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ABSTRACT

One of the most important aspects of the demonstration of the safety of liquid metal cooled fast reactors (LMFBR) is the assurance of adequate cooling of the reactor and critical structures during all potential or hypothetical events. To this end, a design philosophy has developed in which alternate heat removal systems are included in a plant in order to provide independent, diverse, and redundant cooling from the normal heat transport system. In most all designs, at least one of these alternate systems is intended to operate passively, that is, by natural convective flow without any required active intervention. The confirmation of the adequacy of such passive systems to meet desirable performance criteria has necessitated analytical and experimental studies of the dynamics of the transition from forced to natural convective flow in the complex geometries existing in fast reactors.

Recent studies, e.g., (1-7), have examined various aspects of natural convective flow in such systems and preliminary conclusions from these efforts have indicated that the basic phenomena are reasonably well understood and predictable in most cases. However, in certain situations where significant thermal stratification occurs, the resulting buoyancy-driven flow patterns can become quite complex and as a result, the confidence in computer simulations diminishes.

The purpose of this paper is to present the results of an experimental and analytical study of one class of such problems in which the development of natural convective flow requires a transition through an unstably stratified condition. The experiments were conducted in the Experimental Breeder Reactor II (EBR-II) and the analytical tool used

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was the NATDEMO system simulation code (8). The transient studied was initiated from a normal at-power condition of the reactor plant wherein the reactor core, cold- and hot-legs of the heat transport system were at nominal temperatures. At time zero, the reactor was scrammed, resulting in a rapid drop in power generation to a low level; the coolant pumps were permitted to continue operating for a predetermined length of time at full flow causing a cooldown of the reactor and hot-leg of the heat transport circuit. At the specified time, the pumps were tripped and coasted down to zero speed; any subsequent flow was driven only by buoyancy forces. However, under these conditions, the only region within the heat transport system where buoyancy could be generated was in the reactor by heating the coolant. This heating in the lower region of the flow circuit resulted in an unstable thermal stratification since the coolant and structures in the rest of the circuit above the reactor remained at much lower temperatures. Ultimate steady-state could not be attained until the heat generated within the reactor could be convected throughout the circuit by accelerating and/or displacing and mixing with the existing cold fluid. This situation can lead to many possible alternative flow patterns as the warm fluid attempts to move upward into the colder fluid.

Since the reactor core consists of many independent flow channels (coupled hydraulically only in the lower and upper plenum regions), each one of which has a different power level, hydraulic resistance, and heat capacity, the potential for temporary flow stagnation and various modes of flow recirculation exists in this region. The analytical predictions based upon the EBR-II plant parameters indicate that this class of transients results a temporary total reversal of flow within the radial blanket of the reactor (a region of relatively low power and high heat capacity) which persists for several seconds to several minutes, while the flow through the fueled region (higher power and lower heat capacity) always continues in the normal upward direction.

Experiments conducted utilizing EBR-II and special instrumented probes supported these analytical predictions. Detailed transient measurements of local coolant and fuel temperatures, local and total coolant flow rates, and heat transport circuit temperatures were obtained. Although only several plant tests were conducted, the measured flow rates and temperatures clearly revealed radial blanket flow reversal with excellent quantitative agreement with the analysis. A typical comparison of these measurements and analytical results is shown in Figs. 1, 2, and 3. A complete discussion of these experimental observations as well as the analytical results will be presented in the paper. Special emphasis will be placed upon the observed physical phenomenology, the analytical modeling required to fully describe transients of this type and the resulting implications of passive nuclear reactor cooling during accidents.

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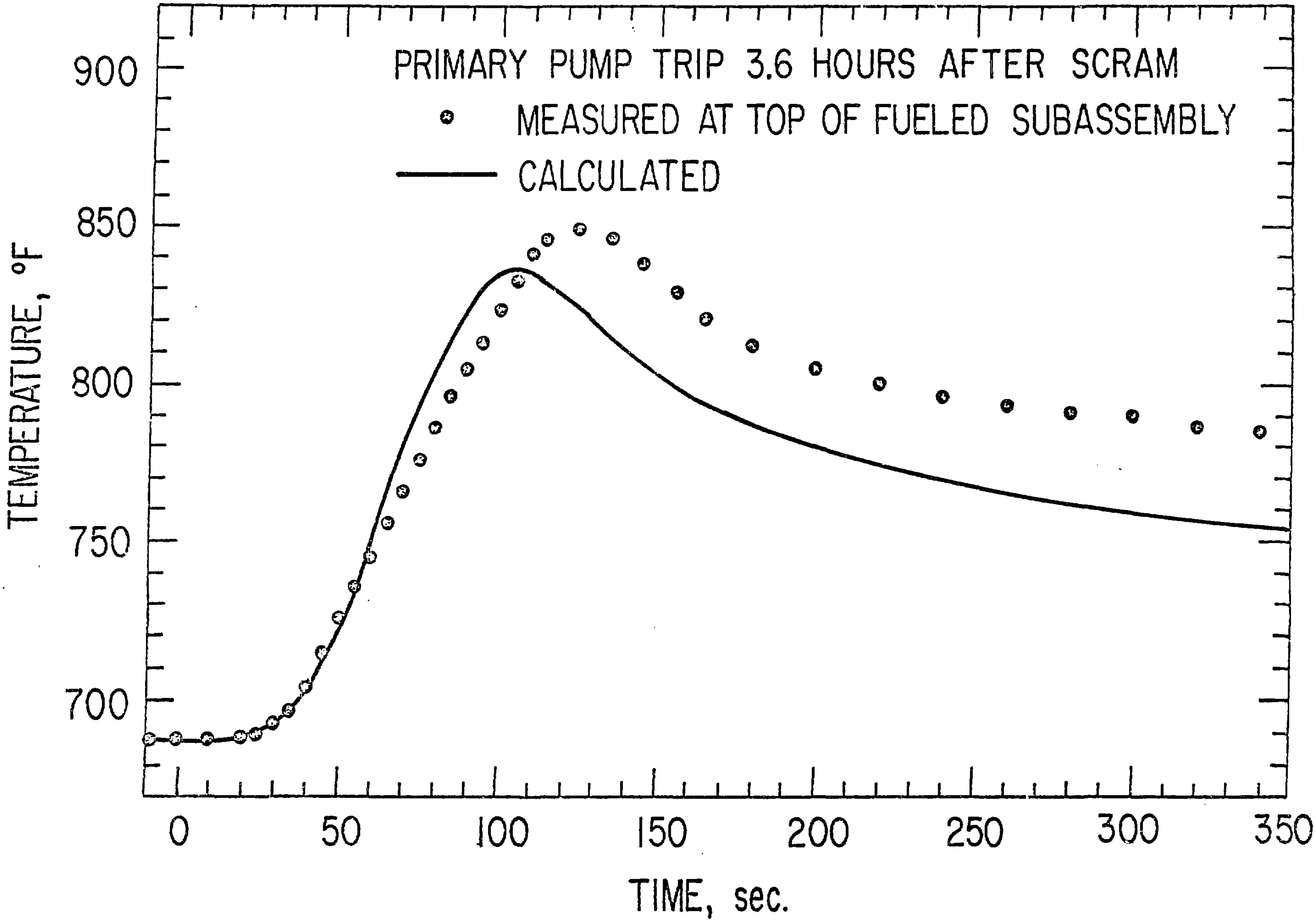

Fig. 1. Comparison Between Measured and Calculated Coolant Temperature at Top of Fueled Subassembly Following Loss of Forced Flow

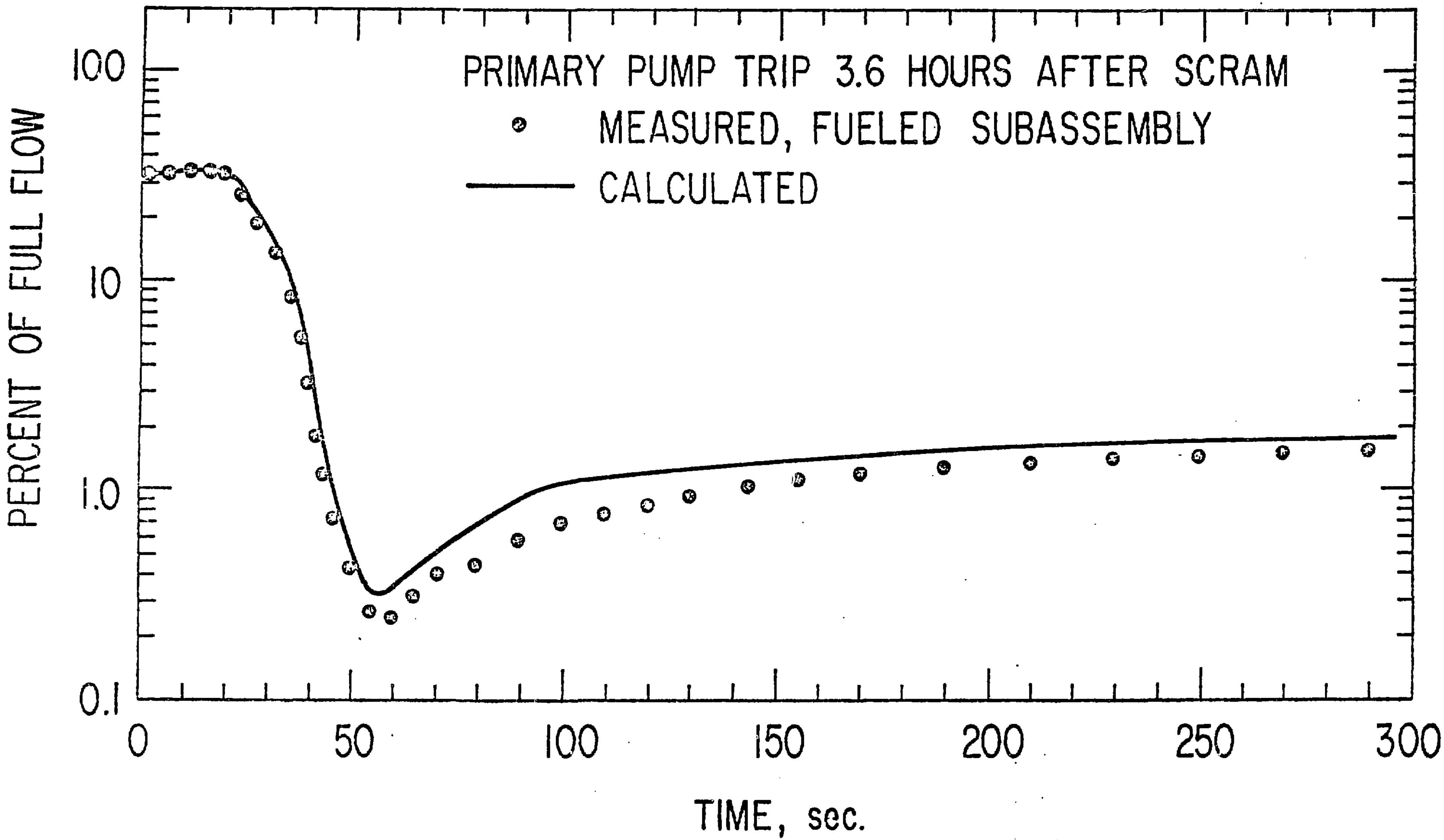

Fig. 2. Comparison Between Measured and Calculated Coolant Flow Rate in a Fueled Subassembly Following Lost of Forced Flow

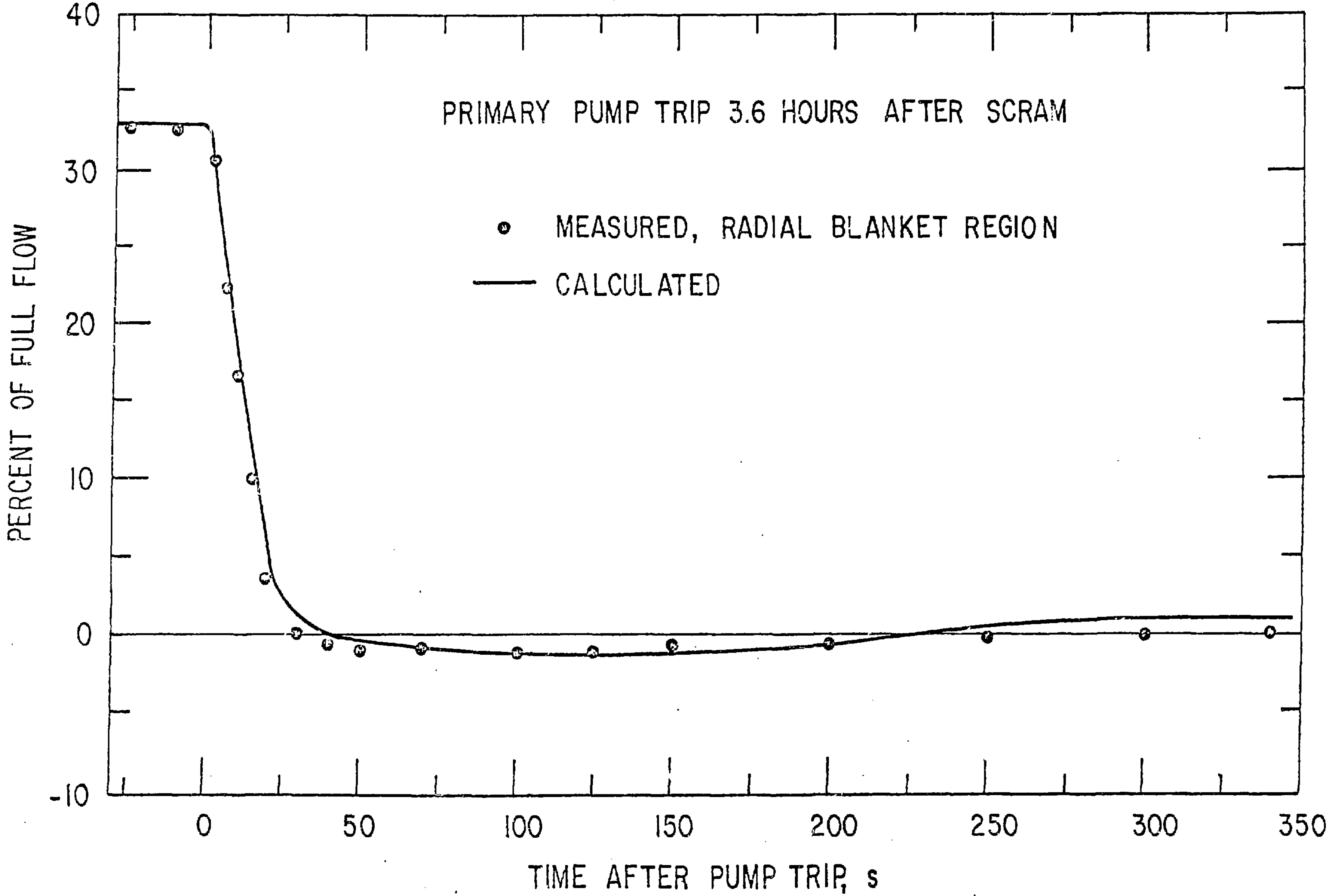

Fig. 3. Comparison Between Measured and Calculated Coolant Flow Rate in the Radial Blanket Region Following Lost of Forced Flow