ENCLOSURE 5

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Title: LARGE-BREAK LOSS-OF-COOLANT ACCIDENTS IN THE UPDATED PIUS 600 ADVANCED REACTOR DESIGN

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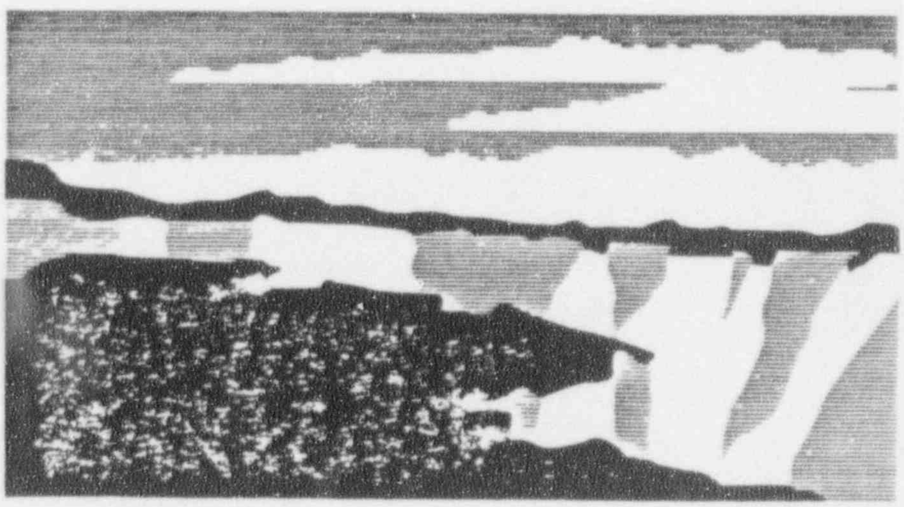

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LARGE-BREAK LOSS-OF-COOLANT ACCIDENTS IN THE UPDATED PIUS 600 ADVANCED REACTOR DESIGN*

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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 normally 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. In 1990, ABB submitted a Preliminary Safety Information Document to the US Nuclear Regulatory Commission (NRC) for preapplication safety review. Early in 1992, ABB submitted a Supplemental Information Package describing recent design modifications. Review and confirmation of these 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. Both multidimensional and one-dimensional (1D) Transient Reactor Analysis Code (TRAC) baseline calculations of the PIUS Supplement design were performed for large break loss-of-coolant accident in the PIUS reactor. Sensitivity studies were performed to explore the vulnerability of the PIUS concept to additional severe off-normal conditions. The sensitivity study results provide insights into the robustness of the design. TRAC and RIGEL calculated results for a cold-leg break just outside the reactor vessel are compared. RIGEL is a 1D thermal-hydraulic system code developed by ABB Atom for PIUS reactor analysis.

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 primary coolant pumps 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

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primary system. PIUS also has a passive scram system that functions should one or more of the reactor coolant pumps (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. The passive scram system can also be activated, even while the RCPs continue to operate, if sufficiently large pressure upsets occur. In either case, 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 reactor coolant pumps. 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 reactor trip and shutdown phenomena following a passive system scram 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 that were performed for large-break loss-of-coolant accidents (LBLOCA) in the PIUS reactor. Sensitivity studies were performed to explore the robustness of the PIUS concept when exposed to severe off-normal conditions following a LBLOCA initiator. The TRAC calculations were performed with a multidimensional (3D) model with four, 1D loops and a fully 1D, four-loop model. Sensitivity studies were performed to explore the vulnerability of the PIUS design to severe off-normal conditions associated with these events. The sensitivity study results provide insights into the resiliency of the design.

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 3D 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, (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 and coolant temperature. All major trends and phenomena were correctly predicted. However, the calculated results were frequently outside the data uncertainty. None of the tests conducted in ATLE simulated severe transients such as loss-of-coolant accidents. Thus, the assessment effort did not simulate some of the key processes and phenomena that are unique to loss-of-coolant accidents.

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 an identical SBLOCA initiator. These comparisons show reasonable qualitative and quantitative agreement in most, but not all, respects. The ability of TRAC to model key PIUS systems, components, processes and phenomena is supported, if not fully demonstrated, by benchmarking TRAC to the RIGEL code.[4] The results of this SBLOCA benchmark comparison will be discussed at an appropriate point in this paper.

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

A schematic of the PIUS reactor design is displayed in Fig. 1. The fully 1D, four-loop TRAC model consists of 74 hydrodynamic components (727 computational fluid cells) and 1 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 (Fig. 2) and 16 components for the reactor vessel (Fig. 3), with the remaining 26 components representing the pool, steam dome, density locks, safety valves, 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 3D four-loop TRAC model includes two vessel components. Each axial segment of the 2 vessels has 4 azimuthal sectors and 4 radial rings. There are a total of 448 computational cells in the 2 vessel components that model the reactor core and internal flow structures. The 4 coolant loops are identical to those in the fully 1D TRAC model.

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
Total core mass flow (kg/s)	12,822	12,880
DC - Riser leakage flow (kg/s)	200.2	200
Loop flow (kg/s)	3,255	3,266
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 with a primary loop boron concentration of 375 parts per million (ppm) and 100% power. The boron concentration in the reactor pool is initially 2,200 ppm. Upon generation of a trip signal, the scram valves open over a period of 2 s, remain fully 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 4.0 MPa (steam drum).

1D BASELINE LBLOCA

The initiating event for the baseline transient is a double-ended guillotine break in one cold leg just outside the steel pressure vessel (loop 3 of the TRAC model). The break flows from the vessel side and the RCP side of the break are shown in Fig. 4 . The larger flow is from the vessel side, and this flow peaks within 3 s at 17,800 kg/s. The RCP-side flow peaks at 8,925 kg/s. Both flows decline rapidly as the primary system pressure decreases (Fig. 5) and voiding in the break flows increases. Immediately after the start of the LBLOCA, flows in both the core and downcomer reverse (Fig. 6). The lower density lock activates (Fig. 7), but the density lock flow joins with the reversed core flow and passes upward through the downcomer to the vessel side of the break. The flow reversal lasts approximately 10 s, and during this period a large fraction of the core reaches saturation temperatures (Fig. 8) and voids (Fig. 9). This period of core voiding is terminated when the downcomer and core flows reverse again and coolant enters the core from the lower plenum. The reversal occurs when flows from the intact cold legs entering the cold-leg plenum and flowing to the break can fully supply the rapidly decreasing vessel-side break flow. Prior to that time, vessel inventory is needed, in addition to the flows from the intact loops, to supply the break flow.

The core power rapidly decreases immediately following the LBLOCA initiator, experiences a sharp rise of 2 s duration beginning at 15 s, and remains at decay levels throughout the remainder of the calculated transient (Fig. 10). Voiding in the core is the single largest negative reactivity insertion early in the transient (Fig. 11). The brief period of criticality beginning at 15 s occurs when the core refills (Fig. 9). The negative void reactivity is eliminated and positive reactivity is inserted by the primary coolant and pool water reentering the bottom of the core. Although the pool water is highly borated and inserts negative reactivity, the primary coolant inserts positive reactivity because it reduces the core fluid temperature, and there is a brief period when the core is critical, resulting in the sharp power rise beginning at 15 s. The power increase, however, causes voiding in the core, which then reduces the power. We note that the point kinetics model may not be adequate for resolving this criticality event. The active scram system is activated shortly after the LBLOCA initiating event. However, the active scram system is only effective for the first 11 s of the transient when the reactor pool level is above the scram-line takeoff from the pool; also, most of the pool water injected through the scram lines is discharged out the break.

A second core flow reversal begins at approximately 20 s and continues until 30 s. Prior to this time, the inlets of the RCPs begin to void and RCP performance degrades (Fig. 12). With the sharp decrease in pumped flow, saturation temperatures are reached in much of the core (Fig. 8), and the resultant void generation (Fig. 9) causes the core flow to reverse (Fig. 6). The magnitude of the reverse core flow peaks at 25 s as shown in Fig. 6. After this time, hot riser fluid entering the core from the top vaporizes in the core (Fig. 9) reducing the downward mass flow at the core inlet (Fig. 6). At approximately 30 s, the voids in the core collapse (Fig. 9) allowing lower plenum fluid to briefly surge into the bottom of the core as shown in Fig. 6. After 30 s the core remains liquid full (Fig. 9), the break flows are significantly reduced (Fig. 4), and a manometer flow oscillation develops between the core and downcomer as shown in Fig. 6. The oscillations continue for the remainder of the calculation, and by the end of the calculation at 200 s the amplitude is reduced because of the very low break flows (Fig. 4) and depressurization rate (Fig. 5). The decay heat at the end of the calculation is removed by the break flow and by pool water circulation into the lower density lock and out of the upper density lock. Neither core dryout nor cladding temperature heatup excursions are calculated (Fig. 13) during the transient. The collapsed liquid level within the internal flow structure containing the core, riser, and pressurizer is presented in Fig. 14. The minimum collapsed liquid level occurs at 55 s. This level is 11 m above the top of the core. The liquid level is generally increasing thereafter.

3D BASELINE LBLOCA

The second baseline LBLOCA calculation was performed with the 3D input model. In major phenomena and trends, the 1D and 3D calculations were similar, although there were some differences in detail. There were no differences that could be specifically attributed to the multidimensional model. We do note, however, that because TRAC currently has only a point kinetics model, potential couplings between multidimensional core kinetics and multidimensional core flows could not be examined in the calculation. The calculated peak break flows for the 1D and 3D baseline transients were similar (Fig. 15), however, the vessel side break flow remained higher in the 3D calculation after the transition to two-phase break flow at 18 s. The higher vessel side break flow resulted in a faster depressurization in the 3D calculation as shown in Fig. 16. The core power exhibited an early decrease to decay heat levels followed by a subsequent power increase to about 920 MWt at about 18 s. The predicted core power increase is slightly less than in the 1D baseline calculation and occurred about 3 s later. The initial core flow reversal (Fig. 17) lasted about 7 s and was terminated when the vessel-side break flow could be supplied by the coolant flows through the intact loops. The subsequent positive core flow was terminated when the inlets of the RCPs voided and pump performance degraded. A second period of reverse core flow then occurred and was terminated at the end of the power increase as voids collapsed in the core. These phenomena were the same as those in the 1D baseline. The following differences were noted. A third period of reverse core flow occurred in the 3D calculation, causing voiding in the core from 55 to 62 s. Core voiding was also predicted from 78 to 110 s because of the lower system pressure in the 3D calculation. The core inlet flow rate displayed smaller oscillations than in the 1D baseline. The peak cladding temperature was approximately 590 K, about 10 K lower than in the 1D baseline.

We completed several sensitivity studies using the 1D model; calculated parameters from the sensitivity studies are provided in Addenda 1–5. The first study examined the response of the PIUS reactor to the baseline LBLOCA initiator concurrent with a 75% blockage of the lower density lock. The phenomena occurring during this low probability transient were similar to the baseline. The same core flow reversal pattern occurred and for the same reasons presented in the baseline discussion. However, during the periods of positive core flow, the flow rates through the core were smaller because the flow entering the primary through the lower density lock was reduced by the lower density lock flow blockage. The amount of boron entering the core through the lower density lock was reduced. The amount of voiding in the core was larger during the second and third core flow reversal periods. Thus, void contributed more to the total negative core reactivity, and boron contributed less during the calculated transient. After the initial decrease in core power immediately following the LBLOCA initiator, a power increase was again calculated. The power increase was to about 1100 MWt, less than in the baseline. The peak cladding temperature during the transient was 600 K. Neither cladding dryout or cladding heatup were predicted. The second sensitivity study examined the response of the PIUS reactor to the baseline LBLOCA initiator concurrent with a reactor pool boron concentration of 1800 ppm. The course of this transient was nearly identical to the baseline with one exception. The core power increase beginning about 15 is more severe than in the baseline because there is less negative reactivity inserted into the core from the pool at 1800 ppm. There is, however, no core dryout or heatup. The peak cladding temperature is again about 600 K. The third sensitivity study examined the response of the PIUS reactor to the baseline LBLOCA concurrent with a failure of the active scram system. As discussed for the baseline transient, the active scram system is only effective for the first 11 s of the transient, after which the reactor pool drops below the level of the scram-line takeoff from the pool. Because the core flow is reversed for the first 6.5 s of the transient, the active scram system has limited impact on the course of the baseline transient. Thus, the course of the transient for the sensitivity calculation was nearly identical to the baseline calculation.

A RIGEL calculation of a LBLOCA in the cold leg piping was reported in Refs. 10 and 11. Several results from the RIGEL calculations have been co-plotted with the TRAC-calculated results for this transient. In general, the TRAC- and RIGEL-calculated results display the same phenomena and trends. There are, however, differences in the details. Explanations are provided, when possible, but the limited number of plots in Ref. 11 does not permit a detailed comparison of all phenomena and component behaviors. The calculated break flows are compared in Fig. 4 (Frame 33). The RCP-side break flows are similar. The RIGEL-calculated peak vessel-side break flow is about 23,000 kg/s, while the TRAC-calculated maximum flow is 17,800 kg/s. This result suggests that there may be differences between the RIGEL and TRAC critical flow models. However, the overall break flow trends calculated by the two codes are similar. A comparison of the primary system pressures is provided in Fig. 5 (Frame 2). Again the trends are similar, although the TRAC-calculated pressure is less than the RIGEL calculated pressure throughout most of the transient. An immediate reversal of the downcomer and core flows and the complete bypass of the lower density lock flow is predicted by both codes. However, the magnitude of the RIGEL-calculated peak reversed core flow is greater than the TRAC-calculated peak flow; the flows are approximately 10,000 and 3,700 kg/s, respectively. This result is consistent with the peak vessel-side break flow calculated by RIGEL, which was approximately 5,200 kg/s larger than that calculated by TRAC. Both the TRAC and RIGEL-calculated core flow reversals last until nearly 10 s, leading to a similar increase in core voiding (Fig. 9). RIGEL, however, calculates a heap of the hot rod (Fig. 13) since the RIGEL core model has separate channels for the high power and average power core regions, with the high power rod connected to the smaller hot channel. The TRAC model has a single average channel for both hot and average power rods, and average channel fluid conditions are used to determine the heat transfer from the high power rod. Because of this modeling difference, RIGEL calculated a heatup of the high power rods, which did not occur in the TRAC calculation.

Once RIGEL predicts the termination of the reverse core flow at approximately 10 s, a positive core flow is established, which lasts until 23 s. As explained above, TRAC predicts that the core flow is reduced once the inlets to the RCPs void and the RCP performance degrades. As a consequence of this difference in core flows and the lower TRAC system pressure (Fig. 5), voiding once again begins in the core at 16 s and continues to 33 s. RIGEL does not predict the occurrence of this event. It may be that the RCP pump performance in the RIGEL calculation degrades later than in the TRAC calculation, but Ref. 11 does not contain this information. We conclude with the observation that the both TRAC and RIGEL predicted the same major phenomena and processes. There were important differences in details, particularly with respect to the magnitude of the vessel-side break flow. These early differences were related to subsequent differences in the calculated results. Although these are important details, we do note that the two codes predicted the same major processes and the same end state, a shutdown reactor with no core damage.

SUMMARY OBSERVATIONS

1. The passive scram system successfully accommodates the baseline LBLOCA 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 designed, is characterized by compensating shutdown mechanisms. The primary shutdown mechanism in an LBLOCA transient is the negative reactivity from voiding in the core. When there are no voids in the core and highly borated pool water enters the primary through the lower density lock, the negative reactivity associated with the boron is the primary mechanism for decreasing the core power. The moderator and fuel temperature contributions reactivity are positive in such circumstances. However, negative reactivities are inserted via both the moderator and fuel temperatures when the boron entering the core is not sufficient to prevent reactor power increases.

3. Our confidence in the baseline simulations is upheld by the assessment activity performed using ATLE data. The ATLE processes and phenomena were correctly predicted by TRAC. However, the phenomena in the ATLE tests conducted to date are not fully representative of LBLOCA conditions, as no test simulates a loss-of-coolant accident or voided primary system conditions. Moreover, there are quantitative discrepancies between key TRAC-calculated parameter values and the ATLE data. We would like to better understand the reasons for these differences should the PIUS design certification effort resume. 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 predicted outcomes of the baseline simulations is enhanced by the code benchmark comparisons that were performed for the cold-leg LBLOCA. 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. We do not feel that the differences are of sufficient import to alter the summary observations presented herein.

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 nominal operating conditions and to successfully bring the reactor to a hot shutdown condition. 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 core neutronic and thermal-hydraulic effects and multidimensional effects are important. We believe that it will be important to investigate such effects if the PIUS reactor moves to the design certification stage.

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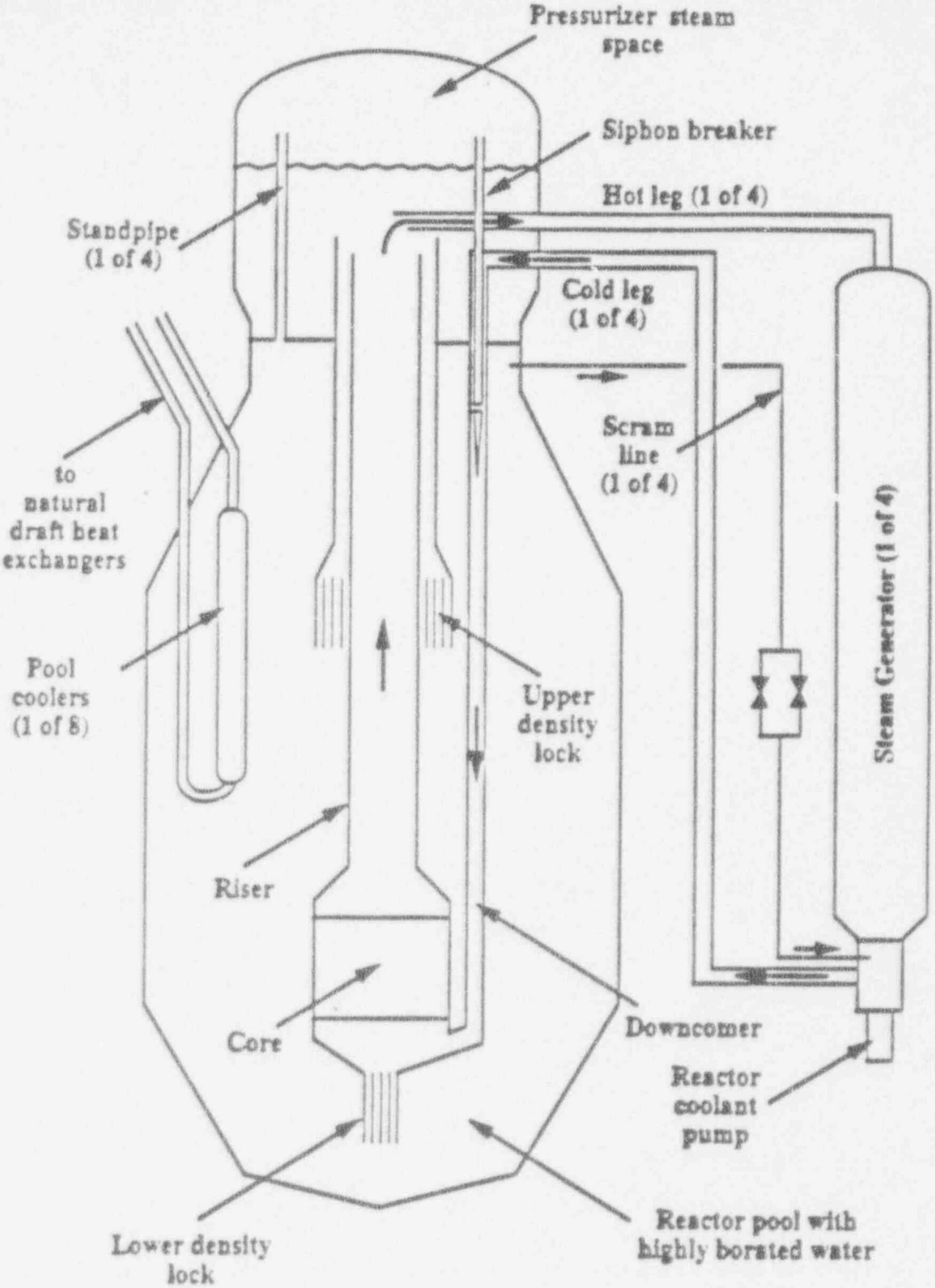

Fig. 1.
Schematic of PIUS 600 reactor design.

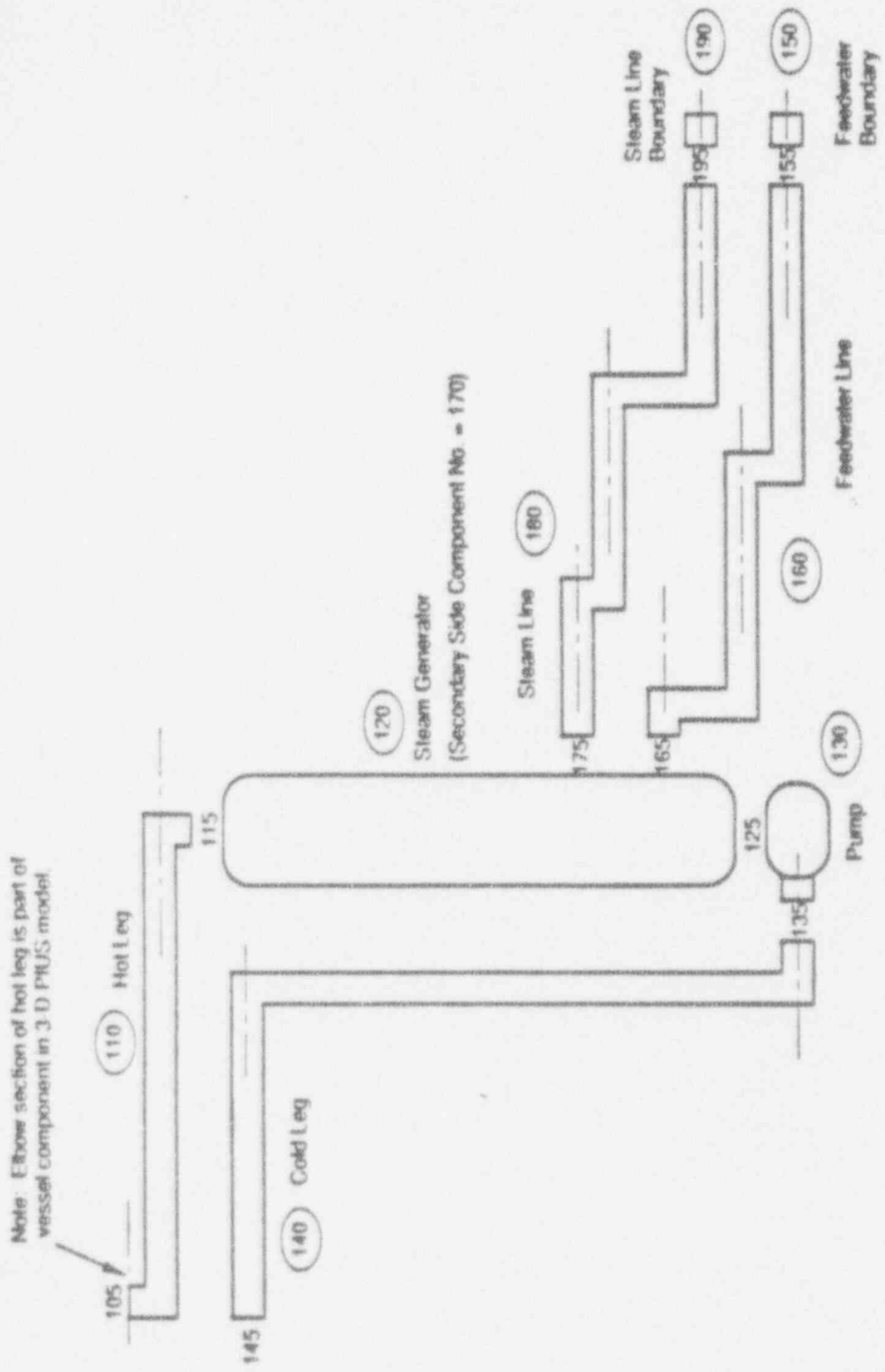

Fig. 2.
TRAC model of PIUS reactor coolant loop.

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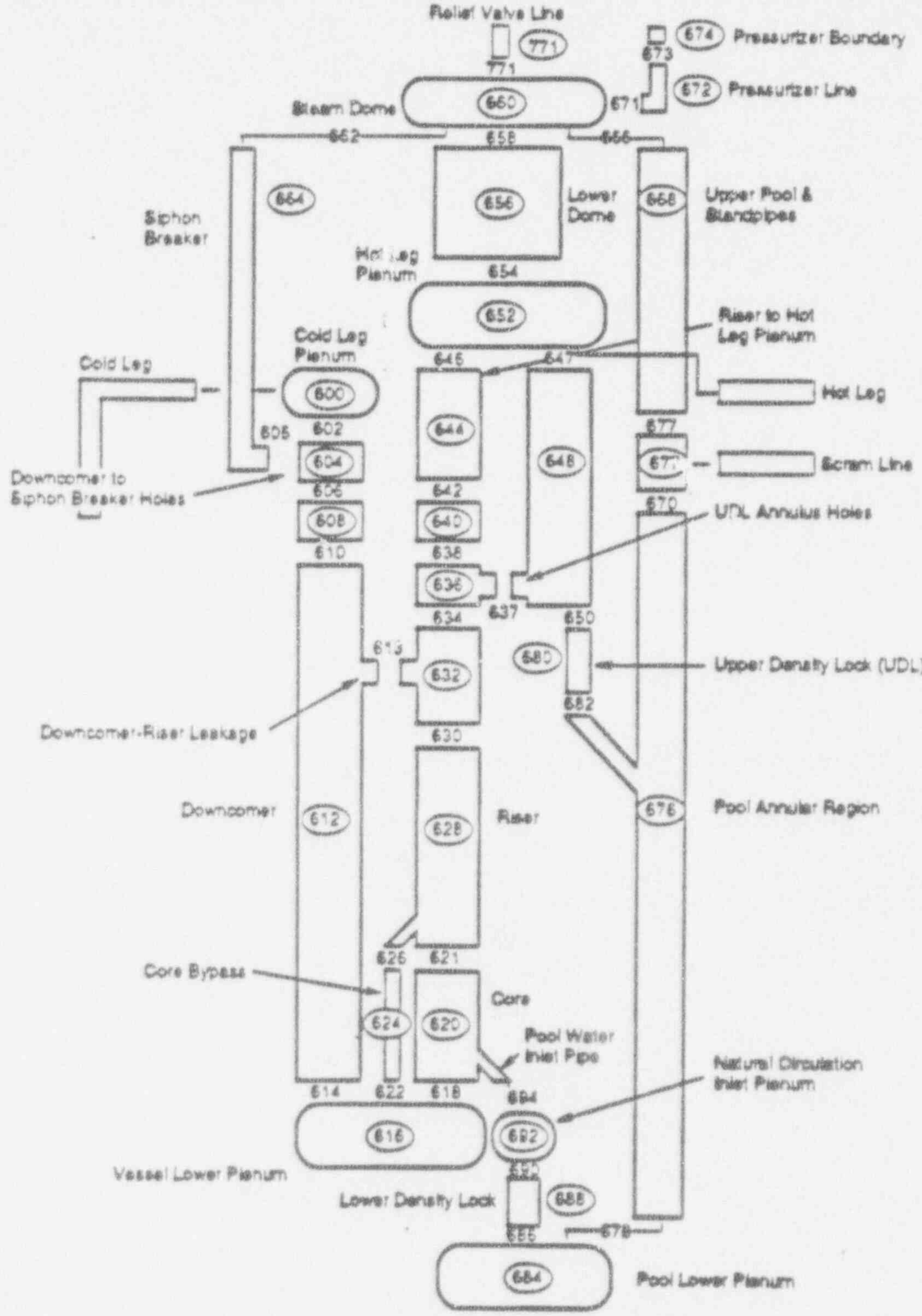

Fig. 3.
TRAC 1D Model of PIUS vessel and pool.

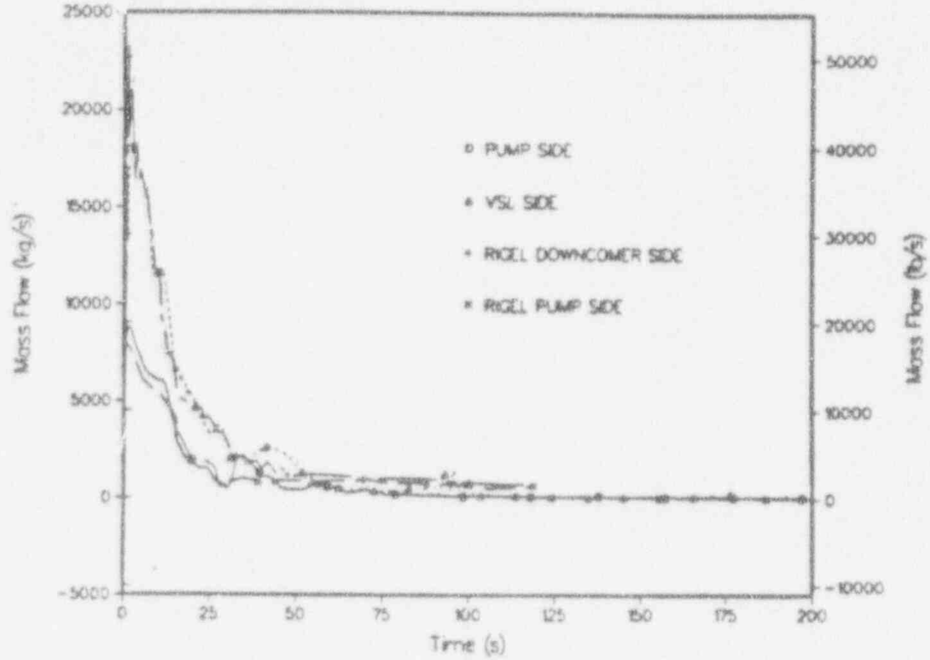

Fig. 4.
Break flow rate, four loop 1D model.

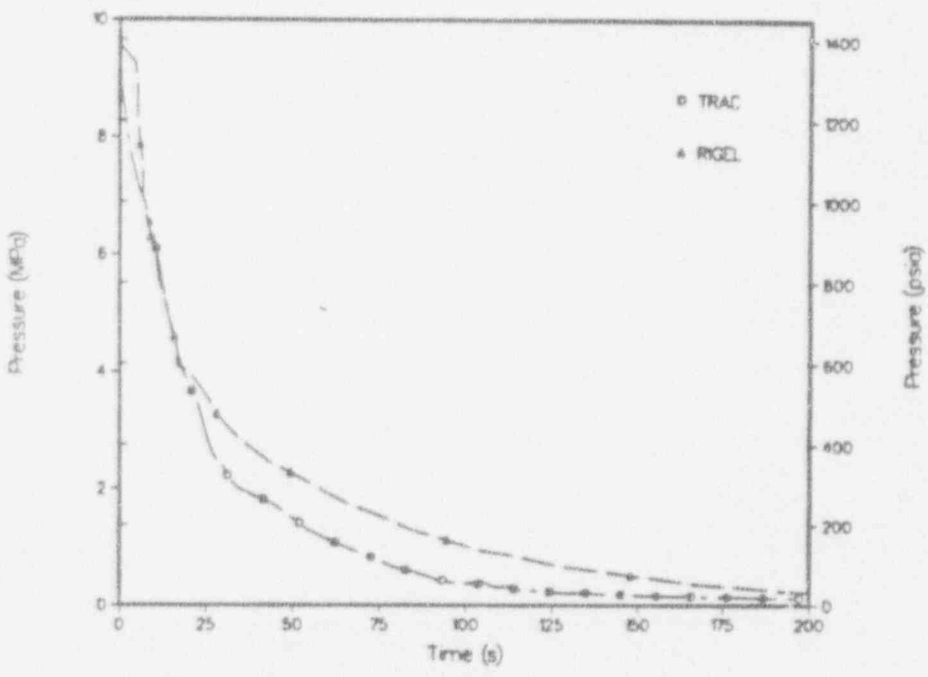

Fig. 5.
Primary system pressure, four loop 1D model.

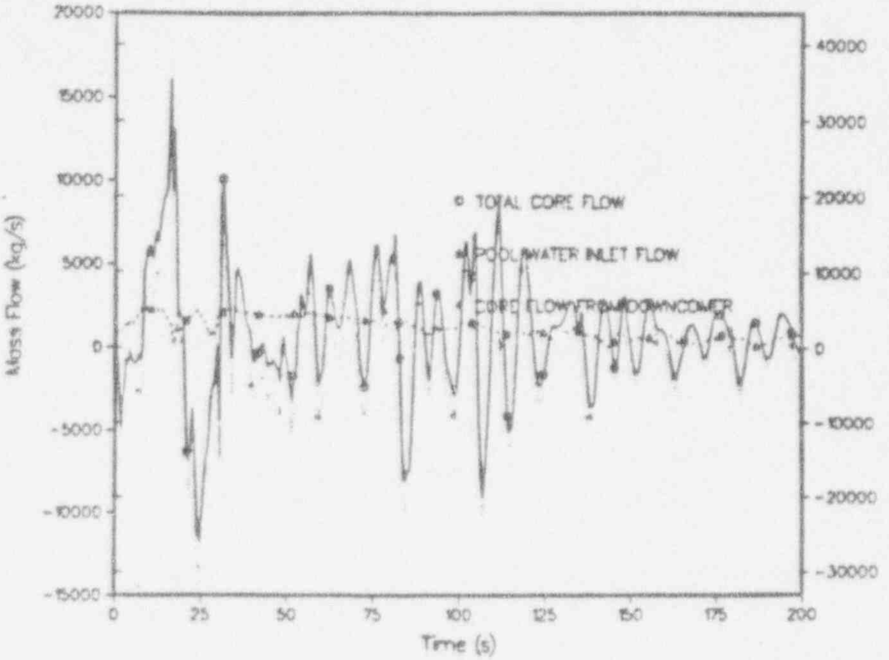

Fig. 6.
Core flow rate, four loop 1D model.

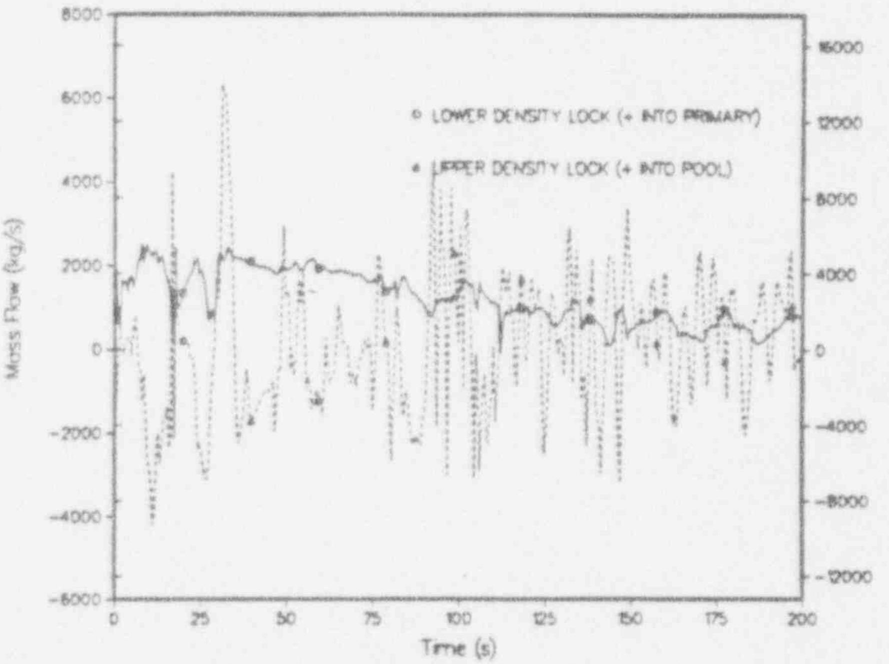

Fig. 7.
Density lock mass flows, four loop 1D model.

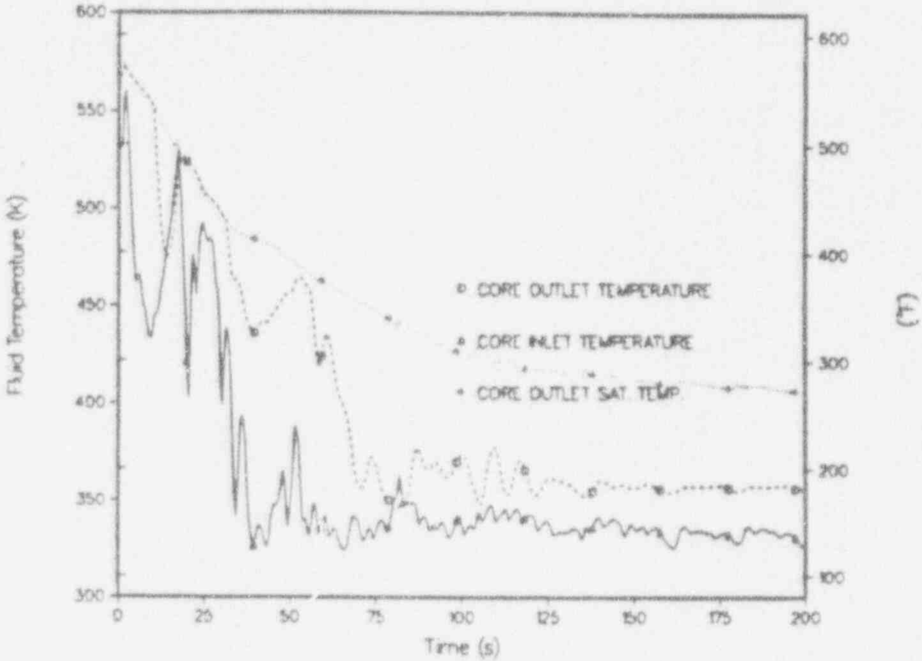

Fig. 8.
Core temperatures, four loop 1D model.

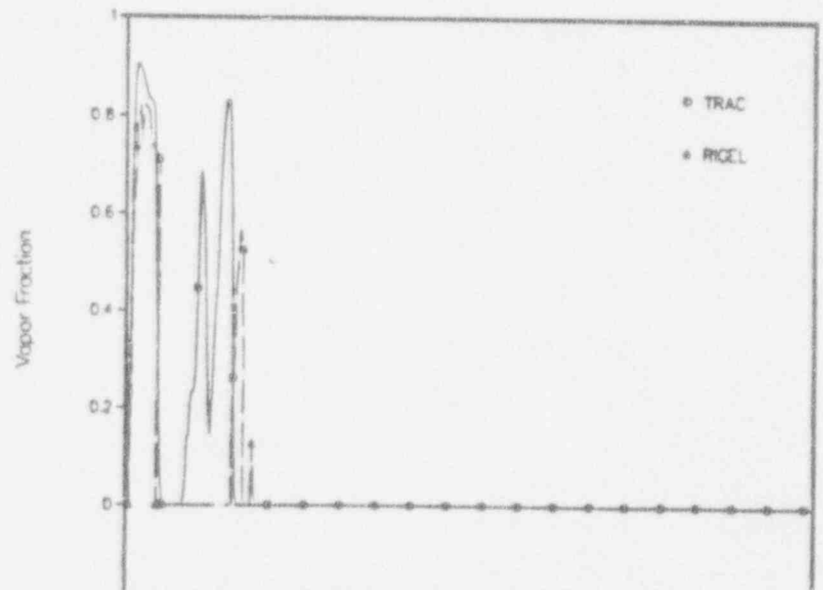

Fig. 9.
Core average void fraction, four loop 1D model.

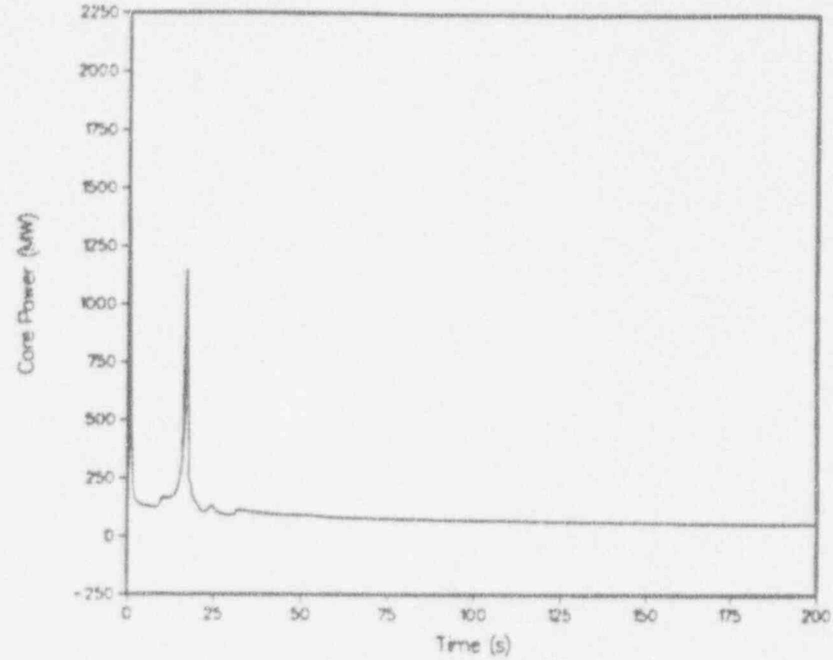

Fig. 10.
Reactor power, four loop 1D model.

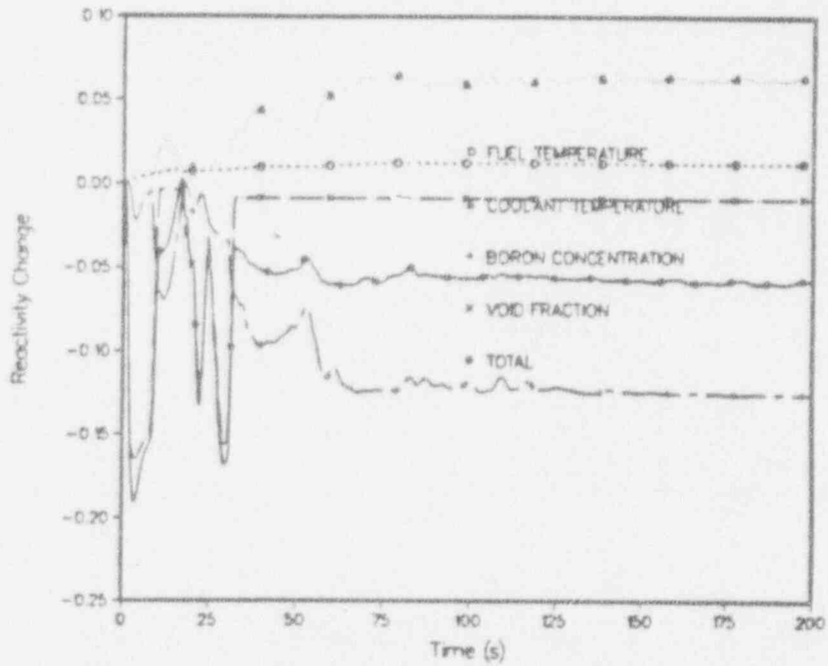

Fig. 11.
Individual reactivity changes, four loop 1D model.

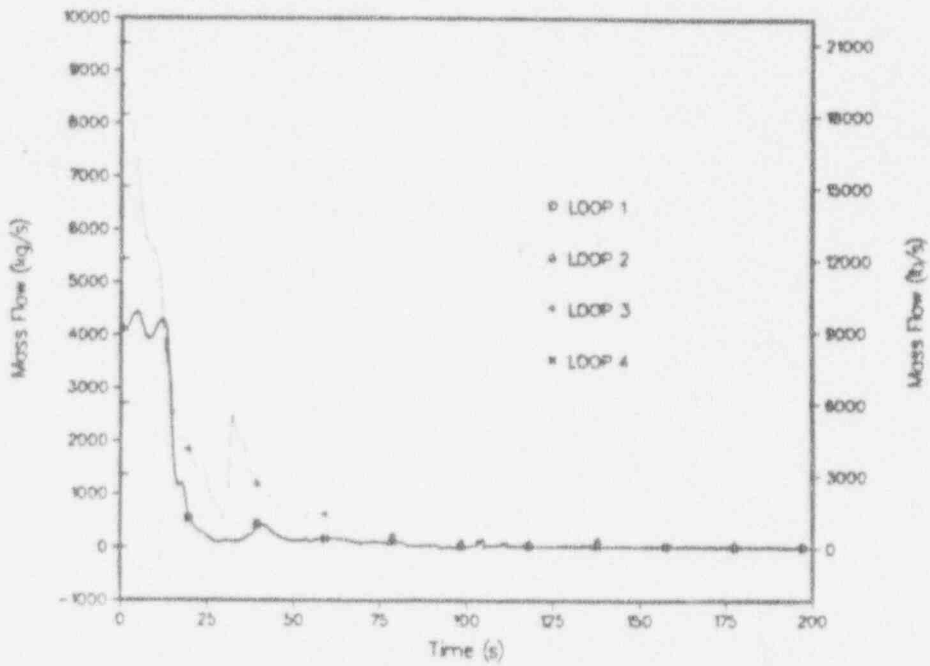

Fig. 12.
Pump mass flow, four loop 1D model.

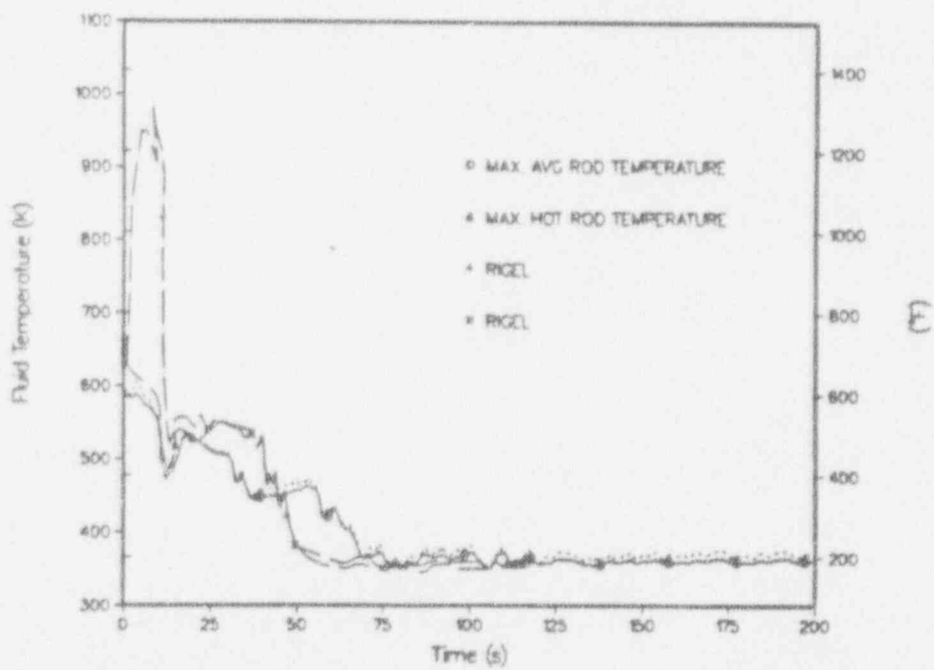

Fig. 13.
Rod temperatures, four loop 1D model.

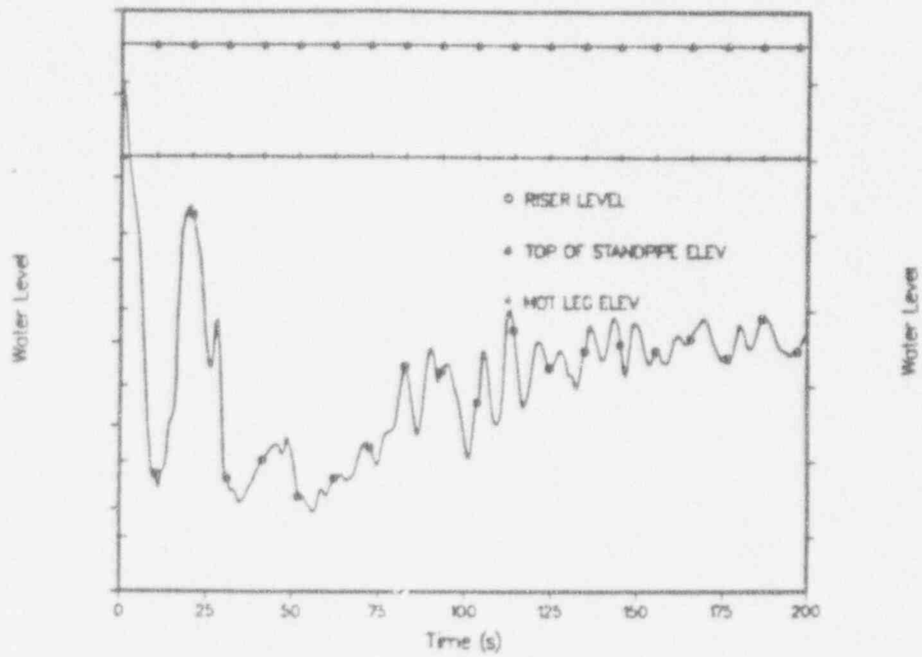

Fig. 14.
Bottom of core to top of steam dome collapsed liquid level, four loop 1D model.

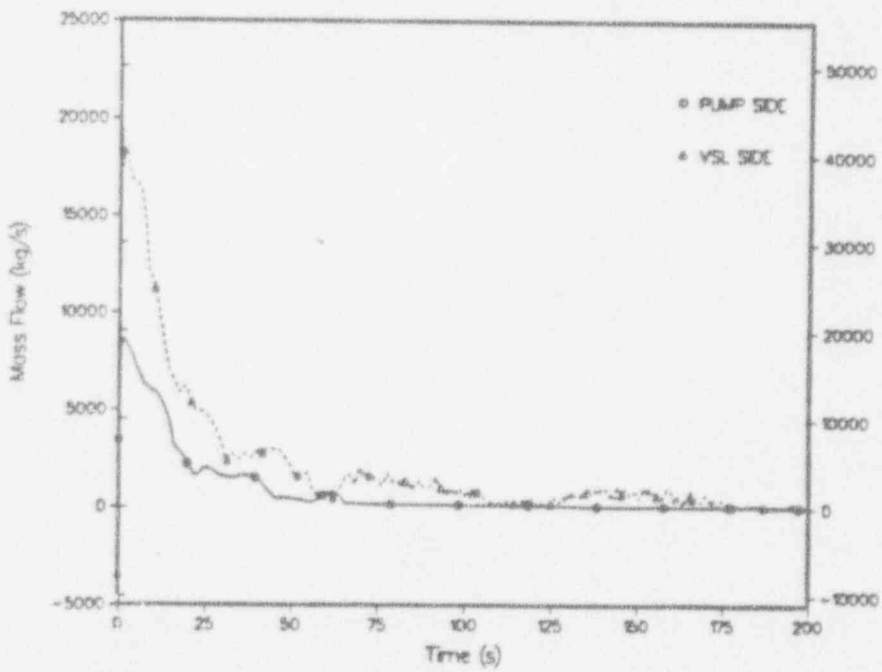

Fig. 15.
Break flow rate, four loop 3D model.

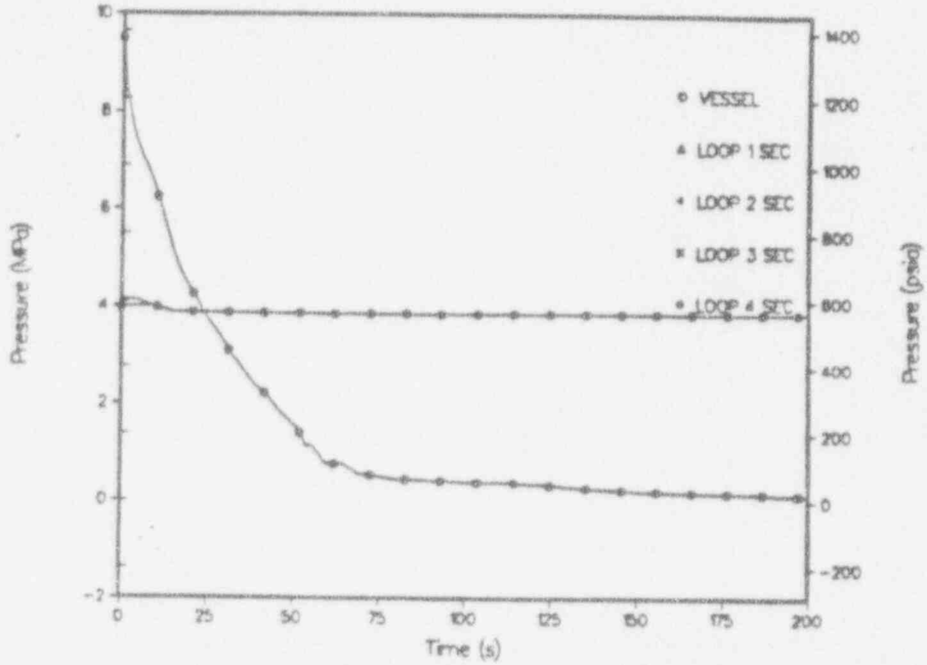

Fig. 16.
Vessel and steam generator secondary pressures, four loop 3D model.

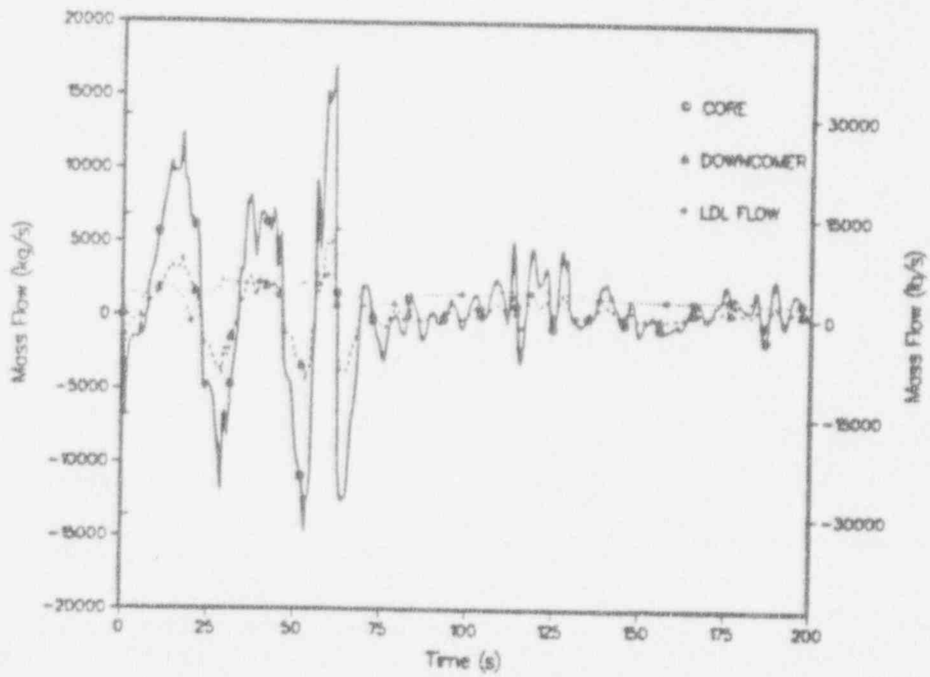

Fig. 17.
Core, downcomer, and lower density lock mass flow rates, four loop 3D model.

January 31, 1994

If you have concerns regarding these papers, please contact me at (301) 492-1104.

Sincerely,

Original signed by:

Dino C. Scaletti, Senior Project Manager
Advanced Reactors Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

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