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Development of One Dimensional Kinetics Program

Chan Bock Lee and Chang Hyun Chung
Seoul National University

Bub Dong Chung
Korea Advanced Energy Research Institute

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일차원 동특성 프로그램 개발

이 찬 복·정 창 현
서울대학교

정 법 동
한국에너지연구소

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Abstract

A one dimensional neutron kinetics program, B1K which is applicable to the safety analyses of PWR's is developed to analyze the reactor core in axial dimension. The B1K employs the finite difference technique in space and θ-time integration method in time. Detailed models for the Doppler and moderator feedbacks and control rod motion are included. The benchmark of the nuclear model is carried out through the ANL benchmark problem and the time dependent nuclear power change in the rod ejection accident of KNU1 is calculated by B1K code.

The results indicate that the B1K can predict the neutron dynamics with fair accuracy within the limits of one dimensional analysis and it is useful for the safety analyses of PWR's.

요 약

원자로 노심을 축방향으로 일차원 해석을 하고, 가압경수로 형원자로의 안전성 해석에 적용할 수 있는 중성자 동특성프로그램 B1K를 개발하였다. B1K프로그램내에서 공간변수에 대해서는 유한차분법이, 시간변수에 대해서는 θ-시간적분법이 채택되었다. 또한 도플러 및 감속재 궤환과 제어봉구동 등을 자세히 묘사하는 모델들이 포함되었다. 핵모델의 검증은 ANL검증문제를 통해 이루어졌고, 고리 1호기의 제어봉 인출사고시의 노심출력 변화를 계산하였다.

이상의 계산결과 B1K동특성프로그램이 노심의 중성자 속 변화를 일차원해석의 한계내에서 비교적 정확하게 묘사할 수 있으며, 가압경수로형 원자로의 안전성 해석에 유용하게 사용될수 있다는 것이 증명되었다.

1. Introduction

The knowledge of a transient power behavior of a core is essential to the safety analysis of the reactor. Normally, one point approximation of a core has been used frequently because of its economy in computing time and it is well known[1,2] that it leads to more conservative results than actual. However, as the core power density increases, it becomes necessary to determine accurately the true dynamic response of the reactor in order to reduce the unnecessary safety margins in the reactor design. Although higher dimensional analysis gives the most accurate results, it isn't used usually in the analysis of an actual core, because it consumes a large amount of computing time.

In this work, a one dimensional kinetics program B1K is developed which analyzes a core in axially one dimension and can predict the control rod motion and the moderator feedback more efficiently than radially two dimensional analysis of a core.

To analyze the transient behavior of a core, both the thermal hydraulic and neutronic behavior should be modelled and the resulting equations should be solved simultaneously. The driving terms and coefficients in the thermal hydraulic model are connected to the time dependent neutron fluxes. The coefficients in neutron equa tions such as D's and Σ' s are in turn functions of the local thermal hydraulic condition. This nonlinear problem in both the thermal hydraulic and neutron flux parameters can be solved such that over a sequence of small time intervals a thermal hydraulic computation is done by using the local flux at the beginning of the interval, and then a neutron flux is calculated based upon the just-calculated thermal hydraulic conditions such as local temperature and density.

The B1K employs the finite difference technique for the solution of both the two group neutron diffusion equations and the transient heat transfer equations. In each mesh, the fuel temperature and fuel to moderator heat transfer are calculated by using volume averaged heat generation rate.

Since Doppler and moderator feedback impose strong influences upon the transient core behavior, the detailed fuel-clad-coolant heat transfer model is included, and the same time step size is used for both the nuclear and heat transfer calculations. This program also employs other features to describe the control rod motion, xenon concentration change and decay heat generation.

2. Description of the B1K Program

2.1. Nuclear Model

Time dependent two group neutron diffusion equations are as follows.

Fast Group

$$\frac{1}{v_1}\frac{d}{dt}\phi_1=\frac{1}{k}(1-\beta)[\nu_1\Sigma_{f1}\phi_1+\nu_2\Sigma_{f2}\phi_2]+\bar{I}\sum_{i=1}^{6}\lambda_iC_i-\nabla\cdot J_1-(\Sigma_{a1}+\Sigma_{r1})\phi_1, \quad (1)$$

Thermal Group

$$\frac{1}{v_2}\frac{d}{dt}\phi_2=\Sigma_{r1}\phi_1-\nabla\cdot J_2-\Sigma_{a2}\phi_2, \quad (2)$$

Precursors

$$\frac{d}{dt}C_i=\frac{1}{k}\beta_i[\nu_1\Sigma_{f1}\phi_1+\nu_2\Sigma_{f2}\phi_2]-\lambda_iC_i \quad (3)$$

Application of the finite difference method to the space variable and θ-difference method to the time variable results in finally the following tridiagonal matrix equation. Detailed derivations are given in Ref. (3).

$$C_n+A_n\phi_{n+1}+B_n\phi_{n-1}-\alpha_n\phi_n=0 \quad (4)$$

where A_n, B_n, C_n and α_n are 2×2 matrices and ϕ_n is a vector composed of ϕ_1 and ϕ_2.

The solution procedure for Eq. (4) is as follows.

$$Q_n=\beta_nA_n,$$
$$M_n=\beta_n(C_n+B_nM_{n-1}),$$
$$\beta_n=(\alpha_n-B_nQ_{n-1})^{-1},$$
$$Q_0=M_0=0. \quad (5)$$

The M_n and Q_n are calculated by the forward sweep and then ϕ_n is calculated by back substitution in the backward sweep as follows.

$$\phi_n=M_n+Q_n\phi_{n+1}.$$

2.2. Thermal Hydraulic Model

The one dimensional transient energy equation

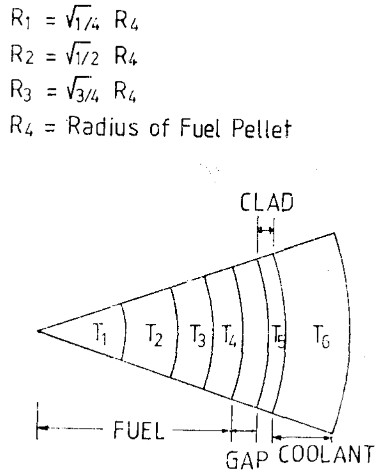

Fig. 1. Six Region Fule Model

for each region in six region fuel model as shown in Fig. 1 is given as follows.

$$\rho\frac{\partial}{\partial t}h-q'''-\nabla_r\cdot k\nabla_r T=0,\quad \text{Fuel}$$

$$-\nabla_r\cdot k\nabla_r T=0,\quad \text{Gap}$$

$$\rho\frac{\partial}{\partial t}h-\nabla_r\cdot k\nabla_r T=0,\quad \text{Clad}$$

$$\frac{\partial}{\partial t}h+G\nabla_x h-q'''$$
$$-\nabla_r\cdot k\nabla_r T-\frac{1}{J}\cdot\frac{\partial}{\partial t}P=0.\quad \text{Coolant} \qquad (6)$$

In each of the eq. (6), application of the finite difference technique to the space and the time results in the following tridiagonal matrix equation.

$$C_n+a_nT_{n+1}+b_nT_{n-1}-\alpha_nT_n=0, \qquad (7)$$

where a_n, b_n, c_n and α_n are coefficients and T_n is the average temperature in the n-mesh.

The solution procedure for Eq. (7) is equal to that for Eq. (4).

2.3. Feedback and Control Rod Model

From the SHA's correlation,[4] the resonance effective temperature for Doppler feedback is calculated by

$$T_{eff}=(1-w_c)T_{mod}+w_c[w_p/4\ (T^1_{fuel}+T^2_{fuel}$$
$$+T^3_{fuel})+(1-3w_p/4)T^4_{fuel}], \qquad (8)$$

where the factors w_p and w_c are the pellet weighting factor and the core statistical weighting factor respectively.

The correction of the fast absorption cross section is given by

$$\Sigma_{a1}=\Sigma^{ref}_{a1}+b\ (T^{1/2}_{eff}-T^{1/2}_{ref}), \qquad (9)$$

where the temperatures T_{eff} and T_{ref} are in degrees absolute and the factor b is a proportional constant.

The corrections of the macroscopic cross sections due to the change of moderator density are given as follows.

$$\Sigma_{a1}=\Sigma^{ref}_{a1}+\delta N^W\ \sigma^W_{a1}+\delta N^B\ \sigma^B_{a1},$$

$$\Sigma_{a2}=\Sigma^{ref}_{a2}+(\delta N^W\sigma^W_{a2}w_a+\delta N^B\ \sigma^B_{a2}w_a),$$

$$D_1=D^{ref}_1/(1+3\times D^{ref}_1\delta N^W\ \sigma^W_{tr,1}),$$

$$D_2=D^{ref}_2/(1+3\times D^{ref}_2\delta N^W\ \sigma^W_{tr,2}w_a), \qquad (10)$$

where the factor w_a is the water advantage factor and the δN^W and δN^b are the number density changes of the water and boron that are homogenized over the mesh.[5]

The corrections of the fast group cross sections due to the neutron energy spectrum hardening by the decrease of the moderator density are given by

$$D_1=D_1-D^{ref}_1\times\delta\Sigma/(1+\delta\Sigma),$$

$$\Sigma_{a1}=\Sigma_{a1}+\delta\Sigma\times\Sigma^{ref}_{a1},$$

$$\nu_1\Sigma_{f1}=\nu_1\ \Sigma^{ref}_{f1}+\delta\Sigma\times\nu_1\Sigma^{ref}_{f1}, \qquad (11)$$

where

$$\delta\Sigma=\alpha\times(\rho_{actual}/\rho_{ref}-1.0)$$

and α is a proportional constant.

The correction of the removal cross section due to the change of the fast group absorption cross section is given by[6]

$$\Sigma_r=\Sigma_{a1}/(\mathrm{EXP}(\Sigma_{a1}/\Sigma_m-1)], \qquad (12)$$

where Σ_m is the moderation cross section and its correction for the water number density change is given by

$$\Sigma_m=\Sigma^{ref}_m+\sigma^W_m\delta N_W w_a. \qquad (13)$$

The cross section changes due to the control rod insertion are described by[7]

$$\Sigma_{a2}=\Sigma_{a2}+\text{ABSCON}\times\text{FCON},$$
$$\Sigma_{a1}=\Sigma_{a1}+\text{TCON}\times\text{ABSCON}\times\text{FCON},$$
$$D_1=D_1\times(1+\text{SCON}(1)\times\text{ABSCON}\times\text{FCON}),$$
$$D_2=D_2\times(1+\text{SCON}(2)\times\text{ABSCON}\times\text{FCON}),\qquad(14)$$

where ABSCON, TCON, SCON(1) and SCON (2) are the correction factors for the control rod insertion and FCON is the control rod insertion fraction in the mesh.

3. Benchmark of Nuclear Model

ANL benchmark problem 7416[8] tests for the one dimensional kinetics solution of infinite slab reactor model which does not have the feedback effect.

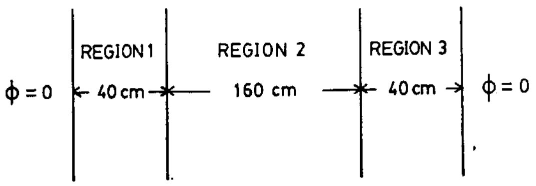

Fig. 2. BSS-6 Configuration and Boundary Condition

The reactor has three sections as shown in Fig. 2. ; a low enrichment central region and two identical high enrichment end sections.

Table 1. Total Power Change vs. Time Step Size (Δt)

Δt	10^{-3}(sec)			10^{-2}(sec)			10^{-1}(sec)		
Time	RAUM-ZEIT	B1K		RAUM-ZEIT	B1K		RAUM-ZEIT	B1K	
		θ=0.5	θ=1.0		θ=0.5	θ=1.0		θ=0.5	θ=1.0
0.0	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
0.1	0.9299	0.9296	0.9299	0.9299	0.9268	0.9299	0.9321	0.8637	0.9301
0.2	0.8733	0.8732	0.8734	0.8733	0.8709	0.8734	0.8718	0.8803	0.8736
0.5	0.7597	0.7598	0.7599	0.7597	0.7585	0.7599	0.7608	0.7205	0.7600
1.0	0.6588	0.6590	0.6591	0.6588	0.6584	0.6591	0.6584	0.6712	0.6592
1.5	0.6432	0.6435	0.6435	0.6432	0.6434	0.6435	0.6435	0.6311	0.6436
2.0	0.6307	0.6310	0.6310	0.6307	0.6309	0.6310	0.6305	0.6388	0.6310

Table 2. Region and Total Power Change ($\Delta t=10^{-3}$ sec)

Time	Wigle				QX1				B1K			
	Total-Power	Region Power			Total-Power	Region Power			Total-Power	Region Power		
		1	2	3		1	2	3		1	2	3
0.0	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
0.1	0.9298	0.8621	0.9339	0.9910	0.9298	0.8621	0.9340	0.9910	0.9299	0.8622	0.9341	0.9911
0.2	0.8732	0.7520	0.8804	0.9830	0.8733	0.7521	0.8805	0.9831	0.8734	0.7522	0.8806	0.9832
0.5	0.7596	0.5336	0.7724	0.9655	0.7597	0.5336	0.7724	0.9655	0.7599	0.5339	0.7726	0.9658
1.0	0.6588	0.3452	0.6753	0.9462	0.6588	0.3452	0.6753	0.9463	0.6591	0.3454	0.6756	0.9467
1.5	0.6432	0.3235	0.6587	0.9381	0.6433	0.3235	0.6588	0.9383	0.6435	0.3237	0.6591	0.9386
2.0	0.6306	0.3066	0.6455	0.9311	0.6307	0.3066	0.6454	0.9312	0.6310	0.3068	0.6458	0.9317
		Initia Region Power				Initial Region Power				Initial Region Power		
		1	2	3		1	2	3		1	2	3
		0.2790	0.4421	0.2790		0.2790	0.4421	0.2790		0.2791	0.4419	0.2791

The perturbation is that Σ_{a2} in region 1 increases linearly by 3% in 1 second. The calculated results and reference solutions for this subcritical transient problem are given in Tables 1 and 2. The comparisons are made for the region power changes and the sensitivity of the time step size change. The latter as given in Table 1 is compared with the RAUMZEIT[9] that uses the time integration method and the former as given in Table 2 is compared with the WIGLE [10] and QX1[11] that uses quasi-static method.

As can be seen in Table 1, the time difference parameter, $\theta=1.0$ is more stable than $\theta=0.5$ in the change of the time step size. The present results are in general agreements with the reference solutions within 0.5% difference.

The verification of the feedback model is carried out by the calculation of the reactivity coefficients and the comparison with the results of KNU1 design report. This is shown in Ref. (15)

4. Calculation of Rod Ejection Accident

By using the nuclear data of KNU1 obtained from the KIDD[12] output and the thermal hydraulic data from the TOODEE2[13] and FACTRAN[14] codes, Doppler and moderator reactivity coefficient and control rod worth in normal operational condition are calculated. The results which are in good agreement with Westinghouse design values, are given in Ref. (15). Using these reactivity coefficients and the initial conditions given in FSAR 15.4, KNU1, nuclear power change in the rod ejection accident is calculated by B1K code.

Tables 3 and 4 show the sensitivity of both the time step size and θ-value at the HZP condition. Comparisons are made of the peak power levels reached in the HZP and HFP rod ejection. Since the peak power is sensitive to both the calculational model and thermal hydraulic feedback model, it is an indicator of the accuracy of the calculation. It is seen from Tables 3 and 4 that the time difference paramcter $\theta=0.5$ is a more desirable choice over $\theta=1.0$ and from Table 3 that once a suitable time step size is selected, a further reduction in the time step size does not improve the results significantly.

As shown in Table 4, if a time step size, Δt, is greater than a specified value, the solutions diverge. The reason is that as the time step size increases, the solutions are overestimated due to the overshooting phenomenon, and the overestimated power results in the more negative feedback effect enough to make the solutions diverge. Since, as generally known [17], the degree of overestimation in case $\theta=1.0$, is greater than

Table 3. Peak Power vs. Time Step Size at HFP Rod Ejection Accident

Δt	$\theta=0.5$			$\theta=1.0$		
	Peak Time	Peak Power	Energy+ Release	Peak Time	Peak Power	Energy Release
0.05	0.1	1.5681	0.3703	0.2	1.4828	0.3603
0.01	0.11	1.4796	0.3525	0.12	1.4790	0.3506
0.005	0.105	1.4783	0.3503	0.115	1.4785	0.3493
0.002	0.106	1.4781	0.3489	0.108	1.4782	0.3485
0.001	0.106	1.4780	0.3484	0.107	1.4781	0.3482
0.0005	0.106	1.4780	0.3482	0.1065	1.4781	0.3481
0.0002	0.106	1.4780	0.3481	0.1062	1.4781	0.3480
0.0001	0.1059	1.4780	0.3480	0.106	1.4780	0.3480

+Integrated Power by 0.25 sec. (Full Power Seconds)

Table 4. Peak Power vs. Time Step Size at HZP Rod Ejection Accident

Δt	$\theta=0.5$			$\theta=1.0$		
	Peak Time	Peak Power	Energy+ Release	Peak Time	Peak Power	Energy Release
0.01	div++	div	div	div	div	div
0.005	0.155	112.28	1.8647	div	div	div
0.002	0.162	105.03	1.8529	0.14	125.83	2.0465
0.001	0.163	104.16	1.8517	0.153	113.35	1.9398
0.0005	0.1635	103.99	1.8516	0.158	108.44	1.8941
0.0002	0.1632	103.98	1.8517	0.1612	105.69	1.8684
0.0001	0.1632	103.98	1.8518	0.1622	104.83	1.8601

+Integrated Power by 0.25 se. (Full Power Seconds)
++Diverge

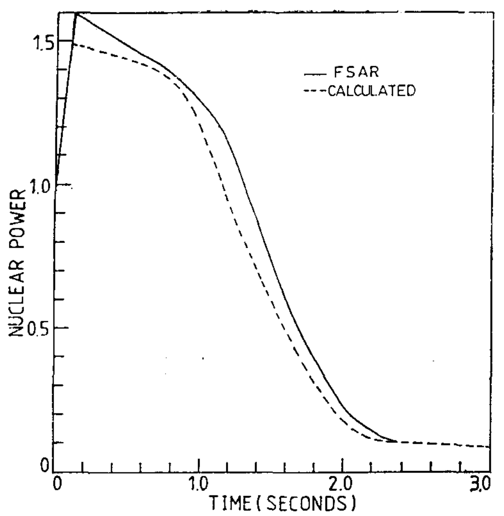

Fig. 3. Nuclear Power Transient BOL, HZP rod Ejection Accident

in case $\theta=0.5$, the divergence comes earlier in case $\theta=1.0$. From Tables 3 and 4, we can deduce the optimum time difference parameter and the optimum time step size, that is, $\theta=0.5$ and $\Delta t=10^{-3}$ sec. at the HFP and HZP rod ejection accident. Fig. 3 shows the nuclear power transient in the HFP rod ejection accident, which is compared with FSAR values[16].

The calculated results of Fig. 3 are lower than the FSAR values. This is due to the difference that the calculation is done with the actual reactivity coefficient at the BOL and the equilibrium

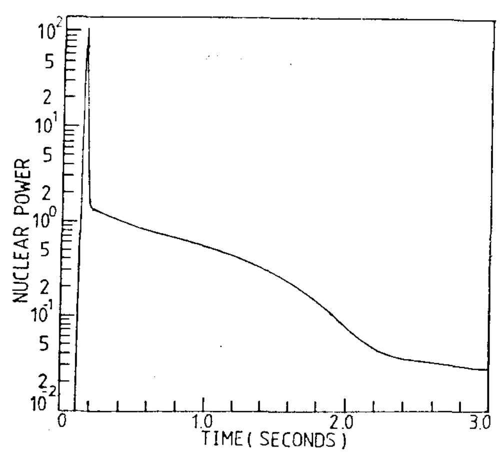

Fig. 4. Nuclear Power Transient BOL, HZP rod Ejection Accident

xenon distribution, while the analysis in the FSAR was made with all the conservatisms such as the conservative reactivity coefficient that is less negative than actual, and the worst xenon distribution, etc. Fig. 4 shows the nuclear power transient in the HZP rod ejection.

5. Conclusion

The objective of this work is to develop the kinetics program that analyze a core axially and be applicable to the safety analyses of PWR's. The program BIK employs the finite difference method for the solutions of both the two group neutron diffusion equations and the transient heat transfer equations. The results of the benchmark of the nuclear model are satisfactory within 0.5% deviation.

The developed program B1K is applied to the rod ejection accident at the HZP and HFP condition. There is a little difference between the calculated results and FSAR values. This arises from the fact that the analysis of the FSAR employs conservative reactivity coefficients. Therefore, the improvement in the selection of appropriate reactivity parameters is required for the safety analysis. In conclusion, it is proved that the axially one dimensional kinetics program B1K can predict the neutron dynamics with fair accuracy within limits of one dimensional analysis and may be useful for the safety analysis.

References

1. W. Ullrich and W. Frisch, "Investigations of Anticipated Transients without Reactor Scram and Other Selected Safety Devices", Nucl. Technology, 41, 185 (1978).
2. L. Moberg et al., "RAMONA Analysis of Peach Bottom-2 Turbine Trip Transients", EPRI NP-1869 (1981).
3. R.F. Barry and D.H. Risher, "TWINKLE: A

Multi-Dimensional Neutron Kinetics Computer Code", WCAP-8028 (1975).

4. W.T. Sha, "Effective Resonance Temperature Correlation of UO_2", Nucl. Appl., 1538 (1965).
5. R.F. Barry and S. Altomare, "The TURTLE 24.0 Diffusion Depletion code", WCAP-7758 (1971).
6. B.J. Jun et al., "KICC: A Space Dependent Cell Spectrum and Depletion Program for Analysis of Light Water Reactor Lattices", KAERI (1983).
7. H.J. Moon, "DD1:D: A One Dimensional Two Group Steady State Neutron Diffusion Theory Program", KAERI (1981).
8. W.M. Stacey and D.A. Meneley, "ANL Benchmark", ANL-7416 (1971).
9. C.H. Adams and W.M. Stacey, Jr. "RAUMZEIT: A Program to Solve Coupled Time-Dependent Neutron Diffusion Equations in One Space Dimension", KAPL-M-6728 (1967).
10. A.F. Henry and A. V. Vota, "WIGLE 2:A Program for the Solution of the One Dimensional Two Group Space Time Diffusion Equations Accounting for Temperature, Xenon and Control Feedback", WAPD-TM-532 (1965).
11. D.A. Meneley, K.O. Ott and E.S. Wiener, "Fast Reactor Kinetic-The QX1 Code", ANL-7769 (1971).
12. S.K. Lee, et al., "KIDD: A KAERI Improved Diffusion Depletion Program for Nuclear Reactor Analysis", KAERI (to be published).
13. G.N. Lauben, "TOODEE2: A Two Dimensional Time Dependent Fuel Element Thermal Analysis Program", NUREG-751 (1975).
14. H.G. Hargrove, "FACTRAN: A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908 (1972).
15. C.B. Lee, "Development of one dimensional kinetics program and its application to KNU1", Master Thesis at S.N.U. (1985).
16. Ko-Ri 1 FSAR Chapter 15.4.
17. O.C. Zienkiewicz, "The Finite Element Method", McGraw-Hll (1977).