ENCLOSURE 4

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Title: REACTOR SCRAM EVENTS IN THE UPDATED PIUS 600 ADVANCED REACTOR DESIGN

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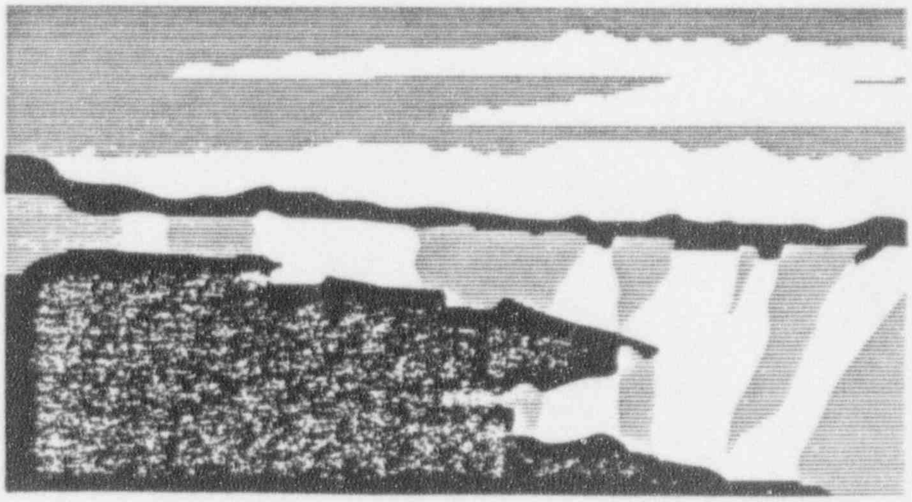

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REACTOR SCRAM EVENTS IN THE UPDATED PIUS 600 ADVANCED REACTOR DESIGN*

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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 a 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. TRAC baseline calculations of the recently modified PIUS design were performed for both active and passive system reactor trips. Sensitivity studies were performed to explore the robustness of the PIUS concept to severe off-normal conditions following active-system trips. The TRAC-calculated baseline active and passive scram results were compared to the 1D analyses prepared by ABB for the recently modified PIUS design. 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 RCPs lose their motive power, thereby eliminating the balance between the primary coolant loop

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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. Because the PIUS reactor does not have the usual rod-based shutdown systems of existing and planned light water reactors, the behavior of the PIUS and shutdown phenomena following active and passive system scrams must be understood. 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 both an active-system and example passive-system reactor trip. Sensitivity studies were performed to explore the robustness of the PIUS concept to severe off-normal conditions following active-system trips. 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 noncondensable gas), two-fluid (liquid and gas), and nonequilibrium modeling of thermal-hydraulic behavior. TRAC features flow-regime-dependent constitutive equations, component modularity, multi-dimensional 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 proved 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, (3) conduct a bottom-up review of the individual TRAC models and correlations, and (4) conduct 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 conducted 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 many of the basic TRAC models and correlations are adequate, although some needed code modifications were also identified. 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.

Benchmarking against another validated code is a second approach to demonstrating adequacy. In this paper, we will provide comparisons of TRAC- and RIGEL-calculated results for the two baseline trip transients. These comparisons show reasonable qualitative and quantitative agreement in most, but not all, respects.

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.[9] 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 do note a reservation. The most important physical processes in PIUS are related to reactor shutdown because the PIUS reactor does not contain mechanical 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 used, 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).

ACTIVE SCRAM TRANSIENT

Essentially all important phenomena in this transient result from operation of the active scram system and termination of feedwater flow to the steam generators. Following generation of the scram signal at time zero, the active scram system is activated. Thus, there is no further injection of highly borated pool water into the primary through the scram lines after 212 s. Feedwater flow to all four loops is decreased linearly to zero over a 20-s interval following receipt of the scram signal at time zero. The steam generators no longer serve as heat sinks after approximately 120 s and core-generated power can no longer be rejected to them. The reactor coolant pumps continue to operate throughout the transient.

A shutdown in reactor power was achieved, as shown in Fig. 4 (Frame 3). The total flow of highly borated pool water (2200 ppm) passing through the scram lines rapidly peaks at 750 kg/s and then declines to about 635 kg/s at 182 s when the scram valves begin to close. The water passing through the scram lines displaces water in the primary through the upper and lower density locks as shown in Fig. 5 (Frame 7). Most of the displaced primary inventory flows to the reactor pool through the upper density lock. A much smaller amount flows into the reactor pool through the lower density lock. The flows through the density locks cease when the scram valves are closed. The primary loop boron concentration increases rapidly while the scram valves are open as, shown in Fig. 6 (Frame 11). After the valves shut, the flow of highly borated pool water is terminated and the primary boron concentration stabilizes at about 840 ppm. The period of the rapidly decaying oscillations in the boron concentration after 212 s is characteristic of the primary circuit transport time. Figure 7 (Frame 6) shows the reactivity changes resulting from fuel temperature, coolant temperature, voiding, boron concentration, and the net total of these components. Positive reactivity insertions arise from the fuel and coolant temperatures, which are decreasing during the period the scram valves are open, as shown in Fig. 8 (Frame 12). The negative reactivity inserted by the boron is larger than the positive fuel and moderator temperature contributions, causing a total negative reactivity insertion and reduction in core power to decay heat levels, as shown in Fig. 4 (Frame 3). Following closure of the scram valves at 212 s, neither pool water nor boron are entering the primary system. Forced flows through the upper and lower density locks are also terminated. Control of the thermal interface in the lower density lock is recovered and no subsequent flows through the density lock occur. The steam generators do not function as heat sinks after 120 s. Thus, the core decay heat is deposited in the primary coolant and fuel, and coolant temperatures begin a near-linear increase, as shown in Fig. 8 (Frame 12). ABB has not indicated how it intends to terminate this event. Should no action be taken, the primary would continue to heat, the primary coolant pumps would increase speed until their overspeed limit of 115% was reached, and the density locks would activate to initiate natural circulation between the primary system and the reactor pool. The pool would be cooled by either active (nonsafety grade) or passive (fully safety grade) pool cooling systems

A RIGEL calculation of the active system scram was prepared and reported by ABB in Ref. 3. Several results from the RIGEL calculations have been co-plotted with the TRAC-calculated results for this transient. The RIGEL calculations were terminated at 300 s while the TRAC calculations were terminated at 1200 s. For RIGEL, the flow through a single scram valve immediately peaks at about 175 kg/s and then declines to about 150 kg/s at the time the scram valves begin to close. For TRAC, the flow immediately peaks at 182 kg/s and declines to about 160 kg/s at the time the scram valves begin to close. The TRAC- and RIGEL-calculated core powers are shown in Fig. 4. The upper and lower density lock flows are compared in Fig. 5 and the primary loop boron concentrations are compared in Fig. 6. Primary coolant temperatures are compared in Fig. 8. The TRAC- and RIGEL-calculated results are both qualitatively and quantitatively similar, and are therefore in reasonable agreement. Because the two code methods were independently developed, this reasonable agreement provides an added element of confidence that the major trends and processes associated with the active are correctly represented, within the inherent capabilities of the 1D thermal-hydraulics and the point kinetics models.

Sensitivity studies were performed to explore the robustness of the PIUS concept 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. As might be expected, given the minimal flows through the lower density lock shown in Fig. 5 (Frame 7), even a total blockage of the lower density lock produces only a minor impact on the course of the transient. As a further assessment of the robustness of the PIUS concept, total blockages of both the upper and lower density locks were assumed. A shutdown in reactor power again was achieved. However, with both density locks blocked, the amount of pool water injected through the scram lines is reduced compared with the baseline because primary inventory can only be displaced into the reactor pool through the small standpipes that connect the pressurizer steam space and the reactor pool. The total scram-line flow increases to 800 kg/s immediately after the scram valves open, but the flow decreases to 300 kg/s by 45 s and remains there until 135 s. A two-phase mixture is entering the standpoint during this interval. At 135 s, the collapsed liquid level reaches the top of the standpoint and the total scram-line flow increases to 500 kg/s as liquid water flows through the standpoint into the reactor pool. With the reduced scram-line flow, the primary boron concentration increased to only 480 ppm before the scram valves closed. Given the limited boron injected into the core, the core power decreases more slowly and the fuel and primary coolant temperatures remain higher, as shown in Fig. 9 (Frame 12). After 1000 s, the increasing moderator temperature results in the largest negative reactivity contribution to the total reactivity, as shown in Fig. 10 (Frame 6).

Calculations were also performed to examine the effect of reduced pool boron concentrations below the 2200 ppm specified by ABB as the normal operating condition. 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. Active scrams with pool boron concentrations of 1800 and 1000 ppm were analyzed. For a pool boron concentration of 1800 ppm, reactor power decreases at a slightly slower rate than for the baseline case but the power levels are indistinguishable by 200 s. The primary loop boron concentration stabilizes at about 765 ppm following closure of the scram valves at 212 s. The primary system temperature trends are the same as in the baseline calculation, but the initial cooldown is about 5 K less because the core power decrease is slowed initially. The active system scram with the pool boron concentration at 1000 ppm also leads to a shutdown to decay heat levels, although the phenomena are markedly different. The reactor power decreases at a slower rate than in the baseline but reaches the same level as the baseline by 400 s. Consequently, the amount of decay heat deposited in the primary causes the primary system to heat and pressurize. The pressure relief system safety valves open twice during the period the scram valves are open when the 10.3-MPa setpoint is reached. After the scram valves are closed, the safety valves continue to actuate for about 110 s. Following closure of the scram valves, the primary loop boron concentration stabilizes at about 550 ppm compared with 840 ppm in the baseline. Because the core power decreases more slowly, the fuel and primary coolant temperatures remain higher, as shown in Fig. 11 (Frame 12). After 750 s, the increasing moderator temperature begins to contribute negative reactivity to the total reactivity, as shown in Fig. 12 (Frame 6).

PASSIVE SCRAM TRANSIENT

The event selected to demonstrate the passive scram function is the loss of a single RCP. The primary reason for selecting this event is that a RIGEL calculation for this event is presented in the Supplemental Information Package.[3] In addition, the trip of a single RCP was the programmed scram operation in the original PSID design. Essentially all important phenomena in this transient result from tripping of a single RCP (loop 3 for the TRAC calculation) and the behavior of the loop 3 steam generator. The scram-valve system does not operate during this event, following the assumption given by ABB. The RCPs in the remaining three loops continue to operate throughout the transient.

There are several immediate outcomes of the trip of the loop 3 RCP. The tripped RCP coasts down while the remaining three RCPs increase speed, rapidly reaching their overspeed limit of 115% in an attempt to maintain control of the lower density lock interface. Flows in the loops with operating RCPs increase while the flow in the loop with the tripped RCP reverses, as shown in Fig. 13 (Frame 14). The steam generators in the loops with operating RCPs no longer function as heat sinks after 85 s. The steam generator in the loop with the tripped RCP continues to function as a heat sink until 300 s because the primary flow rate and temperature are lower than in the loops with the operating RCPs. The imbalance caused by loss of one RCP is, by design, too large for the pump speed control and the lower and upper density locks activate (Fig. 14, Frame 7). The flow through the lower density lock peaks at 550 kg/s on the first insurge after activation. Following two early oscillations, the lower density lock flow smoothly decreases in concert with the core power, approaching 40 kg/s as the power reaches decay heat levels. The upper density lock flow follows the same pattern. Boron enters the primary through the lower density lock (Fig. 15, Frame 11) and the core power declines accordingly (Fig. 16, Frame 3), reaching 5% by 200 s. The primary system boron concentration increases rapidly while the lower density lock flow is high and then continues to increase at a lower rate as a stable natural circulation flow rate is established. Fluid temperatures in the primary are shown in Fig. 17 (Frame 12).

A RIGEL calculation of the passive system scram was prepared and reported by ABB in Ref. 10. Several results from the RIGEL calculations have been co-plotted with the TRAC-calculated results for this transient. The RIGEL calculations were terminated at 600 s while the

TRAC calculations were terminated at 1200 s. The RIGEL-calculated results differ from the TRAC results in one particularly important way. RIGEL predicts a higher peak lower density lock flow (Fig. 14). The TRAC-calculated peak value is about 40% lower. The same trend was noted in a TRAC assessment activity using data from the ATLE facility. The assessment was conducted for a two-pump trip. The calculated lower density lock flow rates calculated by RIGEL and TRAC are compared to the measured lower density lock flow in Fig. 18. The TRAC-calculated initial insurge flow is about 25% less than measured. Although we have conducted extensive sensitivity studies, we have been unable to identify the cause for the underprediction. There are several consequences. Less boron enters the primary system early in the transient (Fig. 15). Thus, the decrease in core power is slower than predicted by RIGEL (Fig. 16) and the decrease in primary system temperatures is smaller (Fig. 17). In the prediction of all important phenomena and trends, the calculations from the two codes are similar. A natural circulation flow is established with pool water entering the primary through the lower density lock and reentering the pool through the upper density lock.

SUMMARY OF OBSERVATIONS

1. Two systems exist for reactor shutdown—active and passive. Each system is effective in scramming the reactor. 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 with 1D thermal-hydraulic and point kinetics models.

2. The PIUS core, as presently designed, is characterized by compensating shutdown mechanisms. When highly borated pool water enters the primary through either the scram lines or 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 temperature contribution to reactivity is positive in such circumstances. However, if the flow of boron into the core is reduced, the primary coolant temperature will rise and the initially positive moderator temperature reactivity will decrease and eventually insert sufficient negative reactivity to reduce the core power to decay heat levels.

3. 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.

4. Our confidence in the baseline simulations is enhanced by the benchmark activity that shows the RIGEL- and TRAC-calculated results for the active scram and pump trip transients to be in reasonable agreement. 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.

5. 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.

6. 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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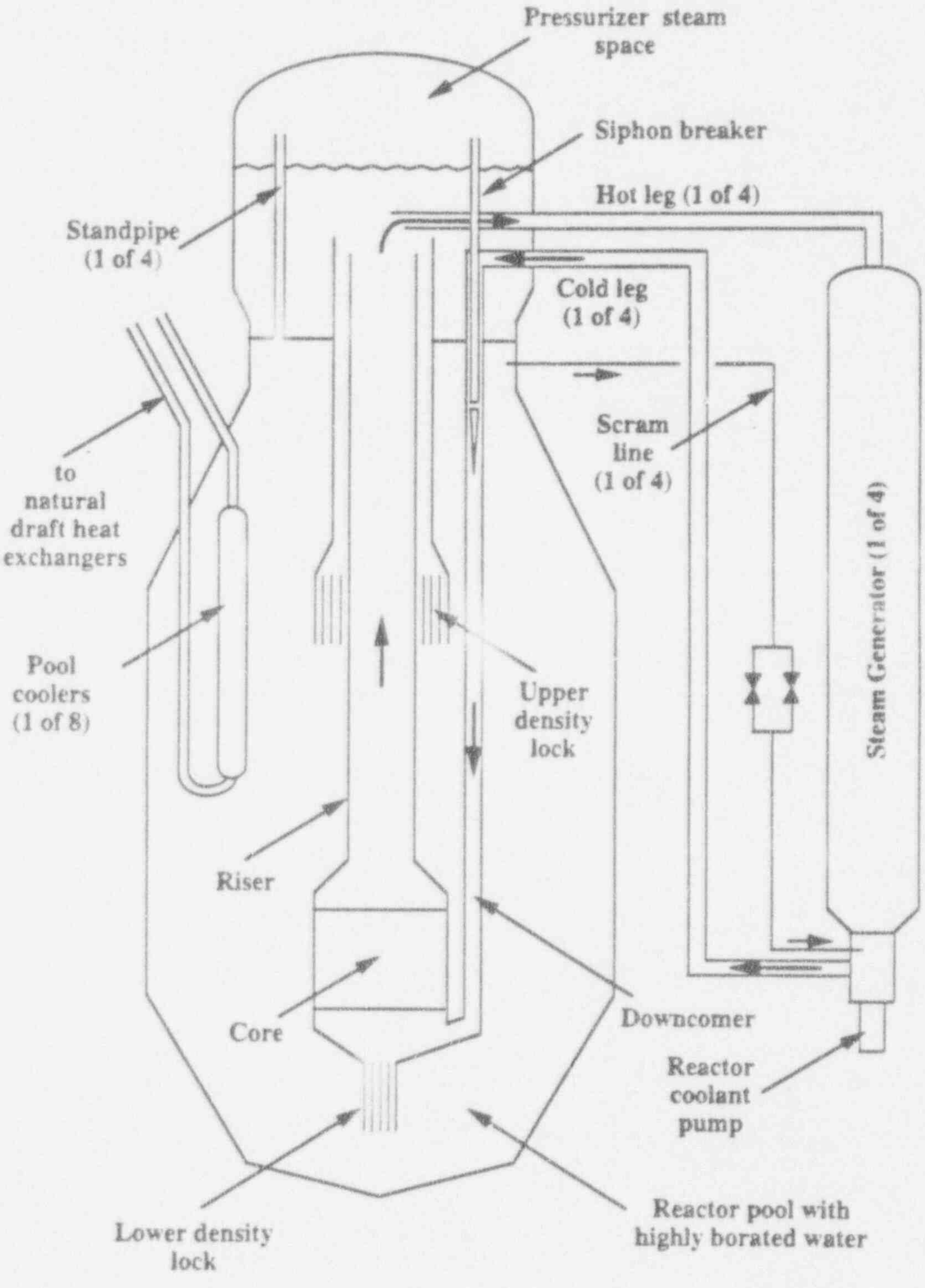

Fig. 1.
Schematic of the PIUS Reactor.

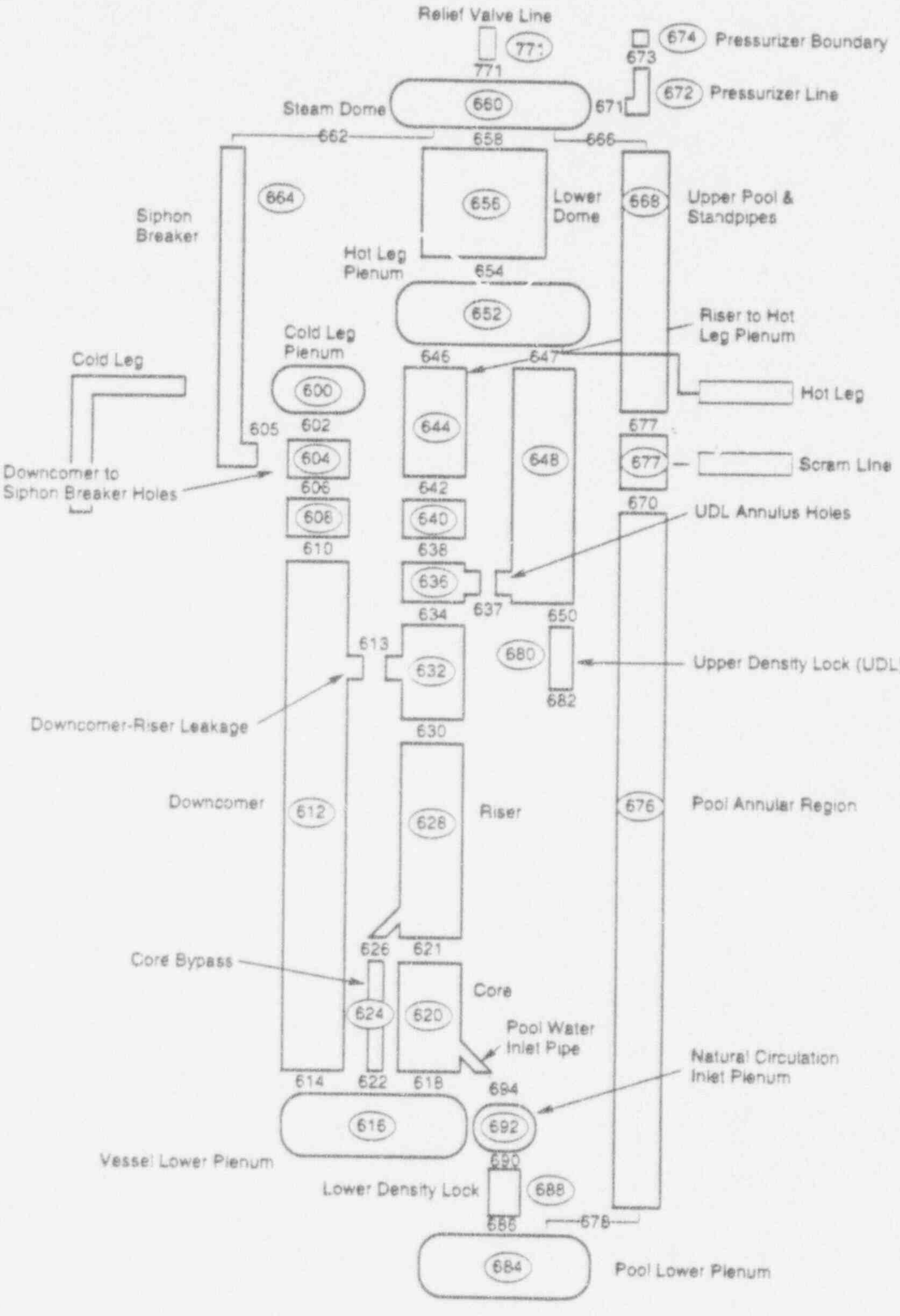

Fig. 2.
TRAC 1D model of the PIUS vessel and pool.

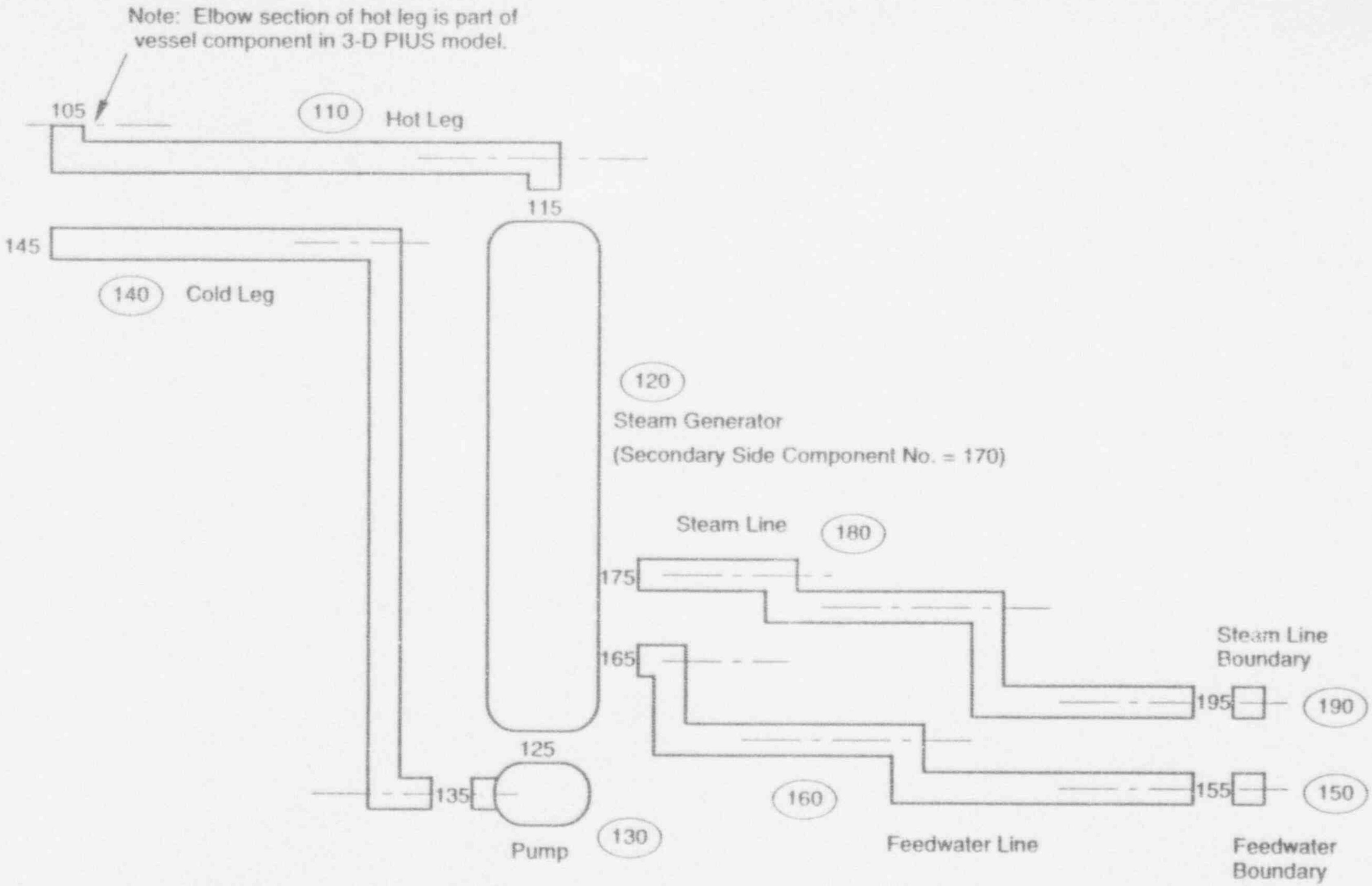

Fig. 3.
TRAC 1D model of the PIUS coolant loops.

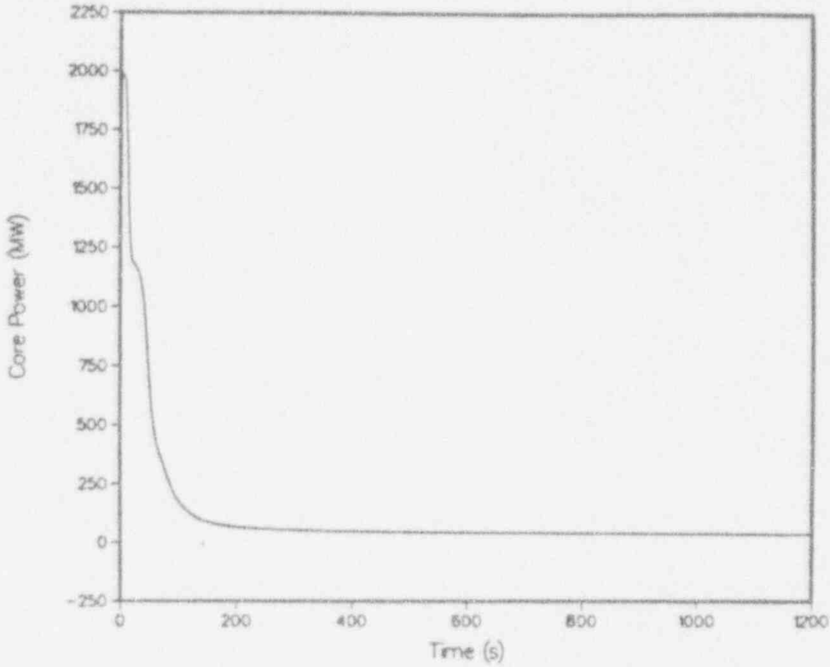

Fig. 4.
Reactor power for active scram system baseline case.

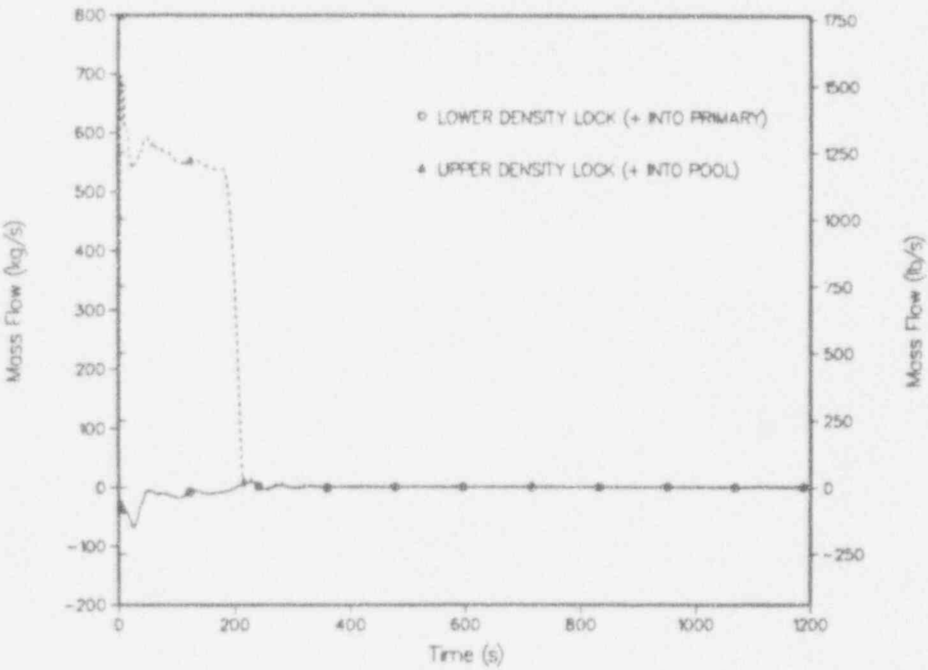

Fig. 5.
Density lock flows for the active scram system baseline case.

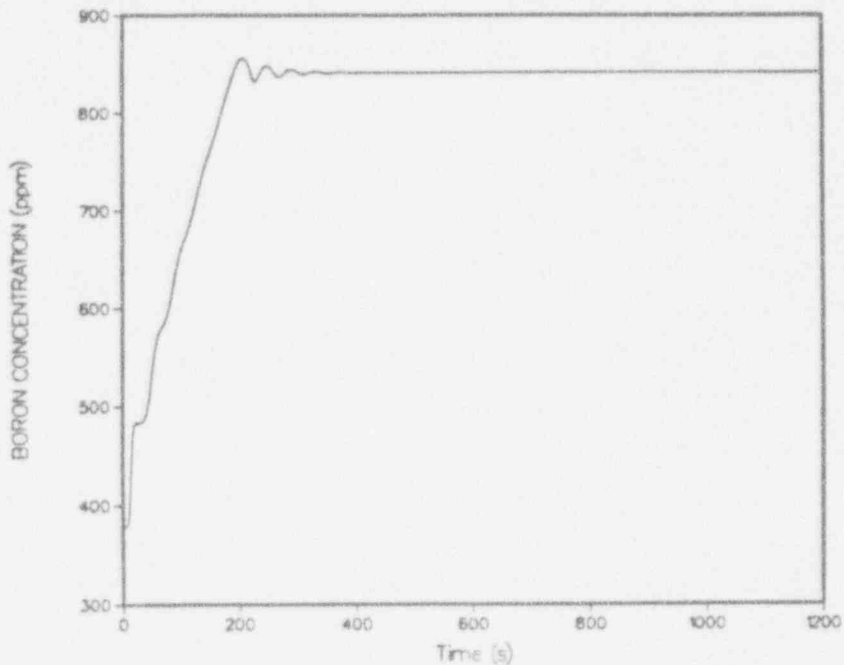

Fig. 6.
Primary boron concentration for the active scram system baseline case.

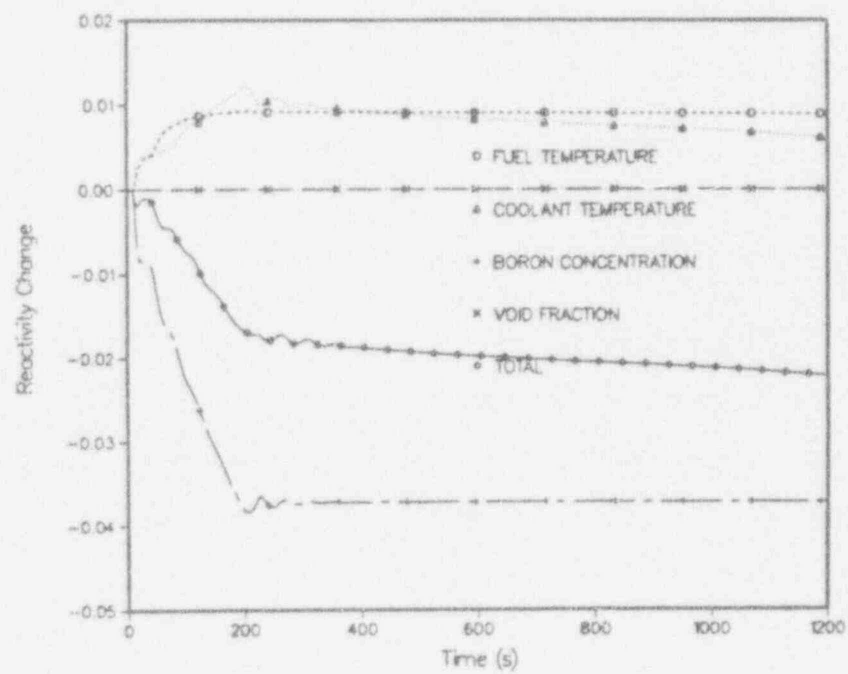

Fig. 7.
Core reactivity changes for the active scram system baseline case.

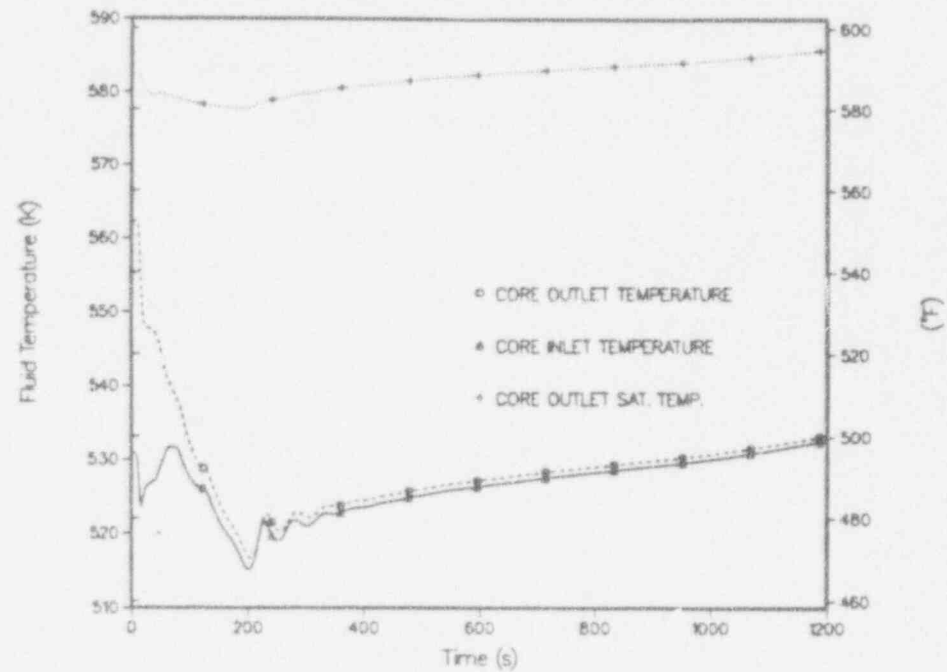

Fig. 8.
Core coolant temperatures for the active scram system baseline case.

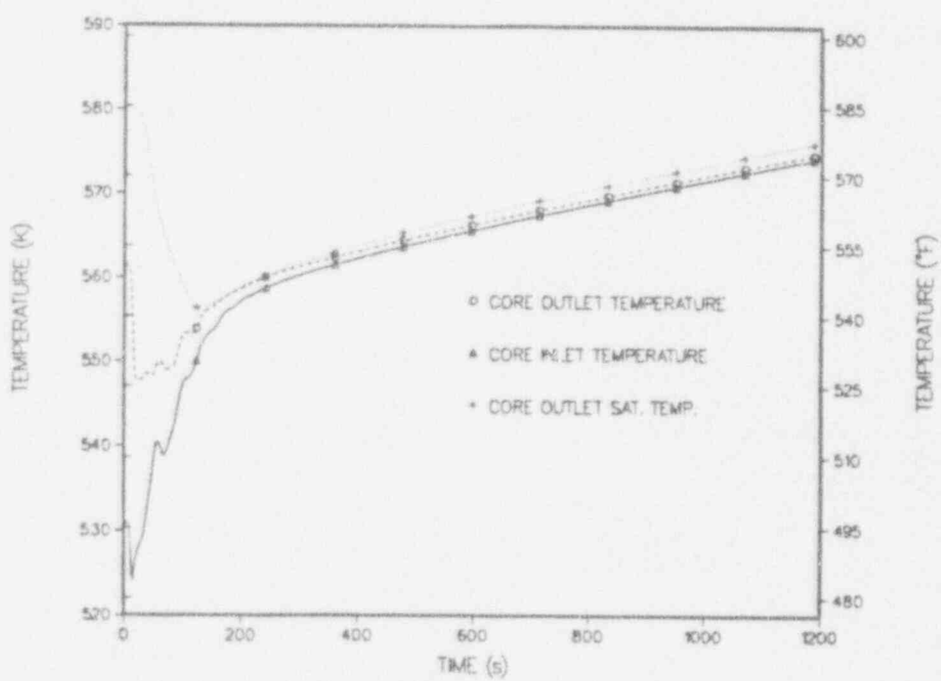

Fig. 9.
Core coolant temperatures for an active scram with complete blockage of lower and upper density locks.

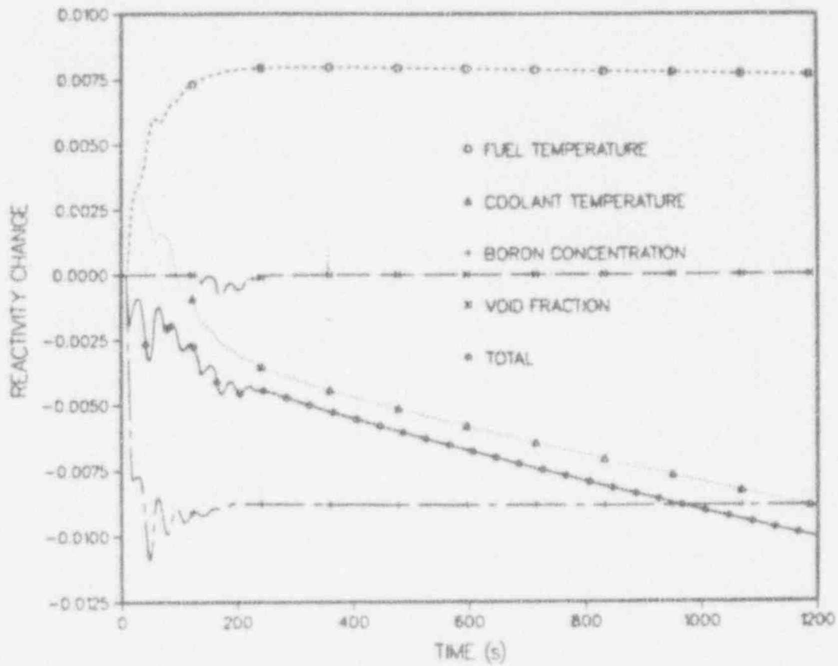

Fig. 10.
Core reactivity changes for an active scram with complete blockage of lower and upper density locks.

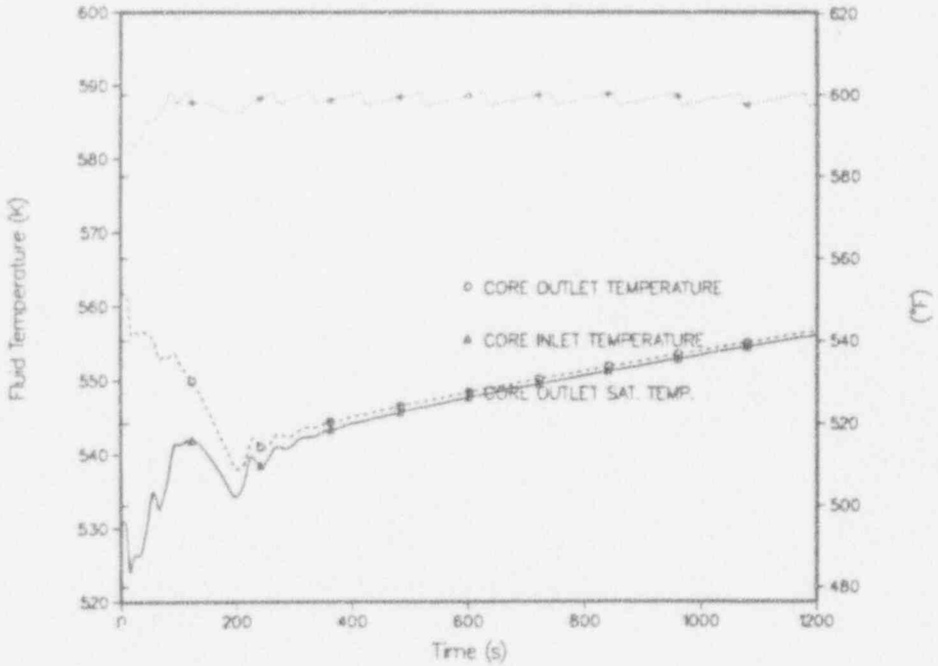

Fig. 11.
Core coolant temperatures for an active scram with a pool boron concentration of 1000 ppm.

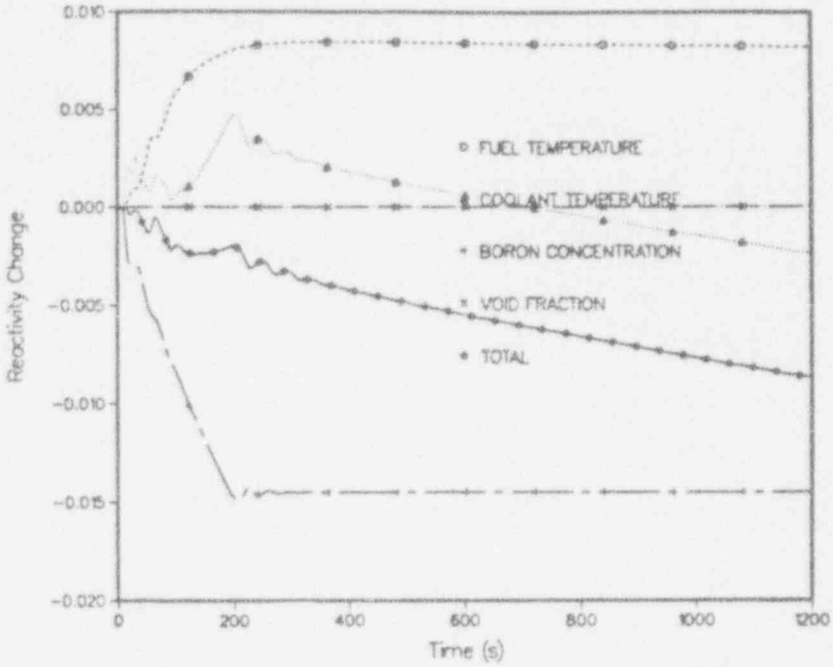

Fig. 12.
Core reactivity changes for an active scram with a pool boron concentration of 1000 ppm.

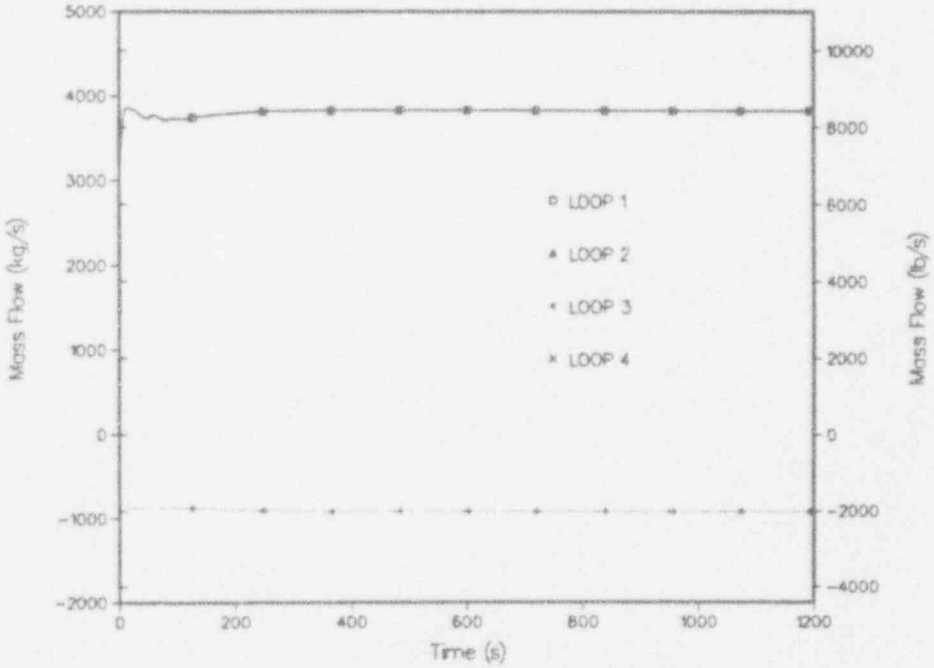

Fig. 13.
Loop flows for a reactor scram with loss of a single reactor coolant pump.

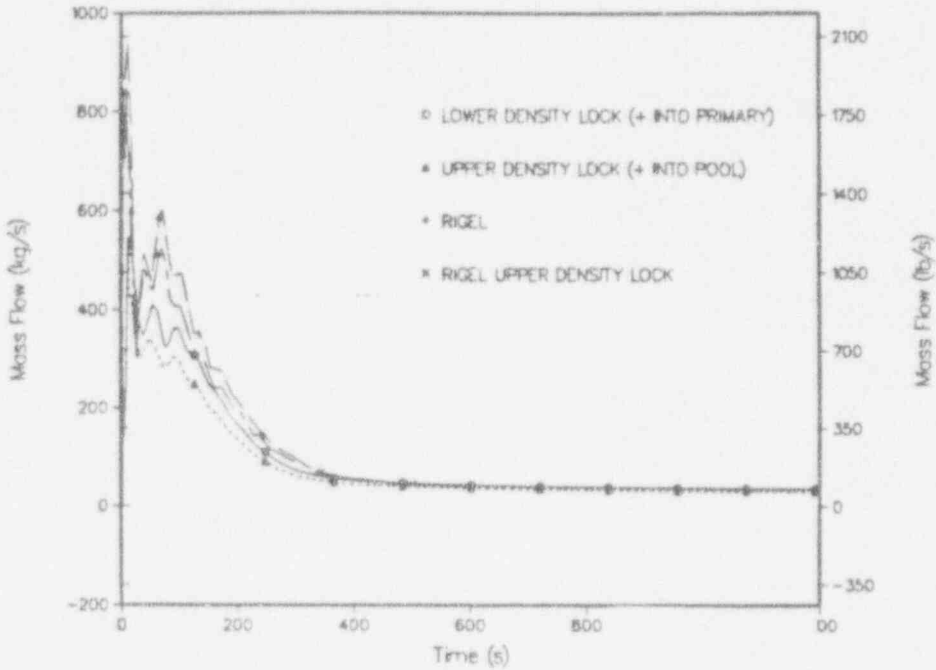

Fig. 14.
Density lock flows for a reactor scram with loss of a single reactor coolant pump.

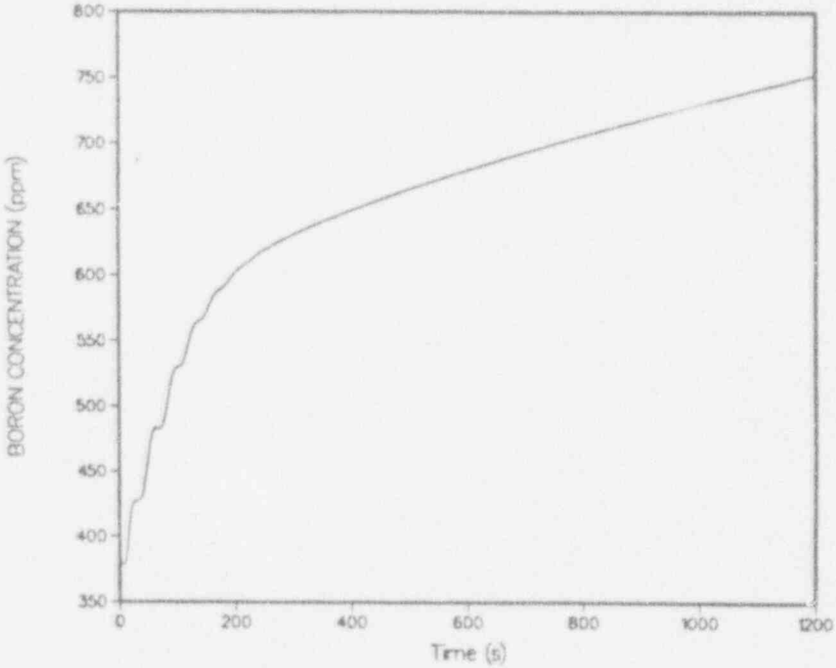

Fig. 15.
Primary boron concentration for a reactor scram with loss of a single reactor coolant pump.

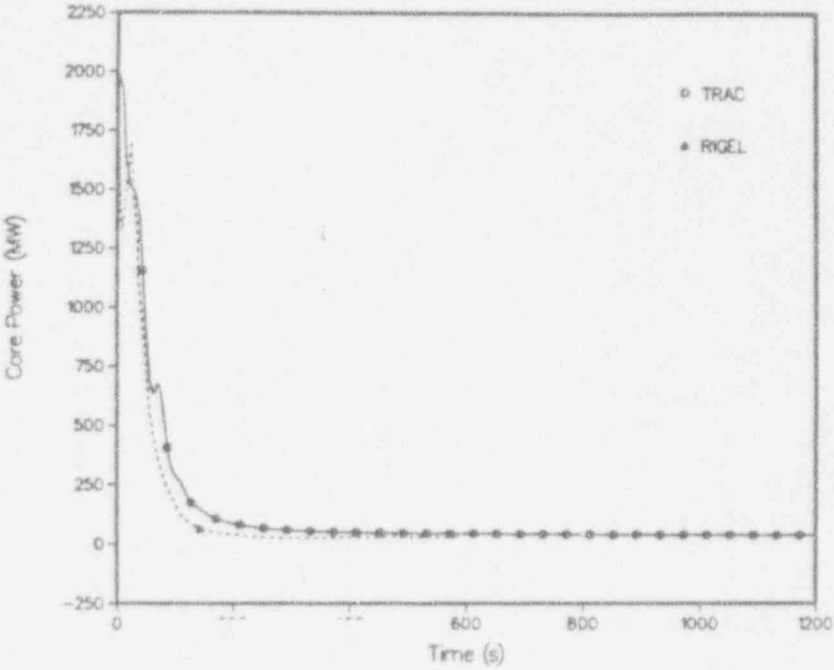

Fig. 16.
Reactor power for a reactor scram with loss of a single reactor coolant pump.

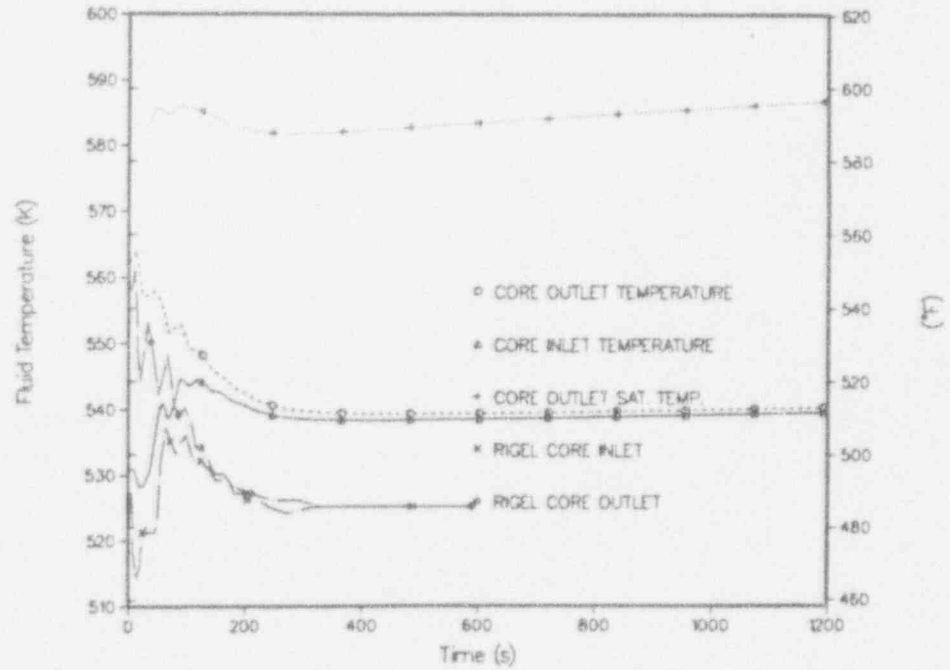

Fig. 17.
Core coolant temperatures for a reactor scram with loss of a single reactor coolant pump.

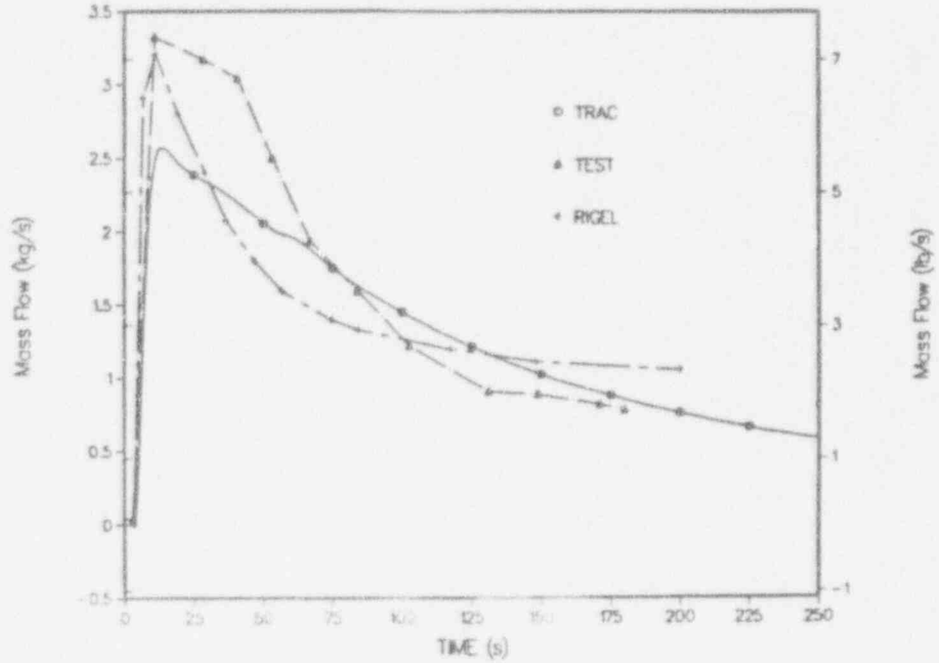

Fig. 18.
Comparison of code-calculated and ATLE lower density lock flows.