古里原子爐 核燃料의 裝塡方法에 대한 多群擴散的 效果分析

논 문
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Multi-group Diffusion Analysis on Kori Reactor's Fuel Loading Patterns

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Abstract

The multi-group diffusion theory is applied to the analysis of the currently constructing Kori reactor core which is to be refuelled by 3-region fuel loading pattern and also to the comparative study on a conceptually designed 5-region reactor core, under the condition that, apart from the thermal-hydraulic considerations, all the input data referred to herein are assumed to be identical for both cases.

The numerical calculation is carried out for quantitative analysis of the characteristics of the two fuel loading patterns in details, and the calculated results show that, so far as the nuclear aspects are concerned, the characteristics of the 5-region reactor core are proved to be superior to those of Kori's 3-region reactor core in general.

1. Introduction

Along with the development of a considerable number of computer codes for reactor design and management, a good many reactor engineers have conducted the reactor calculation with great success by solving the multi-group diffusion equation in numerical methods. An analytical study is made in this paper with regard to the numerical results obtained from computer calculation in an attempt to compare the nuclear characteristics of Kori reactor's 3-region core with those of a 5-region reactor core which is conceptually designed for this comparative analysis.

For the convenience' sake, the nuclear input data for the conceptually designed 5-region core are assmed to be identical with those for Kori reator's 3-region core; however, thermal-hydraulic considerations are exempted from being taken into account herein. Main efforts are employed for the clarification of quantitative descriptions of the neutron flux, power distribution, peaking factor, uranium burn-up rate and the production of fissionable nuclides for two types of reactor core each refuelled in different loading patterns by means of a one dimensional computer code, entitled RELOAD FEVER, developed and compiled by Gulf General Atomic, Inc.

To begin with, the multi-group diffusion theory is discussed, on the ground that this theory is the very backbone in dealing with the present topic from the theoretical aspects. At the same time, discussion is made regarding the reactor models chosen for the reactor calculation. The results of the calculation are then presented both in tables and figures to considerable details. Finally these results are analysed and reviewed with the conclusion that the 5-reigon reactor core is advantageous over the 3-region reactor core by and large; however, the former could result in some unfavorable factors than Kori reactor core from the viewpoints of refuelling frequency or consequent reactor shutdowns and the like.

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2. Multi-group Diffusion Theory

One of the most powerful means of treating a multi-region reactor is to make use of the group diffusion theory. In this theory the energies of all the neutrons are divided into a number of groups. The diffusion equation for each energy group in case of time independent is given by[1]

$$D_1\nabla^2\phi_1(\vec{r})-\Sigma_{a1}\phi_1(\vec{r})-[\sum_{h=2}^{N}\Sigma(1\rightarrow h)\phi_1(\vec{r})$$

$$+X_1\sum_{h=1}^{N}\Sigma_{fh}\phi_h(\vec{r})=0$$

and

$$D_g\nabla^2\phi_g(\vec{r})-\Sigma_{ag}\phi_g(\vec{r})-\sum_{h>g}^{N}\Sigma(g\rightarrow h)\ \phi_g(\vec{r})$$

$$+\sum_{h=1}^{g-1}\Sigma(h\rightarrow g)\phi_h(\vec{r})$$

$$+X_g\sum_{h=1}^{N}\nu_h\Sigma_{fh}\phi_h(\vec{r})=0 \qquad (1)$$

where

the first term : diffusion term

the second term : the loss of neutrons due to absorption,

the third term : the loss of neutrons as the result of scattering from the g_{th} group to all the higher numbered (lower energy) groups.

the fourth term : the number of neutrons scattered into the g_{th} group from all the lower numbered (higher energy) groups:

the last term : the number of fission neutrons produced in the g_{th} group resulted from fissions in all other groups.

X_g : fission spectrum of the g_{th} group

ν_h : number of neutrons produced per fission

Σ_{fh} : macroscopic fission cross-section of the h_{th} group.

Equation (1) represents a set of N-coupled partial differential equation; there is one set of such equations for each region of the reactor core. To proceed with the calculation, it is assumed that the extrapolationdistance at the surface of the reactor is independent of energy and hence is same for all groups, that is;

$$\phi_1(R)=\phi_2(R)=\cdots\cdots=\phi_N(R)=0 \qquad (2)$$

where R is the extrapolated radius. It is practically almost impossible to solve these equations for a multi-region reactor by a conventional method analytically, so the numerical solution has to be carried out on a computer.[11]

3. Reactor Models

Let us consider a cylindrically shaped reactor which consists of fuel loading patterns as shown in Fig. 1, and Kori reactor core is to be loaded in such a fuel arrangement.[2][3] In order to carry out the calculation as accurate as possible, the reactor core is divided into 8 different annulus regions as in Fig. 2.

The main objective of such division in annulus regions is to calculate the respective neutron flux distribution of each energy group as well as the power distribution by solving Eq.(1). The solution of this equation then gives the burn-up rate of uranium and the production rate of fissionable nuclides at respective time step.

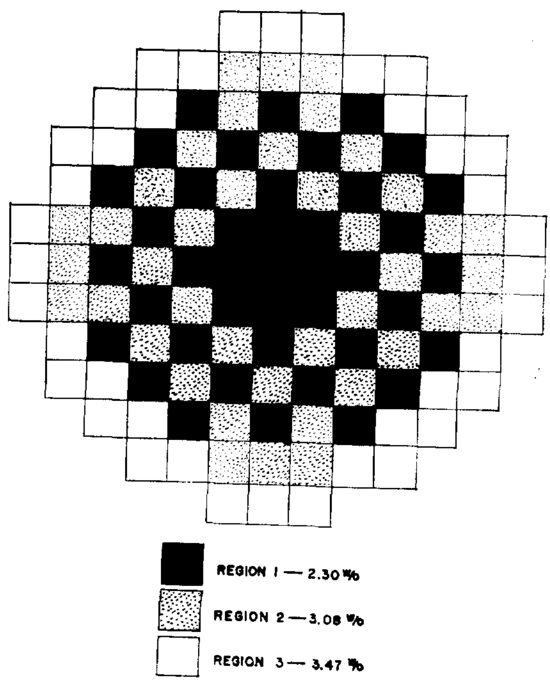
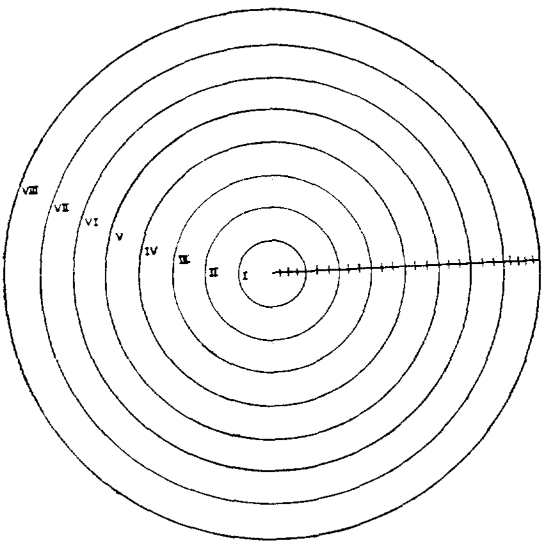


그림 1. 3영역식 연료장전

Fig. 1. Three region core loading.

As mentioned in section 2, Eq. (1) satisfies the conditions of each region, and the neutron energy spectrum is assumed to be continuous. In treating this problem, the entire neutron spectra are divided into 4 groups, namely, two fast energy groups, one epithermal energy group and thermal energy group.



Region	Outer Radius(cm)
I	40.02
II	55.71
III	71.34
IV	84.12
V	92.55
VI	100.27
VII	122.56
VIII	155.58

그림 2. 각 영역의 실제반경
Fig. 2. Effective radius of each region.

In Eq. (1), the constants $\Sigma(g\rightarrow h)$ are the group transfer cross-sections which describe the transfer of neutrons by scattering from one group to another. These are defined so that $\Sigma(g\rightarrow h)\phi_g(\vec{r})$ is equal to the number of neutrons transferred from g_{th} to h_{th} group per cm^3/sec at the point $\vec{r}$.

The value of these constants depends on both the nuclear properties of materials in the reactor and the number of groups adapted in the calculation.[4][5]

It is assumed that neutrons from one group can only be scattered into the adjacent group; i.e., they cannot skip the neighboring energy group. In this case

$\Sigma\,(g\rightarrow h)$ is zero unless $h=g+1$. Hence Eq. (1) reduces to

$$D_1\nabla^2\phi_1(\vec{r})-\Sigma_{a1}\phi_1(\vec{r})-\Sigma(1\rightarrow 2)\phi_1(\vec{r})$$
$$+X_1\sum_{h=1}^{4}\nu_h\Sigma_{fh}\phi_h(\vec{r})=0$$

and

$$D_g\nabla^2\phi_g(\vec{r})-\Sigma_{ag}\phi_g(\vec{r})-\Sigma(g\rightarrow g+1)\phi_g(\vec{r})$$
$$+\Sigma(g-1\rightarrow g)\phi_{g-1}(\vec{r})$$
$$+X_g\sum_{h=1}^{4}\nu_h\Sigma_{fh}\phi_h(\vec{r})=0 \qquad (3)$$

Using Eq. (3), it can be made possible to undertake the approximate reactor calculation for any reactor geometry. By means of this equation, calculation is made for both reactor cores, namely, 3-region and 5-region, and then the calculated results of these two reactor systems are analysed and compared in pros and cons.

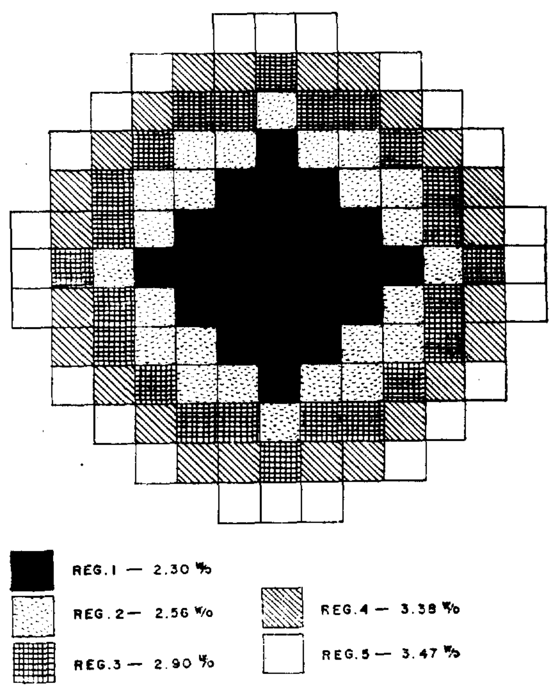


그림 3. 5 영역식 연료장전
Fig. 3. Five region core loading.

The fuel loading pattern of 5-region reactor core is shown in Fig. 3.

4. Calculation Results and Comparison

As described above, Eq. (3) is applied in this section for the approximate reactor calculation for the 3-region (Figs. 4, 5) and 5-region reactor core (Figs. 4, 5), respectively. The calculation results show how the dominant nuclear factors in power reactors, such as neutron flux distribution, power distribution, power peaking factor, burn-up rate, the production of fissionable nuclides, etc., are affected by the fuel loading patterns. Since the present computation work is beyond the boundary of the conventional manual calculation method, all the work herein is carried out by a computer using the following input data supplied by Korea Electric Company and Westinghouse Electric Corporation:

Table 1. Input data supplied by KEC & WEICO
표 1. 한전과 WEICO에서 제공한 입력데이타

Reactor	
Output	1728.5 MWt
Diameter (actual)	96.5 in
Height (actual)	144 in
Fuel	
Material	UO_2
No. of assembly	121
Fuel rods per assembly	179
Clading material	Zircaloy
Refueling	
Method	Out-in refueling
Enrichment of feed material	3.4%
Shut-down time	15 days
Plant factor	0.8

4—1. Neutron flux distribution

Equation (1) describes the neutron flux for a multi-group reactor. However, since the neutron energy spectra here are subdivided only into four energy groups, Eq. (1) is reduced to Eq. (3) for generalization. Solving Eq. (3), it is then possible to obtain $\phi_i(\vec{r})$ as well as the total neutron flux $\Phi(\vec{r})$.

$$\Phi(\vec{r})=\sum_{i=1}^{4}\phi_i(\vec{r}) \tag{4}$$

where $\phi_i(\vec{r})$ is the neutron flux of the i_{th} group.

Using this equation and making use of the computer code entitled RELOAD FEVER together with data from REVISED GAD, totally 192 sets of computation are performed for the neutron flux distribution each consisting of 50 mesh points for the two different reactor systems in question. [8][9][10] For the simplification, however, only one representative figure describing the total neutron flux, which is drawn up by combining all the calculated data of energy range from the fast to the thermal group, is plotted in Fig. 4.

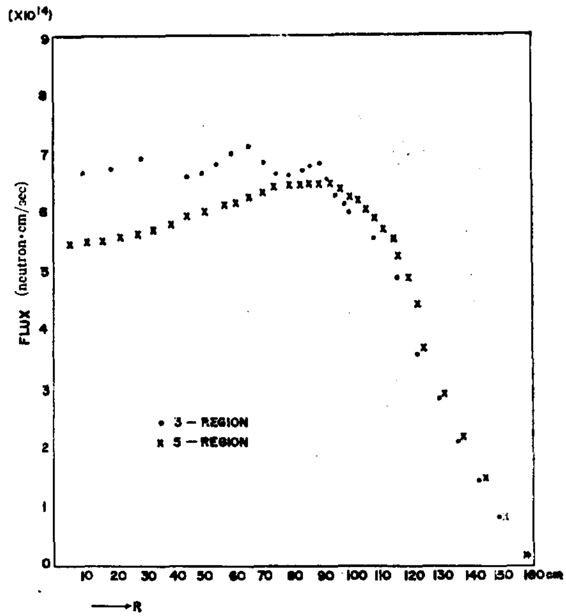


그림 4. 평형상태에서의 중성자속분포
Fig. 4. Flux distribution in equilibrium state.

The figure shows that the neutron flux distribution of the 5-region core is more smoothly flattened than that of the 3-region, especially in the middle part of the reactor core, and this phenomenon is more obvious in the case of the initial loading. But the flux distribution near the edge of the reactor is almost same for both cases.

4-2. Power distribution and peaking factor

The power distribution is as important as the neutron flux distribution in the operation and management of power reactors.[6][7] In general it is expressed by the following equation:

$$P(\vec{r})=\gamma\sum_{i=1}^{4}\Sigma_{fi}(E_{if},\vec{r})\phi_i(\vec{r}) \tag{5}$$

where γ is the recoverable energy and $\Sigma_{fi}(E_i,\vec{r})$ is the fission cross-section of th i_{th} group. Fig. 5 shows the power distribution when the reactor is assumed to be in operation at the equilibrium state.

As shown in Fig. 5, three peakings appear in the 3-region reactor core, and, as is well known, these phenomena are most undesirable in the power reactors. Moreover, the worst peaking factors are seen in the initial loading of the 3-region core, in which the peak-to-average power density ratio reaches 1.321, while that of the 5-region at the same time step is merely 1.217. Such peakings should be lowered by all means.

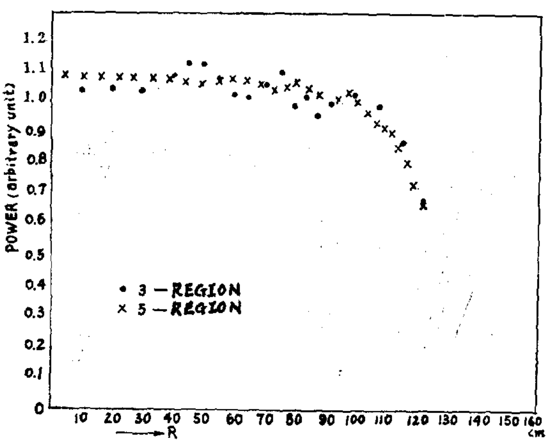


그림 5. 평형상태에서의 power 분포
Fig. 5. Total power distribution at equilibrium state.

Despite that 50 mesh point readings are covered ranging from the core center up to 160 cm distance for the neutron flux distribution there are no datum readings between 120-160 cm line for the power distribution, owing to the fact that power is actually generated mainly in the active core which stretches out only up to the 7th annulus region (122.56 cm from the center as in Fig. 2)

4-3. Burn-up rate and the production of fissionable nuclides

By means of utilizing the results of neutron flux distribution obtained in the previous section, it now becomes possible to get the fuel burn-up rate and the production rate of fissionable nuclides from the following equation.

$$\frac{dN}{dt}=-N(\sigma_a\phi+\lambda)+N'(\sigma_c\phi+\lambda)'+N'(\sigma_c\phi+\lambda)'' \tag{6}$$

where the first term on the right hand represents the depletion of nuclide N and the second and third terms account for the production of nuclide N by capture or decay of N' and N'', respectively.

What is important in this section is how to achieve the most effective fuel burn-up when aiming at the generation of given power output, and also how to produce the maximum possible Pu-239 nuclide which is one of the most indispensable fissionable plutonium isotope for the nuclear weaponry.

Tables 2 and 3 show the burn-up and the production of Pu-239 and Pu-241 nuclides calculated by Eq. (6). Hence, Table 2 deals with the case of 3-region reactor, whereas Table 3 represents the results of 5-region reactor core.

The refuelling time for the 3-region reactor differs from that for the 5-region. For instance, the ending time of the fourth reload fuel for the 3-region approximately corresponds to the ending time of the sixth reload fuel for the 5-region reactor. Therefore, comparative analysis is attempted herein for U-235 consumption and plutonium production between these two cases, namely, as of the end fourth reload of the former and the end of the sixth reload of the latter. The computed results show that the entire consumption of U-235 from the initial up until the end of the fourth reload for the case

Table 2. Depletion of U-235 and Production of Fissionable Nuclides in 3-region Reactor Core
표 2. 3영역식 원자로에서의 U-235의 연소량과 핵분열성 물질의 생성량

	Time Step	Time (days) Calculated	Time (days) Calendar	U-235 (kg)	Pu-239 (kg)	Pu-241 (kg)
Initial Loading	0	0	0	1,387.10	0.00	0.00
	1	124.8	156	1,122.90	64.54	1.49
	2	249.6	312	896.30	107.19	7.47
	3	374.4	468	702.75	133.01	16.88
1st Reloading	0	389.4	483	1,079.00	89.41	9.39
	1	476.6	592	924.84	113.13	13.86
	2	563.8	701	786.00	129.77	19.13
	3	651.0	820	661.75	142.69	25.05
2nd Reloading	0	666.0	583	1,062.10	92.59	12.42
	1	758.8	951	900.45	115.02	16.77
	2	851.6	1,067	756.09	133.63	22.01
	3	944.4	1,183	628.05	146.02	27.97
3rd Reloading	0	959.4	1,198	1,050.40	94.11	12.54
	1	1,053.0	1,315	888.46	116.24	17.10
	2	1,146.6	1,412	744.17	134.56	22.50
	3	1,240.2	1,549	616.50	146.66	28.56
4th Reloading	0	1,255.2	1,564	1,038.90	94.89	13.24
	1	1,348.0	1,680	879.07	116.72	17.71
	2	1,440.8	1,796	736.78	134.80	22.90
	3	1,533.6	1,912	610.90	146.74	28.92
TOTAL PRODUCTION				−2,397.55	344.12	79.79

Table 3. Depletion of U-235 and Production of Fissionable Nuclides in 5-region Reactor Core.
표 3. 5영역식 원자로에서의 U-235의 연소량과 핵분열성 물질의 생성량

	Time Step	Time (days) Calculated	Time (days) Calendar	U-235 (kg)	Pu-239 (kg)	Pu-241 (kg)
Initial Loading	0	0	0	1,405.30	0.00	0.00
	1	74.88	93.6	1,241.10	42.09	0.37
	2	149.76	187.2	1,091.50	76.19	2.18
	3	224.64	280.8	954.51	101.75	5.68
1st Relading	0	239.64	295.8	1,128.70	81.21	3.85
	1	291.96	361.2	1,031.60	95.44	6.33
	2	344.28	426.6	939.86	110.09	9.30
	3	396.60	492.0	853.51	122.17	12.75
2nd Reloding	0	411.60	507.0	1,059.40	96.80	8.65
	1	467.28	576.6	960.62	109.48	11.61
	2	522.96	646.2	867.96	122.61	14.9
	3	578.64	715.8	781.31	133.16	18.66

3rd Reloading	0	593.64	730.8	1,006.50	105.28	12.55
	1	649.80	801.0	909.65	116.94	15.52
	2	705.96	871.2	819.27	129.03	18.83
	3	762.16	941.4	735.06	138.57	22.47
4th Relading	0	777.12	956.4	966.83	109.00	14.91
	1	832.80	1,026.0	872.51	120.07	17.83
	2	888.48	1.095.6	784.73	131.61	21.06
	3	944.16	1,165.2	703.15	140.63	24.59
5th Reloading	0	959.16	1,180.2	948.01	110.49	15.84
	1	1,014.84	1,249.8	854.63	121.30	18.77
	2	1.070.52	1,319.4	767.85	132.60	21.99
	3	1,126.20	1,389.0	687.32	141.40	25.50
6th Reloading	0	1,141.20	1,404.0	935.21	111.36	16.63
	1	1,196.88	1,473.6	842.34	122.07	19.53
	2	1,252.56	1,543.2	756.15	133.26	22.72
	3	1,308.24	1,612.8	676.28	141.93	26.20
TOTAL PRODUCTION				−2,012.56	305.47	63.42

of 5-region turns out to be 2,012.56 kg which is 385 kg less than the case of 3-region during the same period of time. On the other hand, the production of Pu-239 and Pu-241 nuclides for the case of 5-region is 305.47 kg and 63.42 kg which are 38.65 kg and 16.37 kg less than the case for the 3-region reactor, respectively.

The reason how the same thermal rating could be achieved with less amount of U-235 consumption in the case of 5-region reactor is ascribed to the fact that the difference in U-235 consumption between two systems is filled up with the burn-up of plutonium nuclides which have been produced from U-238 during the reactor operation. Nevertheless the production rate of fissionable plutonium muclides per U-235 consumption is better for the case of 5-region.

6. Conclusion

In this paper, the multi-group diffusion equation is used for the reactor calculation. With the numerical results obtained from the solution of this equation, effort is employed for the analysis and comparison of the characteristics of two reactor concepts in question so as to clarify the advantages and disadvantages of the two types of fuel loading patterns in a pressurized water reactor. It is attributed to this study that, aside from the thermal-hydraulic and metallurgical considerations, the nuclear performance characteristics of the 5-region reactor core is better than that of the 3-region, especially from the viewpoints of enrichment and burn-up rate. Since the enrichment, burn-up, neutron flux distribution, plutonium production, etc., are directly associated with fuel economy, and that peaking factor is related with reactor safety, the proposed 5-region reactor core can be said to be advantageous over Kori's 3-region reactor, the reason being that the former could be operatable with less amount of identically enriched U-235 loading, or enrichment could be lowered than for the latter, in the case of achieving the same thermal output.

At the same time it is discovered as the result of analysing the neutron flux and the power distribution of the two reactor cores that the peaking factor in the fuel assemblies of the 5-region core is lower than that of the 3-region, especially right after the reactor start-up, which naturally brings about the improvement of reactor safety to a great extent. In addition, the computation regarding the fuel burn-up rate and the production of fissionable nuclides shows that better achievement of fuel burn-up could be made possible by loading fuels in 5-region pattern, and the relative production rate of plutonium for the 5-region reactor is more than that fo the 3-region reactor.

Theoretically it is more than true that a reactor could benefit from having more number of fuel regions if and when the nuclear aspects are of the prime importance. However, such beneficial consideration cannot be extrapolated further in a light water reactor, owing to the fact that the increase in number of fuel regions certainly calls for more refuelling frequency, and needless to say, more refuelling frequency or more reactor shutdown endangers the plant factor of nuclear power station. These two factors are thus contrary to each other. Therefore, it is advisable that an attempt should be made for the optimization of the number of fuel regions and the refuelling frequency.

Furthermore, all other different aspects, such as thermal, hydraulic, metallurgical, operational, economical, etc. must be combined together with nuclear considerations so that the best possible conditions for the reactor design and operational performance might be sought through the reiteration of reactor calculation. Due to the limited information and especially because of tremendous complication and work volume, this work remains untouched in this paper, for which continuous attention is preferred to be paid in the future.

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