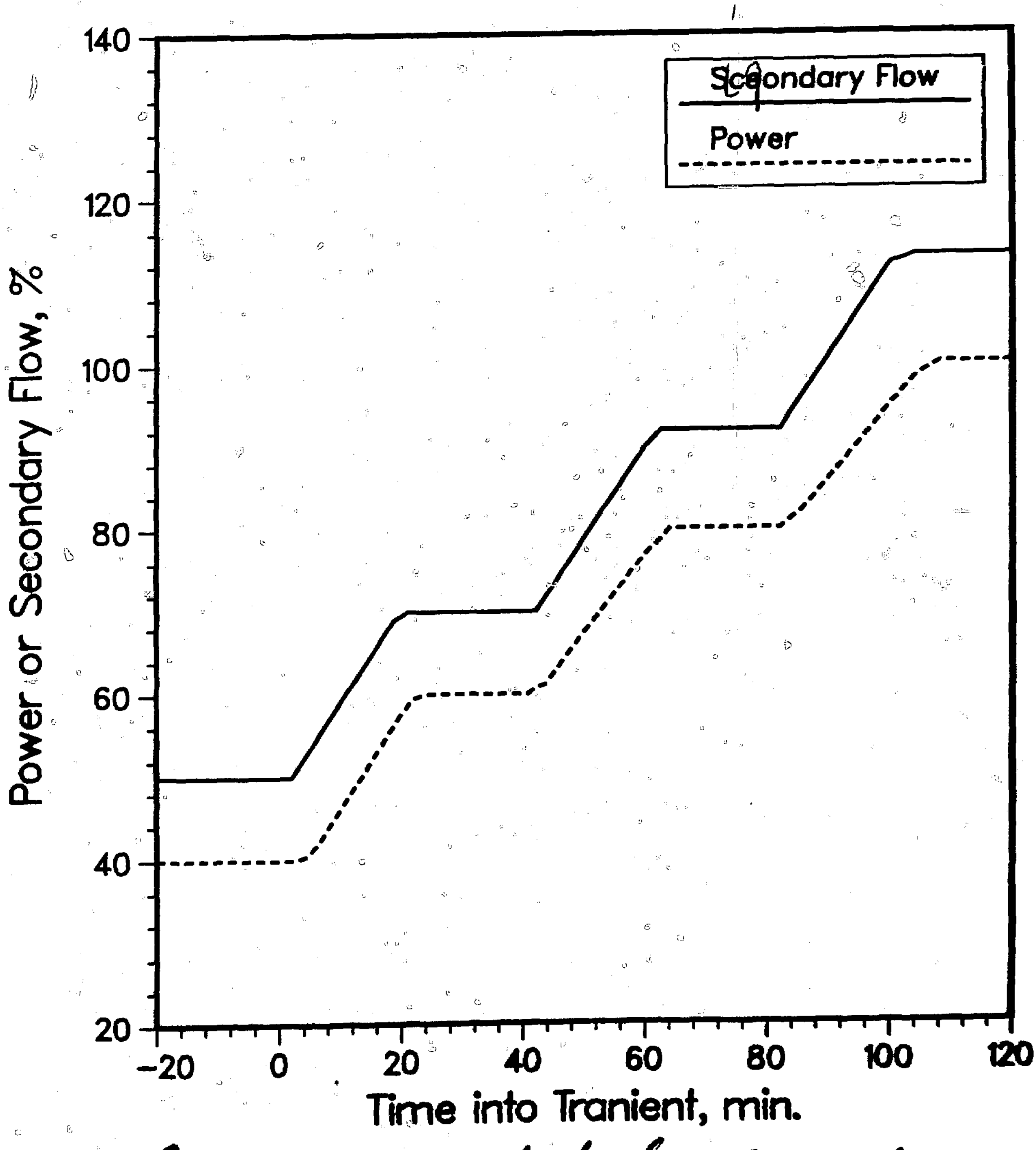

Figure 8 - Power and Secondary Flow of Test No. 3

Reactor Temperature/Test No.3

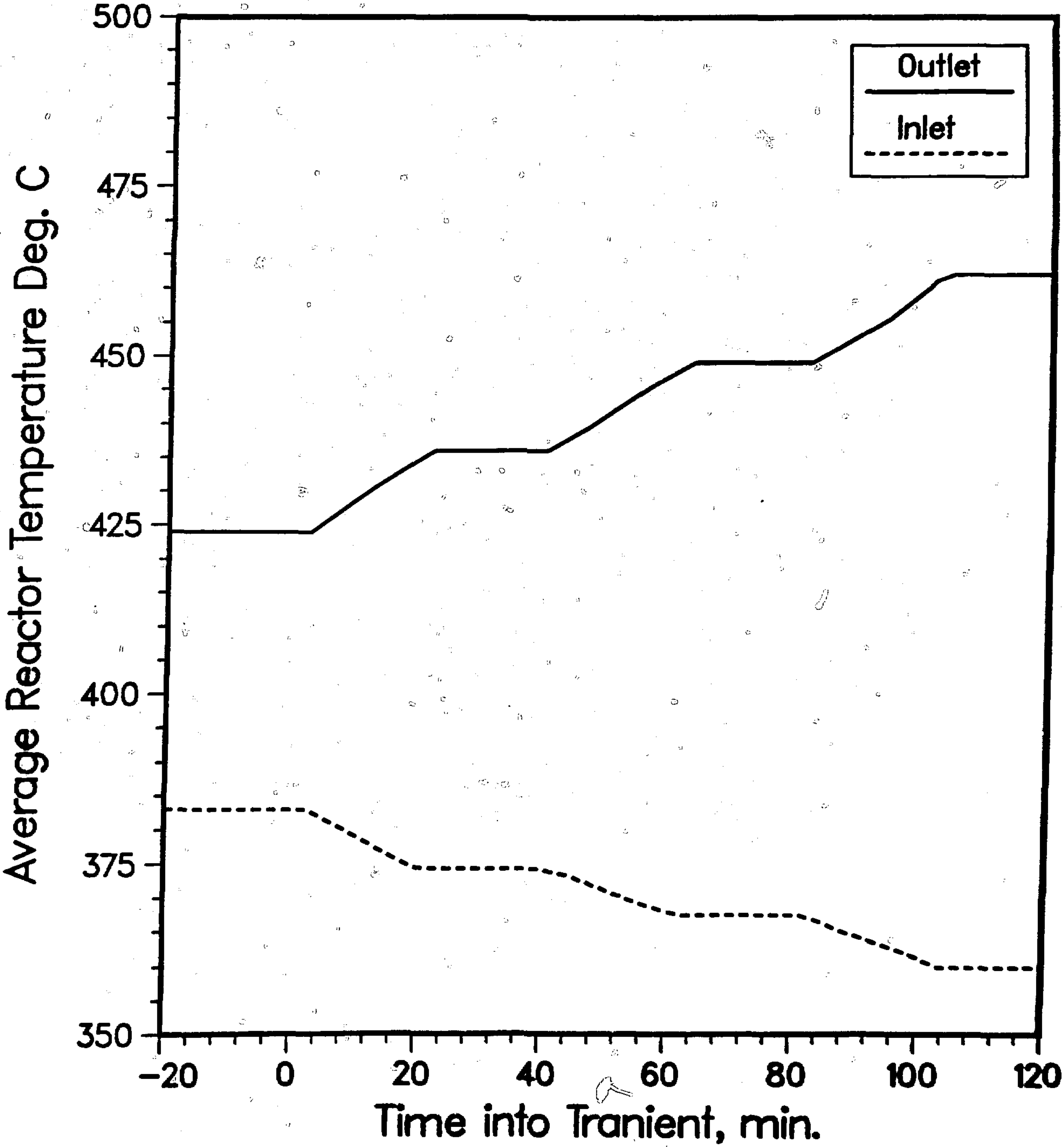

Figure 9- Reactor inlet and outlet Temperatures of Test

Power, Flow, and Pressure/No.4

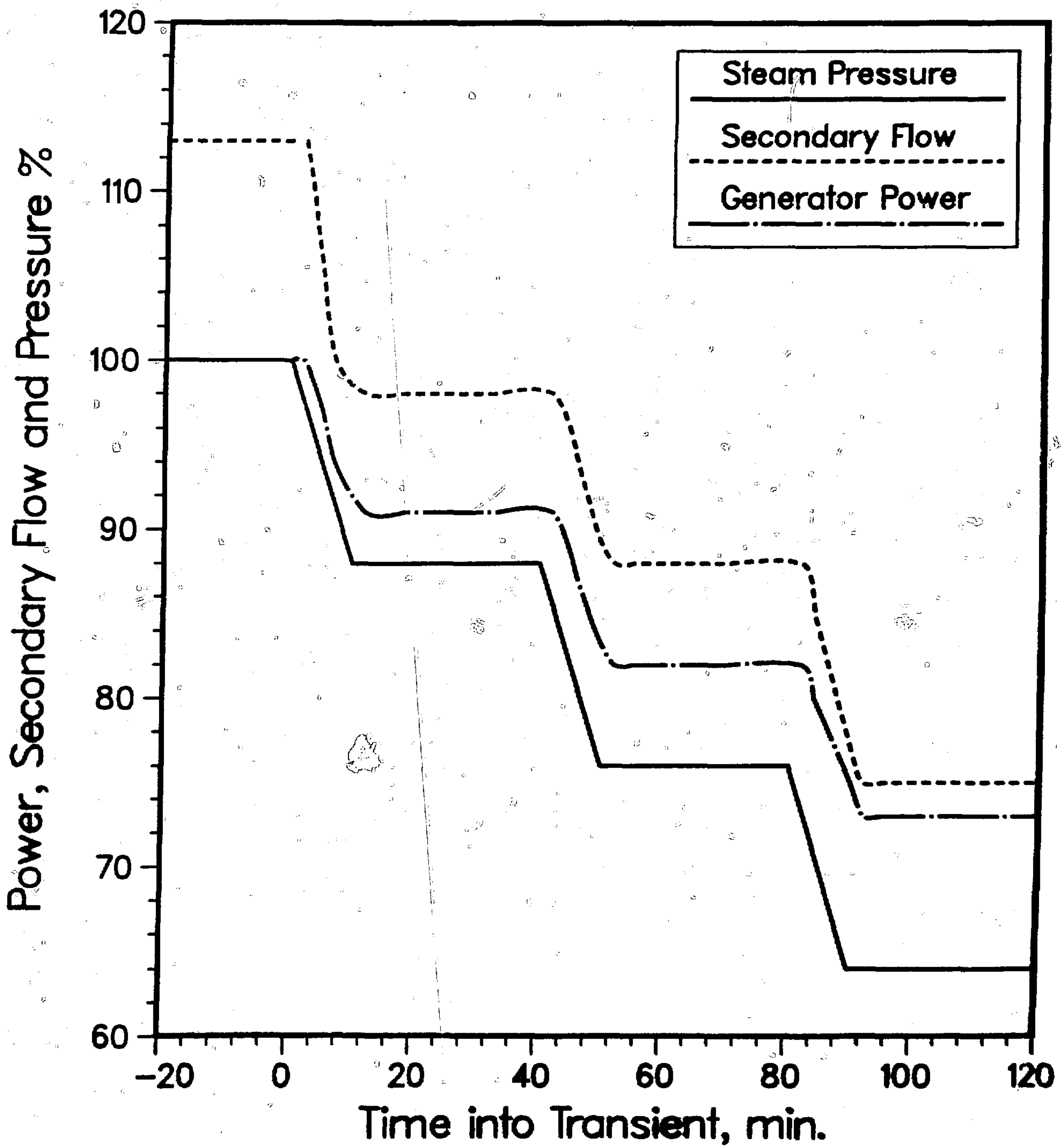

Figure 10 – Plant Response of Test No. 4