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SUMMARY

In this paper, the complete testing program at EBR-II directed towards transient thermal-hydraulic performance is described. The program, which was initially directed towards an understanding of the dynamics of natural convective flow and the validation of related computer codes, has evolved into studies of unprotected transients. These later tests are intended to provide experimental data as well as to directly demonstrate the inherently safe response of an LMR to transients which in the recent past were thought to lead to core disruption. Typical results and conclusions from the series of protected natural circulation tests are also presented.

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INTRODUCTION AND BACKGROUND

In support of the continued safe and reliable operation of EBR-II as well as to contribute to the design and performance assessment of other liquid-metal-cooled reactors, an experimental and supporting analytical program has been and is continuing to be conducted utilizing the EBR-II plant. These efforts, which essentially started in 1974, have been primarily directed towards understanding the detailed response of EBR-II to a wide variety of upset conditions and utilizing this knowledge to validate general purpose thermal-hydraulic-neutronic computer codes. Initial emphasis was placed upon reactor and primary heat transport system phenomena, and more recently, the focus of the work has been on whole-plant dynamic behavior. The success of this program has been immeasurably aided by the availability of fully-instrumented and calibrated in-core fueled and non-fueled assemblies, XX07, XX08, XX09 and XX10 [3]. These assemblies, with their extensive temperature and flowrate measuring capabilities, have permitted the generation and documentation of comprehensive data sets that have been used to validate codes modeling single and multiple assemblies, and whole core behavior. Plant instrumentation has been upgraded so that flowrates and temperatures in the primary, secondary, and steam systems can be measured. Additional control systems have been or are in the process of being added to facilitate the conduct of whole plant dynamic testing. In this paper, we will summarize the evolution of these efforts which started from mild steady-state natural circulation tests and will culminate with unprotected (no scram) transients involving total losses of forced flow or normal heat sinks. In addition, some of the applications of these data in code validation studies will be described.

The in-core instrumented assemblies XX07 and XX08 have been previously described in [1,2] while XX09 and XX10 (the current installed units) were discussed in [3]. However, due to the importance of these instrumented probes to the conduct and interpretation of the testing, it is useful to briefly summarize their main features. The fueled assembly, XX09, contains 61 elements, 59 of which are Mark-II metal fuel and 2 serve as hollow conduits for below-core instrumentation leads. There are 28 thermocouples measuring the three-dimensional temperature field throughout the assembly, ranging from below core to assembly outlet, within the fueled region, and in the inter-assembly bypass flow region. Two permanent-magnet flowmeters are located in tandem within the assembly below the core and have been calibrated over a flowrate range covering rated conditions down to both upward and downward natural convective flow (i.e., from -0.3 to +3.2 ℓ/s. The fuel element O.D. is 4.42 mm, and the wire wrap diameter and pitch are 1.24 mm and 152. mm, re-respectively. Its normal operating conditions are 468 kW (25.6 kW/m, peak), 3.14 ℓ/s, and 136°C temperature rise.

In XX10, there are 19 elements, 18 of which are solid stainless steel (type 316) and 1 serving as a hollow conduit for below-core instrument leads. As with XX09, there are 2 below-core permanent-magnet flowmeters calibrated over the expected flowrate range from -0.04 to +0.44 ℓ/s. A total of 26 thermocouples are included providing full coverage from below-core to assembly exit regions. The particular choice of materials and dimensions used for XX10 was based upon standard thermal-hydraulic scaling laws so that the relative dynamic performances of XX09 and XX10 would closely approximate that of fueled and blanket assemblies in a large reactor [4].

Previous testing experience. The experimental program at EBR-II has of necessity been evolutionary in nature, due both to the need of conducting the mildest tests first in order to avoid any potential risk to the on-going operation of the plant, as well as to the learning process which generated new tests specifically designed to answer questions or resolve uncertainties arising from completed tests and analyses. Accordingly, the emphasis of the early tests (with XX07 installed) was on steady-state natural circulation under conditions of either decay or fission power [5]. The post-test analyses not only permitted the EBR-II system simulation code NATDEMO [6] to be validated for low flowrate conditions, but perhaps more importantly, quantified for the first time the strong inter-assembly flow redistribution that occurs during natural circulation. This phenomenon, which was not modeled in the then available core thermal-hydraulic codes, resulted in a substantial decrease in the cross-core (as well as intra-assembly) temperature peaking factors used in natural circulation safety analyses [7].

The only transient test conducted with XX07 installed, which was a loss of auxiliary pump forced flow at hot-standby conditions [2], demonstrated that the primary coolant flowrate could decrease to a very low value after the cessation of forced flow and prior to the development of buoyantly-driven flow. This observation lead to the possibility of total flow reversal in cooler regions of the core during a transition to natural circulation for certain types of loss of forced flow transients.

Accordingly, following the end of the useful life of XX07, the program was continued with a new instrumented assembly XX08, and emphasis was placed upon the dynamics of the transition from forced to natural convective flow from a wide variety of initial operating conditions and sequencing of scram, primary and secondary pump trip times. Loss of forced flow transients were conducted from full primary flow at hot-standby conditions, from auxiliary pump flow at decay power levels, and from various reduced levels of fission power and primary pump flow. The analysis of these test results clearly demonstrated that transients initiated from conditions in which relatively normal temperature gradients existed in the system (i.e., power-to-flow ratios close to unity) lead to a smooth and benign transition to natural circulation [8] However, when the transient imposed resulted in a significant degradation of the thermal head in the primary circuit prior to the loss of forced flow, a substantial and temporarily sustained total reversal of flow occurred in the lower power regions of EBR-II (i.e., the radial blanket and reflector regions) [9]. An additional result of the post-test analyses revealed that the validation of the NATDEMO code accomplished at steady-state natural circulation conditions was valid for the full range of transients studied.

In addition to providing verification of the EBR-II NATDEMO code, the data generated from these tests were used to validate codes used to support the FFTF natural circulation acceptance testing and the CRBRP natural circulation evaluations. As examples of these efforts, the IHX and secondary loop hot leg piping models in the CRBRP simulation code DEMO were verified from transient data measured in the corresponding parts of the EBR-II plant [10]. Data from the instrumented assemblies XX07 and XX08 during various natural circulation transients were used to verify the whole core/assembly code COBRA-WC and the hot channel code FORE-2M [11,12]. The three-dimensional thermal-hydraulics code, COMMIX-1A, was similarly verified by modeling the XX08 test in which total flow reversal occurred in the radial blanket/ reflector region during a natural circulation transient [13]. Verification studied were also conducted with SASSYS [14] SSC [15], MINET [16], and CORA, an FFTF code [17]. The success of these efforts, the improved understanding of the types of new data required for code validation, and the increased emphasis of DOE and reactor designers on inherent safety, lead to the initiation of the XX09/XX10 SHRT program. This program, which is intended to extend previous testing to the maximum capability of EBR-II as well as to provide detailed data on inter-assembly flow redistribution and inherent reactor shutdown by reactivity feedbacks, is discussed in the following section.

THE SHUTDOWN HEAT REMOVAL TEST (SHRT) PROGRAM

This test program, in which the first test was conducted in June, 1984, utilizes the two instrumented assemblies XX09 and XX10 described earlier. Testing will be conducted over an approximate two year period, limited only by the useful life of the fueled assembly XX09. The transients that are included in this testing cover a full spectrum of upsets in the primary, secondary and steam system including both protected (with reactor scram) and unprotected events.

Simultaneous LOF/Scram to Natural Circulation. The essential parameters that were studied in this sequence are the initial power and primary flowrate and the trip/non-trip of the secondary pump. The first test, which duplicates one that was conducted earlier with XX08, was initiated from 36% of full power and 40% of full flow by tripping the primary pumps. In addition, several different combinations of primary pump rundown times and delays in the secondary pump trip were studied. In all cases, the final state of the plant is natural circulation in both the primary and secondary systems. The subsequent test initial conditions were power and primary flow levels of 75%/100%, 75%/75%, and 100%/100%, respectively. For each of these test condition (with the auxiliary pump turned off), the secondary flow was in the first case slowly reduced to about 20% or in the second case permitted to rapidly coastdown to natural circulation. Thus, the final test in this group directly simulated a total loss of pumping power (normal and emergency) to the plant while operating at full power and flow.

Scram with Delayed LOF to Natural Circulation. These tests were all initiated from normal full power and flow where the first step was to scram the reactor while maintaining either full primary flow or auxiliary pump (~5%) flow. After a prescribed delay during which time the primary system approached isothermal conditions, the primary pumps (or the auxiliary pump) was tripped and the flow coasted down to natural circulation. The effect of the level and rate of coastdown of secondary flow was also studied. During these transients, the coolant flow in the radial blanket/reflector region either temporarily stagnated or reversed due to the adverse thermal conditions delaying the transition to upward natural convection.

Reactivity Feedbacks. These tests were designed to provide measurements of the dynamic plant response to flow and inlet temperature perturbations in order to permit validation of the reactivity feedback models in NATDEMO. A series of thermal-hydraulic disturbances were imposed upon the core, maintaining fixed (or precisely changed) control rod positions, in order to measure the resulting power changes and thereby to compute the effective reactivity feedbacks through an inverse kinetics calculation. The first type of disturbance was a change in primary flowrate, holding the reactor inlet temperature and control rod positions fixed. This causes the reactor ΔT to initially change, affecting the reactivity balance, which drives a power change. Initial power levels of 25%, 50%, and 70% were used.

The second type of disturbance was a change (either decrease or increase) in the reactor inlet temperature (caused by changes in the secondary flowrate), while the primary flowrate and control rod positions were fixed. Initial power levels of 25%, 50%, and 70% were used. A variation of the two types of disturbances employed was a flowrate or inlet temperature disturbance followed by a change in control rod position sufficient to return the power to its initial value. The results of some of these feedback tests is described in this Conference [18].

Loss of Primary Flow Without Reactor Scram. This sequence of tests all involve primary pump trips without permitting a reactor scram. The initial six tests will involve different combinations of rate of primary pump coastdown, trip or non-trip of the secondary pump, and operation of the auxiliary pump on emergency power, battery power, or not operated at all. All of these tests will be initiated from reduced power and flowrates. mildest transients, in terms of maximum core temperatures, will be conducted first, with the severity gradually increased. Depending upon the ultimate agreement between the measured test results with the NATDEMO predictions, additional tests of this type will be conducted at higher initial power and flow condi-tions, limited by the current requirements of maintaining peak fuel-clad temperatures below the U-Fe eutectic temperature. However, the initial six tests will include an unprotected LOF to natural circulation as well as an unprotected station blackout (loss of electric power without scram).

Loss of Normal Heat Sink Without Reactor Scram. These tests are closely related to the preceding group in that the objective is to demonstrate inherent reactor shutdown following a cooling system fault coupled with a total failure of the RSS (no scram). In this case, one test will involve a rapid steam depressurization without reactor scram. This event results in a decrease in the water saturation temperature, thereby reducing the secondary sodium temperature exiting the evaporators. This reduced secondary sodium temperature will overcool the primary sodium in the IHX, and thus reducing the reactor inlet temperature. Due to the negative reactivity feedbacks, the reactor power will increase causing an increase in core temperatures. However, in spite of the potential seriousness of this transient, it is anticipated (as confirmed by NATDEMO calculations), that the core ΔT will experience only a modest increase.

The second type of transient that will be conducted is that of an essentially total loss of secondary flow (a decrease to about 0.5% of normal within about 20 seconds) with the reactor operating at full power conditions. No other changes will be made and the control rods will not be scrammed. These particular conditions were chosen to umbrella all possible loss of heat sink transients. In this case, the loss of heat removal from the primary system will cause the reactor inlet temperature to increase and thus the reactor power to decrease due to the negative reactivity feedbacks. Although the inlet temperature is predicted to rise substaintally, the resulting power reduction is sufficiently large and rapid to cause a reduction in the outlet temperature (at about 40 minutes into the transient), so that the core ΔT actually decreases to only several degrees and the entire primary system approaches an isothermal condition. The asymptotic temperature reached (with the reactor just critical at zero power) is sometimes referred to as the reactor "quenching temperature".

Dynamic Frequency Response Tests. These tests have been designed in order to generate data over a wide disturbance frequency range that can be used to validate whole plant dynamic simulation models. There will be two types of cyclic tests in which a relatively small periodic disturbance is added to either control rod inserted reactivity or secondary loop electromagnetic pump voltage. This periodic disturbance would consist of a fundamental frequency and a number of harmonics. The propagation and attenuation of these frequencies in the form of power, temperature, flow, and pressure disturbances would be measured throughout the plant. The cyclic tests would also be simulated with the analytical codes. Spectral analysis would be applied to both the measured output and the simulation results, providing transfer functions for each. A comparison of the two sets of results (measured vs calculated) is a direct measurement of the efficacy of the calculational models.

In the first group the automatic control rod driver system (ACRDS) would be used to produce a periodic disturbance in reactivity. All of the tests would be performed at a relative power-to-flow ratio near unity. Tests would be done near full power and near half power. Although several frequencies would be contained in the periodic input wave of each test, two tests would be required at each power level to adequately cover the frequency range of interest.

The second group of tests would be analogous to the first and again would require two power levels and two tests at each power level. However, instead of using the ACRDS to disturb reactivity, the appropriate periodic disturbances would be added to the secondary EM pump voltage. An effort is currently underway to identify and obtain the necessary computer hardware and software to generate these periodic disturbances.

PRELIMINARY RESULTS

In this section, some typical experimental results obtained from the three types of protected loss of flow to natural circulation tests that were conducted will be described. These tests essentially differ in the sequence of events which lead to a transition from forced convection to natural convection cooling of the reactor core. The range of conditions covered by these tests encompass transients initiated from normal power operation to hot-standby conditions. Since the unprotected transients discussed earlier have not yet been conducted, their presentation will be the subject of future papers.

Loss of Flow from Power Operation. Two types of transients involving a loss of forced flow were conducted resulting in significantly different dynamic behavior. The first type was initiated by an essentially simultaneous trip of the pumps and a reactor scram, while the second had a pump trip substantially delayed after the scram. Thus, while the transition to natural circulation was added by the presence of the pre-existing thermal head in the primary heat transport circuit during the first type of test, the absence of any thermal heads when the pumps were tripped in the second test type (due to overcooling after the scram) resulted in a difficult development of convective flow.

The simultaneous LOF/scram tests were conducted over a wide range of initial reactor powers and flows, and utilized a variety of secondary flowrate transients. Typical temperature transients, measured by a wire-wrap thermocouple located in a central coolant subchannel at the top of the core (top of fueled region), are shown in Figure 1 for a variety of initial reactor conditions. These results cover tests conducted over the period of 1978 to 1984, utilizing the XX08, XX09, and XX10 instrumented assemblies and all involved relatively slow secondary flowrate transients. As can be seen, the coolant temperature response in these fueled assemblies to a natural circulation transient is relatively benign, especially since even the normal operating temperatures are only slightly exceeded. The parametric effects of increasing reactor power, as well as increased assembly power at the same reactor power, upon the thermal transients are as expected.

One especially interesting result obtained from these tests was the relative responses of the two different instrumented assemblies, XX09 (fueled) and XX10 (blanket simulation). As an illustration, some detailed data from the 100% power and flow case from Figure 1 are presented in Figures 2, 3, and 4. The measured flowrates in XX09 and XX10 are shown in Figure 2, while the top-of-core central subchannel coolant temperatures are shown in Figure 3 and the subassembly outlet in Figure 4. As can be seen, although these two assemblies substantially differ in their initial thermal-hydraulic conditions, under natural convective conditions, their behavior becomes more similar. In fact, at the subassembly outlet (the thermocouple location was within the assembly hex can), the XX09 and XX10 coolant temperatures converge. Although substantial inter-assembly flow redistribution has occurred, it was insufficient to cause the coolant temperatures at core-top to converge. Thus, the behavior at the higher elevation is almost certainly due to inter-assembly heat transfer. Hence, the data obtained from this and related tests can be used to directly validate the models of combined inter-assembly flow redistribution and heat transfer used in available core and system thermal-hydraulic codes.

The second type of transient LOF from power operation that was conducted involved an initial reactor scram, maintaining the power to the primary pumps. This continued forced flow cooled down the entire primary heat transport circuit to an essentially isothermal condition. At this point, when the pumps were tripped, there was no thermal buoyancy available in the circuit to aid the establishment of natural circulation, and the core temperatures increased until sufficient buoyancy within the core developed to cause convective flow. However, during this period, the flow within the core essentially stagnated and within the radial blanket region (connected to the low pressure plenum, LPP), the coolant flow reversed. These observations are illustrated with some typical data in Figures 5 and 6. The flow in the LPP or radial blanket region remained in the downward direction for periods ranging from tens of seconds to many minutes, depending upon the actual test conditions. Within the instrumented assemblies, which had highly sensitive flowmeters precisely calibrated at low flowrate conditions, coolant velocities generally remained positive, but at values of the order of 1 mm/sec and less.

However, even under these seemingly serious flow conditions, it was quite impossible to overheat any assemblies. This was due to the fact that as soon as the coolant temperatures sufficiently increased, convective flow started and limited further temperature rises. These tests, supported by extensive analyses, have demonstrated that it is possible to establish convective flow in a reactor even under strongly adverse thermal conditions.

Loss of Flow from Hot-Standby Conditions. At many times during the normal operational cycle of a power plant, the shutdown reactor core is cooled by primary pumps driven by pony motors (or in the case of EBR-II, by the auxiliary pump). If electrical power is lost under these conditions, the combination of a potentially rapid pump rundown and relatively small initial temperature rises (i.e., low initial buoyancy) may result in temperatures as large as or even greater than that occurring during similar events initiated from normal power operation. Accordingly, this transient has been studied in EBR-II over the period of 1976 to 1984 covering a wide range of decay power levels, reactor inlet temperatures, and secondary flowrate conditions.

A typical set of data from such tests are shown in Figures 7 and 8. The only parameter varied in these tests was the initial decay power level, covering shutdown periods ranging from about one hour to over one week. The secondary flowrate in all cases was maintained constant at a low value. The measured coolant temperatures (central subchannel at core-top) increased with increasing decay heat level in a monotonic manner. However, the transient flowrates varied in a somewhat more complex manner due to the range of initial flowrates used and the resulting different initial primary circuit buoyancy forces. However, all of these results have been well predicted using the

NATDEMO code. Thus, although the transient temperatures occurring following a LOF under these conditions can be substantial, they are predictable and not necessarily larger than those occurring under other reactor operating conditions.

CONCLUSIONS

An extensive and on-going experimental program at EBR-II is generating experience as well as code validation quality data on the transient performance of an LMR under a wide variety of upset conditions. The completed natural circulation test program was described and several of the important conclusions were summarized, but the essential result was that it was possible to accurately predict the dynamic natural circulation performance of EBR-II during every type of transient conducted. This accomplishment has strongly supported the case being made by the current advanced reactor designers in the US that natural convection is a highly reliable and predictable method of decay heat removal. In addition, the planned unprotected loss of flow and loss of heat sink tests were described. It is expected that these tests will confirm the calculations of the availability of strong negative reactivity feedbacks for power reduction during such transients. These tests, along with pre- and post-test analyses, will provide essential support for many of the inherent safety arguments being made by advanced reactor designers.

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Fig. 1. Effect of Initial Power and Flow Conditions on the Coolant Temperature in a Central Subchannel at Core Top During a Protected LOF to Natural Circulation

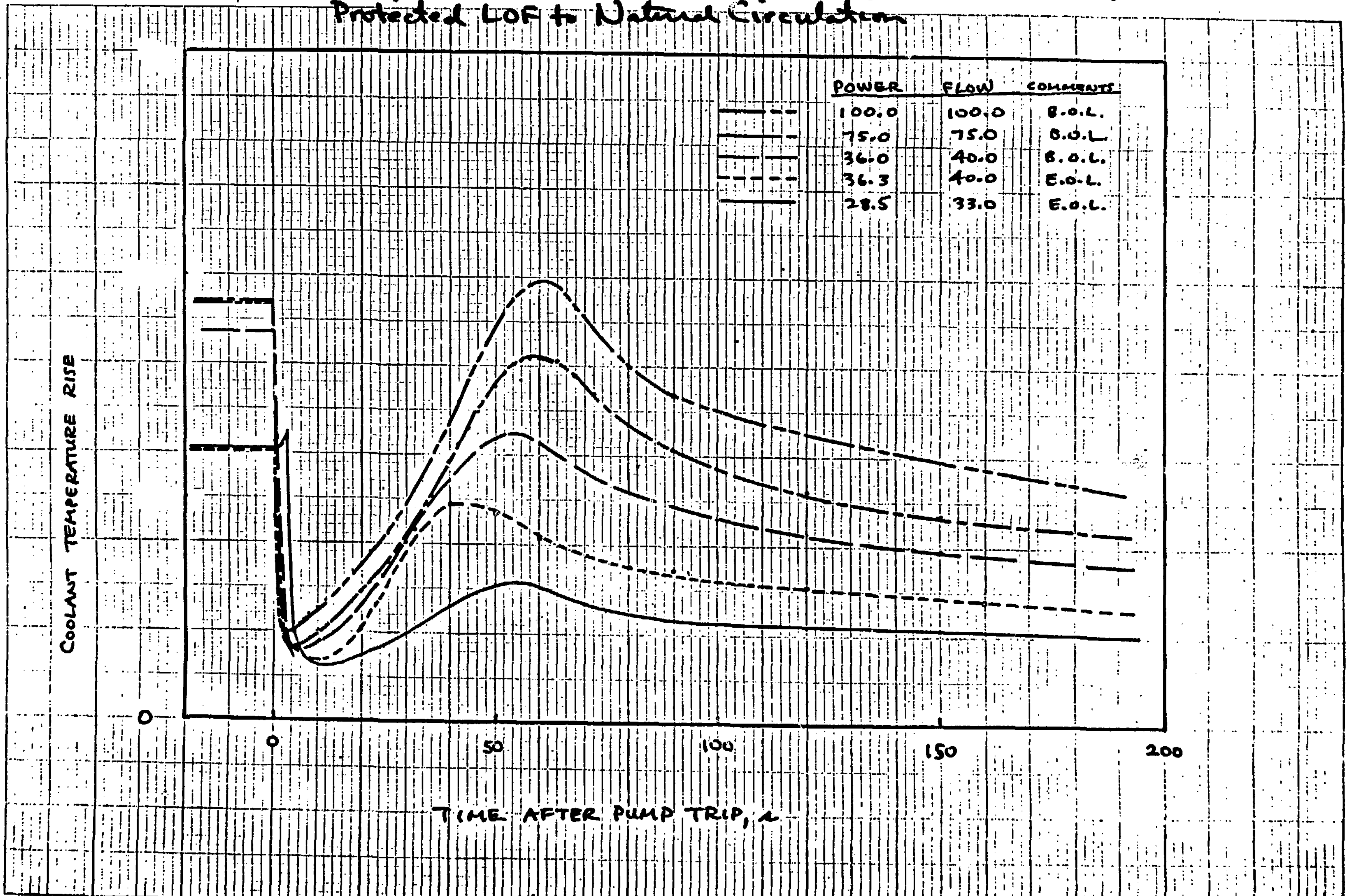

Figure 2. Local Coolant Flowrates Measured During a Protected LOF to Natural Circulation from Full Power and Flow

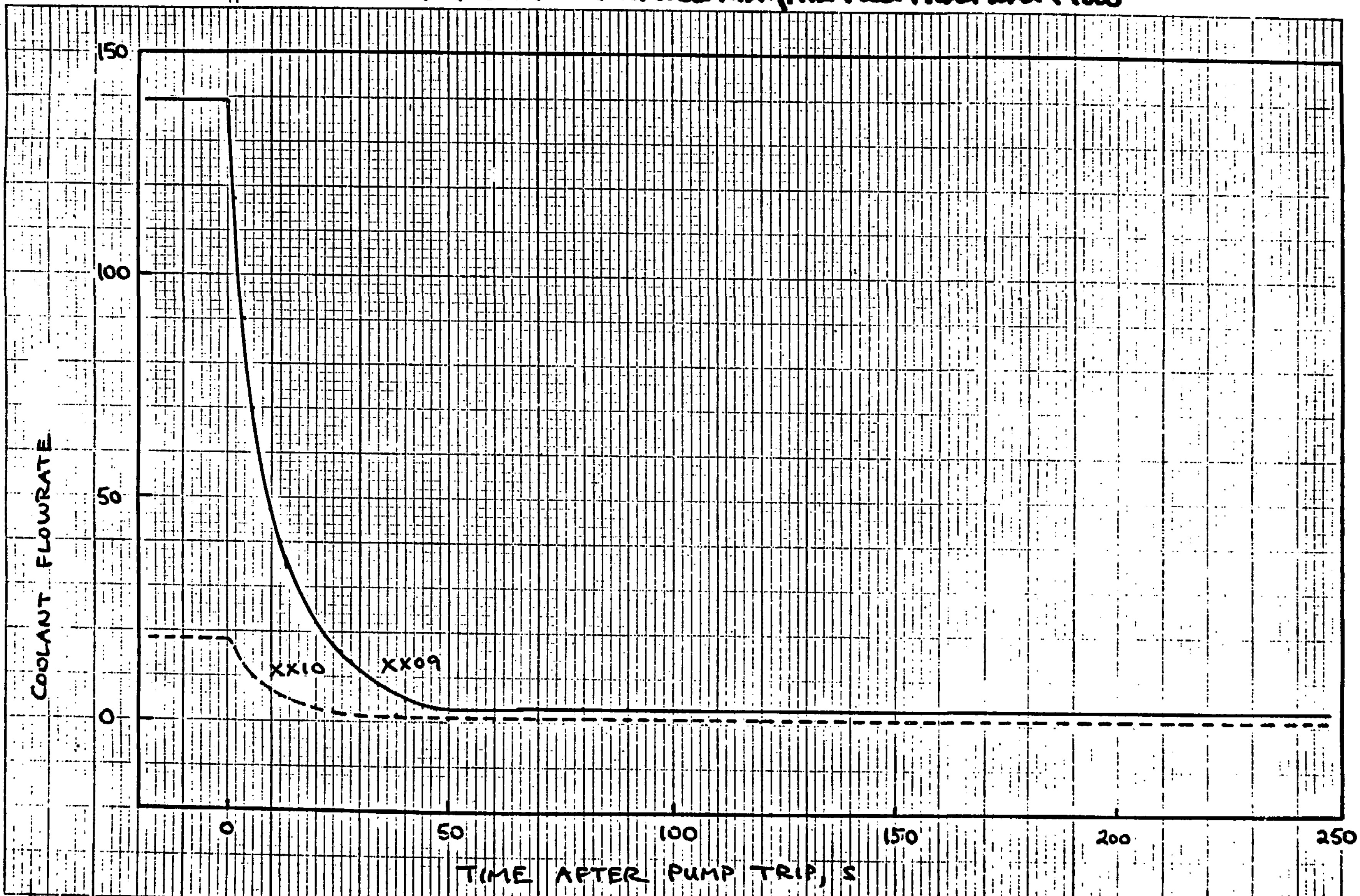

Figure 3. Local Coolant Temperatures Measured During a Protected LOF to Natural Circulation from Full Power and Flow (Top-of-Core)

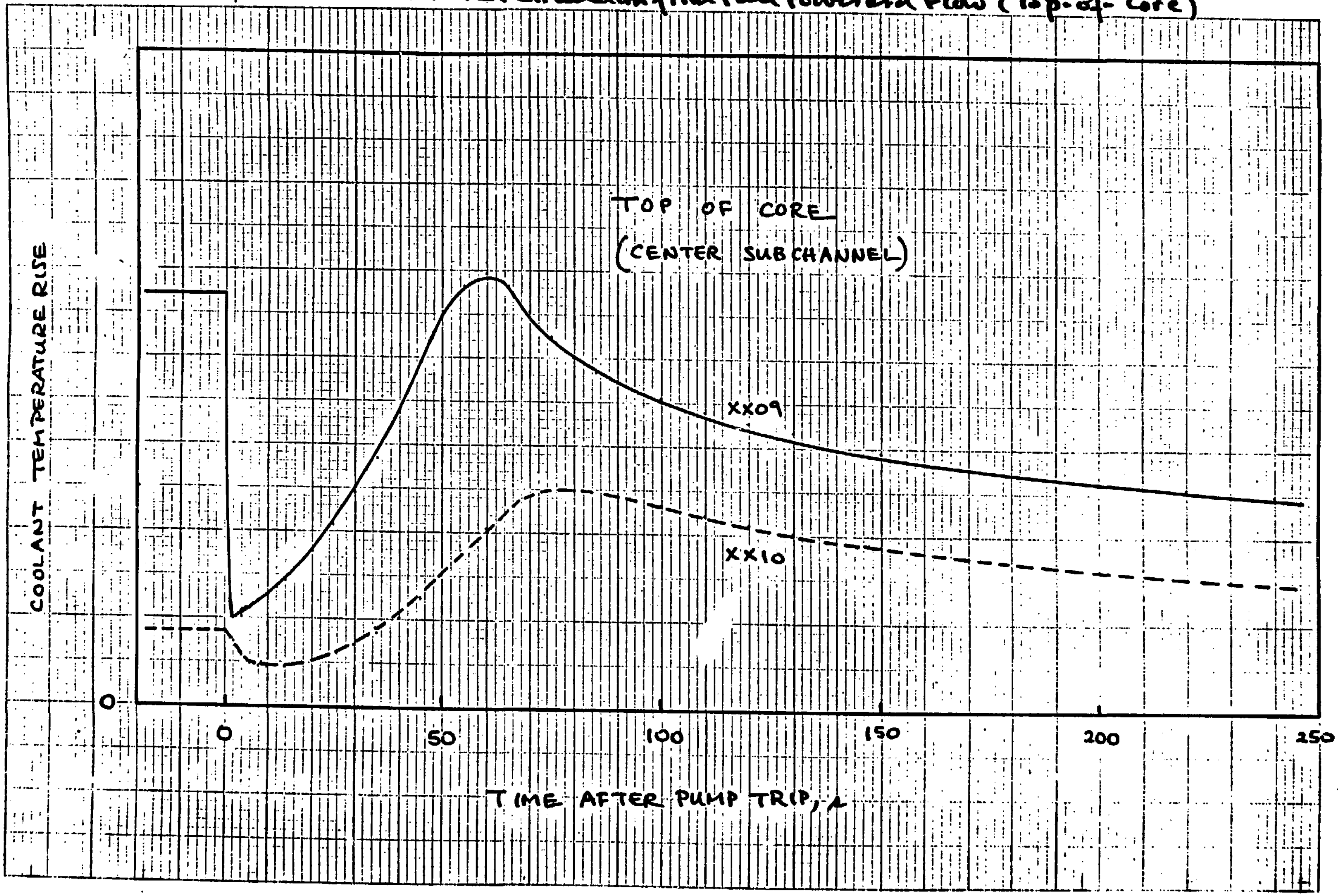

Figure 4. Local Coolant Temperatures Measured During a Protected LOF from Full Power and Flow (Subassembly Outlet)

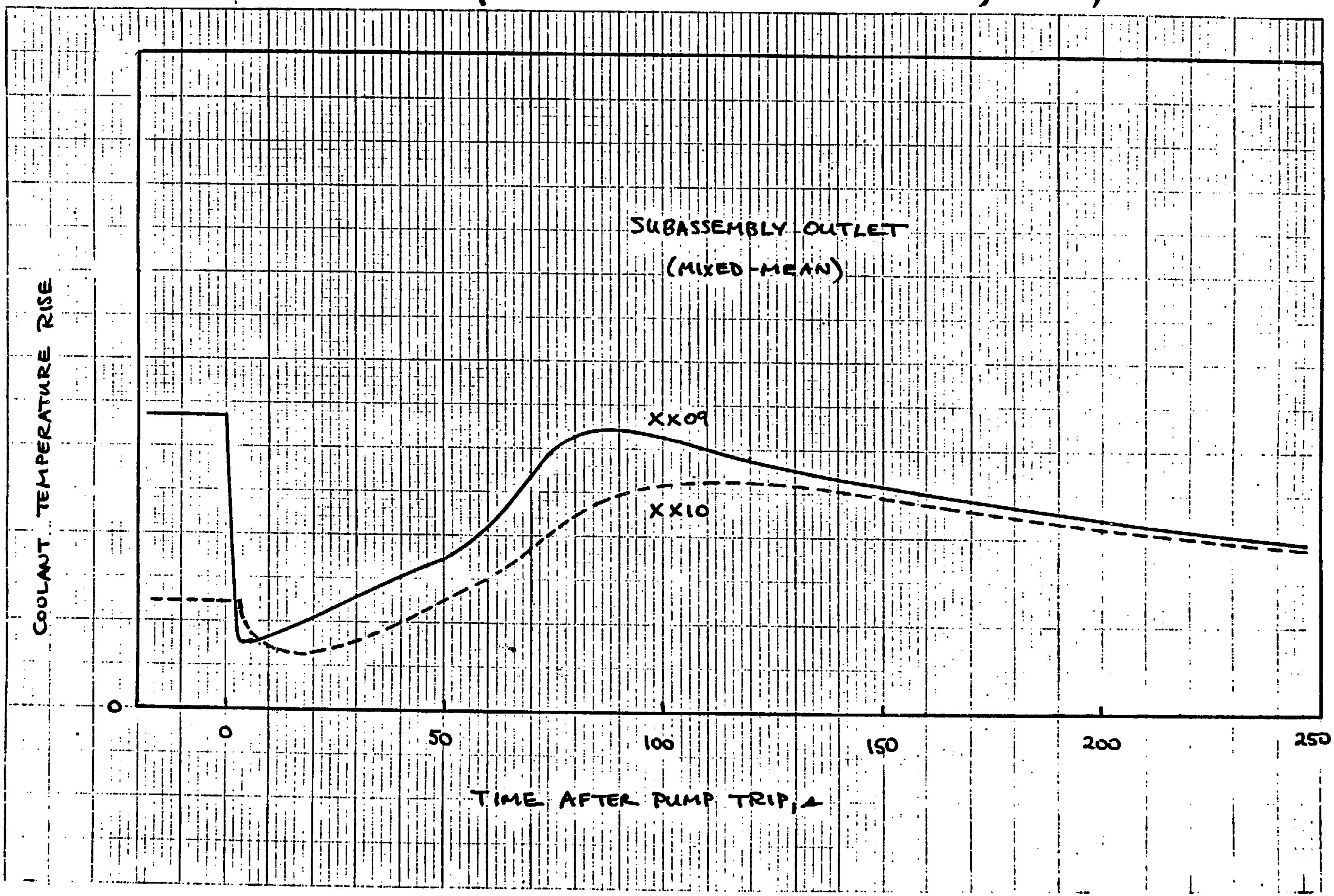

Figure 5. Local Coolant Flowrate Measured During a LOF Delayed After a Scram From Full Power

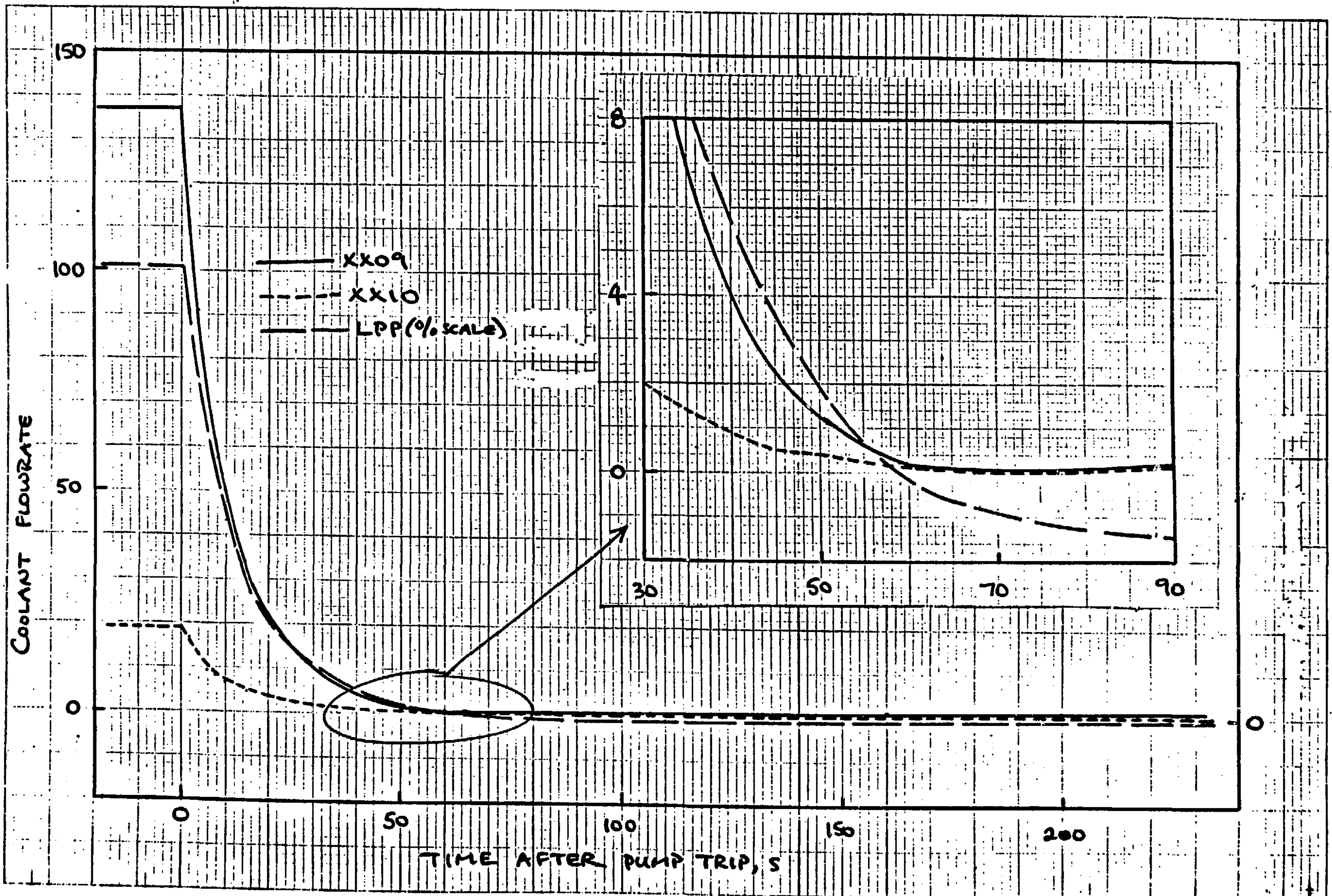

Figure 6. Local Coolant Temperatures Measured During a LOF Delayed After a Scram From Full Power

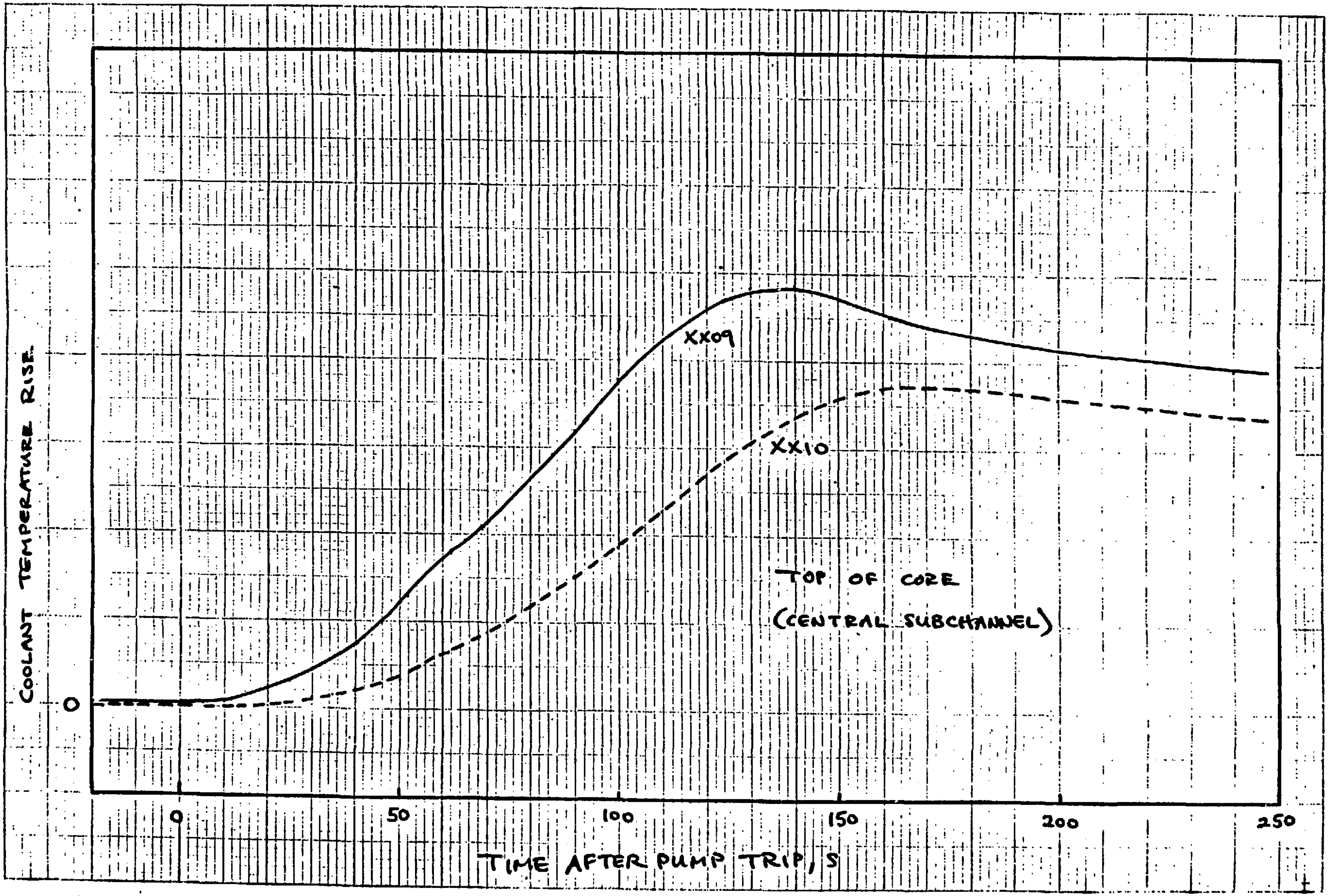

Figure 7. Measured Top-of-Core Coolant Temperatures Measured During a Loss of Power to the Auxiliary Pump

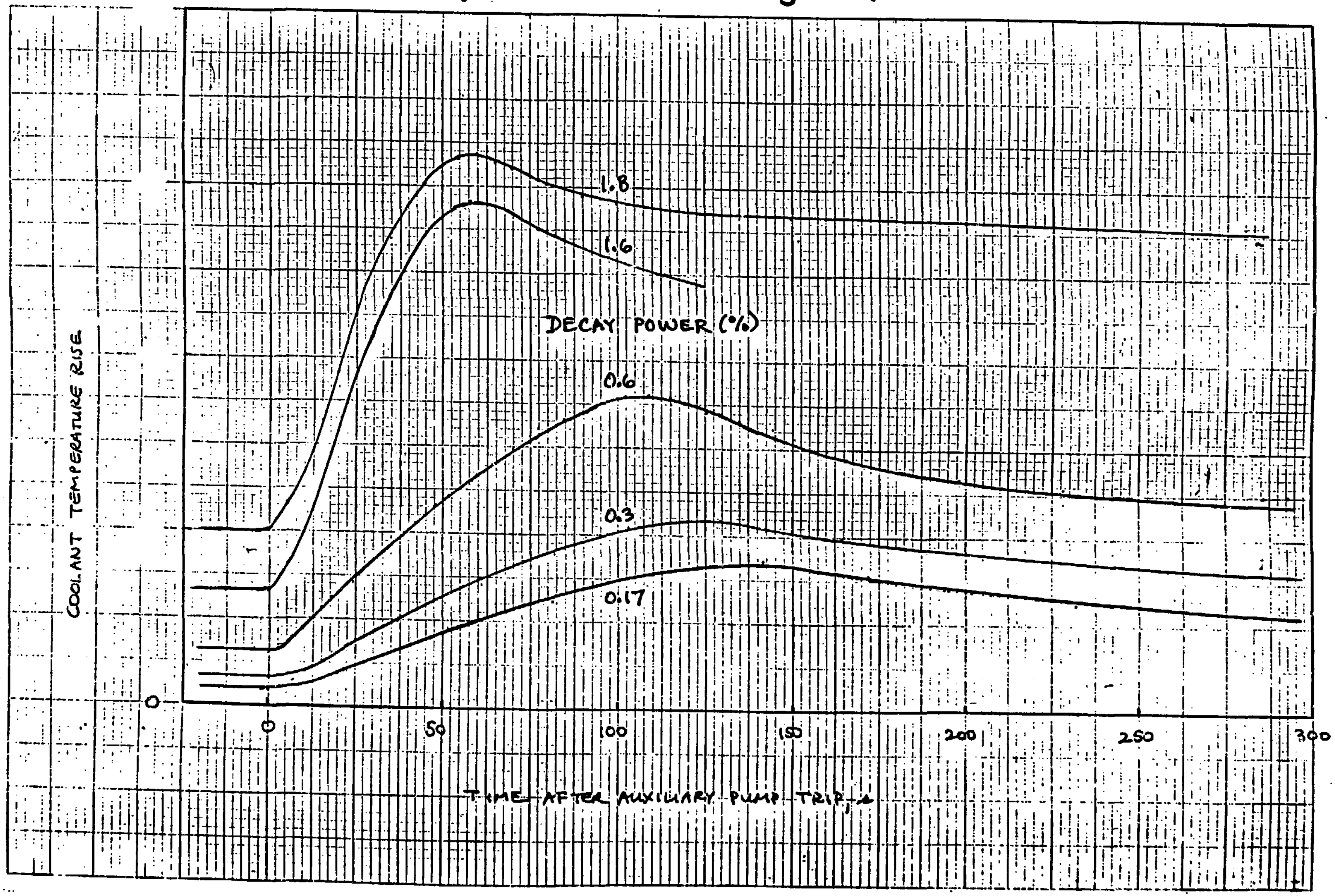

Figure 8. Measured Local Coolant Flowrate During a Loss of Power to the Auxiliary Pump