Development and Application for the Main Feedwater Line Break Mass and Energy Releases Analysis Methodology with RETRAN-3D Code

Dong-Soo Song, Young-Chan Park, Hwang-Yong Jun
Koerea Electric Power Research Institute
103-16 MunJi-dong, Yusong-Gu, Daejeon, KOREA, 305-380

1. Introduction

The analysis for a feedwater line break for outside containment should be performed to justify the structural integrity and equipment qualification in accordance with revision 1 of Reg. Guide 1.89. Rev 1(1984), which is also required as part of obtaining the extended operating license for Kori Unit 2 Nuclear Power Plant. Currently, a feedwater line break M/E analysis methodology is not available.

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown or heatup. Both the cooldown and heatup types of events must be considered. The transient is analyzed to determine the worst set of mass and energy(M/E) releases that impact the EQ aspects of safety related equipment and flooding outside containment. The most limiting single failure in this event is determined by a sensitivity study. The feedwater line break event is analyzed for a full set of power conditions and break sizes.

2. Methodology Development

Analysis methodology of a feedwater line break M/E release is described in this section. The developed methodology is applied to Westinghouse nuclear power plants.

2.1 Methodology

Previously accepted by USNRC safety analysis standards(SAS) existed for the analysis of core response and mass and energy release for the main steamline break along with the core response for a main feedwater line break inside Containment. The additional safety analysis standards for the M/E release of the main feedwater line break were developed by comparing the major differences between the safety analysis standards for the core response[1] and the safety analysis standards for the M/E release for the main steam line break case were compared to establish their major differences.

Table 1 and 2 are shown comparison of each methodology.

Table 1. Comparison of each methodology (1)

	MSLB M/E (SAS ME)[2]	FLB Core (SAS 16.0)[3]	FLB M/E
. Initial Condition			
Power (Mwt)	1913.52 (102%)	1913.52 (102%)	1913.52 (102%)
RCS Flow (kg/sec)	14,849(TDF)	14,849(TDF)	14,849(TDF)
AVG Temp (C)	306.55 + 2.8	306.55 + 2.8	306.55 + 2.8
PZR Pressure (Mpa)	15.513	15.513	15.513
PZR Level(%)	60	60	60
SG Pressure (Mpa)	6.549	6.549	6.549
SG Level (%)	58.81/58.81 (max)	50.0/50.0 (Nom.)	58.81/58.81 (max)
FW Enthalpy (kJ/kg)	959.10 / 959.10	959.10 / 959.10	959.10 / 959.10
FW/Steam Flow(faulted/intact)	Nominal(High/Low)	**Nominal/Nominal**	**Nom.->max/nom.->zero**
Input Break Area Type	Double Ended	Double Ended	Double Ended
. Reactor Kintics			*(Sensitivity)*
MTC	HZP stuck rod(max)	Max	Max
ATF	Max	Max	Max
DOP	Min	Min	Min
BETA	Min	Min	Min
. Control Systems			
Control Rod	OFF	OFF	OFF
PZR Prop. Heater	OFF	OFF	OFF
PZR Backup Heater	OFF	OFF	OFF
Spray	OFF	OFF	OFF

Table 2. Comparison of each methodology (2)

	MSLB M/E (SAS ME)	FLB Core (SAS 16.0)	FLB M/E
. Reactor Trip			
Low RCS Flow	OFF	OFF	OFF
High Neutron Flux	OFF	OFF	OFF
High PZR Water Level	OFF	OFF	OFF
High PZR Pressure	OFF	OFF	OFF
Low PZR Pressure	OFF	OFF	OFF
Lo-Lo SG Water level	OFF	ON	ON
Safety Injection	ON	OFF	OFF
OPDT / OTD	ON	OFF	OFF
. SG Isolation			
FW Isolation	ON / 7.0 sec	ON / 5.5 sec	ON / 5.5 sec
Steam Line Isolation	ON / 10.0 sec	ON / 10.0 sec	ON / 10.0 sec
. Safety System			
Safety Injection	ON / 12.0	ON / 12.0	ON / 12.0
Flow rate	Min	Min	Min
Enthalpy	Max -> Min(Max)	Max	Max
AFW Injection	High	Low	High
Flow rate(faulted/intact)	Table(High/Low)	Low / Low	Table(High/Low)
Enthalpy	Max	Max	Max/Min
PZR PORV & SV	OFF	Min	Min
SG SV	Min	Min	Min

The initial power is assumed to be 102, 70, 30, and 0%. The initial water level in the steam generator is assumed to be +10% above the normal operating value for the narrow range level the normal operating value. The actuation setpoints and delays for the reactor trip and feedwater isolation are assumed to be conservatively large. The break is modeled just inside the subcompartment of main feedwater piping area. A double-ended guillotine break location of the feedwater line is taken as the intermediate location which maximizes blowdown. Credit is taken for the Feedwater Isolation Valve in the faulted loop.

2.2 Nodalization of Feedwater Line

For the feedwater line break M/E analysis, two figures are shown. The figure 1 is feedwater line system description. This is modeled from deaerator to steam generator inlet nozzle. The main feedwater system consists of three booster pumps and three main feedwater pumps along with two intermediate pressure heaters and two high pressure heaters. Normally, only two booster and two main feedwater pumps are operating; however for the current model, all three Main feedwater pumps and three of the booster pumps are assumed to be operating initially. Both of the intermediate pressure heaters and both of the high pressure heaters are assumed to be initially operating. The bypass lines of the pressure heaters are initially closed and remain closed throughout the transient. The water source of the auxiliary feedwater system is the condensate Storage tank. The pressure and temperature in the condensate storage tank are assumed to be 0.1MPa and 48.89 ^{o}C.

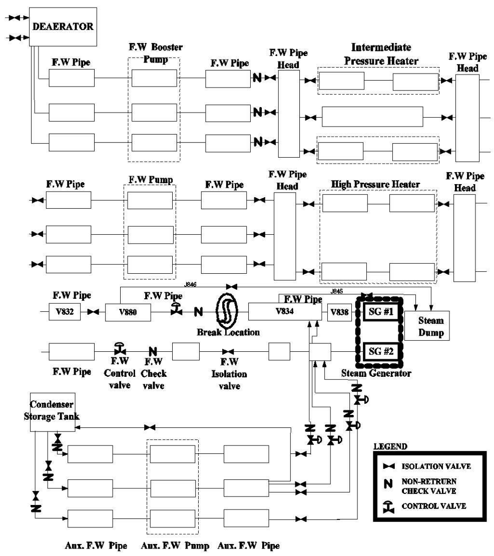

Figure 1. Feedwer line water system nodalization of KORI Unit 2

The RETRAN-3D nodalization of the thermal hydraulic model for the primary and secondary side is shown in Figure 2[4].

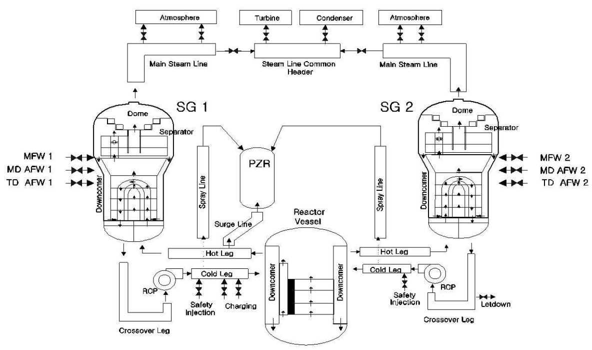

Figure 2. Primary and secondary side nodalization of KORI Unit 2

The thermal hydraulic model of the main feedwater system begins at the deaerator that is assumed to be a constant pressure volume. RETRAN-3D model to analyze feedwater line break consists of 196 volumes, 182 junctions and 3 reactor core heat conductors for primary and secondary side. And to model the reactor control and protection systems, 162 trip cards and 450 control block description cards were used. The steam generators are modeled as 13 nodes, respectively. The steamline was modeled to the turbine stop valve. To consider the effect of the stored energy in RCS, the thick metals were modeled with 43 heat conductors. The models required to represent the reactivity feedback and boron concentration of safety injection flow were implemented by the general data tables and control block cards in RETRAN-3D code.

3. Analysis Result

Figure 3 is shown feedwater line break mass flowrate for each core power level. Following the feedwater line break, blowdown proceeds from both ends of the broken pipe. The M/E release is developed using the Henry-Fauske and Moody critical flow correlations for subcooled and saturated state, respectively. This is done by bounding the RETRAN-3D calculated break point pressure with an enveloping pressure curve. This enveloping pressure curve is applied to determine the enveloping break flow.

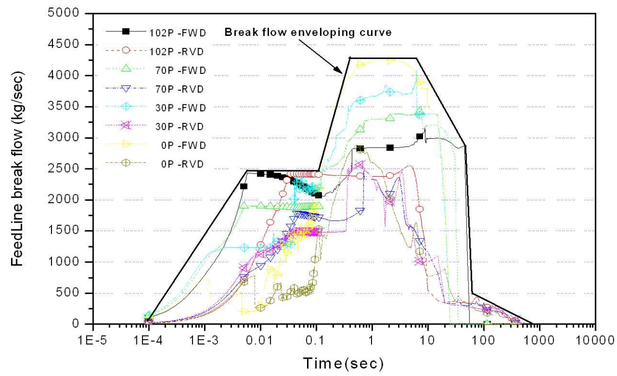

Figure 3. Feed line break mass flowrator of KORI Unit 2

4. Conclusion

The main feedwater line break M/E analysis methodology was developed by comparing the major differences between the safety analysis standards for the core response and the safety analysis standards for the MSLB M/E release. And, throughout spectrum analysis for the core power, a 102% core power level has been found that was the limit case.

REFERENCES

[1] Westinghouse, Steamline Rupture – Core Response, Safety Analyses Standard 12.0, Rev. 3, (1983).

[2] Westinghouse, Mass and Energy Release to Containment Following a Steamline Rupture, Westinghouse Safety Analyses Standard 12.2, Rev. 6, (1996).

[3] Westinghouse, Feedline Rupture, Westinghouse Safety Analyses Standard 16, Rev.5, (1999).

[4] L. J. Agee, RETRAN-3D A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Electric Power Research Institute (EPRI), (1997).