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by

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I. INTRODUCTION

Recent testing in the Experimental Breeder Reactor-II (EBR-II) has demonstrated the potential for passively accomplishing basic reactor safety functions. Tests have shown that reactor feedbacks can safely shut down the reactor for loss of flow[1] and loss of heat sink accidents[2] and that natural convection can subsequently cool the reactor without aid of active components.[3] Analysis has indicated that passive safety for the transient overpower and loss of coolant events can also be achieved for a larger LMR with metal fuel.[4] The analyses and experiments suggest that there will be both special constraints and opportunities for the design of automatic control and protection systems for inherently safe reactor plants. The constraints are generally a restriction on the "control band" of active components so that they cannot override the reactivity feedbacks, or natural convective heads which otherwise inherently carry out the safety functions.[3]

The opportunities for improvement of reactor controls are generally in two areas. First the complexity of safety systems (which evolves from the philosophy of safety reliability through redundancy, diversity and independence) can be reduced. The reliability and diverse nature of passive shutdown, convective heat removal and hydrostatic mitigation of leaks in the primary boundary allow considerable simplification or deletion of active control and protection system, while at the same time improving reliability of the safety functions.

Second, the operability of reactors may be improved by emphasizing passive response. If inherent safety features can be made to be an outer

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bound to a control system, then there will be more freedom to innovate and improve plant control with advanced diagnostics and control methods.

We in EBR-II first identified constraints on control system design for inherent safety while doing the safety analysis for the inherent safety demonstration tests. It was necessary to deenergize the control rods (while still preserving their trip capability) for the LOFWS tests so that failures could not cause them to override the feedbacks and prevent a safe passive shutdown.[5] A similar solution could also be used during normal operations to "passively" prevent a transient overpower accident. First, however, an alternate way to control (load follow) the plant must be developed. Tests conducted in November 1987 showed several possibilities for controlling the plant with minimum control rods. Further tests are planned to investigate these alternate control system designs and identify issues and tradeoffs in inherent safety, complexity, and plant operability.

The balance of this paper is thus divided into three sections, passive safety considerations and a summary of the earlier test results, the origin of focus on control and supporting tests in EBR-II, and future tests.

PASSIVE SAFETY

The basic safety goal in any nuclear power plant is to match heat removal to heat generation in the reactor under all operating conditions, both normal and accidental. The role of passive (or inherent) safety is to call into play highly reliable physical processes to drive down reactor power and maintain adequate cooling subsequently, in case an undercooling or overpower accident is initiated and the plant protection system fails to function properly.

There are two general classes of undercooling accidents, loss of primary flow (LOF) and loss of heat sink (LOHS). A transient overpower (TOP) accident may be initiated in a liquid metal reactor (LMR) by primary pump runup, sudden increase in power demand in the balance of plant, or by the run-out of control rods. If the plant protection system does not respond as required to one of these accidents, the latter are referred to as "unprotected", or "without scram."

If it can be shown that a metal fueled LMR plant can survive any unprotected accident falling within the categories above, and be subsequently passively cooled by natural convection, such a plant would pose minimal safety concern. If it can be further shown that the plant is immediately restartable following any of these unprotected accident scenarios, both the safety posture and economics of the plant will be greatly enhanced. The goals of the plant testing program at EBR-II are to demonstrate the role of passive safety features in mitigating unprotected accident scenarios in this plant, and to indicate by appropriate analysis that the results are extrapolable to larger size metal-fueled LMR's being developed at Argonne National Laboratory in the Integral Fast Reactor Program.

Loss of primary flow without scram tests were run from 100% initial power in EBR-II in February and April 1986. These tests were the climax of extensive feasibility analyses, driver fuel qualification tests (both out-of-pile and in-pile), specific safety analyses, and prior tests from lower initial powers.[6] These tests were both run with special trips in place to scram the reactor in case of too high outlet temperatures or too rapid a pump coastdown rate being measured during the course of the transient. In addition, the control rods were de-energized just prior to the tests, so they could not be moved inadvertently during the tests, although their scram capability was preserved.

With the reactor initially at normal full power and flow, the normal loss-of-flow trips were bypassed. Then the electrical power to the two primary pumps and the secondary pump was simultaneously turned off, and the primary pumps allowed to coast down in about 100 seconds. During this time the core temperatures increased, peaked, and then decreased as feedback due mainly to thermal expansion effects drove down the power. The peak temperature of driver fuel cladding was about 800°C, inferred by analysis based upon coolant temperatures measured near the top of a special driver subassembly.

The only fuel changeout in the time interval between the two tests was that required by some fuel reaching its peak allowable burnup in the interval. Subsequent analyses and post-irradiation examination indicated no perceptible fuel damage from the two tests.[7]

Two unprotected-loss-of-heat-sink tests were also run from 100% initial power.[6] These tests were much easier to run than the unprotected loss-of-flow tests, because the peak temperatures in the former were much lower than in the latter. No bypasses of the plant protection system were thus required, and no backup trips were needed. The tests were run simply by turning off the secondary pump with the plant initially at full power. As the primary pumps continued to operate and dump increasingly hot primary sodium back into the reactor inlet, the entire primary system increased in temperature, driving the reactor power down by thermal expansion feedback. The power went to essentially zero when the primary system inlet temperature increased by about 42°C.

CONTROL

Even though the unprotected LOF and LOHS tests have both been successfully run in EBR-II from 100% initial power, there remains the unprotected TOP. As noted earlier, there are really three sub-categories of TOP events for an LMR, the familiar control rod withdrawal, but also primary pump run-up and sudden increase in power demand in the balance of plant. Focusing on the rod withdrawal (rod insertion in EBR-II, with its fuel-bearing control rods), it is known that only about half of the power reactivity decrement (reactivity addition needed to go from hot critical to full power) could be inserted from initial full-power conditions without taking the driver fuel above currently approved EBR-II safety limit temperatures. This is only about 1/5 of the worth of one control rod. As increasing amounts of reactivity would be added, there would be an increasing level of fuel damage.

The solution to this problem is to limit the total worth of control rods by controlling power largely by some other means. This is the substance of the Plant Inherent Control Tests to be discussed shortly.

But controlling power by other means requires the development of one or more control strategies. That is, the ability to conduct meaningful (limiting) rod withdrawal tests, as well as tests in the other two subcategories of unprotected TOP's, requires the development of a compatible control strategy.

There are two other critically important reasons for work on a control strategy. First, control must be carefully designed not to override inherent safety characteristics of a plant. We have encountered this problem in utilizing our automatic control rod drive system for EBR-II. Second, the control system must be designed to accommodate passively the malfunction of automatic controllers. Thus, preparing to run an unprotected TOP in EBR-II is a broad-based activity.

Quasi Static Control Tests

The ability to quasi-statically control reactor power with changes in primary flow, and/or in a turbine power, load following mode were shown in the EBR-II 1987 tests.[8] Subsequent analysis has shown that power can be controlled with a control rod with limited reactivity in such a way to preserve the capability to passively shutdown for a loss of flow without scram.

The reactor power change during the tests can be explained by considering changes in reactivity ($\delta\rho$), power (δP), power/flow ratio ($\delta(P/F)$), and reactor inlet temperature (δT_i). Reactivity changes due to fissile atom depletion are neglected as is control rod reactivity. A quasi-static (linearized) approximation for the reactivity perturbation can be expressed as:

$$\delta\rho = A\delta P + B\delta(P/F) + C\delta T_i, \tag{1}$$

in which A is the power coefficient representing reactivity feedbacks proportional to power change alone, B is the coefficient representing the reactivity feedbacks proportional to the power-to-flow ratio (P/F) change, and C is the coefficient representing reactivity feedbacks related to reactor inlet temperature variation.

With the control-rod-drive mechanism deenergized, the net reactivity change from one steady state to another would be zero ($\delta\rho = 0$). For the tests in which reactor inlet temperature is kept constant, and where power is controlled with primary flow, then $\delta T_i = 0$. By substituting $\delta T_i = \delta\rho = 0$ into Eq. (1), the relationship of P/F between two equilibrium states, 1 and 2, can be expressed as:

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

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, loading conditions and the bowing reactivity components.

For the load following tests, the primary flow is kept constant (i.e., $F_1 = F_2$), and the reactor power responds to changes in reactor inlet temperature. The relationship between power and reactor inlet temperature at two equilibrium states is:

$$P_2 - P_1 = C \times (T_{i1} - T_{i2})/(A+B), \tag{3}$$

where (A+B) is the approximate power reactivity decrement (PRD), i.e., the reactivity addition required to raise the power from zero power hot critical to 100% power at 100% flow. The PRD in EBR-II is about 0.28$ depending on loading conditions, and C is about 0.007$ per °C based on data gathered from reactor inlet temperature perturbation tests and LOHSWS tests in EBR-II. The final equilibrium conditions of the PICT tests can be estimated using Eqs. (1) to (3).

PICT 1 - Control of Reactor Power with Flow

The purpose of this test was to study the feasibility of controlling reactor power using primary flow. Referring to Fig. 1, the primary pump speed (C_2) was controlled to a prescribed speed by computer software, and the secondary flow was regulated by the secondary EM pump (C_3) through a secondary flow/tank temperature controller to maintain a constant reactor inlet temperature. The turbine admission valve controller was used to maintain a constant steam header pressure by adjusting the Turbine Admission Valve (TAV) position (C_4).

The initial reactor inlet temperature and turbine header pressure were controlled to the normal constant operating values of about 371°C and 8.7 MPa, respectively, and these values were controlled to remain essentially constant throughout the test. The initial conditions of the test (96% rated power 110% rated flow) provide a comfortable margin below normal operating conditions. The reactor flow, the forcing function in this test, was reduced to about 42% at 1% per minute in three steps and then the flow was returned to 110% in a similar manner. The intermediate values were 77 and 58% flows. When the flow was reduced, the reactor temperature increased, which caused the reactor power to decrease due to negative reactivity feedback as explained in Eq. (2). The normalized flow and power profiles of the test are illustrated in Fig. 2. In order to control the reactor inlet temperature, the secondary flow tended to follow the primary flow and power variation. It was noted during the transient that the reactor inlet temperature remained nearly constant as demanded, with a deviation of no greater than 2°C from the initial value as shown in Fig. 2. This deviation was somewhat reflected in the transient power response since the reactor power varies about 2.7% for every 1°C change. Although the reactor flow and the inlet temperature at the end of the test were very close to the initial conditions, the final power was about 3% lower than the initial power as shown in Fig. 2. This was caused by driver fuel burnup during the test period. It was noted in Fig. 2 that in PICT 1 the power-to-flow ratio was below the normal operation value throughout the test. After the test a calibrated control rod was moved to obtain the initial power and thereby to measure reactivity loss due to burnup. If the reactor power was controlled to be constant by varying the reactor flow, the burnup would be manifested as an increase in primary flow.

The results indicated that reactor power can be regulated using primary flow. However, if a precise transient reactor power profile is required, the secondary flow/tank temperature controller should be more precisely tuned, if possible, such that reactor inlet temperature variation can be reduced during the power and flow maneuvers.

PICTs 3 and 4 - Load Following

PICTs 3 and 4 demonstrated the slow reactor power change and load-following (reactor power follows the turbine-generator load demand) maneuvers involving reactor power, inlet temperature and turbine generator output demand changes. These plant disturbances, in turn, were controlled by the secondary pump (C_3) and the TAV position (C_4). The reactor power in these tests were maneuvered from 96% to about 50% (PICT 3) and then back to about 96% (PICT 4). For both tests, the primary pump speed was controlled to maintain a constant reactor flow. In PICT 3, the demanded reactor inlet temperature setpoint was first set (see Fig. 3) and the secondary flow/tank temperature controller thus responded by regulating the secondary flow. At the same time, the turbine admission valve controller was used to maintain a nearly constant steam header pressure by adjusting TAV position. In PICT 4, the TAV was controlled to attain desired electric output. The secondary pump was controlled to keep the steam header pressure constant.

The primary flow and turbine header pressure of both tests were maintained constant at 96% and 8.7 MPa, respectively. The initial power of PICT 3 was 96%, and the test was initiated by increasing the reactor inlet temperature setpoint from the initial 363°C to 368°C at 0.56°C per minute. This was accomplished by changing the secondary flow as described above. As the reactor inlet temperature increased, the reactor power decreased due to negative reactivity feedback as indicated in Eq. (3). After the plant parameters stabilized, the reactor inlet temperature demand was increased to 375°C, and then to 383°C, which caused the reactor power to decrease. The reactor inlet temperature of PICTs 3 and 4 is given in Fig. 3. In the figure the first 6 h, approximately, is PICT 3. The remainder is PICT 4. An increase/decrease in reactor inlet temperature corresponded to a decrease/increase in reactor and generator powers as indicated in Fig. 4. The results indicate that the power can be easily controlled by the reactor inlet temperature. The secondary flow controller realignment occurred between 6 and 7.5 h in the figures. In the realignment, the secondary flow/tank temperature controller and the TURBINE ADMISSION VALVE were replaced by the secondary flow/turbine pressure and the generator power controllers. The temporary disturbance shown at ~ 7 h was due to the above controller changes.

PICT 4 which followed shortly after the completion of the controller realignments, is essentially a mirror image of PICT 3. The mode of control, however, was changed between PICTs 3 and 4. PICT 4 was basically a load-following maneuver. The measured turbine power outputs at the intermediate steps recorded in PICT 3, as shown in early portion of Fig. 4, served as the demanded turbine load output for the corresponding steps in PICT 4.

In PICT 4, the test was initiated by increasing the turbine-generator output demand from 9.7 MWe to 13.4 MWe at 0.2 MWe per minute using the generator power controller. As the turbine-generator output demand was increased, the TAV was automatically opened to meet the output power demand. The increased steam caused by the TAV opening tended to decrease steam header pressure. The secondary flow/turbine pressure controller thus responded by increasing the secondary flow in an attempt to keep the steam header pressure constant at 8.7 MPa. The increase of secondary flow to meet the higher pressure demand caused the tank temperature to decrease and reactor power to increase because of reactivity feedback. The turbine-generator output demands for the three steps were 13.4, 16.5 and 17.7 MWe. The reactor and generator powers of PICTs 3 and 4 are given in Fig. 4 which show very similar response patterns, and indicate that plant efficiency drops only slightly at low power conditions. The results indicated that the nuclear power output can be adequately controlled by the combination of secondary flow and generator output controllers.

The final power of PICT 4 was originally planned to be the initial power of PICT 3. Fuel burnup and the attendant reactivity losses during the test were thus to have been accommodated for the planned condition with a lower reactor inlet temperature. At the end of PICT 4 more secondary flow would therefore have been required than at the beginning of PICT 3 in order to obtain the required (lower) reactor inlet temperature. Due to limited secondary pump capacity, however, the final power of PICT 4 was somewhat lower than planned. Had the test been run from a slightly higher initial reactor inlet temperature the control scheme would have adequately compensated for fuel burnup reactivity loss over the whole operating range.

Power Control with Limited Control Rod Reactivity

Recent analysis supports the idea that control rod reactivity can be limited so there would still be passive shutdown for loss-of-flow-without-scram (LOFWS). For the traditional way of controlling reactor power. A fuel element damage evaluation was performed for two LOFWS events for the new EBR-II Mk-III fuel. Two cases were examined. In case 1, the reactor is assumed to operate without the Automatic Control Rod Drive System (ACRDS), while in case 2 the ACRDS was assumed to operate with a maximum reactivity worth of 4¢. During a LOFWS event, the reactor temperature increases and causes negative reactivity feedbacks, which reduce the reactor power. If the ACRDS is in operation, reactivity would be inserted to maintain a constant power. In case 2, the reactivity worth of 4¢ is inserted during the LOFWS transient.

The system simulation code NATDEMO was used to predict the plant responses for the transients, and the thermal-hydraulic code HOTCHAN was employed to determine temperatures of individual subassemblies. The hottest subassembly allowed by the Technical Specification (T.S.) was considered in both temperature and damage calculations.

The peak sodium temperature for the two cases were below the sodium boiling temperature. A failure analyses of the cladding indicated that it would not fail due to either eutectic penetration of the clad or stress rupture. Thus if the ACRDS is used with a rod with less than 4¢ reactivity over that for full power, the reactor is passively safe for LOFWS.

Protection Systems for Inherently Safe Reactors - Availability

Consideration of passive safety features in the design of control and protection systems can increase the availability of metal fueled reactors. Singer[9] analyzed primary pump binding events in EBR-II and found that a power-to-flow scram function would have enabled EBR-II to ride through the temporary flow reduction accompanying the pump binding. The reactivity feedback would have reduced power keeping the power-to-flow ratio within the scram envelop and keeping reactor temperature low.

The primary pump binding incident occurred at EBR-II on March 25, 1987. This event resulted in a slow increase in the power supplied to one pump motor up to the control system limit, at which time the speed of the affected pump began to decrease. The decrease in the speed of pump 1 was sufficient to cause the reactor flow to drop to its setpoint, and the reactor automatically scrammed on this low-flow signal. However, the pumps were not secured and continued to operate as their speeds continued to decrease. Approximately 13 to 14 min. after the initial indication of an abnormal condition, the pump speeds started to return to their original values without any intervention by the operators. During the entire time period prior to scram, the reactor temperatures remained essentially unchanged. After postevent evaluations, it was determined that some foreign material, most likely sodium oxides, had accumulated in the labyrinth seal area of pump 1, causing increased friction. Due to continued operation of the pump, this material somehow broke loose, permitting the pump to return to its original operational state with any additional operator action.

The events of the pump binding were simulated with the NATDEMO code with the additional assumption that no scram occurred. Two cases were considered, differing only in the type of secondary flow control assumed in order to evaluate the importance of reactivity feedbacks dependent on the core inlet temperature. The first case analyzed represented the actual plant conditions that occurred during the pump binding, except that a scram was not allowed and a constant secondary flow was maintained after the scram (during the event, secondary flow was constant until the scram, then tripped to a low level). The second case was identical to the first, except that the secondary flow was controlled to maintain a constant reactor inlet temperature.

For the first case the power was seen to initially decrease due to the slight increase in core outlet temperature. After ~ 7 min, however, the power start to increase. This increase was caused by the reactivity feedback associated with the decreasing core inlet temperature. The inlet temperature decrease because of the mismatch in primary and secondary flow coupled with the reduced reactor power. Ultimately, as the primary flow increase due to recovery of the affected primary pump, the reactor outlet temperature decrease and the inlet temperature increase. These two variations in temperature occur somewhat out of phase, resulting initially in a net positive reactivity change driving the power up and then a net negative reactivity feedback effect to terminate the power increase. The maximum outlet temperature reached was 497°C. This ~ 17°C increase in temperature would result in exceeding the assembly outlet temperature trip point and a scram still would have occurred.

In the second case, however, where the inlet temperature was held fixed by controlling the secondary flow, the positive reactivity feedback associated with the inlet temperature decrease was eliminated, and the power monotonically decrease so long as the primary flow decrease. The power start to increase again after the flow start to increase a ~ 3°C and maintain a large margin to its trip setpoint.

From these results, it is clear that a scram is not required to protect the reactor from overheating if the plant control system is such that the secondary flow is controlled to maintain a constant core inlet temperature. Thus, the loss of reactor availability and the sustaining of a scram-induced thermal transient was caused by the use of a low flow trip that was unnecessary from the standpoint of reactor safety.

Test Plans

A series of tests are being planned or have been done to characterize EBR-II for control failures which could lead to overpower. Also, we are planning to dynamically test plant control methods which emphasize passive safety.

Lehto, Dean and Fryer[10] conducted primary pump run-up tests which showed that the increase in power due to increase in primary flow was acceptable. Primary flow was increased from 32% to 100% in 20 seconds from an initial power to flow ratio of 1.0. Power followed flow and leveled off at about 90%. Thus the final P/F ratio was less than 1.0 and core exit temperature was less than at the starting point. During the experiment the secondary flow was conservatively controlled to keep the inlet temperature nearly constant. Lehto also showed by analyses that the power increase would be even less with a control strategy that allowed reactor inlet temperature to increase as a natural consequence of the increase in primary. Thus the transient overpower caused by primary pump runout has been shown by analysis and test to not be a safety problem for EBR-II. This conclusion is also true generally for metal fuel LMRs.

We are planning tests that will examine how the plant limits the effects of large increases in the turbine load that could occur as a result of controller failures. In plant inherent control test 7 - PICT 7 the plant will be in the normal control mode. Referring to Fig. 2, the control rods (C_1) the primary pumps (C_2) and secondary pumps (C_3) will be in manual control and presumed to not be adjusted during the transient.

The turbine admission valve (TAV) is assumed to fail wide open. The analysis shows that the steam pressure will decrease, the secondary cold leg sodium temperature will decrease, and ultimately this will cause a decrease in reactor inlet temperature and an increase in power. The capacity of the TAV and IHX however limit the power increase to about 10%.

PICT 8 is similar to PICT 7 except that the secondary pump (C_3) is automatically controlled to keep the reactor inlet temperature T_1 constant. When the TAV opens and the steam pressure and temperature decreases the secondary flow will decrease thus effectively isolating the reactor from the increased power demand of the turbine. The end result is expected to be near constant reactor power, a steam depressurization and depending on characteristics of the turbine and generator either a turbine trip or less efficient operation at low pressure.

In PICT 9 the plant will initially be in a load follow mode. The turbine throttle C_4 will be controlling electrical load, the secondary pump C_3 will be controlling steam pressure P_1. The control rods and primary pumps will not be adjusted. The turbine admission valve will then be failed wide open. Our predictions show that the secondary pump will run to its upper limit attempting to keep steam/pressure constant. This transmits part of the increase in load demand to the reactor. However, the limited capacity of the turbine admission valve and secondary pump limit the power increase.

PICT 10 will investigate the passive safety characteristics of EBR-II which would limit reactor power during secondary flow runout events. The plant will be operated in the normal mode prior to the event -- the primary pumps (C_2) providing constant flow, and the turbine admission valve (C_4) controlled to maintain constant steam pressure. The control rods will not be adjusted during the test. The secondary pump (C_3) controller is assumed to fail and produce maximum flow. The pretest analysis indicates the plant "sees" the failure as an increased energy transport rate from the reactor to the steam generator. The secondary pump and heat transport system capacities limit the power increase to about 25%. The increase in temperature at the reactor exit is limited and not a problem because the transient is driven by a reactor inlet temperature decrease.

The traditional transient overpower caused by a control rod runout will be simulated in PICT 11. The plant will be configured with the controllers in their normal lineup except the controlling rod will be operated with the automatic rod control rod drive system. The reactivity in this rod will be limited so that it can add only a limited amount (~ 4¢) over the simulated full power point. The limited reactivity is chosen to provide adequate reactivity for maneuvering and daily reactivity burnup compensation and at the same time pressure passively safety for loss-of-flow without scram. Power maneuvers with the limited reactivity control rod will then be demonstrated. A failure of the control rod will then be simulated by rapidly ramping the control rod to the end of its travel. The analysis shows the power will increase about 15% and stabilize if the other controllers act to remove the excess power. If other control schemes (such as steam load following) were used such that secondary flow did not increase to keep reactor inlet temperature constant, the power and temperature extreme would be even less.

Further testing of control methods which enhance passive safety are also being planned. In the near term a dynamic version of the turbine load following control scheme will be tested. In addition to dynamic validation of this control scheme, the test will also investigate the ability to passively compensate for fissile burnup with temperature adjustment and other operational aspects of this control scheme.

A modern control approach to EBR-II is also being developed. In the near term we are planning an "integral test" by applying the approach to control of the reactor inlet temperature. The problem is physically interesting since it involves the nonlinear behavior of the IHX and coolant stratification in the primary tank. On the other hand it is simple enough to use and test control hardware and software interfaces in the plant environment.

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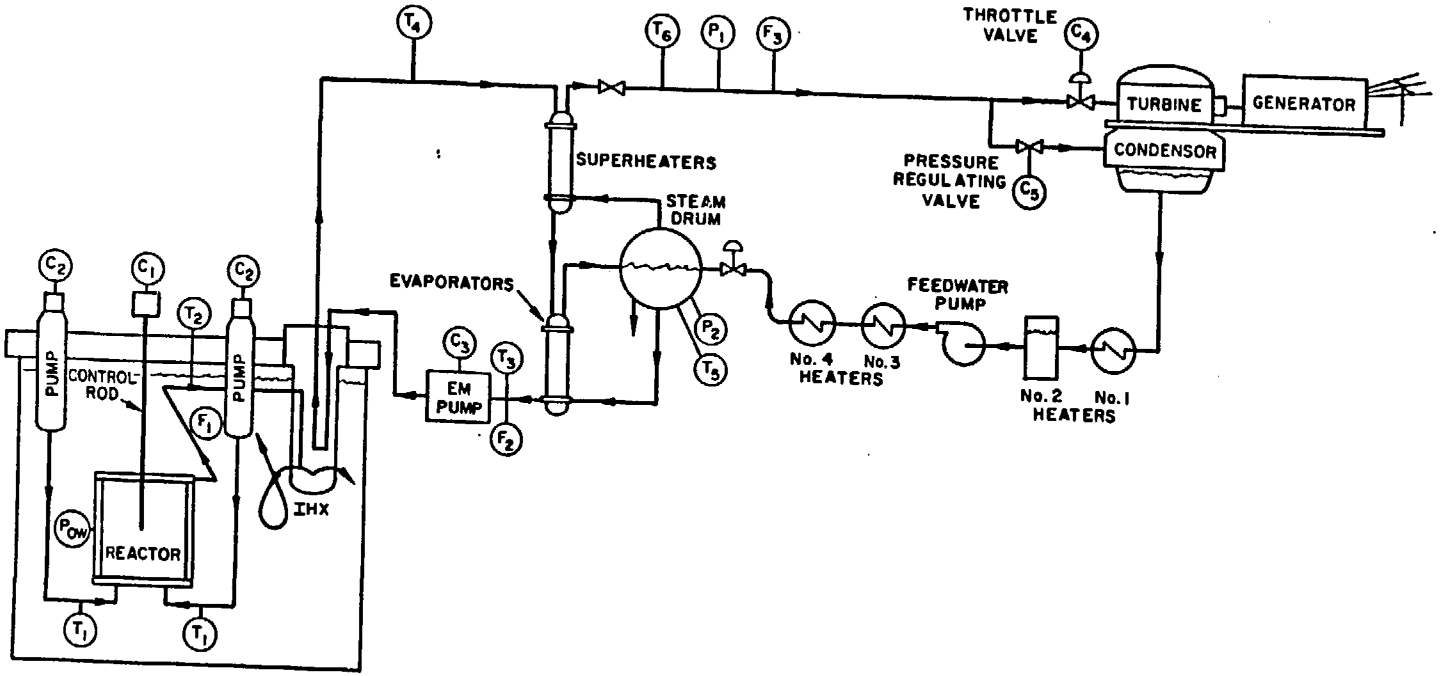

Fig. 1. Schematic of EBR-II Plant and Control Systems

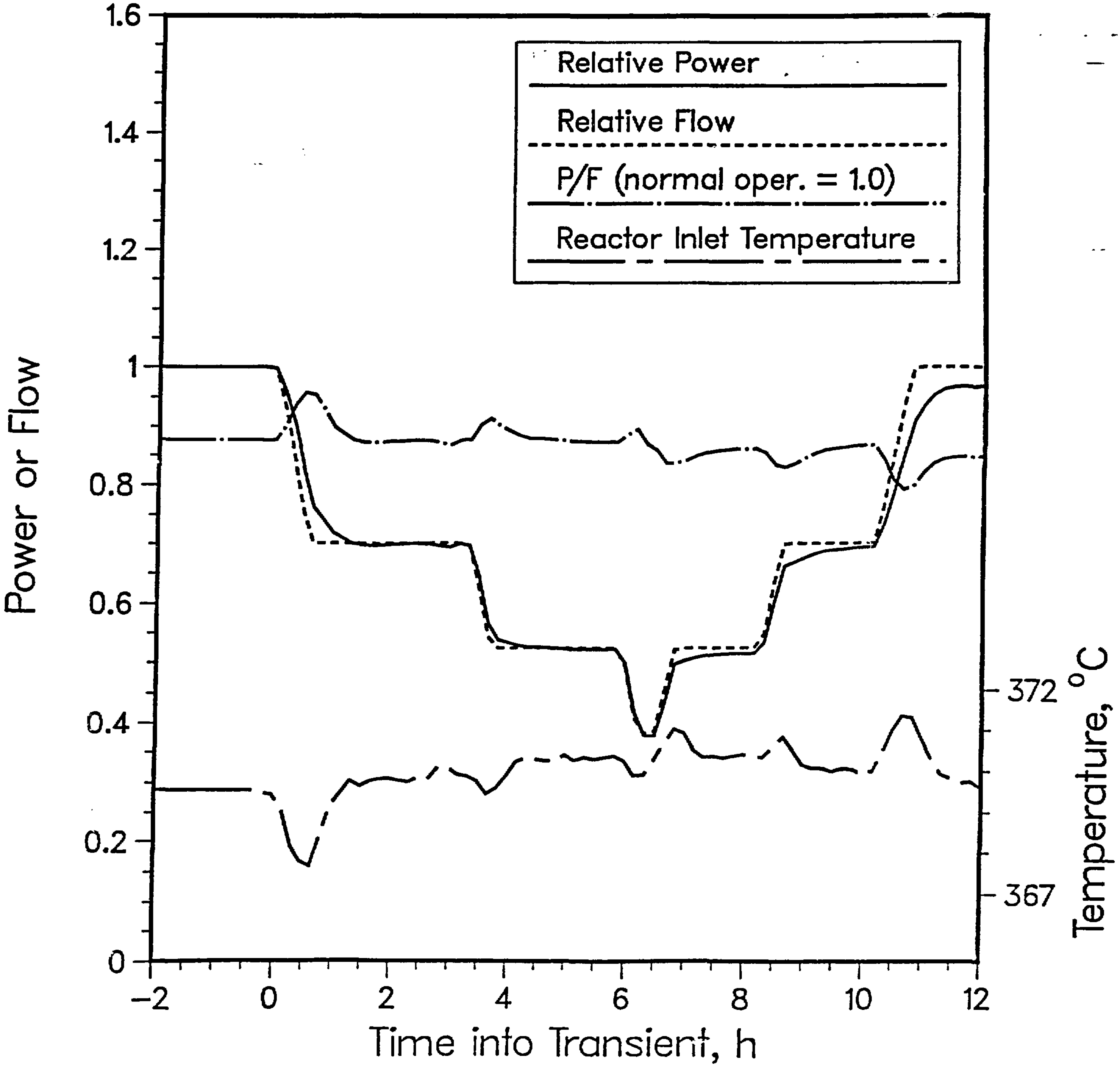

Fig. 2. Normalized Power and Flow, and Reactor Inlet Temperature of PICT 1

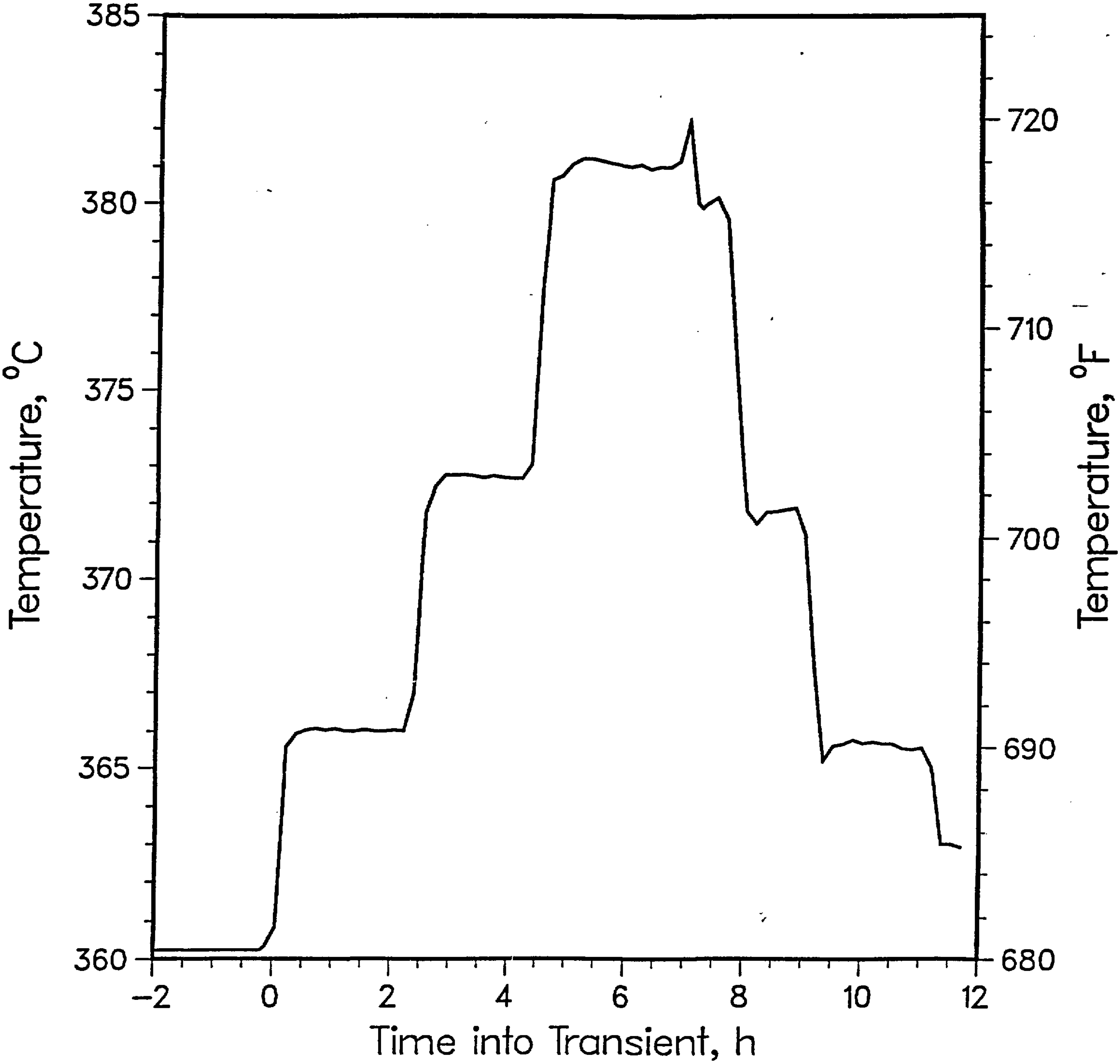

Fig. 3. Inlet Temperature of PICTs 3 and 4

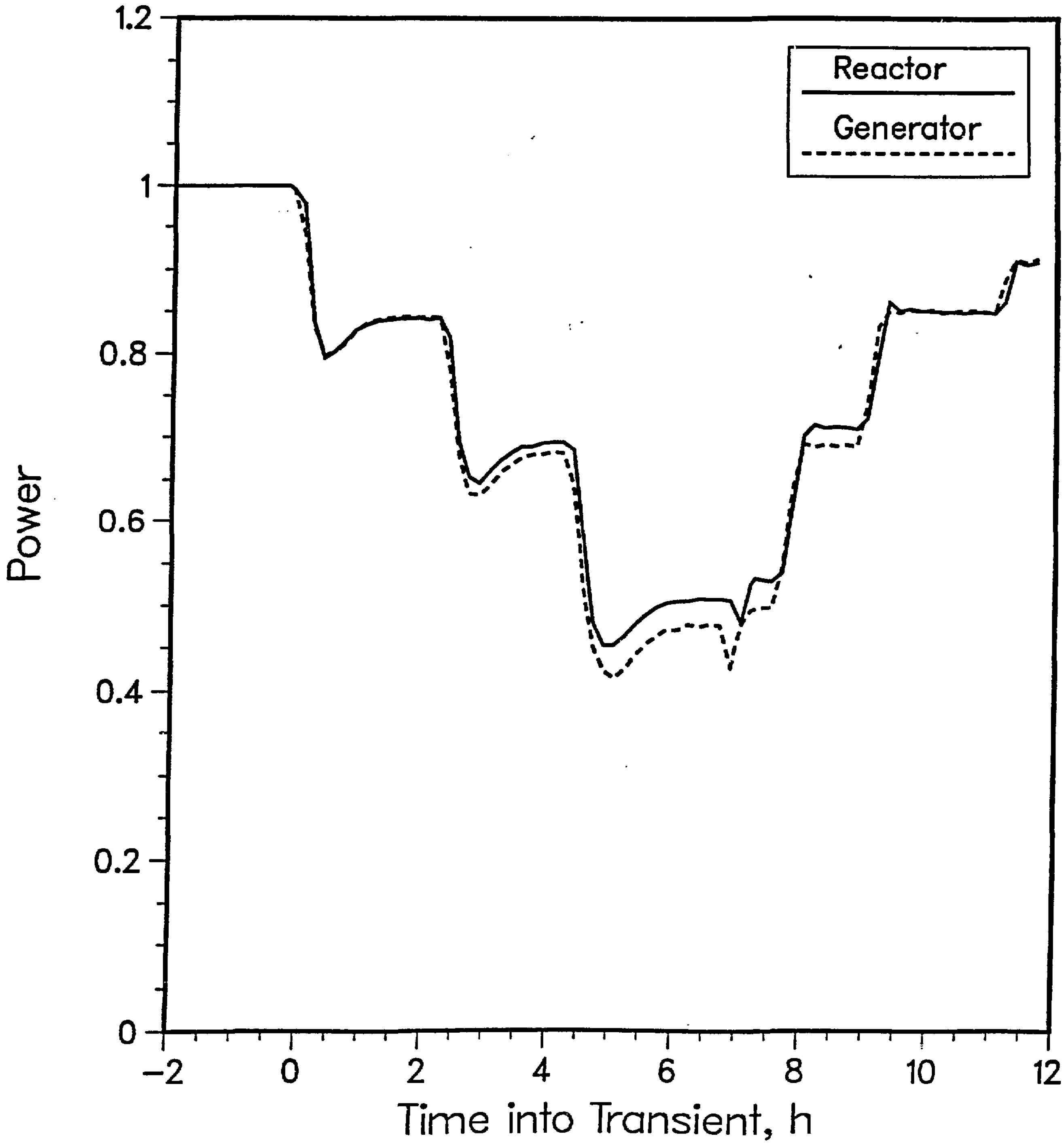

Fig. 4. Normalized Reactor and Generator Power Responses of PICTs 3 and 4