DEVELOPMENT OF PREDICTION SYSTEM OF DOSE EQUIVALENT RATE AROUND A PACKAGE

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Introduction

Spent fuels have had to be transported from nuclear reactor to reprocessing factory. As it is increasingly the high-burn-up type of spent fuel that is being transported, it becomes more and more important to learn how to keep the radiation dose equivalent rate surrounding the fuel package as low as possible, for the purpose of reducing the effects of exposure on the operators that work with the packages.
It is therefore necessary first to develop a new system that can evaluate the radiation strength of the source in detail, on the basis of the irradiation history of each fuel assembly in a TN-12 or 12A package, and then to determine the best way to organize the assemblies in the package so that the dose equivalent rate around a package is kept to a minimum.

Constitution of the system

The constitution of the system developed by the authors is shown in the flow chart in Fig. 1. A general description of each of the programs involved in the system is provided below.

ACDIP : This is the program for inputting the data about the fuel assemblies to be transported. This includes data such as the number of assemblies, the uranium weight, the enrichment level, the irradiation history, the cooling time required for each, and the manufacturer's serial number on the package for transportation. All the input can be conducted according to the instructions on the CRT. An example of the CRT directions and a display showing input data is given in Fig.2.

GROUP : This is the program for calculating the radiation source grouping. In this system, the longest time is required for the calculation of the source radiation strength. To shorten the time needed, the calculation is conducted with similar specification types grouped into one category. Radiation source grouping is based on factors such as cycle operation pattern, enrichment and burn-up.

SOURCE : This is the program used to calculate the radiation source. The radiation source calculation is done by the same calculation method as the ORIGEN 2 code, based on the grouped radiation source specifications of the fuel assemblies (see next section).

ODER : This program is used to determine the order of the radiation strength levels of the assemblies, in order to plan the arrangement of the fuel assemblies in the container, as illustrated in Fig.3. In this case the radiation strength order has to be calculated on the basis of the reduced value, since the dose equivalent rate contributions of neutron and gamma radiation are different at each point in the package. As described above, the arrangement of assemblies in the package should be done in an order that assures the lowest possible dose equivalent rate around the package.

EDIT : This program is used to derive the dose equivalent rates of the packages. On the basis of the location the fuel assemblies will have in the containers, the calculation uses the data base obtained from a detailed shielding analysis to determine the dose equivalent rate surrounding the package.

PRIN : This is the printing program. The results of the above calculations should be output in report style. The printout should include such contents as 1) a summary of the dose equivalent rate around the package; 2) a comparison of these results with the values presented in the Safety Analysis Report; and 3) a list of the fuel assemblies to be packaged. Examples of such reports are shown in Table 1.

Radiation source strength calculation code SOURCE

In the Safety Analysis Report, the ORIGEN 2 code is used to carry out the calculation of source radiation strength. A new system for predicting the dose equivalent rate has been developed which takes into consideration the memory capacity and the calculation time, using the same method as the ORIGEN 2 code for personal computers, with only the function of the source radiation strength calculation with the minimum number of nuclides to be treated. This allows the source strength to be calculated in only a few seconds.
Bateman's equation (shown below) is used as the basic formula for calculating the generated quantity of the nuclides from the neutron and gamma radiation source.

$$N_N = C_1 e^{-\Lambda_1 t} + C_2 e^{-\Lambda_2 t} + \cdots\cdots + C_N e^{-\Lambda_N t}$$

$$C_n = \frac{\Lambda_1^* \Lambda_2^* \cdots\cdots \Lambda_n^*}{(\Lambda_1 - \Lambda_n)(\Lambda_2 - \Lambda_n)\cdots\cdots(\Lambda_{n-1} - \Lambda_n)(\Lambda_{n+1} - \Lambda_n)\cdots\cdots(\Lambda_{N-n})} = N_1^0$$

$\Lambda_n = \lambda_n + \sigma\phi$ (Corrected decay constant when quenching is considered)

$\Lambda_n^* = \lambda_n^* + \sigma^*\phi$ (Decay constant when generation of next nuclide is considered)

N_1^0 : Quantity of basic nuclide that exists at initial stage.

(Bateman's equation is usable only when $N_2^0 = N_3^0 = \cdots\cdots N_n^0 = 0$)

The SOURCE code uses Bateman's equation, which is also used within the ORIGEN 2 code, treating 11 nuclides of actinoid and 10 nuclides of fission products. Using this code, the source neutron and gamma radiation strength can be calculated with quite high accuracy in a relatively short time.
As shown in Fig.4, the main neutron sources, ^{242}Cm and ^{244}Cm, are calculated through 2 routes of 11 nuclides ; Both of these are the last daughter nuclides of ^{238}U, that is, the parent nuclide.
For a large package such as the TN, the shielding body is thick enough that the influence of the low energy gamma ray can be ignored. Therefore only the five main fissile nuclides, ^{106}Rh, ^{110m}Ag, ^{134}Cs, ^{144}Pr, and ^{154}Eu, which irradiate a high energy gamma ray, are taken into account in these calculations.
In order to shorten the calculation time, it should be assumed that the pre-stage nuclide of the gamma radiation source nuclide is generated directly from the parent nuclide. Five routes develop for the gamma radiation source calculations :

① (Parent nuclide) → ^{106}Ru → ^{106}Rh
② (Parent nuclide) → ^{109}Ag → ^{110m}Ag
③ (Parent nuclide) → ^{133}Cs → ^{134}Cs
④ (Parent nuclide) → ^{144}Ce → ^{144}Pr
⑤ (Parent nuclide) → ^{153}Eu → ^{154}Eu

As the parent nuclide, four fissile nuclides are used: ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu.

To evaluate this newly developed SOURCE code, a calculation of the burn-up was conducted (burn-up; 30GWD/T, a 3-cycle operation of 360 days). The results of the comparison of the SOURCE code with the ORIGEN 2 code are shown in Table 2.

In this calculation of the minority nuclide, the results agree within a 1% error for all neutron generated numbers. As for gamma radiation, in spite of the bold hypothesis, the results also agree within a 1% error range for the high energy group spectrums which play an important role in determining the dose equivalent rate around the package.

Compared to the ORIGEN 2 code, the SOURCE code developed for personal computer is much more accurate. Nevertheless, when this code is used, the following two points should be taken into account ;

1) The code is intended for use in calculating the source radiation levels of a large-sized package which contains a PWR-type fuel assembly.

2) Since the product route of a nuclide is simplified, and short-lived nuclides have been ignored, some unusually large source strength errors might occur just after shutdown. However, the accuracy can be depended on when the cooling time is 100 days or longer.

Preparation of the shielding calculation data

To use this new system to determine the dose equivalent rate, one basic type of data needed is the contribution rates of the fuel assemblies to be contained. In obtaining the contributions ratios, the ANISN code and the DOT 3.5 code are used for neutrons, while the QAD code is used for gamma rays. The analysis model for a TN-12A package is shown in Fig.5. Figure 6 presents an example of the calculated results for spatial distribution of the neutron dose equivalent rates. Thus the contribution rate data obtained from any location in the package can be used as the base. If in addition to this the radiation source for each assembly as calculated by the SOURCE code is given, the external dose equivalent rate can be obtained.

Comparison of calculated values with measured values

As shown in Table 3, the data for TN-12A Type (6 bodies) packaged in 1990 were compared with the values calculated by this system. The results show that this system offers adequate prediction accuracy for the side and bottom areas, with somewhat less accuracy for the lid direction.

Conclusion

This system for minimizing the danger of radiation for operators involved in packaging and transporting spent fuel was developed for personal computer use, to offer ease in handling and high adaptability. The data input is done in dialogue style, with a variety of check functions.

In checks to verify the accuracy of the shielding calculation data in this system by comparing the calculated values with several kinds of measured values, the reliability of this new system has been shown to be very high. Since its high utility has been recognized, the system has already been put into use in actual transportation situations.

References

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DOT3.5......"DOT3.5 TWO DIMENSIONAL DISCRETE ORDINATES RADIATION TRANSPORT CODE" Oak Ridge National Laboratory Radiation Shielding Information Center Code Package CCC-276.

ORIGEN2..."ORIGEN-2 Isotope Generation and Depletion Code MATRIX EXPONENTIAL METHOD" Oak Ridge National Labolatory Radiation Shielding Information Center Code Package, CCC-371.

QAD...........Y. Sakamoto, S.Tanaka, "QAD-CGGP2 and G33-GP2: Revised Version of QAD-CGGP and G33-GP (Code with the Conversion Factors from Exposure to Ambient and Maximum Dose Equivalents)", JERI-M 90-110(1990).

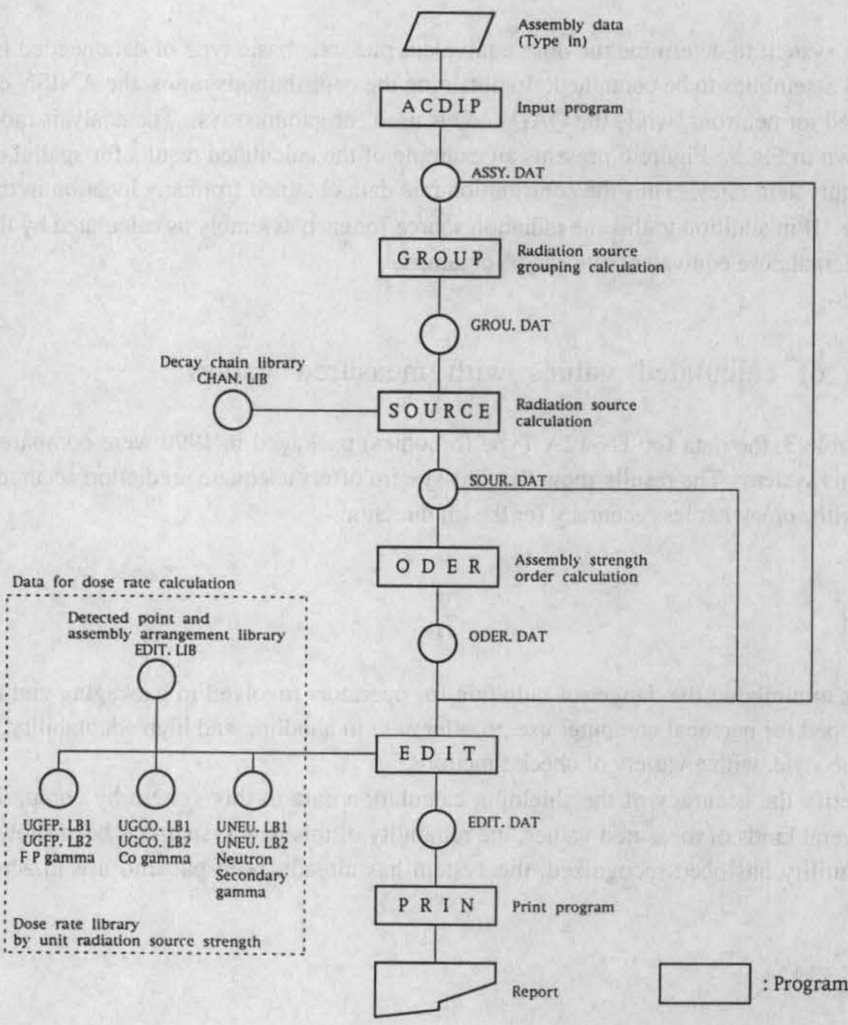

Fig.1 Structural figure of predictive system of cask dose equivalent rate

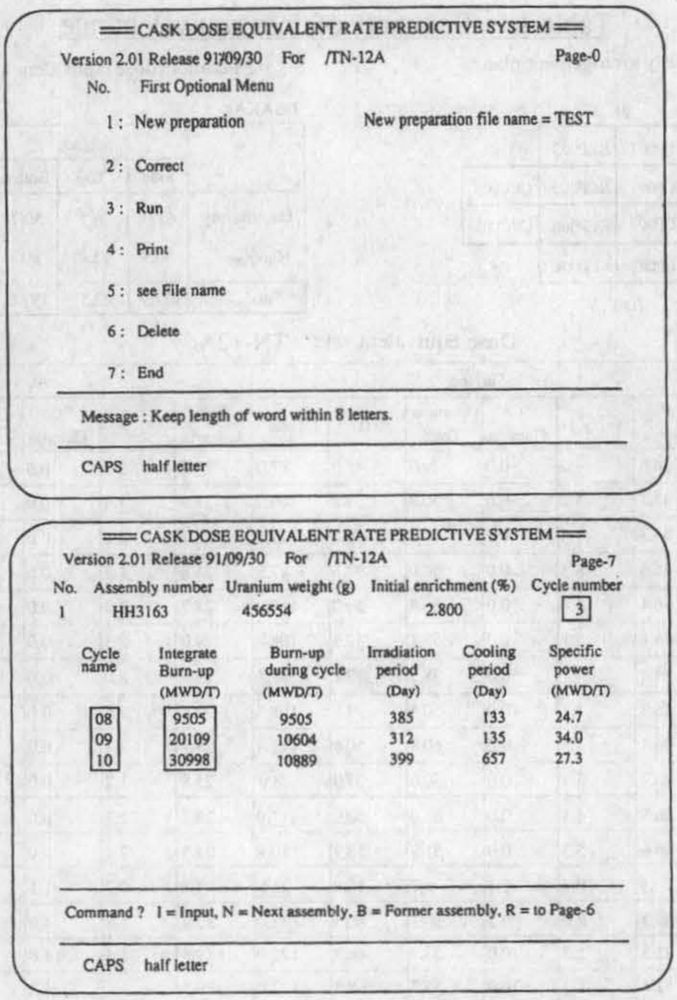

Fig.2 Sample CRT-display of predictive system of cask dose equivalent rate

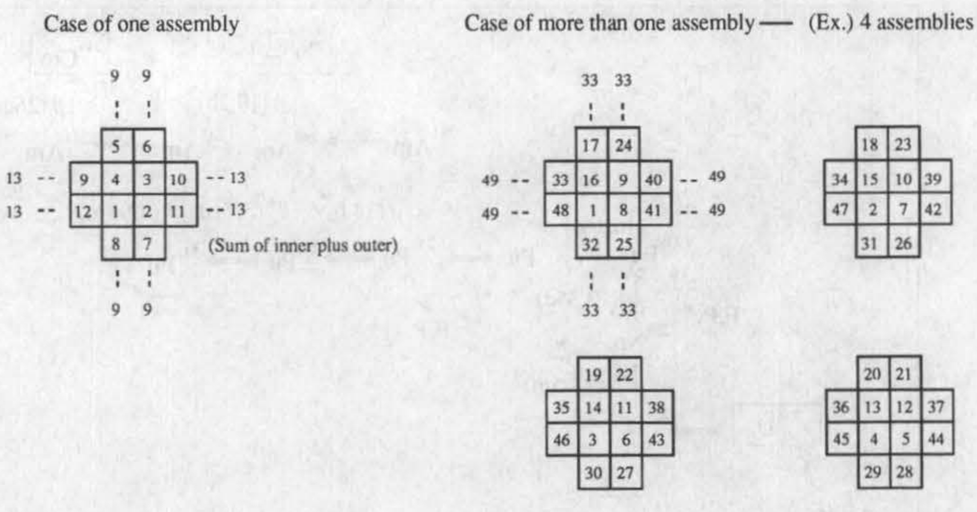

Fig.3 Determination of storage location of fuel assemblies

Table 1 Summary of dose equivalent rate

Fuel assembly arrangement plan

		(0°)			
		KEP007	KEP005		
(270°)	CRC003	KCP001	KCP005	CRC003	(90°)
	CRC001	KCP002	KCP006	CRC001	
		KEP003	KEP008		
		(180°)			

Maximum dose equivalent rate : TN-12A

OSAKA4 (unit: μ Sv/h)

	Surface			At 1m		
	Side	Top	Bottom	Side	Top	Bottom
Gamma ray	61.0	6.7	50.7	30.9	3.6	24.6
Neutron	50.9	15.8	80.3	28.2	5.3	26.4
Total	111.9	22.5	131.0	59.1	8.9	51.0

Dose equivalent rate : TN-12A (Unit: μ Sv/h)

		Surface						At 1 m					
		F.P. Gamma	(n, γ)	^{60}Co Gamma	Gamma Total	n	Total	F.P. Gamma	(n, γ)	^{60}Co Gamma	Gamma Total	n	Total
Side	0°	34.8	4.2	0.0	39.0	48.0	87.0	19.7	2.0	0.0	21.7	27.1	48.8
	27°	45.7	5.2	0.0	50.8	58.2	109.2	23.9	2.4	0.0	26.3	31.9	58.1
	63°	56.3	4.4	0.0	60.8	50.2	111.0	28.7	2.1	0.0	30.7	28.0	58.7
	90°	46.6	3.3	0.0	50.0	37.5	87.5	25.8	1.7	0.0	27.5	22.3	49.8
	117°	56.4	4.4	0.0	60.8	50.0	110.9	28.7	2.0	0.0	30.8	27.8	58.6
	153°	46.1	5.1	0.0	51.2	57.5	108.8	24.0	2.3	0.0	26.4	31.5	57.9
	180°	34.4	4.1	0.0	38.5	47.1	85.7	19.4	2.0	0.0	21.4	26.5	47.9
	207°	45.3	5.2	0.0	50.5	57.8	108.3	23.8	2.4	0.0	26.2	31.7	57.8
	243°	56.4	4.5	0.0	60.8	50.6	111.4	28.7	2.1	0.0	30.8	28.2	59.0
	270°	46.7	3.4	0.0	50.0	37.9	88.0	25.8	1.7	0.0	27.5	22.5	50.0
	297°	56.5	4.5	0.0	61.0	50.9	111.9	28.7	2.1	0.0	30.9	28.2	59.1
	333°	46.6	5.3	0.0	51.9	58.9	110.8	24.3	2.4	0.0	26.7	32.1	58.8
Top	C	3.9	0.7	2.1	6.7	15.8	22.5	1.9	0.7	1.1	3.6	5.3	8.9
Bottom	C	39.2	2.0	9.5	50.7	80.3	131.0	17.9	1.8	4.9	24.6	26.4	51.0
	45°	42.5	1.3	10.0	53.7	68.3	122.0	19.5	1.4	4.8	25.7	24.6	50.2
	135°	42.4	1.3	10.0	53.7	68.3	122.0	19.5	1.4	4.8	25.6	24.6	50.1
	225°	42.4	1.3	10.0	53.7	68.3	122.0	19.4	1.4	4.8	25.6	24.6	50.1
	315°	42.5	1.3	10.0	53.7	68.3	122.0	19.5	1.4	4.8	25.7	24.6	50.2

C : Center n : Neutron

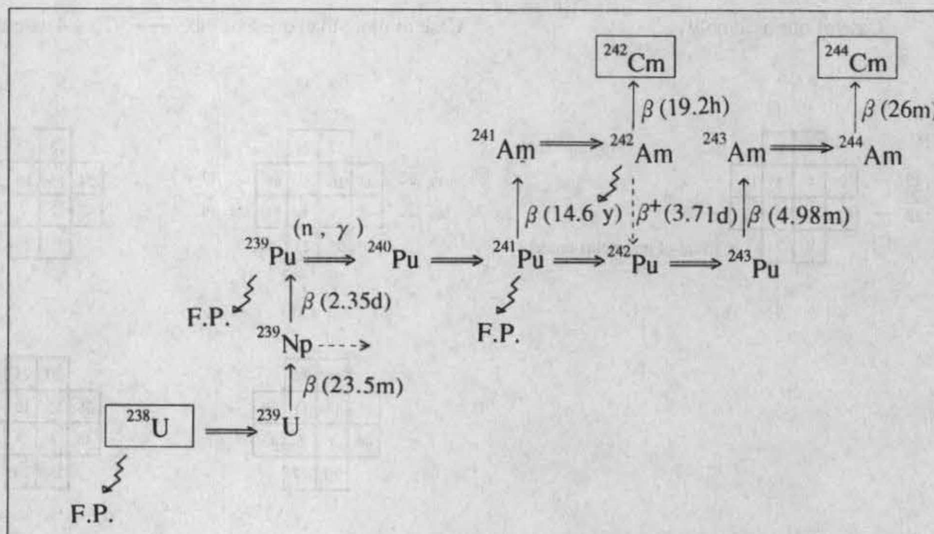

Fig.4 Generative process for ^{242}Cm and ^{244}Cm

Table 2 Comparison of Calcultion Results

[Neutron Radiation Source]

(g-atom/MTU)

Generative Quantity	SOURCE (a)	ORIGEN-2 (b)	a/b
^{239}Pu	2.399×10^{1}	2.361×10^{1}	0.991
^{240}Pu	8.873×10^{0}	9.039×10^{0}	0.982
^{242}Pu	2.011×10^{0}	2.100×10^{0}	0.958
^{241}Am	8.327×10^{-1}	8.469×10^{-1}	0.983
^{242}Cm	5.163×10^{-2}	5.141×10^{-2}	1.004
^{244}Cm	1.083×10^{-1}	1.084×10^{-1}	0.999
All Neutron Gererative Number (n/sec/MTV)	6.251×10^{8}	6.290×10^{8}	0.994

[Gamma Ray Source]

(g-atom/MTU)

Generative Quantity	SOURCE (a)	ORIGEN-2 (b)	a/b
^{106}Ru	4.386×10^{-7}	4.416×10^{-7}	0.993
^{110m}Ag	2.186×10^{-3}	2.437×10^{-3}	0.897
^{134}Cs	4.391×10^{-1}	4.322×10^{-1}	1.016
^{144}Pr	2.084×10^{-5}	2.093×10^{-5}	0.996
^{154}Eu	2.016×10^{-1}	2.225×10^{-1}	0.953

Gamma ray spectrum (ph/sec/MTU)

	SOURCE (a)	ORIGEN-2 (b)	a/b
1 (0.375Mev)	8.080×10^{14}	9.846×10^{14}	0.821
2 (0.575Mev)	6.110×10^{15}	9.471×10^{15}	0.645
3 (0.850Mev)	2.911×10^{15}	5.728×10^{15}	0.508
4 (1.25 Mev)	5.685×10^{14}	5.833×10^{14}	0.975
5 (1.75 Mev)	4.754×10^{13}	4.932×10^{13}	0.964
6 (2.25 Mev)	7.284×10^{13}	7.324×10^{13}	0.995
7 (2.75 Mev)	1.373×10^{12}	1.383×10^{12}	0.993
8 (3.50 Mev)	1.720×10^{11}	1.733×10^{11}	0.992

Table 3 Comparison of the measured values with the analysis values

(TN-12A, Surface)

		Gamma ray			Neutron		
		Measured values	Analysis values	Rate	Measured values	Analysis values	Rate
Side	①	45	48.3	0.93	10	17.4	0.58
	②	40	48.5	0.83	11	17.6	0.63
	③	40	46.7	0.86	10	17.5	0.57
	④	40	46.5	0.86	15	19.7	0.76
	⑤	40	45.5	0.88	20	20.7	0.96
	⑥	40	45.9	0.87	15	20.5	0.73
Bottom	①	35	49.3	0.71	15	62.5	0.24
	②	35	49.0	0.71	15	63.5	0.24
	③	28	47.9	0.59	1	63.0	0.02
	④	35	45.7	0.77	20	65.5	0.31
	⑤	30	42.9	0.70	30	76.5	0.39
	⑥	29	43.4	0.67	20	75.5	0.27
Top	①	4	15.3	0.26	1	27.0	0.04
	②	5	15.3	0.33	1	28.0	0.04
	③	5	15.0	0.33	2	28.0	0.07
	④	4	14.7	0.27	1	29.0	0.03
	⑤	4	14.0	0.29	1	33.0	0.03
	⑥	4	14.0	0.29	1.3	33.0	0.04

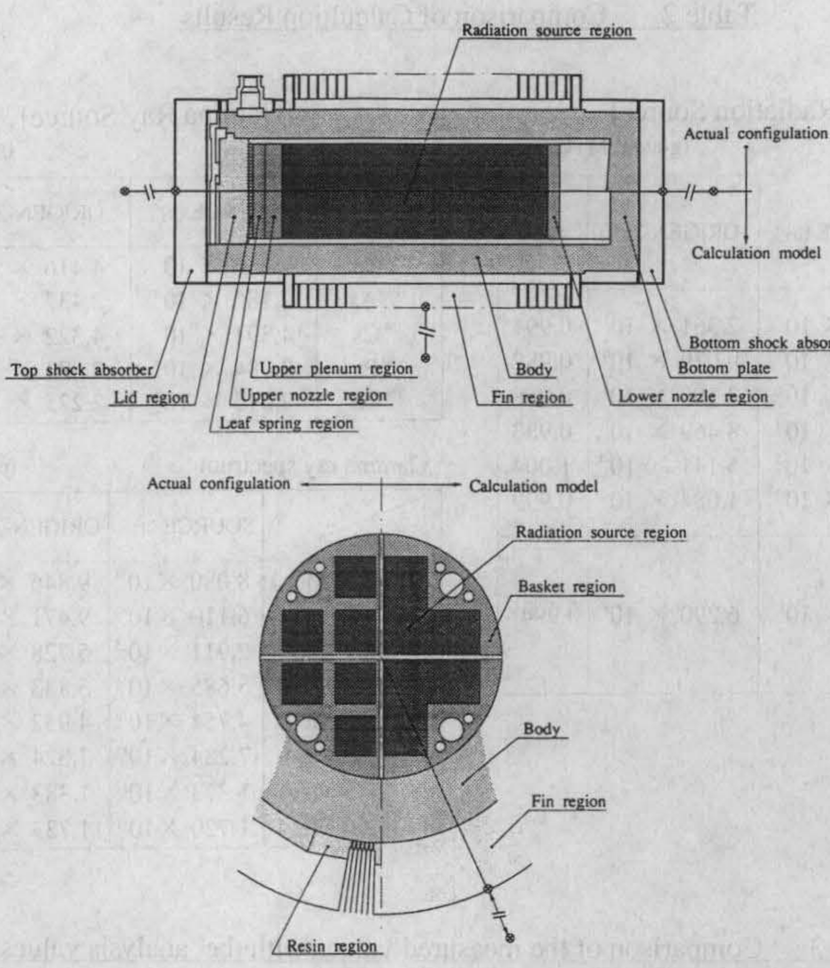

Fig.5 Dose equivalent rate calculation model

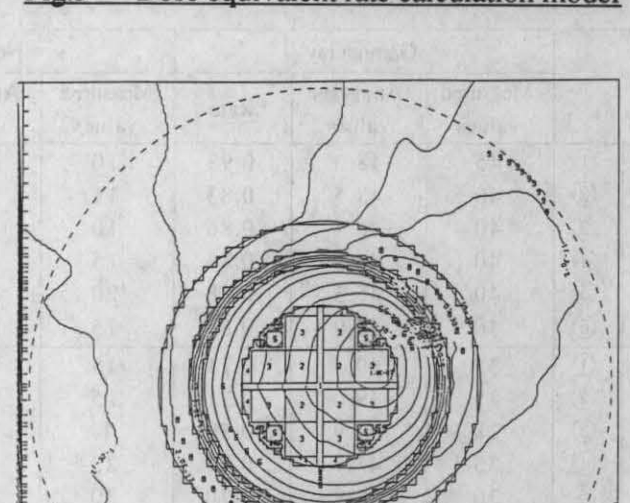

Fig.6 Neutron dose equivalent calculation result