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Defining Design Limits of a Portable Radiation Dispersion Prevention System

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1. Introduction

The disaster at the Fukushima Daiichi nuclear power plant (NPP) prompted the nuclear industry and the regulators to review the existing weaknesses of the NPPs [1]. To avoid another such accident, many countries now either require or recommend installation of safety systems to delay or to avoid over-pressurization of the containment. One such system gaining popularity in the nuclear industry is called the Containment Filtered Venting System (CFVS). Via controlled releases to the environment, the CFVS theoretically prevents the containment failure from over-pressurization and reduces the chance of having uncontrolled radioactive releases to the public [2].

To the eyes of the general public, however, reducing the chance of such accident is not enough. A typical engineer views a risk as a combination of both consequences and likelihoods, whereas an ordinary person may only consider consequences [3]. The implementations of better regulations, improved human operator actions, and installations of extra safety systems may reduce the chance of having uncontrolled accident practically to zero, yet the public still fears having nuclear reactors. For the nuclear industry to gain public acceptance, it may be essential to develop public relief technologies regardless of likelihood of having such severe accidents. Designs of new comprehensive engineered safety systems to avoid or to lessen the effects of uncontrolled releases in NPPs are being investigated at KAIST [4]. One such barrier system is a portable suction-based radiation dispersion prevention system, called Integrated Portable Suction-Centrifugal Filtration System (IPS-CFS).

To design such systems, detailed information about the radioactive source term at the release point to the environment must be available to draw design limits. The preliminary design limits of the IPS-CFS are presented in this paper. Based on MAAP4 code simulations for an extended station blackout (SBO) accident for a typical 1000 MWe Korean reactor, design limits are developed with conservative assumptions.

2. Conceptual Design of the IPS-CFS

The IPS-CFS is a comprehensive portable suction-based radiation dispersion prevention system. In order to prevent (or at least reduce) the dispersion of the radionuclides into the environment, proposed design must collect the released radionuclides and treat the collected radioactive materials simultaneously. The IPS-CFS is comprised of six major parts: 1) a suction-arm that can be attached to the rupture area both manually and remotely by an operator control, 2) an intermediary container with cooling mechanism to reduce the volumetric flow rate of the incoming radioactive materials by cooling, 3) a centrifugal filtration device to separate the radioactive materials by size to increase treatment capability of the system, 4) a compartment for absorbed particles in liquid and solid phase (larger particles), 5) a compartment for filtered releases of absorbed particles in gas phase (smaller particles), and 6) transportation devices (trucks and trailers) for mobility. A schematic of the IPS-CFS is shown in Figure 1.

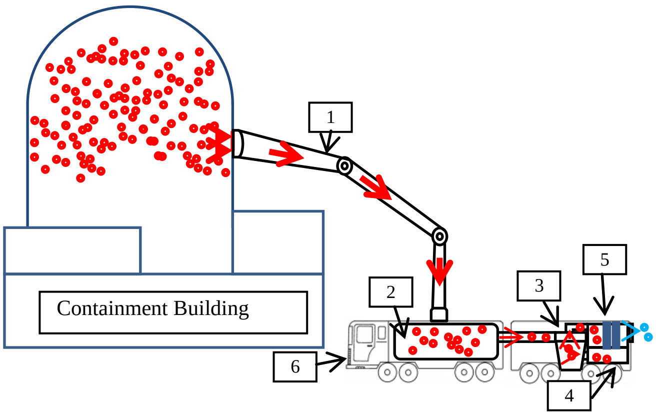

Figure 1. The Configuration of the Integrated Portable Suction-Centrifugal Filtration System

Portability is one of the main requirements considered in designing the IPS-CFS. Huge uncertainty exists in analyzing containment behaviors and predicting containment failure locations in case of severe accidents. Disadvantage of CFVS is that it must be installed inside the containmentt [5]. Having a flexible and mobile radiation dispersion prevention system may lead to the development of a cost-efficient public relief technology. Therefore, the IPS-CFS should have capability to move inside the plant after the exact location of damaged section is detected using unmanned aerial vehicles or other mobile monitoring technologies.

In the following section, preliminary quantitative design limits of the IPS-CFS are derived. Such limits include limits of the temperature of the radioactive release, the mass release rate, the volumetric release rate, and the release speed; they are conservatively derived using the data from the severe accident code simulation for an extended SBO with no recovery of auxiliary feedwater to the steam generators for period of 3 days. Furthermore, dose rates due to the radioactive release were estimated to find recommended values for the decontamination factor so that the dose intake of the workers in operation would not exceed the dose limit set by the United States Nuclear Regulatory Commission (NRC).

3. Methods and Results

3.1 Source Term Simulation using MAAP4.0.6

The simulation of the containment rupture scenario from over-pressurization from extended SBO and loss of auxiliary feedwater system was performed for a 72 hours period, with the plant parameter file that has been provided by Korea Atomic Energy Research Institute for research purposes. According to severe accident management guidelines, the ultimate containment failure pressure for OPR1000 in Korea is 1.01MPa with 5% probability and 95% confidence level [6]. From the containment integrity research at Sandia National Laboratories, a containment rupture is defined as a breach that with representative hole-size of 1.0ft^2, which was used as a reference failure area for a circular containment rupture at the cylindrical wall of the upper compartment [7]. The containment data and the scheme used in this analysis are shown in Table 1 and Table 2.

Table 1. Initial and Boundary Conditions for the Containment

Containment Rupture Failure Pressure	1.01 MPa
Containment Rupture Location	Cylindrical Wall
Containment Rupture Size	9.29 x 10^{-2} m^2 (1 ft^2)
Containment Total Free Volume	77,800 m^3
Reactor Power	2,815 MWt

Table 2. Containment Nodalization and Junction Scheme

#	Corresponding Nodalization	Corresponding Junction
1	Cavity Compartment	Cavity Through Bypass → Lower Compt
2	Lower Compartment	Cavity Through Tunnel → Lower Compt
3	Upper Compartment	Lower Compt → Annular Compt
4	Annular Compartment	Lower Compt → Upper Compt
5	Containment Dome	Annular Compt → Upper Compt
6	Containment Sump	Upper Compt → Dome
7	Auxiliary Building	Annular Compt → Sump Compt
8	Environment	Upper Compt → Environment

For the MAAP4 simulation, following outputs were collected for 72-hours of extended SBO simulation: mass releases and mass fraction releases of each FP group and temperature, pressure, and the density of the gas inside the containment. From the collected data, and the containment rupture size, the radioactive gas volumetric flow rate and the gas release speed were calculated.

The simulation in this report was for the extended SBO accident without auxiliary feedwater system for 72 hours and without any safety system to relieve the containment over-pressurization, which is a very conservative scenario to represent the possible worst-case NPP accident. After the accident initiation, the pressure rapidly builds up inside the containment until the containment rupture occurs about 2 days after the accident initiation, which results in rapid depressurization in first few hours and more stable releases thereafter. Thus, two different time periods during the accident sequences were examined for setting the design limits: 1) the depressurization stage and 2) the stable release stage. The stable stage was defined when the change of rate in containment pressure per hour is less than 1 percent after the depressurization stage. Characteristics of the radioactive release from the simulation are shown in Table 3.

Table 3. Physical Characteristics of the Radioactive Release for the Extended SBO with Loss of Auxiliary Feedwater System

	Depressurization Stage		Stable Stage	
	Max	Min	Max	Min
Temperature (°C)	201.6	169.5	179.4	173.4
Mass Flow Rate (kg/s)	105.2	1.169	0.9732	0.2182
Volumetric Flow Rate (m^3/s)	328.5	3.651	3.039	0.6813
Release Speed (m/s)	226.5	26.17	21.87	7.170

Because such a rapid change of conditions occurs during the depressurization stage and also during first few hours of the stable stage, following design limits for the IPS-CFS were drawn for different stages of the accident. Table 4 shows the durations in which the IPS-CFS must withstand the specified temperature and release flow rates. The limiting values are rounded-up maximum values during the specified time period.

Table 4. Design Limits for the IPS-CFS Simulated for the Extended SBO with Loss of Auxiliary Feedwater System

	Depressurization Stage		Stable Stage	
Time after Release (H)	0~1	1~3	3~5	5~
Duration (H)	1	2	2	Long Term
Temperature (°C)	210	180	180	180
Mass Flow Rate (kg/s)	110	41	1	1
Volumetric Flow Rate (m^3/s)	330	130	5	2
Release Speed (m/s)	230	230	30	15

The results assume that the radioactive release happens at extremely a high pressure, and such an extreme condition lasts only for first few hours after the initial release. Two design challenges for the depressurization stage are 1) designing a suction arm that can withstand initial high momentum of the radioactive releases and 2) designing an intermediary container that can reduce the high volumetric flow rate of the collected releases for the first few hours. The first challenge may be met by designing a suction arm much larger than the anticipated containment rupture size. For the second challenge, the system may use large cryogenic container. However, additional literature review, code simulations, and possible experiments are needed for realistic physical design limits for the IPS-CFS.

3.2 Dose Estimation

By default, MAAP4 code divides fission products (FPs) into 12 different FP groups [8]. The mass of each FP released in each hour were calculated for up to 72 hours from the MAAP4 simulation, which was then used to estimate the radiation dose by calculating the activity of each representative radionuclide and converting the activity to doses. It was conservatively assumed that each FP group consists of radioactive isotopes, listed in Table 5. If several different isotopes of the radionuclides in a FP group exist, the radionuclide with a half-life longer than few days and shorter than 100 years were selected to represent the corresponding FP group for the dose estimation.

Table 5. Information about the Representative Radionuclides

Radionuclide	Half-Life (y)	Decay Constant (s^{-1})	Conversion Ratio with a Point Source Assumption (Bq → mSv/hr)
Cs-137	30.07	7.30946E-10	7.34837E-13
Te-123m	0.3280	6.70220E-08	4.66319E-13
Sr-90	28.78	7.63709E-10	2.60864E-13
Mo-99	0.007527	2.91994E-06	1.05573E-11
Ba-133	10.51	2.09130E-09	3.15959E-13
I-131	0.02198	1.00023E-06	4.56882E-13
La-140	0.004598	4.78073E-06	2.64043E-12
Ce-144	0.7805	2.81598E-08	2.83117E-14
Sb-125	2.758	7.96880E-09	5.27838E-13
U-232	68.90	3.19007E-10	1.50443E-14

For the dose-rate calculation, one must also take into account the distance from the radioactive source as well as the energy of the radioactivity. One cannot simply convert from an activity to a dose rate but rather must do a complicated calculation on an isotope by isotope and a distance by distance basis [9]. Using a program called Rad Pro Calculator, dose rate conversion factors for each radionuclide for 1Bq activity and 10m distance were calculated, included in Table 5. For the distance away from the radioactive source, 10 m was chosen as a reference value because newly administered firefighting vehicle used in Gijang Fire Station has length of 13m as a reference [10]. The estimated factors were then used to calculate the effective dose from the calculated radionuclide activity. The equation to estimate the dose rates from the mass of the FP releases are shown below:

$$D = \lambda \text{ x } (m_i/M_i) \text{ x} 1000 \text{ x } n_{ij} \text{ x } N_A \text{ x } C_{ij} \quad (1)$$

where

D = Estimated dose (mSv/h)
λ = Decay constant (s^{-1})
m_i = Mass released of the fission product i (kg/h)
M_i = Molar mass of the fission product i (g/mol)
n_{ij} = Number of moles of radionuclide ij in a mole of fission product i
N_A = Avogadro's number = 0.6022 x10^{24} (molecules/mole)
C_{ij} = Conversion ratio from activity to dose for defined distance (mSv/h·Bq)

The dose information about each representative radionuclides are shown in Table 6 for the total dose from the release and the dose from the release during the depressurization period.

Table 6. Dose Estimation for each Radionuclide during the Simulation

Radionuclide	Total Dose from the Release (mSv)	Dose from the Release during the Depressurization Period (mSv)	Relative Portion of the Doses from the Release during the Depressurization Period (%)
Cs	1906.585	1727.189	90.59073
I	177253	152634.6	86.11112
Te	3596.904	1757.325	48.85660
Sr	35.77389	33.35865	93.24859
Mo	20397.95	19058.43	93.43306
Ba	5.480482	4.877834	89.00374
La	223769.7	209105.9	93.44691
Ce	19.42319	18.15141	93.45227
Sb	146.5083	83.27252	56.83810
U	0.037215	0.025147	67.57315
Total	427131.4	384423.1	90.00113

As shown in Table 6, about 90 percent of the total dose is contributed to the release during the depressurization stage. So the IPS-CFS should ideally have short preparation time, so that once the radioactive release is detected, it can travel to the exact location within minutes to stop the release. Hourly doses of the released radionuclides after containment failure are shown in Figure 1.

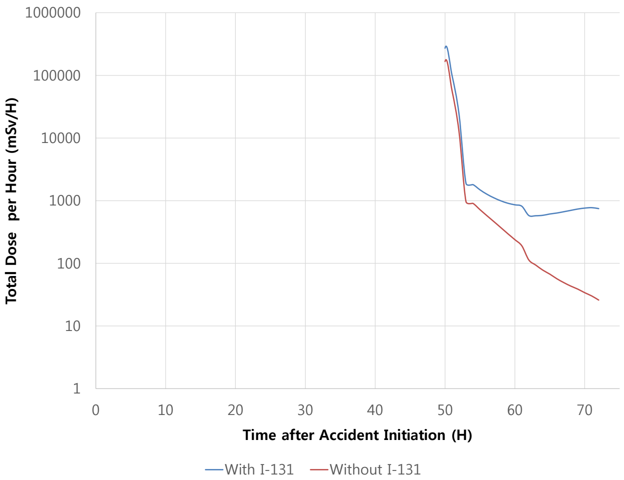

Figure 1. Hourly Dose Rate versus Time

Iodine accounts for most of the doses, so the hourly doses are plotted with and without I-131 for the comparison purpose. If the IPS-CFS can treat the radioactive iodine releases through series of filtration before releasing the treated materials into the environment, it will significantly reduce the radiation dose from the collected radioactive materials.

4. Conclusions

The analysis used in this paper used many conservative assumptions, including accident scenarios, and conservative radioactivity assumptions in defining the limits of the IPS-CFS. It may seem challenging to design a comprehensive radioactive dispersion system that can successfully prevent such extreme accident conditions, especially due to the releases from high pressure. However, as more technologies develop and more realistic source term analyses are performed, it may be possible to develop such a public relief technology. With the development of such technology that can effectively prevent the dispersion of the uncontrolled radioactive releases in case of another Fukushima-like accident, it will result in increased safety of the nuclear power plants for both the public and the workers and may contribute to the increase in the public acceptance of nuclear energy.

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