Proceedings of Korean Nuclear Society Spring Meeting,
Pohang, Korea, May 1999.

Evaluation of Analytically Scaled Model for Small Break Loss of Coolant Accident at Low Power

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ABSTRACT

The present paper performs analytical evaluations for the potential distortions caused by the scaled models using RELAP5/MOD3 computer codes. By use of scaling analysis, two scaled models with same volumetric ratio are constructed for Korean Next Generation Reactor (KNGR), which is an advanced light water reactor. The scaling methodology adopted in this paper preserves two-phase natural circulation similarities between prototype and scaled models. One scaled model is at full height with reduced flow area. The other model is at reduced height with reduced flow area. By using appropriate scale factors the RELAP5/MOD3 input models are developed. Then, the transient responses of two ideal scaled models are simulated for Small Break Loss of Coolant Accident (SBLOCA) by using RELAP5/MOD3 computer code. The transient responses of two scaled models are compared with those of the prototype. The results indicate that qualitative and quantitative similarities are well preserved for both models during SBLOCA with different break sizes.

1. INTRODUCTION

For the design of advanced light water reactor, it is very necessary to confirm the thermal-hydraulic performance of new safety systems in response to the postulated accidents to ensure public safety. Also the recent focus on the importance of beyond-design basis events including the accidents at low power requires the verification of the performance of safety system in a broader range of operation. The advanced light water reactor and existing plant at non-design basis conditions including low power operation require the operation of safety system at low flow and pressure condition. As the previous researches in nuclear reactor safety were focused on high pressure and high temperature range, good test data or well-verified correlation is few. Therefore, it is necessary to perform integral effect or separate effect thermal-hydraulic test to confirm the performance of safety systems and provide thermal-hydraulic data bank for use in the design of safety system and developing computer codes for reactor safety.

The integral test facility for the simulation of thermal-hydraulic behavior of the nuclear power plant in response to the postulated accidents is usually called as the integral test facility. PUMA test facility [1] and APEX test facility [2], and ROSA test facility [3] are good examples of integral test facilities. Expenses for building test facility and risks of being at prototypic condition lead to building scaled test facility. To design a reduced scale test facility, it is necessary to set up scaling law to determine the non-dimensional parameters. There were pioneering papers for liquid metal reactor [4] and pressurized water reactor [5]. Scaling laws applicable for reduced pressure system [6], flashing system [7], and structured scaling laws for severe accident [8] are developed. For the PUMA and APEX integral test facilities, complicated but structured scaling laws are developed and used in the design [1,2].

However, before the construction of test facility we need to evaluate the validity of scaling laws used in the design. One good way of testing scaling distortion is to rely on the numerical simulation by the computer code. As RELAP5/MOD3 is one of the widely used computer codes for the simulation of two-phase flow thermal-hydraulics of reactor, there were some researches [9,10] with this objective. Larson [9] evaluated the analytically scaled model, where scaling laws for natural circulation of Ishii [5] are applied, in response to Small Break Loss Of Coolant Accident (SBLOCA) by using RELAP5/MOD2 computer code. The responses of scaled model were different from those of prototype and the reasons for the difference were not clearly explained. Ransom [10] did similar approach. He evaluated responses of analytically scaled model of simplified boiling water reactor in response to main steam line break by RELAP5/MOD3 and compared it with those of prototype. The agreement was very good. Ransom used similar scaling laws as those by Larson [9]. But it looks like the thermal-hydraulic behavior of main steam line break of boiling water is simpler than SBLOCA of pressurized water reactor. And different from Larson, Ransom applied scaling laws to heat structure, which may have lead to better results.

In this study, we evaluate the thermal-hydraulic responses of analytically scaled models of Korean Next Generation Reactor (KNGR) to the postulated accident of SBLOCA to provide insights to the design of scaled test facility. Since the various correlation and constitutive relation used in RELAP5/MOD3 computer is not scalable, the actual thermal-hydraulic responses of test facility would be different from those simulated by RELAP5/MOD3 computer code. However, the integral responses will give valuable insights for the design of test facility.

2. SCOPE OF ANALYSIS

For the scaling analysis we considered SBLOCA initiated from natural circulation at 5% power, as the accidents at low power become a very important issue for operating and new plants [11]. Also, it gives us chance to look at the models and correlation at low pressure and low flow used in the RELAP5/MOD3 computer code. Another point is that as the core power is one of the limiting factors in terms of economy for the design of test facility, it will be very nice if there is no major difference in thermal-hydraulic behavior of SBLOCA events at full power and reduced power. Then, we can verify the models and correlation at full power by the test results at low power.

2.1 Typical Event Scenarios for SBLOCA at full power and 5% power

A typical SBLOCA event at full power initial condition and 5% power is analyzed by the RELAP5/MOD3 computer code. The event is initiated by a rupture with equivalent size of 0.2 ft^2 in Direct Vessel line (DVI) connected to the reactor vessel. The initial conditions for full power case and low power case are listed in Table 1. The nodalization diagram used for RELAP5/MOD3 analysis is shown in Figure 1.

Table 1 Initial conditions for full power and natural circulation

Parameters	Full Power	Natural Circulation
Core power	3,914 Mwt	195.7 Mwt
Pressurizer pressure	15.5 MPa (2,250 psia)	15.5 MPa (2,250 psia)
Cold leg /Hot leg temperature	561 K /593 K	565 K /590 K
Loop Flow Rate	10,834 kg/s	1,406 kg/s
Steam generator pressure	6.52 Mpa	7.58 Mpa

Typical phases of SBLOCA scenario can be identified from the plots for the transient pressurizer pressure, steam generator pressure, break flow rate and HPSI flow rate, and Loop seal mass in Fig. 2, Fig. 3, Fig. 4, and Fig. 5 respectively. In the Figures L indicates the low power case. The line without symbol indicates full power case.

Blow-down period: The initial pressure drop from 2250 psia is very rapid. The rate of pressure drop is somewhat decreased when the pressure reaches the saturation pressure of the hottest reactor coolant. The depressurization continues until it approaches steam generator pressure.

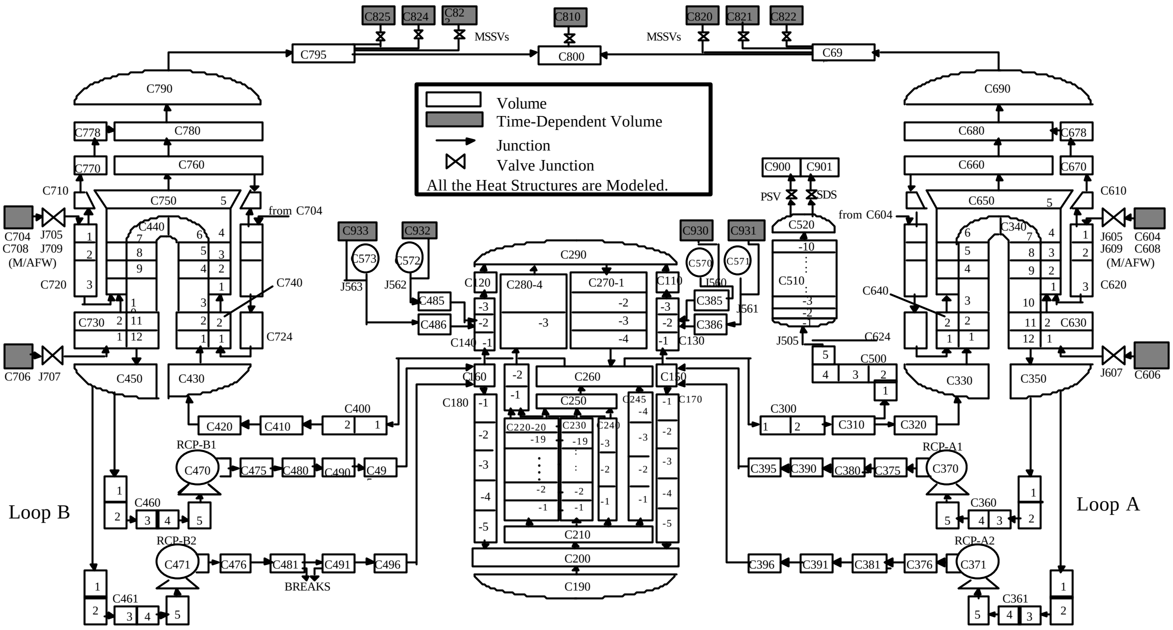

Fig. 1 Nodalization Diagram for KNGR RELAP5/MOD3.2 Model

Pressure Plateau: The halt in the depressurization during 50 to 200 seconds is because the energy removal out of break is less than the energy generation rate in the reactor core. The excess energy is transferred to the secondary water of steam generator, which results in the opening of Main Steam Safety Valves (MSSVs). It requires that the reactor coolant system (RCS) temperature stay somewhat above the steam generator water temperature. This temperature dictates the RCS pressure. The duration of pressure plateau increases as the break size decreases. During this phase, the break flow is two-phase.

Depressurization Phase: Before reaching 200 seconds, the primary mass in the steam generator U-tube and the loop seal is cleared through the break. Then the break is uncovered. As the rate of energy removal out the break increases, the RCS pressure continues to decrease after 200 seconds. The depressurization continues until the break flow returns to two-phase flow as the RCS inventory is recovered. During this phase the break flow is close to single-phase steam. Therefore, the energy removal rate is bigger than the decay heat, which results in RCS depressurization. As the RCS pressure decreases, the inflow from High Pressure Safety Injection (HPSI) pump becomes equal to the break flow. It recovers RCS inventory and the break flow becomes two-phase.

Recovery and Balance Phase: After 600 seconds, the RCS pressure reaches equilibrium state. The break flow matches HPSI flow. It can be easily seen from the figure for the break flow and HPSI inflow. The decay heat from the core equals the energy removal by the break flow and energy required to heat up HPSI flow to RCS average temperature.

In the analysis we did not assume injection of Safety Injection Tanks (SITs). Aalso as the steam generator level was maintained above the Auxiliary Feed Water Actuation (AFWS) level set-point, AFWS is not actuated.

2.2 Comparisons of full power case and natural circulation case

Comparative analyses by RELAP5/MOD3 are performed for SBLOCA at full power and SBLOCA at 5% power. Though the full power case and low power case has different initial conditions, such as, core power, cold leg and hot leg temperatures, the transient behaviors are quite similar as shown in the plots for pressure and break flow. The duration of pressure plateau is a little bit shorter in the full power case as shown in Fig. 2. The break flow rate during 200-250 seconds shows more oscillatory behavior in low power case. This behavior is closely related to loop seal mass, core inlet flow rate, and collapsed core level behavior. For the full power case, as the core inlet flow is maintained rather high during the Reactor Coolant Pump (RCP) coast-down as shown in Fig. 7, the loop seal mass is depleted much faster (Figure 6). It results in early break uncovery and subsequent increase in the quality of break flow. The core collapsed level for the low power case is much lower than that of full power. This is due to bigger mass loss in the core by the bigger break flow and reverse core flow during this period.

The steam generator pressures are different. This is partly due to the difference in initial pressures. The full power case started from the nominal design pressure of 950 psia and the low power case started from 1100 psia.

The present analysis results indicate that the SBLOCA at low power is conservative in terms of the core inventory, which is primary safety criterion. It justifies our use of SBLOCA at low power as a base case for scaling and test.

3. SCALED MODELS FOR RELAP5/MOD3 ANALYSES

We construct analytically scaled model for KNGR. One is full height and the other is reduced height model at a given volumetric ratio. We intend to design test facilities to simulate SBLOCA at low power and investigate the validity of scaled design by the numerical simulations using RELAP5/MOD3.

3.1 Prototype

A natural circulation is simulated for Korean Next Generation Reactor. The core power is maintained at 5% of nominal power and steam flow and feed water flows are matched to this condition. The steam generator pressure is at 1100 psia. The feed water is supplied at 200 ^{o}F and 1200 psia. Hydraulic diameter of SG U-tube is taken as the diameter of U-tube. MSSVs are modeled in the steam generator secondary side to consider the effect of primary-to-secondary heat transfer during SBLOCA transient simulation. The resultant steady state conditions are as follows

Core Power: 195.7 Mwt
Hot Leg Temperature: 590.1 K, Cold Leg Temperature: 565.4 K
Loop Flow Rate: 1406 kg/s
Pressurizer Pressure: 15.5 Mpa, Steam Generator Pressure: 7.58 Mpa
Pressure difference between bottom of reactor vessel to top of steam generator U-tube primary side: 811 kPa
Primary Volume: 719.91 m^3, Secondary Volume: 321.98 m^3
Primary Mass: 495,046 kg, Secondary Mass: 100,000 kg

3.2 Scaling Analysis

The purpose of integral test facility is to investigate the performance of reactor systems and safety system in response to the postulated accidents. As the system response before the reactor trip is usually not complicated, the interested period is after the reactor trip. Therefore, test condition is usually taken from the condition after the reactor trip with various initiating events. Also as it is expensive to build test facilities at prototypic conditions, they either use analytical models to set up initial conditions for test [1] or use pressure scaling methodology [2]. Here, we take natural circulation condition as the prototypic condition. And we will construct ideally scaled model. Then, we will perform numerical simulation for the natural circulation and the small break loss of coolant accident.

From the loop continuity and momentum balance equations, we obtain the following requirements per Ishii's scaling model [2] based on the two phase natural circulation,

$$(a_i/a_o)_R = 1, \quad (1)$$
$$[\, \Sigma(fl/d + K)\,(a_o/a_i)^2]_R = 1\,, \quad (2)$$
$$(l_i/l_o)_R = 1, \quad (3)$$
$$q_R = 1/\sqrt{l_R}, \quad (4)$$
$$\tau_R = l_o/u_o = \sqrt{l_R}\,, \quad (5)$$
$$u_R = \sqrt{l_R} \quad (6)$$

For the similarity of the heat structure, such as, core fuels and U-tube metals, we can derive similarity parameters similar to Ransom's analysis [10] as follows

$$Q_m/Q_p = (C_p m\Delta T/t)_m/(C_p m\Delta T/t)_p = 2m_m/m_p \quad (7)$$
$$Q_m/Q_p = (hA\Delta T)_m/(hA\Delta T)_p = A_m/A_p = P_m L_m/P_p L_p \quad (8)$$
$$Fo_m/Fo_p = [(k/\rho c_p)t/D^2]_m/[(k/\rho c_p)t/D^2]_p = 1 \quad (9)$$

3.3 Scaled Model with Full Height

As $l_R = 1$, the time scale, the volumetric power scale, and the resultant velocity rate ratio become 1. Then, the area of scaled model should be 1/400 of prototype. The mass flow rate, the break area, and the power are scaled as 1/400. We maintained the K-factors the same as prototype. For the fuels and U-tube metals, we obtain following relations from equations (7), (8), and (9).

$$D_m / D_p = 1 \quad (10)$$
$$k_m/k_p = 1 \quad (11)$$
$$n_m/n_p = 1/400 \quad (12)$$

Consequently, the number of fuel rods and U-tubes are reduced per volume scale. However, the hydraulic diameters of fuels and steam generator U-tubes are maintained the same as prototype. Also, the conductivity of fuels and U-tubes are not changed. The area of Main Steam Safety Valves (MSSVs) is scaled as 1/400. Injection flow from the HPSI has been reduced to 1/400. The steady state natural circulation results obtained by RELAP5/MOD3 are as follows

Core Power: 0.48925 Mwt
Hot Leg Temperature: 597.8 K , Cold Leg Temperature: 564.9 K
Loop Flow Rate: 2.58 kg/s
Pressurizer Pressure: 15.5 Mpa, Steam Generator Pressure: 7.58 Mpa
Pressure difference between bottom of reactor vessel to top of steam generator U-tube primary side: 806 kPa
Primary Volume: 1.7974 m^3, Secondary Volume: 0.80494 m^3
Primary Mass: 1238 kg, Secondary Mass: 257 kg

It can been seen that the primary and secondary masses and volumes are scaled properly. The pressure of primary and secondary are same as prototype. The pressure difference for the lowest point and highest point is maintained almost same. However, the resultant loop mass flow rate of 2.58 kg/s is much smaller than the scaled flow rate of 3.515 kg/s. This is because the loop resistance from skin friction has been increased due to much higher l/d ratio than that of the prototype. As the loop mass flow rate has been reduced the temperature difference between core inlet and out let has been increased. As the core inlet temperature is controlled by steam generator secondary pressure, the core inlet temperature is close to that of the prototype.

3.4 Scaled Model with Reduced Height

Let $l_R = 1/4$, then $q_R = 2$, $\tau_R = 1/2$. The time at the scaled model is two times faster than the prototypic time. As the volume ratio is 1/400, the area ratio becomes 1/100. The resultant velocity scale and mass flow rate becomes 1/2 and 1/200 respectively. Then the choke flow area and power ratio becomes 1/200. We maintained the K-factors the same as prototype. For the fuels and U-tube metals, we obtain following relations from equations (7), (8), and (9).

$D_m / D_p = 1/2$ (11)
$k_m / k_p = 1/2$ (12)
$n_m / n_p = 1/200*4*2 = 1/25$ (13)

To preserve the ratio of stored energy release from the core, convective heat transfer, and to maintain equal heat flux at the surface the heat structure are scaled. Here, Dm and Dp are the diameters, kn and kp are conductivity, n_m and n_p are number of heat structures for prototype and model. Therefore, the conductivity of fuels and SG U-tubes is reduced to half. The decrease in diameter is reflected in the heat transfer area, which is scaled to 1/100. The thickness of heat structure is not changed. The hydraulic diameter of scaled model is maintained same as that of prototype. The steady state natural circulation results obtained by RELAP5/MOD3 are as follows

Core Power: 0.48925 Mwt
Hot Leg Temperature: 589.4 K, Cold Leg Temperature: 564.6 K
Loop Flow Rate: 3.548 kg/s
Pressurizer Pressure: 15.5 Mpa, Steam Generator Pressure: 7.58 Mpa
Pressure difference between bottom of reactor vessel to top of steam generator U-tube primary side: 212 kPa
Primary Volume: 1.8070 m^3, Secondary Volume: 0.80494 m^3
Primary Mass: 1246 kg, Secondary Mass: 244 kg

It is shown that the primary and secondary masses and volumes are scaled properly. The pressure of primary and secondary are same as prototype. The pressure difference for the lowest point and highest point is 1/4, which is due to reduced height scaling. The resultant loop mass flow rate of 3.548 kg/s is very close to the ideally scaled flow rate of 3.515 kg/s. It is due to slight decrease in skin friction portion of loop resistance. As the loop mass flow rate is close to ideally scaled condition, the core inlet temperature and core out let temperatures are close to those of prototype.

4. SIMULATION OF SBLOCA TRANSIENT FOR SCALED MODELS

SBLOCA scenarios at low power are simulated for prototype and scaled models. For each model two break sizes of 0.2 ft^2 and 0.1 ft^2 are considered to look at the effect of break sizes.

The pressurizer pressure, steam generator pressure, break flow rate, and core collapsed level, break node quality, and flow regime at the core exit node for the cases with 0.2 ft^2 break size is shown in figure 8, 9, 10, 11, 12, and 13 respectively. In the figures R denotes the results by the reduced height model. Does F and P full height and prototype respectively.

Surprisingly the overall behaviors of scaled models are very similar to those of the prototype, though they started from different initial conditions. It means that the scaling parameters are appropriate enough to suppress initial disturbances. The pressure transient of reduced height model is found to be much closer to that of prototype. The full height model results in slower depressurization at the initial blow down phase. It is due to much higher hot leg temperature than that of the prototype. Figure 10 and 12 show that the break is uncovered around 200 seconds. After the break uncovery the break flow becomes single-phase steam. The quality of break flow of full height model is smallest among three cases, which explains higher break flow rate and slower depressurization during 300 to 600 seconds. Also, it results in less core inventory as shown in the figure of core collapsed level in Figure 11.

The steam generator pressure is shown in Figure 9. The depressurization rate of reduced height model is slowest. As the reduced height case has more mass release through the MSSVs during forward heat transfer period, the steam generator has lee inventory than the prototype. It results in slower depressurization rate than that of the prototype. This distortion comes from the fact that we did not explicitly maintain the similarity of pool boiling in the secondary side. However, the response of steam generator pressure is less important than other parameters in terms of the primary safety criterion of core inventory.

The change in flow regime for three cases is shown in Figure 13. The flow regime starts from the bubbly flow (indicated by 4). Then it becomes slug flow (indicated by 5), annular-mist flow (indicated by 6), then it returns to slug flow. The overall trends of scaled models and prototype are same. But the details of change during blow down, and pressure plateau phases are slightly different. The reduced model does not evolve to annular-mist flow regime. However, it does not make noticeable difference in the overall behavior.

As the above findings and discussions demonstrate the validity of scaling methodology used in this paper, we looked at smaller break size case to generalize our findings. We considered the SBLOCA scenario at 0.1 ft^2 break size. The transient responses of pressurizer pressure, steam generator pressure, break flow rate, and core collapsed level is shown in Fig. 14, 15, 16, and 17 respectively.

As can be seen from figures, the overall trends are very similar to those of 0.2 ft^2 case. The pressurizer pressure and break flow rate of scaled model at reduced height are closer to those of prototype than full height model. The full height model resulted in minimum core inventory. The qualitative and quantitative trends are similar to those of 0.2 ft^2 case.

These findings are very valuable, since previous study [9] showed rather big differences in the transient responses between prototype and scaled model for SBLOCA. The pressurizer pressure was very different and the cause of the difference was not identified.

Above results demonstrate that the proposed simple scaling parameters based on the similarity of two-phase natural circulation preserves complicated thermal-hydraulic behaviors during SBLOCA. It gives us valuable insight for the design of test facilities. Both of the scaled models at low power seem to be acceptable. However, to select the full height model we need to consider the effect of excessive heat loss and multi-dimensional effect. We should maintain certain aspect ratio to preserve the multi-dimensional effect of the prototype, which cannot be modeled by RELAP5/MOD3 computer code. And the heat loss should be much smaller than core power. On the other hand, the reduced height model is better in these aspect.

5. CONCLUSIONS

The present study proposed scaling parameters for the design of test facilities to simulate SBLOCA transients. By performing simulations of SBLOCA for the two scale models by RELAP5/MOD3 computer codes, it is shown that the proposed scaling parameters preserve the major thermal-hydraulic behaviors during SBLOCA. Also the use of low power SBLOCA for scaling analysis is proposed rather than using full power case as a prototypic condition. These findings give valuable insights for the design of integral

ACKNOWLEDGMENTS

The authors would like to appreciate the support from Ministry Of Science and Technology of Korean Government.

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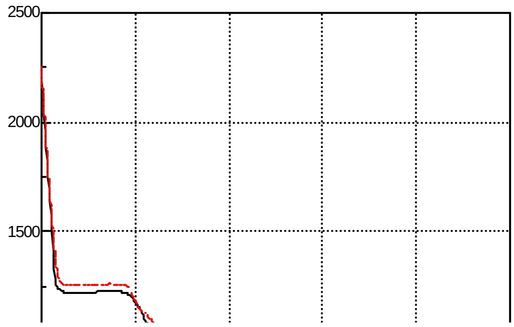
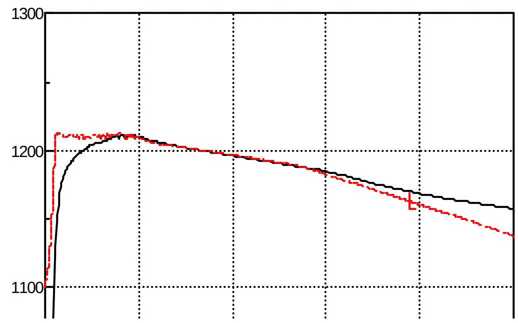




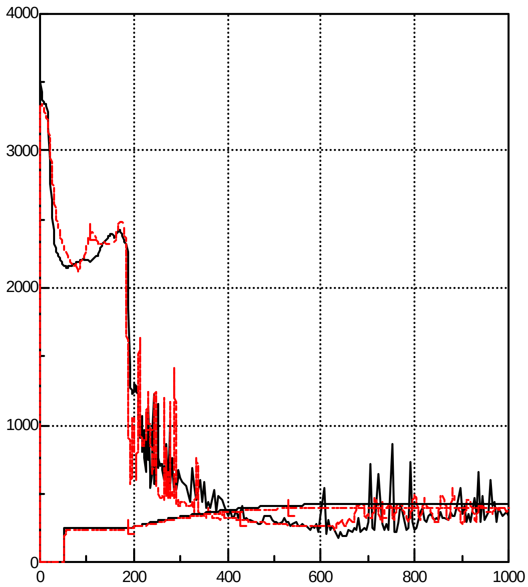


Fig. 4 Break and HPSI Flow Rate (kg/s)

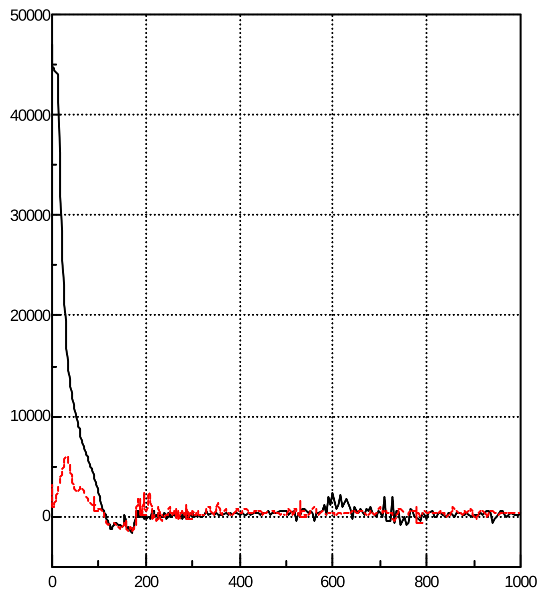


Fig. 7 Core Inlet Flow Rate (kg/s)

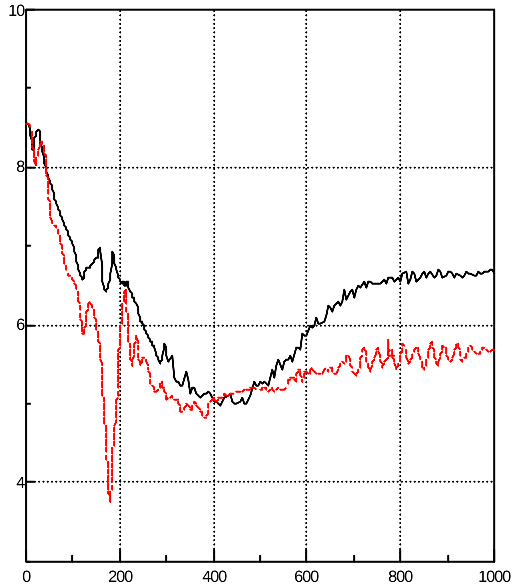


Fig. 6 Core Collapsed Level (m)

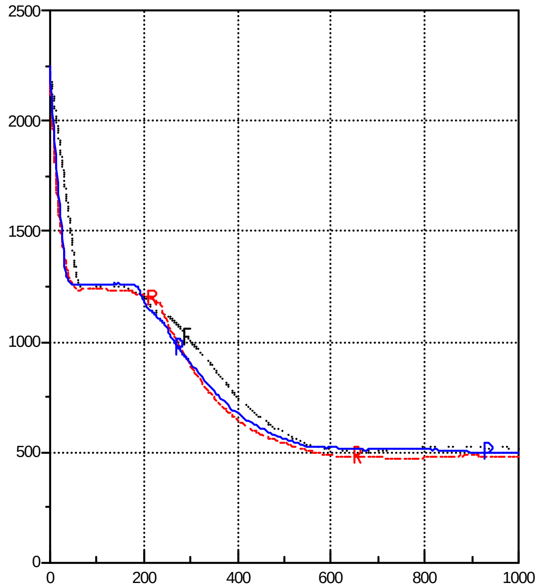
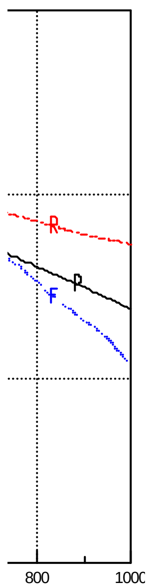


Fig. 8 Pressurizer Pressure (psia)



e (psia)

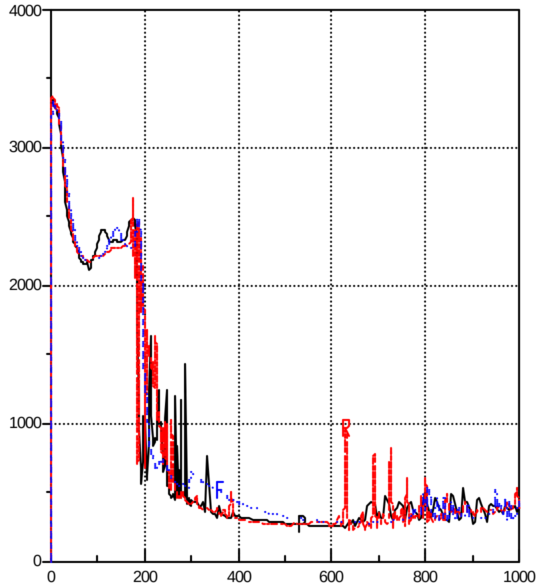


Fig. 10 Break Flow Rate (kg/s)

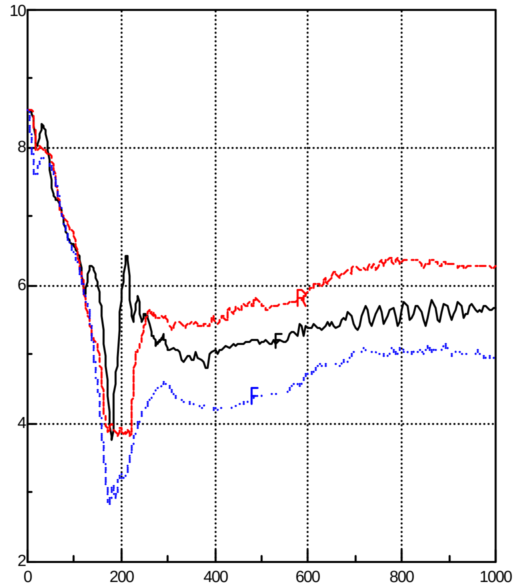


Fig. 11 Core Collapsed Level (m)

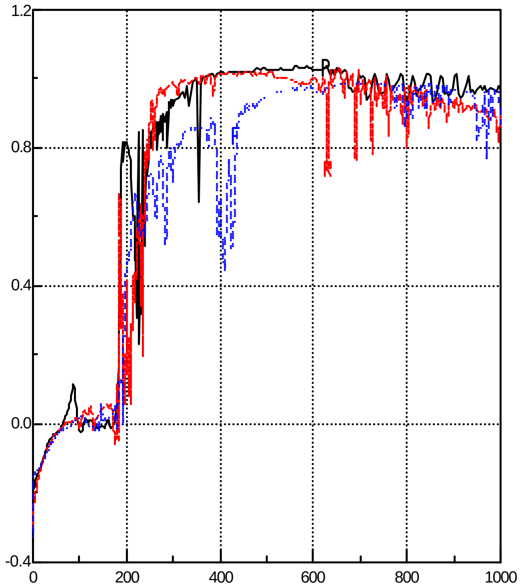


Fig. 12 Break Node Quality

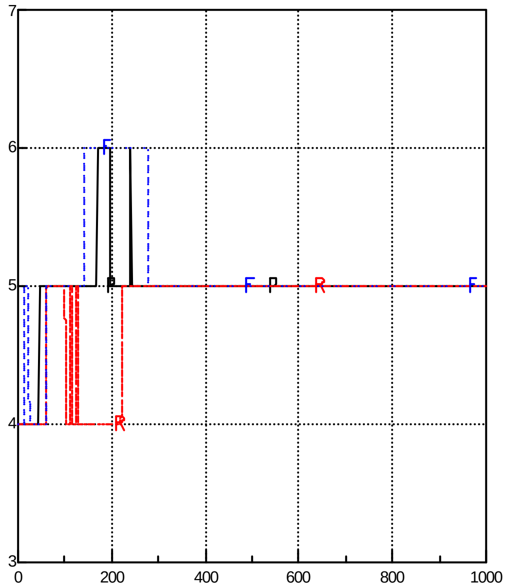


Fig. 13 Flow Regimes at Core Exit

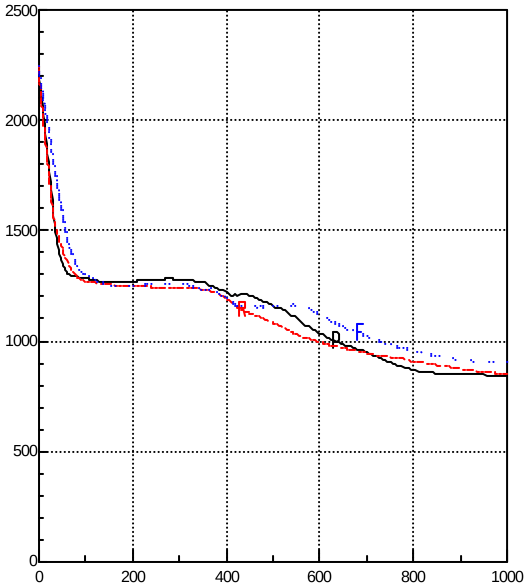


Fig. 14 Pressurizer Pressure

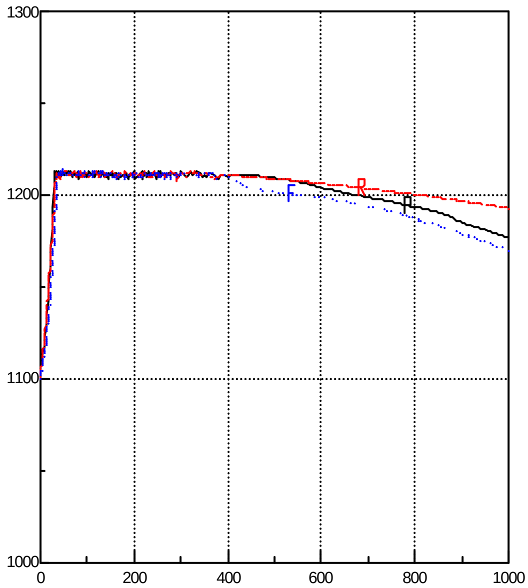


Fig. 15 Steam Generator Pressure

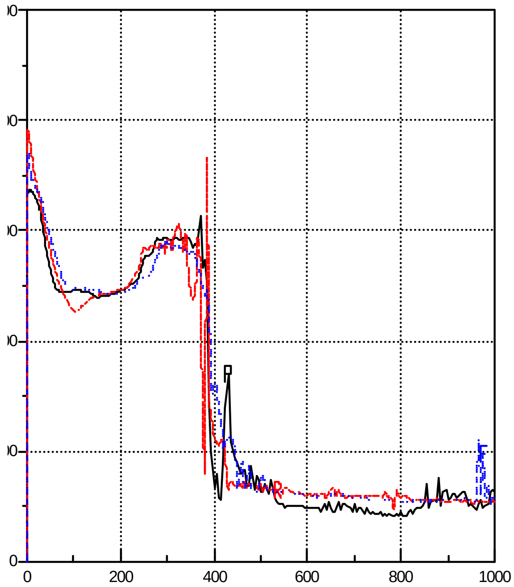
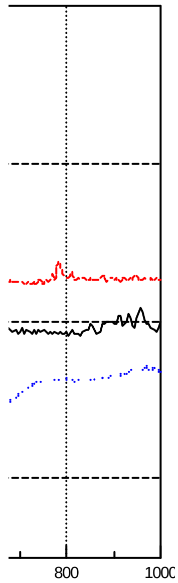


Fig. 16 Break Flow Rate



Level