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# Development of Computational Code for Occupational Radiation Dose and Off-Site Dose at Normal Operating Condition of Nuclear Power Plant

Dong-Hoon Shin, Young-Wook Lee, Hyun-Seok Ko and Chang-Sun Kang

*Seoul National University*
*56-1, Shillim Kwanak-gu, Seou 151-742, Korea*

Byung-Soo and Joo-Hyun Moon

*Korea Institute of Nuclear Safety,*
*19 Kusung, Yusung, Taejon 305-338, Korea*

Moon-Soo Park

*FNC Tech. Co.Bldg. 130-201 Seoul National University,*
*56-1, Shillim Kwanak-gu, Seou 151-742, Korea*

## Abstract

An integrated code is compiled, which is for assessing both the occupational radiation dose (ORD) and the offsite radiation dose resulting from the normal operation of nuclear power plant (NPP). The computer code, called INSREC, consists of three sub-modules: source term evaluation, ORD assessment and offsite dose assessment. To evaluate the source term, the new evaluation program, SAEP (Specific Activity Evaluation Program), is developed and incorporated. To estimate the annual ORD, the estimation method in Reg. Guide 8.19 and the database of ORD that is assembled for Kori Units 3 &4 are used. And to assess the offsite radiological consequences, the program is developed and incorporated, whose method is based on those in Reg. Guide 1.109 and 111. For user's convenience, Graphic User Interface is embedded to the computer code, which is made with Visual C++.

## 1. Introduction

Pursuant to the existing practices to assess the occupational or the offsite radiation doses resulting from the normal operation of nuclear power plant (NPP), the uses of the different analysis tools are required. Due to this fact, the hardship arises that the analyst who wants to assess both occupational and offsite doses due to the normal operation of NPP should have good knowledge for and much experiences to run at least several computational systems, and should take a large amount of time for being familiarized with the computational systems. Hence, an integrated and easy-to-use computational code, INSREC is compiled, which is able to assess both ORD and the offsite radiation dose resulting from the normal operation of NPP.

The code of INSREC has several distinct features different from those of the existing codes: First, INSREC includes the source term evaluation program to evaluate the source terms during normal operation, regardless of reactor type; Second, for user’s convenience, INSREC is provided with an easy-to-use graphic user interface that is made with Visual C++; Third, INSREC has the powerful analysis tool to analyze the calculation results, which can show the calculation results in 2-D images or in text mode according to user’s choice.

## 2. Evaluation Methods

### 2.1. Source Term Evaluation

Having some assumptions valid for only a particular reactor type, the existing code such as PWR-GALE is available only for the particular reactor type. Hence, the new model is developed to evaluate the source terms regardless of reactor type. In the model, the components such as reactor vessel, primary coolant system and secondary coolant system are modeled without any assumptions and simplifications. Based on the model, the computer program, SAEP, is developed.

First, it is assumed that the systems consist of two regions with two different nuclides as shown in presented in Fig. 1.

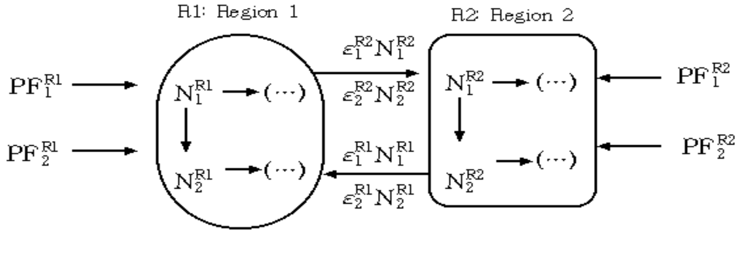


Fig. 1. System Model

The model can be described with systems of first-order differential equations as follows:

[Region 1]

$$\frac{d}{dt}N_1^{R1} = PF_1^{R1} + f_1^{R1}N_2^{R1} - (d_1^{R1} + e_1^{R1})N_1^{R1} + e_1^{R2}N_1^{R2} \tag{1}$$

$$\frac{d}{dt}N_2^{R1} = PF_2^{R1} + f_2^{R1}N_1^{R1} - (d_2^{R1} + e_2^{R1})N_2^{R1} + e_2^{R2}N_2^{R2} \tag{2}$$

where $d_i^{R1} = l_i + CVR^{R1} \cdot \overline{\Phi}^{R1} s_i + R_i^{R1}$

$$f_i^{R1} = l_{j \to i}\, l_i + f_{j \to i}\, CVR^{R1} \cdot \overline{\Phi}^{R1} s_j$$

[Region 2]

$$\frac{d}{dt}N_1^{R2} = PF_1^{R2} + f_1^{R2}N_2^{R2} - (d_1^{R2} + e_1^{R2})N_1^{R2} + e_1^{R1}N_1^{R1} \tag{3}$$

$$\frac{d}{dt}N_2^{R2} = PF_2^{R2} + f_2^{R2}N_1^{R2} - (d_2^{R2} + e_2^{R2})N_2^{R2} + e_2^{R1}N_2^{R1} \tag{4}$$

where $d_i^{R2} = l_i + CVR^{R2} \cdot \overline{\Phi}^{R2} s_i + R_i^{R2}$

$$f_i^{R2} = l_{j \to i}\, l_i + f_{j \to i}\, CVR^{R2} \cdot \overline{\Phi}^{R2} s_j$$

The above linear systems can be expressed in matrix form as follows:

[Region 1]

$$\frac{d}{dt}\begin{pmatrix} N_1^{R1} \\ N_2^{R1} \end{pmatrix} = \begin{pmatrix} PF_1^{R1} \\ PF_2^{R1} \end{pmatrix} + \begin{pmatrix} -(d_1^{R1} + e_1^{R1}) & f_1^{R1} \\ f_2^{R1} & -(d_2^{R1} + e_2^{R1}) \end{pmatrix}\begin{pmatrix} N_1^{R1} \\ N_2^{R1} \end{pmatrix} + \begin{pmatrix} e_1^{R2} & 0 \\ 0 & e_2^{R2} \end{pmatrix}\begin{pmatrix} N_1^{R2} \\ N_2^{R2} \end{pmatrix} \tag{5}$$

$$\frac{d}{dt}N^{R1} = PF^{R1} + A^{R1}N^{R1} + C^{R2}N^{R2} \tag{6}$$

[Region 2]

$$\frac{d}{dt}\begin{pmatrix} N_1^{R2} \\ N_2^{R2} \end{pmatrix} = \begin{pmatrix} PF_1^{R2} \\ PF_2^{R2} \end{pmatrix} + \begin{pmatrix} -(d_1^{R2} + e_1^{R2}) & f_1^{R2} \\ f_2^{R2} & -(d_2^{R2} + e_2^{R2}) \end{pmatrix}\begin{pmatrix} N_1^{R2} \\ N_2^{R2} \end{pmatrix} + \begin{pmatrix} e_1^{R1} & 0 \\ 0 & e_2^{R1} \end{pmatrix}\begin{pmatrix} N_1^{R1} \\ N_2^{R1} \end{pmatrix} \tag{7}$$

$$\frac{d}{dt}N^{R2} = PF^{R2} + A^{R2}N^{R2} + C^{R1}N^{R1} \tag{8}$$

Let the original system with two regions and two different nuclides be considered as the system with one region and four different nuclides. Then, the governing equations are modified as follows;

$$\frac{d}{dt}\begin{pmatrix} N_1^{R1} \\ N_2^{R1} \\ N_1^{R2} \\ N_2^{R2} \end{pmatrix} = \begin{pmatrix} PF_1^{R1} \\ PF_2^{R1} \\ PF_1^{R2} \\ PF_2^{R2} \end{pmatrix} + \begin{pmatrix} P & f_1^{R1} & e_1^{R2} & 0 \\ f_2^{R1} & Q & 0 & e_2^{R2} \\ e_1^{R1} & 0 & R & f_1^{R2} \\ 0 & e_2^{R1} & f_2^{R2} & S \end{pmatrix} \begin{pmatrix} N_1^{R1} \\ N_2^{R1} \\ N_2^{R1} \\ N_2^{R2} \end{pmatrix}$$

where

$$\begin{aligned} P &= -(d_1^{R1} + e_1^{R1}) \\ Q &= -(d_2^{R1} + e_2^{R1}) \\ R &= -(d_1^{R2} + e_1^{R2}) \\ S &= -(d_2^{R2} + e_2^{R2}) \end{aligned} \tag{9}$$

Hence, Eq. (9) can be written briefly in matrix notation.

$$\frac{d}{dt}\begin{pmatrix} N^{R1} \\ N^{R2} \end{pmatrix} = \begin{pmatrix} PF^{R1} \\ PF^{R2} \end{pmatrix} + \begin{pmatrix} A^{R1} & C^{R2} \\ C^{R1} & A^{R2} \end{pmatrix} \begin{pmatrix} N^{R1} \\ N^{R2} \end{pmatrix}$$

$$\boldsymbol{\frac{d}{dt} N = PF + AN \Rightarrow \dot{X} = AX + B} \tag{10}$$

If the equation of (9) is extended to a general system with M regions, the governing equation can be modified as follows:

[<u>Change rate of nuclide in region m</u> ( $\dot{X}_m$ )] = [<u>Production & removal in region m</u> ( $A_m X_m$ )] + [<u>Outside source term</u> ( $B_m$ )] + [<u>Production transferred to other region</u>]

( $\sum_{j \neq m} C_{j \to m} X_j$ )

$$\dot{X}_1 = A_1 X_1 + B_1 + \sum_{j \neq 1} C_{j \to 1} X_j$$

$$\dot{X}_2 = A_2 X_2 + B_2 + \sum_{j \neq 2} C_{j \to 2} X_j$$

$$\vdots \qquad \vdots \qquad \vdots$$

$$\dot{X}_M = A_M X_M + B_M + \sum_{j \neq M} C_{j \to M} X_j$$

The above equations are simply arranged in matrix form as follows;

$$\begin{pmatrix} \dot{X}_1 \\ \dot{X}_2 \\ \vdots \\ \dot{X}_M \end{pmatrix} = \begin{pmatrix} A_1 & C_{2\to1} & \cdots & C_{M\to1} \\ C_{1\to2} & A_2 & \cdots & C_{M\to2} \\ \vdots & \vdots & \vdots & \vdots \\ C_{1\to M} & C_{2\to M} & \cdots & A_M \end{pmatrix} \begin{pmatrix} X_1 \\ X_2 \\ \vdots \\ X_M \end{pmatrix} + \begin{pmatrix} B_1 \\ B_2 \\ \vdots \\ B_M \end{pmatrix} \Rightarrow \dot{X} = AX + B \qquad (11)$$

where, $\dot{X}$ is a [N×M]×1 column vector representing radioactivity, $\boldsymbol{A}$ is a [N×M]×[N×M] matrix including production and removal terms of fission products and transfer information of each region, and $\boldsymbol{B}$ is [N×M]×1 column vector representing the outside source term.

Since Eq. (11) has a stiffness, it is could be calculated using some numerical solvers such as Gear-type routines.

## 2.2. Occupational Dose Assessment

The method to estimate the annual occupational radiation dose is based on that in Reg. Guide 8.19. To estimate the occupational doses for the radiation job procedures, the ORD database that includes the past ORD data for a total of 72 maintenance job procedures over the past 10 years (1986-1995) at Kori unit 3&4 is developed and incorporated into INSREC.

A total of 4,335 maintenance jobs were occurred over the past 10 years. The variables relevant to ORD data assembled in the database are unit number, job date, main job, detailed job, job description, radiation dose rate, job manpower, job time, maximum individual dose and collective dose.

The logic structure of the database, as shown in Fig. 2, consists of three parts: data input; data management and calculation; and data output. The statistical analysis tool is embedded into the module of data management and calculation.

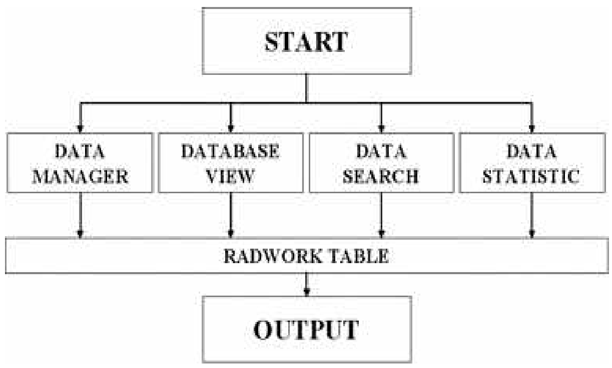


Fig. 2. Database Logical Structure

### 2.3. Offsite Dose Assessment

Prior to assessing the offsite dose, it is necessary to assess the environmental transport behaviors of the radioactive releases from NPP. Regarding the environmental transport assessment for the airborne radioactive releases, the method in Reg. Guide 1.111 is applied. The method of assessing the offsite dose is based on that in Reg. Guide 1.109. Hence, in this paper, it is assumed that the contributors to the resultant offsite dose are dose from noble gases, external dose due to the radioactive materials deposited to the ground, dose from inspiration of radionuclides in atmosphere, and dose from ingestion through food chain.

## 3. Computer Code

Based on the evaluation methods in Section 2, the computer program, INSREC, is developed to evaluate the annual ORD and the annual offsite dose resulting from the normal operation of NPP. The code structure is shown in Fig. 3. INSREC is provided with an easy-to-use GUI (Graphic User Interface) that is made with Visual C++.

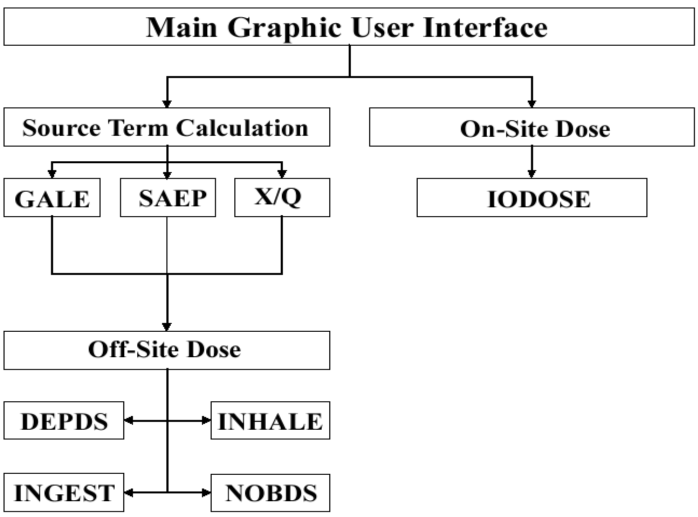


Fig. 3. Computer Code Structure

The main GUI of INSREC is shown in Fig. 4. The main purpose of GUI is to provide a user-friendly and easy-to-use environment. On the screen of GUI, a user can choose any option.

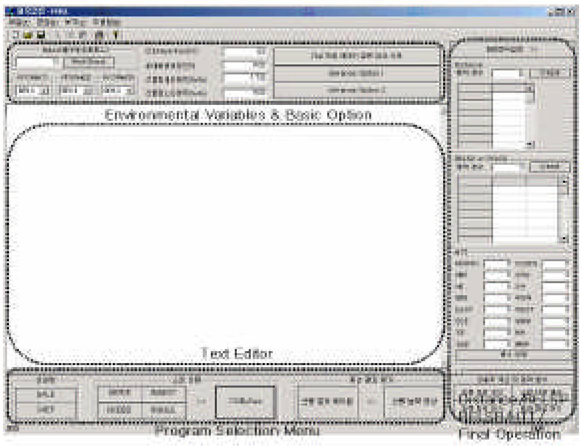


Fig. 4. Initial Screen of Main GUI

Fig. 5 illustrates the GUI of the source term evaluation program, SAEP. The major inputs are plant specific data such as failed fuel ratio, escape rate, reactor coolant weight, clean-up rate and so on. Using SAEP, the activity of core, primary system and secondary system could be evaluated.

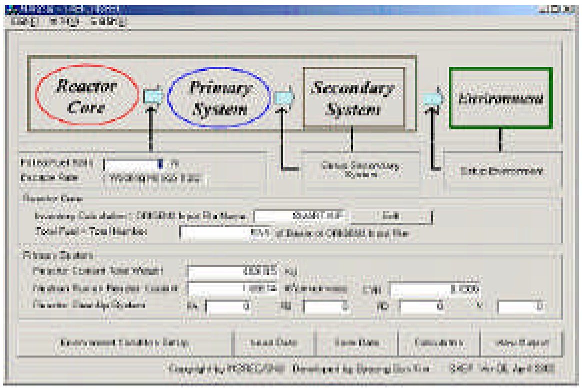


Fig. 5. Initial Screen of SAEP

The annual ORD can be assessed by selecting the menu of IODOSE. Fig. 6 shows the initial screen of the program of IODOSE. This program consists of three parts: data output box, database table and database statistical tool. Many useful data such as an average, maximum, minimum dose and dose rate could be obtained by using database statistical function.

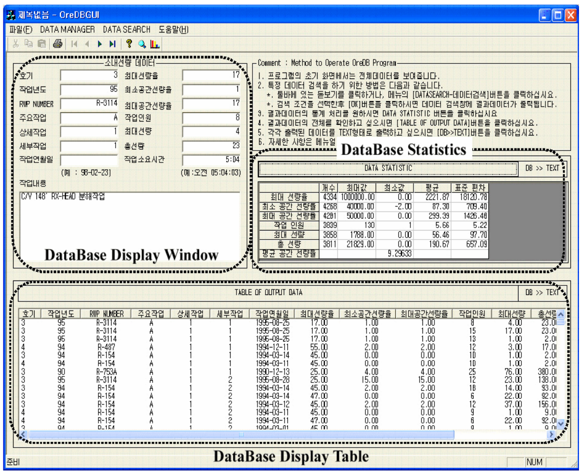


Fig. 6. Occupational dose assessment

The offsite dose can be assessed as a result of the evaluation of source terms and atmospheric dispersion factor. Hence, the offsite dose can be obtained after running those evaluating programs. For user's convenience, the calculation results of the offsite doses can be represented in text mode as well as in 2-D image mode. Fig. 7 and 8 show the display screens in text and image mode, respectively. In Fig. 8, the center of image is the nuclear power plant and the color is changed according to the offsite dose values

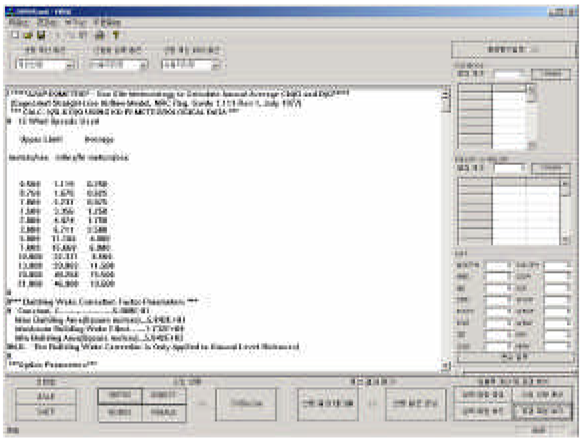

Fig. 7. Output Screen (Text Mode)

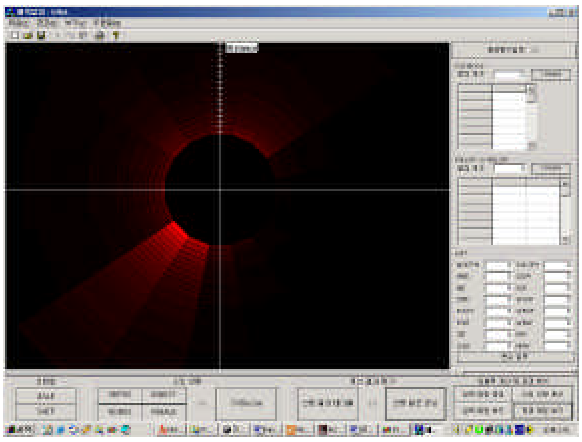

Fig. 8. Output Screen (2-D Image)

## 4. Conclusions

An integrated code, INSREC is developed and presented, which can assess both the annual

ORD and the offsite doses resulting from the normal operation of NPP. Also, INSREC is provided with an easy-to-use GUI (Graphic User Interface) made with Visual C++, which makes to enhance the user’s convenience. In near future, it is planned to develop the computer program of assessing the offsite dose during accident conditions and incorporate into INSREC in near future.

## Acknowledgement

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## Nomenclature

| | |
|---|---|
| $N_i^{R_j}$ | Atom density of nuclide i in region j, |
| $PF_i^{R_j}$ | Source term of nuclide i into region j, |
| $e_i^{R_j}$ | Escape rate of nuclide i into region j. |
| $l_i$ | Transformation const. of nuclide i, $sec^{-1}$ |
| $CVR^{Ri}$ | Volumetric ratio of coolant in region i over coolant in region j, fraction |
| $\overline{\Phi}^R$ | Average thermal flux in region R, $n/cm^2$-sec |
| $s_i$ | Neutron absorption cross-section, $cm^2$ |
| $R_i$ | Reactor coolant clean-up rate of nuclide i |
| $l_{j\to i}$, $f_{j\to i}$ | Branching fraction from nuclide j to i, fraction |

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