Risk Profiles from Internal Events PSA for HANARO Research Reactor

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1. Introduction

HANARO is the 30MW pool-type research reactor for utilizing a neutron source rather than a thermal energy. It can be characterized by operation in atmospheric pressure under the saturation temperature, ultimate core cooling by natural convection with no external power, and so on. Thus, no fuel damage is expected actually in most of postulated initiating events due to the intrinsic safety features for HANARO. According to the requirements of the Citizen Verification Team (2017.4 ~ 2018.3), however, a research project was launched in 2019 to prove that the operating research facilities are fully satisfied with the domestic nuclear safety goals (e.g., less than 0.1% of individual risks) through the risk profile assessment of the research site.

A risk profile for nuclear facilities can be derived from a probabilistic risk assessment (PSA) as a presentation tool to show how risks vary across comparable entities. The risk profiles can be generally expressed in a log-log scale of complementary cumulative density function (CCDF) as a multiplication of off-site release frequency (Level 1&2 PSA results) and population-weighted risk (Level 3 PSA results). In a mathematical meaning, the integral value of the CCDF corresponds to the average individual risk.

The paper focuses on the risk profile based on the level 1/2/3 PSA for internal events at the HANARO research reactor.

2. Development and Results of the Risk Profile for Internal Events of HANARO Research Reactor

2.1 General Mathematical Formulation for PSA

The level 1/2/3 PSA can be simply represented as equation 1, using the terminology of NUREG-1150 [1].

$$SUR_m = \sum_{l=1}^{nSTC} \left\{ f\left(\sum_{k=1}^{nAPB} \sum_{j=1}^{nPDS} \sum_{i=1}^{nIE} IE_i \cdot PDS_{ij} \cdot APB_{jk} \cdot STC_{kl} \right) \cdot cSTC_{lm} \right\} \tag{1}$$

where,
SUR_m = annual single unit risk (per reactor year) for consequence measure *m* (e.g., early fatalities, latent cancer fatalities, etc.),
$f(\cdot)$ = annual release frequency (per reactor year) by the *l*-th source term category (STC_l),
IE_i = *i*-th initiating event,
PDS_{ij} = accident sequences that IE_i will be propagated to the *j*-th plant damage state (PDS_j),
APB_{jk} = accident sequences that PDS_j will result in *k*-th accident progression bin (APB_k),
STC_{kl} = accident sequences that APB_k will be assigned to the *l*-th source term category (STC_l),
$cSTC_{lm}$ = mean (over weather variability, practically) for consequence measure *m* of the *l*-th source term category (STC_l),
$nXXX$ = the number of *XXX* representing *IE, PDS, APB* and *STC,* respectively.

2.2 Determination of Accident Sequences for HANARO Internal Events

First, the level 1 & 2 PSA model for internal events at HANARO facilities was developed in order to obtain the major off-site release accident scenarios and quantify their frequencies:

1) Development of full-power PSA model for internal events at HANARO [2]
2) Qualitative assessment of low power and shutdown PSA model for HANARO (screening-out) [3]
3) Qualitative assessment of HANARO spent fuel pool including bounding thermal-hydrauric analysis (screening-out) [4]
4) Severe accident analysis for HANARO using MELCOR code [3,5]

As a result, the frequency and release characteristics of each major accident scenario included in the risk profile are summarized in Table 1. The release characteristics of radioactive materials by accident type of HANARO facility were divided into five categories:

STC 1) No release (NR),
STC 2) Early ground release (EG),
STC 3) Early release through chimney (EC),
STC 4) Late ground release (LG),
STC 5) Late release through chimney (LC).

According to the MELCOR results, first of all, it should be noted that the core damage accident scenarios defined in the level 1 PSA model do not have a source term release due to no core damage [3]. The release time for all source term categories is assumed very conservatively to be 1 hour after accident occurrence, even though all accident sequences have a lot of time to core damage without any mitigation measures due to the design characteristics of research reactor. According to whether the emergency ventilation system is operated or not, the source term is assumed to be released through chimney or ground (building wall), respectively. The offsite release amount of the source term through chimney or ground is assumed to be same, but the health effect can be different due to air dispersion. It is assumed that the fraction of the chimney release is 10% in this study. The release amount was conservatively determined by the results of MELCOR simulations under the very conservative assumptions that all fission products of the core inventory are released from core to inside reactor building.

Table 1. The Results of Accident Sequences in the Internal Events PSA for HANARO Research Reactor

EVENT	IE	CD Sequence	IE Frequency	L1 CDF	L2 분율	L2 CDF	CD	Early/Late	STC
INTERNAL	%BT-LOCA	#BT-LOCA-2	6.85E-06	4.49E-07	0.90	4.04E-07	O	L	4
	%BT-LOCA	#BT-LOCA-3	6.85E-06	4.49E-07	0.10	4.49E-08	O	L	5
	%BT-LOCA	#BT-LOCA-4	6.85E-06	1.31E-11	0.90	1.18E-11	O	L	4
	%BT-LOCA	#BT-LOCA-5	6.85E-06	1.31E-11	0.10	1.31E-12	O	L	5
	%BT-LOCA	#BT-LOCA-6	6.85E-06	4.50E-14	0.90	4.05E-14	O	E	2
	%BT-LOCA	#BT-LOCA-7	6.85E-06	4.50E-14	0.10	4.50E-15	O	E	3
	%GTRN-AT	#GTRN-AT-3	5.65E+00	1.90E-09	1.00	1.90E-09	X	L	1
	%GTRN-AT	#GTRN-AT-4	5.65E+00	2.56E-08	0.90	2.30E-08	O	E	2
	%GTRN-AT	#GTRN-AT-5	5.65E+00	2.56E-08	0.10	2.56E-09	O	E	3
	%GTRN-MT	#GTRN-MT-3	1.43E+00	4.80E-10	1.00	4.80E-10	X	L	1
	%GTRN-MT	#GTRN-MT-4	1.43E+00	1.70E-12	0.90	1.53E-12	O	E	2
	%GTRN-MT	#GTRN-MT-5	1.43E+00	1.70E-12	0.10	1.70E-13	O	E	3
	%LOCA	#LOCA-2	9.89E-04	1.89E-09	0.90	1.70E-09	O	L	4
	%LOCA	#LOCA-3	9.89E-04	1.89E-09	0.10	1.89E-10	O	L	5
	%LOCA	#LOCA-4	9.89E-04	6.50E-12	0.90	5.85E-12	O	E	2
	%LOCA	#LOCA-5	9.89E-04	6.50E-12	0.10	6.50E-13	O	E	3
	%LOEP	#LOEP-2	1.92E+00	3.68E-06	1.00	3.68E-06	X	L	1
	%LOPCS	#LOPCS-2	6.20E-02	1.19E-07	0.90	1.07E-07	O	L	4
	%LOPCS	#LOPCS-3	6.20E-02	1.19E-07	0.10	1.19E-08	O	L	5
	%LOPCS	#LOPCS-4	6.20E-02	1.40E-09	0.90	1.26E-09	O	E	2
	%LOPCS	#LOPCS-5	6.20E-02	1.40E-09	0.10	1.40E-10	O	E	3
	%LOSCS	#LOSCS-3	6.20E-02	2.08E-11	1.00	2.08E-11	X	L	1
	%LOSCS	#LOSCS-4	6.20E-02	2.81E-10	0.90	2.53E-10	O	E	2
	%LOSCS	#LOSCS-5	6.20E-02	2.81E-10	0.10	2.81E-11	O	E	3
	%RIA	#RIA-3	1.67E+00	5.60E-10	1.00	5.60E-10	X	L	1
	%RIA	#RIA-4	1.67E+00	7.57E-09	0.90	6.81E-09	O	E	2
	%RIA	#RIA-5	1.67E+00	7.57E-09	0.10	7.57E-10	O	E	3
	%SCFB	#SCFB-3	1.30E-05	4.23E-15	1.00	4.23E-15	X	L	1
	%SCFB	#SCFB-5	1.30E-05	2.62E-07	0.90	2.35E-07	O	E	2
	%SCFB	#SCFB-6	1.30E-05	2.62E-07	0.10	2.62E-08	O	E	3

*) source term category: 1(no release), 2(Ground early release), 3(Chimney early release), 4(Ground late release), 5(Chimney late release)

2.3 Modelling and Quantification of Population-Weighted Risk

A site-specific MACCS2[1] input model for HANARO facilities [8] was developed to estimate the health effects of the surrounding population caused by the release of source terms. The results of health effect are usually used by population-weighted risks, i.e., acute fatality (EF) and latent cancer fatality (CF), which are the results of MACCS2 execution.

Table 2. The results of Average Individual Risk for HANARO Internal Events

STC*	Internal CDF(/RY)(a)	population-weighted risk (b)(/person)		Average Individual Risk (c=a*b)(/person-Ry)	
		EF(~5km)	CF(~20Km)	EF(~5km)	CF(~20Km)
1	3.68E-06	0	0	0.00E+00	0.00E+00
2	2.67E-07	0	1.83E-07	0.00E+00	4.88E-14
3	2.96E-08	0	1.83E-07	0.00E+00	5.43E-15
4	5.13E-07	0	1.83E-07	0.00E+00	9.38E-14
5	5.70E-08	0	1.83E-07	0.00E+00	1.04E-14
Total (A=∑a)	4.55E-06	Sum of Individual Risk (C=∑c)		0.00E+00	1.58E-13
Total Average Individual Risk (∑C)				1.58E-13	

*) source term category: 1(no release), 2(Ground early release), 3(Chimney early release), 4(Ground late release), 5(Chimney late release)

2.4 Results of the Risk Profile for HANARO Internal events

In this study, a 5 km radius for EF and 20 km radius for CF were applied around HANARO reactor for population-weighted risk assessment. As a result, the average individual risk for HANARO facilities were evaluated as 1.58e-13/yr as shown in Table 2. Note that no acute fatality was estimated. This figure is comparable to the safety goal reference (0.1% rule), and according to the literature [9] it was reported that the comparative reference was 5e-7/yr for EF and 1e-6/yr for CF. (>> 1.58e-13/yr (negligible)).

Finally, the risk profiles (CF only within 20km) for source term categories of internal events are shown in Figure 1, respectively. Similar shapes of risk profiles (CCDF) for all STC are because most of fission products except noble gas is captured by reactor building wall or chimney with emergency ventilation system.

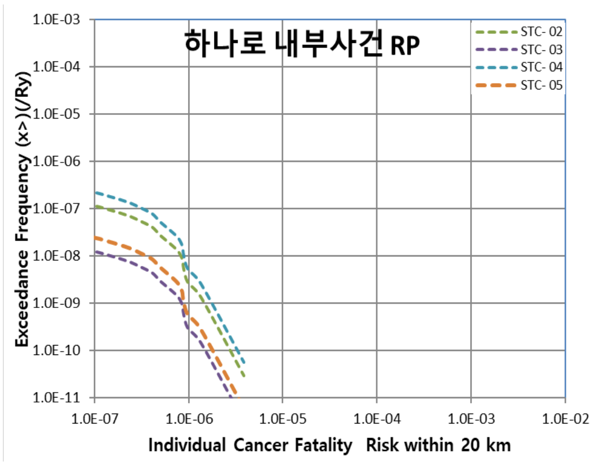

Figure 1. The Risk Profiles for HANARO Internal Events

3. Conclusions

The risk profile for internal events for the HANARO research reactor was developed based on the conservative results of the level 1/2/3 PSA. As a result, the average individual risk for internal events of the HANARO facilities were evaluated as 1.58e-13/yr,

[1] MACCS2 (MELCOR Accident Consequence Code System Version 2) [6,7]

which can be regarded to be insignificant through the comparison on the regulatory-side safety goal reference [9].

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