Pre-Study of Off-site Consequence Analysis in Level 3 PSA of Wolsong Unit 1

Won-Jik Kim [a*], Ho-Chang Yang [a], Seong-Soo Choi [a]
[a]ACT, #406, IT Venture Town, 35, Techno 9-ro, Yuseong-gu, Daejeon, 305-510, Korea
[*]Corresponding author: kimwj@actbest.com

1. Introduction

As a post-action after Fukushima accident, Korea Hydro & Nuclear Power Co., LTD(KHNP) develops Severe Accident Management Guidelines considering risk insights, and upgrades Probabilistic Safety Assessment(PSA) model of all of operating Nuclear Power Plant(NPP). KHNP improves the Plant Reliability Data Information System(PRinS) and performs Low-Power and Shutdown(LPSD) PSA for All of NPP. So far, work scope is level 2 PSA, but with reference to NPP life extension, it is necessary that level 3 PSA perform in advance.

In order to perform level 3 PSA, MACCS II [1] (MELCOR Accident Consequence Code System 2) is needed. MACCS II is used in PSA for plants in order to evaluate population dose that is the effects on health and environment caused by released radioisotopes after an accident.

In this study, Steam Generator Tube Rupture (SGTR) event in CANDU-6 plants is evaluated population dose that is the effects on health and environment caused by released radioisotopes after an accident.

2. Methods and Results

MACCS II is used in PSA in order to evaluate population dose that is the effects on health and environment caused by released radioisotopes after an accident. MACCS II is using Gaussian Plume model to evaluate atmospheric dispersion factors.

2.1 MACCS II code structure [1]

MACCS2 is divided into three primary modules: ATMOS, EARLY, and CHRONC. Figure 1 shows the structure of MACCS II code calculation.

First, ATMOS performs calculation about atmospheric transport, dispersion, and deposition. And air and ground concentrations, plume size, and timing information for all plume segments as a function of downwind distance are calculated in the ATMOS. EARLY calculation accounts for consequences due to exposure to radiation in the emergency phase (between 1 and 7 days) of the accident. CHRONC's purpose is to calculate consequences due to exposure to radiation subsequent to the emergency phase of the postulated accident and for computing decontamination and other economic impacts incurred because of the accident.

Output has results of simulation obtained population dose and early fatalities and cancer fatalities, each organs fatalities, and so on.

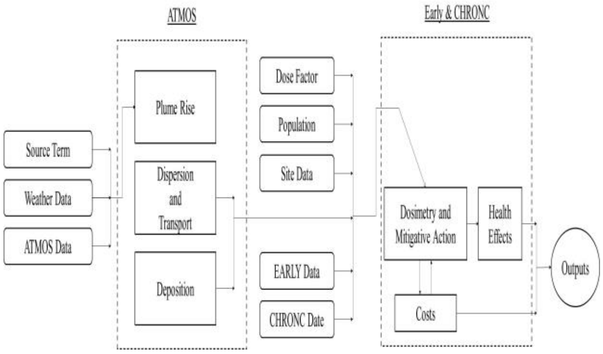

Figure 1. Structure of MACCS II code calculation

2.2 Input data

In order to evaluate population dose, SGTR event was selected as input data. As a part of the reactor building bypass scenario, the SGTR event is the greatest contributor to the Large Early Release Frequency (LEFR).[2]

Figure 2 shows the details of the event tree for the SGTR scenario considered in ISSAC[3]. Figure 2 indicates that the reactor shutdown, steam generator pressure control, and loop isolation are available while the Primary Heat Transport (PHT) D2O make up, SG crash cool-down, and emergency core cooling are unavailable.

MACCS2 requires the population distribution within an 80km radius from the plant site. The population distribution for the Wolsong site obtained from data by Statistics Korea is shown in Table I. The release faction of the SGTR scenario considered in ISSAC was utilized in calculating the population dose. Table II shows the results of the release fractions for 9 radioisotope groups from the SGTR scenario.

It was assumed that 95% of people would evacuate outside of the 16km radius from the plant site while the rest would remain within the 16km radius from the plant site. For a year, site meteorological data measured at Wolsong NPP was generated and then was used for input data. Also, default value was used for ground data and other input data.

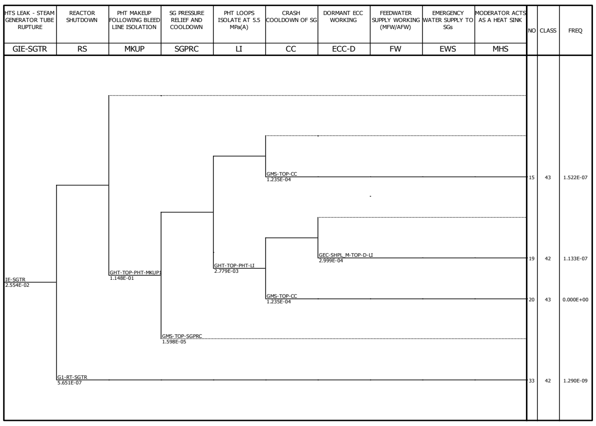

Figure 2. Steam generator tube rupture event tree

Table I: Population distribution for Wolsong site

Wolsong site (person)	0.0-1.6 (km)	1.6-3.2 (km)	3.2-4.8 (km)	4.8-6.4 (km)	6.4-8.0 (km)
	1,073	2,790	1,947	1,493	1,710
	8.0-16 (km)	16-32 (km)	32-48 (km)	48-64 (km)	64-80 (km)
	131,605	1,177,153	690,286	882,494	3,996,236

Table II: Release fraction for isotope group in release

	Xe / Kr	I	Cs	Te	Sr	Ru	La	Ce	Ba
SGTR	9.97 E-01	1.45 E-01	2.93 E-01	2.64 E-02	3.72 E-04	3.24 E-03	5.22 E-05	4.42 E-05	5.88 E-04

2.3Result

The results of this study are summarized as follows :

- Early fatalities and cancer fatalities within a radius of 80 km
- Early and latent population dose within a radius of 80 km

The results of early and cancer fatalities are shown at Table III which indicates that early fatalities for both early and latent show same value. However, for cancer fatalities, the value of early is much less than that of latent. Table IV shows the early and latent population doses.

The results above are obtained by multiplying each population dose (person-rem) by the event frequency (2.10E-07).

Table III: Early fatalities and cancer fatalities within a radius of 80 km

SGTR case	Early Fatalities		
		Mean	99.5%
	Early	4.19E+01	5.35E+02
	Latent	4.19E+01	5.35E+02
	Cancer Fatalities		
		Mean	99.5%
	Early	5.31E+02	2.33E+03
	Latent	2.51E+03	1.11E+04

Table IV: Early and latent population dose within a radius of 80 km

SGTR case	Population dose (person-rem/yr)		
	L-EDEWBODY TOT LIF 0-80.0 km		
		Mean	99.5%
	Early	2.44E-01	1.25E+00
	Latent	1.17E+00	5.00E+00

3. Conclusions

In this study, Steam Generator Tube Rupture (SGTR) event has been evaluated by using Level 1 PSA result and Level 2 PSA result(ISSAC) and MACCS II. As a result, We are obtained the following conclusion.

- Early maximum early fatalities is 5.35E+02 equal to latent maximum early fatalities.(99.5%)
- Early and latent maximum cancer fatalities are 2.33E+03 and 1.11E+04, respectively. (99.5%)
- Early and latent maximum population doses are 1.25 and 5.00 person-rem/yr, respectively. (99.5%)

Other study has shown that MACCS II was performed evaluation for Wolsong NPP.[4] Small Break Loss of Coolant Accident(SBLOCA) event is selected by other study. The results of early and cancer fatalities applied similar assumption were 3.02E+00 and 1.89E+03, respectively. This study’s results are higher than other study’s result. Because, basis input data is different each studies, and event frequency are different (This study : 2.10E-07/ Other study : 4.93E-09). Also, assumption is not exactly same.

As a first step of Level 3 PSA study, uncertainty of input data, source terms, local meteorological data, needs to be performed for more reliable level 3 PSA.

The study results are expected to present useful information in level 3 PSA for CANDU-6 plants.

REFERENCES

[1] USNRC, Code Manual for MACCS2: Volume 1, User's Guide, NUREG/CR-6613 SAND97-0594, May 1998.
[2] KHNP, Probabilistic Safety Assessment Report of Wolsong Unit 1, 2007.
[3] KAERI, ISAAC Computer Code User's Manual TR-3645/2008, 2008.
[4] Ho-Jun Jeon, Moon-Goo Chi, Seok-Won Hwang, Off-Site Consequence Analysis for PWR and PHWR Types of Nuclear Power Plants Using MACCS II Code, Journal of The Korean Society of Safety Vol.26, No.5, September 2011.