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Title: LOSS OF OFFSITE POWER TRANSIENTS IN THE UPDATED PIUS 600 ADVANCED REACTOR DESIGN

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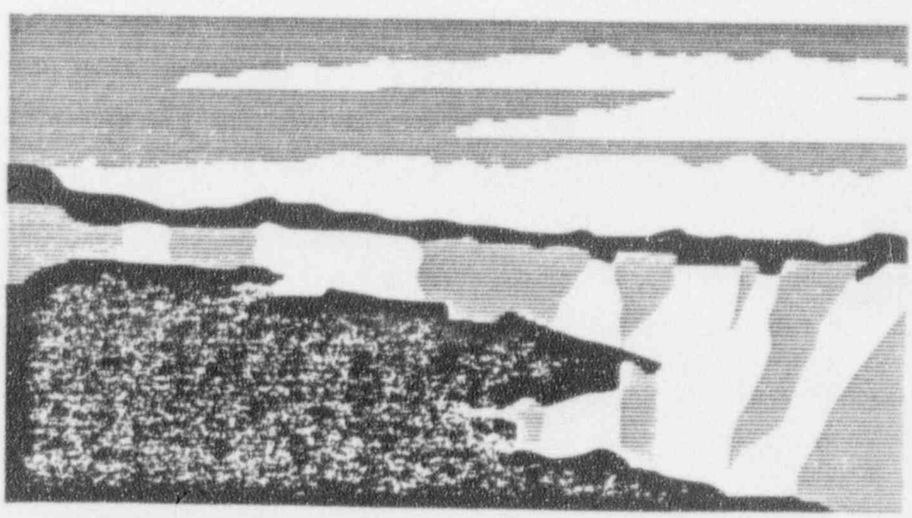

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LOSS OF OFFSITE POWER TRANSIENTS IN THE UPDATED PIUS 600 ADVANCED REACTOR DESIGN*

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ABSTRACT

The PIUS advanced reactor is a 640-MWe pressurized water reactor developed by Asea Brown Boveri (ABB). A unique feature of the PIUS concept is the absence of mechanical control and shutdown rods. Reactivity is controlled by coolant boron concentration and the temperature of the moderator coolant. As part of the preapplication and eventual design certification process, advanced reactor applicants are required to submit neutronic and thermal-hydraulic safety analyses over a sufficient range of normal operation, transient conditions, and specified accident sequences. Los Alamos is supporting the US Nuclear Regulatory Commission's preapplication review of the PIUS reactor. A fully one-dimensional (1D) model of the PIUS reactor has been developed for the Transient Reactor Analysis Code, TRAC-PF1/MOD2. Early in 1992, ABB submitted a Supplemental Information Package describing recent design modifications. An important feature of the PIUS Supplement design was the addition of an active scram system that will function for most transient and accident conditions. However, the active scram system requires operation of the reactor coolant pumps and these pumps are not available following a loss of offsite power. Using TRAC and the 1D PIUS model, baseline calculations of the PIUS Supplement design were performed for a loss-of-offsite power initiator. In addition, sensitivity studies were performed to explore the robustness of the PIUS concept to severe off-normal conditions following a loss of offsite power. The sensitivity studies have examined flow blockage and boron dilution events, and these studies provide insights into the robustness of the design.

INTRODUCTION

The PIUS advanced reactor is a four-loop, Asea Brown Boveri (ABB) designed pressurized water reactor with a nominal core rating of 2000 MWt and 640 MWe.[1] A primary design objective was to eliminate any possibility of a core degradation accident. A schematic of the basic PIUS reactor arrangement is shown in Fig. 1. Reactivity is controlled by coolant boron concentration and temperature, and there are no mechanical control or shutdown rods. The core is submerged in a large pool of highly borated water, and the core is in continuous communication with the pool water through pipe openings called density locks. The density locks provide a continuously open flow path between the primary system and the reactor pool. The reactor coolant pumps (RCPs) are operated so that there is a hydraulic balance in the density locks between the primary coolant loop and the pool, keeping the pool water and primary coolant separated during normal operation. Hot primary system water is stably stratified over cold pool water in the density locks. PIUS contains an active scram system. The active scram system consists of four valved lines, one for each primary coolant loop, connecting the reactor pool to the inlets of the reactor coolant pumps. Although the active scram piping and valves are safety-class equipment, operation of the nonsafety-class reactor coolant pumps is required for effective delivery of pool water to the primary system. PIUS also has a passive scram system that functions should one or more of the

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RCPs lose their motive power, thereby eliminating the balance between the primary coolant loop and the pool, and activating flow through the lower and upper density locks. Highly borated water from the pool enters the primary coolant via natural circulation, and this process produces a reactor shutdown. The reactor pool can be cooled by either an active, nonsafety-class system, or a fully passive, safety-class system.

As part of the preapplication and eventual design certification process, advanced reactor applicants are required to submit neutronic and thermal-hydraulic safety analyses over a sufficient range of normal operation, transient conditions, and specified accident sequences. ABB submitted a Preliminary Safety Information Document (PSID)[2] to the US Nuclear Regulatory Commission (NRC) for preapplication safety review in 1990. Early in 1992, ABB submitted a Supplemental Information Package to the NRC to reflect recent design modifications.[3] The ABB safety analyses are based on results from the RIGEL code,[4] a one-dimensional (1D) thermal-hydraulic system analysis code developed at ABB Atom for PIUS reactor analysis. An important feature of the PIUS Supplement design was the addition of the previously described active scram system that will function for most transient and accident conditions. However, this system cannot meet all scram requirements because the performance of the active scram system depends on the operation of the RCPs. Thus, the passive scram system of the original PSID design was retained. Review and confirmation of the ABB safety analyses for the PIUS design constitute an important activity in the NRC's preapplication review. Los Alamos is supporting the NRC's preapplication review of the PIUS reactor. This paper summarizes the results of Transient Reactor Analysis Code (TRAC)[5] baseline calculations of the PIUS Supplement design for a loss of offsite power (LOSP) transient in which the passive scram system must function due to the unavailability of the active scram system. Sensitivity studies were performed to explore the robustness of the PIUS concept to severe off-normal conditions following a LOSP. The TRAC calculations were performed with a fully 1D, four-loop model. Core neutronic performance was modeled with the TRAC point kinetics model.

TRAC ADEQUACY FOR THE PIUS APPLICATION

The TRAC-PF1/MOD2 code[5] was used for each calculation. The TRAC code series was developed at Los Alamos to provide advanced, best-estimate predictions for postulated accidents in pressurized-water reactors. The code incorporates four-component (liquid water, water vapor, liquid solute, and noncondensible gas), two-fluid (liquid and gas), and nonequilibrium modeling of thermal-hydraulic behavior. TRAC features flow-regime dependent constitutive equations, component modularity, multidimensional fluid dynamics, generalized heat structure modeling, and a complete control systems modeling capability. The code also features a three-dimensional, stability-enhancing, two-step method, which removes the Courant time-step limit within the vessel solution. Many of the features just identified have proven useful in modeling the PIUS reactor.

It is important that the issue of code adequacy for the PIUS application be addressed. If the TRAC analyses were supporting a design certification activity, a formal and structured code-adequacy demonstration would be desirable. One such approach would be to (1) identify representative PIUS transient and accidents sequences, (2) identify the key systems, components, processes and phenomena associated with the sequences, and (3) conduct a bottom-up review of the individual TRAC models and correlations and a top-down review of the total or integrated code performance relative to the needs assessed in steps 1 and 2. The bottom-up review determines the technical adequacy of each model by considering its pedigree, applicability, and fidelity to experimental separate effect or component data. The top-down review determines the technical adequacy of the integrated code by considering code applicability and fidelity to data taken in integral test facilities.

Because the NRC was conducting a preapplication rather than a certification review, the NRC and Los Alamos concluded that a less extensive demonstration of code adequacy would suffice. Steps 1 and 2 were performed and documented in Ref. 6. A bottom-up review specific to the PIUS reactor was not conducted. However, the bottom-up review of TRAC conducted for another reactor type[7] provided some confidence that the basic TRAC models and correlations are adequate and also identified some needed code modifications. A complete top-down review was not conducted. However, the ability of TRAC to model key PIUS systems, components, processes and phenomena was demonstrated in an assessment activity[8] using integral data from the ATLE facility.[4] ATLE is a 1/308 volume scale integral test facility that simulates the PIUS reactor. Key safety features and components were simulated in ATLE, including the upper and lower density locks, the reactor pool, pressurizer, core, riser, downcomer, reactor coolant pumps, and steam generators. Key processes were simulated in ATLE including natural circulation through the upper and lower density locks, boron transport into the core (simulated with sodium sulfate), and control of the density lock interface. Core kinetics were indirectly simulated through a point kinetics computer model that calculated and controlled the core power based upon the core solute concentration, coolant temperature, and heater rod temperature. The TRAC-calculated results were in reasonable agreement with the experimental data. Reasonable agreement means the code provided an acceptable prediction. All major trends and phenomena were correctly predicted. However, the calculated results were frequently outside the data uncertainty. There were two major discrepancies. First, TRAC-PF1/MOD2 calculates a too rapid diffusion of the solute. A higher order numerical method has been implemented for the solute field in the MOD3 version of the code, but this capability is not yet available in the NRC-version of the code (MOD2). MOD3 calculations of ATLE transients better represent the solute characteristics observed in the experiment. Second, in one case, a two-pump trip scram experiment, the initial surge flow from the reactor pool through the lower density lock is underpredicted by about 25%. Although the underprediction would influence the early course, but not the final or end state, of a similar transient in the PIUS reactor, the discrepancy is, nevertheless, of concern. This discrepancy will be discussed further in the section of this paper describing the PIUS baseline LOSP event.

Benchmarking against another validated code is a second approach to demonstrating adequacy. In this paper, we will provide qualitative comparisons of TRAC and RIGEL calculated results for a LOSP event. ABB did not published an updated analysis of the LOSP event in the PSID Supplement. Therefore, it is not possible to provide quantitative code-to-code comparisons for the LOSP transient. However, it is possible to provide a qualitative comparison because the same processes and phenomena occur in both the original PSID design and the later PSID Supplement design. Direct code-to-code comparisons have been prepared for other transients for which ABB calculations of the PSID Supplement design are available.[9,10]

TRAC includes the capability for multidimensional modeling of the PIUS reactor. Indeed, multidimensional analyses of the passive scram via trip of one reactor coolant pump were completed for the original PSID design.[11] That study concluded that well-designed orificing of the pool water inlet pipes would minimize multidimensional effects. As a result of these earlier studies, we have concluded that 1D modeling has the potential for adequately representing many PIUS transients and accidents. We believe that the LOSP event is adequately characterized with 1D modeling. We do note a reservation. The most important physical processes in PIUS are related to reactor shutdown because the PIUS reactor does not contain control and shutdown rods. Coupled core neutronic and thermal-hydraulic effects are possible, including multidimensional interactions arising from nonuniform introduction of boron across the core. ATLE does not simulate multidimensional effects. The RIGEL thermal-hydraulic model is 1D and a point kinetics model is used. Although both 1D and multidimensional TRAC thermal-hydraulic models have been applied for selected accident analyses, core neutronics are simulated with a point kinetics model. At the present time, it is not known whether coupled core neutronic and thermal-hydraulic effects and multidimensional effects are important. We offer this important reservation along with the results that follow.

TRAC MODEL OF THE PIUS REACTOR

Figures 2 and 3 display the reactor vessel and coolant loop components of the TRAC 1D model. The four-loop TRAC model consists of 74 hydrodynamic components (727 computational fluid cells) and one heat-structure component representing the fuel rods. The reactor power is calculated with a space-independent point-kinetics model. The hydrodynamic model has 8 components in each coolant loop and 16 components for the reactor vessel, with the remaining 26 components representing the pool, steam dome, density locks, and pressurizer line. The TRAC 1D model is more finely noded than the RIGEL model because of Los Alamos' modeling preferences, but no particular merit is attributed to the finer noding.

The TRAC steady-state and transient calculations were performed with TRAC-PF1/MOD2, version 5.3.05. The TRAC-calculated and PSID Supplement steady-state values are tabulated below for comparison.

	TRAC	PSID Supplement
Core mass flow (kg/s)	12822	12880
Core bypass flow (kg/s)	200.2	200
Loop flow (kg/s)	3255	3266
Cold-leg temperature (K)	531.0	527.1
Hot-leg temperature (K)	560.7	557.3
Pressurizer pressure (MPa)	9.5	9.5
Steam exit pressure (MPa)	4.0	4.0
Steam exit temperature (K)	540.3	543
Steam flow superheat (°C)	15.3	20
Steam and feedwater mass flow (kg/s)	243	243

Additional initial conditions for the calculated transients are as follows, except where otherwise noted for the sensitivity studies. The reactor is operating at beginning of cycle (BOC) with a primary loop boron concentration of 375 parts per million (ppm) and 100% power. The boron concentration in the reactor pool is initially 2200 ppm. If the active scram system is activated, the scram valves open over a period of 2 s following event initiation, remain open for 180 s, and close over a period of 30 s. The feedwater pumps are tripped as the scram is initiated and the feedwater flow rate decreases linearly to zero in 20 s. The steam pressure on the steam generator secondary side is kept constant at 3.88 MPa (steam drum).

BASELINE LOSP TRANSIENT

With the loss of motive power to all RCPs, the pumps coast down and the loop flows rapidly decrease, develop a small reverse flow beginning at 50 s, and effectively stagnate by 300 s (Fig. 4, Frame 14). The steam generators continue to accept heat from the primary for 70 s, after which primary-to-secondary heat transfer is terminated. The hydraulic balance in the density locks between the primary coolant loop and the pool is upset with the tripping of the RCPs. There is a rapid inflow of water into the primary system through the lower density lock, and a corresponding but lower flow from the primary back to the reactor pool through the upper density lock (Fig. 5, Frame 7). The difference between the two flows replaces the volumetric shrinkage of the primary system coolant as fluid temperatures decrease. The lower density lock flow peaks at 1225 kg/s shortly after the LOSP initiator, and decreases until the flow rate required to remove core decay heat (about 200 kg/s) is established. The large influx of water passing from the reactor pool into the primary through the lower density lock, rapidly lowers the primary system boron concentration (Fig. 6, Frame 11) and temperatures (Fig. 7, Frame 12) at the core inlet to the conditions in the pool. The rapid decrease in fuel and coolant temperatures lead to positive reactivity insertions. However, the negative reactivity insertion by the boron is larger than the positive contributions, and the total reactivity is negative (Fig. 8, Frame 6). The decline in reactor power to a decay heat level is rapid, as shown in Fig. 9 (Frame 3).

A RIGEL calculation of the LOSP sequences for the original PSID design was reported by ABB in Ref. 2. ABB did not update the analysis of the LOSP event in the PSID Supplement. There are significant differences between the original PSID design and the PSID Supplement design. For example, the outer diameter of the lower density lock was reduced from slightly over 1.3 m to just under 1 m, resulting in reduced lower density lock flows. Given such design differences, it is not possible to provide quantitative code-to-code comparisons. A more direct code-to-code benchmark for the LOSP transient is desirable for this transient but not possible. However, it is instructive to compare the main qualitative features of the calculations for the two designs. These are similar in all key respects. The RIGEL calculation of the LOSP event for the original PSID design is characterized by these features. Following the coastdown of the RCPs, the loop flows reverse. Steam generator primary-to-secondary heat transfer is terminated early in the transient. There is a rapid inflow of water into the primary system through the lower density lock, and a corresponding but lower flow from the primary back to the reactor pool through the upper density lock. The large influx of water passing from the reactor pool into the primary through the lower density lock rapidly lowers the primary system boron concentration and temperatures at the core inlet to the conditions in the pool. The decline in reactor power to decay heat levels is rapid.

One test in the ATLE facility was very similar to a LOSP transient in the PIUS reactor. This test was a total loss of pumping power when both recirculation pumps are tripped. As the recirculation pump speeds decrease, the pressure balance in the lower density lock is disturbed and the pool water enters the primary system from the pool. The primary flow decreases rapidly and the core outlet temperature rises rapidly. The increased solute concentration in the primary system is measured and this measurement, along with the measurements of coolant temperature, are supplied to a computational point kinetics model where a new power setting for the electrically heated core is calculated. Key TRAC-calculated results of the assessment calculation are presented in Figs. 10-12, the results are compared with the ATLE data, and results are calculated with the RIGEL code. A comparison of the measured and code-calculated lower density lock flows is presented in Fig. 10a. The TRAC-calculated peak lower density lock flow is about 25% less than measured. The TRAC-calculated natural circulation flow rate at the end of the test is about 12% less than measured. The RIGEL-calculated peak flow is within 2% of the measured value. The RIGEL-calculated natural circulation flow rate at the end of the test is about 30% greater than measured. The TRAC-calculated heater rod power is compared with the measured and RIGEL-calculated values in Fig. 11. The start of the decrease in the TRAC-calculated power is delayed, but then falls at a faster rate than measured. The TRAC-calculated core outlet temperature is compared with the measured and RIGEL-calculated values in Fig. 12. The relative differences in core outlet temperatures are consistent with the density lock flow and core power discrepancies discussed previously. We performed many sensitivity studies to identify the source of the underpredicted lower density lock flow. One study included using the measured heater rod power as an input. The lower density lock flow was generally insensitive to all parametric variations with the exception of one. We determined that a small increase (15%) in the minimum flow area in the flow path between the riser and the upper density lock led to reasonable agreement with the data (Fig. 10b). We believe a 15% error in our estimation of the flow area is possible since the flow area was calculated using scaled measurements from a facility drawing. We have concluded that our modeling of the ATLE heater rod control system is incorrect. In both the ATLE test and the TRAC calculation, the test solute equivalent of boron is the only shutdown mechanism. The only source of the solute (boron) is the entry of pool flow into the primary through the lower density lock. The TRAC-calculated heater rod power decays much more rapidly than the measured power. However, TRAC calculates a smaller lower density lock flow, which should lead to a slower power decline. We have concluded that our modeling of the ATLE heater rod power control model is incorrect. We have not been able to identify the source of the error in subsequent discussions with ABB personnel.

SENSITIVITY CASES

Sensitivity studies were performed to explore the robustness of the PIUS design to severe off-normal conditions following active-system trips. The most severe of these conditions are very low probability events. Calculations were performed to examine the effect of lower density lock blockage fractions of 75% and 100%. We first review the 75% blockage case and compare the calculated results with those in the baseline LOSP transient. The flows through the lower density lock from the pool to the primary are shown in Fig. 13 (Frame 7). The peak lower density lock flow of 450 kg/s compares with a peak flow of 1225 kg/s for the baseline transient. This has several consequences. The rate at which boron is introduced into the core is delayed. The core inlet boron concentration reaches the pool value of 2200 ppm about 100 s later than in the baseline. The core inlet temperature follows a similar trend (Fig. 14, Frame 12). The core outlet temperature reaches the saturation temperature shortly after the start of the transient and there is a brief period of voiding in the core. The voiding lasts only a few seconds and there is no core dryout. The decline in reactor power to decay heat levels is only slightly slower in the blockage case. The same core power levels are reached after approximately 100 s.

We next review the 100% or complete blockage of the lower density lock and compare the calculated results to those in the baseline LOSP transient. Although this is a very challenging transient with regards to phenomena, the PIUS reactor successfully accommodates this transient. The density lock flows are shown in Fig. 15 (Frame 7). The lower density lock is blocked to flow per the problem specification. The upper density lock is open to the reactor pool and the interface is agitated for the first 375 s. However, the net flow from the primary to the pool is negligible (about 1000 kg). During this interval the primary pressurizes (Fig. 16, Frame 2) and heats up (Fig. 17, Frame 12). The safety valves open repeatedly after the opening setpoint of 12.3 MPa is reached. Some voiding occurs in the core; the core average voiding peaks at slightly under 7%. With the exception of a brief interval between 200 and 300 s, some void is present in the core throughout the calculated transient, but there is no core dryout. Because the active scram system does not function when the RCPs are inoperable and there is no flow from the reactor pool to the primary system through the upper density lock, the only early negative reactivity insertion is via the void (Fig. 18, Frame 6). The core power is reduced (Fig. 19, Frame 3) but remains above 500 MWt until 200 s. Primary-to-secondary heat transfer continues in the steam generators until they dry out at 235 s. The core inlet temperature increases rapidly following steam generator secondary dryout, and the increasing moderator temperature inserts sufficient negative reactivity to further reduce the power. At 375 s, the upper density lock activates after the pressurizer liquid level has swelled to the point that primary coolant spills into the standpoints and a flow path from the primary to the reactor pool is established (Fig. 20, Frame 1-extra). Flow through the upper density lock replenishes the primary coolant flowing through the standpipes. By 600 s, a stable primary system flow circulation has been established. This circulation consists of a primary and a secondary circulation. The primary circulation follows the normal flow through the primary loops. The secondary circulation is the means by which boron from the reactor pool enters the primary system. The flow entering the upper density lock merges with a larger recirculation flow passing downward through the upper density lock annulus. The merged flow passes into the riser through the overlapping joint (gap) between the riser and the upper density lock annulus. Once in the riser, the bulk of the flow passes into the hot legs and through the steam generators, stationary RCPs, and cold legs; it passes through the downcomer, and then through the core into the riser. The riser flow entering the hot leg plenum splits. As previously mentioned, the bulk of the flow enters the hot legs. The larger fraction of the remainder passes downward into the upper density lock annulus and a smaller flow passes upward into the pressurizer pool.

Additional sensitivity calculations were performed to examine the effect of pool boron concentrations. ABB has stated that a reactor scram will occur if the pool boron concentration decreases to 1800 ppm (Ref. 3). ABB further notes that at a pool boron concentration of 1000 ppm, a critical core could be achieved at cold shutdown conditions and BOC. Sensitivity calculations were performed at these two pool boron concentrations. We first review the 1800 ppm case and compare the calculated results with those in the baseline LOSP transient. The differences between the calculated baseline and 1800 ppm pool concentration case are small. The lower and upper density lock flows are nearly identical. However, the core inlet boron concentration can only increase to the concentration in the pool, or 1800 ppm. Thus, the negative reactivity inserted by the boron, and the total reactivity, are less than the baseline. However, the difference is small, and the core power decreases at a rate only slightly slower than in the baseline. The decline in reactor power to decay heat levels is rapid. The PIUS reactor successfully accommodates a LOSP initiator with pool boron concentration at 1800 ppm or 400 ppm below the nominal value.

We next review the case with the pool boron concentration at 1000 ppm and compare the calculated results to those in the baseline LOSP transient. The phenomena of the LOSP transient with the pool boron concentration at 1000 ppm are markedly different than either the baseline or the 1800 ppm case. The lower and upper density lock flows are similar to those in the baseline. However, the core inlet boron concentration can only increase to the concentration of the boron in the pool or 1000 ppm (Fig. 21, Frame 11). The negative reactivity inserted by the boron is sufficient to produce an initial reduction in core power but is insufficient to keep the core subcritical (Fig. 22, Frame 6). The core power oscillates once between 1000 and 250 MWt and then settles to a near constant value of 500 MWt by 200 s (Fig. 23, Frame 3). The primary pressure begins to increase shortly after the LOSP but first slows and then stabilizes as the power decreases from 1000 to 250 MWt (Fig. 24, Frame 2). However, once the steam generators cease to function as heat sinks at 85 s and the power again increases from 250 to 500 MWt, the primary pressure resumes its increase. The pressure rises to 12.3 MPa and the safety valves open. The safeties continue to cycle to the end of the calculated transient at 1200 s. Although a stable condition has been reached, the power level remains high at 500 MWt and this energy is carried to the reactor pool. The reactor pool is cooled by both a non-safety active system and a completely passive safety-grade system. However, to reach stable, decay heat levels, additional boron must be inserted at some point into the primary system.

SUMMARY OF OBSERVATIONS

1. The passive scram system successfully accommodates the baseline LOSP transient. The predicted key trends and processes for the baseline transients can be expected to occur in PIUS to the extent that they are accurately represented in 1D and by point kinetics models.

2. The PIUS core, as presently conceived, has inherent, compensating neutronic shutdown mechanisms. When highly borated pool water enters the primary through the lower density locks under baseline conditions, the negative reactivity associated with the boron is the primary mechanism for decreasing core power to decay heat levels. The moderator and fuel temperature contributions reactivity are positive in such circumstances. There is no voiding. However, negative reactivities are inserted via both the moderator temperature and the void when the boron entering the core is not sufficient to prevent fuel and coolant temperature increases. Neither operator nor active system actions are needed to accomplish

reactor shutdown, even for LOSP initiators combined with very low probability flow path blockage or pool dilution occurrences.

3. The PIUS concept, as presently conceived, has multiple flow paths between the primary system and reactor pool. Following a LOSP initiator, a natural circulation path would be established with reactor pool water entering the primary system through the lower density lock and reentering the pool through the upper density lock. However, alternate flow paths exist should even complete blockage of one or other of the density locks occur. Neither operator nor active system actions are needed to accomplish reactor shutdown, even for LOSP initiators combined with very low probability flow path blockage occurrences.

4. Our confidence in the baseline simulations is enhanced by the assessment activity performed using ATLE data. The ATLE processes and phenomena were correctly predicted by TRAC. However, there are quantitative discrepancies between key TRAC-calculated parameter values and the ATLE data, and we would like to better understand the reasons for these differences. More effort is required to identify whether the reasons for the discrepancies lie in our knowledge of the facility, modeling decisions made in preparing the TRAC input model of ATLE, or deficiencies in the TRAC models and correlations.

5. Our confidence in the baseline simulations is enhanced by the qualitative benchmark activity, which shows the RIGEL and TRAC-calculated results for the LOSP transient display the same processes and phenomena. The quantitative benchmark activity for other transients further increases our confidence.[10,11] The RIGEL and TRAC-calculated results display many areas of similarity and agreement. However, there are also differences in the details of the transients and accidents calculated by the two codes, and we would like to better understand the reasons for these differences. It is desirable that the reasons for these differences be explored if the PIUS reactor progresses to the design certification stage. Although it is desirable to understand the reasons for the differences, we have concluded that they affect the detailed course of the predicted sequences rather than the predicted end states of the transient and accident sequences.

6. Although the sensitivity calculations move beyond both the assessment activity using ATLE data and the code-to-code benchmark activity with RIGEL, the PIUS design appears to accommodate marked departures from the baseline transient and accident conditions, including very low probability combination events. The studies of extremely low pool boron concentrations and complete blockages of the lower density lock are characteristic of very low probability events, yet these events appear to be successfully accommodated. No phenomenological "cliffs" were encountered for the sensitivity studies conducted.

7. At the present time, it is not known whether coupled multidimensional core neutronic and thermal-hydraulic effects are important. We believe that it will be important to investigate such effects should the PIUS reactor progress to the design certification stage.

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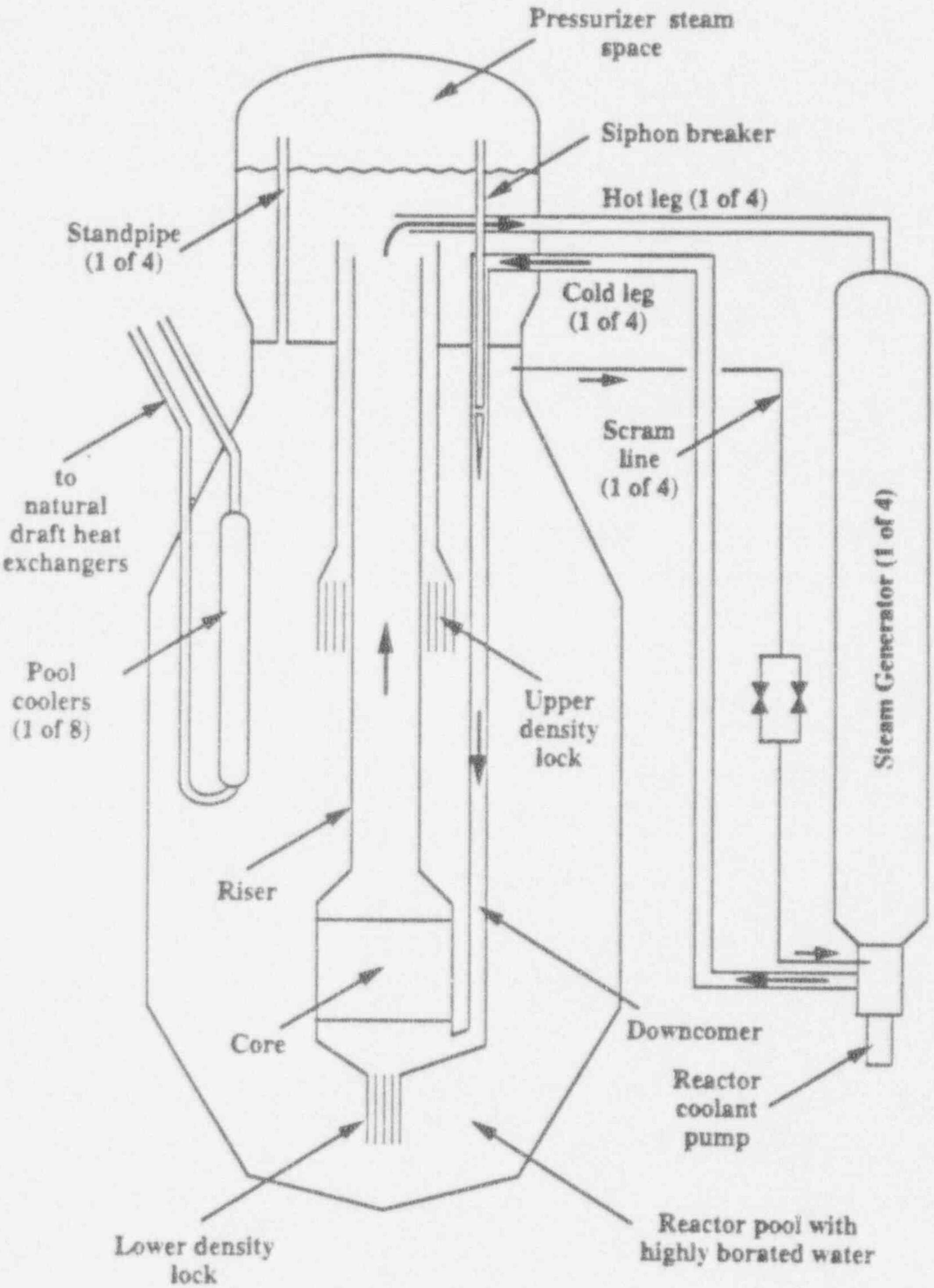

Fig. 1.
Schematic of the PIUS Reactor.

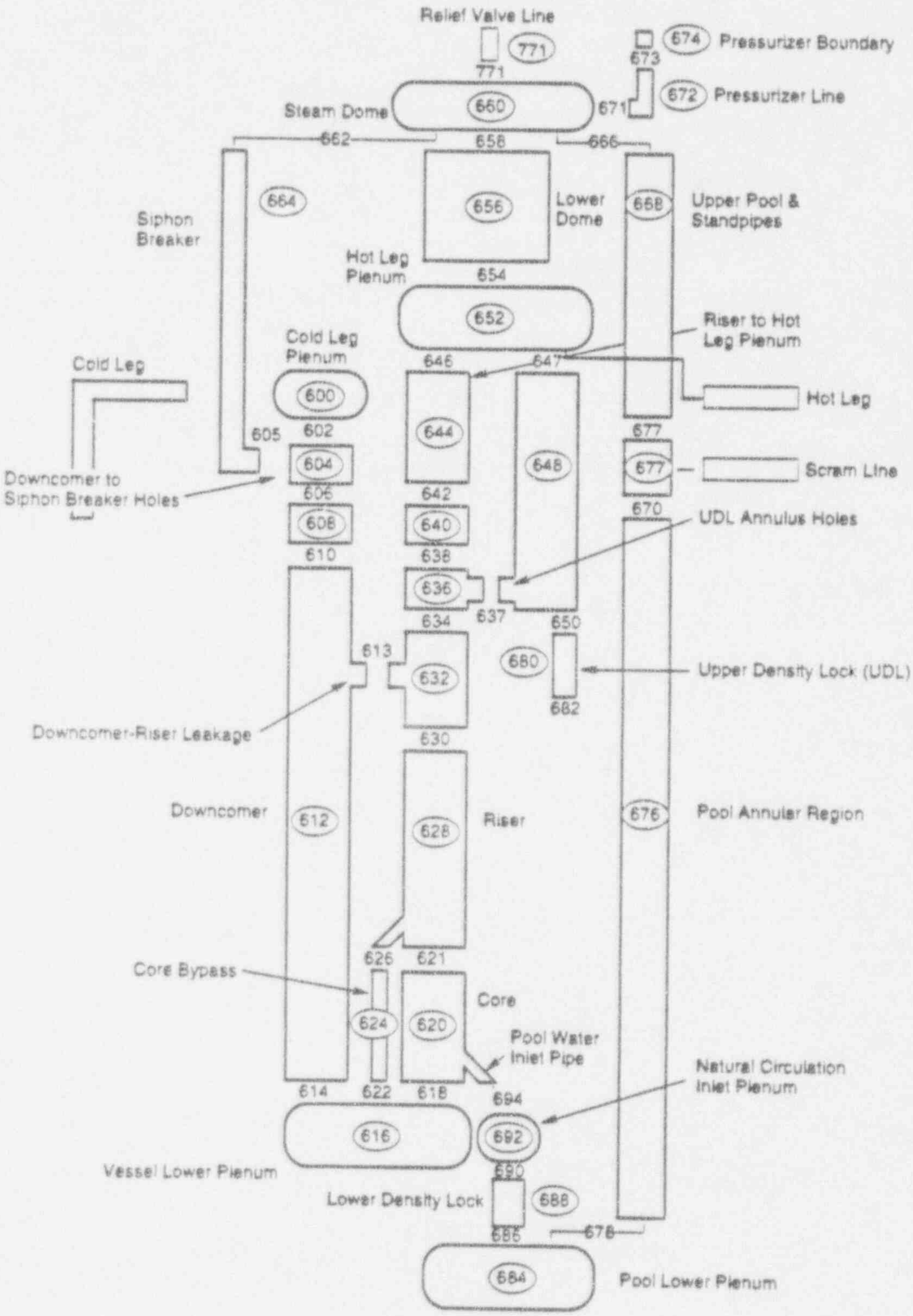

Fig. 2.
TRAC 1D model of the PIUS vessel and pool.

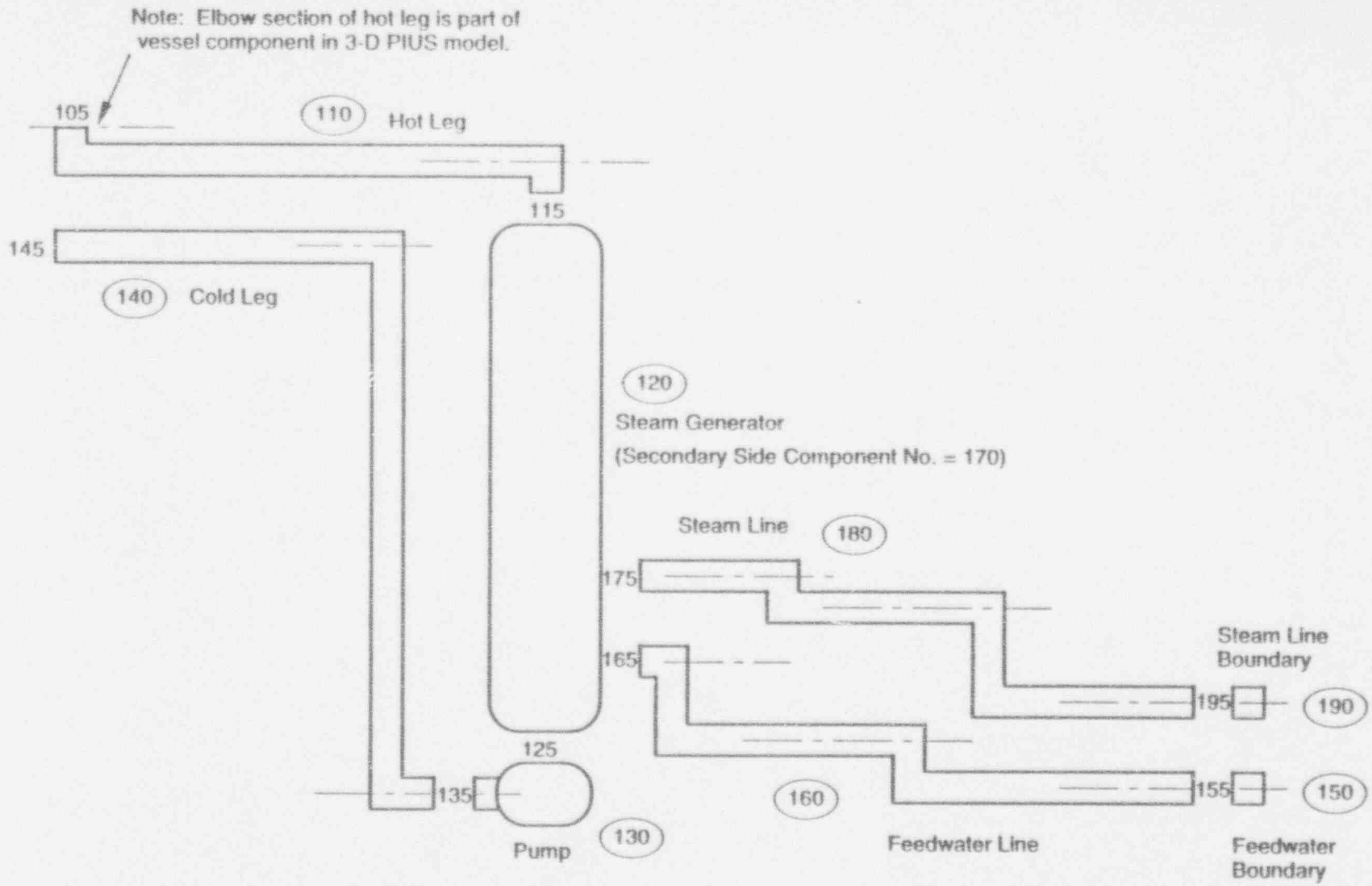

Fig. 3.
TRAC 1D model of the PIUS coolant loops.

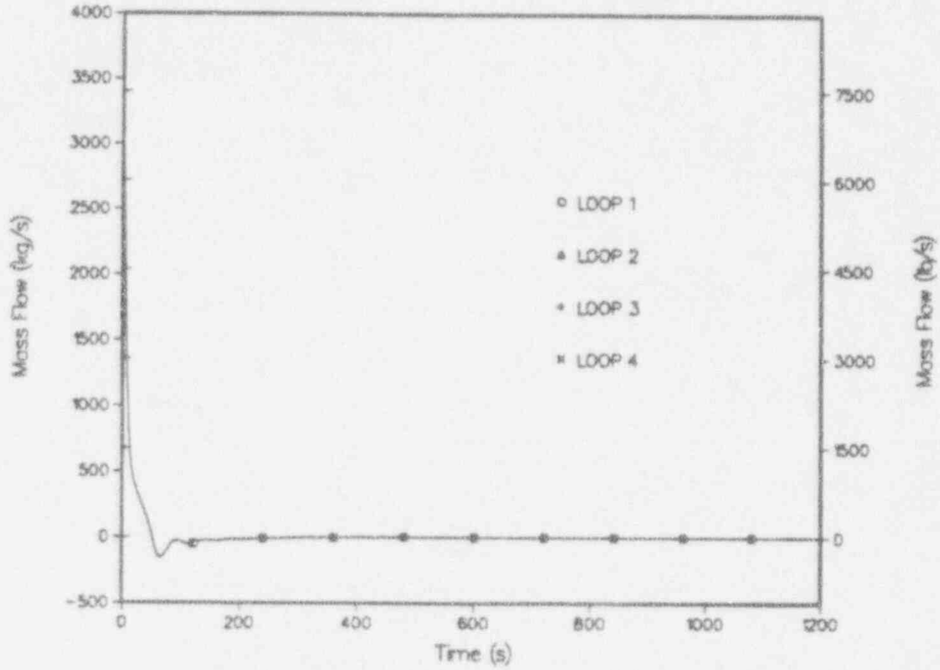

Fig. 4.

Loop flows for the LOSP baseline case.

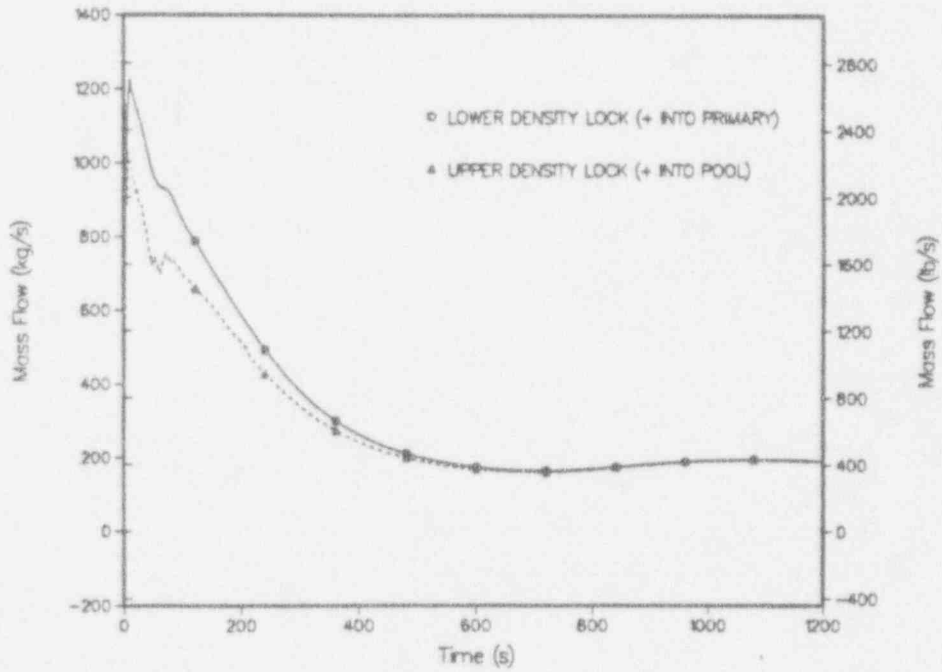

Fig. 5.

Density lock flows for the LOSP baseline case.

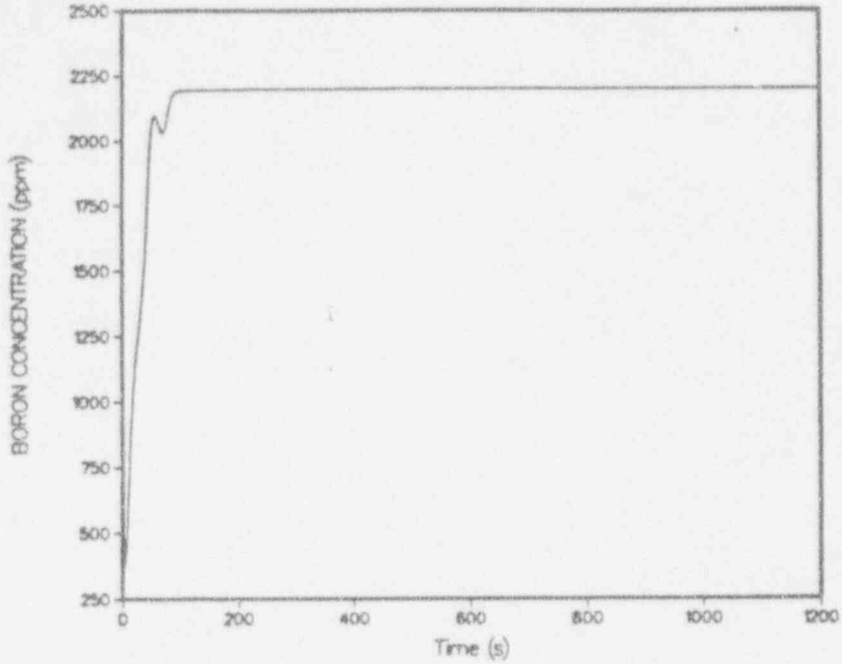

Fig. 6.

Primary boron concentration for the LOSP baseline case.

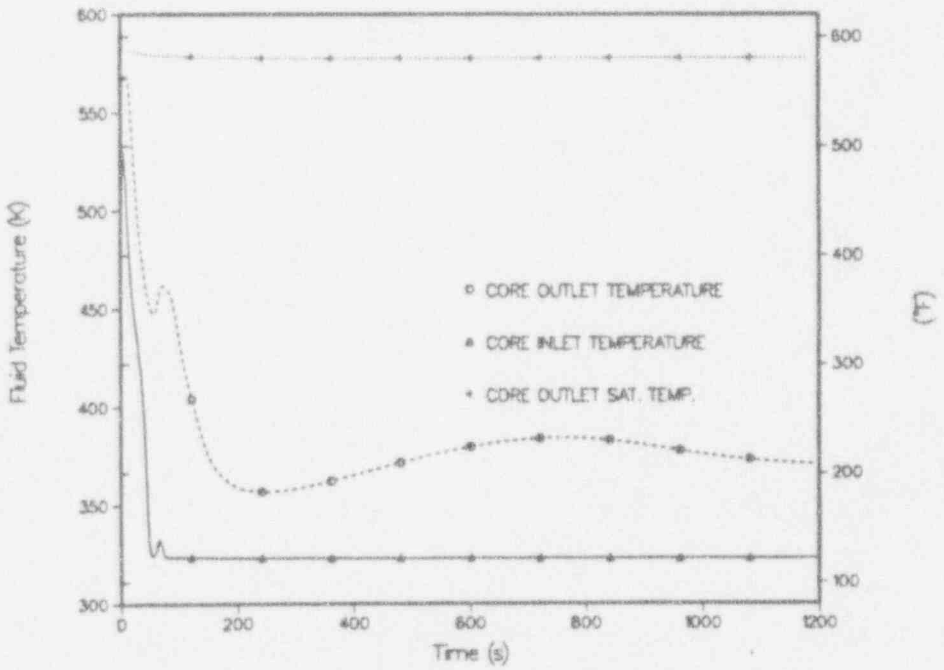

Fig. 7.

Core coolant temperatures for the LOSP baseline case.

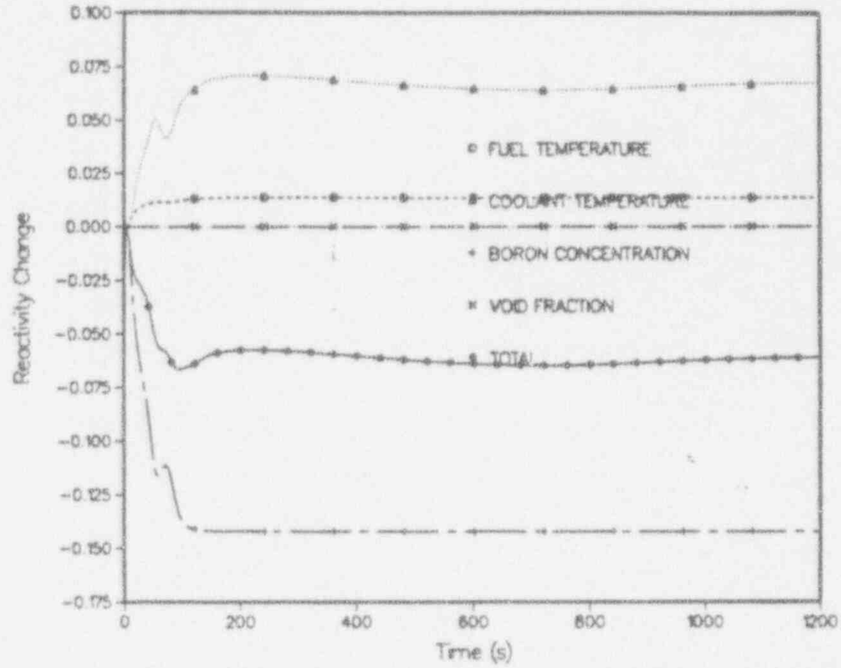

Fig. 8.

Core reactivity changes for the LOSP baseline case.

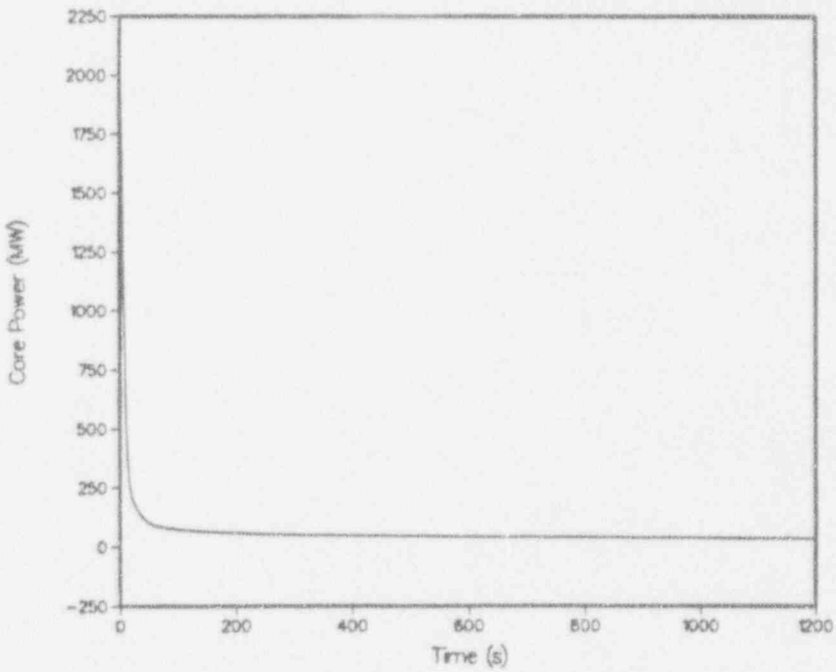

Fig. 9.

Reactor power for the LOSP baseline case.

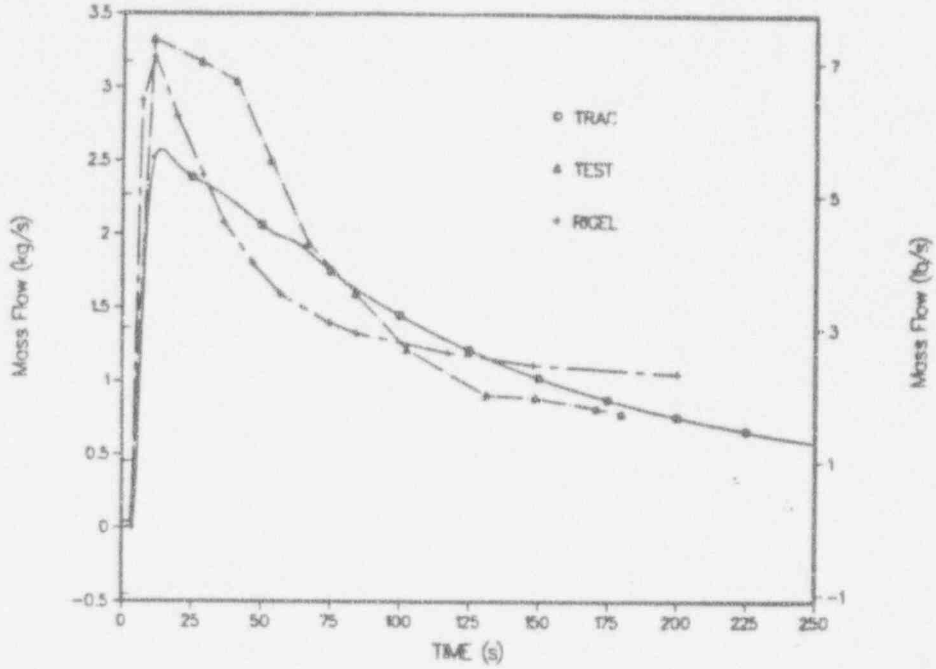

Fig. 10a.

Comparison of code-calculated and ATLE lower density lock flows

(baseline ATLE model).

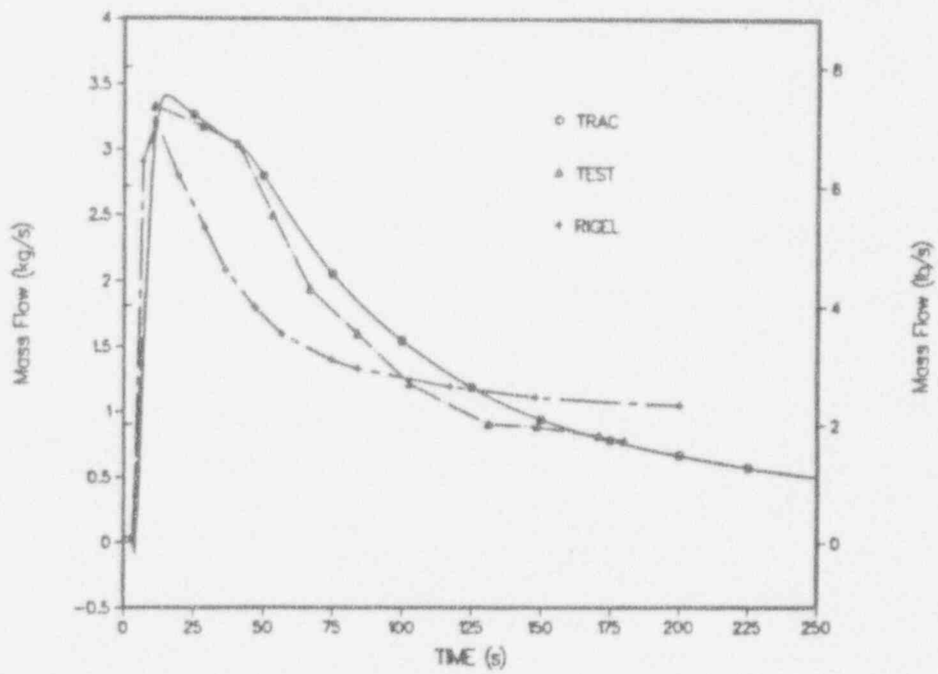

Fig. 10b.

Comparison of code-calculated and ATLE lower density lock flows

(modified ATLE model).

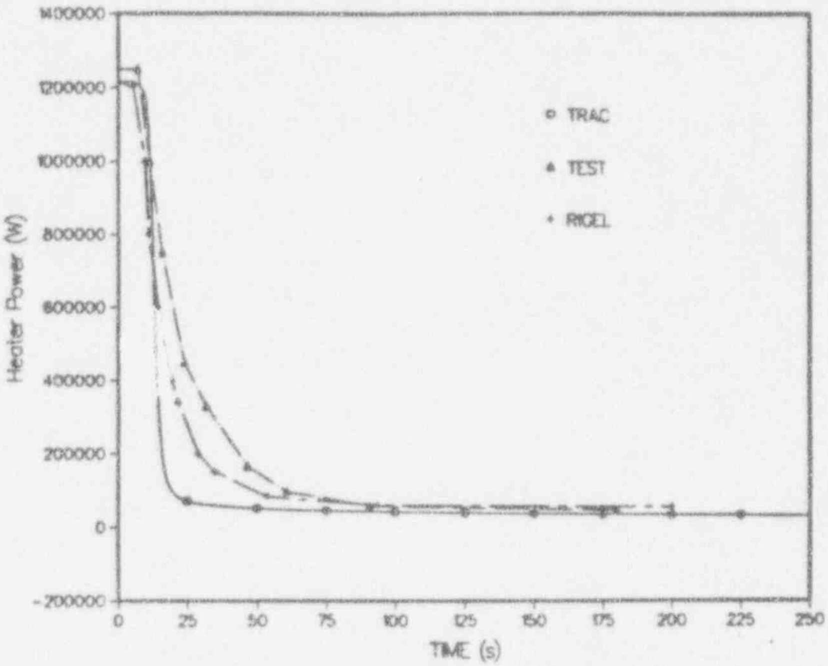

Fig. 11.

Comparison of code-calculated ATLE heater rod powers (baseline ATLE model).

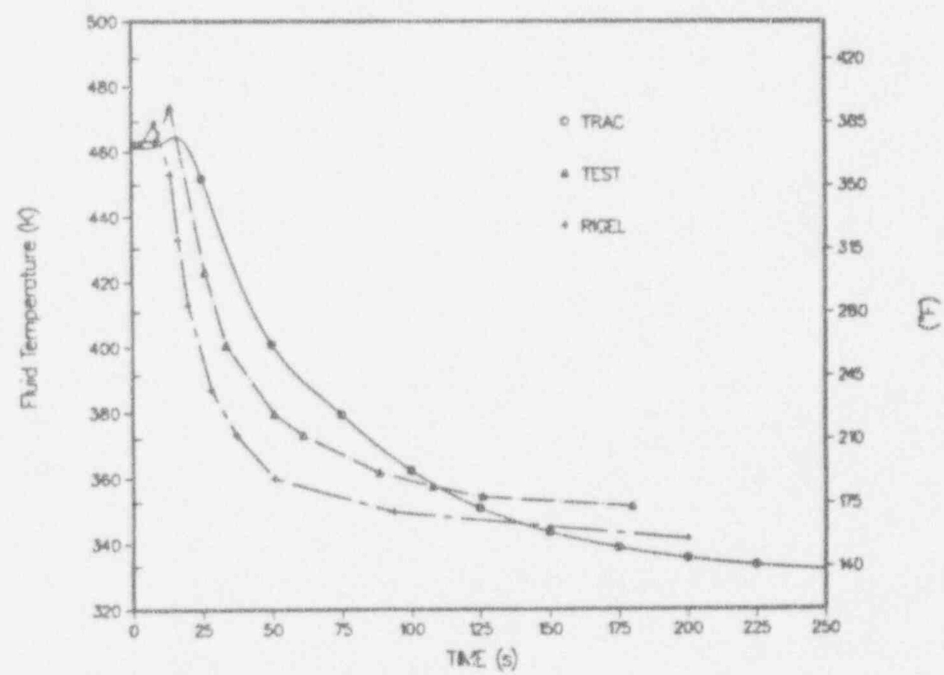

Fig. 12.

Comparison of code-calculated ATLE core outlet temperatures (baseline ATLE model).

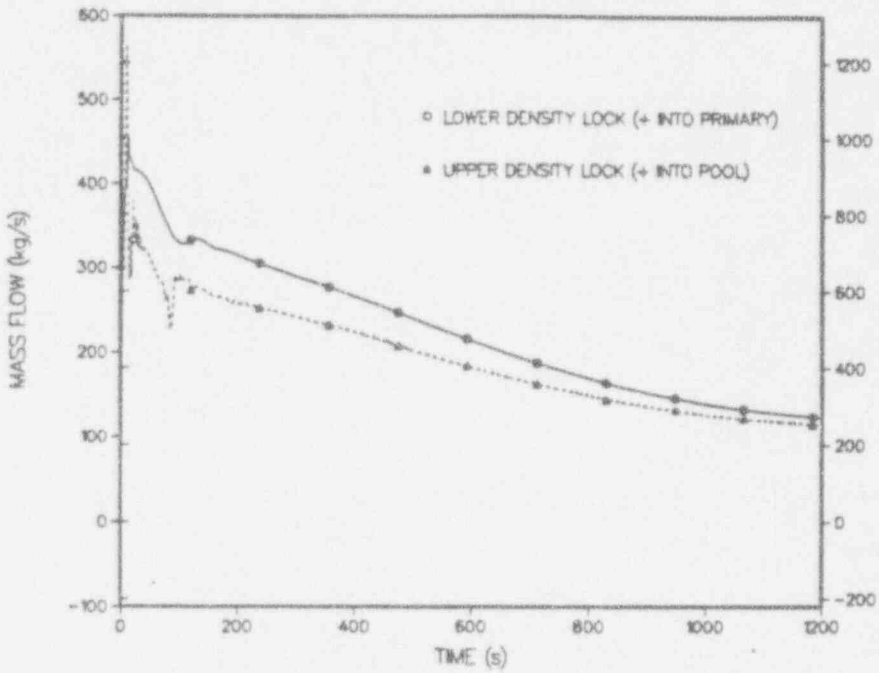

Fig. 13.

Density lock flows for a LOSP with 75% blockage of the lower density lock.

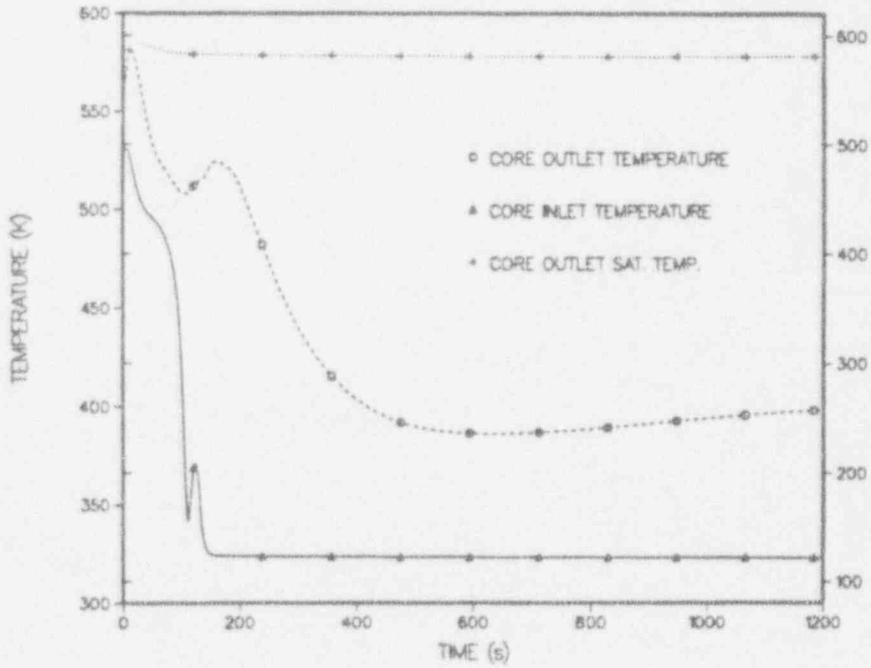

Fig. 14.

Core coolant temperatures for a LOSP with 75% blockage of the lower density lock.

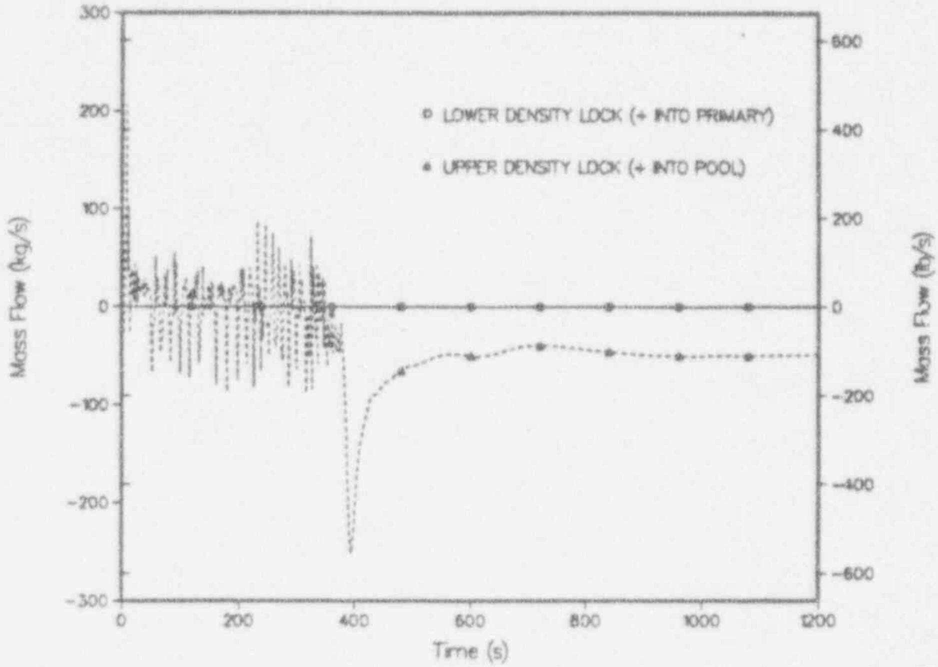

Fig. 15.

Density lock flows for a LOSP with 100% blockage of the lower density lock.

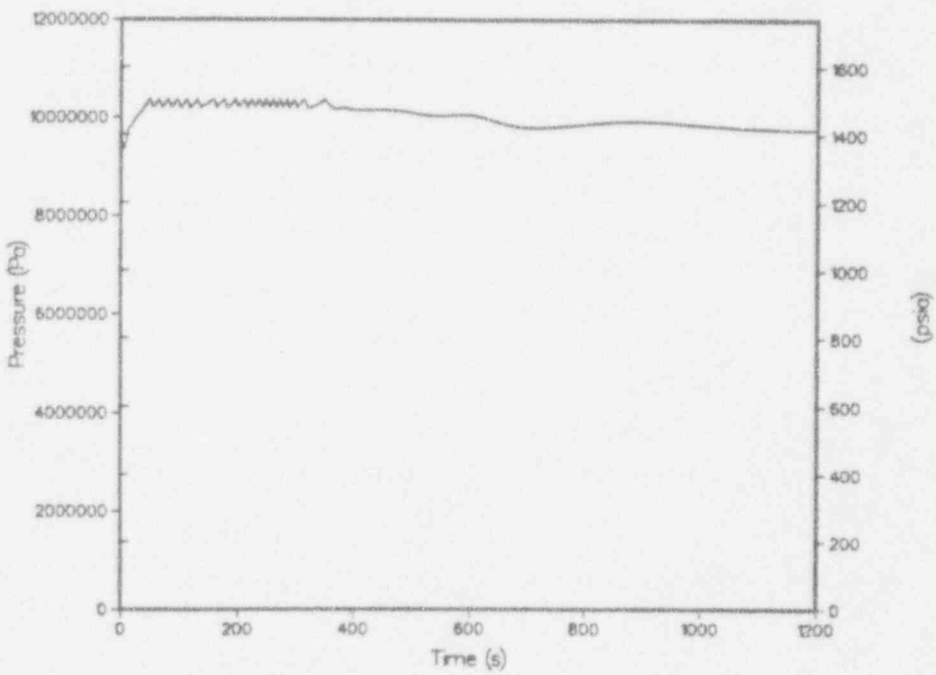

Fig. 16.

Pressurizer pressure for a LOSP with 100% blockage of the lower density lock.

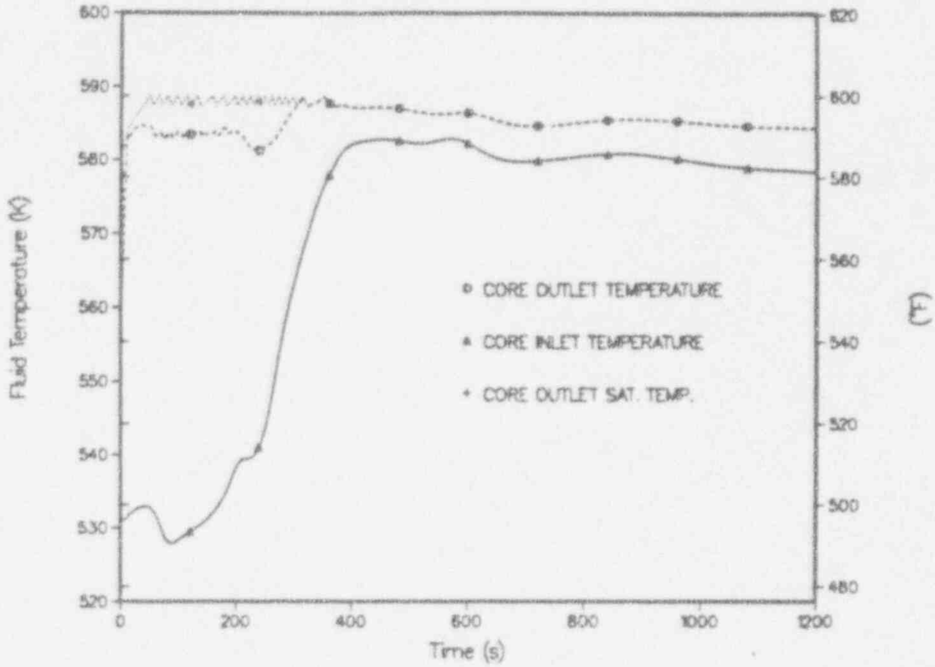

Fig. 17.

Core coolant temperatures for a LOSP with 100% blockage of the lower density lock.

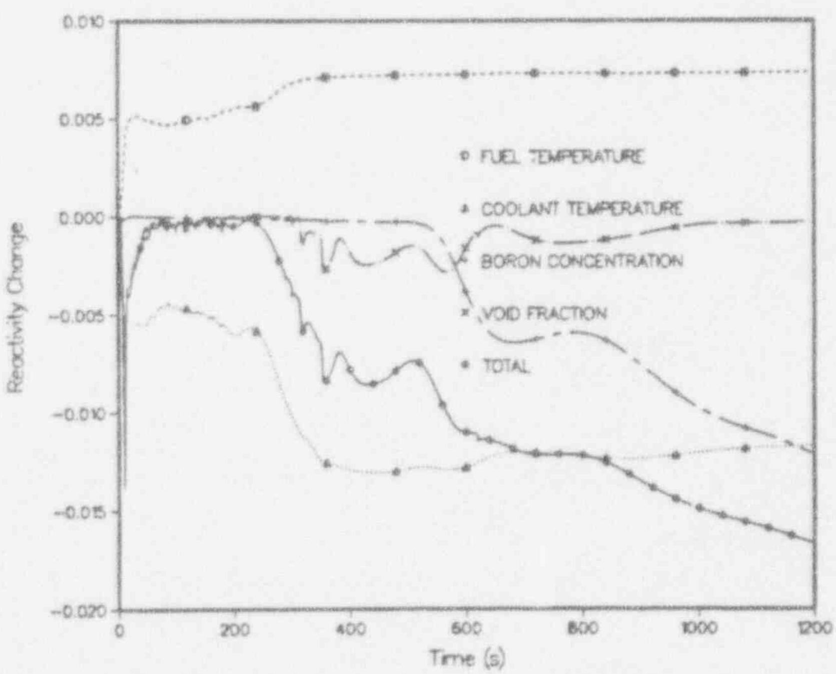

Fig. 18.

Core reactivity changes for a LOSP with 100% blockage of the lower density lock.

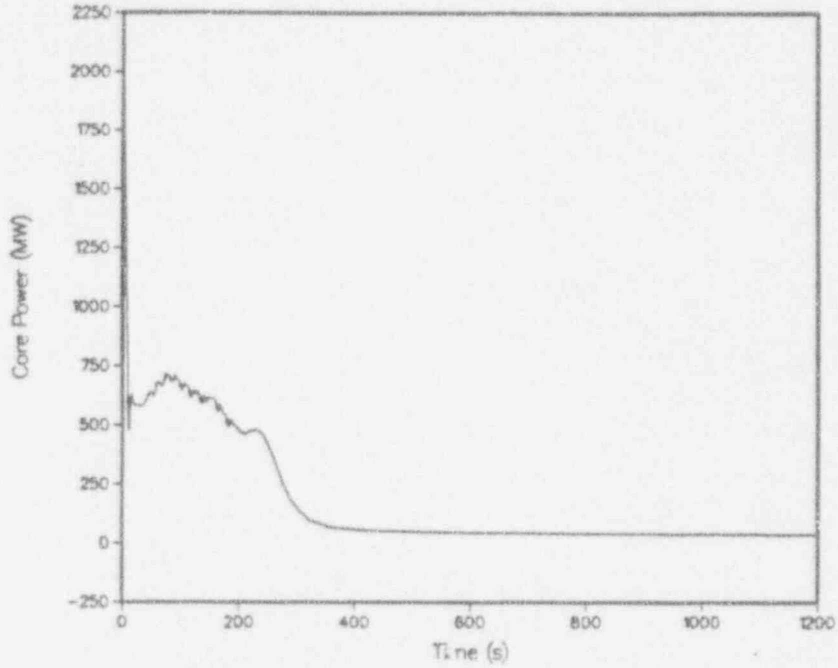

Fig. 19.

Reactor power for a LOSP with 100% blockage of the lower density lock.

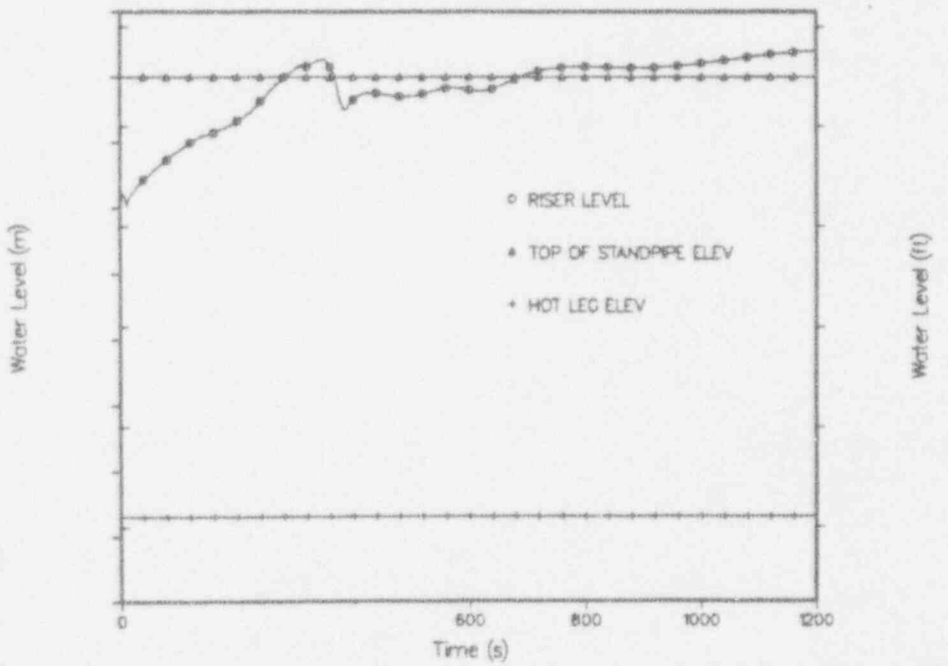

Fig. 20.

Collapsed liquid level within PIUS flow structure for a LOSP with 100% blockage of the lower density lock.

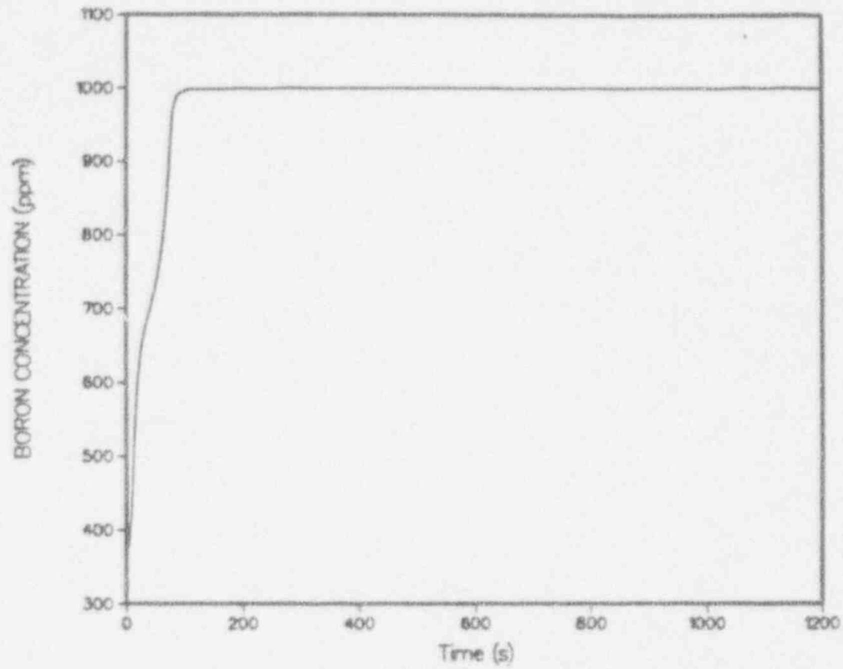

Fig. 21.

Primary boron concentration for a LOSP with a pool boron concentration of 1000 ppm.

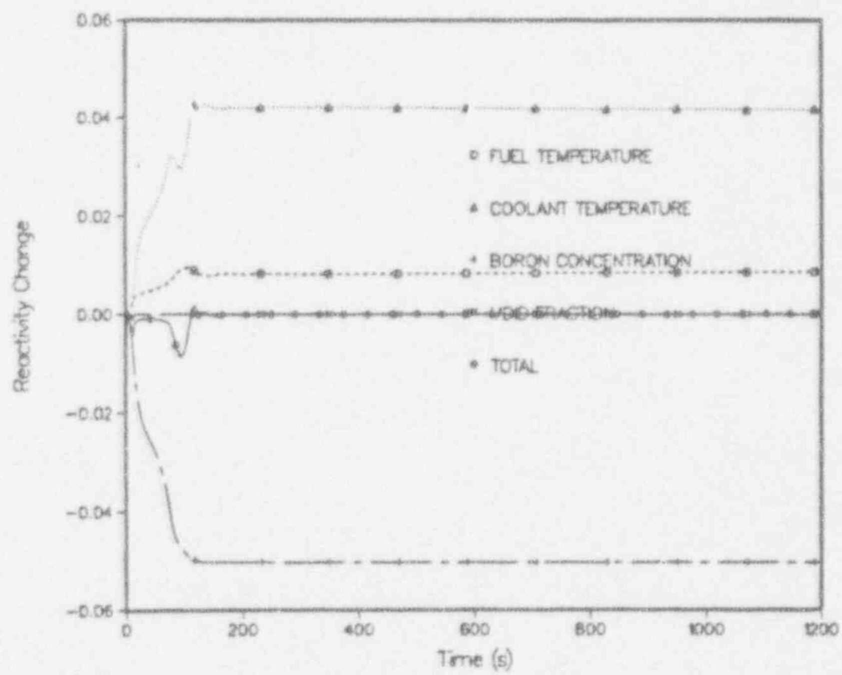

Fig. 22.

Core reactivity changes for a LOSP with a pool boron concentration of 1000 ppm.

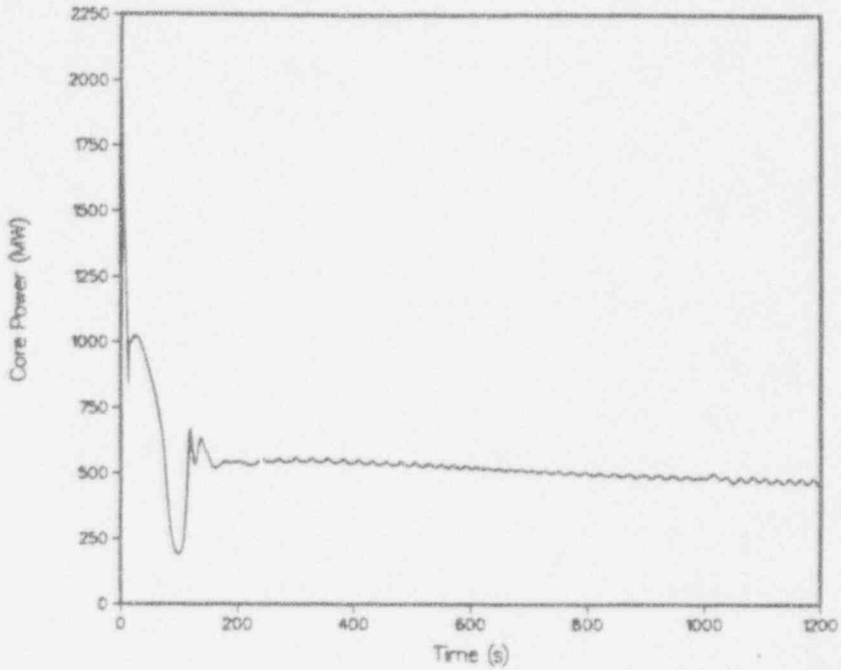

Fig. 23.

Reactor power for a LOSP with a pool boron concentration of 1000 ppm.

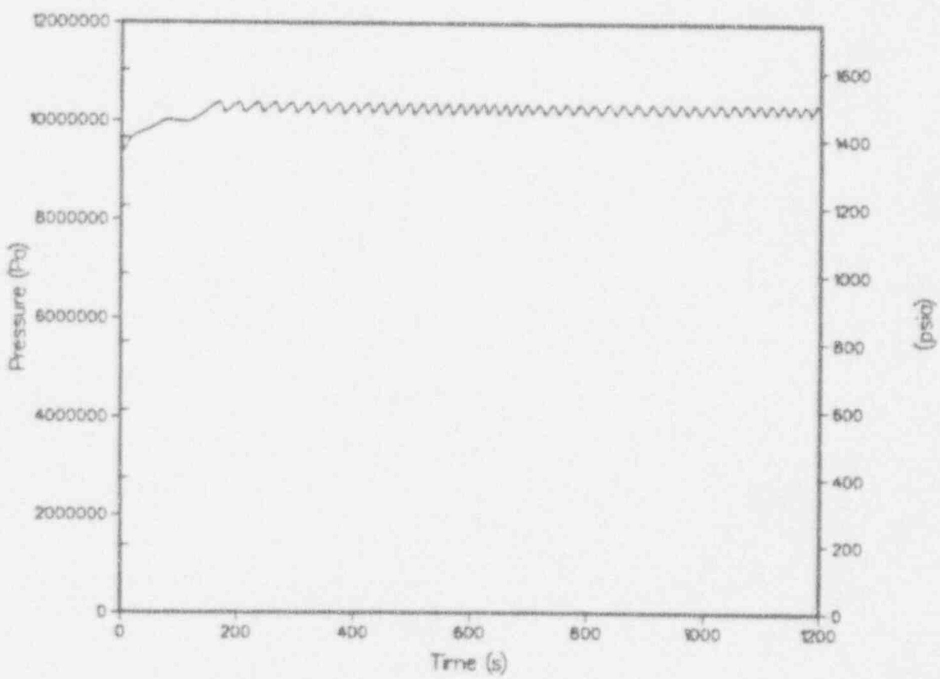

Fig. 24.

Pressurizer pressure for a LOSP with a pool boron concentration of 1000 ppm.