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■Papers■

DEVELOPMENT OF A FRAMEWORK FOR ASSESSING RADIATION SOURCE TERMS IN NUCLEAR POWER PLANTS

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Abstract - A risk analysis consists of a triplet, <Si, Pi, Xi>, where Si is the scenario identification; Pi is the probability of each scenario; and Xi is the consequences of each scenario. A new computing framework, OMAM (ORIGEN-MAAP4-MAACS), has been developed and applied for assessing the risk of a reference plant as well as radiation source terms using the concept of risk triplet. The result of this study using the OMAM framework presented in this paper, can contribute to producing domestic nuclear power plant's risk data base as well as to establishing severe accident management plans.

INTRODUCTION

The risk is in general involves both uncertainty and some kind of damage that might be received[1]. Symbolically, we can write this as the following Eqn. (1):

$$\text{Risk} = \text{Uncertainty} \times \text{Damage} \quad (1)$$

It is useful to draw a distinction between the ideas of risk and hazard. The risk is relative to the observer because it is subjective thing. Risk is the possibility of injury or damage, whereas hazard is defined as a source of danger[2,3]. A risk analysis, therefore, consists of an answer to the following three questions. "What can go wrong?", "How likely will it happen?", and "If it does happen, what are the consequences?" To answer these questions, it is required to make a list of scenarios as suggested in Table 1. The ith line in Table 1 can be thought as the following triplet of Eqn. (2).

$$R = \{<S_i, P_i, X_i>\}, \quad i = 1,2,3,...N \quad (2)$$

where,

S_i = Scenario identification
P_i = Probability of ith scenario
X_i = Consequence of ith scenario

Risk curve can be obtained by arranging the scenarios in order of increasing severity of damage as given in Eqn. (3). By adding the 4_{th} column of Table 1, (i.e.) the cumulative probability, adding from the bottom, we have the result presented in the 4_{th} column. If the plot points are known, $<X_i, P_i>$, the staircase function can be obtained, which is typically called a risk curve.

$$X_1 \le X_2 \le X_3 \le ...... \le X_N \quad (3)$$

Table 1. A risk triplet with cumulative probability

Scenarios	Probabilities	Consequences	Cumulative Probabilities
S1	P1	X_1	$P_1 = P_2 + P_1$
S2	P2	X_2	$P_2 = P_3 + P_2$
.	.	.	.
.	.	.	.

In this study, a OMAM (ORIGEN-MAAP4-MACCS) framework has been developed and applied for assessing the risk of a reference plant using the concept of the risk triplet described above. Ulchin nuclear power plant unit 3&4 is used as the reference plant for quantifying the radiation risk. Its risk is quantitatively evaluated using the OMAM framework as shown in Figure 1. The IPE (individual plan evaluation) result of the reference plant is utilized for deducing and quantifying accident scenarios[4,5]. The first element of the framework, ORIGEN-S code, is used for estimating the fractions of in-core inventories of radioactive nuclides[6]. And the sum of each nuclides' fraction released to the outside of containment has been calculated by MAAP4 code[7]. Also MACCS code has been run to determine the extent of radioactive nuclides' diffusion to the environment and the consequence of early / latent cancer effects[8].

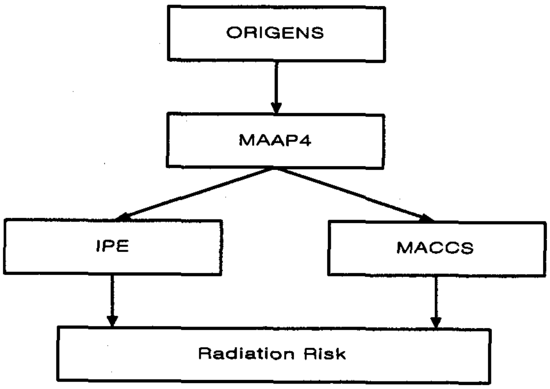


Fig. 1. A OMAM (ORIGEN-MAAP4-MACCS) Framework Developed for Assessing Radiation Source Terms.

IN-CORE INVENTORY CALCULATIONS

ORIGEN-S code calculations have been performed to estimate in-core inventory of radioactive nuclides. ORIGEN-S was originally developed at Oak Ridge National Laboratories for the calculations of fission products, cladding, fuel, and radioactivity behaviors as well as getting data from cross-section library using NITAWL-II and XSDRNPM module[6].

In determining the time dependent nuclide concentrations, ORIGEN-S considers the time rate of change of the concentration for a particular nuclide, Ni, as given in Eqn. (4).

$$\frac{dN_i}{dt} = \sum_j \gamma_{ji}\sigma_{f,j}N_j\phi + \sigma_{c,i-1}N_{i-1}\phi + \lambda_i' N_i' - \sigma_{f,i}N_i\phi - \sigma_{c,i}N_i\phi - \lambda_i N_i \qquad (4)$$

where　i : 1, 2, ... , I,

$\sum_j \gamma_{ji}\sigma_{f,j}N_j\phi$: the yield rate of Ni due to the fission of all nuclides N_j

$\sigma_{c,i-1}N_{i-1}\phi$: the rate of transmutation into Ni due to radioactive neutron capture by nuclide, Ni-1

$\lambda_i' N'$: the rate of formation of Ni due to the radioactive decay of nuclide Ni-1

$\sigma_{f,i}N_i\phi$: the destruction rate of Ni due to fission

$\sigma_{c,i}N_i\phi$: the destruction rate of Ni due to all forms of neutron capture(n,(, n,(, n, p, n,2n)

$\lambda_i N_i$: the radioactive decay rate of Ni

Φ : a space-energy-averaged neutron flux

The core configuration of the reference plant

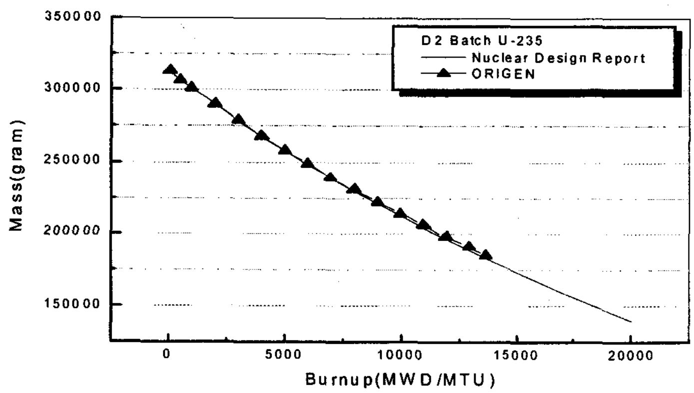


Fig 2. The benchmark calculation result for the D2 batch of the reference with respect to U-235.

typically has nine batches, such as A, B, B1, B2, C, C1, D, D1, and D2 fuel assembly. Each batch has different mass of uranium dioxide as well as enrichment of each fissile material. For verifying the code accuracy of the ORIGEN-S, nine batch inventory calculations for Uranium have been performed and compared with the NDR (Nuclear Design Report) of the reference plant. One of the results is shown in Figure 2, where the fuel burn-up is assumed to be 1 cycle, 371 days (13,661 MWD/MTU). The benchmark calculations show that the ORIGEN-S code for most nuclides is reliable.

CALCULATIONS OF RELEASE FRACTION FOR RADIOACTIVE NUCLIDES

The results of inventory calculations are used as an input data to the MAAP4 and the MACCS code as well. Since those codes have different input formats, they need each unique nuclide data. The twelve fission product species through the ORIGEN-S code are incorporated into the MAAP4 code. The chemical states in the MAAP4 code affects transition rates among solid, liquid, and vapor forms. They are important in determining when a given material forms an aerosol. The physical state also affects the deposition and the retention of fission products. In this code, fission products are released in-vessel through the module of FPRATP, and ex-vessel by the equilibrium model of MEXTONA. Once fission products leave the fuel matrix of in-vessel or core debris of ex-vessel, the chemical state become "frozen" and defined by the species shown below. Equilibrium between vapor and aerosol for these species is determined by the correlations in subroutine GROSSE; Specie (1): Noble gases and radioactivity inert aerosols, Specie (2): CsI + RbI, Specie (3): TeO_2, Specie (4): SrO, Specie (5): MoO_2 , Specie (6): CsOH + RbOH, Specie (7): BaO, Specie (8): La_2O_3 + Pr_2O_3 + Nd_2O_3 +Sm_2O_3 + Y_2O_3, Specie (9): CeO_2 , Specie (10): Sb, Specie (11): Te_2, and Specie (12): UO2 + NpO_2 + PuO_2 .

MAAP4[9] calculates the release fractions of radioactive nuclides using STCs (Source Term Categories)[4], which was regrouped with respect to the similar source term characteristics as shown in Figure 3. The STCs are derived from CETs (Containment Event Trees), which is developed through Level-2 probabilistic safety assessments (PSAs) as well as PDSs (Plant Damage States), which is developed through Level-1 PSAs. The containment failure mode and time associated with the STC-1 include core melt stopped before reactor vessel failure, SIT injection Success, LPSIS injection failre, HPSIS injection success, and re-circulation cooling using CSS success. The containment failure mode and time associated with the STC-2 are reactor vessel failed, containment do not failed, reactor trip success, AFW failure, bleed RCS failure, LPSIS injection success, LPSIS re-circulation success, CTMNT spray injection success, and re-çirculation cooling using CSS success. In the same manner, the containment failure mode and time for STC-3 include early containment failure, leak; early containment failure, rupture for STC-4, etc [10,11].

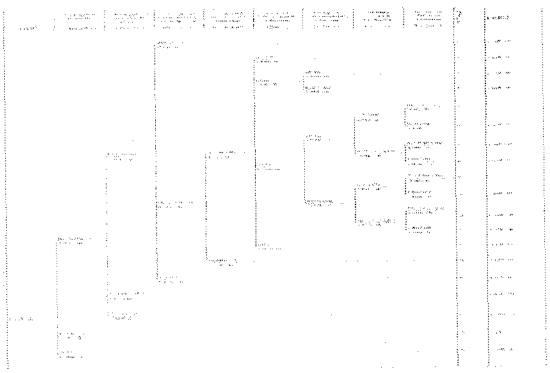


Fig. 3. A Source Term Logic Diagram for the reference plant.

The MACCS (MELCOR Accident Consequence Code System), which is the code that Sandia National Laboratories has developed for the USNRC, calculates the offsite consequence by the amount of an atmospheric release of radioactive nuclides[8]. It allows a release of radioactive materials to the atmosphere to be divided into successive plume segments, which can have different compositions, release time, durations, release height, and energies. It also models seven exposure pathways: exposure to the passing plume, exposure to materials deposited on the ground, exposure to materials deposited on skin (skin deposition), inhalation of materials directly from the passing plume (cloud inhalation), inhalation of materials re-suspended from the ground by natural and mechanical processes, ingestion of contaminated foodstuffs (food ingestion), and ingestion of contaminated water (water ingestion). Health effects are then calculated from doses to specific organs. Doses to specific organs are calculated using dose appropriate conversion factors. Early injuries and fatalities are estimated using nonlinear dose-response models incorporated in the MACCS.

CALCULATIONS OF SOURCE TERMS AND RISK

Using a new computing framework, OMAM, source terms and risk have been calculated for the reference plant as well as radiation source terms. The calculation is based on the concept of risk triplet. Table 2 presents the release fractions of the source terms calculated with respect to each radioisotope nuclides. A time duration of 10 hours and an are of 64km are considered in this study. The off-site dose rate, which is calculated in distance [km] from the center of the reference plant, is shown in Figure 4. The yearly site weather and the population density used are the plant specific data. To determine the risk curve, Eqn. (1) has been be utilized. As explained in previous Section, the STC occurrence probabilities obtained from the level 2 PSA has been multiplied by the consequences, which are 'early fatality' and 'latent cancer fatality'. Figure 5 shows the calculated risks of the early fatality and the latent cancer fatality using the OMAM framework developed in this study. The total risk value of the reference plant is 1.54×10^{-11} /Reactor-Year for the early fatality and 4.08×10^{-5} /RY for the latent cancer fatality, respectively.

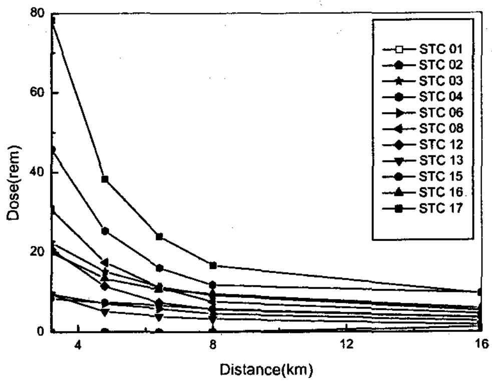
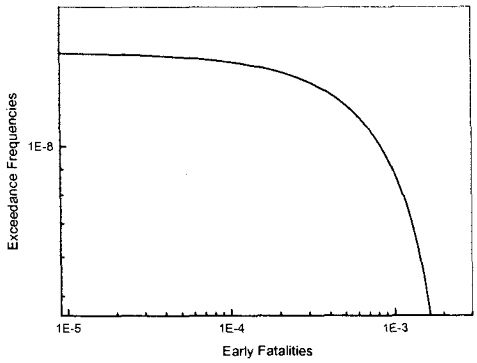


Fig. 4. Calculated off-site dose rate in distance from the center of the reference plant.

Table 2. Source term release fractions calculated through the OMAM framework.

STC	XE/KR	I	CS	TE	SR	LA	CE	BA
1	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
2	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
3	9.646E-01	5.672E-03	1.255E-02	2.088E-02	4.304E-05	2.126E-05	3.795E-05	7.270E-04
4	9.944E-01	2.986E-02	6.733E-02	5.968E-02	2.448E-04	1.209E-04	2.165E-04	4.140E-03
6	0.000E+00	2.004E-03	5.591E-03	0.000E+00	9.569E-05	1.590E-04	1.809E-04	6.950E-04
8	9.081E-01	5.572E-04	3.128E-03	1.103E-01	1.253E-08	2.476E-09	2.249E-10	4.590E-07
10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
12	9.120E-01	2.488E-04	1.432E-03	4.071E-02	8.376E-09	1.668E-09	1.166E-10	3.280E-07
13	0.000E+00	1.571E-04	4.722E-04	4.894E-02	1.237E-08	3.322E-09	1.865E-10	3.820E-07
14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
15	6.296E-02	2.183E-03	1.072E-02	0.000E+00	5.793E-06	6.594E-06	1.019E+00	6.530E-05
16	9.998E-01	5.776E-03	1.789E-02	0.000E+00	6.024E-06	6.503E-06	6.208E-06	7.210E-05
17	9.616E-01	6.416E-02	1.768E-01	3.291E-02	5.174E-04	5.172E-04	6.555E-04	5.120E-03
18	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
19	7.303E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00



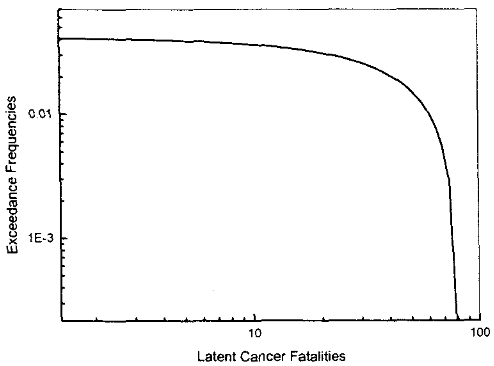


Fig. 5. Calculated risk of the early fatality and the latent cancer fatality, respectively.

CONCLUSIONS

In this paper, a new computing framework, OMAM, has been introduced and applied for assessing the risk of a nuclear power plant as well as radiation source terms using the concept of a risk triplet. This framework demonstrated in this work systematically applicable, flexible and can be applied to any type of nuclear power plants. The results of this study can contribute to producing risk DB (data base) as well as to establish severe accident management plans in all types of NPPs, Korea. For future work, the efforts to reduce the uncertainties in the OMAM framework are required. The uncertainties include model uncertainty, analyst uncertainty, data uncertainty, and completeness uncertainty associated with accident scenarios (S_i).

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