TUF VALIDATION AGAINST REACTOR TRIP DATA AT DARLINGTON NGS

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ABSTRACT

The physical parameters in the reactor controller, reactor physics and thermal-hydraulics models of the TUF code, associated with a reactor trip event are discussed in detail. To check the realism and the fidelity of the TUF performance, the prediction of a reactor trip event is compared with the plant data at Darlington NGS. The prediction of the system parameters is in reasonable agreement with the plant data. This simulation demonstrates the overall code performance in the operational support analysis for CANDU reactors.

1. INTRODUCTION

There is a substantial amount of operating plant data including steady state tests up to full power and commissioning tests in each reactor power plant. These data can be used to validate the reactor system analysis codes. Simulations of operating plant transients by reactor system analysis codes are of great interest for many reasons. Their significant role has been recognized in the nuclear industry. After all, operating plant information is what it is all about. To increase the code reliability for plant operational and safety analyses, predictions of certain plant transients are required for reactor system analysis codes. Operating reactors clearly provide the only full scale and integrated test data for reactor system analysis codes. Plant data can provide the following useful information for the reactor system analysis codes: responses of all the reactor controller systems, overall thermal-hydraulics, correlations for wall frictional and heat transfers, interactions among various reactor systems and input data. After the reports on the TUF validation against the Class IV power failure event (Reference 1) and the LRV test data (Reference 2) at Darlington NGS, several other plant operational data have been used to validate the TUF code at Darlington Nuclear Generation Division. TUF has been used to support short term modifications to the process control system.

Commissioning tests on the effectiveness of shut-off rods (or manual reactor trip) are required by the regulatory authority for every operating CANDU reactor plant in Canada since the function of shut-down systems is the most important issue in the safety of reactor operation. The data generated from these tests during the initial transient (about two minutes) can provide useful information for reactor system codes to verify the following physical parameters: the point kinetics model (illustrated by the reactor power), the pressurizer model during an outsurge process (illustrated by pressurizer pressure and level), the overall wall heat transfer rates in the channels (illustrated by ROH pressures) and heat exchangers (illustrated by steam drum pressures), and the controller responses in the secondary side system (illustrated by boiler water level and feed flow). According to ASME classification, a reactor trip belongs to Level B service limits. Even though a reactor trip is a relatively mild operational transient, simulation of this event has been treated as a base case for reactor system or simulator codes in the utility industry since the reactor trip has been initiated in many other events simulated in safety analysis. For the reactor simulator codes, plant data can be used to adjust the transport or relaxation time constants built in the codes. For the reactor system codes which are based on the basic thermal-hydraulic equations, plant data can be used to verify the correlations used in the code.

The system behaviour of a reactor trip in CANDU reactors is well documented in the design and operational descriptions. When a reactor trip from high or full power occurs, there will be a rapid depressurization and a reduction in coolant temperature in the primary heat transport (PHT) system because of the shrinkage brought about by the rapid reduction of heat transfer from the fuel pins to coolant. The system pressure is still sufficiently high to meet the NPSH requirements for the PHT pumps. As the pressurizer supplies the make-up coolant to the PHT system, the PHT system pressure will stabilize. Following the reactor trip, the PHT pressure controller will respond to the negative pressure error by turning on the heaters in the pressurizer until the PHT system pressure reaches the setpoint (about 10 minutes after a reactor trip for Darlington NGS). The governor valve closes in an attempt to maintain the steam drum pressure at its setpoint. Initially, the steam drum pressures will increase slightly due to the mismatch between the boiler heat transfer rate and the turbine power. Then, the drop in the PHT temperature causes a secondary side cool-down and depressurization. The steam and feed flows will drop. The steam drum pressures and temperatures will then return to their nominal values.

The comparison of system code prediction with plant data for the reactor trip event is desirable because of its simplicity in examining the associated physical parameters implemented in the code. In this paper, these physical parameters in a TUF simulation are discussed. The initial transients (up to two minutes) are compared with the reactor trip data from 100% FP at Darlington NGS by using the TUF version 1.0.3.

2. PHYSICAL PARAMETERS IN TUF SIMULATION

A general description of the physical parameters in the TUF models was presented in Reference 3. In this paper, only the relevant parameters associated with the reactor trip event are discussed below.

SDS1 and Reactivity Control Devices

The primary method of quickly terminating CANDU reactor power when it is required for safety reasons is the shut-down systems (SDS1 and SDS2). The reactor can also be tripped manually at the control room operator's discretion. The standard methods for reactor trip for shut-down system No.1 include the following: (1) shut-off rods (SORs) are dropped, (2) four mechanical control absorber (MCA) rods are dropped due to the action of the stepback program on SDS trip signal, (3) light water zone controller compartments fill at maximum rate, and (4) mode of control of reactor power setpoint resets to alternate-manual (turbine follows reactor) and stays in alternate on trip reset.

There are 32 (separated into two banks) stainless steel-cadmium-stainless steel sandwich SORs in the form of tubes with an active length of 5.72 m and an active outside diameter of 112.7 mm. In their fully inserted position the rods are symmetrical about the horizontal mid-plane of the reactor core. Shut-down system No.1 employs an independent triplicated logic system, which senses the requirement for reactor trip and de-energizes the clutches to release the SORs. The total reactivity for all rods when fully inserted is -71 mk for equilibrium fuelling conditions. Total time of a SOR gravity drop, from fully withdrawn to fully inserted, is about 2 seconds. In the safety analysis, this interval is partitioned into four time periods: (1) A delay of approximately 0.2 s before motion begins; (2) acceleration period from 0.2 s to 0.7 s; (3) constant velocity period from 0.7 s to 1.8 s (the rod is spring-assisted to near its terminal velocity); and (4) deceleration period from 1.8 s to 2.2 s where the rods are cushioned by a hydraulic damper. The rod velocity v can be calculated from the following equation:

$$\frac{dv}{dt} = g\left[1 - \frac{v}{v_t}\right] \qquad [1]$$

where t is time, g is the gravity acceleration and v_t is the rod terminal velocity resulting from the external friction. A large terminal velocity results from the case where a free fall of rod has a small frictional force. The initial rod velocity is calculated from the force generated by the spring accelerator.

Based on the above equation, the rod position, as a function of time after drop starts, is set up as a table in the code for each station. The actual reactivity being inserted by the rods is calculated from their normalized insertion values by multiplying the total bank worth. The table of normalized reactivity as a function of rod position is obtained from the design data or the reactor physics code for each station. The uncertainty in the table of rod position as a function of time is usually greater than that of the reactivity table as a function of rod position. For example, the initial delay time period of 0.2 s and the acceleration period of 0.5 s for rod motion may be too conservative. In reality, due to the spring assisted effect the rods may quickly reach their terminal velocity before 0.7 s. Nevertheless, the effect of SOR motion on the reactor core power only shifts the transient results by the time difference (at most 0.5 second) in the SOR motions if the delay and acceleration periods are reduced.

The four mechanical control absorber (MCA) rods are also dropped due to the action of reactor stepback. The simulation of MCA rods is similar to that of the SOR, except that MCA rods do not have spring accelerators and are filled with orifices to reduce rod terminal velocity (around 3.5 m/s). The total reactivity for MCA rods is -9.5 mk. The zone control system serves as the primary means of short-term reactivity control in the reactor, both for control of total reactor power and for control of spatial flux variations. The spatial flux variation control is not simulated in the code. There are 14 light water zone controllers spatially distributed in each power zone. In the code, the zone controllers are simulated by an average level. The water level is regulated by a valve which controls the water inflow rate. The demand valve position is a function of effective power error. The actual valve position is calculated from the valve sizing equation (Reference 3). The zone water level and the rate of reactivity change are calculated as a function of the valve position. The reactivity change over the time step is then calculated. The total reactivity for the light water zone controller is -6 mk. Due to its limited capacity and slow controller response, the light water zone controllers do not play an important role in the reactor power reduction during a reactor trip.

SG Pressure and Level Controls

Whenever the steam drum pressure changes, the saturation temperature changes simultaneously. Water and metal temperatures will follow with time lags which are complex but generally fairly short. The rate of pressure rise or drop in the steam drum is roughly proportional to the difference between the rate of evaporation in the boiling section of the steam generator and the rate of steam flow out of the steam drum. Therefore, the pressure error between the steam drum pressure (after the rationality and validity check, a single steam drum pressure is selected from four steam generators in Darlington NGS) and the pressure setpoint is used in the feedback terms for calculating the reactor power setpoint, the turbine load setpoint, and the ASDV and CSDV lifts. In the normal or auto mode of overall unit control, the steam generator pressure control program calculates the reactor power setpoint and the setpoint is used by the demand power routine for reactor power manoeuvring, while ASDV and CSDV are being used for pressure trimming. In the alternate mode, as chosen by the operator or selected by logic under certain abnormal conditions such as reactor or turbine trip or reactor setback or stepback, the steam generator pressure control program loses control of the reactor power setpoint, but must maintain the steam generator pressure at its setpoint by controlling the plant load (i.e. turbine), the ASDV or the CSDV operations.

Water level in the steam generator here means the level which would be shown in a gauge glass that correctly indicates the head above the water balance connection on the drum. It is the collapsed water level. In actual fact the water will be turbulent and full of bubbles, but this does not affect the mass of water above the tapping point and hence the level shown in the glass. In general, a boiler at normal operation conditions contains a larger volume of submerged steam outside the generator tubes at high load than at low load, and therefore contains less water. The gradient of the curve of equilibrium water content against evaporation rate represents the change in mass of water in the boiler per unit change in evaporation rate. The simplest possible idealized feedwater regulator would govern the feed inflow rate in proportion to the level error between the level setpoint and the distance of the sensed water level below a reference level. This is the single element control logic used in the code. In this control logic, instability will result if the action of the feedwater regulator is too powerful. This is because the regulator in changing the feed flow also changes the evaporation rate. The resulting swell effect produces a further disturbance in level. If this is too great, the system will be unstable. It is common practice to add to the controller output a signal proportional to the difference between the steam off-take rate and the feed flow rate. This gives the well-known three-element feed regulator, with action based on the water level, steam flow and feed flow.

The level in each steam generator is maintained by three level control valves: one small (20%) and two large (100%) each) valves. At high power operations, only one large control valve is normally kept in service at any one time, the other large valve being on standby. The signal representing the measured level is compared with the setpoint level, thereby generating an error signal. This error signal is sent through a proportional plus integral (PI) controller which eliminates steady state level errors. The output from the PI controller is sent to the three-mode valve controller. The output of the PI controller is the main feedwater valve position signal. A single element algorithm is used for control at low loads (using the small valve). A three element algorithm is used for control at higher loads (using the large valve). When the flow components of the three element algorithm are not available, a default single element algorithm is substituted. The selection of the control algorithms mainly depends on the reactor thermal power.

The level control setpoint is common to all four steam generators and is an increasing function of plant load. This function is chosen such that the locus of the points at various loads will coincide with the swell curve. The level setpoint L_{set} is calculated from the following equation:

$$L_{set} = C_0 + C_1 \, P_R^{1/2} \qquad [2]$$

where P_R is the filtered reactor power, either the average steam flow in units of percentage of full power (at high steam flow) or the percentage reactor power (at low steam flows), and C_0 and C_1 are the constants determined from the commissioning tests.

Governor Valve Control

The turbine load setpoint processed by Turbotrol can be provided either from the steam generator pressure controller or from the unit power regulator. When the steam generator pressure control program is in control of the turbine load setpoint after the reactor trip, the program calculates the load setpoint demand rate E (in velocity algorithm form),

$$E = K_1 \left(p_{set} - p_{sg} \right) + K_2 \frac{dp_{sg}}{dt} - K_3 \frac{dP_R}{dt} \qquad [3]$$

where p_{set} and p_{sg} are the pressure setpoint and the steam drum pressure, respectively, P_R is the reactor thermal power and the constants K_1, K_2 and K_3 are obtained from the commissioning tests. The rates of steam drum pressure and thermal reactor power are calculated using a third-order difference.

The load setpoint demand rate E is then processed by Turbotrol which manipulates the rate of position change for the turbine governor valve. The turbine load setpoint processed by Turbotrol is entered in one of the following two manners: 1 %/s channel (the load program device of Turbotrol is in auto and DCC control is on) and -5 %/s channel (the load program device is switched to manual by Turbotrol). The second manner is primarily used for fast unloading by the steam generator pressure control program, such as following a reactor trip or a stepback.

Heat Transfer Rates in Fuel Pins and Steam Generators

The steady state stored heat within a fuel pin cannot be precisely determined. The effects of thermal cycles, fuel burn-up and other parameters on the gap conductance between fuel and sheath cannot be precisely included in the correlation. The TUF input models an average gap conductance for all fuel pins in a channel. Thus, uncertainty in the initial core stored heat exists. Simulations of different plant conditions with different gap conductances can reduce this uncertainty. The value (10 Kw/m/m/C) used in the previous analyses (References 1 and 2) and safety analysis is also applied to this simulation.

It would be a very expensive model to include all the reactor channels in the thermal-hydraulic circuit of the CANDU reactors. Normally, the averaged channel approach is adopted. For example, each reactor core pass can be simulated by one region (or zone) with averaged channels or several regions each having its own averaged channels. In this simulation, for example, seven regions in the north loop and two regions in the south loop were modelled, each core pass having five nodes axially within each region. This average channel approach is usually adequate for operational support analysis.

The pressure transient at any control volume is given by

$$\frac{dp}{dt} = \frac{\partial p}{\partial M}\frac{dM}{dt} + \frac{\partial p}{\partial U}\frac{dU}{dt} + \frac{\partial p}{\partial M_g}\frac{dM_g}{dt} + \frac{\partial p}{\partial U_g}\frac{dU_g}{dt} \qquad [4]$$

where $\partial p/\partial X$'s are the pressure derivatives, dM is the change in total mass, dU is the change in total internal energy, dM_g is the change in total vapour mass and dU_g is the change in total vapour internal energy. In this simulation, the pressure transients in the reactor channels and steam generators can be expressed approximately by

$$\frac{dp}{dt} \approx \frac{\partial p}{\partial U} dQ_w \qquad [5]$$

where Q_w is the wall heat transfer rate.

The pressure transients at the ROH and the steam drums are strongly influenced by the heat transfer rates in the channels and steam generators. Several options for wall heat transfer correlations are available in the code. Different correlations are applied in different options, depending on the phase condition (single or two-phase) and heat transfer path (positive or negative). A general description of the correlations and their applicable ranges is given in Reference 4. In the pre-dryout region, which is the region encountered during a reactor trip transient, the following correlations are available in the liquid phase: Dittus-Boelter, Kays and Short correlations. For two-phase flow, the following correlations are available: Chen, Thom, Roshenow, Ananiev and McAdams correlations. For steam flow, the Dittus-Boelter, Hadaller, Heinemen, Groeneveld-Delorme, or Short correlation can be selected. In this simulation, the heat transfer options used for fuel channels and steam generators are identical to those used in the safety analysis.

Thermal-Hydraulics Model

In CANDU reactors, two-phase flows occur only in certain areas of reactor piping and components during the normal operation conditions. The two-fluid effects are not important in overall thermal-hydraulics in the design analysis. Therefore, the codes with a one-fluid model have been successfully used in reactor design analysis. In operating support analysis, the two-fluid effect on the transients of system components (boilers, bleed condenser and pressurizer) may be noticed even though its effect on the PHT system is small (Reference 1). In the loss of coolant accidents of safety analysis, two-fluid effects become important either when the phase slip is compatible with the mixture velocity or after the activation of the ECI system. The thermal-hydraulic model implemented in the code has been described in Reference 5.

Similar to pressurized light water reactors, the pressurizer in CANDU reactors (except Pickering NGS) is a particularly important component. Several theoretical models of a pressurizer can be found in the literature. Most of the models require at least two control volumes in the representation of a pressurizer. In TUF, the pressurizer is modelled as a single control volume with a distinct phase separation. A thermal non-equilibrium model is applied to the pressurizer. The interfacial heat transfer rate is dependent on the amount of entrained bubbles and the operating condition of heaters. The void fraction of entrained bubbles in the liquid region is calculated from the level swell model. The total energy of the heaters (four ON/OFF and two variable heaters), which are located at the lower region of the pressurizer, is calculated from a first-order delay equation. A series of tests on pressurizer behaviour have been conducted at the Nuclear Power Demonstration in Ontario Hydro in 1985. It has been found that the transient behaviour of two phase temperatures are different during the in-surge and out-surge processes. In the present case, only the out-surge process is encountered. Also, it should be noted that the gravity head in the pressurizer should be included in the momentum equations since the pressure at the top of the vapour region of the pressurizer is used in the reactor controllers.

Heat Loss from End Shields

In addition to mechanically supporting end fittings and calandria tubes, the end shields provide gamma-ray shielding from the radioactive core. The end shields are made of steel slabs which are light water cooled. Cooling is required because of the heat produced in the steel from the absorption of neutrons and gamma radiation as well as conduction from the hot fuel channels during full reactor power operation. Therefore, the coolant in the dead-end volume of the end fittings will be subcooled due to heat loss from the end shields. In the safety analysis, the water in the dead-end volume of the end fittings is assumed close to the saturation temperature in the steady state condition. However, it has been found from sensitivity studies that the simulation with subcooled (about 10 C) water in the dead-end volume of the end fittings produces a better agreement in the ROH pressure compared with the plant data for the present analysis. Since the heat loss from the end shields has not been included in the simulation, it is reasonable to assume subcooled water in the dead-end volume of the end fittings in the steady state simulation. This modification is not important in the LOCA analyses since its effect on the overall system behaviour is small.

3. COMPARISONS WITH PLANT DATA

On June 28, 1996, while the reactor trip channel E in Unit 3 of Darlington NGS was open due to regularly scheduled testing, a trip signal from the channel D trip computer watchdog occurred. It resulted in a SDS1 trip. Prior to the event, Unit 3 was operating at 100 %FP. In this paper, this reactor trip event is considered. In the plant data logs, data were recorded every six seconds. The station measured error bar is about 100 kPa for pressure and 1 C for temperature. The plant circuit (total 358 nodes and 438 links, see Figure 1 of Reference 2) used in this simulation is identical to those used in References 1 and 2. Except for the coolant enthalpies at the dead-end volumes of the end-fittings, the steady state conditions are identical to those used in Reference 1: fuel power 2651 MW, core flow per pass 2893 kg/s, RIH pressure 11.29 MPa(a), RIH temperature 265 C, ROH pressure 9.93 MPa(a), ROH temperature 309.5 C, ROH quality 0.08 %, channel outlet quality 0 %, boiler pressure 5.07 MPa(a) and boiler feedwater temperature 172.6 C. Also the options for wall frictional and heat transfer correlations are identical to those used in the safety analysis. It is noted that there are six reactor zones in the NE core pass and one zone in the other core passes (NW, SE and SW quadrants) in the network circuit.

Reactor Core Power

In the station, the reactor power level is determined from flux distribution measurements across the core in the horizontal and vertical planes. From the resultant thermal neutron flux distributions the reactor power can be calculated by integrating across the flux distributions and multiplying by the thermal neutron fission cross section. Also, reactor heat balances are carried out on a routine basis in order to check the accuracy of the nuclear power recording instruments. It is known that the neutron flux measurements are not accurate. The thermal power measurement (based on temperature rise across the core and the channel flow rates) is the most accurate method of determining the power output of the core at a high power operating condition. Therefore, there are three types of reactor power measuring devices in Darlington NGS: 6 ionization chambers for low power range, 14 in-core flux detectors for high power range and 44 instrumented channels to measure thermal power. The thermal power measurement is also used for continuing on-line calibration of the in-core flux detectors and ionization chamber signals. In the code, the check on the rationality of thermal power measurement is not performed.

Figure 1 shows the comparison of TUF predicted reactor power with the plant data. It shows that the power drop recorded in the plant data is slightly faster than the code prediction. The possible reason for this discrepancy may be due to a faster SOR speed in the actual plant operation. The force acting on SORs due to spring accelerators may be larger than that estimated in the design data. Since the overall effect of SOR speed on the circuit thermal-hydraulics is not significant in this simulation, the sensitivity study of the SOR speed on the power transient has not been performed. In the safety analysis of large LOCA, the SOR speed is an important factor in the calculation of the reactor power pulse. Based on the result presented here, it can be concluded that in additional to the conservatism in the void reactivity, an extra conservative factor for the SOR speed has been inserted in the simulation of SDS1 in the safety analysis.

Primary Heat Transport System

After the reactor trip from a full power operational condition occurs at time zero with four PHT pumps running, the heat transfer rate from fuel pins to coolant decreases. As a result, there is a rapid depressurization and a reduction in coolant temperature in the PHT system. Figure 2 plots the boiler and preheater powers for the steam generators. The ROH pressure and temperature drops from 9.9 MPa(a) to 8.9 MPa(a) and from 309.5 C to 260 C, respectively, in 50 seconds. Figures 3 to 6 show the pressure transients at four ROHs. At about 7 seconds, TUF predicts a sudden pressure dip (about 100 kPa) in the ROH pressure. This pressure dip does not show up in the plant data because of the 6 seconds interval in recording plant data. However, this phenomenon is observed in other plant data for similar cases (reactor trip from 61 %FP). After the reactor trip, the core pass flow rate slightly increases from 2894 kg/s to 2976 kg/s at time 50 seconds as the coolant density in the channels increases.

The pressurizer supplies the make-up coolant to the PHT system from time zero to about 50 seconds because of the shrinkage brought about by the rapid heat reduction. As a result, the PHT system pressure is stabilized at about 50 seconds. Figures 7 and 8 show the comparisons of the pressurizer level and pressure transients, respectively. Following the reactor trip, the pressurizer heaters try to re-pressurize the PHT system up to its nominal pressure. Therefore, the pressures in the pressurizer and PHT system gradually increase after 50 seconds. The pressurizer heaters provide 1430 kW of energy at 120 seconds.

The comparisons between the code predictions and the plant data in the PHT system parameters have demonstrated the code capabilities in the following areas: (1) the action of the SDS1 system, (2) overall wall frictional pressure drop, (3) overall heat transfer rates in the channels and boilers, and (4) the HT system pressure and inventory controller actions.

Secondary Side System

After the reactor trip, the turbine runs down using the -5 %/s channel by Turbotrol. Initially, the decrease in the boiler power is slower than that for the turbine power. As a result, the steam generator pressures increase initially. After the drop in the PHT pressure and temperature, the secondary side is cooled down and depressurized. The steam drum pressures, shown in Figures 9 and 10 for the North loop, drop from 5.07 MPa(a) to 4.68 MPa(a), and temperature from 264 C to 259 C, respectively, in 50 seconds as the governor valve closes in an attempt to maintain the secondary side pressure at its setpoint. It shows that TUF over predicts the steam drum pressures during the transient. This may result from the lower steam flow predicted by the code. The turbine run-down rate in the simulation may not be close to that in the station operation. The boiler water levels decrease from 14.4 m at time zero to about 12.3 m at time 50 seconds as shown in Figures 11 and 12, respectively for SG1 (NW) and SG2 (NE). After the PHT pressure stabilizes after 50 seconds, the steam drum water levels start to increase slowly toward the level setpoint. As reflected in the steam drum pressures, the steam drum water levels are under predicted. However, the trend of increasing boiler water levels after 53 seconds is predicted well. As a result of the reduced heat transfer rate from the PHT system, the feed water flows also decrease due to the action of the steam generator level controller as shown in Figures 13 and 14, respectively for SG1 and SG2. A sudden drop in the feedwater flows at about 4 seconds is predicted by the code but not observed in the plant data. This phenomenon may result from the SG level control algorithms implemented in the Darlington version.

The overall agreement in the secondary side system parameters between prediction and plant data is not as good as that for the PHT system. It may be initiated either by the steam generator level control algorithms in the TUF Darlington version or by the characteristics of the level control valves used in the simulation.

4. DISCUSSIONS

In this simulation, the following observations are described:

(1). The results of the physical parameters in the NE quadrant (HD1 and SG1) which has six reactor zones are very close to those in the NW quadrant (HD3 and SG2) which has only one zone. It indicates that each core pass can be adequately modelled by using the average channel approach in operational support analysis.

(2). There is a dip in the ROH pressure transient in the prediction, which is not shown in the plant data because of the six seconds interval in recording plant data. However, the dip has been observed in the plant data for another case where the reactor was initially tripped at 61 %FP. Also this dip has been observed in other events (the pump trip or reactor stepback) in CANDU reactors. For example, the single PHT pump trip tests at Darlington and Point-Lepreau NGS have shown a large pressure dip at the ROHs (about 500 kPa).

(3). The control of feedwater flow in the TUF Darlington version requires further investigation, especially in the SG level control algorithms. Other than that, the overall performances by TUF are excellent.

5. CONCLUDING REMARKS

The physical parameters that are relevant to the event of a reactor trip have been discussed. From the results presented here, it can be concluded that the TUF predictions are in reasonable agreement with the plant data for the event of a reactor trip. This simulation continues to enlarge the code qualification base in the operational support analysis for CANDU reactors.

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5. W.S. Liu, R.K. Leung and J.C. Luxat, Two-fluid effects on reactor thermal-hydraulics, 5th International Conference on Simulation Methods in Nuclear Engineering, Montreal, Quebec, September 8-11, 1996.

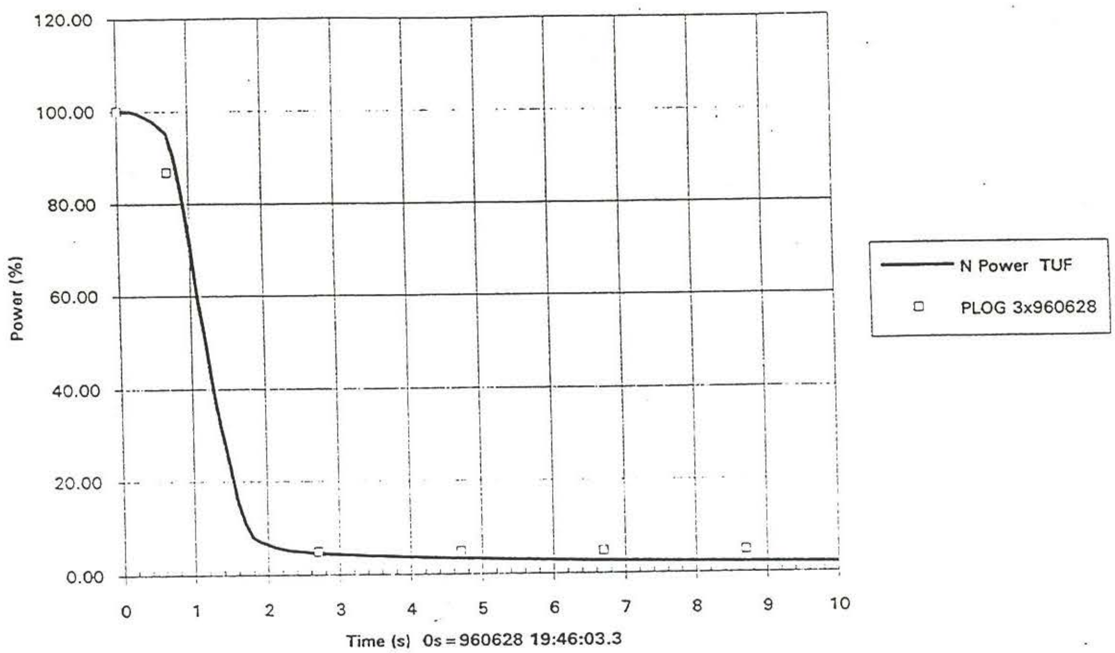

Figure 1. Comparison of neutron power with plant data

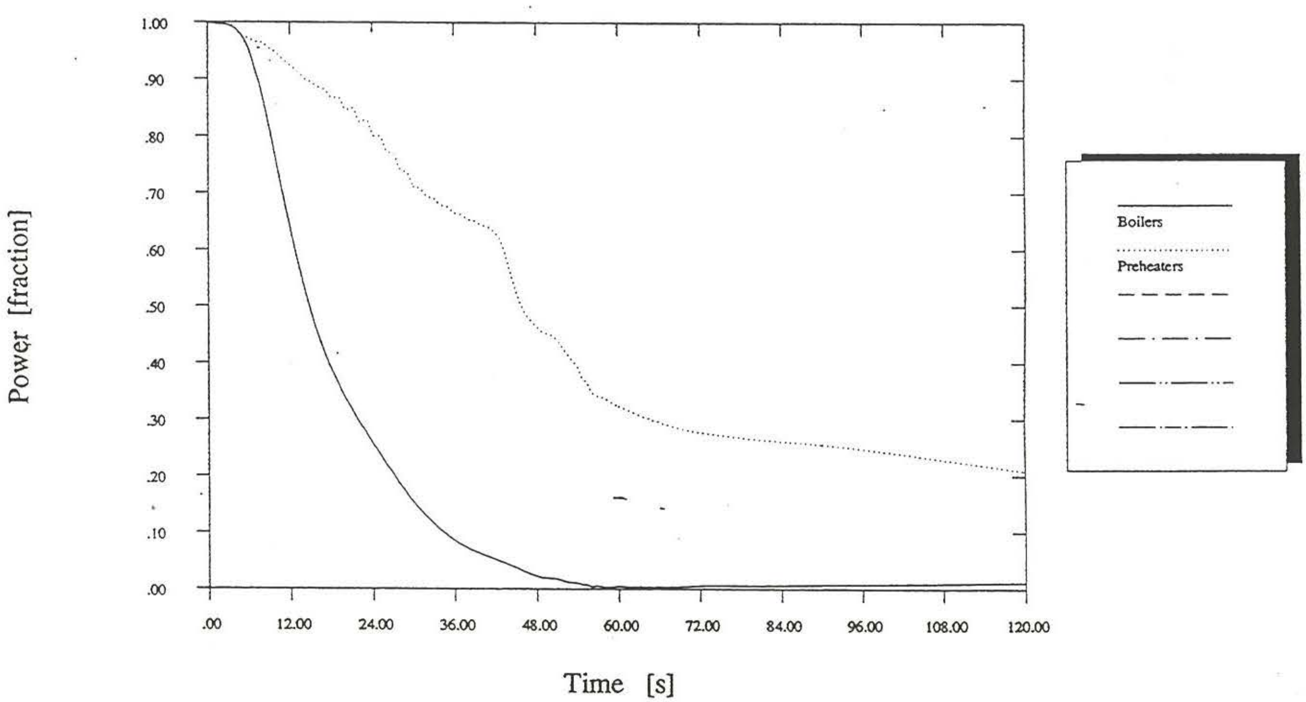

Figure 2. Transients of boiler and preheater powers

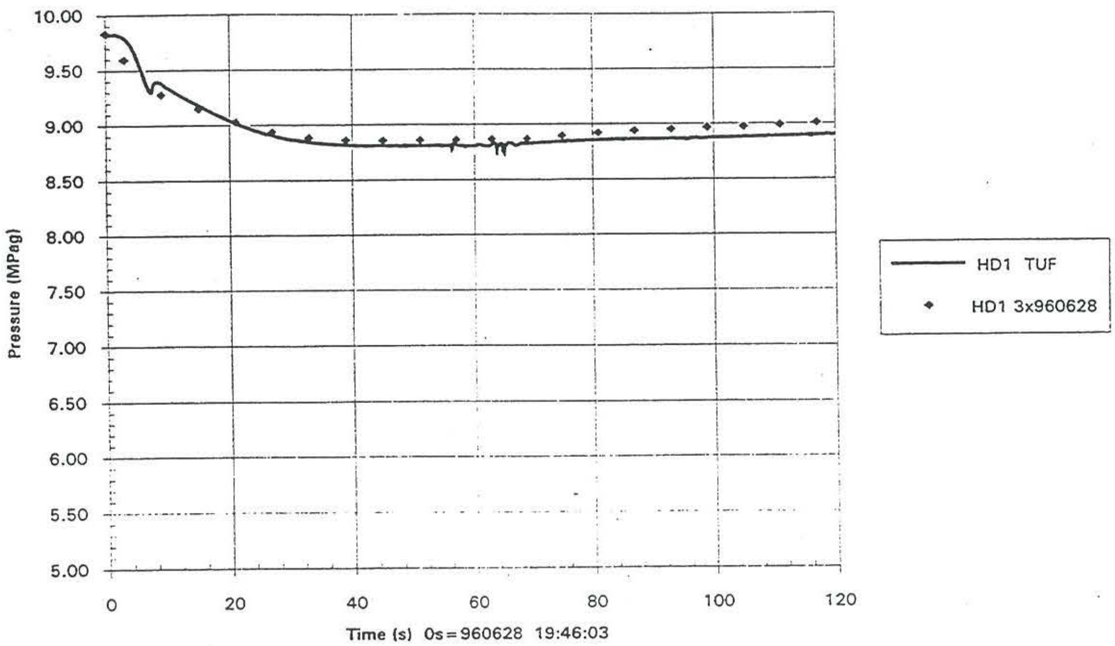

Figure 3. Comparison of ROH pressure with plant data for HD1 (NW)

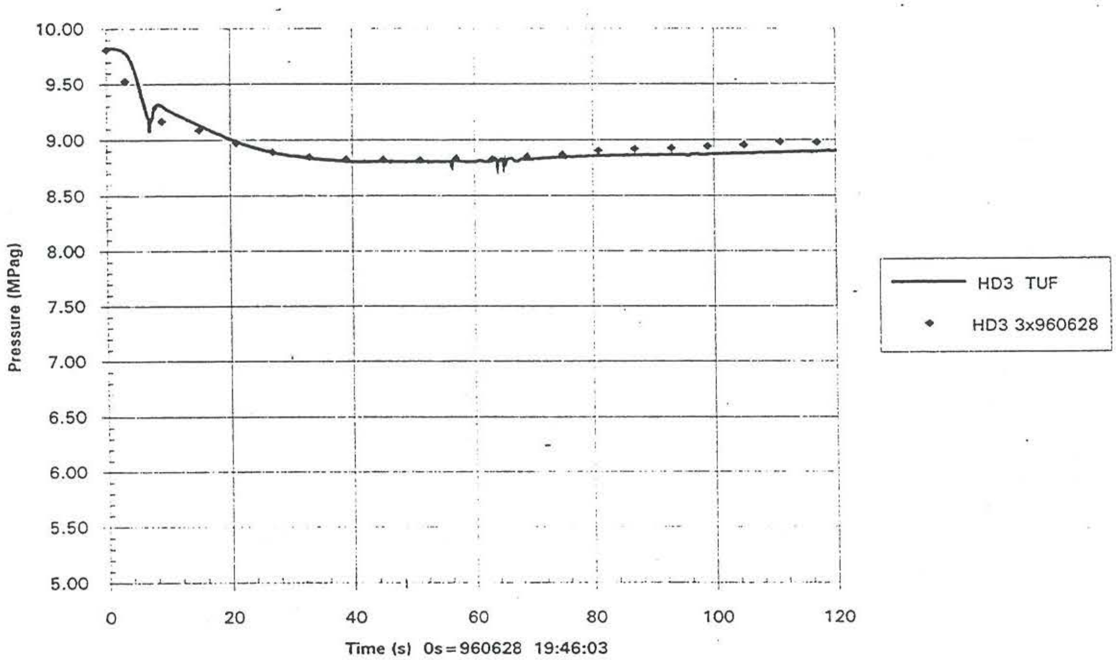

Figure 4. Comparison of ROH pressure with plant data for HD3 (NE)

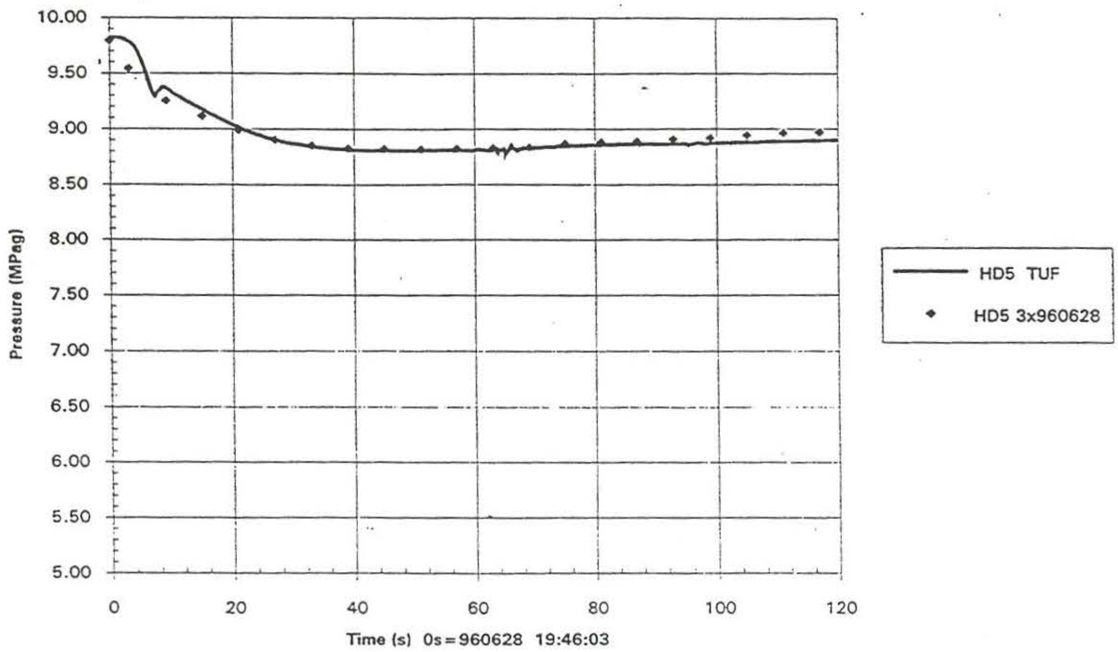

Figure 5. Comparison of ROH pressure with plant data for HD5 (SW)

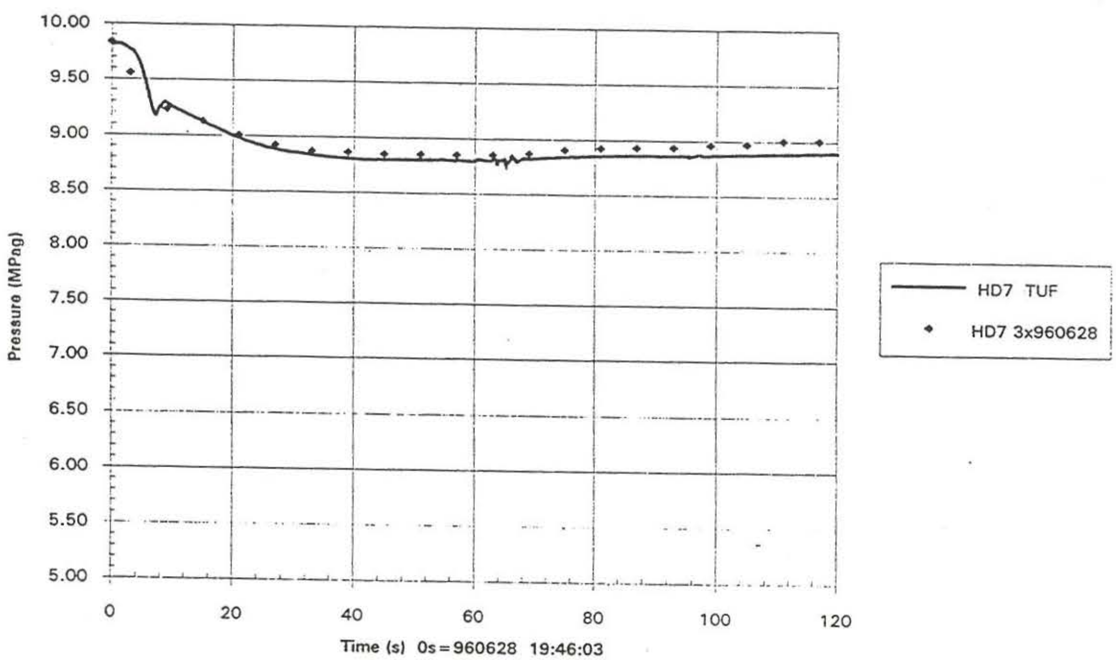

Figure 6. Comparison of ROH pressure with plant data for HD7 (SE)

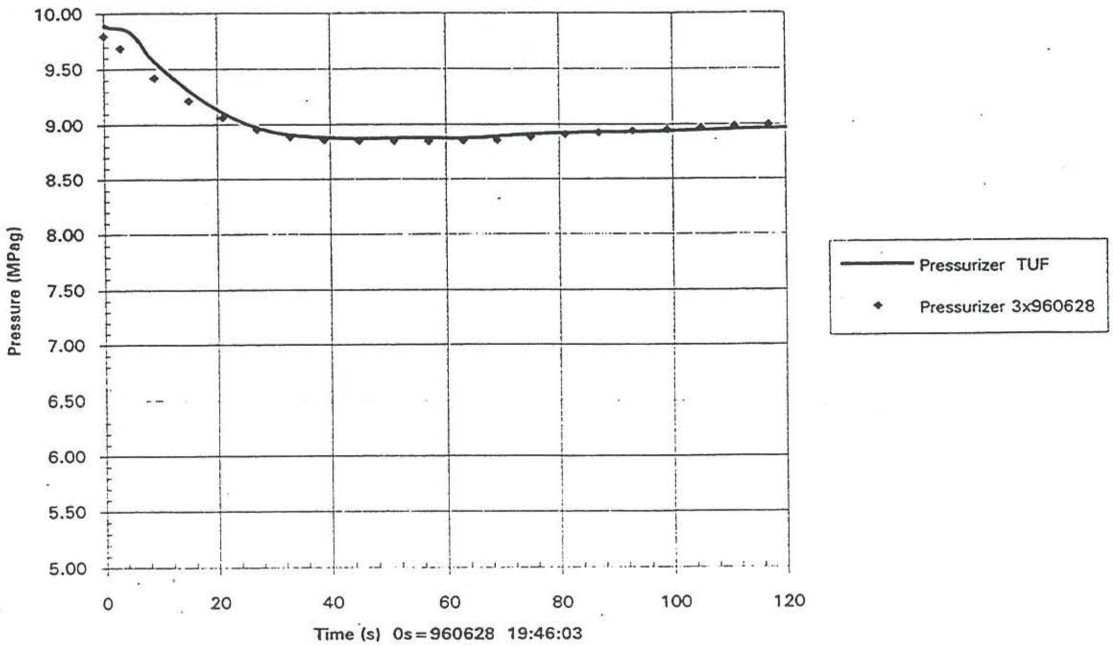

Figure 7. Comparison of pressurizer pressure with plant data

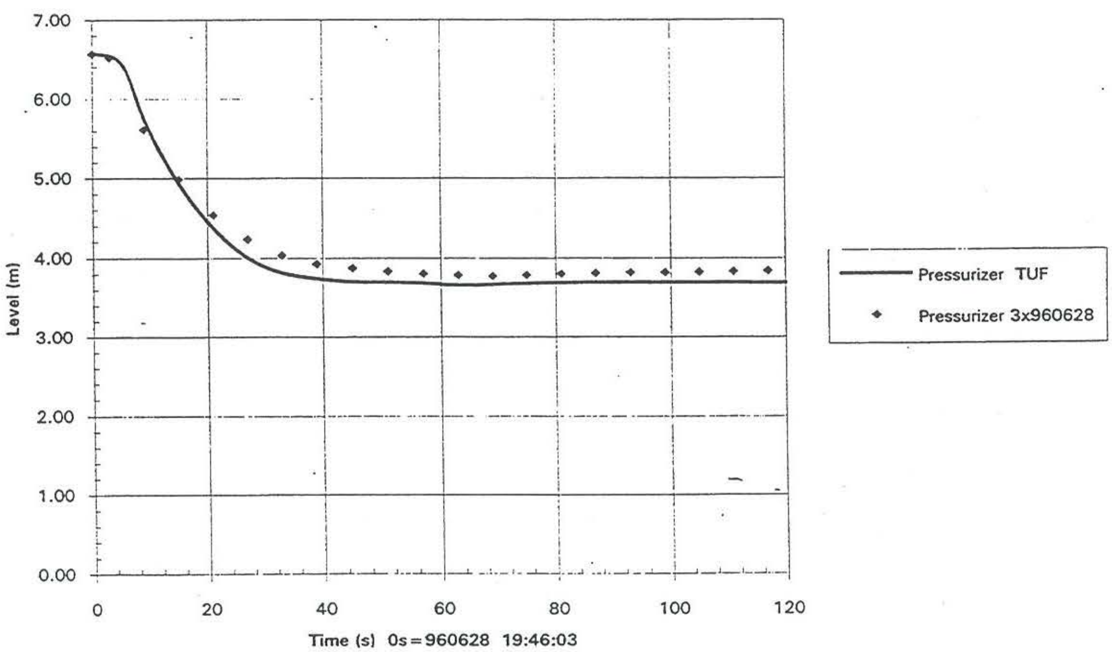

Figure 8. Comparison of pressurizer water level with plant data

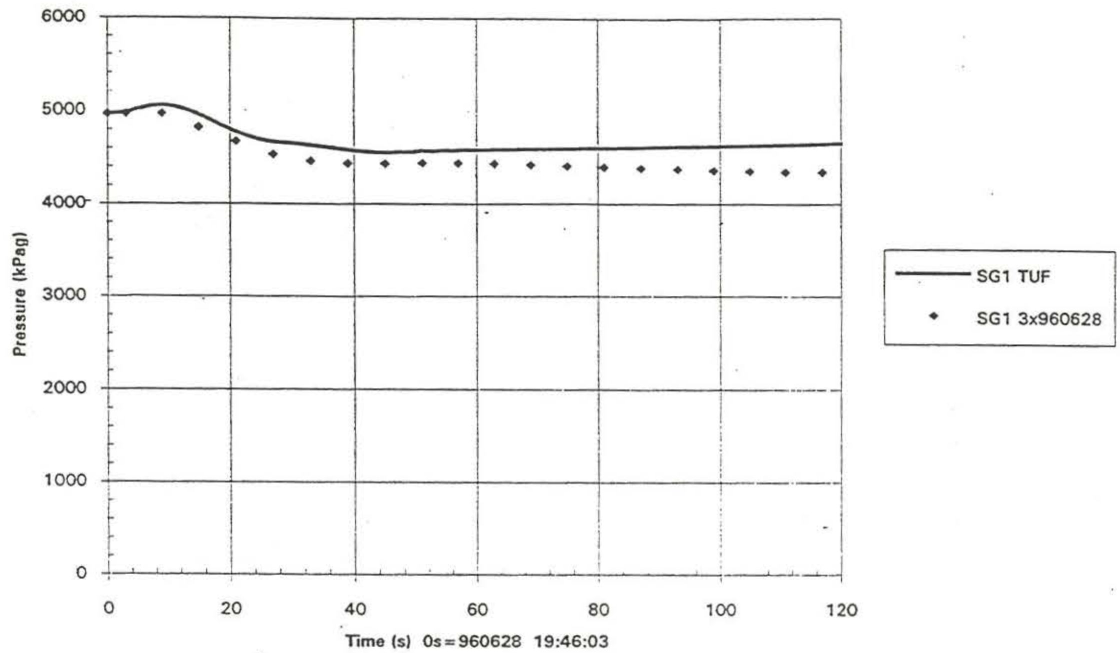

Figure 9. Comparison of steam drum pressure with plant data for SG1 (NW)

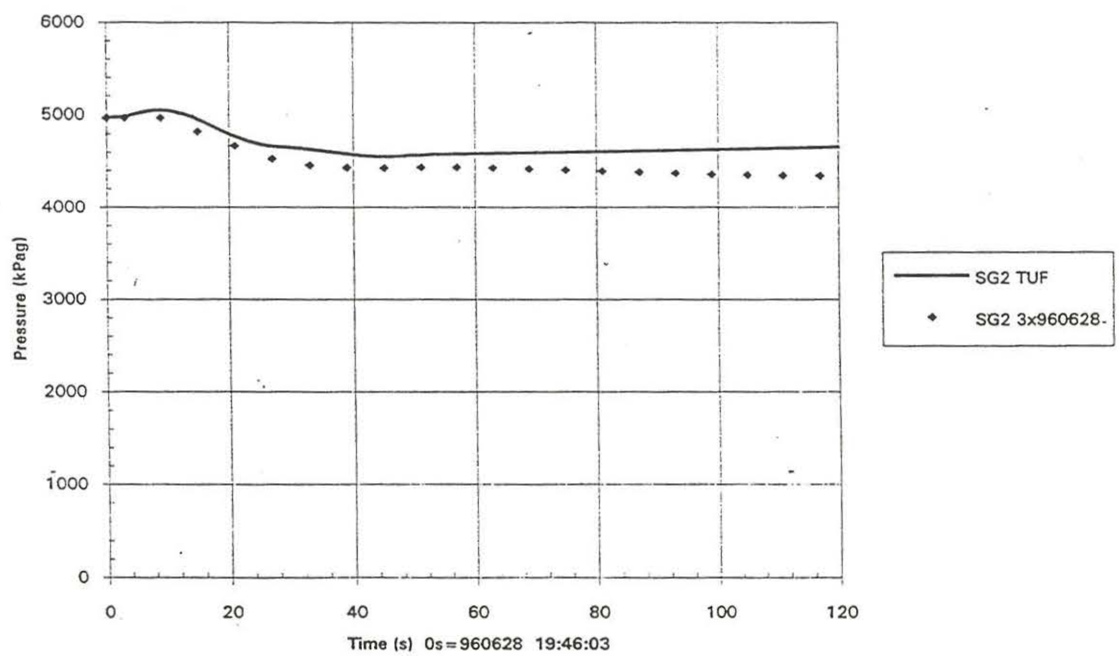

Figure 10. Comparison of steam drum pressure with plant data for SG2 (NE)

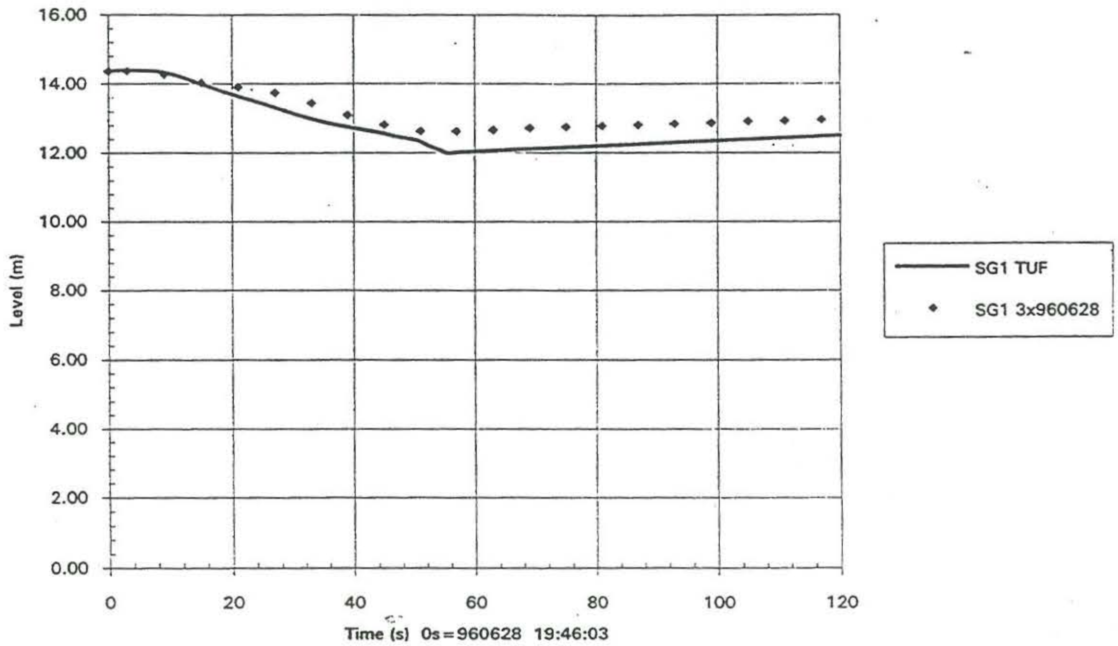

Figure 11. Comparison of steam drum water level with plant data for SG1

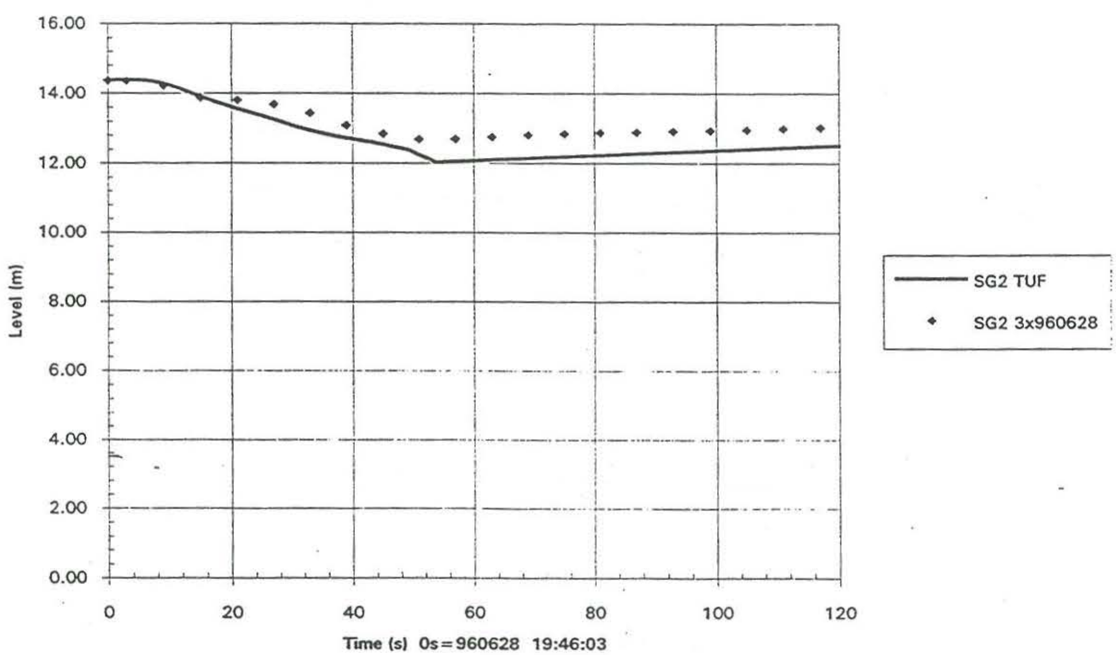

Figure 12. Comparison of steam drum water level with plant data for SG2

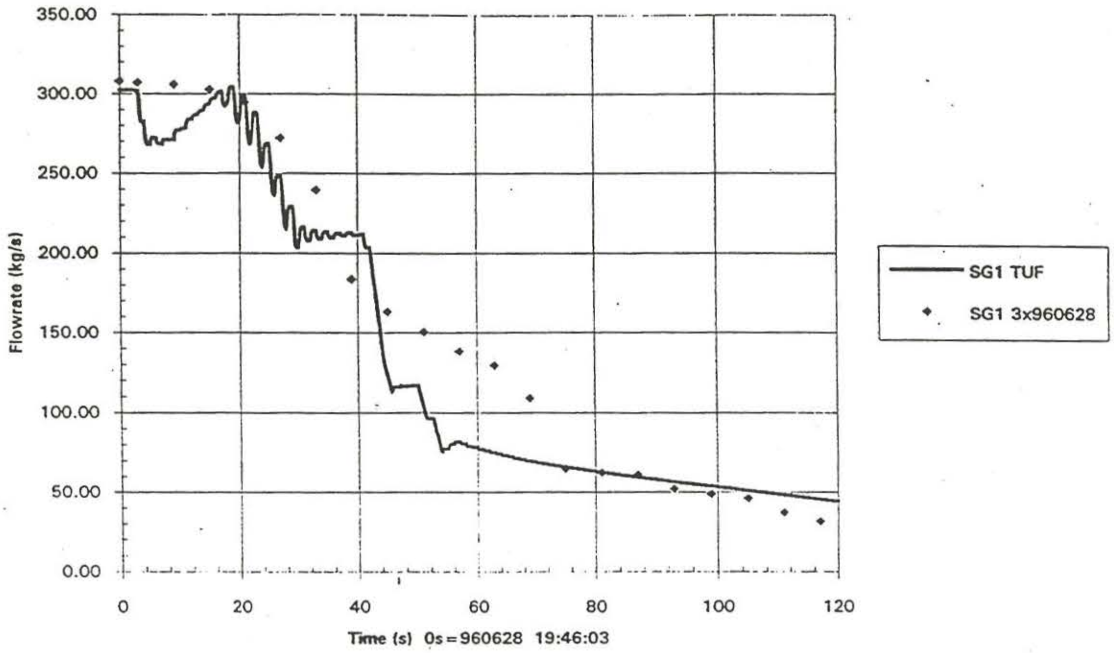

Figure 13. Comparison of feedwater flow with plant data for SG1

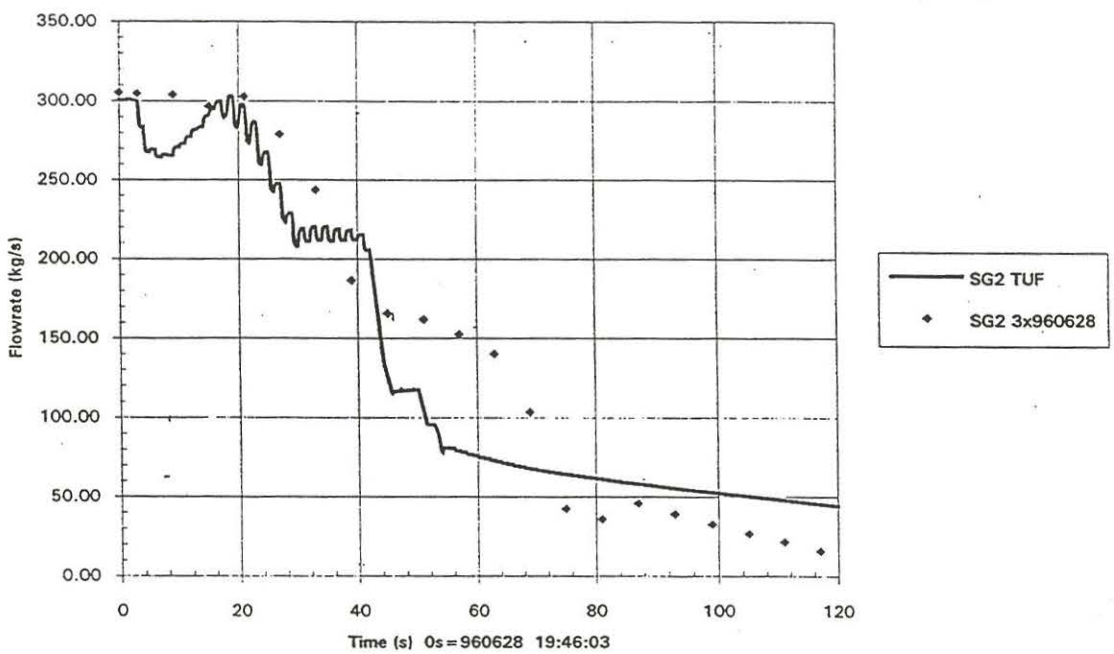

Figure 14. Comparison of feedwater flow with plant data for SG2