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Passive Safety and the Advanced Liquid Metal Reactors*

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by

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ABSTRACT

Advanced Liquid Metal Reactors being developed today in the USA are designed to make maximum use of passive safety features. Much of the LMR safety work at Argonne National Laboratory is concerned with demonstrating, both theoretically and experimentally, the effectiveness of the passive safety features. The characteristics that contribute to passive safety are discussed, with particular emphasis on decay heat removal systems, together with examples of Argonne's theoretical and experimental programs in this area.

INTRODUCTION

The Advanced Liquid Metal Reactors (LMR) currently being developed in the USA, PRISM (1) and SAFR (2), are designed to make maximum use of passive safety features. These passive safety features arise from a pool configuration, advanced ternary alloy metallic fuel, enhanced feedback characteristics and passive decay heat removal systems. As currently envisaged, these LMRs share a great many features with the Integral Fast Reactor (IFR) concept being developed at Argonne National Laboratory in a program sponsored by the U.S. Department of Energy (3).

In LMRs, the liquid sodium coolant operates at atmospheric pressure, and maintains a margin to boiling greater than 400 K (700 deg F). This eliminates the need for a pressurized primary system and thick-walled pressure vessels. The use of sodium coolant in a pool reactor, combining the high thermal conductivity of sodium and large thermal heat capacity of the pool, allows removal of the decay heat from the core by natural circulation and large grace times for corrective action. At nominal operating conditions, liquid metal cooling permits a compact core configuration that compliments the neutronic advantages provided by a fast neutron energy spectrum.

Safety is assessed by the predicted (or actual) response of the design to a spectrum of anticipated transients, with or without action of the plant protection system, (protected and unprotected accidents). The concept of passive safety requires that the reactor respond to a transient and bring itself to a stable quiescent state without operator action, or application of engineered safety features which rely on instrumentation recognizing the need for action. Analysis of these transients is performed using a coupled neutronic and systems thermal hydraulic code such as SASSYS-1, developed at Argonne National Laboratory (4).

PASSIVE SAFETY CHARACTERISTICS OF DESIGN CHOICES

Pool Configuration

To meet the requirements of passive safety the primary system is arranged with the core, the intermediate heat exchangers, and the primary pumps all submerged in a large pool of liquid sodium. All radioactive materials, and coolant that travels through the core, are confined within a large, thin-walled reactor vessel. A back-up guard vessel guarantees that all major components will remain submerged in the event of a reactor vessel leak. This pool-type arrangement minimizes the amount of high pressure piping to the pipes (several per pump) which run between the pump outlet and the core inlet, and reduces the consequences of a pipe rupture accident. In such an accident, no coolant spills are possible and core cooling is maintained even if one pipe should suffer a double-ended guillotine break. With the whole primary system submerged, natural circulation paths are assured for all normal, as well as, abnormal operating conditions. In addition, the large heat capacity of the sodium pool provides long grace periods (hours) for response to cooling system faults. It is necessary to ensure that for any postulated transient initiator, natural circulation can be easily attained without exceeding temperature limits in the core and structural components. This is achieved by design which allows for appropriate relative positioning of the major components, for example, placing the IHX outlet at an elevation above that of the core to guarantee sufficient thermal head, and by suitable choice of pump behavior in the cases where the pumps are assumed to cease to function, as a result of loss of power, bearing seisure events, siesmic events, etc.

Metal Fuel

Both advanced LMR designs currently plan to use metal fuel, (IFR fuel). This is a ternary metal alloy of Uranium, Plutonium and Zirconium. Many of the superior safety performance characteristics of the IFR ternary alloy fuel design can be traced to its thermal and mechanical properties, with the most important of these being its very high thermal conductivity. At operating temperatures, typical fresh IFR fuel has a thermal conductivity of around 20 W/m-K. This yields a very low radial temperature drop across the fuel at operating conditions (less than 200 K), and a comparatively low stored heat. The low temperature gradient across the fuel gives a correspondingly small Doppler reactivity swing between zero power and full power states, resulting in reduced control reactivity requirements and less external reactivity available for accidental insertion. The low operating temperature difference also yields a low positive Doppler reactivity feedback in fuel cooling transients. For unprotected transients such as loss-of-heat-sink (LOHS) and loss-of-flow (LOF), this permits other naturally occurring negative reactivity feedbacks, such as axial and radial core thermal expansion, to overcome the positive Doppler component, resulting in self-adjustment of the reactor core power to equal available decay heat removal capacity at acceptable temperatures. The low stored heat in the metallic fuel leads to reduced system heating in multiple-fault accidents, allowing increased time for operator action to correct flow or cooling deficiencies.

Feedback Characteristics

The harder neutron spectrum attendant the metallic fuel form has two important effects on reactivity feedback coefficients. The negative Doppler reactivity coefficient T dk/dT is reduced by about a third relative to oxide fueled systems. The positive sodium density coefficient becomes more positive by about 1/3. While the second effect is unfavorable from a reactor safety perspective, the effect of the reduced Doppler reactivity vested in temperature rise in the fuel relative to the coolant dominates and the overall impact on safety performance is a marked improvement.

To assist in attaining a self-limiting response to accident conditions specific design choices can be made. Because in most unprotected accidents the coolant temperature in the core rises rapidly initially, it is expedient to utilize feedbacks dependent upon this quantity to control the response. Two such mechanisms are radial core expansion and control rod driveline expansion.

For radial expansion to be effective, it is necessary that the radial core restraint system be configured to provide contact at the load pad plane during normal operation. In addition, provision must be made to allow for thermal expansion during the transient. These requirements lead to the 'limited free bow' restraint system which allows for duct expansion and load pad growth as the coolant temperature rises, providing a negative reactivity feedback. To enhance differential control rod expansion, core outlet coolant flow may be ducted around rod drive-lines, and core-support members can be located in areas that heat relatively slowing during an accident. This assures that the differential movement of control rods early in the transient acts to insert control material into the reactor. Both these feedbacks have characteristic times associated with them and the choice of pump coastdown behavior must properly reflect the need to avoid short term power to flow mismatch.

Fig. 1a, 1b, 1c,
this column

PASSIVE DECAY HEAT REMOVAL SYSTEMS

The function of these systems is to ensure adequate heat removal so that, in the event of an accident in which the normal heat removal capability is lost, the reactor can be cooled. This implies that this capability must be adequate to remove the decay heat at some time after the neutronic shutdown.

The primary method of removing the decay heat is with the steam generator (an active system). The unavailability on demand of the steam generator for decay heat removal is of the order of 10^{-2}/year and thus back up decay heat removal system are needed. Passive backup decay heat removal systems are favored over active systems because they have an inherently higher reliability. Passive decay heat removal can be achieved in several different ways. The advanced reactors use three different types, see Fig. 1. The first type is a Direct Reactor Auxiliary Cooling System (DRACS) which consists basically of a heat exchanger inserted into the sodium pool. A naturally circulating Na or NaK loop removes heat directly from the sodium pool and releases the energy to atmospheric air via a natural draft heat exchanger. The second type is called an Auxiliary Cooling System (ACS) and consists of a natural draft heat exchanger surrounding the steam generator. Energy is transferred by natural convection from the core to the steam generator, passes through the walls of the steam generator by conduction and is convectively transfered to the natural circulating air. The third type is a Reactor Vessel Air Cooling system or a Reactor Auxiliary Cooling System, (RVACS/RACS), depending on whether one is describing PRISM or SAFR. In this concept ambient natural circulating air is used to cool the guard vessel, with the energy transferred from the reactor vessel to the guard vessel by radiation.

The containment boundary of the advanced reactors is the guard vessel. Previous risk assessments of reactors with active decay heat removal systems indicate that the probability of core melt is dominantly caused by failure of the decay heat removal systems. It is believed that the use of passive systems, particularly the RVACS/RACS, can lower the probability of core melt due to the loss of decay heat removal to levels significantly less than 10^{-6}/year. The reliability of RVACS/RACS is a result of the natural robustness of the system: its large passages, and its large thermal mass allowing long times for corrective action. The use of RVACS/RACS on the SAFR and PRISM designs results in a very compact containment, however, use of a conventional visible containment is not precluded as it appears feasible to use a double RVACS/RACS system (see Fig. 1). The double RVACS/RACS is similar to a single RVACS/RACS except that the air that cools the guard vessel is within the containment and transfers its energy to the containment shell and atmospheric air. The practical limit of reactor sizes for application of the double RVACS/RACS has not been determined.

PROOF OF SAFETY FEATURES EFFECTIVENESS

Much of the LMR safety work performed at Argonne National Laboratory is concerned with demonstrating both theoretically and experimentally the effectiveness of the passive safety features. This includes, for example, analysis of transients using the SASSYS code to demonstrate that design goals can be met, and experimental programs designed to evaluate the performance of specific features.

As an example of the former, Fig. 2 gives the results of a SASSYS analysis of the unprotected LOF for the PRISM reactor. In this case the primary pump is electromagnetic and so the flow drops to 40% of nominal in one second. The resulting negative feedback from radial expansion caused by the overheating core reduces the power rapidly and in the long term there is ≈ 400 K margin to boiling, Fig. 3. All of the phenomena discussed earlier contribute to this result.

Fig. 2

Fig. 3

One example of an experimental program is the Natural Circulation Shutdown Heat Removal Test Facility (NSTF) at Argonne. The NSTF is a full scale simulation of the RVACS/RACS concept. A circumferential segment of the guard vessel/duct wall natural convecting air system has been fabricated. It consists of a ≈ 1.3 meter wide by 6.7 m. tall heated zone with a 18 m. insulated chimney. It is capable of operating with guard vessel temperatures to 900 K and heat fluxes to 20 kw/m^2 (conditions in excess of demands of PRISM and SAFR). Tests of the PRISM configuration have been completed and tests of the SAFR system are presently underway. The primary goal of the NSTF experiments is to provide passive heat removal performance data characteristic of a full-scale LMR design. Performance predictions of the facility have been made and refined through the design phase. Figure 4 shows the predicted and experimental results of system flow (Reynold's) versus guard vessel heat flux for various system loss parameters for the PRISM configuration. The overall hydraulic resistance, KLOSS expressed in terms of inlet velocity head, is a dominate parameter affecting system performance. In the figure 90% of the data points lie above the predicted performance with a maximum deviation of (+) 20%, (-) 3%.

Fig. 4

A second example is the completion of full power loss-of-flow tests without reactor scram in EBR II in April 1986 (5). In these tests the reactor was at full power, the power to the pumps was terminated and allowed to coastdown, the average core temperature increased slightly, resulting in a negative reactivity feedback causing a power reduction to decay power levels, while maintaining safe temperatures well within margins. Experiments such as these are valuable not only as demonstrations of the effectiveness of these passive safety principles but also as experiments against which safety codes can be validated. SASSYS

Fig. 5

has been applied to analysis of the above experiments, in order that the application to the advanced LMR's, described above, can be demonstrated to have a firm foundation. Since the NSTF simulates a RVACS/RACS concept, SASSYS should be above to describe the behavior accurately. Figure 5 shows the results of one simulation (6). Because of the initial overshoot in power level, the guard vessel temperatures overshoot and peak at ~10000 s and then drop back to final equilibrium values. The duct wall and air tempertures agree well with the measured values. The calculated air flow rates agreed with the measured values to within 9%.

Another example of SASSYS validation is analysis of the SHRT 45R experiment in EBR-II. This transient was the unprotected loss of flow from full power and flow performed as part of the Inherent Safety Demonstration Tests. This transient was analyzed using SASSYS with the models used to simulate SHRT 17 and SHRT 25 earlier (7). There are no adjustable parameters and so the calculation can be considered a test of the predictive capability of the code. Figure 6 compares the predicted reactivity history for this experiment with the observed values. Figure 7 gives the measured temperature response of the XX09 thermocouples for the coolant at the top of the fuel location, compared to the SASSYS computed coolant temperature at that location. In both cases agreemennt is good. Other plant variables have been compared and agreement is good there also.

Fig. 6

Fig. 7

On the basis of these and other calculations it is believed that a good understanding of the passive safety features of the advanced LMR's exists and is embodied in SASSYS.

SUMMARY

Passive safety is being built in to the design of the Advanced Liquid Metal Reactors in the U.S. today. This is achieved in a number of ways some of which have been discussed in this paper. These include using a pool configuration, metallic fuel, enhanced feedback and passive Decay Heat Removal Systems. These features provide naturally corrective responses to plant and reactor upset conditions as a result of their inherent hydraulic, thermal, neutronic, and mechanical properties. The potential effectiveness of these design concepts has been investigated at Argonne National Laboratory with full scale experiments and integrated analyses.

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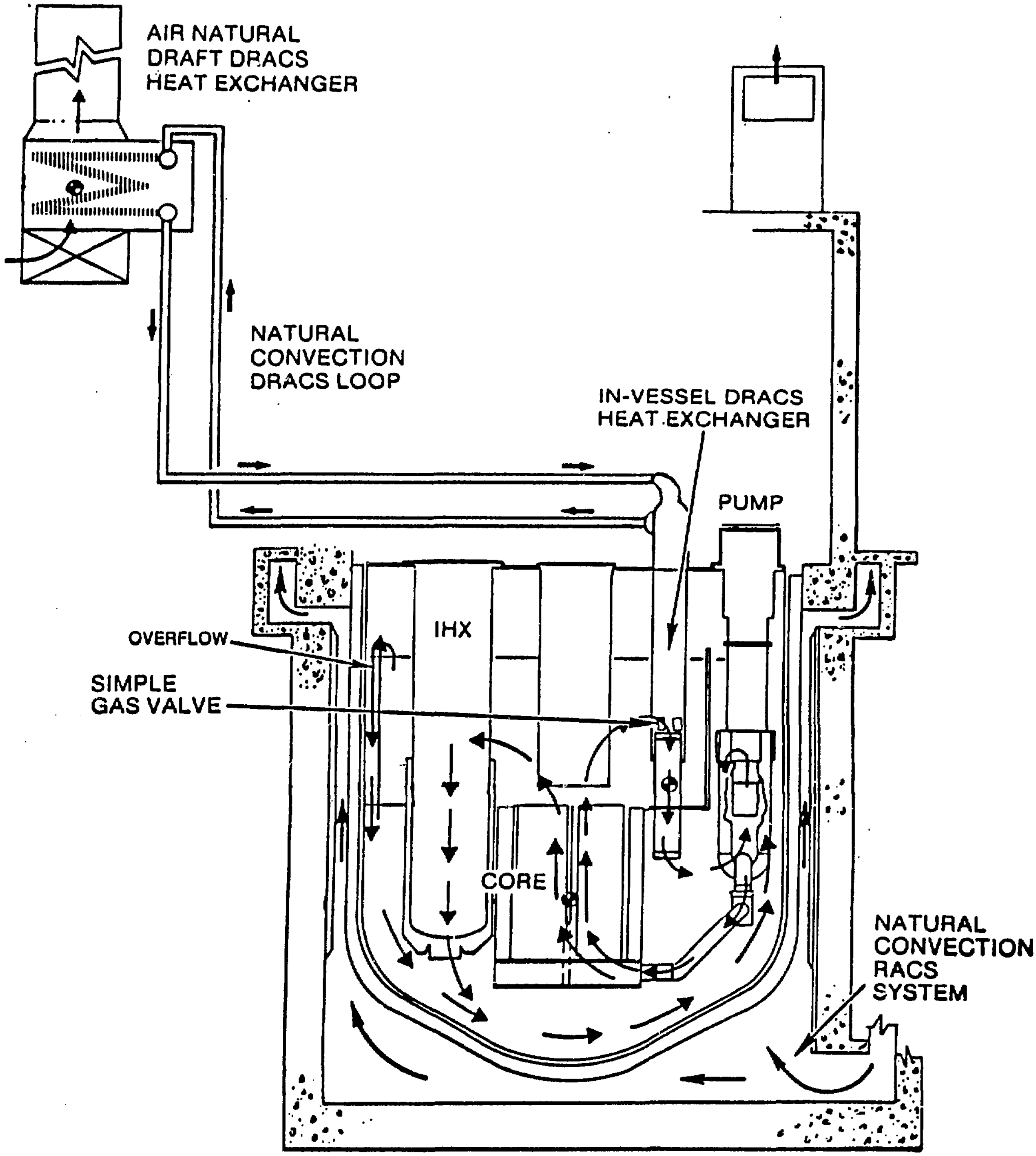
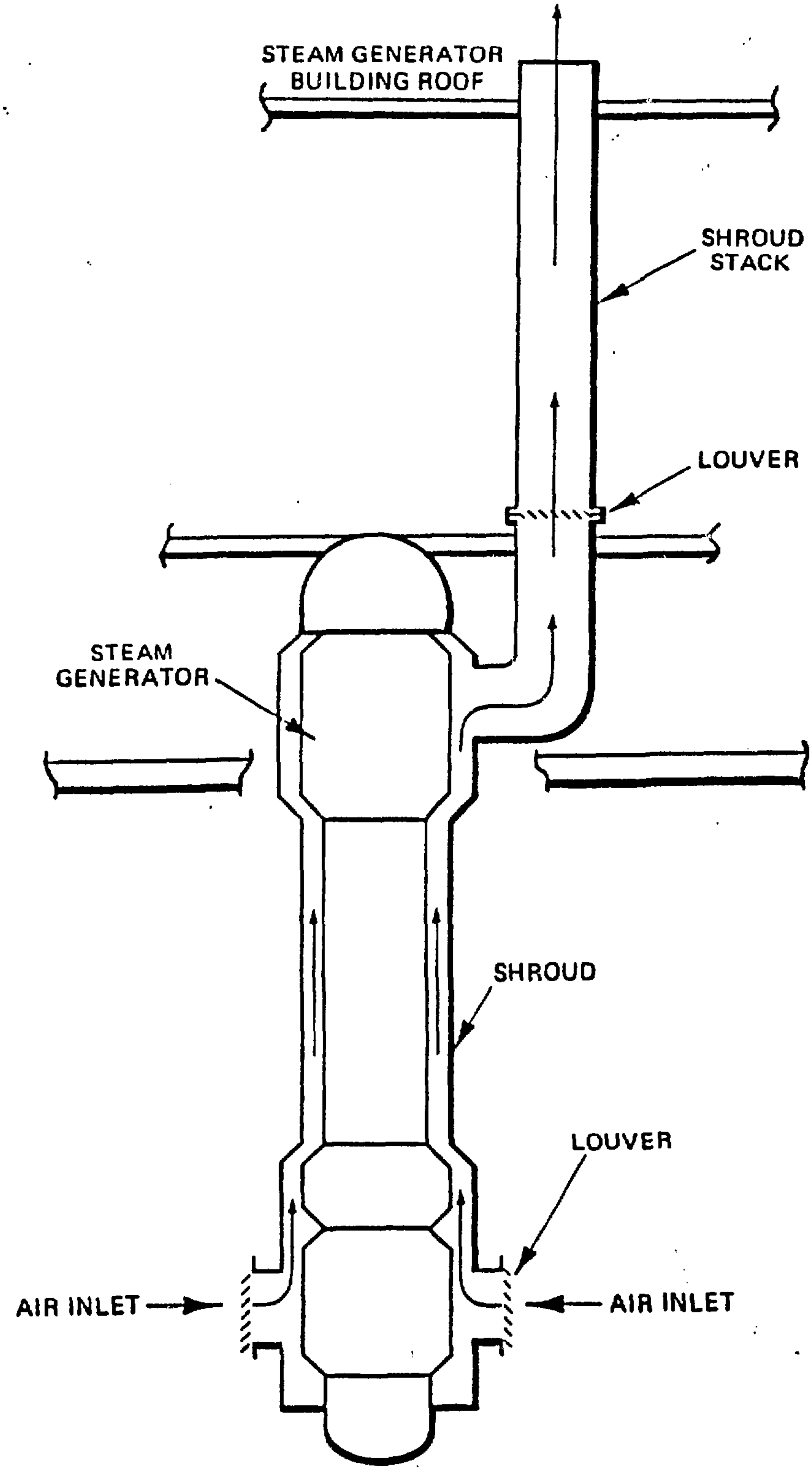
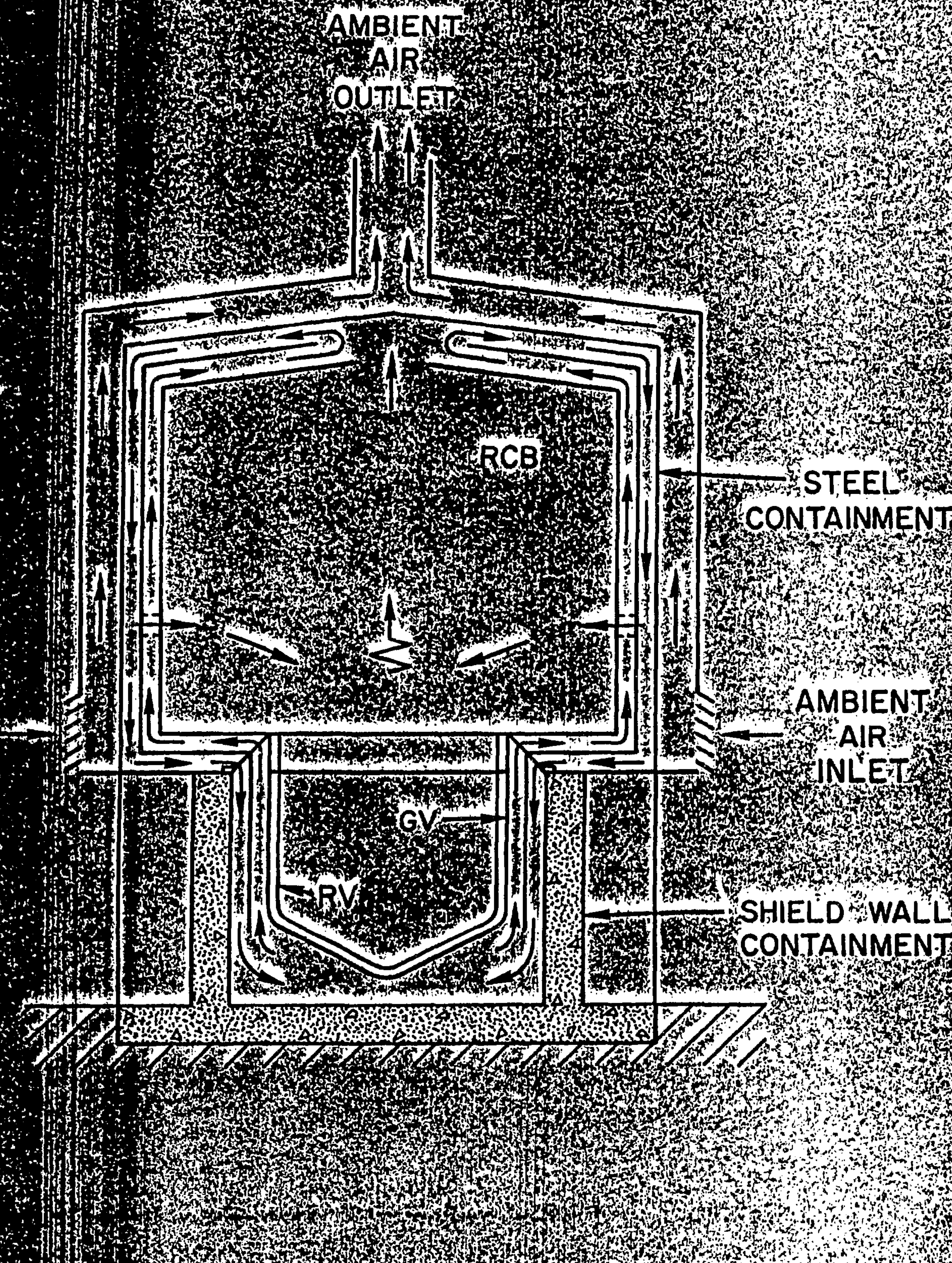


1a DRACS + RACS


STEAM GENERATOR
BUILDING ROOF
SHROUD
STACK
LOUVER
STEAM
GENERATOR
SHROUD
LOUVER
AIR INLET
AIR INLET

16 ACS



1c Double RACS

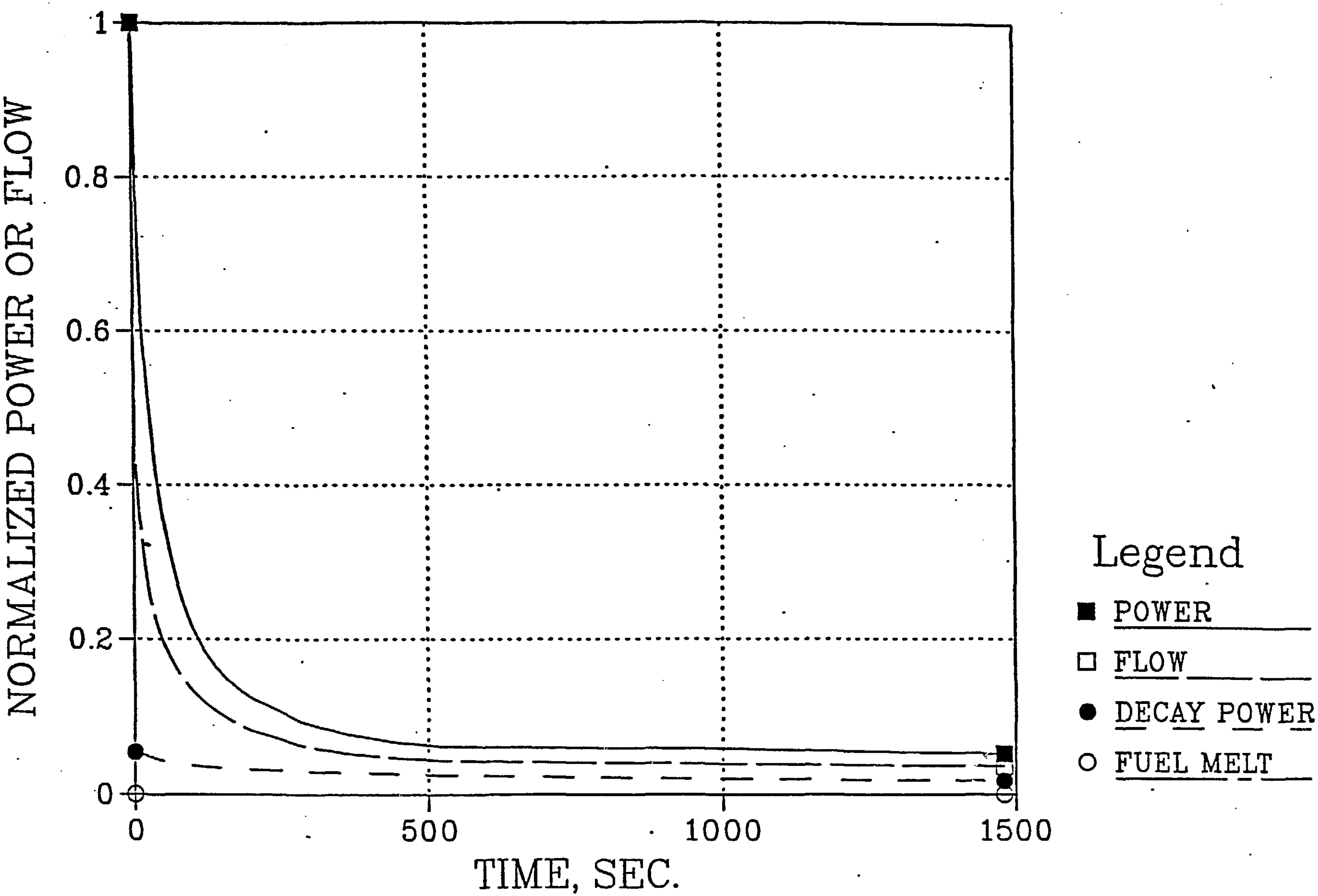


Fig. 2. Normalized Power and Flow During an Unprotected LOF Transient in PRISM with an EOEC Core

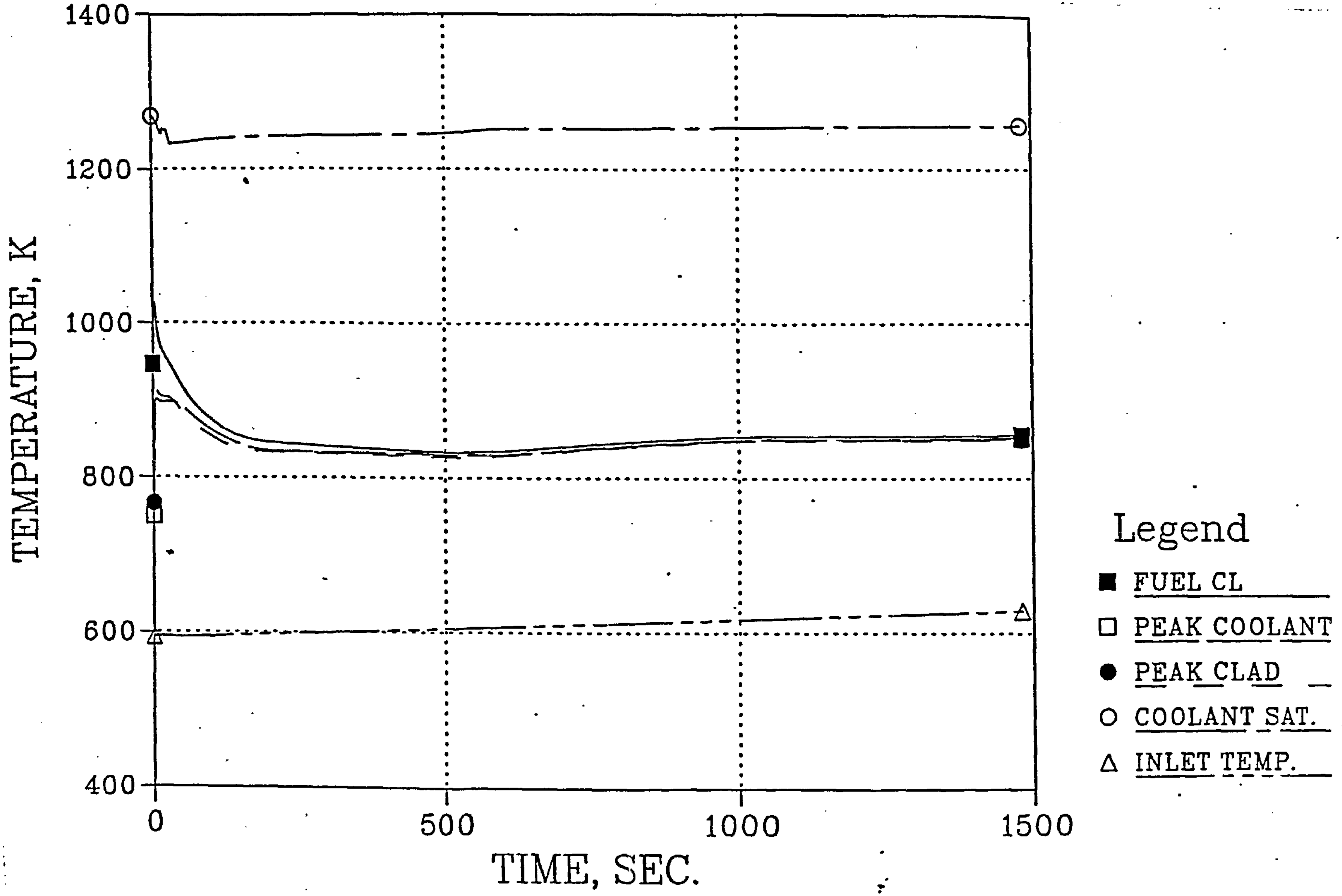
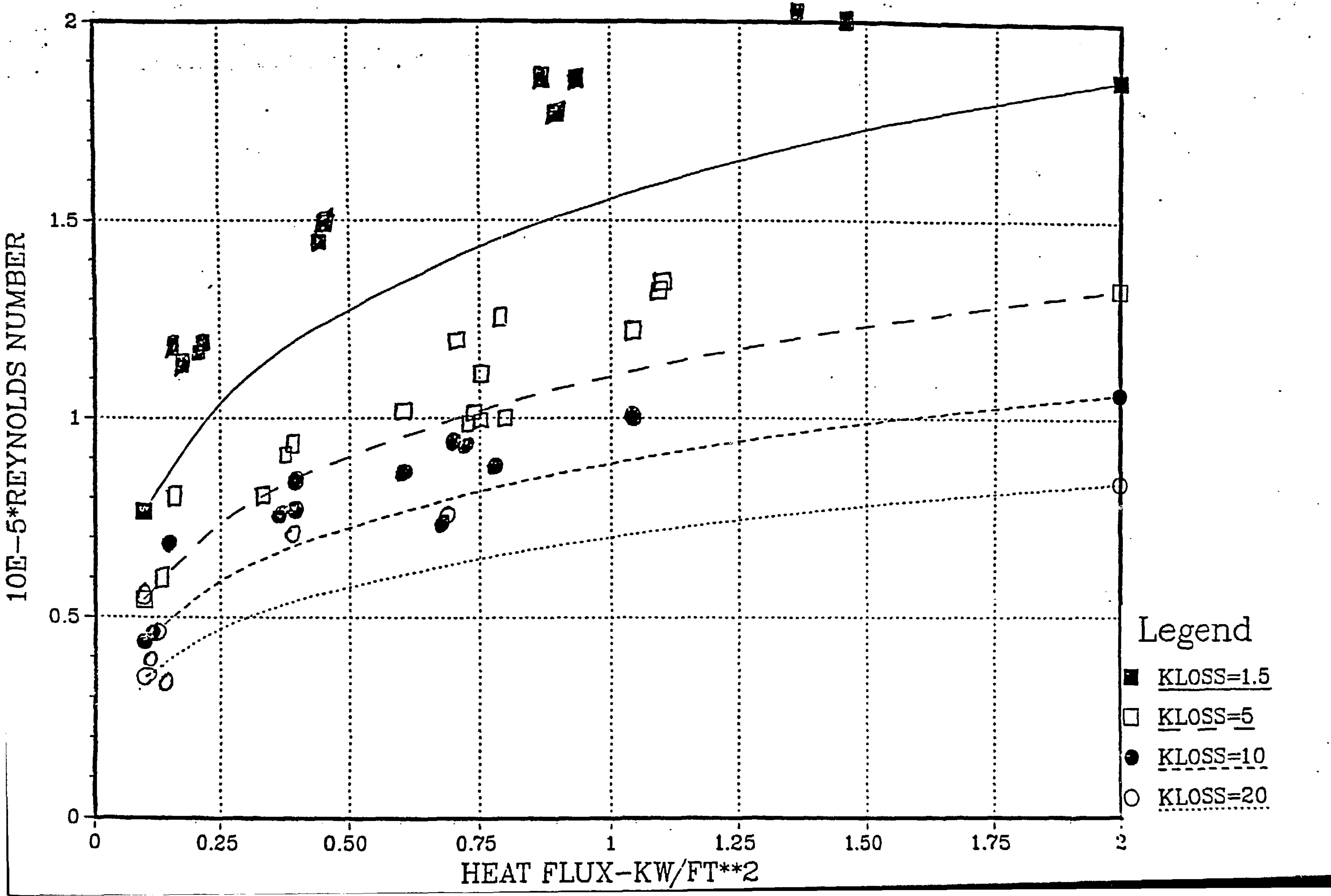


Fig. 3. Core Temperatures During an Unprotected LOF Transient in PRISM with an EOEC Core.

FIGURE ~~3-2~~ 4 NSTF PERFORMANCE FOR VARIOUS VALUES OF APPLIED HEAT FLUX AIR INLET TEMP=70 DEGF—EMISSIVITY=0.7



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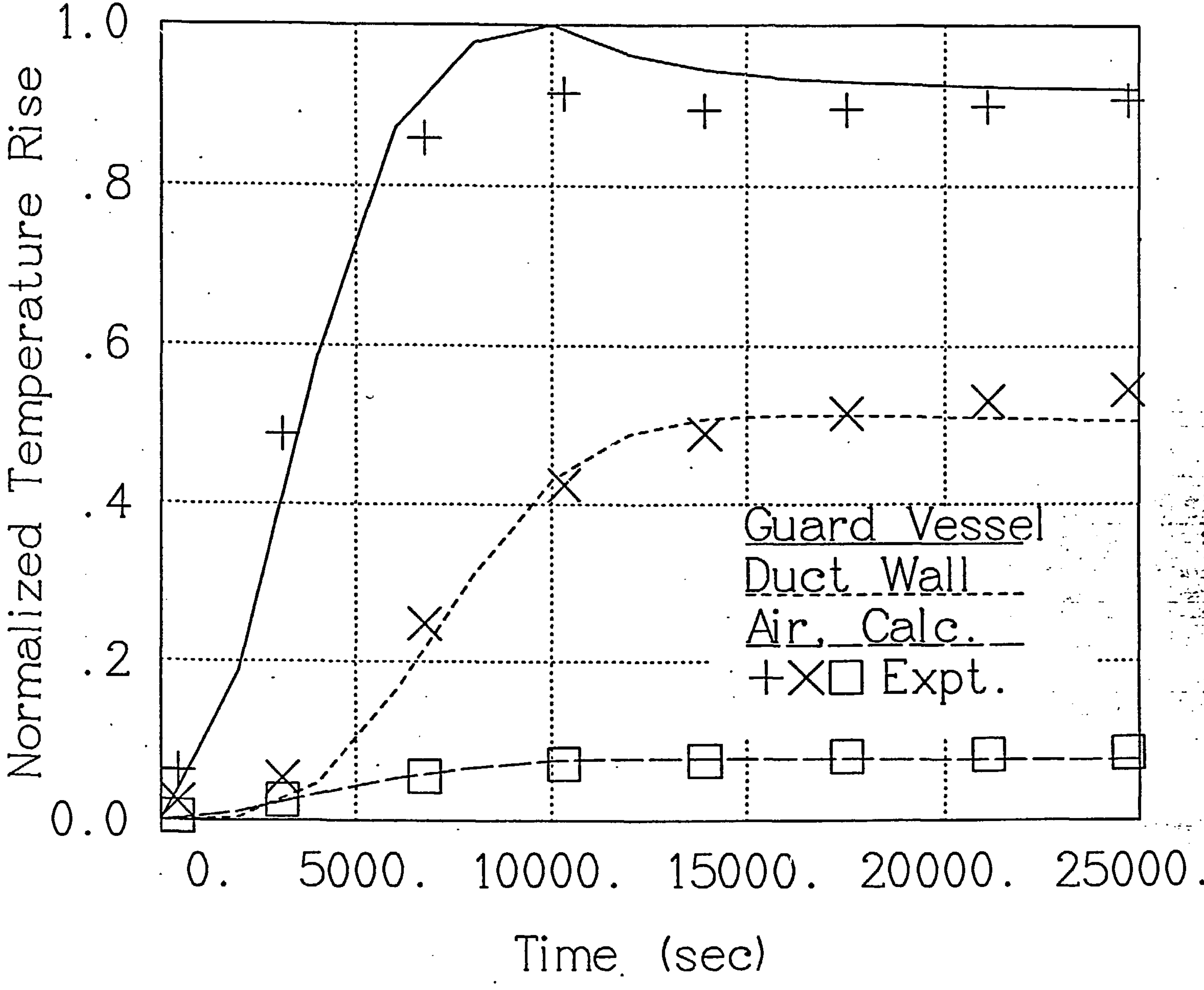
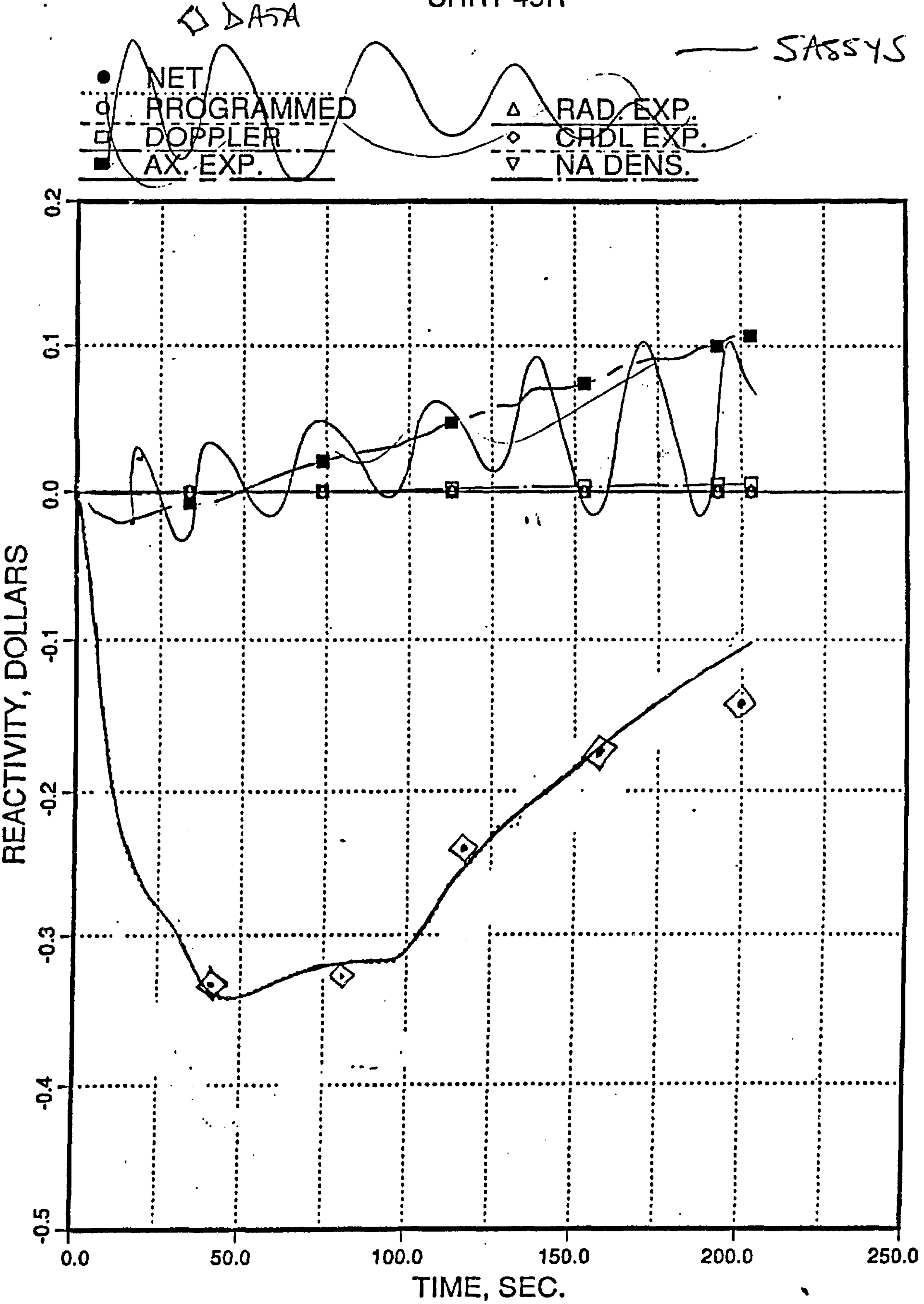


1.0
.8
.6
.4
.2
0.0
Normalized Temperature Rise
0.
5000.
10000.
15000.
20000.
25000.
Time (sec)
Guard Vessel
Duct Wall
Air, Calc.
+×□ Expt.

SHRT 45R



◇ DATA
— SASSYS
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PROGRAMMED
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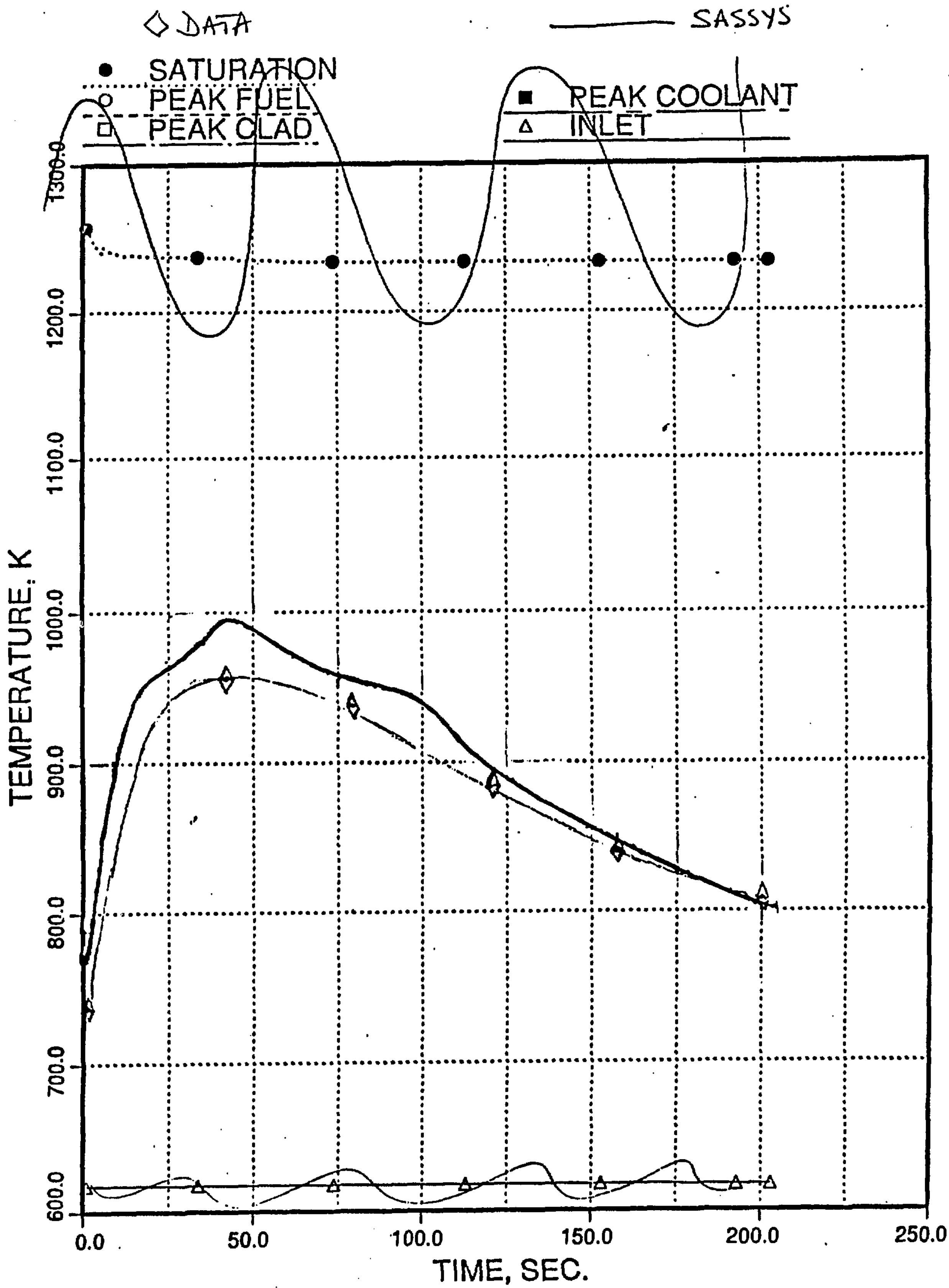
RAD. EXP.
CRDL EXP.
NA DENS.
0.2
0.1
0.0
-0.1
-0.2
-0.3
-0.4
-0.5
REACTIVITY, DOLLARS
0.0
50.0
100.0
150.0
200.0
250.0
TIME, SEC.

6

SHRT 45R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PEAK CLAD
PEAK COOLANT
INLET
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1200.0
1100.0
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900.0
800.0
700.0
600.0
TEMPERATURE, K
0.0
50.0
100.0
150.0
200.0
250.0
TIME, SEC.

7