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A Study on the Evaluation of the Radioactive Source Term for Korean Next Generation Reactor

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Abstract

The amount of radioactive materials released from nuclear power plant must be evaluated before construction stage for the shielding design and radioactive systems. Since 1984, any new methodologies for source term evaluation are not provided. At present, a few codes such as PWR-GALE used for evaluation of source term have some limitations for application to the next generation plants of Korea. The purpose of this study is to provide the method and evaluation tool for radionuclide concentrations at reactor primary coolant systems and radioactive material released from NPP, and to compare the results with those of the well-known and recognized tools. The evaluation method for radionuclide concentrations at RCS is suggested and a corresponding code for source term evaluation is developed. The code named as Visual GALE is able to predict the radionuclide concentration of fission product at primary with various reactor design parameters based on the simple calculations. Also, Visual GALE is able to calculate the radioactive materials released from nuclear power plant with various waste treatments components. Visual GALE uses the simplified equation with the assumption of steady state condition for the fission product concentrations at RCS. At fuel pellet region, ORIGEN 2 code was used for the activity of fission product. For reflection of radwaste system of next generation reactor, waste treatment system is divided as waste input, radionuclide removal process and discharge rate. The reference system is pressurized water reactor with U-tube steam generator and the formal radwaste treatment system. By applying Visual-GALE code to YGN unit 3, 4, the results are compared with the actual data measured from the reference plants and calculation results of PWR-GALE and FSAR of YGN 3, 4. In this study, specifically, the expected fuel defect rate and the concentration distribution of the fission product was focussed for the analysis in detail. The comparison has shown that the well-known and recognized tools relatively overestimate radionuclide concentrations at the reactor coolant systems. Also, the resultant concentration distribution of the fission product from Visual-GALE is similar to the that of actual data measured from the reference plants and calculation results of PWR-GALE and FSAR (Final Safety Analysis Report) of YGN 3, 4. In conclusion, Visual-GALE can be used to evaluate the radioactive source term due to change of fuel inventory. Despite of the simple method, Visual-GALE showed reasonably good results as compared to the well-known and recognized tools such as PWR-GALE. For evaluation of the source term with various fuel defect rate, the Visual-GALE is turned out to be useful and applicable, for next generation reactor, to evaluate the radioactive source term.

1. Introduction

Typically, radioactive source term is used for shielding design, design of radioactive systems and calculation of expected gaseous and liquid releases from the plant during normal operation. Several methodologies to predict the radioactive source term are developed from the mid of 1970s to the mid of 1980s. Since 1984, any new methodologies for source term evaluation are not provided.

Although, at present, a few codes such as PWR-GALE, PROFIP3 and DAMSAM are used for evaluation of source term, these codes have some limitations at applying to plant of Korea because their base-data measured are not updated. Especially, radionuclide concentration in reactor coolant system should be modified for using MOX fuel and extended fuel cycle because radionuclide concentration in reactor coolant system is affected by fuel inventory and water chemistry conditions.

PWR-GALE is a computerized mathematical model for the expected source terms that radioactive material released in gaseous and liquid from water reactors. The calculations are based on data generated from operating reactors, field and laboratory tests and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment during normal operation, including anticipated operational occurrences. PWR-GALE use realistic model for reactor coolant activity based upon the data measured in USA during the 1980s. So, it is not proper to apply PWR-GALE to the next generation plants of Korea without updating database.

At present, the well-known and recognized tools are not proper for estimating source term of nuclear power plant of Korea. Also, it is necessary to evaluate radionuclide concentration at RCS before construction due to change of RCS conditions such as fuel defect rate, reactor design parameters and fuel inventory to describe the overall change in radwaste effluent from the fuel to the environment. Especially, because many nations including KOREA have scheduled to construct next generation reactor and to use of MOX(mixed Oxide) fuel and extended fuel cycle, the radionuclide concentrations become the predominant factor in estimation of the radioactivity in the principal fluid streams of a light water reactor over its lifetime.

The purpose of this study is to provide the method and evaluation tool for radionuclide concentrations at reactor primary coolant systems and radioactive material released from NPP, and to compare the results with those of the well-known and recognized tools respectively. Also, the evaluation method for radionuclide concentrations at reactor coolant systems is suggested and a corresponding code for source term evaluation is developed. The code named as Visual GALE is able to predict the radionuclide concentration of fission product at coolant region with various reactor design parameters based on the simple calculations.

For a simple calculation, we have selected the simplified equation by assumption of steady state. The simplified equation has used ORIGEN 2 code and modified ANSI/ANS-18.1 as the evaluation method for radionuclide concentrations at reactor coolant system with change of fuel irradiation condition. The modified ANSI/ANS-18.1 is used in radioactivity at coolant region. ORIGEN 2 is used in calculation of the effective core inventory. With this method, Visual-GALE is developed to predict the radionuclide concentration based on reactor design conditions and fuel inventory.

2. Method and Code Description

The concentration of fission products at RCS is determined by the application of appropriate mathematical removal rate equations and the fission product inventory determined in the two separate regions of the fuel pellet region and the reactor coolant region. This mathematical model has many calculation and nuclear data.

Therefore, in this study, the simplified equation is used for source term evaluation with the assumption of the steady state. This assumption is appropriate for the expected source term because it is based normal operational condition. For fission product, the simplified equation provides radionuclide concentration at primary coolant region based on radioactivity at fuel pellet region.

The equation for the reactor coolant region is as following

$$\frac{dC_i}{dt} = \frac{\nu_i \times \eta_i \times A_i}{WP} - (R_i + \lambda_i + \sum_{ai} \phi) C_i - \frac{L_i}{WP} C_i \tag{1}$$

Except some nuclides having a large absorption neutron cross-section such as Kr-85, and on the assumption of steady state, the equation (1) becomes

$$C_i = \frac{\nu_i \times \eta_i \times A_i}{(R_i + \lambda_i) \times WP} \tag{2}$$

In equation (2), the radionuclide concentration at primary coolant region is a function of radioactivity at fuel pellet region, fuel defect rate, escape coefficient from fuel, reactor coolant mass, decay constant and the radionuclide removal rate. In this study, for radioactivity at the fuel pellet region, ORIGEN 2 code is used. The radionuclide removal rate represents the reactor coolant purification system. For calculation of the radionuclide removal rate, the radionuclides are divided into six groups by their characteristics. Table 1 shows the classification of the radionuclide and figure 1 represents the main input windows of Visual GALE for evaluation of radioactive source term at reactor coolant system.

In equation (2), the removal rate (R_i) of the radionuclide from the system due to demineralization, leakage and etc. is as following

- For noble gases

$$R_1 = \frac{FB + (FD - FB)Y}{WP} \tag{3}$$

- For other nuclides

$$R_i = \frac{FD \times NB + (1 - NB)(FB + FA \times NA)}{WP} \tag{4}$$

3. Results and Discussion

The proposed radioactive source term evaluation methodology have been applied to the YGN 3,4(Young-Gwang Units 3 and 4). Input data was grouped as two categories. One is the fixed data and the other is the variable data. The fixed data are the nuclide data, plant specific parameters and experimental data such as half-life, escape coefficient, efficiency of coolant purification component and others. The variable data are some design parameters and experimental data such as letdown flow rate, fuel defect rate and so forth.

In this study, for the effective fuel inventory, ORIGEN 2 code is used. Table 2 shows the equilibrium fuel composition of YGN 3, 4. With this composition, ORIGEN 2 provides the equilibrium fuel inventory.

In this study, the resultant radionuclide concentrations are compared with those of PWR-GALE and FSAR of YGN 3, 4. PWR-GALE doesn't include the fuel defect. Typically, however, 0.12% fuel defect rate is considered for the expected source term. On the other hand, FSAR of YGN 3, 4 uses only 1% fuel defect rate. Also, EPRI-URD uses 0.25% fuel defect rate.

In this study, the expected fuel defect rate and the concentration distribution of the fission product was focussed to be analyzed in detail. The results were compared to the actual data measured from the object plants and calculation results of PWR-GALE and FSAR of YGN 3, 4 with various fuel defect rates.

Figure 2 and 3 show that the well-known and recognized tool overestimates radionuclide concentrations as compared to the actual concentrations. The actual data is the average value of YGN 3, 4 during full power day at once cycle and the result of PWR-GALE is adjusted with the system parameters of the object plant. In figure 2, for noble gas at primary coolant, the results of PWR-GALE are 100-1000 times higher than the actual noble concentrations. In figure 3, for corrosion product, shows that the results of PWR-GALE are 50-1000 times higher than the actual concentrations. Although PWR-GALE has used a realistic model based the measured data in USA during 1980s, the base-data was not updated. However, it is prospected that the fuel defect rate is decreased by improvement of the fuel manufacturing technology. In addition to the decrease of the fuel defect rate, the improvement of water chemistry, at present, the actual radionuclide concentrations at reactor coolant system is lower than those of PWR-GALE.

The comparison of results having 0.12% fuel defect rate with PWR-GALE is shown as figure 4 and 5. For fission products, the resultant distribution of radionuclide concentration and their values are similar to those of PWR-GALE at primary region. In this case, the some nuclides such as Sr-89, Sr-91, Y-91, Zr-95, Nb-95, Mo-99, Te-129 and Ba-140 are very different from that of PWR-GALE. It is due to that activation of the irradiated structural material plays a considerable role at coolant concentration in case of some nuclides. (Sr-89, Sr-91, Y-91, Nb-95, Mo-99, Te-129 and Ba-140)

In figure 6 and 7, for the fission product at primary coolant region, the results for 1% fuel defect rate are compared to the reference data-the those of FSAR. Because it is not easy to access DAMSAM, the results of DAMSAM are taken to FSAR of YGN 3, 4. In case of 1% fuel defect rate, code results are very similar to the reference data except some nuclides such as Sr-89, Sr-90, Y-91, Zr-95 and Mo-99.

As described above and shown at figure 2 - 7, it is a reasonable to make use of Visual-GALE for the radioactive source term evaluation at reactor coolant region. Especially, for fission product, Visual-GALE was comparable to the well-known and recognized codes such as PWR-GALE and FSAR.

However, for some radionuclide such as Sr-89, Sr-91, Y-91, Zr-95, Nb-95, Mo-99, Te-129 and Ba-140, there was a wide disparity between the results of Visual-GALE and the commercial codes. In table 3, these nuclides are generated with various sources, i.e., fission and activation. For most fission products, the product of activation including structural materials is ignored in comparison with production by fission. However, some radionuclides such as Zr-95, Nb-95, Mo-99 are not generated only by fission also by activation of structural materials. Because Visual-GALE has not considered the activation production, radionuclides such as Zr-95, Nb-95, Mo-95 should be excepted. Another reason of a disparity in Sr-89, Sr-91, Y-91, Te-129 and Ba-140 is that they have a short half-life. For almost all fission products, the decay constant is lower than radionuclide removal rate of purification system. Therefore, the purification system is the predominant factor. However, although some nuclides with short half-life are saturated, because of their short half-life, their decay constant has a considerable role for removal rate.

Figure 8 shows the fission product concentrations with various fuel defect rates and the comparison with the reference data. The results of PWR-GALE were the range of 1%-0.12% fuel defect rate. Coincidence of their distribution and the fuel defect rate range show that Visual-GALE has a reasonable result.

In figure 9, the expected fuel defect rate of YGN 3,4 was estimated as 3.93E-3%. By the comparison with the actual data measured at the reference plants, some radionuclides such as Kr, Xe and I were selected for the prediction of the fuel defect rate. Because these nuclides are coincident with the commercial codes with various fuel defect rate, they were compared with the measured data of YGN 3, 4. The resultant fuel defect rate has a range of 1.57E-3 - 9.00E-3% and the average value is 3.93E-3%. This range is very low as compared to the value used to the commercial codes. As the YGN 3, 4 have recently constructed, this means a reasonable range.

Figure 10 shows the input window of the Visual-GALE for liquid waste treatment system. In Visual-GALE, the radwaste treatment system is divided as waste input, removal process and discharge rate. Waste input is characterized by amount of waste input and its PCA(primary coolant activity) ratio for each stream. Also, radionuclide removal process is divided as DF of system and collection and treatment time for decay. As figure 10, Visual-GALE are developed with Visual Basic for GUI(Graphic User Interface) and can evaluate the radioactive source term at RCS and estimate the amount of radioactive material released from nuclear power plant.

4. Conclusion

In this study, the evaluation method for radionuclide concentration at PWR reactor coolant system was suggested and a corresponding computer code named as Visual-GALE was developed.

By applying SYCOS code to YGN 3 & 4 power plants, and comparing the code results with the actual data measured at the reference plants and calculation results of PWR-GALE and FSAR (Final Safety Analysis Report) of YGN 3, 4, the following results were obtained.

1. The well-known and recognized tools generally overestimate radionuclide concentration of the fission products at reactor coolant system.
2. For the fission products, the concentrations and their concentration distribution trends at the reactor coolant system are well agreed with the results of the well-known and recognized tools.

In conclusion, despite of a simple approach, Visual-GALE code shows quite reasonable results as compared to the well-known and recognized tools such as PWR-GALE and FSAR. For the fission products, especially, Visual-GALE code can predict the fuel defect rates with sufficient appropriate data and operational histories. Also, it is concluded that Visual-GALE is also applicable to the estimation of the expected source term for KNGR with respect to MOX fuel and radwaste treatment system.

In relation to the future research topics, it is recommended that further work should be carried out on the followings

1. Generating and updating the appropriate and good database for the source term evaluation of Korean nuclear power plant
2. Development of the more reliable and detailed computer code depending on the various fuel conditions.

Acknowledgement

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References

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[3] NRC, *Calculation of Release of Radioactive Materials in Gaseous and Liquid Effluents from the Pressurized Water Reactors*, 1976

[4] NRC, *Calculation of Release of Radioactive Materials in Gaseous and Liquid Effluents from the Pressurized Water Reactors*, 1985

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[6] KEPCO, *YGN 3, 4 FSAR*

< Table 1. Classification of the radionuclide >

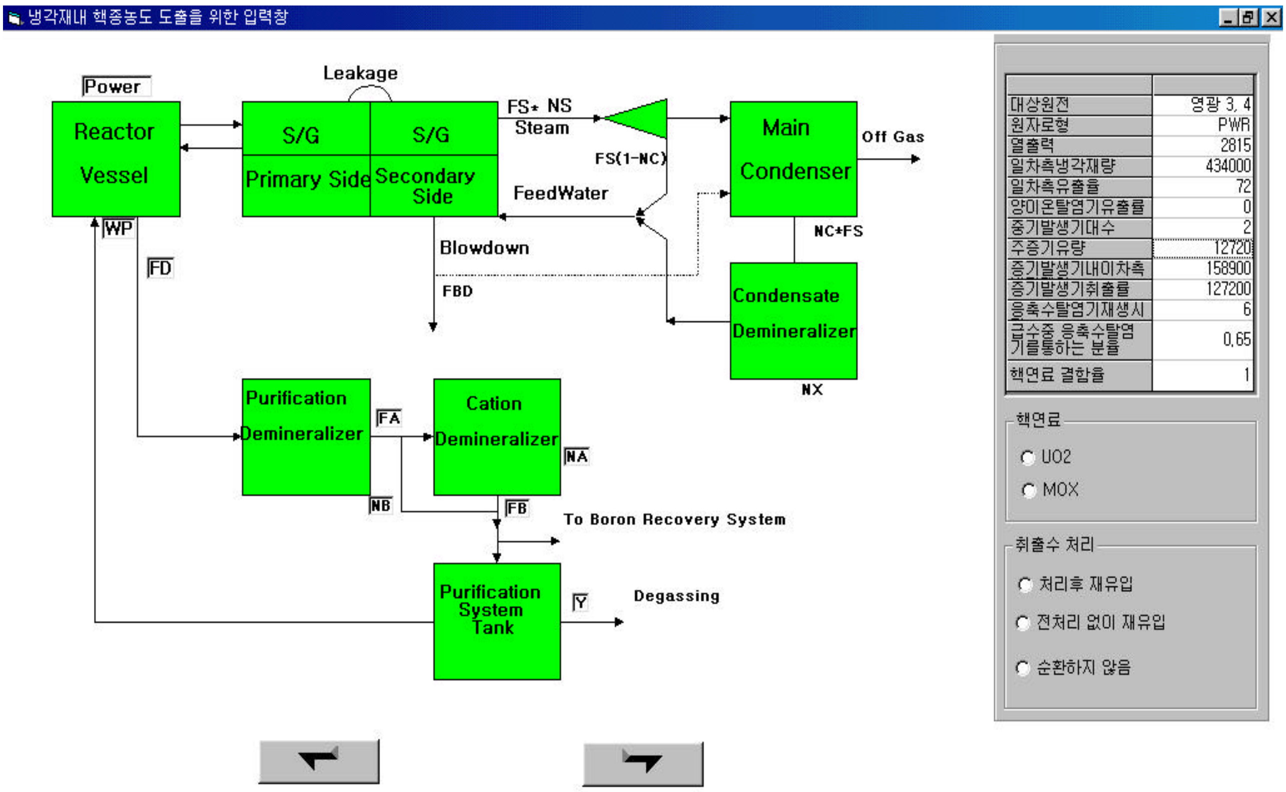
Class 1	Kr, Xe	Class 4	N-16
Class 2	Br, I	Class 5	H-3
Class 3	Rb, Cs	Class 6	others

< Table 2. Equilibrium fuel composition of YGN 3, 4 >

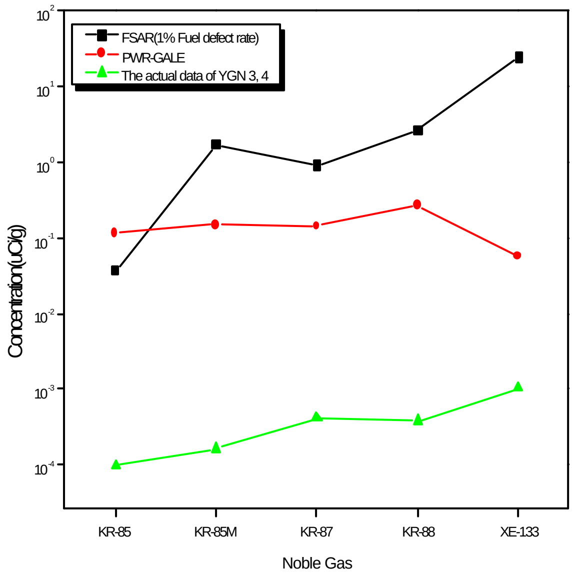
Fuel Type	Quantity	U-235 Enrichment	Rod/Assembly
A	45	1.28	236
B	20	2.34	236
B1	8	2.34	176
	8	1.28	52
B2	16	2.34	232
C	12	2.84	184
	12	2.34	52
C1	32	2.84	176
	32	2.34	52
D	12	3.34	184
	12	2.84	52
D1	8	3.34	176
	8	2.84	52
D2	24	3.34	128
	24	2.84	100

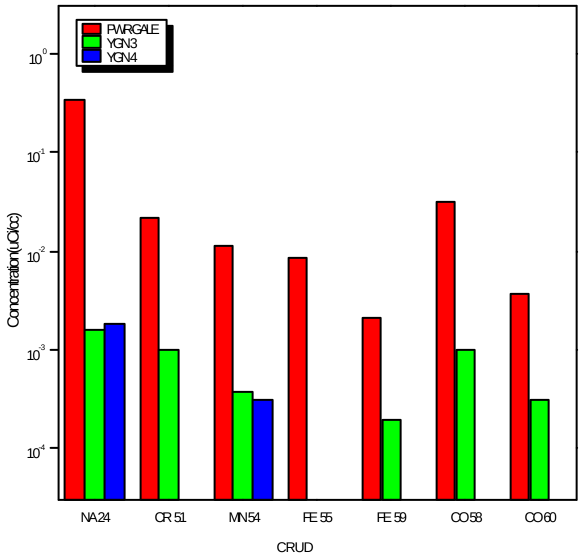
< Table 3. Inventory due to the radioactive source >

Radionuclide	At fuel[Ci]		At Structural material[Ci]	
	Fission	**Activation**	**Fission**	**Activation**
Sr-89	8.914E+05	1.356E-07	2.043E-02	5.445E+01
Sr-90	7.253E+04	8.761E-11	1.338E-03	1.195E-03
Y-91	1.167E+06	2.207E-08	2.799E-02	1.374E+02
Zr-95	1.656E+06	2.983E-03	4.735E-02	4.676E+04
Nb-95	1.672E+06	7.834E-03	4.590E-02	4.685E+04
Mo-99	1.818E+06	3.590E+01	6.010E-02	1.382E+03



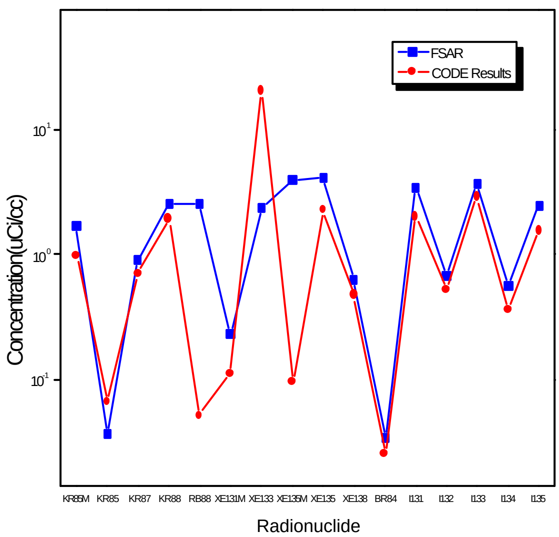
<Figure1. Main input windows of Visual GALE >

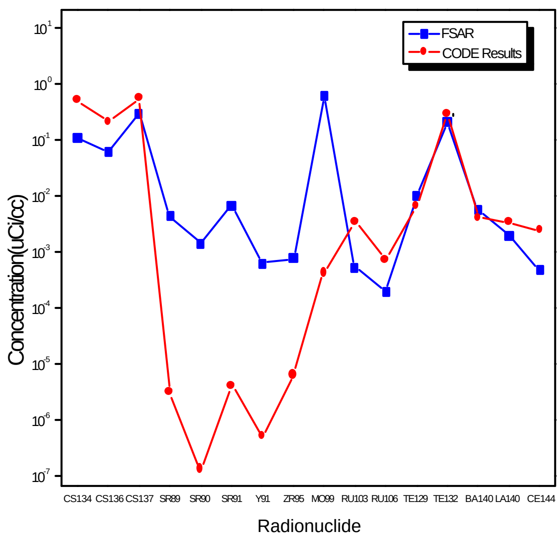




< Figure2 . Comparison with the reference data (1)>

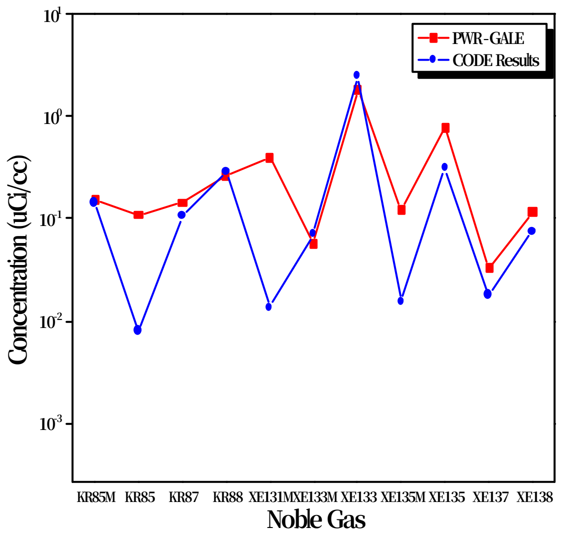
< Figure3. Comparison with the reference data (2)>



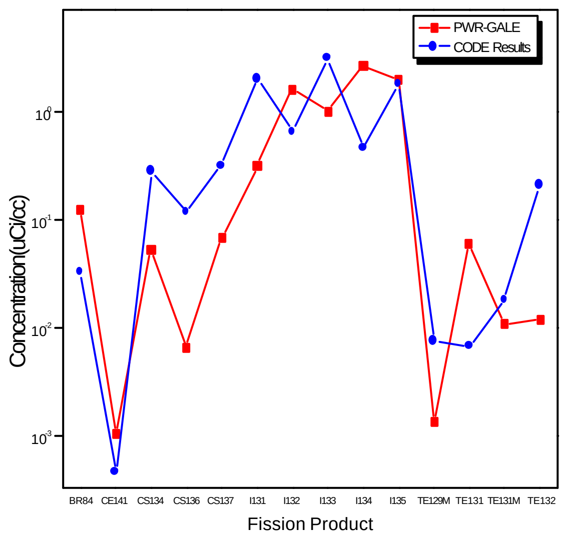


<Figure4 . Comparison with PWR-GALE (1)>

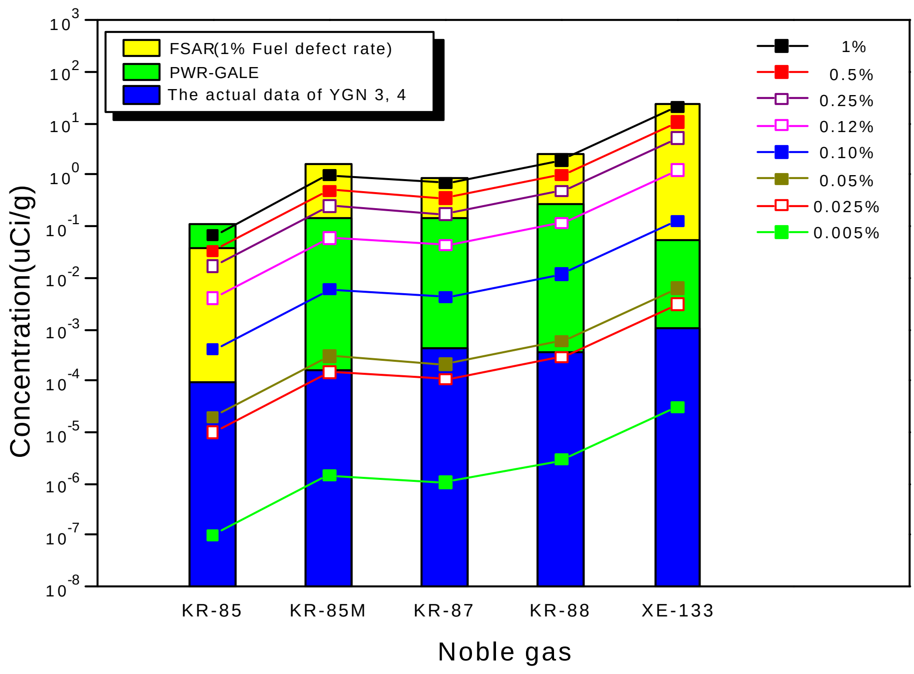
< Figure5. Comparison with PWR-GALE (2)>



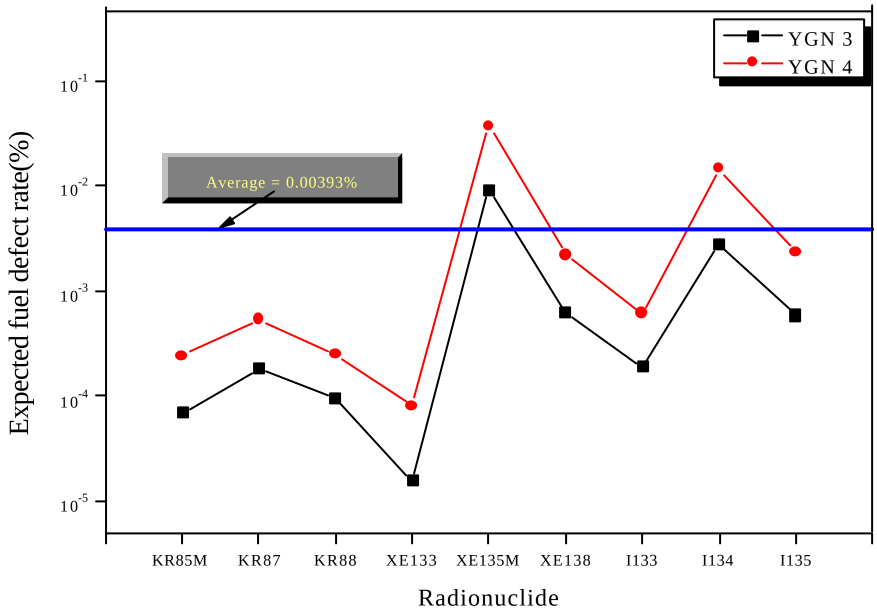
< Figure6 . Comparison with FSAR(1) >



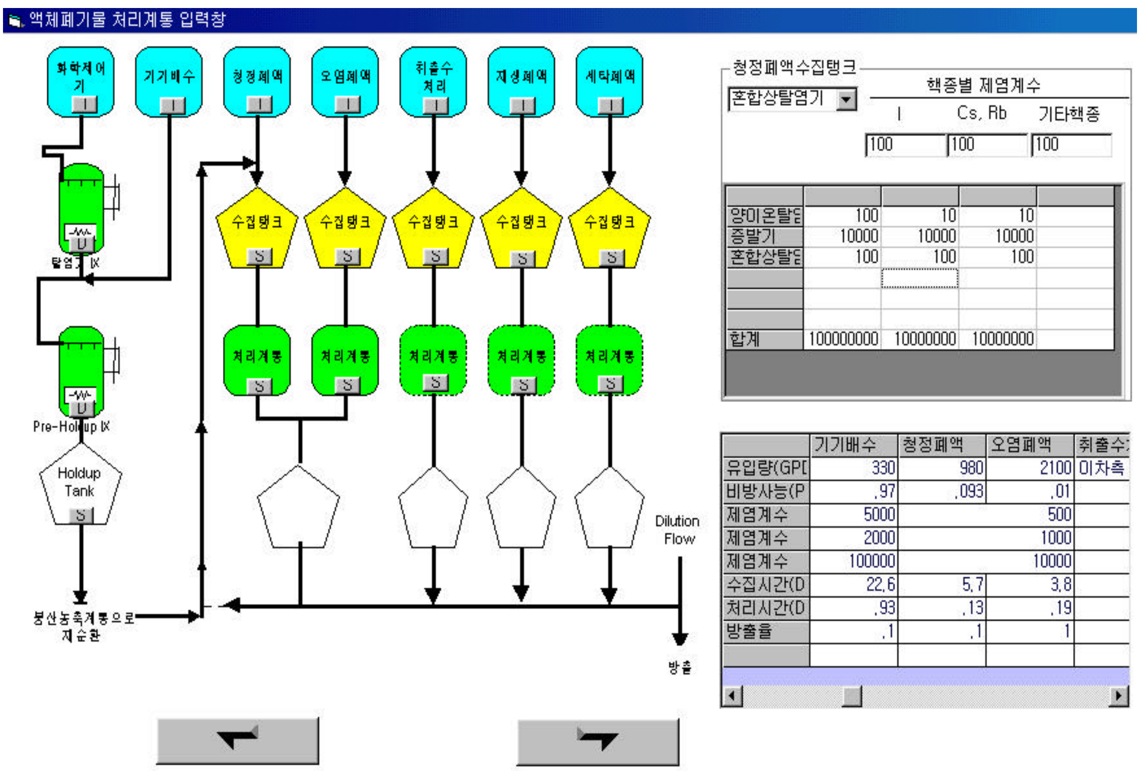
< Figure7. Comparison with FSAR(2) >



< Figure 8. Fission product concentration with various fuel defect rate >



< Figure9. Expected fuel defect rate >



< Figure10. Input window of Visual-GALE >