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ONE-DIMENSIONAL TRAC CALCULATIONS OF A PUMP-TRIP SCRAM FOR THE PIUS 600 ADVANCED REACTOR DESIGN*

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SUMMARY

One-dimensional TRAC transient calculations of the process inherent ultimate safety (PIUS) advanced reactor design were performed for a pump-trip SCRAM. The TRAC calculations showed that the reactor power response and shutdown were in qualitative agreement with the one-dimensional analyses presented in the PIUS Preliminary Safety Information Document (PSID) submitted by Asea Brown Boveri (ABB) to the US Nuclear Regulatory Commission for preapplication safety review. The PSID analyses were performed with the ABB-developed RIGEL code. The TRAC-calculated phenomena and trends were also similar to those calculated with another one-dimensional PIUS model, the Brookhaven National Laboratory developed PIPA code. A TRAC pump-trip SCRAM transient has also been calculated with a TRAC model containing a multi-dimensional representation of the PIUS internal flow structures and core region. The results obtained using the TRAC fully one-dimensional PIUS model are compared to the RIGEL, PIPA, and TRAC multi-dimensional results.

INTRODUCTION

PIUS is a four-loop pressurized water reactor with a nominal core rating of 2000 MWt and 640 MWe (Ref. 1); the design is being developed by Asea Brown Boveri (ABB). The basic PIUS primary system design arrangement is shown in Fig. 1. A primary design objective was to eliminate any possibility of a core degradation accident. Reactivity is controlled by coolant boron concentration and temperature, and there are no mechanical control 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. There are no mechanical devices (e.g., valves) that separate the primary system and the pool; the density locks provide a continuously open flow path. 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. A reactor SCRAM is accomplished by tripping one of four primary coolant pumps, thereby eliminating the balance between the primary coolant loop and the pool. Highly borated water from the pool enters the primary coolant via natural circulation, and this process produces a reactor shutdown. 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 must be understood.

ABB has submitted a Preliminary Safety Information Document (PSID) to the US Nuclear Regulatory Commission (NRC) for preapplication safety review (Ref. 2). 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

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operation, transient conditions, and specified accident sequences. Review and confirmation of these safety analyses for the PIUS design constitutes an important activity in the NRC's preapplication review. The PSID describes the PIUS design and safety characteristics. It includes safety analyses based in part upon plant transient calculations performed with the ABB proprietary code RIGEL, which is based upon a fully one-dimensional (1-D), four-loop model of the PIUS reactor.

This paper summarizes the results of TRAC calculations of a key PIUS transient, the pump-trip SCRAM with a fully 1-D, four-loop and single lumped-loop models. The former model is similar to the RIGEL model of the PIUS reactor developed by ABB. The latter model is similar to the PIPA model of the PIUS reactor developed by Brookhaven National Laboratory (Ref. 3). The pump-trip SCRAM transient was selected as the first application of the TRAC fully one-dimensional models because PIUS is the only commercial or advanced reactor concept to rely on boron injection for the primary means of reactivity control. A similar TRAC pump-trip SCRAM transient calculation has been completed with a TRAC model containing a multi-dimensional representation of the PIUS the internal flow structures and core region (Ref. 4). A comparison of the results obtained using the TRAC fully one-dimensional, four-loop PIUS model and the TRAC four-loop model containing a multi-dimensional model of the PIUS internal structures and core region is provided.

TRAC MODEL DESCRIPTION

Two 1-D TRAC models were developed for PIUS, a four-loop model and a one-loop model with a single loop used to represent the four PIUS loops. Figures 2 and 3 display the reactor vessel and coolant loop components of the TRAC 1-D model. The four-loop TRAC model consists of 57 hydrodynamic components (746 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 model has eight hydrodynamic components in each coolant loop, 17 components for the reactor vessel, and the remaining eight components representing the pool, steam dome, density locks, and pressurizer line. The geometric input data for the coolant loop and secondary components in the one loop model were scaled to represent all four PIUS loops. The TRAC 1-D models are more finely noded than either the RIGEL or PIPA models.

A steady-state calculation was performed and compared with the steady-state specifications in the PSID. The TRAC-calculated and PSID steady-state values are tabulated below for comparison.

	TRAC	PSID
Core mass flow (kg/s)	13032	13000
Core bypass flow (kg/s)	792	800
Core inlet temperature (K)	536.3	533.2
Core outlet temperature (K)	568.9	563.2
Pressurizer pressure (MPa)	9.0	9.0
Steam exit pressure (MPa)	3.97	4.0
Steam exit temperature (K)	548.3	543
Steam flow superheat (°C)	25.6	20
Steam and feedwater mass flow (kg/s)	253	253

PUMP-TRIP SCRAM TRANSIENT (FOUR-LOOP MODEL)

A full SCRAM of the PIUS reactor is initiated by tripping one of the main coolant pumps while the remaining three pumps continue to operate in a normal way. For the pump-trip SCRAM calculation, the following assumptions were made:

(a) Initial boron concentration in the primary system was set to 375 ppm, representing a beginning-of-cycle (BOC) concentration, to conform with PIUS PSID calculation assumptions. The pool boron concentration was set to 2110 ppm.

(b) The reactor coolant (RC) pump in loop 3 was tripped, and the pump speed was assumed to decrease to zero rpm in 6 s. Reverse pump-impeller rotation was not allowed.

(c) The non-tripped pumps were assumed to increase speed to 105% of nominal in 4 s following the single pump trip.

(d) Feedwater flow to all steam generators was decreased linearly from full flow to zero flow in 20 s at time of pump trip.

(e) A zero-flow fill boundary was used for a pressurizer boundary to simulate an inactive pressurizer pressure control system.

(f) The steam line pressure boundaries were maintained constant at 3.9 MPa for the entire transient.

(g) The initial pool water temperature was assumed to be 303 K. The water in the natural-circulation inlet plenum was calculated to be the same as the core inlet temperature, 535.6 K.

The pump-trip SCRAM transient was calculated to 1200 s. The sequence of events for this transient is given in Table I. Following the pump trip in loop 3, feedwater flow to all four loops was decreased linearly to zero over a 20 s interval. The pumps in the other three loops increased in speed to 105% in 4 s and remained at a constant speed thereafter. The flows in the loops with operating pumps increased slightly with the increased pump speed. The flow in the loop with the tripped pump reversed. The net effect of the pump trip was a decrease in core flow and a pressure imbalance between the primary system and the pool, that allowed highly borated pool water to flow in through the lower density lock thereby shutting the reactor down.

Calculation results for the first 150 s of the transient show the thermal-hydraulic behavior associated the pressure imbalance and boron injection. A shutdown in reactor power was achieved as shown in Fig. 4. An initial decrease in power (first 5 s of the transient) was due to the reduced core flow, which caused the fuel and core coolant temperatures to increase slightly, inserting negative reactivity. A few seconds later, the highly borated pool water reached the core, causing a sharp drop in power. Figure 5 shows the boron concentration at the core inlet. The boron concentration closely follows the lower density lock flow shown in Fig. 6. Figure 7 shows the reactivity changes due to fuel temperature, coolant temperature, voiding, boron concentration, and the net total of these components. The only negative reactivity after 10 s is from boron addition.

The power reversal starting at 12 s was due to a decrease in pool water inlet flow caused by colder core inlet flow that comes from the mixing of pool water flow (temperature of 303 K) with reactor coolant flow (temperature of 535.6 K). There are two effects from the colder coolant flow. First, the colder coolant flow is more dense and increases the pressure forces acting on the lower density lock, which decreased the pool water inlet flow. Second, the colder coolant flow increases the reactor power because of positive reactivity insertion from the decrease in coolant temperature. The power increased from 48% to 74% and then decreased for the rest of the transient.

The flows into the primary system from the pool through the lower density lock and from the primary system into the pool through the upper density lock equalized at about 700 s (Fig. 8). At this point, the conditions of stable hot-shutdown were established. Core inlet boron concentration, Fig. 9, continued to increase from the natural circulation flow established between the pool and primary system. The fuel and core coolant temperatures are shown in Fig. 10. Core fluid inlet, fluid outlet, and maximum fuel temperatures exist within a narrow band; the core has effectively reached a hot-shutdown condition early in the transient. The reactor power was reduced to 1.9% of full power at 1200 s, with most of the power reduction occurring in the first 200 s following scram initiation.

At the same time the pump-trip SCRAM was initiated, the steam generator feedwater flows were terminated, decreasing to zero over an interval of 20 s. The steam generator secondaries in loops 1, 2, and 4 dried out 65 s following event initiation. The loop 3 steam generator secondary is dried out by 175 s; the tripped-loop steam generator dry out took longer because the primary-side flow was reversed, feeding the steam generator with fluid from the cold legs of the other three loops instead of hot fluid from the core exit. After 175 s, the steam generators could no longer receive heat from the primary system and all core-generated heat was rejected to another heat sink, the highly borated pool in which the reactor core and internal structures are immersed. The interval between SCRAM initiation and 175 s was a period of transition in which a decreasing amount of core-generated thermal energy was rejected to the steam generators and an increasing fraction of the heat load was deposited in the pool. The PIUS design includes both active and passive pool cooling systems. Neither of these systems are in the current TRAC model. However, the thermal mass of the pool is sufficiently large that the lack of models for these systems does not compromise the calculated results over the relatively short time intervals analyzed.

The pump-trip SCRAM results calculated with the four-loop RIGEL model were in reasonable agreement with the four-loop TRAC 1-D results. The same phenomena and trends were calculated although there were differences in the magnitude and timing of events. The major modeling differences and calculation results are compared in Table II. Several differences of note are the comparative magnitudes of the core power reversals, the reverse flows in the tripped loop, and the core boron concentration. There are three possible causes of the discrepancies: (1) different interpretation of the PIUS design dimensions, features, etc., (2) different models (e.g., noding) of the PIUS design, and (c) differences in the codes used. The reasons for the discrepancies are not known at the present; we do emphasize, however, that the same major phenomena and trends were calculated by the two codes. We also note that the RIGEL code has been assessed against experimental data from a scaled experimental facility of PIUS while the TRAC code has not. TRAC assessment using data from this facility is currently in progress.

PUMP-TRIP SCRAM WITH SINGLE-LOOP AND MULTI-DIMENSIONAL TRAC MODELS

In addition to the one-loop and four-loop 1-D models, a multi-dimensional TRAC model of the PIUS reactor was developed. The upper and lower sections of the reactor vessel were modeled in three dimensions with two VESSEL components. The rest of the reactor coolant system was modeled with one-dimensional components. A point-kinetics model of the core neutronics is used. Pump-trip SCRAM calculation results for the multi-dimensional TRAC model are reported in Ref. 4. A comparison of the core power calculated with the three TRAC models transients is presented in Fig. 12. The pump-trip SCRAM results are in reasonable agreement with the calculated results of RIGEL and PIPA. Again, the same phenomena and trends were calculated although there were differences in the magnitude and timing of events. The major modeling differences and calculation results are compared in Table II. The major difference of note is the comparative magnitudes of the core boron concentration.

Although major calculation results are similar, the multi-dimensional TRAC model showed several multi-dimensional effects that would not be calculated with a one-dimensional model. The 3-D TRAC calculation showed a nonuniform distribution of boron across the core inlet caused by a recirculation flow between the lower core and the natural-circulation pool water plenum through the pool water inlet pipes. The non-uniform boron distribution cannot be calculated with one-dimensional models. Because of the potential for uneven boron distribution in the core, we analyzed a pump-trip SCRAM with pool water inlet orificing. Figure 12 compares the reactor power calculated with and without pool water inlet pipe orificing. The orificing effectively damped out the power reversal without significantly affecting the overall reactor shutdown.

PRELIMINARY OBSERVATIONS

TRAC calculations have shown that for the pump-trip SCRAM, it is likely that the reactor can be brought to a safe hot shutdown condition. We note, however, that a formal TRAC adequacy assessment for the PIUS application has not been conducted, nor has the code been benchmarked against data from a PIUS prototypic test facility. An NRC-funded effort to benchmark TRAC using data from the ATLE test facility (Ref. 5) is currently in progress.

For the single-pump SCRAM, the TRAC one-dimensional and multi-dimensional results are in reasonable agreement with those calculated by other codes. The same key phenomena and processes are calculated by all codes, differing primarily in the magnitude and timing of parameter values. Non-uniform boron concentrations in the core were calculated with the TRAC multi-dimensional model. Orificing of the pool water inlet pipes by radial sectors may be necessary to obtain a uniform distribution of boron entering the core through the lower density lock. The impact of axial and radial variations of boron concentrations in the core on core neutronic performance during the pump-trip SCRAM are not currently known, since a point-kinetics model of the core neutronics was used. We plan to repeat the pump-trip SCRAM analysis using a version of TRAC with a three-dimensional neutronics capability in the near future.

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TABLE I

SEQUENCE OF EVENTS FOR THE PUMP-TRIP SCRAM WITHOUT POOL WATER INLET PIPE ORIFICING

Time	Event
0 s	Reactor coolant pump tripped. Decrease in feedwater flow to all steam generators initiated. Pump speed in other loops start to increase.
3 s	Power starts to decrease as fuel and core temperatures increase from decrease in core flow (negative reactivity is inserted when fuel and coolant temperature increase).
4 s	Non-tripped pumps reach 105% speed and remain at that speed for rest of transient.
4 s	Highly borated pool water flow reaches core; power continues to decrease.
6 s	Tripped pump ceases to rotate.
7 s	Coolant flow reverses in loop with tripped pump.
10 s	Inflow of pool water into primary system reaches a maximum of ~1400 kg/s, and then starts to decrease rapidly. The decrease in pool water flow is because of higher opposing pressure forces on lower density lock from higher coolant density.
12 s	Power decreases to 48% and then starts to increase when colder coolant flow reaches core.
20 s	Feedwater flow decreases to zero.
22 s	Power increases to 74% and then starts to decrease as core temperatures increase. Power continues to decrease for rest of transient.
65 s	Steam-generator secondary side voids in loops with pumps operating.
150 s	Reactor power at 5.2%.
175 s	Steam-generator secondary side voids in loop with tripped pump.
700 s	Lower and upper density lock flows equal
1200 s	Calculation terminated, reactor power at 1.9%

TABLE II

COMPARISON OF TRAC PUMP-TRIP SCRAM CALCULATIONS TO PUMP-TRIP SCRAM CALCULATIONS BY OTHER CODES

MODEL:	TRAC-1D FOUR LOOP MODEL	RIGEL	TRAC-1D ONE LOOP MODEL	PIPA	TRAC-3D
Basic Reactor Model	4 loops, 1-D vessel	4 loops, 1-D vessel	1 loop, 1-D vessel	1 loop, 1-D vessel	4 loops, 3-D vessel
Feedwater Flow	Trip at time zero, decrease linearly to zero flow in 20 s	Trip at time zero, decrease linearly to zero flow in 20 s	Trip at time zero, decrease linearly to zero flow in 20 s	Trip at 18 s, rate decrease not known	Trip at time zero, decrease linearly to zero flow in 20 s
Pressurizer pressure control system modeled?	Deactivated	Deactivated	Deactivated	Yes	Deactivated
Pump coastdown time	6 s to zero rpm	6 s to -35% speed	6 s to 73% speed (1)	10 s to 73% speed (1)	6 s to zero rpm
Pump reverse rotation	Not allowed	Yes	Not applicable	Not applicable	Not allowed
Maximum reverse flow in tripped-pump loop.	-907 kg/s	-600 kg/s	Not applicable	Not applicable	-1000 kg/s
Magnitude of power reversal due to decrease in pool water flow (2)	48% to 74%	32% to 92%	42% to 82%	38% to 86%	50% to 81%
Core average temperature at 150 s	534.9 K	526 K	533.3 K	Not applicable	548.6 K
Core boron concentration at 150 s	647 ppm	765 ppm	644 ppm	770 ppm	538 ppm
Reactor power at 150 s	5.2%	5%	6.1%	4%	7.9%

(1) Tripped pump is modeled by reducing speed of the one modeled pump to approximately 3/4 speed.
(2) Each code calculation showed a power reversal in the first 30 s of the transient, caused by a decrease in pool water inlet flow as a consequence of colder coolant flow in the riser.

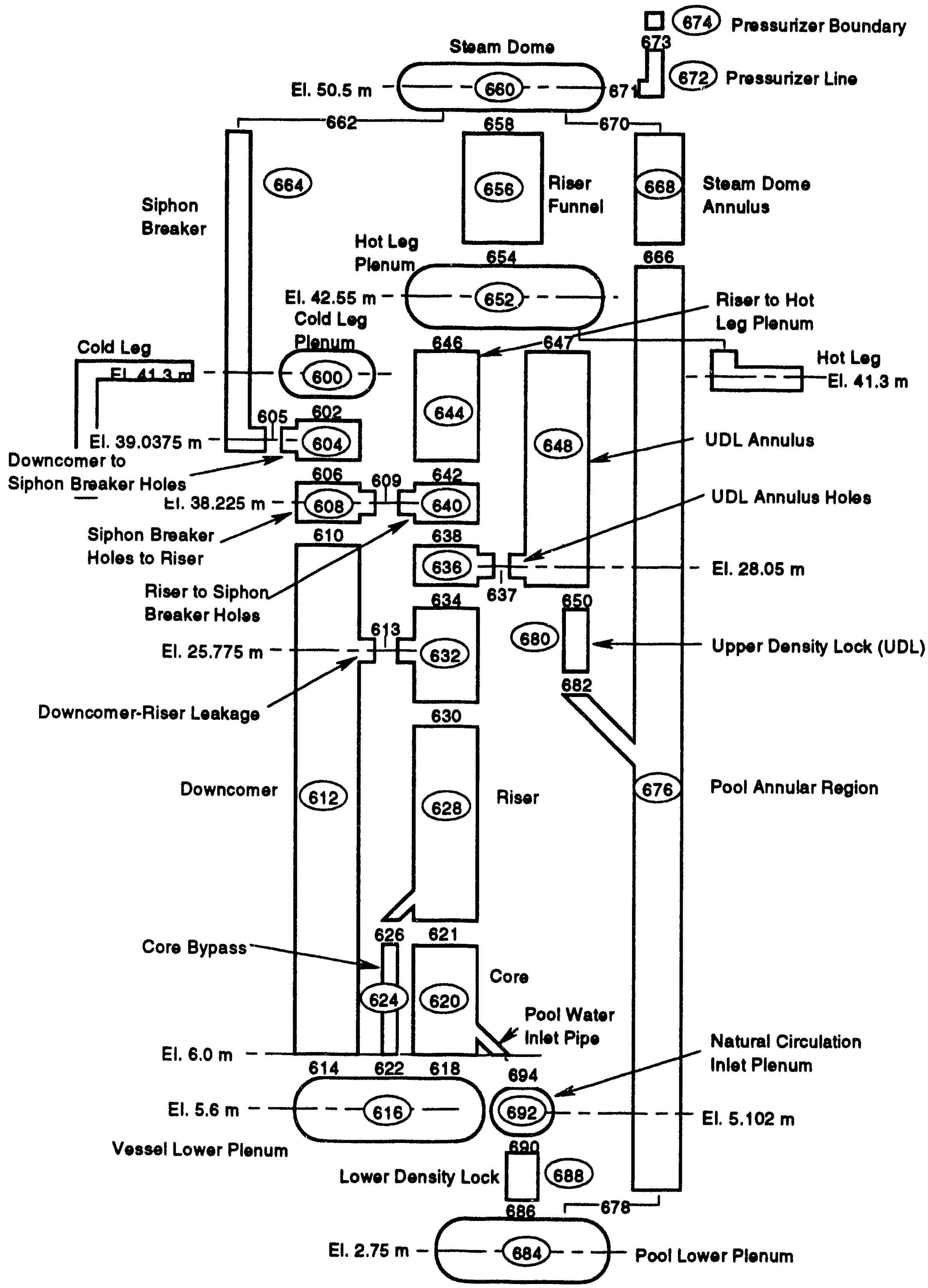

Fig. 2. TRAC 1-D Model of PIUS vessel and pool.

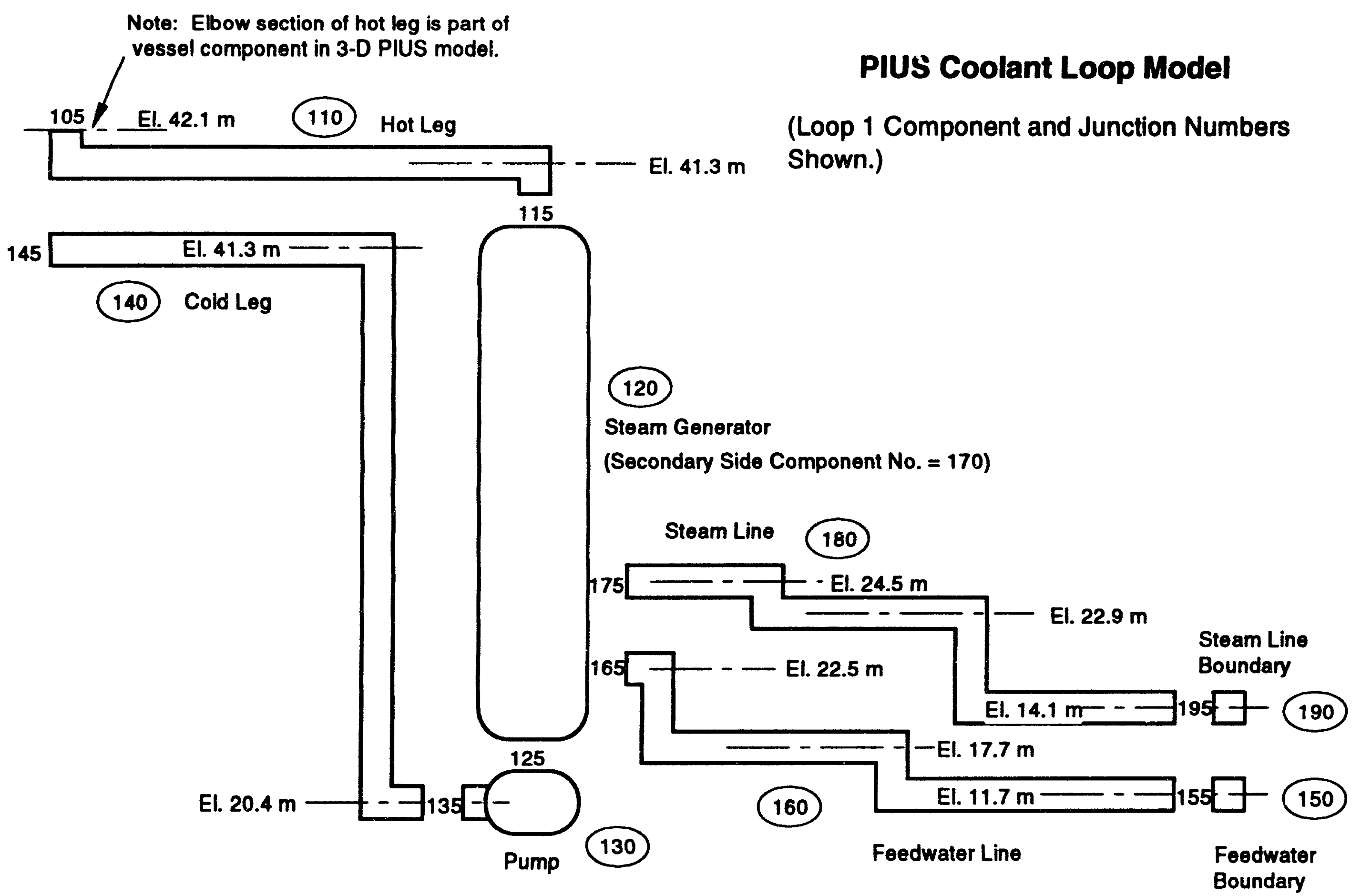

Fig. 3. TRAC model of PIUS reactor-coolant loop.

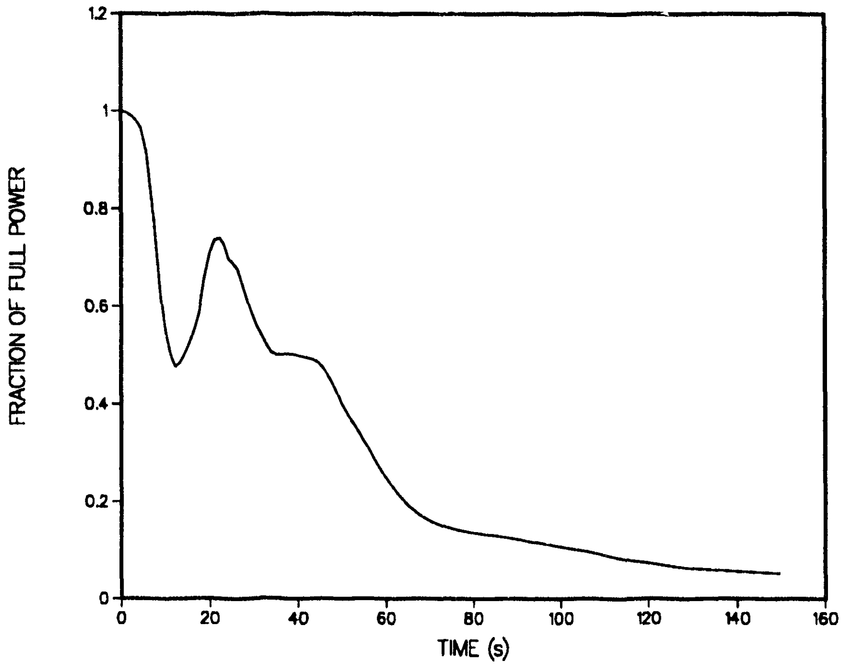

Fig. 4. Reactor power,
pump-trip SCRAM transient with four-loop 1-D model.

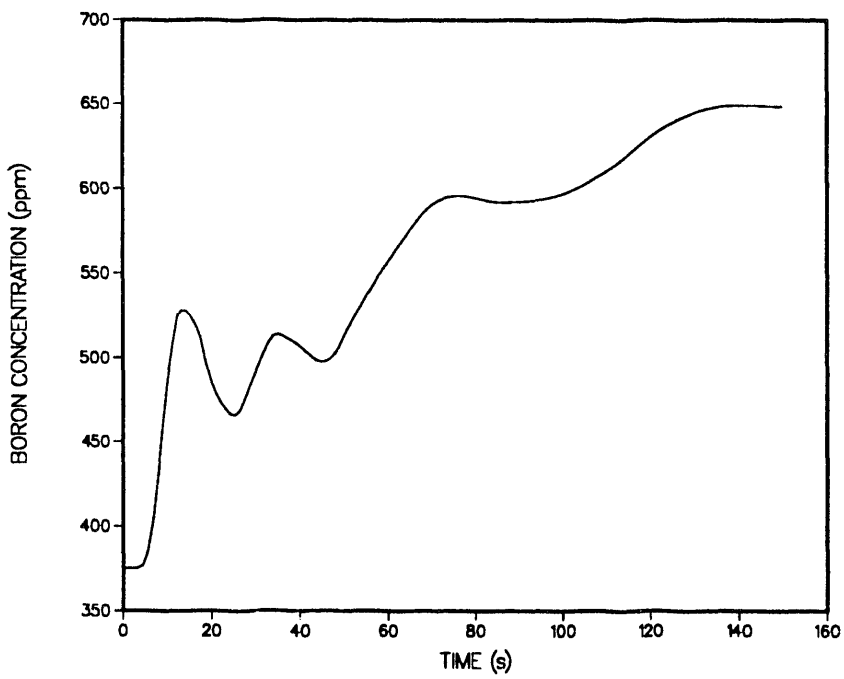

Fig. 5. Core inlet boron concentration,
pump-trip SCRAM transient with four-loop 1-D model.

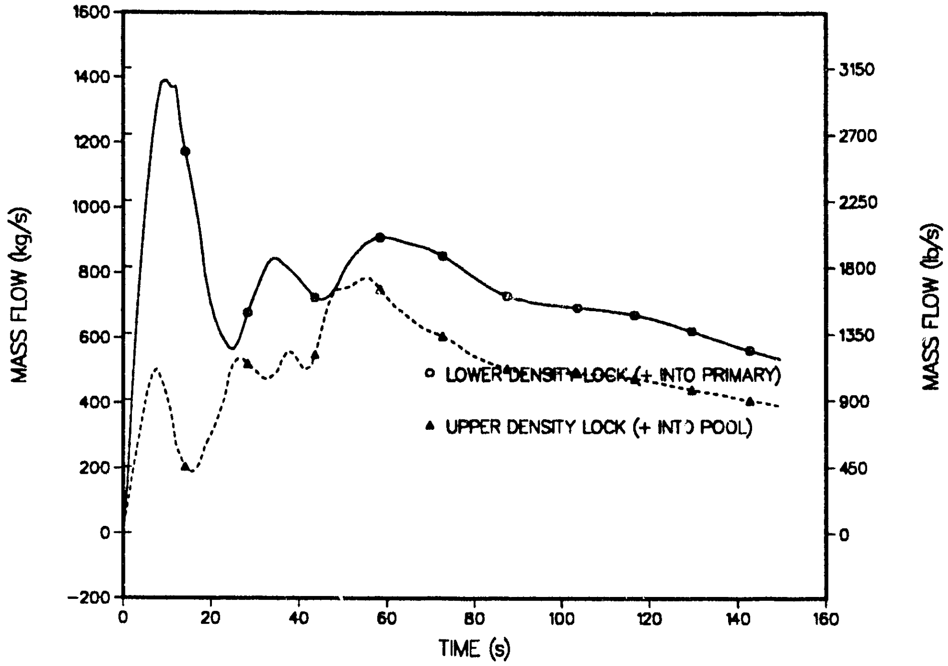

Fig. 6. Upper and lower density lock mass flows,
pump-trip SCRAM transient with four-loop 1-D model.

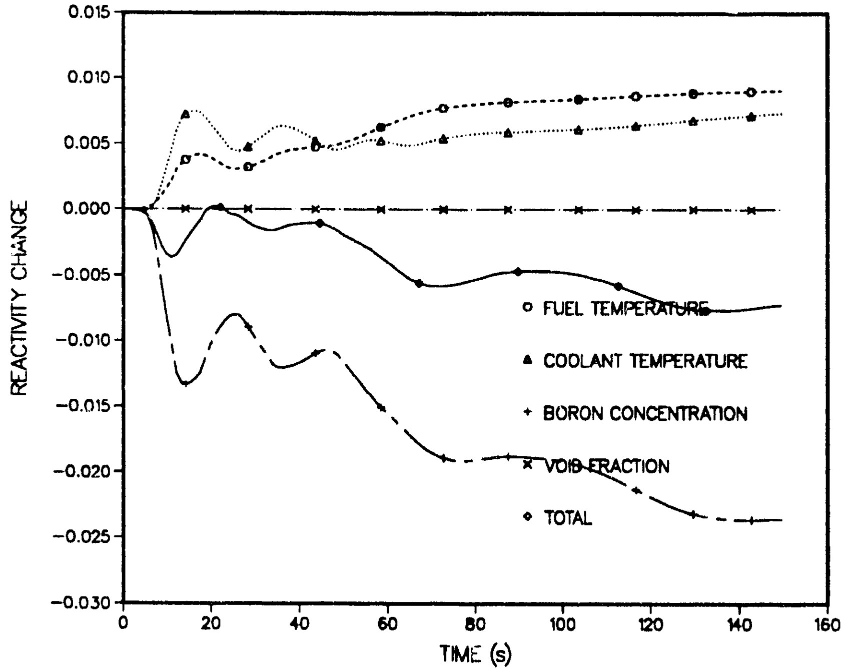

Fig. 7. Reactivity changes,
pump-trip SCRAM transient with four-loop 1-D model.

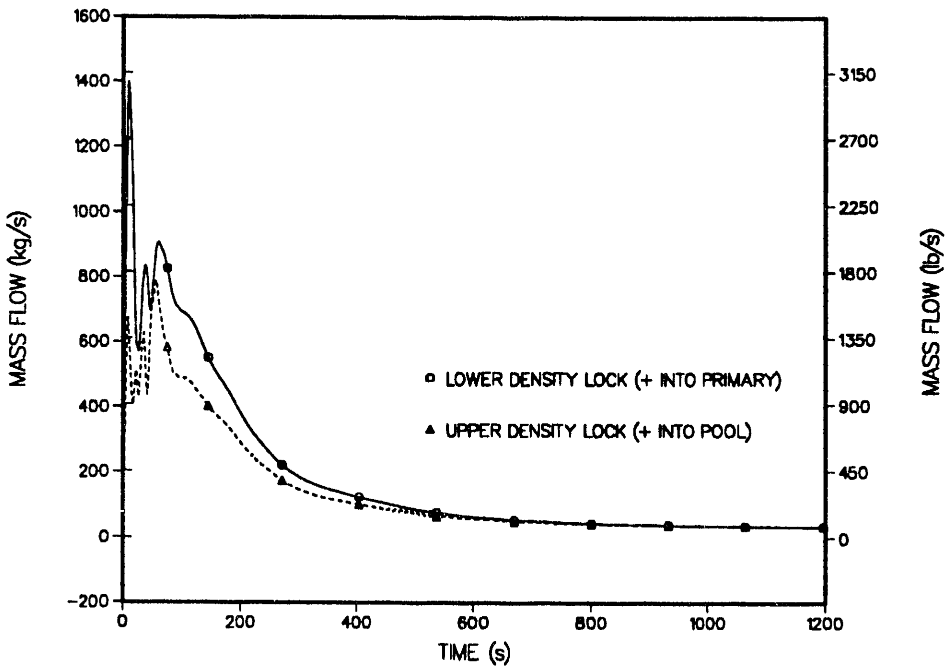

Fig. 8. Upper and lower density lock mass flows from 0 to 1200 s, pump-trip SCRAM transient with four-loop 1-D model.

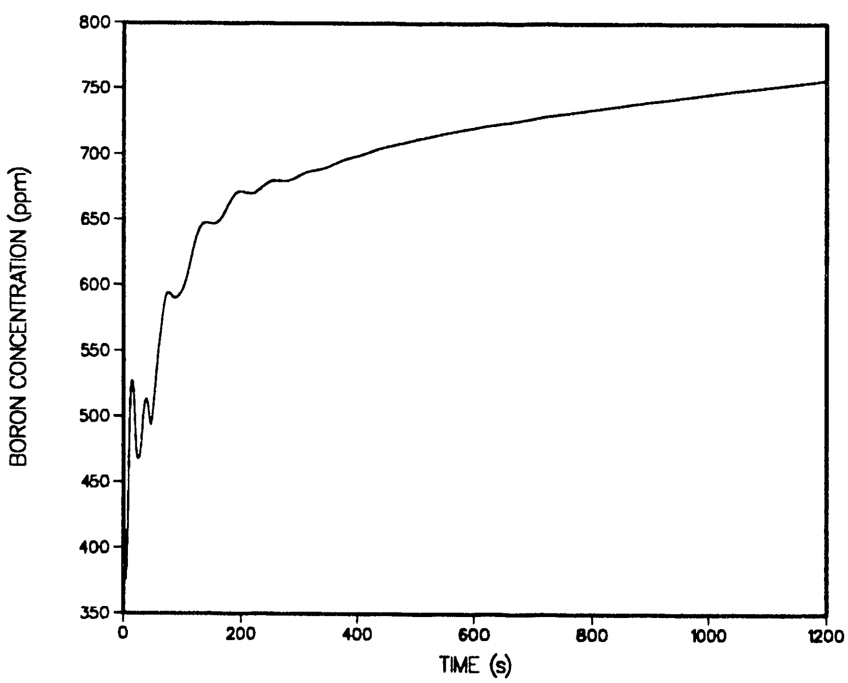

Fig. 9. Core inlet boron concentration from 0 to 1200 s, pump-trip SCRAM transient with four-loop 1-D model.

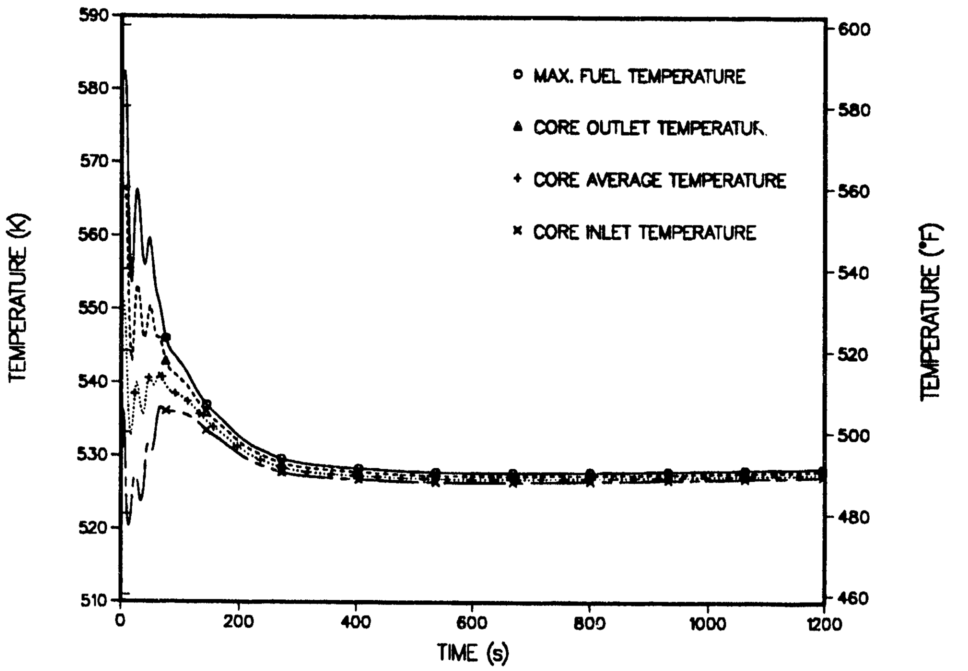

Fig. 10. Core coolant and fuel temperatures, pump-trip SCRAM transient with four-loop 1-D model.

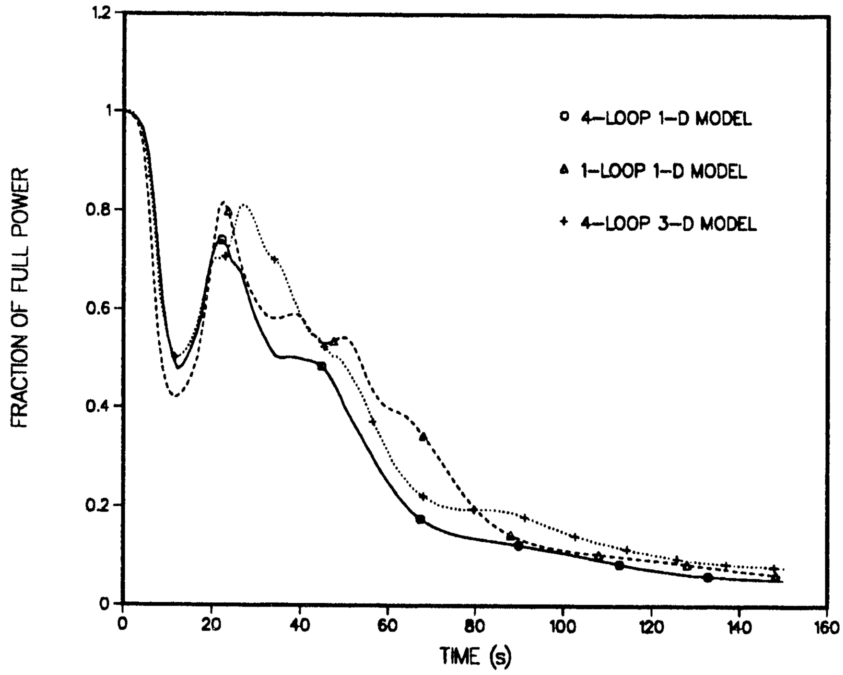

Fig. 11. Comparision of the calculated reactor power for a pump-trip SCRAM transient for different TRAC models of the PIUS reactor.

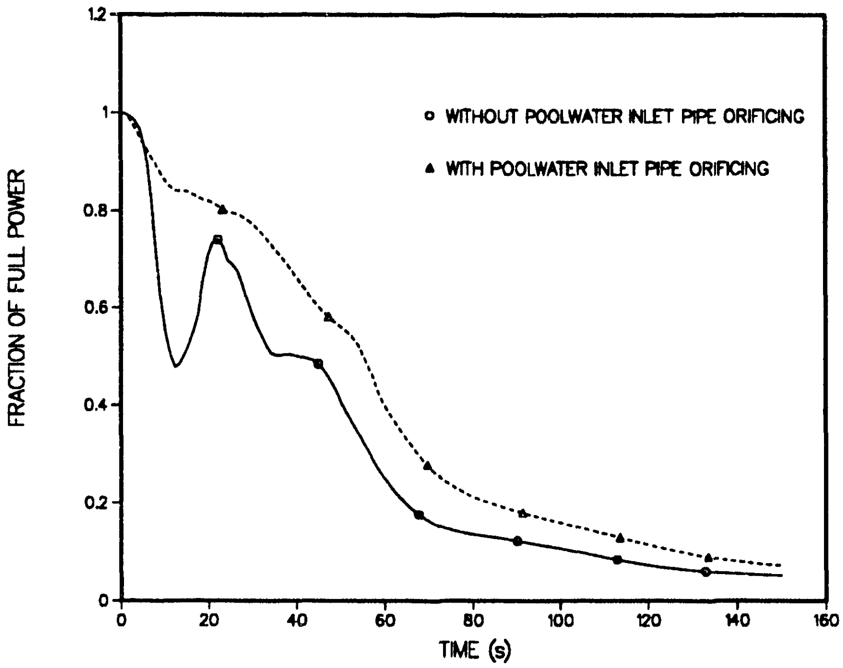

Fig. 12. Reactor power for a pump-trip SCRAM without and with poolwater inlet pipe orificing.

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