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7.1Thermal Hydraulics under Transient Conditions

Validation of a RELAP5 Computer Model For a VVER-1000 Nuclear Power Plant *

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ABSTRACT

This paper describes a computer model that has been developed for a VVER-1000 nuclear power plant for use with the RELAP5/MOD3.1.1 computer code in the analyses of operational occurrences, abnormal events, and design basis scenarios. This model will provide a significant analytical capability for the Bulgarian nuclear regulatory body (Committee on the Use of Atomic Energy For Peaceful Purposes) and the Bulgarian technical specialists located at the power plant site (Kozloduy Nuclear Power Plant). In addition, the initial validation of computer model has been completed and is described in the paper. The analytical results are compared with data obtained during planned testing at the power plant; the test performed was the trip of a single main coolant pump. In addition,

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the paper provides a discussion of various other RELAP5 parameters calculated for the main coolant pump trip scenario. This model development and validation analysis represents an important accomplishment in the analyses of Russian designed nuclear power plants with computer codes developed and used in Western countries. The results indicate that RELAP5 can predict the thermal-hydraulic behavior of the VVER-1000 reactor for the class of transients represented by test results.

INTRODUCTION

The reference power plant for this analysis is Unit 6 at the Kozloduy site. This plant is a VVER-1000, Model V320, pressurized water reactor that produces 3000 MW thermal power and generates 1000 MW electric. Bulgaria has two VVER-1000's located at Kozloduy. The plant design has four primary coolant loops, each including a main circulation pump and a horizontal steam generator. The horizontal steam generator represents an important difference between VVER designs and Western reactor designs; the transient response of horizontal steam generators can be very different than that of Western type vertical steam generators due to the larger water mass in horizontal steam generators. This larger water mass can affect the reactor transient response particularly during secondary side occurrences.

The thermal-hydraulic computer code, RELAP5/MOD3.1.1 was used in the calculation of the experiment. RELAP5/MOD3.1.1 has been used extensively in the Western countries for the analyses of operational occurrences, anticipated transients, and abnormal events. In addition, a comprehensive validation program has been completed with RELAP5 by the U.S. Nuclear Regulatory Commission.

DISCLAIMER

REACTOR MODEL DESCRIPTION

In the VVER-1000 primary system, coolant enters into the reactor vessel through the four inlet pipes associated with the four primary loops. The flow then passes into the downcomer between the reactor vessel and the inner vessel. The flow enters the lower plenum of the reactor vessel and passes through orifices in the inner vessel and then enters slots in the fuel support structures that lead directly to the fuel assemblies. The flow passes through the open bundles of the core. The pressure drop in the core is approximately 1.8 atm at rated flow conditions. The fuel assemblies are in the configuration of a hexagon with each containing 317 fuel rods. There are 163 fuel assemblies of which 61 have control rods. After the exiting the reactor core, the flow moves into the upper plenum, which contains the shielding block, and then out to the hot legs of each of the four primary loops in the system.

In the RELAP5 model (shown schematically in Figures 1 and 2), the primary system has been modeled using two coolant loops with one loop representing three reactor loops and the second loop representing the reactor loop with the tripped pump. Overall, the RELAP5 model includes 120 control volumes, 187 flow junctions, and 256 heat structures; this model configuration provides a detailed representation of the primary, secondary, and safety systems. The following models are included:

1. Reactor vessel including a downcomer, lower plenum, and outlet plenum.
2. Core region represented by a hot and average heated flow paths. There is also a bypass channel. Each core channels has ten axial nodes. The heated hot and average channels have eight heated axial nodes plus one non- heated inlet and outlet node.
3. Pressurizer system with heaters, spray, and pressurizer relief capability.
4. Safety system representation including the accumulators, high and low pressure

injection systems, and the reactor scram system.

5. Makeup and blowdown system including the associated control systems.

In each primary loop, the steam generator tubing is modeled by 3 horizontal channels, each divided into 5 axial volumes. The vertical hot and cold flow distribution headers are each modeled with 4 nodes. The steam volumes above the heat exchanger tubes are modeled with three nodes. For each steam generator, the upper steam volume is modeled as a steam separator. The tube area secondary side is modeled with three nodes. The steam generator secondary side downcomers are represented with three separate components to provide modeling for internal steam generator natural circulation.

The initial conditions for the RELAP5 model have been selected to reflect the initial conditions of the power plant at the start of the test. The data is summarized in Table 1.

Table 1: Initial Conditions for the RELAP5 Model and the Experiment

Parameter	Plant Condition	RELAP5 Model
Power Level, MWt	2250.	2250.
Core Flow Rate, kg/s	17696.	17696.
Core Inlet Temperature, C	284.4	284.4
Core Outlet Temperature, C	307.5	307.5
Upper Plenum Pressure, MPa	15.58	15.58
Pressurizer Level, cm	830.	830.
Steam Header Pressure, MPa	5.84	5.84
Steam Generator Level, cm	230.	230.
Feedwater Flow Rate, kg/s	306.	306.

Data and information for the modeling of these systems and components were obtained from Kozloduy documentation [3] and from the power plant staff.

RELAP5 COMPUTER CODE

The RELAP5/MOD3.1.1 computer code [1] has been used to model the VVER-1000 power plant and perform the analysis described in this paper. RELAP5 has been used for safety analyses and the analysis of power plant operational performance for reactors of Western design. The hydrodynamic model in the code is a one- dimensional, two fluid model for the flow of a two phase steam/water mixture. The basic field equations include the continuity, momentum, and energy equations solved for each of the two phases. Heat transfer is modeled with a one-dimensional heat conduction equation that is coupled with the hydrodynamic calculation through a number of surface heat transfer regimes. Process models are available for choked flow, branching, and the representation of pipe ruptures. Component models can be used to represent pumps, valves, and accumulators. The extensive previous experience and the code flexibility for the representation of complex systems make RELAP5 well suited for the analysis of VVER-1000 power plants.

DESCRIPTION OF EXPERIMENT

The experimental investigation is part of "Program for Dynamical Research of VVER Reactors During Switch Off of Main Circulation Pumps" [8] . This was an experiment, performed jointly by Bulgarian and Russian specialists at the Kozloduy Nuclear Power Plant (Unit 6), where one of the four main circulation pumps was tripped and the power level was reduced from 75% [2250 MW] to 50% [1500 MW]. The purpose of the experiment was to explicitly establish the response of the plant to a single pump trip and to provide a data base for bench marking thermal hydraulic codes.

During the transient, the plant staff did not interact with the operation of the automatic control systems. The response of the primary and secondary side control systems did not reach the reactor scram setpoints. Experimenters indicated that the steam dump to condenser facility (BRU- K), steam dump to atmosphere (BRU-A), and spray systems from the cold leg piping (lines Du 180) did not activate.

In base - load mode of NPP unit operation the Reactor Power Controller (RPC) operates in "T" mode (secondary circuit pressure stabilization). During the experimental transient, a signal from Reactor Power Limitation Controller (RPLC) generates a Warning Protection 1 (WP-1) signal, the RPC automatically switches to "H" mode of operation (neutron power stabilization), and the RPC is disconnected from operating the control rods (CR) and drives. Rod Bank #10 inserts from position 78% to 43% of the core height in 59 s at the normal operational speed of 2 cm/s. WP-1 is a type of emergency action of the control rods: downward movement of the control rods bank by bank, starting with the "control bank", normally Rod Bank #10. When the initiating signal is cleared, rod movement stops.

Changes of the RPC modes of operation automatically lead to corresponding changes in the Electro-Hydraulic Turbine Controller (EHTC) mode and reduces turbine power corresponding to the reduction in the thermal power of reactor.

When the WP-1 signal is cleared, the RPC continues to work in "H" mode and maintains the neutron power level reached at that time (51.8% power) and switches to controlling the control rods. Thermal power stabilized at the 50% level.

A goal of the transient was to reach a new plant steady state at 50% power by means of pressurizer heaters, feedwater system, make up system and pressurizer relief valves. Reactor power was reduced from 75% to 50% during the transient without the need to initiate a scram. The model development and validation has focused on the applicability of RELAP5 to this type of transient.

The following sequence of events describes the scenario that was followed during the experimental transient and that has been used in the RELAP5 calculations:

+	0 s	Plant stabilized at 75% power [2250MW]
+	0 s	a. Trip one (Loop #3) of four main circulation pumps. b. Power reduction from 75% to 50% power initiated. c. Reduction in feedwater flow to the steam generator initiated.
+	28 s	a. Steam Generator (#3) level dropped 100 mm. b. Auxiliary pumps #1 and #2 automatically initiated to supply feedwater to steam generator.
+	29 s	Pressurizer Heater #1 (Section 1,2, and 3)[1] and Heater

1 Each pressurizer heater section generates 90 kw.

#2 (Section 3) automatically turned off. Pressure reached 15.84 MPa.

+	37 s	a. Pressurizer relief valve (YP13S02) opens at 15.8 MPa b. Pressurizer fine spray (from makeup system) initiated.
+	40 s	a. Pressurizer relief valve (YP13S02) closed at 15.7 MPa. b. Pressurizer fine spray terminates.
+	43 s	a. Pressurizer Heater 1 (Section 2), Heater 2 (Section 1,2, and 3), Heater 3 (Section 1 and 2), and Heater 4 (Section 1,2,3, and 4) initiated due to rate of pressure decrease.
+	60 s	Power reached 50% rated condition [1500MW].
+	98 s	Pressurizer Heater 1 (Section 1 and 3) initiated, system pressure dropped below 15.36 MPa.
+	137 s	Pressurizer Heater 3 (Section 1, and 2) turned off; system pressure reached 15.5 MPa.
+	139 s	Pressurizer Heater 4 (Section 1,2,3, and 4) turned off; system pressure reached 15.51 MPa.

RESULTS

The transient considered in this report can be categorized as of class of transients resulting from power plant equipment failure and perturbing the coolant flow rate through the reactor core. In general, the reason for main coolant pump (MCP) failure could be electrical or mechanical (e.g., loss of electrical power or stuck rotor).The experiment and the RELAP5 analysis have assumed that the MCP failure was due to the loss of electrical power.

The transient calculations are compared with the experimental data in Figures 3- 11.An important parameter is the pressure in the primary circuit, since this parameter is input to many reactor control systems. Figure 3 presents measured primary pressure during the experiment and calculated one. As shown, the calculated parameter is almost identical to measured one. Maximum pressure 15.84 MPa was reached at 37 s during the experiment and at 35 s in the RELAP5. calculation. Corresponding to this pressure, in both cases spray from the makeup system was initiated for approximately for 3 seconds; primary pressure then began to reduce, causing initiation of pressurizer heaters to maintain pressure. In this time makeup injection reached the maximum rate and continued to support pressure in primary side. Minimum pressure in the primary side (15.3 MPa) was reached after 90 s. Pressurizer Heater 3 (Sections 1, and 2) and Pressurizer Heater 4 (Sections 1,2,3, and 4) switch off at 137 and 138 sec in RELAP5 calculation respectively by reaching the pressure trip setpoint (see scenario sequence of events).

Another important characteristic is the coolant temperature in cold and hot legs. As it is seen from Figures 4 and 5 , the calculation closely follows the results obtained from the experiment. Temperature of the hot, tripped loop was initially 307.5 C and reduced to 275.4 C, below the temperature of the operating cold legs; this indicates reverse flow in the tripped loop. The RELAP5 calculation indicated that reverse flow was initiated at 34 s.

Other very important characteristics are water level in the pressurizer and steam generator secondary side; these results are compared with the plant data in Figures 6 and 7. These figures indicate good agreement between the plant data and the RELAP5 calculation. The main coolant pump rundown curve is also an important parameter for the quality of the model predictions. The results from the experiment and RELAP5 are compared on Figure 8.

Figures 9 through 11 provide comparisons of the following parameters:

Figure 9: Secondary side main steam header pressure

Figure 10: Main coolant pump pressure difference in tripped loop

Figure 11: Reactor vessel pressure difference.

In each case, the comparisons indicate good agreement between the RELAP5 results and the experimental data.

CONCLUSIONS

The RELAP5 model developed for the transient analysis of the performance of VVER-1000 nuclear power plants has been used to accurately predict the results obtained during a loss of main coolant pump test performed at the Kozloduy NPP (Unit 6). These results are an important part of the validation of the RELAP5 model developed for Kozloduy NPP. Integrated test results obtained from actual power plant testing provide important data for analytical model validation. Experimental data frequently is obtained from scaled down models of the actual power plant; the scaling can increase the uncertainty of the data relative to the reactor performance. The test results provided in this paper provide an integrated evaluation of the complete RELAP5 VVER-1000 model. The comparisons indicate that RELAP5 predicts the test results very well.

References

[1.] Carlson, K.E., at al., "RELAP5/MOD3 Code Manual," VOL.1,2,3 and 4 Draft, NUREG/CR-5535 (1990)

[2.] International AtomicEnergy Agency, Guidelines For Accident Analysis For WWER Nuclear Power Plants, WWER-SC-094, Vienna (1995)

[3.] Design Data For The Kozloduy Units 5&6 Plant Analyzer Project, BNL-USA (1995)

[4.] Overall Plant design Descriptions VVER, Water-Cooled, Water-Moderated Energy Reactor, DOE/NE-0084, Revision 1, (1987)

[5.] Ustanovka Reactornaia V-320. Poiasnitelnaia Zapiska. Opisanie Proektnyh Rejimov 320.00.00.00.000P31, GKAE. OKB, Gidropress (1971)

[6.] TOB Peti Blok, AEC Kozloduy, Bulgaria

[7.] Development of a VVER-1000 Model Using RELAP5/MOD3.1.1 For Kozloduy Unit 6 Nuclear Power Plant, BNL-USA, Draft (1995)

[8.] AKT No_ 06.NPZ.00.1246.EP Kozloduy Nuclear Power Plant Unit 6, Program for Dynamical Research of VVER Reactors During Switch Off of Main Coolant Pumps, November 9, 1991.

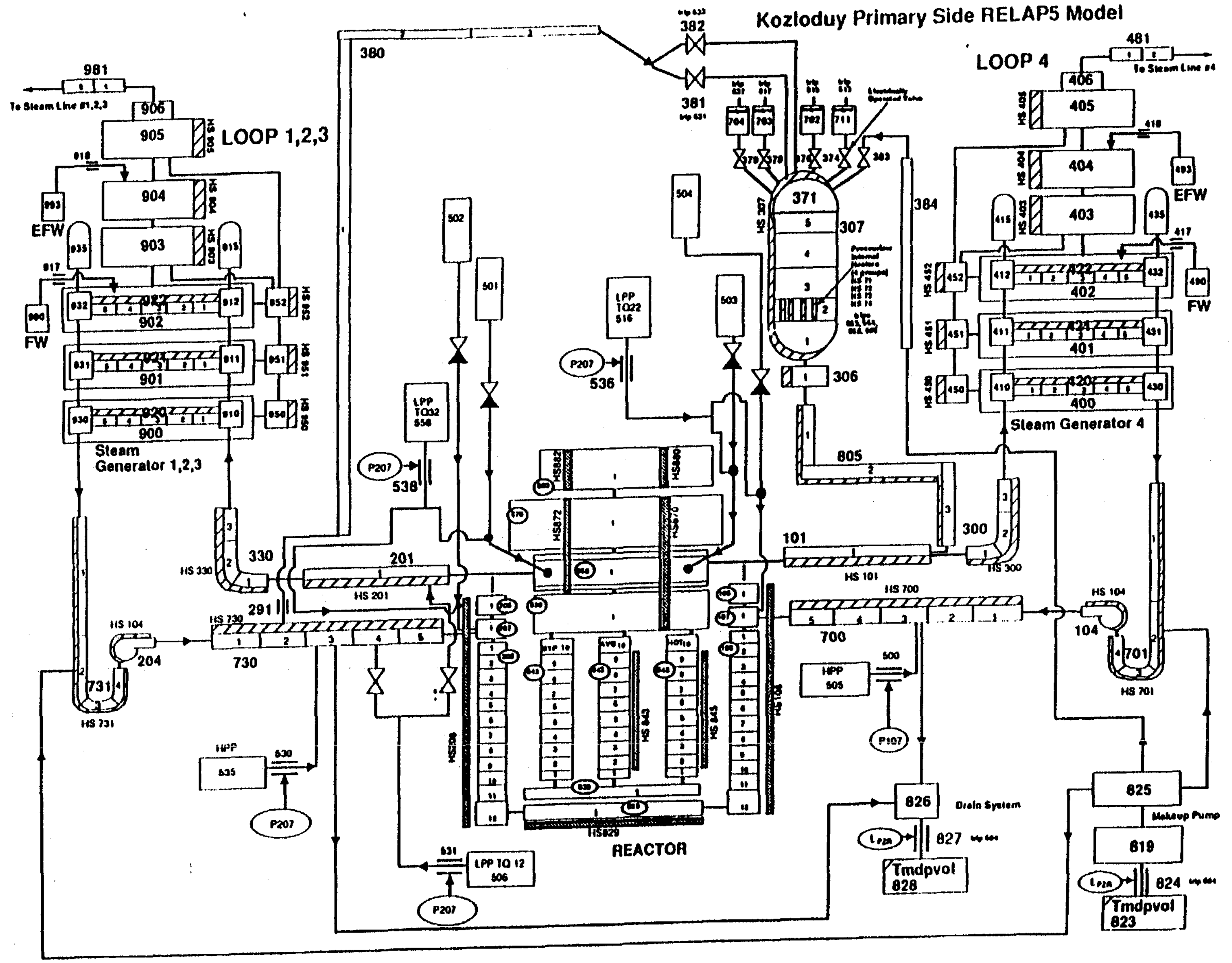

Figure 1: RELAP5 Nodalization for Primary System

Kozloduy Steam Line
RELAP5 Schematic

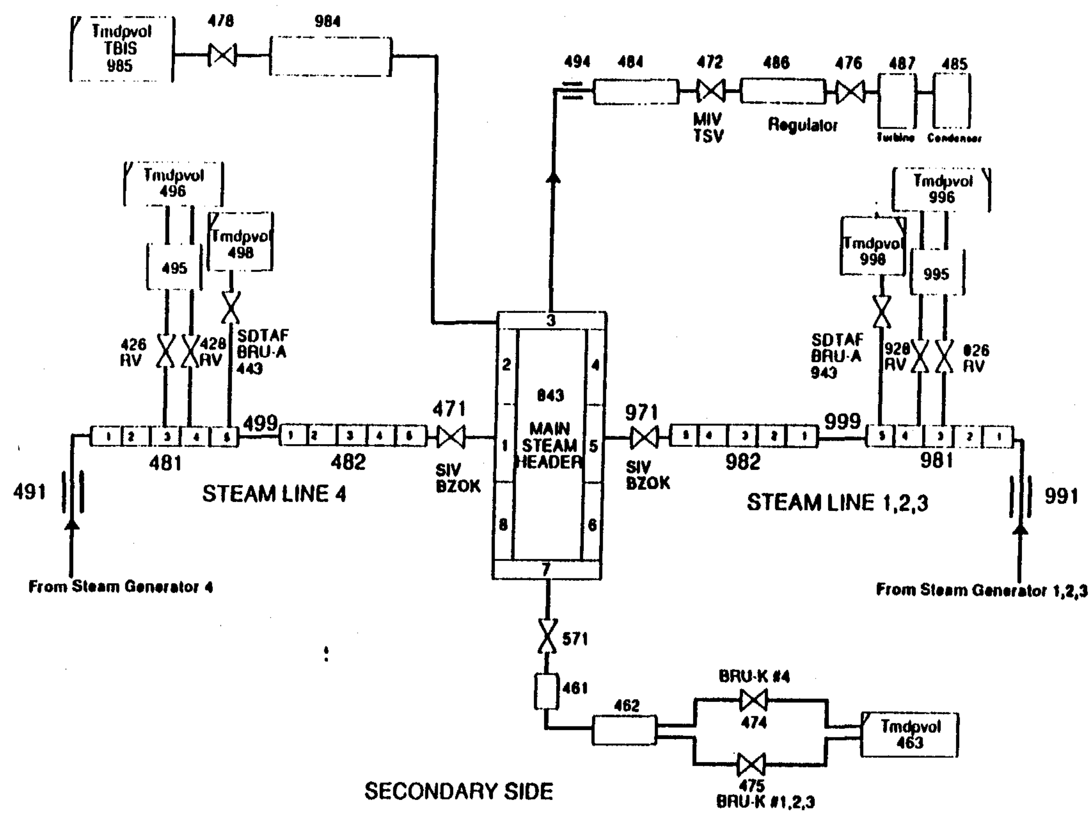

Figure 2: RELAP5 Nodalization for Secondary System

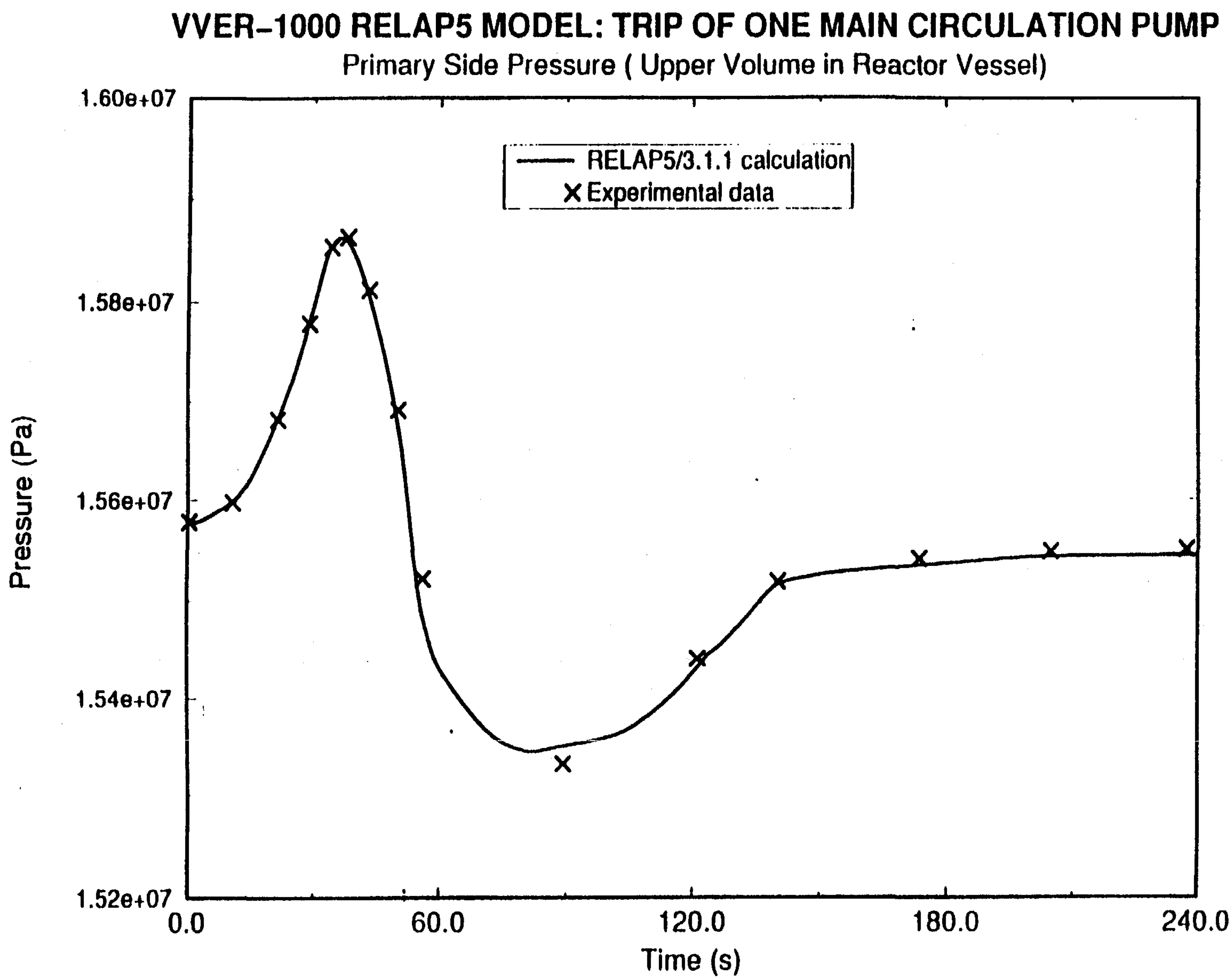

Figure 3: Comparison of Primary System Pressure

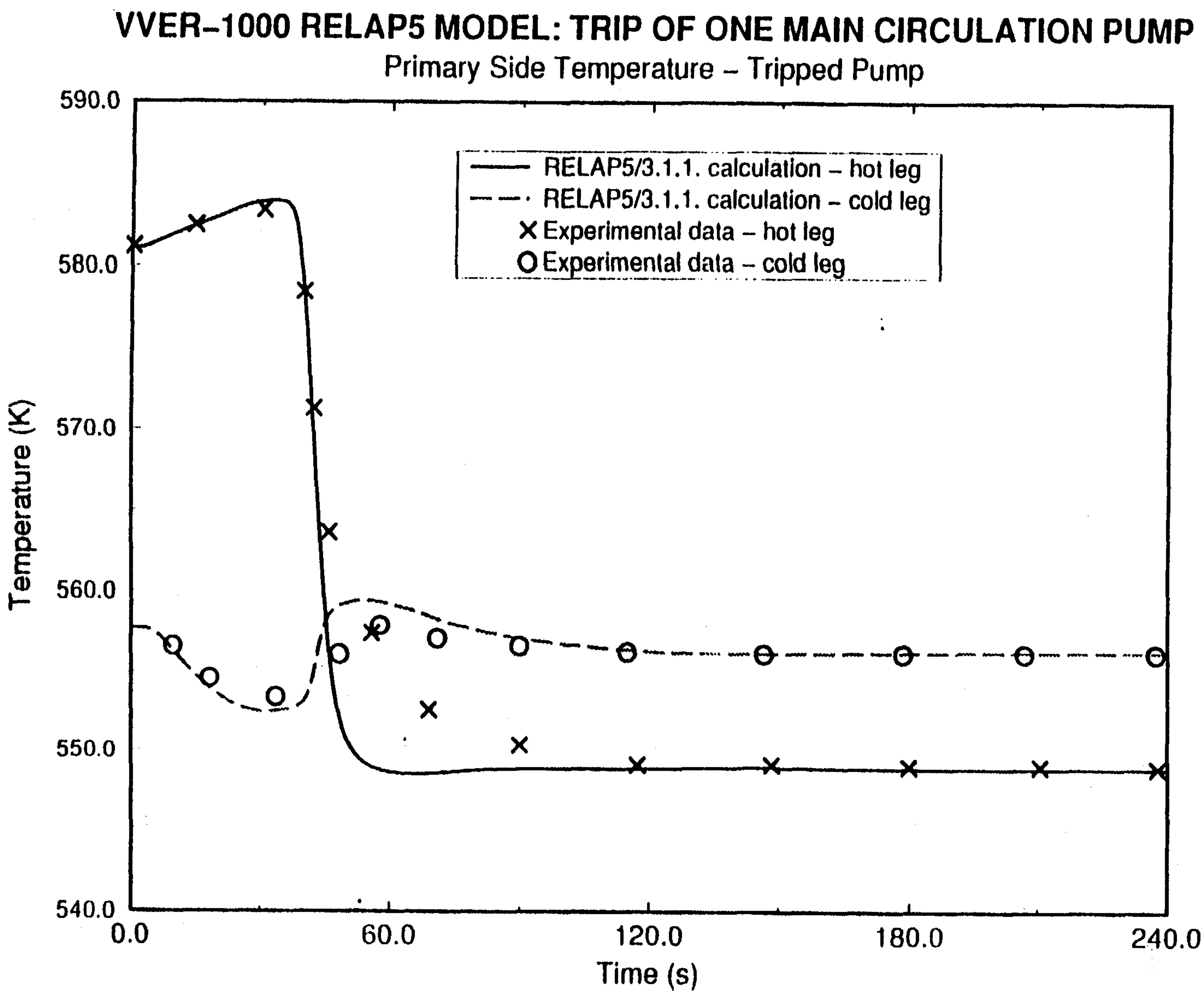

Figure 4: Comparison of Hot and Cold Leg Temperatures: Tripped Loop

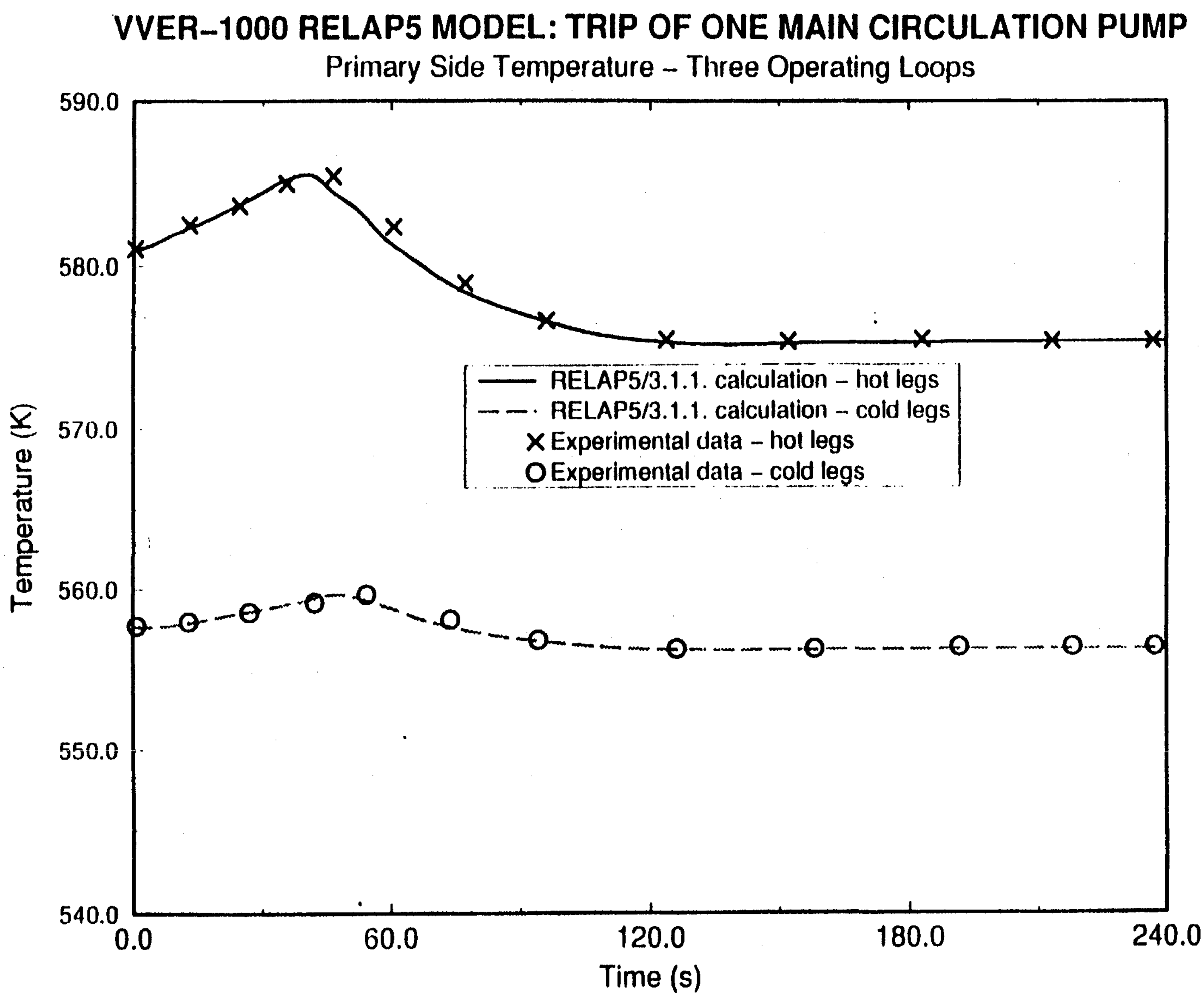

Figure 5: Comparison of Hot and Cold Leg Temperatures: Operating Loops

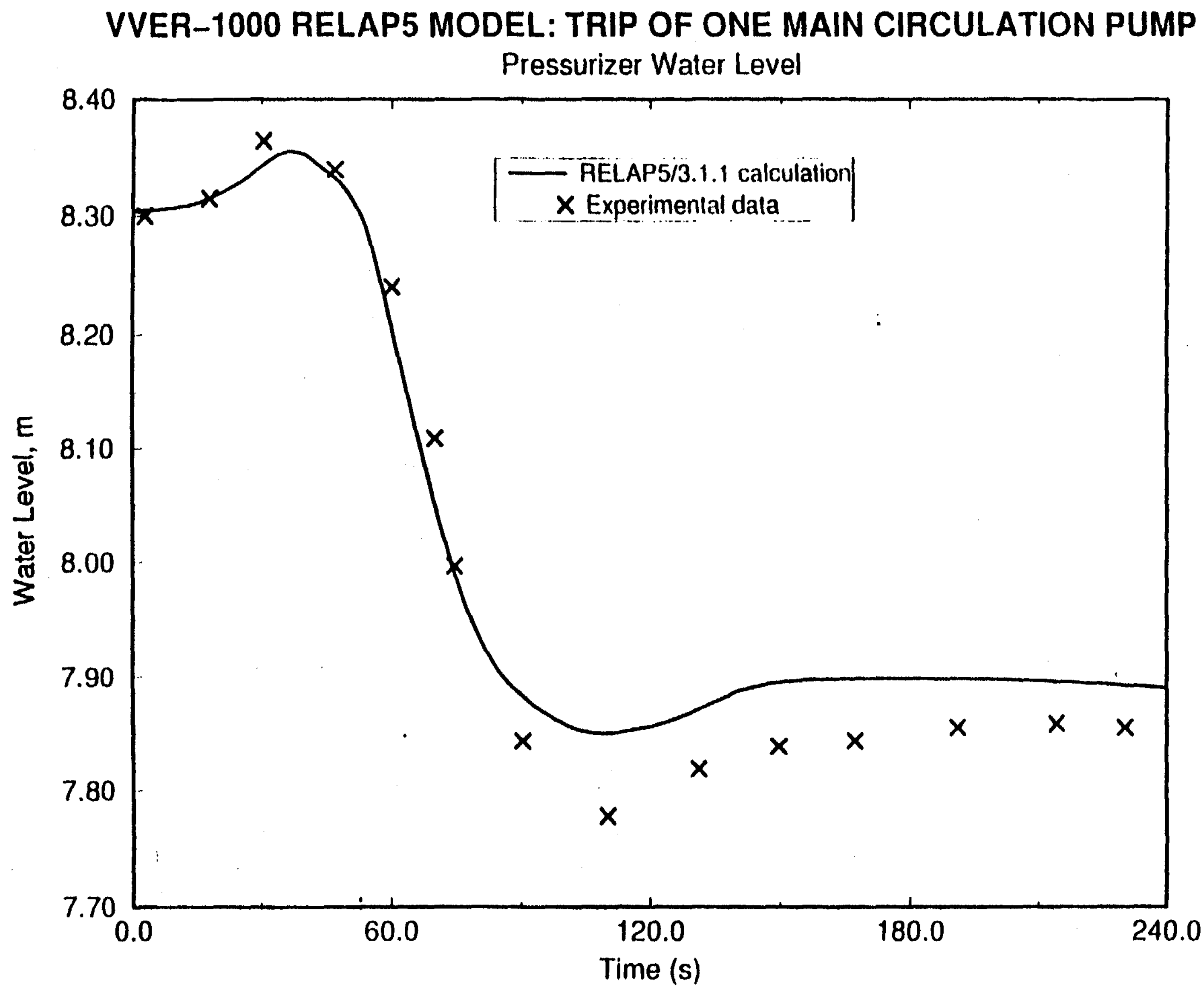

Figure 6: Comparison of Pressurizer Water Level

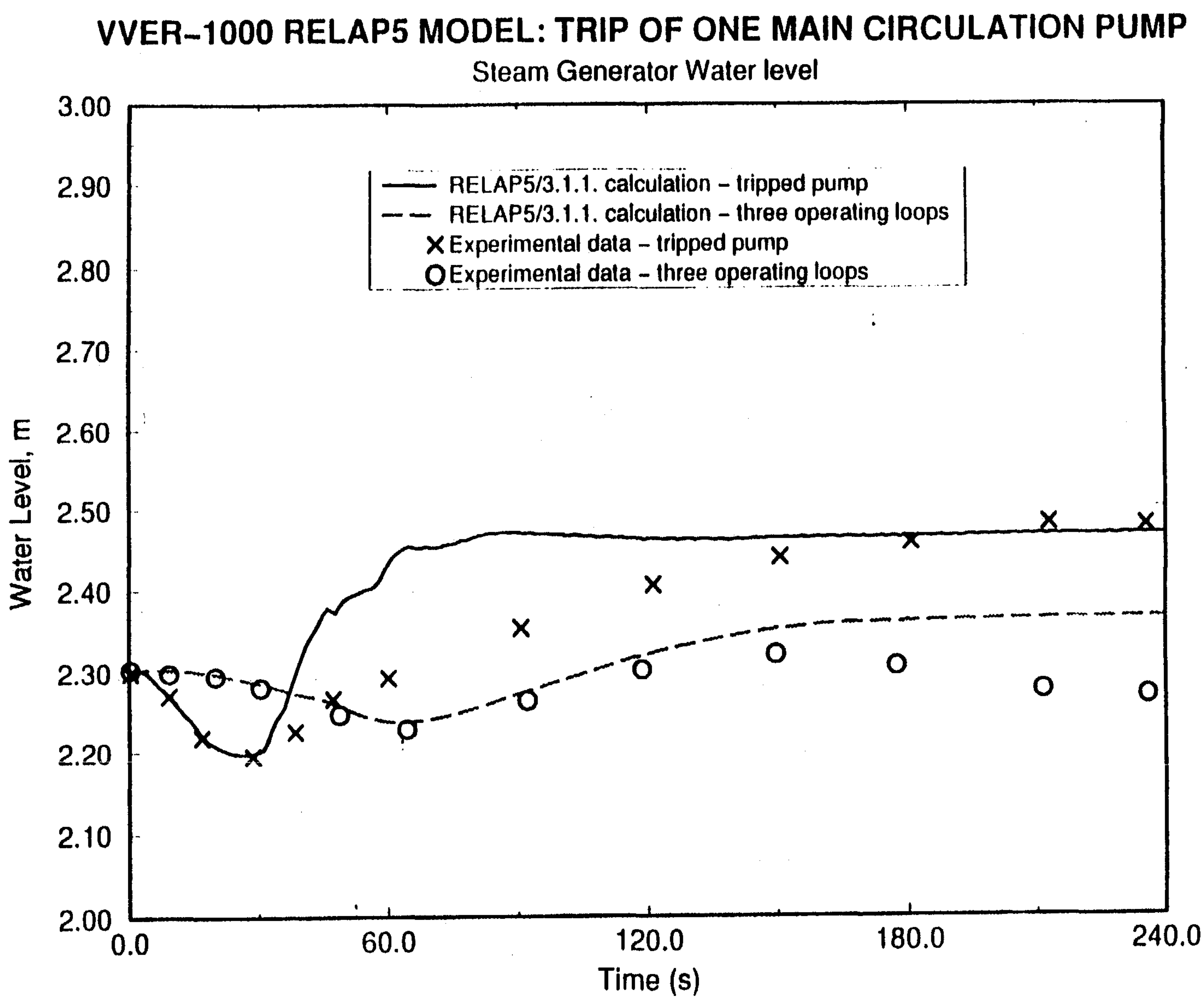

Figure 7: Comparison of Steam Generator Water Level

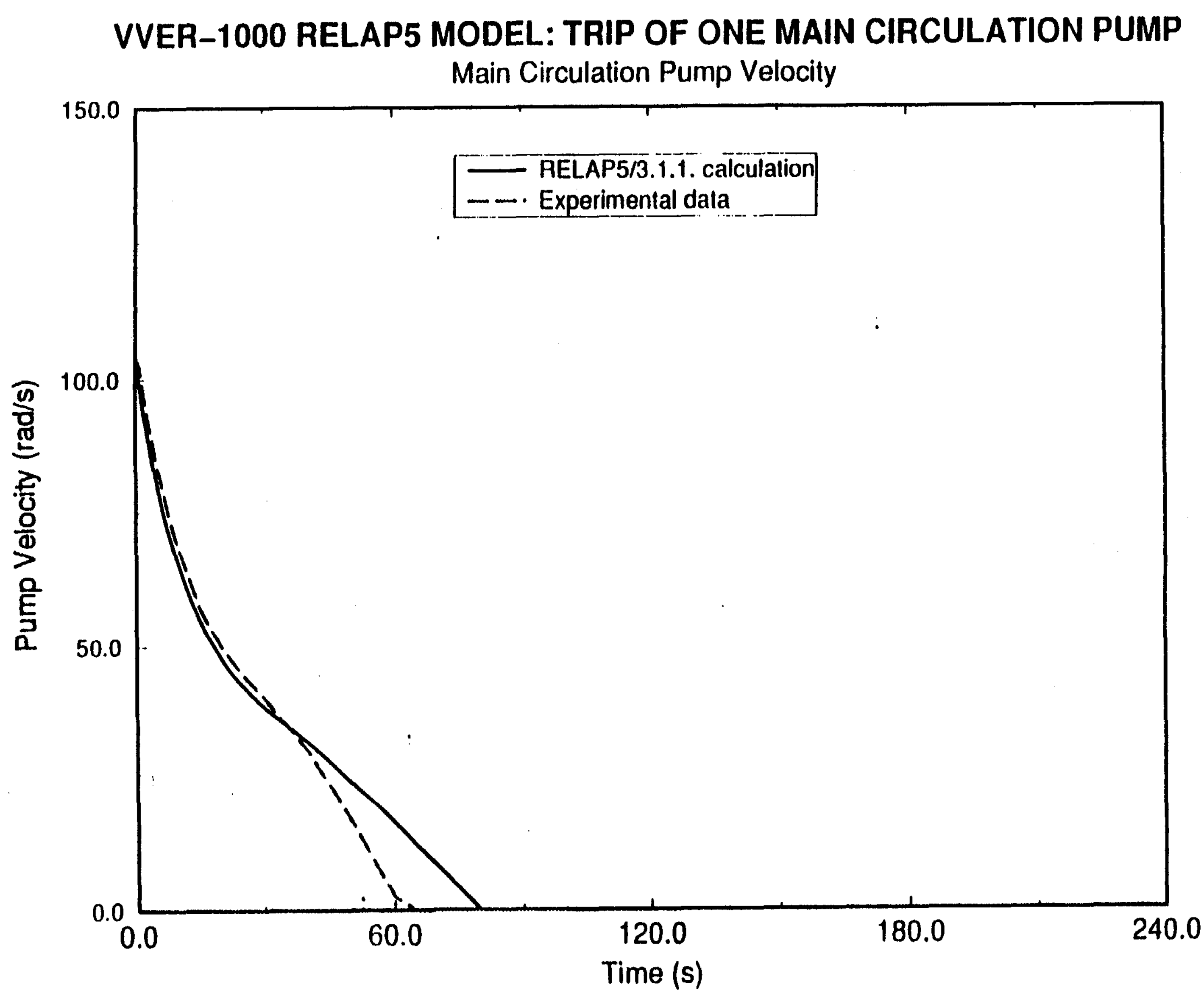

Figure 8: Comparison of Main Coolant Pump Velocity: Tripped Loop

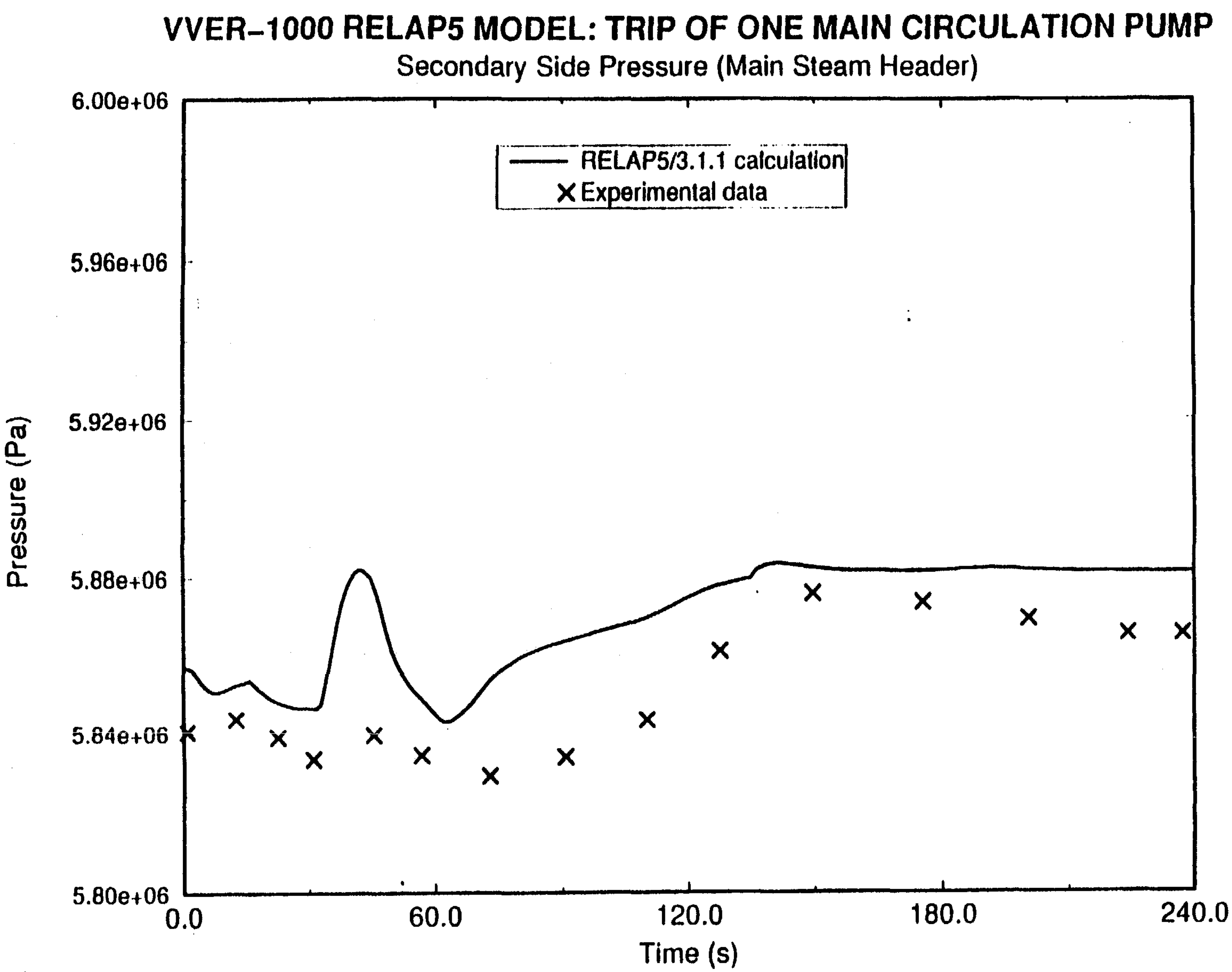

Figure 9: Comparison of Main Steam Header Pressure

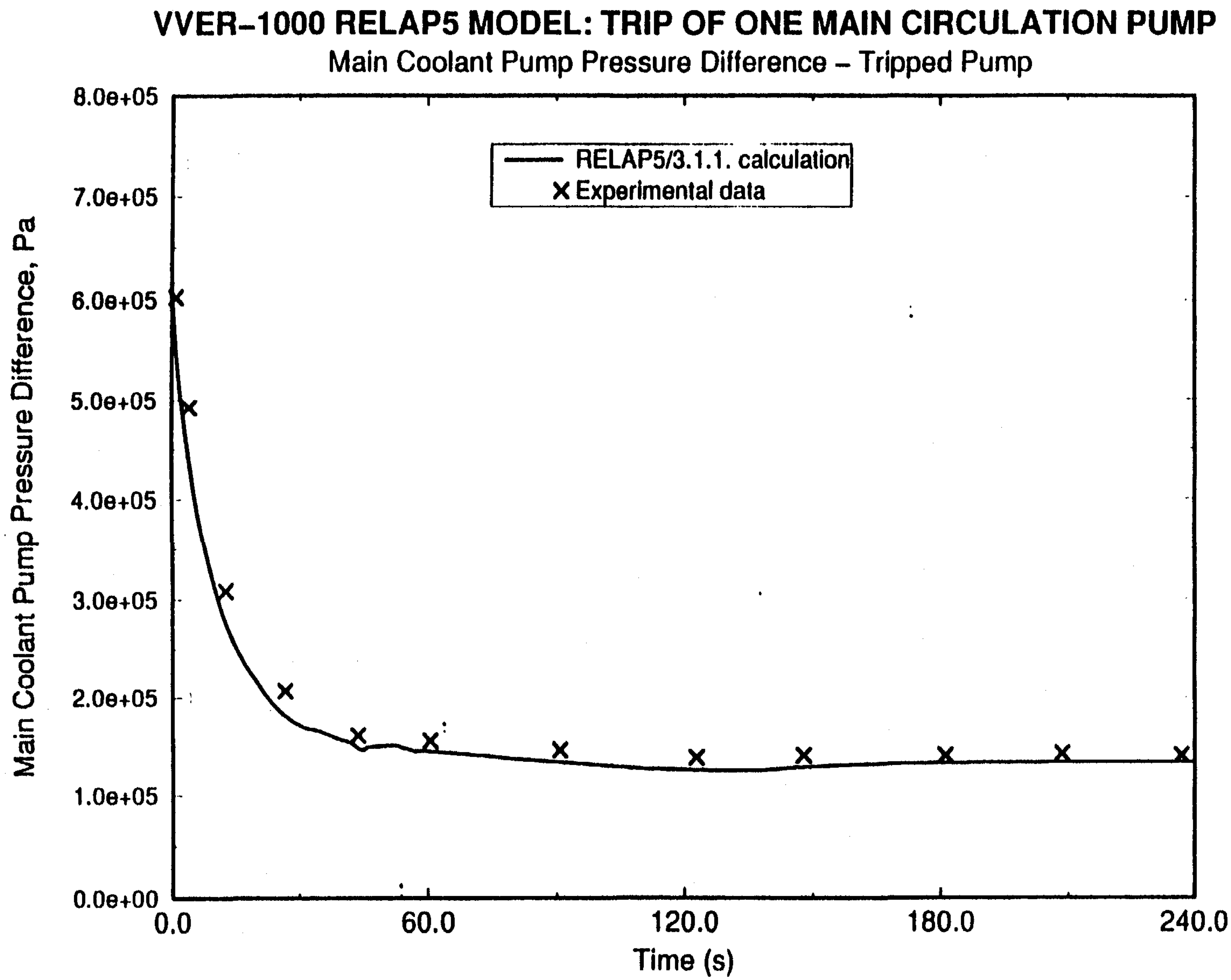

Figure 10: Comparison of Main Coolant Pump Pressure Difference: Tripped Loop

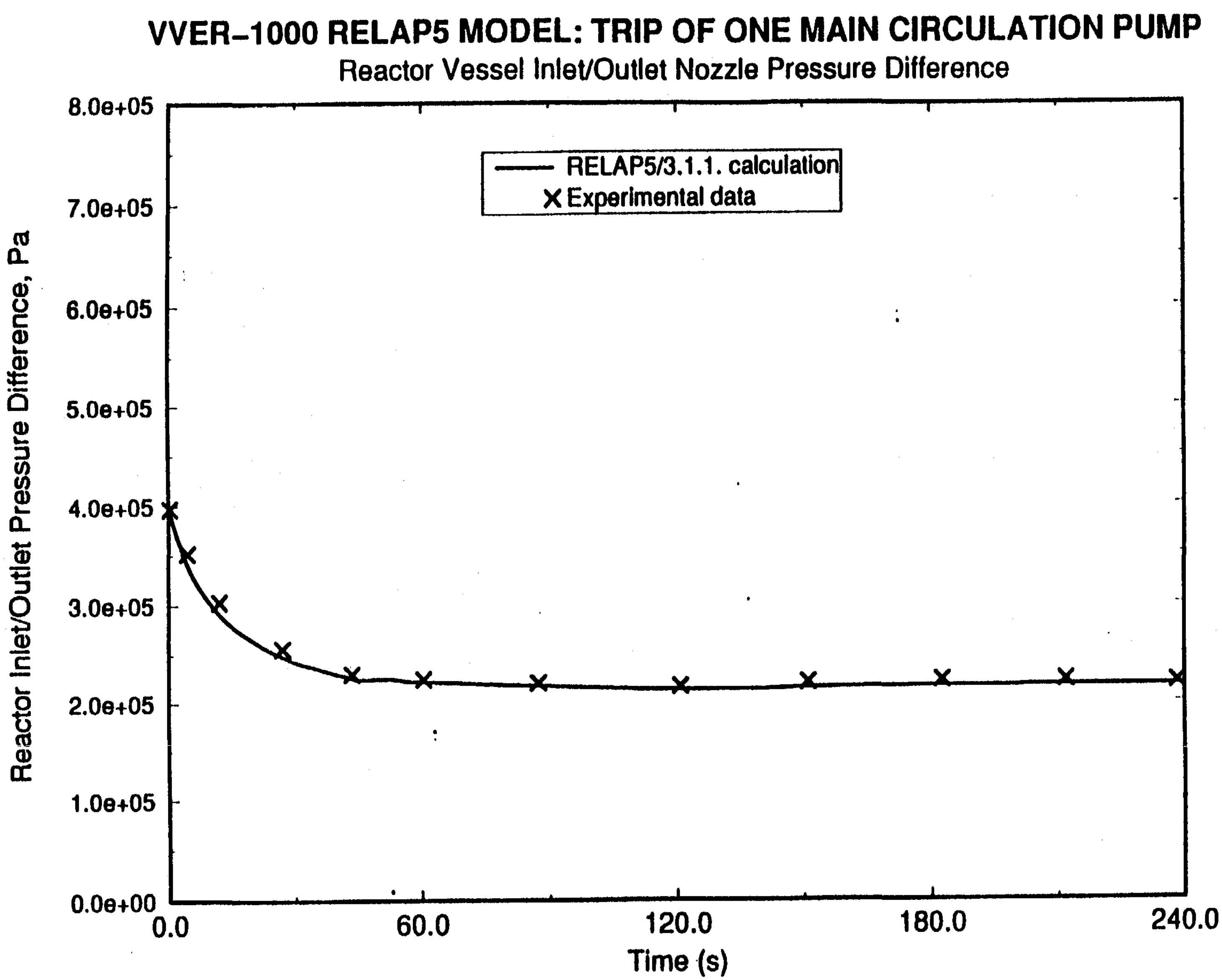

Figure 11: Comparison of Reactor Vessel Pressure Difference