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Demonstration of Passive Safety Features in EBR-II

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ABSTRACT

Two tests of great importance to the design of future commercial nuclear power plants were carried out in the Experimental Breeder Reactor-II on April 3, 1986. These tests, (viewed by about 60 visitors, including 13 foreign LMR specialists) were a loss of flow without scram and a loss of heat sink without scram, both from 100% initial power. In these tests, inherent feedback shut the reactor down without damage to the fuel or other reactor components. This resulted primarily from advantageous characteristics of the metal driver fuel used in EBR-II. Work is currently underway at EBR-II to develop a control strategy that promotes inherent safety characteristics, including survivability of transient overpower accidents. In parallel, work is underway at EBR-II on the development of state-of-the-art plant diagnostic techniques.

INTRODUCTION

On April 3, 1986 two tests of historical importance to commercial nuclear power were carried out by Argonne National Laboratory in the Experimental Breeder Reactor-II (EBR-II). This facility, located at the Idaho National Engineering Laboratory, is a small but complete liquid-metal-cooled reactor (LMR) power plant that produces 20,000 kilowatts of electrical power. The tests showed that natural processes such as thermal expansion of reactor materials and thermal convection of the sodium coolant can shut down the reactor and maintain cooling even if a serious accident were to disable the normal safety systems.

These tests provided valuable information to support the design of advanced LMR concepts that incorporate inherent safety features to greatly enhance the operational safety, and hence licensing process; as well as reduce the capital and operating costs of such plants. Advanced concept LMR plants are being designed by the General Electric Company and the Rockwell International Corporation.

The two tests were a loss of flow without scram and a loss of heat sink without scram. In the loss of flow test the reactor was brought to full power, the normal plant protection system was disabled, and then the primary and secondary coolant pumps were turned off. Inherent physical effects drove the reactor power down to essentially zero without any control rod or operator action. In the loss of heat sink test the reactor was brought back to full power, the normal protection system again disabled, and then the secondary pump turned off, simulating a total loss of heat rejection to the balance of plant. Here the heat retained in the primary system caused the reactor temperature to increase, again causing inherent physical effects to shut the reactor down. The reactor was not damaged by either test.

The EBR-II results are representative of what could be designed into metal-fueled LMRs of all sizes of current commercial interest. Such metal fuel is under development as part of the Integral Fast Reactor (IFR) Program at Argonne National Laboratory.

The IFR Program was discussed in a paper given at the 1986 meeting of the American Power Conference.[1] Subsequently, the experimental IFR driver fuel has attained burnups greater than 100,000 MWd/T, and is on its way to a lifetime burnup of 150,000 MWd/T or greater.[2] Moreover, results have been obtained of transient tests of irradiated IFR fuel in the TREAT reactor. These results demonstrate large margins to cladding breach, which requires powers that are more than four times nominal.

Details of the EBR-II Shutdown Heat Removal Test (SHRT) Program, culminating in the April 3, 1986 demonstration tests, have been reported in the technical literature.[3] Included in these details are:

1. the evolution of thermal-hydraulics testing in EBR-II,
2. modifications of EBR-II to conduct the loss-of-flow-without-scram tests,
3. qualification of the metal driver fuel to conduct the tests,
4. safety analyses to support the conduct of the tests,
5. preliminary results of the loss-of-flow tests,
6. preliminary results of the loss-of-heat-sink tests,
7. post-test analysis of effects of tests on driver fuel, and
8. implications of the tests.

This paper focuses on the fourth, fifth, sixth, and eighth of the above topics.

Figure 1 shows the EBR-II plant. It consists of a primary system and a secondary system, both using liquid sodium as the coolant, and a conventional steam system. The primary system is enclosed in a large double-walled tank, containing about 88,000 gallons of sodium at 700°F under normal operating conditions. The reactor is in a special vessel in the tank. The reactor consists of an array of hexagonal subassemblies 2.290 in. across flats on 2.320 in. centers. The core region extends through row 7, the steel radial reflector through row 10, and the uranium blanket out through row 15. There are two safety rods in row 3 and 8-9 control rods in row 5.

The current driver fuel for EBR-II is a uranium alloy pin, enriched to 66% in ^{235}U, and contained in a type 316 stainless steel tube having an OD of 0.174 in. Free space between the pin and its tubing is filled with sodium as a thermal bond. Each fuel element is wrapped with 0.049 in. dia. stainless steel spacer wire; there are 91 fuel elements in a driver subassembly. The active core height is 13.5 in.

The two primary pumps take their suction directly from the tank and deliver a combined flow to the reactor of 1069 lbs/s at a power of 62.5 MWt. The total flow splits into two streams, one entering the high-pressure plenum that feeds the first seven rows and the other entering the low-pressure plenum that feeds the remaining rows. About 84% of the flow goes into the high pressure plenum. The high and low pressure sodium streams mix in the reactor outlet plenum, go through the outlet pipe to the primary auxiliary pump (a DC electromagnetic device), and then to the intermediate heat exchanger (IHX) in which it transfers its heat to the secondary sodium. The full-power mixedmean temperature of the sodium leaving the reactor is 883°F. The primary sodium leaving the IHX dumps directly back into the primary tank.

The sodium in the secondary system is driven by a single electromagnetic pump at a flowrate of 694 lbs/s. The secondary system heat is transferred to the steam system in seven evaporators and two superheaters, all of the Argonne double-wall heat exchanger tube design.

The steam is used in a conventional turbine-generator to produce electricity. At 62.5 MWt the superheaters deliver 70 lbs/s of steam at 820°F and 1260 psi to the turbine.

Two specially instrumented subassemblies were placed in control rod positions in row 5 of EBR-II for the SHRT Program. One of them, XX09, closely simulated the thermal-hydraulic behavior of a regular driver subassembly and thus played an important monitoring role during the entire test program. XX09 was a 61-fuel element position subassembly; it actually contained driver fuel in 59 of the positions and hollow tubes in the other two positions to carry instrument leads. There were two carefully calibrated flowmeters in tandem in this subassembly, below the core region. In addition, 28 fast-response, RDT standards thermocouples were placed in various positions in the subassembly, including in the flowmeters, at the core midplane, near the top of the core, above the top of the core, and at the subassembly outlet.

RESULTS

A. Loss-of-Flow-Without-Scram Tests[4]

Loss-of-flow-without-scram tests were run from some eighteen different sets of initial conditions. The objective was to identify the main parameters determining the plants' ability to passively shut down if failures reduced or eliminated forced circulation of coolant. As shown in Table I, tests were run from 16% to 100% initial power with pump coastdown times (the time from initiation to pump stop) ranging from 19 sec to 100 sec. Other parameters that were varied included the initial primary flow and the conditions of the secondary and auxiliary pumps.

Test 45 resulted in the highest fuel temperatures. It was run from 100% power in such a way to simulate a station blackout (loss of off-site and diesel emergency AC power) without scram. The test was conducted by 1) placing the auxiliary pump on its battery power supply (simulating a loss of normal and emergency AC power), 2) bypassing the normal loss of flow scram function, 3) inserting a special high-temperature scram (just in case there were equipment failures or operator errors during the test) and 4) tripping AC power to the main coolant pumps.

The response of the reactor plant is summarized by the measurements shown in Fig. 2. As shown, the reactor flow coasted down with the main coolant pumps. At about 100 sec the pumps stopped, and thereafter the flow was provided by natural convection aided by the auxiliary pump. The reduction in flow caused a rapid increase in the reactor and coolant temperatures, which in turn produced negative reactivity feedback. The negative reactivity reduced power and was effective in limiting temperature overshoot. The initial negative reactivity which reduced power and "turned the temperature" was largely proportional to the core ΔT, as can be seen in both Figs. 2 and 3. This features is typical of larger metal fueled LMRs. Most of the negative reactivity came from thermal expansion of the core structures, sodium, and control rod drivelines.

The reactivity feedback from the fuel was smaller than those from the sodium and structure. This is the case in metal fueled LMR's because of the high thermal conductivity of the metal fuel (compared to the low conductivity of oxide fuel), which results in relatively low fuel-to-coolant ΔT and, consequently, small Doppler and fuel expansion feedback. The high thermal conductivity of the metal fuel also tends to link the fuel temperature to the coolant temperature and decouple it from the power. Consequently, as shown in Fig. 3, the initial reduction in flow results in a fast acting increase in the fuel temperature and, as also shown in Fig. 3, a quick contribution of negative reactivity. Then as the power is reduced at larger times the positive reactivity from the collapse of the fuel-to-coolant ΔT is relatively small.

There was good agreement between the pretest predictions and measurements in all the LOFWS tests. Figure 4 shows the temperature predictions and measurements for test 45. The measurements were in the instrumented in-core subassembly XX09 at a position near the top of the core. The predictions were made with the NATDEMO and HOTCHAN computer codes. The maximum and minimum curves included nuclear, thermal, and hydraulic uncertainties as well as uncertainties in reactivity feedback. The "max hot driver clad" curve is the predicted fuel-clad interface temperature for the hottest pin in the core with all uncertainties included. As such it was the safety envelope for the test.

The pump coastdown rate is one of the most important parameters affecting the peak temperature during the LOFWS. Analysis backed by test results show that the flow decrease must be sufficiently gradual that the negative feedbacks have time to reduce power. These findings are summarized in Fig. 4, which shows core temperatures for three LOFWS tests which had pump coastdown times of 100, 300, and 600 sec. All three of the tests were run from 100 percent power. The temperatures were measured in XX09 at the top of the core.

As suggested by the data in the figure, the specification of the pump coastdown time and shape during the design of a plant has important consequences. The desired coastdown time is long compared to nuclear and thermal-hydraulic time constants. A desired coastdown is attainable by tailoring the inertia of the rotating parts of the pump, its motor, and possibly the pump-drive generator. The ability of a pump manufacturer to deliver a pump with an as-specified coastdown was demonstrated in the design and manufacture of the CRBRP and FFTF pumps. In EBR-II the most significant time delay in reducing power is caused by the delayed neutrons. Since the delayed neutron fraction varies by a factor of 2 between uranium and plutonium fuels, the designer should consider fuel type, as well as feedbacks and thermal time constants that influence the feedbacks when specifying a pump coastdown time. This is particularly important if the initial core loading uses ^{235}U, and there is a progressive transition to an equilibrium ^{239}Pu core.

The peak core temperatures were low compared to the operational and safety limits for the fuel. The fuel temperatures, with uncertainty, for all the tests except test 44 and test 45 were below the temperature limits for our most common and least severe transients--anticipated events. The conservative Technical Specification limit for the fuel-clad interface temperature in an anticipated event is 1319°F. Below this temperature there is no clad damage and no limit on continued operation. Operation at higher temperatures is limited because of the possibility of formation of fuel-cladding eutectic, creep rupture, and increased fission gas pressure. Therefore, in order to conduct tests 44 and 45, a less conservative time-at-temperature correlation was required. In these tests, the temperatures in a few fuel assemblies were predicted to exceed 1319°F for a short time. The correlation was developed based on accumulated in-reactor and out-of-reactor experimental data and was verified prior to tests 44 and 45 with an in-reactor-test-until-failure.[5]

In summary, the peak temperatures were below acceptable limits for the LOFWS tests. For moderately long pump coastdowns, the temperatures were less than the fuel-clad eutectic temperature--the conservative limit for an anticipated event. For more rapid coastdowns, the time above eutectic temperature was short, and no fuel failures were observed. In fact, after conducting the LOFWS tests, the fuel was used until it reached its normal burnup limits; thus confirming that there was no significant fuel damage caused by the tests.

B. Loss of Heat Sink Without Scram Tests[6]

The loss of heat sink without scram tests were run from 50% and 100% power. In both cases a worst case test was conducted by stopping secondary flow, thus essentially blocking all heat transfer from the primary pool to the secondary loop and steam system where electricity is generated. The primary flow was kept at its initial value during the tests. No automatic scram was used to shut down the reactor. The tests were designed to study the passive shutdown following the many different equipment failures or errors which can occur in the balance of plant and which can result in a reduction or complete loss of the ability of the balance of plant to accept heat from the reactor. Examples of loss of heat sink accidents are loss of feed water, loss of forced circulation in the secondary loop, and loss of a controller in the feedwater train or secondary heat transport system.

The final test was conducted from 100% power by first blocking the automatic initiation of the shutdown coolers. These units are NaK-air natural circulation that provide long term cooling of the EBR-II sodium pool. The air side dampers automatically open at slightly above normal tank temperatures.

Then the flow in the secondary loop was stopped by tripping electrical power to the secondary pump and applying power with a reverse voltage to the pump to counter the natural circulation head in the secondary loop. It was not necessary to bypass any of the scram functions to conduct these tests because the reactor feedbacks reduced the power and temperature, and the thermal margin to scram actually increased during the test.

Stopping heat transfer to the balance of plant caused an increase of the primary system temperature. As shown in Fig. 6, the rate of increase of temperature was gradual, considering that full reactor power was initially retained in the primary system. The rate of temperature increase was mitigated by the heat capacity of the large sodium pool. The increasing core inlet temperature produced negative reactivity--first from thermal expansion of the lower reflector and then from the expansion of the core support structure. The negative reactivity led to the reduction in reactor power also shown in the figure. The power reduction was sufficient to reduce the reactor outlet temperature and prevent any significant temperature overshoot. An increase in reactor inlet temperature of about 75°F shut down the reactor. The physics of the passive shutdown can be illustrated with the following "back of the envelope" calculation. In EBR-II the power reactivity decrement (the total feedback reactivity between zero power hot critical and full power hot critical) is about 30¢. The reactivity change due to heating the whole reactor 1°F (the inlet temperature coefficient) is about 0.4¢/°F. The quotient, (30/0.4 = 75°F) is the temperature increase at the inlet that will balance the PRD and reduce power to zero. This is sometimes called the quenching temperature.

The relatively small quenching temperature is typical of metal fueled reactors. The dominant factor in determining the low quenching temperature is the high thermal conductivity of fuel. As previously discussed, this results in a small Doppler reactivity and a small PRD.

PASSIVE SAFETY PARAMETER CHARACTERIZATION TESTS

It seems likely that if future reactor plants are to depend on passive accomplishment of the shutdown and decay-heat removal safety functions, then the adequacy of these passive safety features will have to be firmly established by acceptance tests. Further it seems prudent to monitor the passive safety features from time to time during the plant lifetime. The parameters which support passive safety are expected to be more stable than the characteristics of active safety systems--particularly electronic equipment. However, some parameters important to passive safety, notably reactivity feedback coefficients and pump coastdown times, do change with core loading, fuel burnup, and equipment operation. Therefore, a new approach to acceptance testing and monitoring passive safety performance appears to be required. While further work is necessary to develop this area, some of the testing methods developed for the EBR-II tests appear to be applicable.

At the outset of the testing program it was realized that additional surveillance and testing was needed to supplement the standard tests required by our Technical Specifications. Experience had shown that the feedback changed with core loading and burnup and that the EBR-II pump coastdown times vary with several operational factors. The approach we used was to extend the normal startup testing. Typically the total static power reactivity decrement is measured of during each startup. If the measured PRD varies less than 10% from the baseline then we know that the baseline, safety and stability analysis for the reactor are valid. On the other hand if there is a significant change, then rod drop tests are conducted to measure the dynamic components of feedback and thereby assure adequacy of baseline reactor safety and stability analysis.

To make sure there is adequate feedback for a passive shutdown, more information is required. The criteria must include the components as well as the total PRD. As previously explained, for a LOFWS, the component of the PRD which is proportional to reactor ΔT is very important. This can be determined by perturbing the reactor flow rate. Likewise for a LOHSWS, the inlet temperature coefficient is important. This can be determined by perturbing the inlet temperature.

At the start of the testing program a series of reactivity characterization tests were run. These tests and how the measurements were used to validate the NATDEMO reactivity models and data were described by Mohr and Chang[7]. The validated models and data were used in the safety analysis for all the subsequent tests. The characterization tests were repeated prior to each test window to verify that the data on which the safety analysis was based was adequate.

The three reactivity characterizations that were run prior to each test period were 1) a detailed power reactivity decrement test, 2) a primary flow perturbation test and 3) an inlet temperature perturbation test. Additionally, a flow coastdown test was run prior to the startup for each loss of flow without scram test run from high power.

The power reactivity decrement is measured in EBR-II by establishing a constant 100% reactor flow rate, a constant reactor inlet temperature and then during the rise to full power measuring rod position (reactivity) as a function of power level. Figure 7 shows the results of a PRD measurement made prior to the first test window. Note that the reactivity is nearly a linear function of power or reactor ΔT except for a nonlinear deviation starting at about 75% power. This non-linearity, which is believed to be caused by thermally induced bowing of the core subassemblies, could significantly alter the reactivity feedback. Since significant variations in the PRD have been observed over years of EBR-II operation, the PRD data was examined to assure that the measured total feedback compared closely to that used in pre-test analysis.

The flow perturbation test measured the component of the PRD which is proportional to the core ΔT. As discussed in Section II, this component is important in limiting temperatures for the LOFWS. The flow perturbation test was conducted from about 70% power and flow by rapidly increasing primary flow by about 30% while keeping reactor inlet temperature constant. Power and temperature were allowed to freely respond to the change in flow as shown in Fig. 8, reactor feedback acts to increase reactor power. Since a major part of the PRD is proportional to reactor ΔT (power-to-flow-ratio) and a minor part is proportional to fuel-to-coolant-ΔT (power), the power increases to nearly match the flow and tends to keep the reactor ΔT constant.

The inlet temperature perturbation test results are shown in Fig. 9. This test determines the reactivity coefficient of inlet temperature. The test was run by changing the reactor inlet temperature (by varying the heat rejection with secondary system flow rate) while keeping primary flow rate constant. The reactor power and outlet temperature were allowed to freely respond. The power decrease for a given change in inlet temperature in this perturbation test is directly related to the quenching temperature for a complete loss of heat sink, which was discussed in the previous section.

In summary, at the start of the testing program extensive measurements were made from which reactor plant models and their data were verified. Analysis for subsequent tests were done with these models. Prior to each subsequent test period the reactivity characterization tests were repeated to verify that the reactivity feedback (total and individual components) was sufficient to run the tests. In most cases simple measurements of the static components of feedback and total pump coastdown time clearly indicated the adequacy of the model and its data. However, on occasion the tests detected problems-particularly with pump coastdown shape-that further dynamic testing and analysis solved prior to conducting the LOFWS tests. The approach worked well, as evidenced by the excellent agreement between pretest predictions and measurements. It is believed that a similar approach could be used in future LMR's to verify reactor/plant parameters that are necessary for passive shutdown safety.

DISCUSSION

The EBR-II Shutdown Heat Removal Testing series has shown that LMR plants can rely on passive accomplishments of some of the critical safety functions. Passive shutdown was shown for eighteen loss-of-flow-without-scram tests. Natural circulation was shown to effectively remove shutdown heat. Reactor plant design features which promote the passive shutdown include reactivity feedbacks largely proportional to the coolant temperature (attainable with high conductivity metal fuel) and moderately long pump coast-downs. The test results support the current approach of US designers which include passive shutdown for loss of flow accidents and passive natural convection decay heat removal.

Passive shutdown was also shown for a sudden loss of heat sink. The tests here bounded many balance of plant transients such as a loss of feedwater with no scram. The passive shutdown was facilitated by the reactivity feedbacks indigenous to metal fuel and heat capacity of EBR-II's sodium pool. The results of these tests support the design choice to have a non-safety grade balance of plant.

In all the tests, the core temperatures were mild and, even though the transients were extreme, there was no damage to the fuel or reactor. The predicted fuel temperatures agreed well with measurements.

An important part of controlling the risk of the tests was monitoring parameters which were critical to the passive shutdown. Integral reactivity feedbacks and pump coastdown times were monitored and verified prior to conducting the tests. An extension of this approach could be used to verify on a continuing basis the viability of passive safety features in an operating LMR.

The analysis and testing have identified some open issues in "inherent safety". If the promise of a broad base of safety functions being accomplished by passive means is to be realized, these open issues must be addressed in design and verified by test. One open issue is the transient overpower class of accidents. Operational accidents in this class could be caused by a rod withdrawl or excessive power demand from the secondary or steam system. Work is underway to develop and prove passive means of mitigating the effects of overpower accidents. A second open issue is the possibility of an inherently safe response being overridden by action of a non-safety-grade control system. During our tests, we deenergized the control rod normal drives to preclude an inadvertant power control which could have overridden the passive shutdown. Analysis has also shown that malfunctions of pump or steam system controllers can also hinder passive shutdown. This suggests that the architecture and failure modes of the control system must be carefully considered if broadly based passive safety is to be achieved.

As part of the future test program at EBR-II, techniques for measuring important parameters necessary to insure inherently safe responses on LOF, LOHS, and TOP will be further developed to allow on-line measurements. This work will include development and demonstration of the analytic techniques and associated software.

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From French Paper

TABLE 1. Summary of Loss of Flow Without Scram Tests

Test	Initial Power, % of Rated	Initial Primary Flow % of Rated	Primary Pump Coastdown Time (sec)	Secondary Pump	Auxiliary Pump	Predicted Peak Cladding Temperature of Fuel Assembly (with uncertainty)
27	16.7	19	85	on	on	613
28	16.7	19	85	tripped	on	618
29	16.7	19	85	on	off	657
30	16.7	19	85	tripped	off	677
31	16.7	19	19	tripped	on	657
32	16.7	19	19	tripped	battery	679
33	50	100	300	400 s. coastdown	off	585
35	50	50	300	400 s. coastdown	off	625
36	50	50	300	tripped	off	625
37	100	100	600	tripped	off	604
38	100	100	300	400 s. coastdown	off	652
39	100	100	300	tripped	off	672
40	50	100	95	tripped	battery	635
41	50	100	95	tripped	off	622
42	50	100	200	tripped	off	676
43	70	100	95	tripped	battery	713
44	90	100	95	tripped	battery	774
45	100	100	95	tripped	battery	802

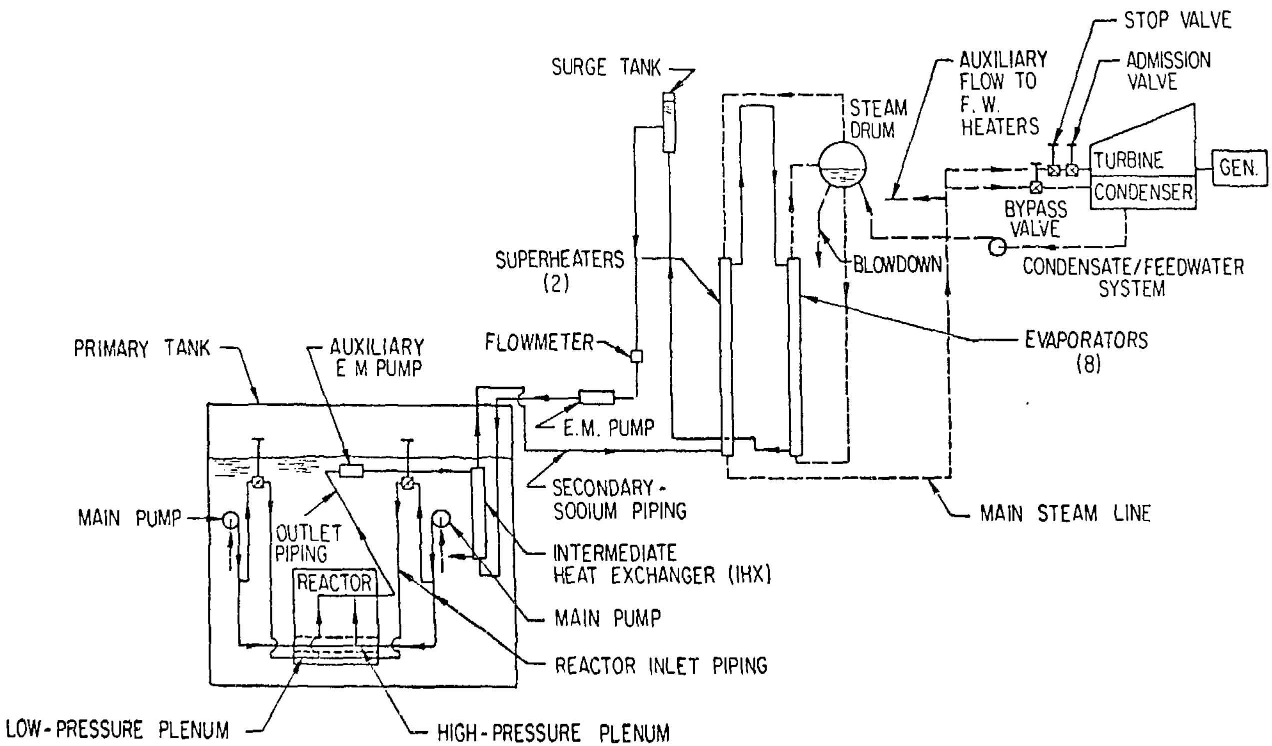
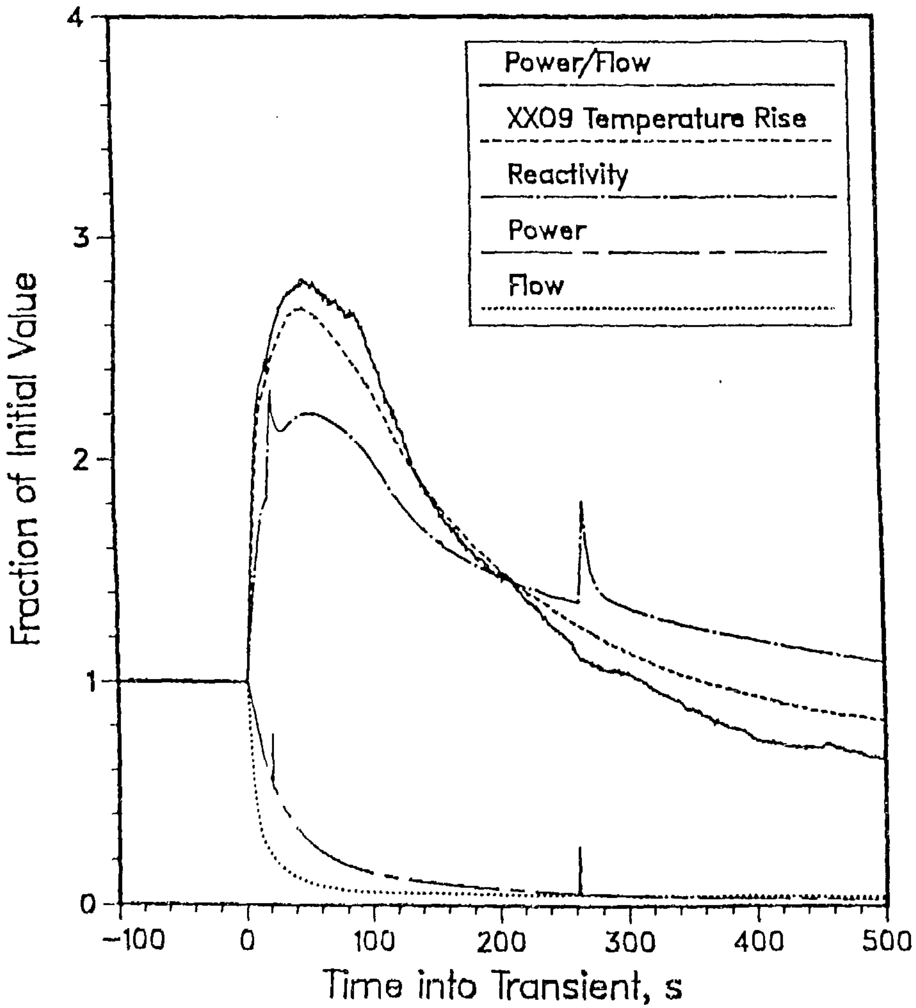


Fig. 1. EBR-II Plant

Fig. 2. Comparison of Power to Flow Ratio, Core ΔT and Reactivity Ratio for Loss ~~of~~ Flow Without Scram. Test ~~39~~ 45. Pump Stop Times ~~300 and~~ 100 ~~sec.~~ Initial power 100%. Auxiliary pump ~~off~~ on Battery Power.



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Reactivity Change from Full Power

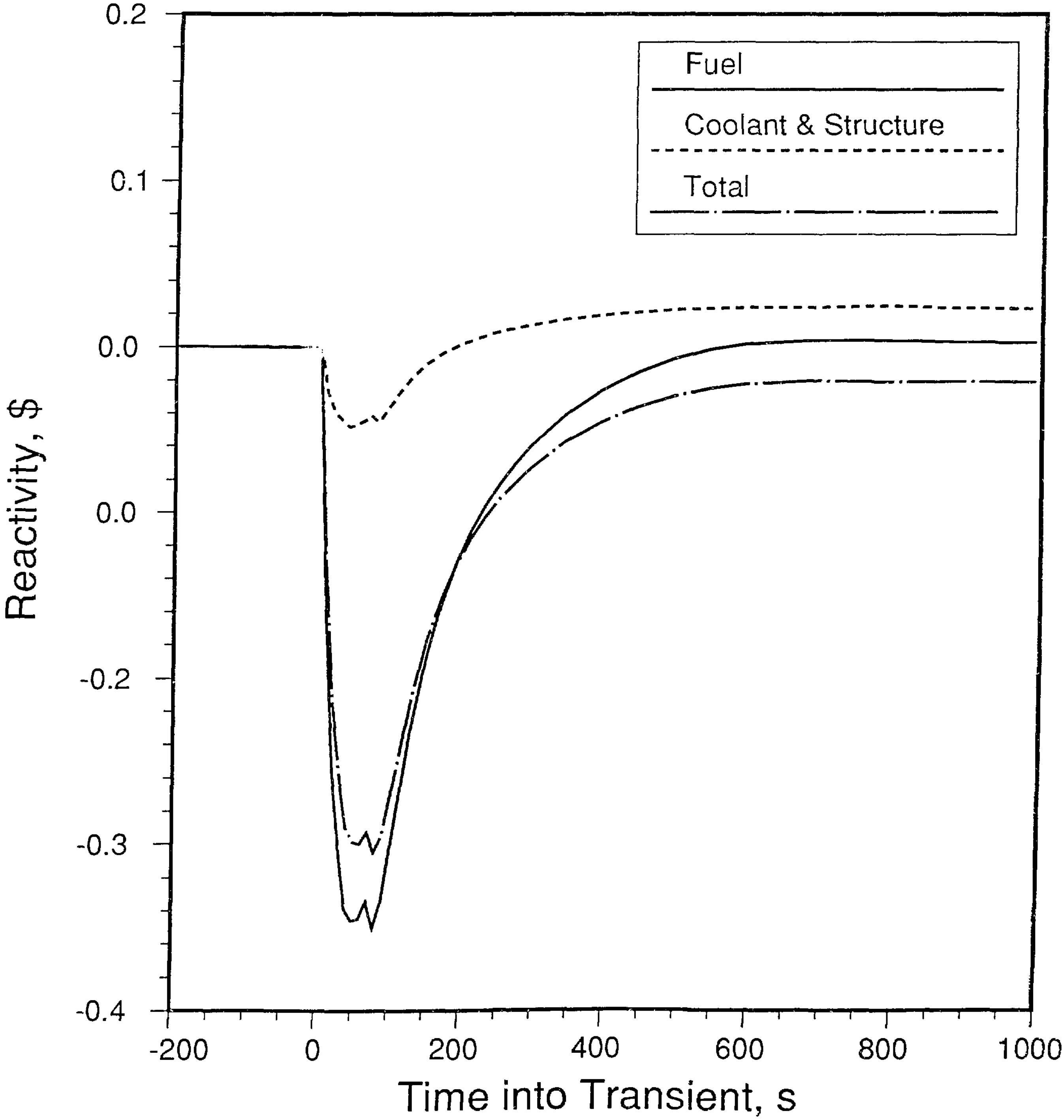
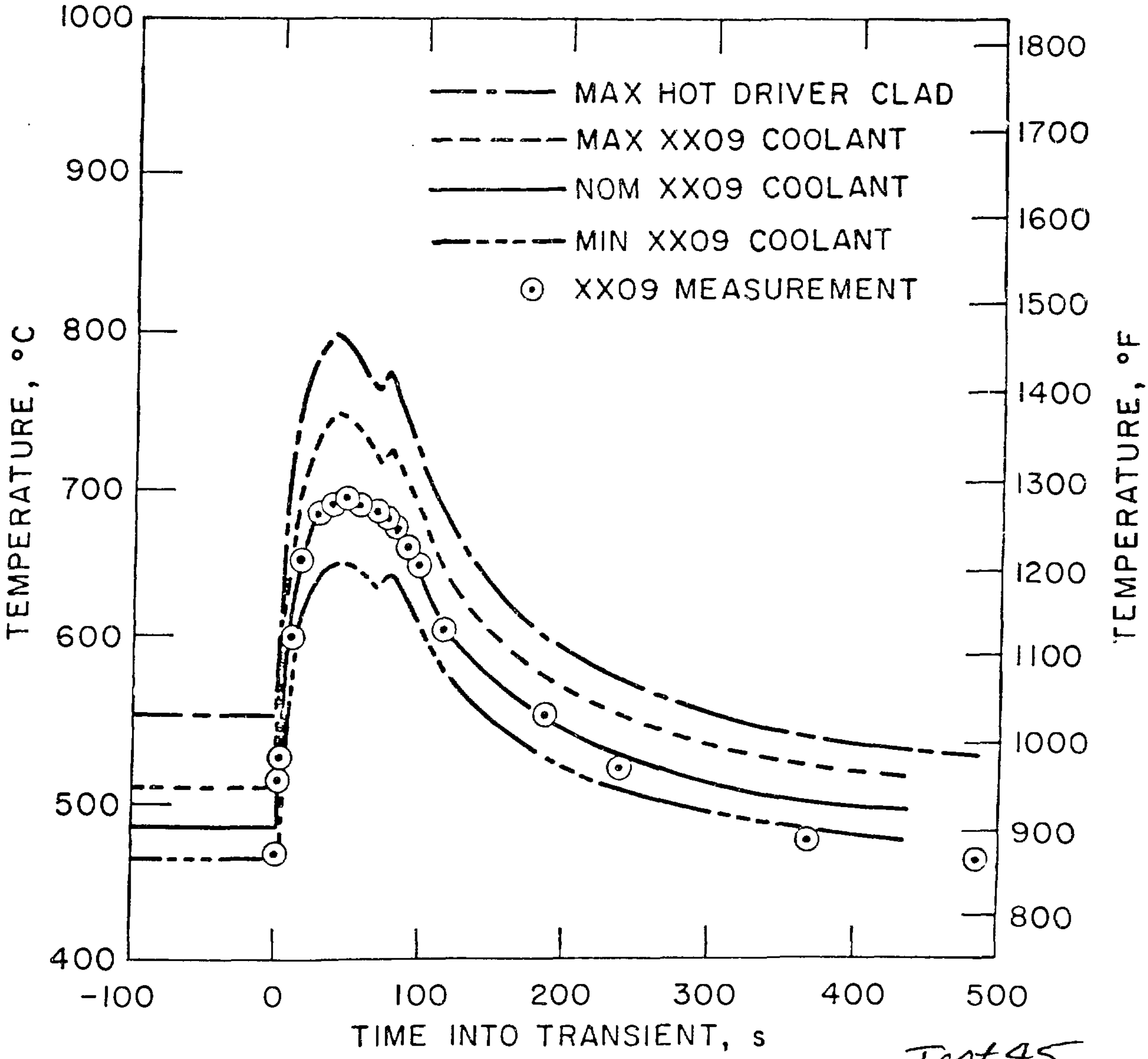


FIG 5 Reactivity Change from full power.



Test 45.

FIG Loss of Flow Without Scram. Simulation of Station Blackout from 100% power. 100 sec pump coastdown. Pretest Predictions and Measurements of In-Core Temperatures.

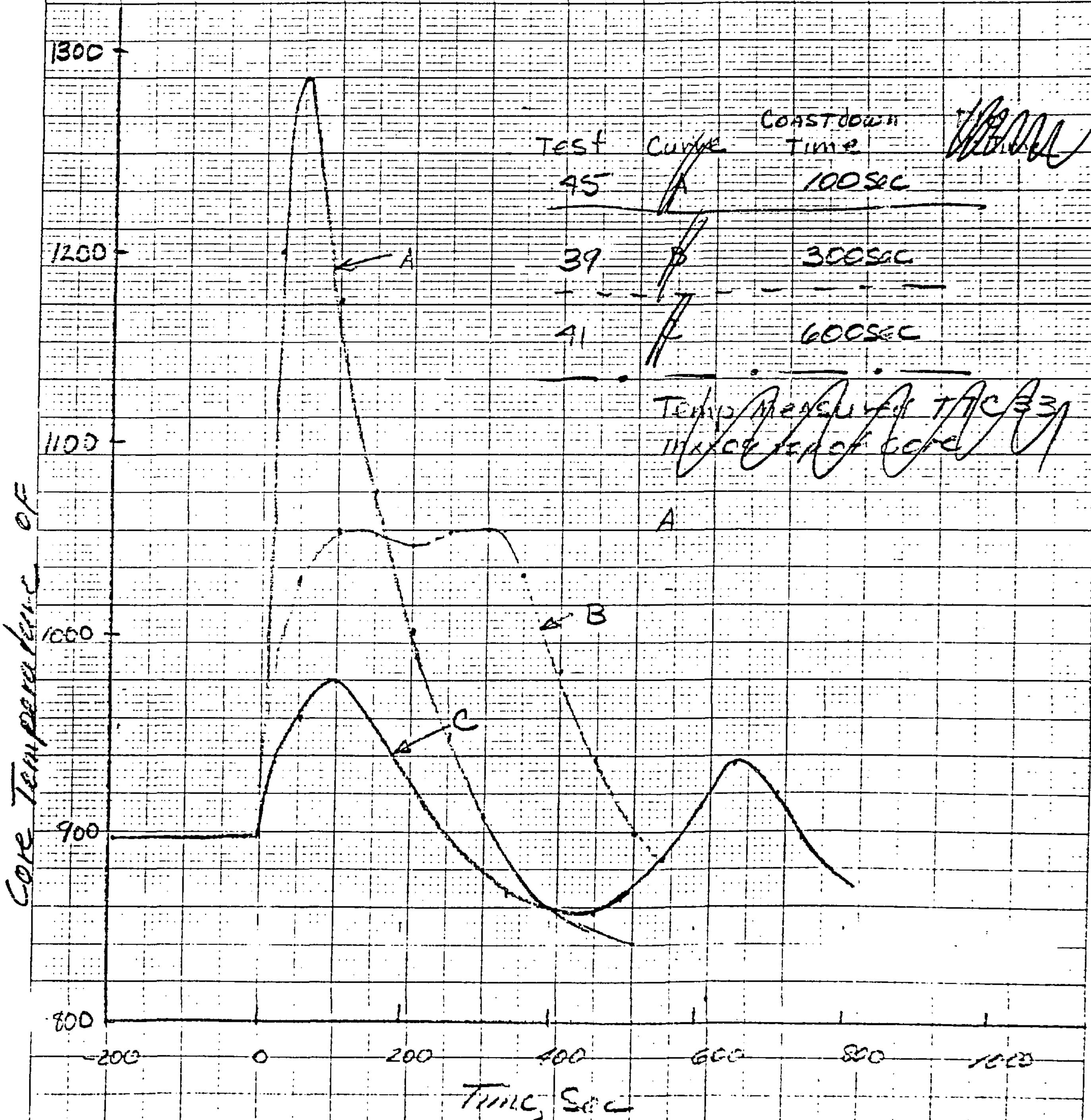


FIG 5 Comparison of Measured Core temperatures for Loss of Flow without scram tests with different Pump Coastdown times.

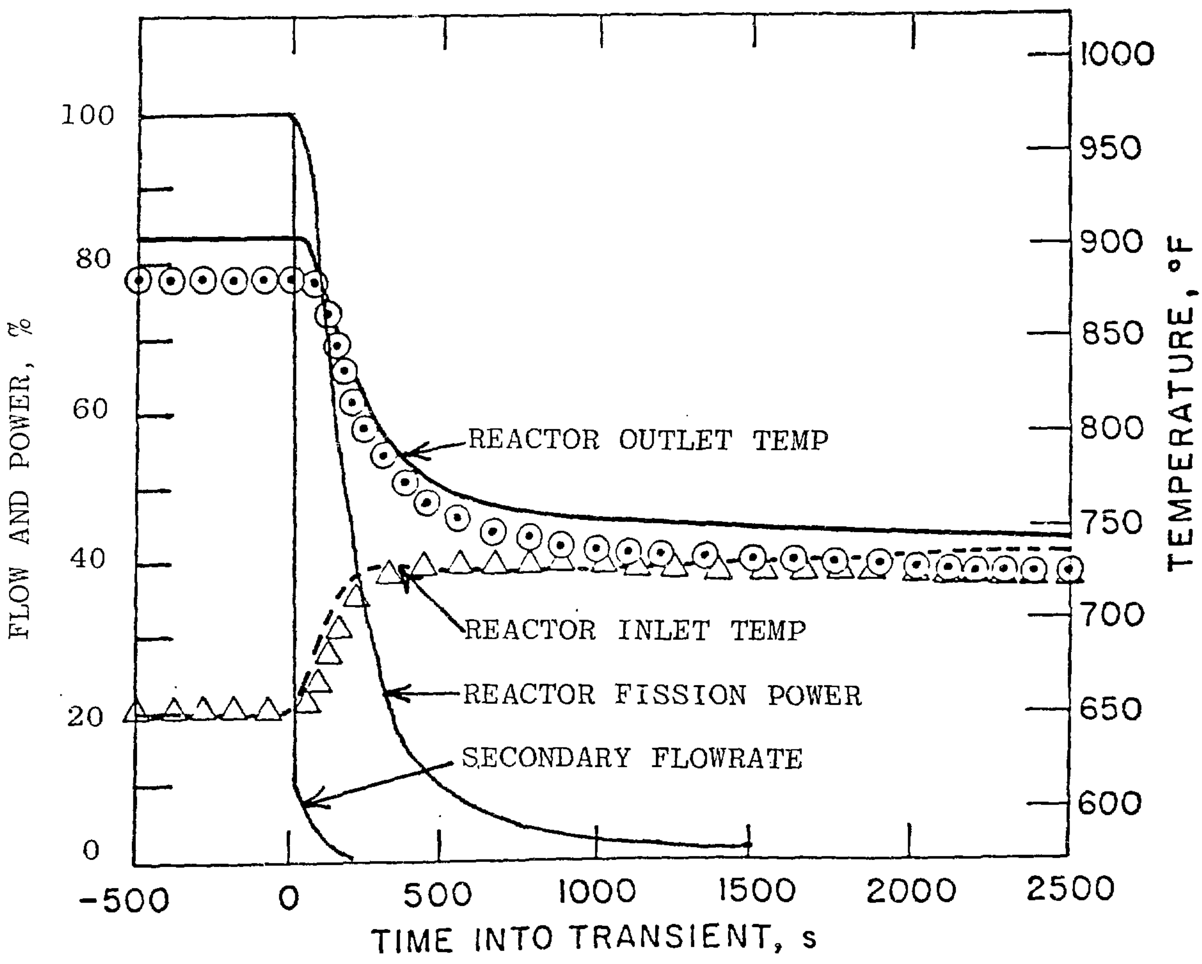


Fig. 6. Loss of Heat Sink Without Scram from 100% Power (Test B302). Pretest Predictions and Measurements of Temperatures. Measured Power and Secondary Flow.

EBR-II PRD (RUN 129)

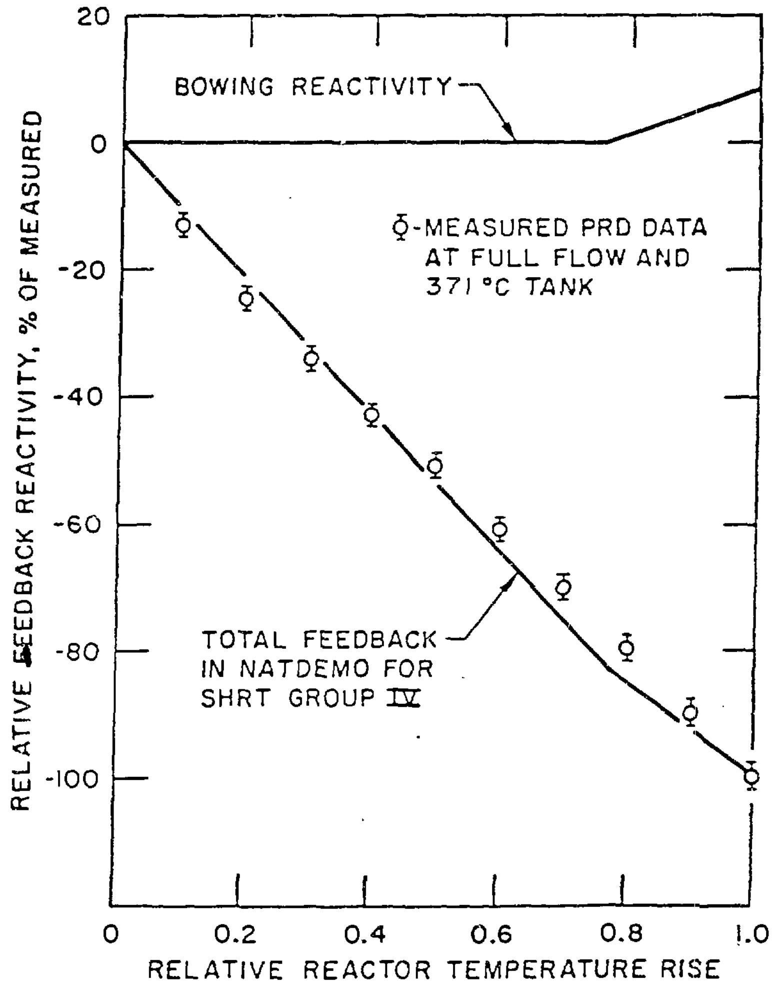


FIG ~~6~~ 7 Measured Power Reactivity Decrement Measured In EBRII. Constant Inlet Temperature and Primary Flow. Run 129.

~~EBR-II RESPONSE TO FLOW PERTURBATION (SHRT 25)~~

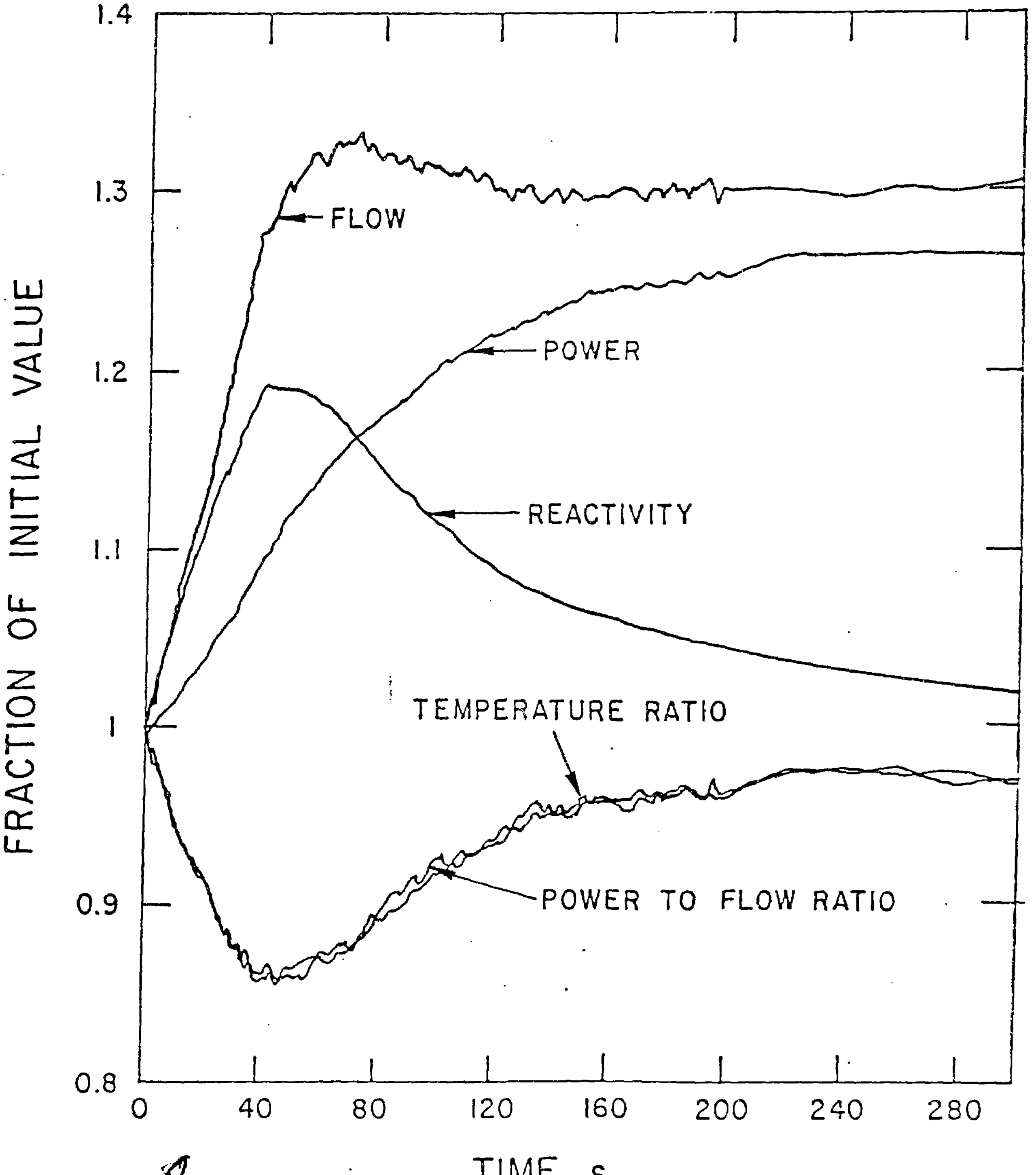
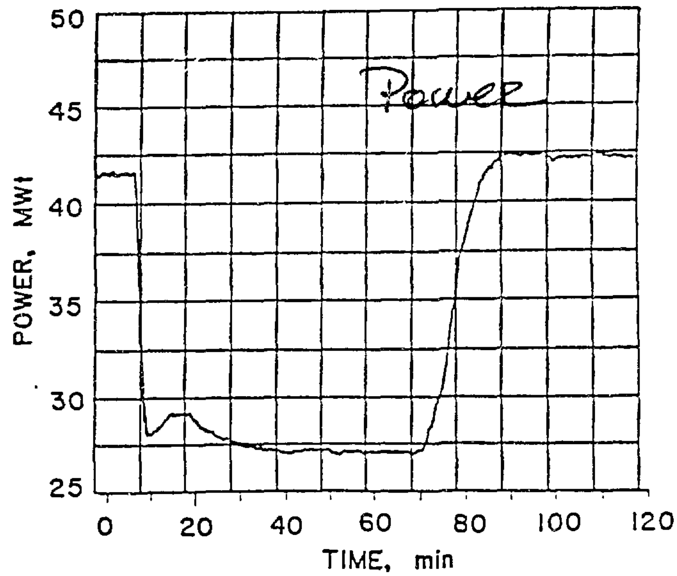
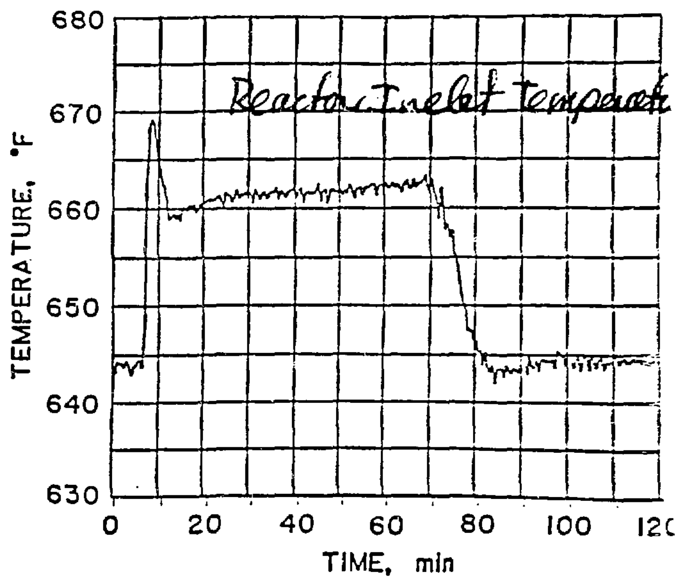


Figure ~~17~~ 8 EBR II ~~Response to~~ Reactivity Characterization, Flow Perturbation ~~Test~~.
Test 25. Constant Inlet Temperature.

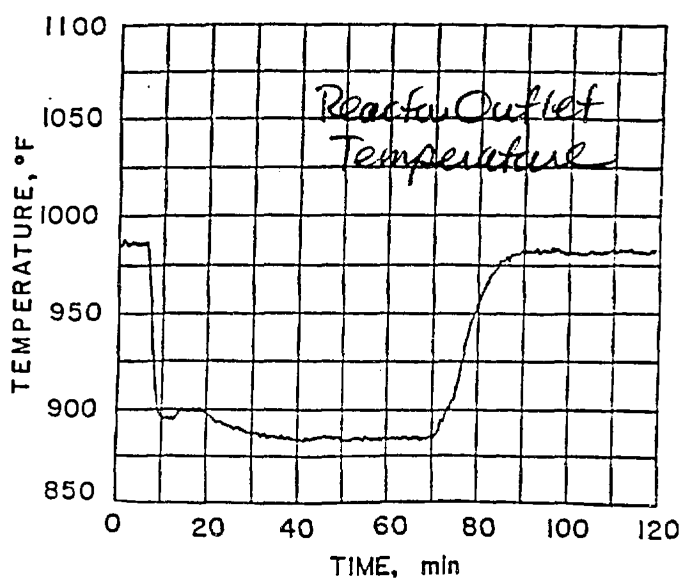
Fig ~~8~~ 9 Reactivity Characterization - EBR-II ~~RESPONSE TO~~ INLET TEMPERATURE PERTURBATION Test. Test 26
Constant Reactor Flow



~~Fig. 12. Reactor Power Measured During SHRT 26~~



~~Fig. 11. Reactor Inlet Temperature Measured in the Instrumented Subassembly XX09 During SHRT 26~~



~~Fig. 13. Top of Core Temperature Measured in XX09 During SHRT 26~~

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