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# Safety Evaluation on RCS Flow Asymmetry for WH Type NPPs

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## Abstract

A safety evaluation was performed for RCS flow asymmetry for Kori units 1/2/3/4 and Yonggwang units 1/2. The amount of RCS flow asymmetry is based on the design uncertainties and the margin of steam generator tube plugging limit. RCP locked rotor event, steam line break event and feedline break event were reanalyzed and the other design basis events in the current FSAR were evaluated. As a result, RCS flow asymmetry affects RCP locked rotor event considerably. The increase of peak RCS pressure during the RCP locked rotor event is about 30 psi and 20 psi for 2-LOOP plants and 3-LOOP plants, respectively. However, all the safety analysis criteria can be met for RCS flow asymmetry phenomena

## 1.0 Introduction

In recent years, Reactor Coolant System (RCS) loop flow asymmetry has been observed at many pressurized-water reactors, including the Westinghouse-designed plants in Korea. The safety analyses for these plants have not accounted for the possible effects of flow asymmetry. The purpose of this paper is to provide a summary of the evaluation of RCS loop flow asymmetry relative to the licensing-basis safety analyses for the Kori 1, Kori 2, Kori 3&4, and Yonggwang 1&2 units.

## 2.0 Evaluation

By directly or indirectly affecting the initial plant conditions or the transient response, loop-to-loop flow asymmetry could cause more limiting results for some safety analysis events. Though the effects may be characterized as relatively small, in the worst case they may be of a magnitude similar to other effects or changes that are explicitly addressed in the safety analyses. Thus, it is appropriate to evaluate RCS loop flow asymmetry relative to the conclusions of the FSAR.

In order to evaluate the effects on the safety analyses it is necessary to define the magnitude of the loop-to-loop flow asymmetry and the resultant effect on total core flow. As described in the NSAL (Reference1), it is concluded that individual loop flow calorimetric measurements are not sufficiently accurate for use in determining the existence or magnitude of flow asymmetry. Therefore, maximum expected actual loop-to-loop flow differences have been determined via engineering analysis. For both 2 loop and 3 loop Westinghouse plants the maximum expected loop-to-loop flow difference (highest to lowest flow) is less than 2% of nominal loop flow[3], assuming equal steam generator tube plugging (SGTP) in each loop. A SGTP imbalance of 10% (e.g., one loop has 5% plugging while another loop has 15% plugging) would result in an additional 3% loop-to-loop flow asymmetry. Therefore, for this evaluation a maximum loop-to-loop flow asymmetry of 5% is assumed, consistent with a maximum SGTP difference of 10%. For events that are directly affected (i.e., assuming the initial "higher" flow loop is faulted or initially idle), this loop flow imbalance would lead to a maximum core flow reduction of 1.1% of the nominal core flow relative to a symmetric loop flow configuration.

Loop flow asymmetry results in individual loops contributing either more or less of their normal design flow fraction. A key assumption in this evaluation is that the initial total core flow remains at or above the minimum design value assumed in the safety analysis. All sensitivity analyses performed in support of this evaluation used the same safety analysis computer codes as previously described in the FSAR and licensed for these plants.

The following sections evaluate the effects of the above-defined flow asymmetry for the safety analysis events.

## 2.1 Feedwater Malfunction

Excessive heat removal may be caused by a reduction in feedwater enthalpy or by excessive feedwater flow addition in one or more loops. Full RCS flow is maintained for these events, so total core flow would not be directly affected by loop flow asymmetry. However, loop flow asymmetry may indirectly affect the analysis since the transient may be asymmetric (i.e., one loop is affected differently than the other(s)). Should the feedwater enthalpy reduction or feedwater flow increase occur in the steam generator associated with the highest RCS loop flow the primary to secondary system heat transfer may be slightly enhanced in this loop relative to the symmetric RCS flow case.

The case of feedwater enthalpy reduction is bounded by the analysis of the excessive load increase incident, which is a symmetric transient (i.e., each loop is affected the same). This case would continue to be bounded for the slight changes that could be introduced by asymmetric RCS flow conditions. Furthermore, the excessive load increase incident typically shows a large margin to the DNBR limit. Therefore, the conclusions of the FSAR remain valid.

For the case of excessive feedwater flow addition, cases are analyzed at full and zero power conditions. Generic sensitivity analyses have shown that the effect of a 5% flow change in the affected RCS loop had an insignificant effect on the calculated DNBR for the full power case. Similarly, changes in loop flow of this magnitude would have an insignificant effect on the reactivity insertion rate calculated for the zero power case. Therefore, the conclusions of the FSAR remain valid.

## 2.2 Excessive Load Increase

This event is analyzed as a symmetric transient in which all reactor coolant loops are affected the same. Therefore, loop flow asymmetry would not adversely affect the results of the analysis. The conclusions of the FSAR remain valid.

## 2.3 Main Steam System Depressurization

The case of a non-uniform credible steam line break at zero power, i.e., the inadvertent opening of a single steam generator relief valve, may be indirectly affected by loop flow asymmetry since it is an asymmetric transient. However, this case is bounded by the Main Steam Line Break (MSLB) evaluation below. Based on the results of the MSLB case, the conclusions of the FSAR remain valid.

## 2.4 Main Steam Line Break

The MSLB accident analysis results may be indirectly affected by loop flow asymmetry, since it is an asymmetric transient in which one loop is affected differently than the other(s). If the faulted steam generator is in a loop with high flow, then the increased flow could enhance primary to secondary heat transfer in that loop during the transient. For the zero power case the slightly increased cooldown in the loop may lead to a higher moderator reactivity feedback and thus a slightly higher peak nuclear power and core heat flux relative to a symmetric loop flow case. For cases analyzed at full power this will affect the transient response such that reactor trip occurs at a different time, and thus the limiting break size and DNBR result may change.

Sensitivity analyses were performed to determine the effect for the Korean plants. For the zero power case, the key transient result is the peak core heat flux. It was found that for the limiting case the peak heat flux for the asymmetric case increased by less than 0.4% for any of these plants, relative to the symmetric flow case. Figure 2.4-1 shows a comparison transient plot of the core heat flux for Kori 1, which showed the largest difference between the symmetric and asymmetric flow cases. Other transient parameters such as coolant temperature and pressure were also not changed significantly. Therefore, the conclusions of the FSAR remain valid.

For the full power case the key transient result is the peak heat flux for the limiting break size, which is the largest break that does not result in an early reactor trip from the low steam pressure SI actuation signal. The sensitivity analyses for each of the Korean plants showed that the asymmetric flow case resulted in a slightly smaller limiting break size (0.01 to 0.02$ft^2$ smaller) relative to the symmetric flow case, but with a virtually identical peak core heat flux. Figure 2.4-2 shows a comparison plot for Kori 1, showing this effect. The other plants exhibited the same behavior. Other transient parameters were similarly not significantly changed. Thus, the results of the full power MSLB analysis are not adversely affected by asymmetric flow and the conclusions of the FSAR remain valid.

Loop flow asymmetry may also indirectly affect the main steam line break mass and energy releases calculated for use in the containment integrity analysis. The loop power imbalance may affect the initial steam generator masses and steam pressure in the individual loops such that the mass and/or energy releases out the break could increase slightly, thus possibly affecting the containment pressure. To determine the potential effect, a sensitivity analysis was performed for Kori 1. Cases were analyzed assuming both a higher and lower flow in the RCS loop associated with the faulted steam generator. The results showed that there was an insignificant effect on the calculated peak containment pressure. Thus, it is concluded that the assumption of symmetric flow is acceptable and the conclusions of the FSAR remain valid.

### 2.5 Loss of Load / Turbine Trip

This event is analyzed as a symmetric transient in which all reactor coolant loops are affected the same. Therefore, loop flow asymmetry would not adversely affect the results of the analysis. The conclusions of the FSAR remain valid.

### 2.6 Loss of Normal Feedwater / Loss of Offsite AC Power

These events are analyzed as symmetric transients in which all reactor coolant loops are affected the same. Therefore, loop flow asymmetry would not adversely affect the results of the analysis. The conclusions of the FSAR remain valid.

### 2.7 Feedline Break

A main feedwater line break accident may be indirectly affected by loop flow asymmetry since the reactor coolant loops are affected differently. Following the break water level in the secondary-side of the faulted steam generator falls rapidly due to the break flow exiting the steam generator. The safety analysis assumes that all main feedwater to the intact steam generators is completely lost at the time of the break. Thus, level in the intact steam generators also begins to fall. The conservative analysis acceptance criterion is that bulk boiling does not occur in the RCS (i.e.,the hot leg temperature remains subcooled) prior to the time that heat removal capability exceeds the core decay and reactor coolant pump heat addition (transient turnaround). This ensures that the core will not become uncovered, thus maintaining core coolability and avoiding more severe consequences. The actuation of emergency feedwater is required to provide cooling to eventually turn around the RCS heatup. However, an important factor that may have an effect on the ultimate margin to hot leg saturation is the total secondary-side inventory available in the intact-loop steam generators at the time of reactor trip, which occurs before emergency feedwater flow is delivered. A loop flow and steam generator tube plugging imbalance may result in a reduction in the total intact-loop mass inventory at the time of trip relative to a symmetric flow initial condition as a result of the loop-to-loop power imbalance, which affects the initial steam generator masses and subsequent steaming rates prior to reactor trip. Sensitivity analyses have shown that the magnitude of the effect of asymmetric flow depends on how long it takes from the time of the break until reactor trip occurs. This is dependent on the specific steam generator design.

For the Kori 1 plant (Model D60 steam generator), a reanalysis of the previously identified limiting feedline break case showed that the minimum margin to hot leg saturation prior to transient turnaround was reduced to about 4°F for asymmetric RCS flow, versus about 10°F for the symmetric flow case. Figure 2.7-1 shows a comparison of the symmetric and asymmetric flow hot leg coolant temperature transients versus the saturation temperature. The other Korean units have Model F steam generators, for which the time to reactor trip is much shorter and the effect much less. A sensitivity analysis for Kori 2 showed that there was no change in the minimum margin to hot leg saturation for asymmetric flow conditions. This is shown in the comparison plot of the limiting Kori 2 case (offsite power not available) presented in Figure 2.7-2. Thus, the conclusions of the FSAR or previous analysis remain valid.

## 2.8 Loss of Forced Reactor Coolant Flow

The limiting loss of flow case is a complete loss of flow in all reactor coolant loops, as a result of either loss of power supply to all reactor coolant pumps or frequency decay. This case is unaffected by RCS loop flow asymmetry since the transient is symmetric, i.e., all loops are affected the same. The total core flow transient will be the same as for the symmetric flow case. The conclusions of the FSAR remain valid.

The partial loss of flow case, in which one reactor coolant pump coasts down, could be directly affected by asymmetric loop flow. In this case, the resultant core flow during the transient would be slightly lower than in the symmetric flow case if the faulted loop initially has higher flow than the other loop(s). However, the DNBR in this case would still be bounded by the more limiting complete loss of flow case, which is not affected by flow asymmetry (see above). Therefore, the DNB design basis would still be met and the conclusions of the FSAR remain valid.

## 2.9 Locked Rotor / RCP Shaft Break

In a Locked Rotor or RCS Shaft Break event the faulted loop flow is lost instantaneously. The unfaulted loop(s) continue providing flow to the core in the crucial few seconds following the accident. This event could be directly affected by asymmetric loop flow. If the locked rotor occurs in a flow loop that initially has higher flow than the other loop(s) the resulting core flow at the limiting point in the transient will be lower than it would be if each loop initially provided an equal flow contribution to the core. This leads to more limiting safety analysis results for the maximum RCS pressure, peak fuel clad temperature, and the percentage of fuel rods in DNB (i.e., the rods that may experience a DNBR lower than the applicable limit value).

Sensitivity analyses were performed for the Korean plants in order to determine the effect of asymmetric flow on the peak RCS pressure. It was found that the calculated peak pressure increased by 33, 31, and 23psi for the Kori 1, Kori 2, and Kori 3&4/ Yonggwang 1&2 analyses, respectively. For the 2loop Kori 1 and Kori 2 plants the peak pressures in these cases were higher than 3000 psia (120% of design RCS pressure). However, these cases include the very conservative assumption that the undamaged pumps begin coasting down instantaneously at the time of turbine trip. When an appropriate small delay of 2.0 seconds is assumed between turbine trip and loss of power to the undamaged pumps, as has been justified for many other plants, the resultant peak RCS pressures are well below 3000 psia. The final conservative peak pressures for these cases were as follows:

| Plant | Peak RCS Pressure (psia) (Asymmetric Flow) |
|---|---|
| Kori 1 | 2963 |
| Kori 2 | 2953 |
| K3&4, Y1&2 | 2758 |

Transient plots for these plants, showing the results for symmetric flow and for asymmetric flow with both a zero and 2.0 second pump trip delay, are presented in Figures 2.9-1, 2.9-2, and 2.9-3. Based on these results, the peak pressure is well below that which would cause the faulted condition stress limits to be exceeded and the conclusions of the FSAR remain valid.

For the peak clad temperature result, it is noted that the current conservative analysis result for the most limiting plant (Kori 1) has over 450°F margin to the 2700°F limit value. From sensitivity analyses performed for other plants the effect of asymmetric loop flow is much less than 100°F. Therefore, an explicit analysis for this criterion was not performed. The conclusions of the FSAR remain valid.

For Kori 1 and Kori 2 a rods in DNB calculation is not performed since the radiological dose analysis conservatively assumes that all of the rods in the core fail for a locked rotor event. For Kori 3&4/ Yonggwang 1&2 an analysis was performed to determine the effect of asymmetric loop flow. The results showed an increase of approximately 3.5 to 4.5% in the calculated rods in DNB for current fuel cycles of these units. However, the total percentage of rods in DNB remains well within the limit value of 40%. The conclusions of the FSAR remain valid.

## 2.10 RCCA Bank Withdrawal from a Subcritical Condition

The analysis of this event assumes a core flow based on one reactor coolant pump not operating, consistent with the plant Technical Specifications for the Hot Standby plant operating mode (Mode 3). If the idle loop normally provides more flow than the operating one(s) under full-flow conditions then the steady-state core flow fraction may be less than the nominal value assumed in the safety analysis. This would lead to more limiting analysis results for DNBR. This effect of asymmetric loop flow was evaluated for the Korean plants by applying a 1.1% core flow reduction penalty in the DNB analysis. The resultant minimum DNBR calculated for each plant was still above the applicable limit value. Thus, the DNB design basis is met and the conclusions of the FSAR remain valid.

### 2.11 RCCA Bank Withdrawal at Power

This event is analyzed as a symmetric transient in which all reactor coolant loops are affected the same. Therefore, loop flow asymmetry would not adversely affect the results of the analysis. The conclusions of the FSAR remain valid.

### 2.12 RCCA Misoperation

These events, including the dropped rod transient, are analyzed as symmetric transients in which all reactor coolant loops are affected the same. Therefore, loop flow asymmetry would not adversely affect the results of the analysis. The conclusions of the FSAR remain valid.

### 2.13 Startup of an Inactive Reactor Coolant Loop

This transient is asymmetric since one reactor coolant loop is affected differently than the other one(s). However, the effect of asymmetric loop flow on the transient results would be insignificant. There is considerable margin to the DNBR limit for this transient. In addition, the event is precluded from happening as analyzed by the plant Technical Specifications, which require that all reactor coolant pumps be in operation at power. Thus, the conclusions of the FSAR remain valid.

### 2.14 Boron Dilution

No specific assumptions are made regarding the RCS loop flows in the calculation of the analysis results for the boron dilution transient. The key analysis results depend upon total RCS volume, not flow rate. Therefore, the analysis is unaffected by loop flow asymmetry and the conclusions of the FSAR remain valid.

### 2.15 RCCA Ejection

Rod ejection is analyzed for both full power and zero power cases. The full power cases assume full RCS flow and all loops are affected the same during the transient. Therefore, loop flow asymmetry would not adversely affect the results of the full power analysis cases.

The zero power cases assume a core flow based on one reactor coolant pump not operating, consistent with the plant Technical Specifications for the Hot Standby plant operating mode (Mode 3). If the idle loop normally provides more flow than the operating one(s) under full-flow conditions then the steady-state core flow fraction may be less than the nominal value assumed in the safety analysis. The maximum core flow reduction would be less than 1.1% of nominal core flow. Sensitivity analyses performed in support of the Westinghouse Rod Ejection topical report (Reference 2) found that changes in core flow of 10% have an insignificant effect on the important analysis results for this event. Therefore, the applicable analysis criteria would still be met for asymmetric flow conditions, and the conclusions of the FSAR remain valid.

### 2.16 Inadvertent Operation of ECCS During Power Operation

This event is analyzed as a symmetric transient in which all reactor coolant loops are affected the same. Therefore, loop flow asymmetry would not adversely affect the results of the analysis. The conclusions of the FSAR remain valid.

### 2.17 RCS Depressurization (Inadvertent Opening of a Pressurizer Relief Valve)

This event is analyzed as a symmetric transient in which all reactor coolant loops are affected the same. Therefore, loop flow asymmetry would not adversely affect the results of the analysis. The conclusions of the FSAR remain valid.

### 2.18 Steam Generator Tube Rupture

The steam generator tube rupture (SGTR) thermal and hydraulic analysis is performed to calculate the amount of primary to secondary break flow and the amount of steam released to the environment. These outputs of the thermal and hydraulic analysis are used along with the initial coolant activities to calculate the radiological consequences of the event.

The flow path of the primary to secondary break is from the RCS into a single steam generator. However, the SGTR transient is essentially symmetric due to the rather small break area of a single steam generator tube. Reactor trip and safety injection signals are generated due to the RCS depressurization. A loss of offsite power is conservatively assumed concurrent with reactor trip and the reactor coolant pumps coastdown and natural circulation in the RCS occurs.

There are potential indirect impacts on the SGTR analysis from the difference in initial conditions in the steam generators, resulting from the flow asymmetry. (1) The initial steam generator pressure could be slightly lower in the ruptured steam generator when the flow asymmetry is considered. This could result in a small increase in break flow during the period until reactor trip. After reactor trip, the steam generator pressure increases rapidly to the main steam safety valve setpoint pressure and the asymmetry would have no further impact on the break flow. The slightly higher break flow would result in a slightly earlier reactor trip. The net impact could be an increase in the total break flow. This increase would be very small and have an insignificant impact on the calculated offsite doses. (2) The flow asymmetry could also have another potential indirect impact on the SGTR by changing the mass of fluid initially in each of the steam generators. This could result in a small increase in steam released from the ruptured steam generator, as there is less water available to absorb decay heat following reactor trip. This increase would have an insignificant impact on the calculated offsite doses.

It is concluded that although a flow asymmetry may have an indirect impact on the SGTR analysis the impact on the offsite doses would be negligible. The conclusions of the FSAR remain valid.

### 2.19 Loss of Coolant Accidents

An evaluation has been made by Westinghouse that concludes that the results of the Large Break and Small Break Loss of Coolant Accident (LOCA) analyses are not sensitive to small amounts of loop flow asymmetry on the order of those determined to be credible for Westinghouse plants. This evaluation assumes that the total core flow assumed in the analysis is met. Thus, the conclusions of the FSAR remain valid.

For the Large Break LOCA mass and energy release calculations used in the containment integrity analysis, the releases are predominantly controlled by the initial RCS temperature conditions that exist at the initiation of the event. Sensitivity analyses for variations in RCS flow have shown a negligible effect on the severity of the mass and energy releases. Considering that the average RCS temperature remains bounded by the average loop temperature when asymmetric RCS loop flow conditions exist, the current design basis analyses for LOCA mass and energy releases to the remain valid. The conclusions of the FSAR remain valid.

## 3.0 Conclusions

Based on the individual evaluations summarized above, it is concluded that the effects of RCS loop flow asymmetry would not have a significant adverse effect on results of the plant safety analyses. In all cases it was determined that the conclusions of the FSAR remain valid. Thus, all safety criteria are met and RCS loop flow asymmetry does not present a safety concern for the Korean plants.

## References

[1] NSAL-00-008, Reactor Coolant Loop Flow Asymmetry, May 22,2000.

[2] WCAP-7588, Rev.1A, An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, January 1975.

[3] TM.99NS05.P2001.090, Evaluation of the RCS Loop Flow Imbalance, KEPRI, 3/01

**Figure 2.4-1  Kori 1 Main Steam Line Break (Zero Power)**

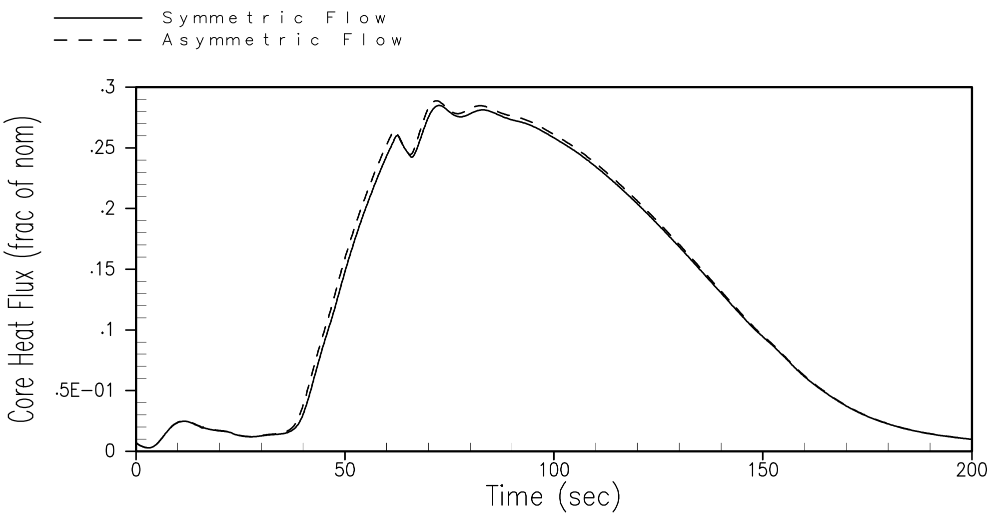


**Figure 2.4-2  Kori 1 Main Steam Line Break (Full Power)**

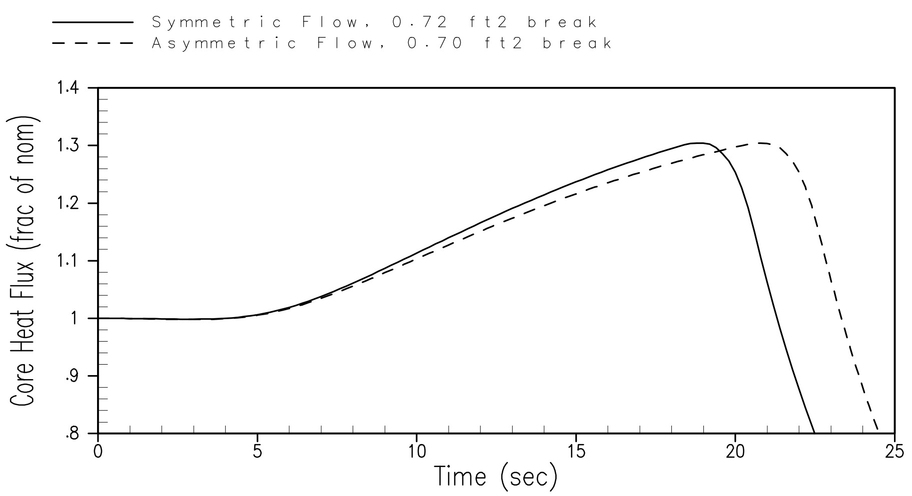

Figure 2.7-1 Kori 1 Feedline Break ( Offsite Power Available )

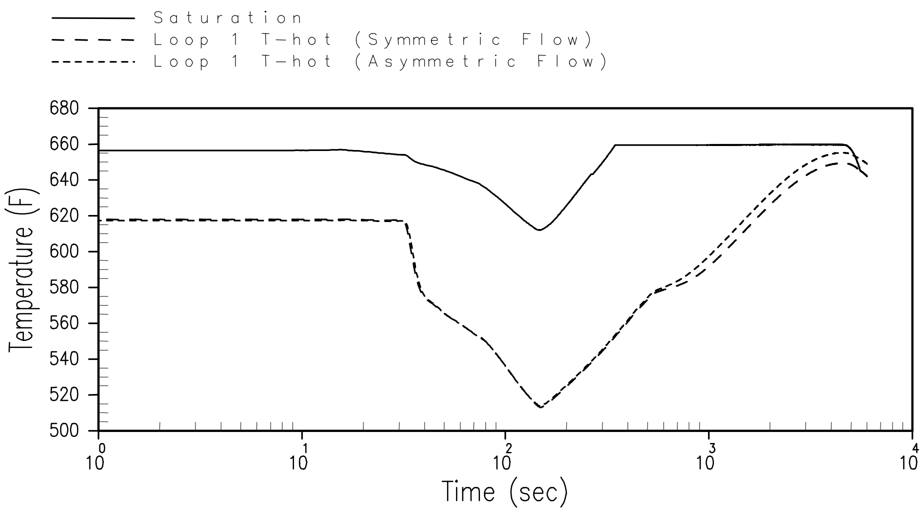


Figure 2.7-2 Kori 2 Feedline Break ( Offsite Power Not Available )

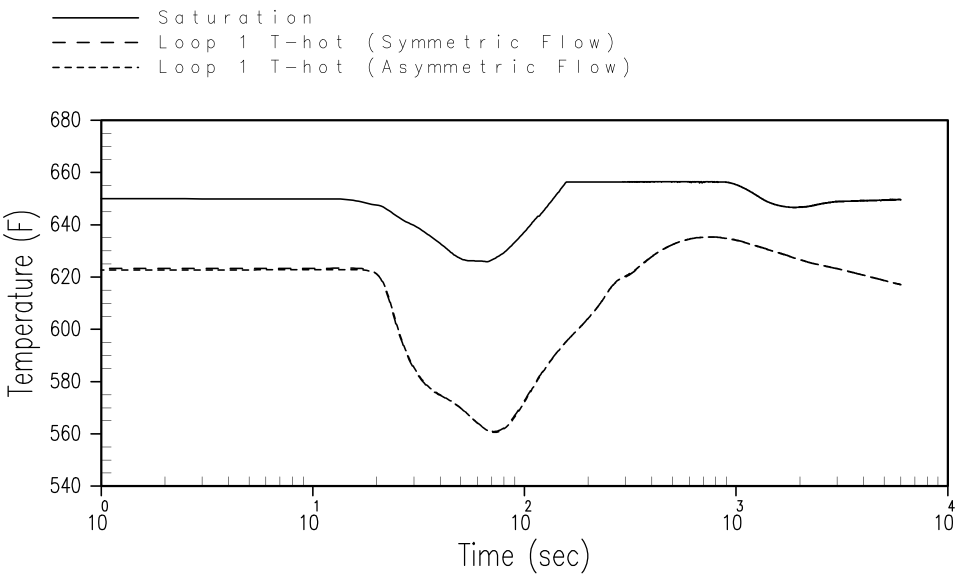

Figure 2.9-1   Kori 1 Locked Rotor/Shaft Break

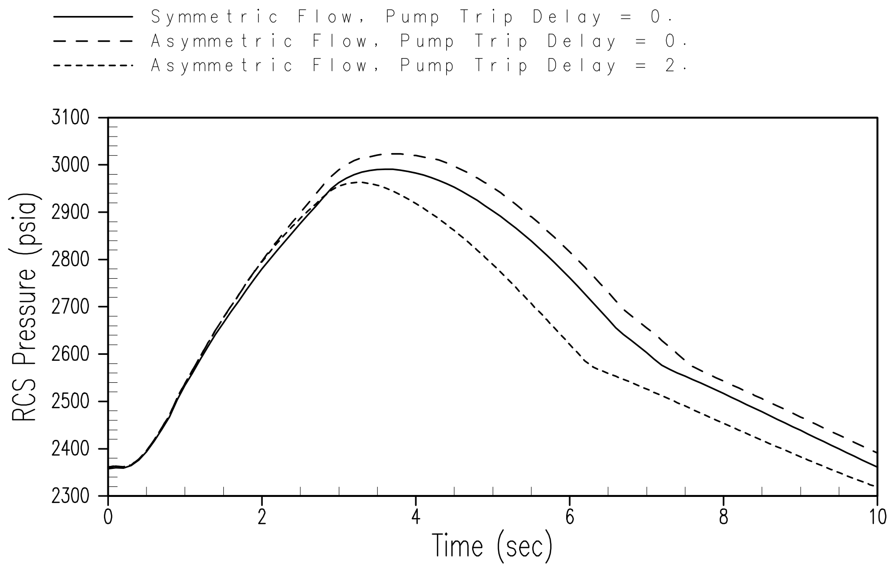


Figure 2.9-2   Kori 2 Locked Rotor/Shaft Break

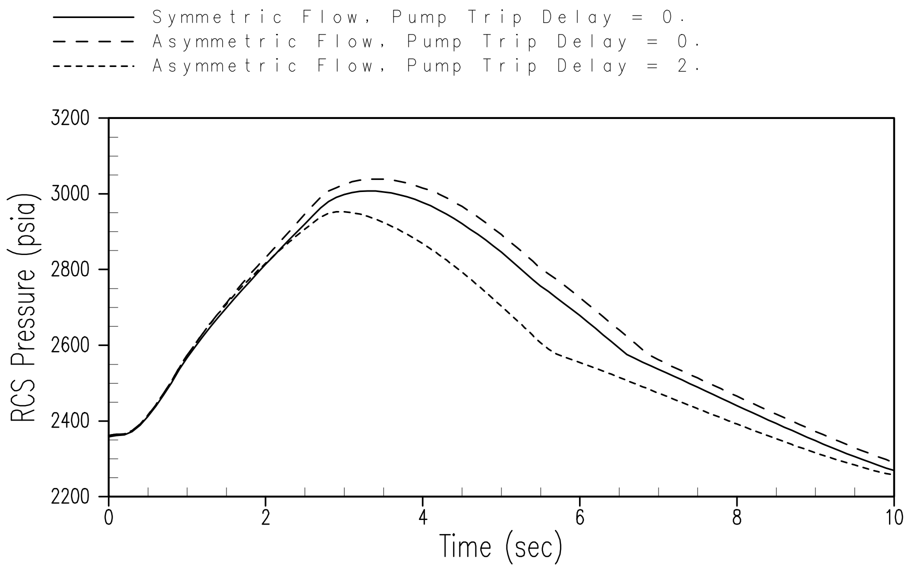

Figure 2.9-3 Kori 3&4, Yonggwang 1&2 Locked Rotor/Shaft Break