

Neutronics Codes Currently Used in Japan for Fast and Thermal Reactor Applications

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Neutronics codes currently used in Japan for core design and core management calculations are reviewed, and the features of the codes are described together with description of reactor core improvement. The review covers pressurized water reactors, boiling water reactors and fast reactors.

KEYWORDS: *neutronics code, Japan, BWR, PWR, FR*

1. Introduction

In Japan there are 29 boiling water reactors (BWRs), 23 pressurized water reactors (PWRs) and 1 fast reactor operating. Besides these, the Japan Atomic Energy Research Institute (JAERI) has several research reactors and the Japan Nuclear Cycle Development Institute (JNC) has an experimented fast reactor JOYO and an advanced thermal reactor FUGEN.

In Japan BWR started in 1970 its operation at TSURUGA-1 reactor. Recently used BWR fuel are step 1~3 fuel. Step 1 fuel is a new 8×8 Zr-lined fuel with assembly averaged burnup of 33GWd/t. The Zr-linear cladding has 0.1mm Zr inside the Zr-2 cladding to improve the integrity for PCI to allow arbitrary rod follow operation. The number of water rod is changed from 1 to 2 to improve void coefficient and local power peaking factor. The use of the new 8×8 fuel together with the uranium saving technique leads to the increase of burnup, and decreases of the fuel cycle cost by about 10 %.

The high burnup 8×8 fuel is called Step-2 fuel. The enrichment is increased together with the design improvement of fuel assemblies. The maximum assembly burnup was increased from 40GWd/t to 50GWd/t. One oversized water rod with volume corresponding to 4 fuel rods is located at the center of an assembly. The water rod leads to decrease of the absolute value of void coefficient.

The 9×9 fuel is called as Step-3 fuel. The maximum assembly burnup is 55 GWd/t, and the average assembly burnup is ~45 GWd/t. The 9×9 type A assembly has two oversized water rod and 74 fuel rods. Among 74 rods, 8 rods are partial length fuel rods with 2/3 height of the standard rods. The 9×9 type B assembly has a central square water channel.

These fuel assemblies are loaded to conventional BWRs and ABWRs. Two ABWR are now operating in Japan from 1996, and several ABWRs are now constructing. ABWRs adopt internal recirculation pump and improved control rod mechanism.

In Japan PWR started its operation in 1970 at MIHAMA-1 reactor. In conventional PWRs, the number of fuel assemblies and primary coolant loops vary with the core's thermal output. There are mainly 4 types of cores, such as 2-loop with 14×14 type 121 assemblies, 3-loop with 15×15 type 157 assemblies, 3-loop with 17×17 type 157 assemblies and 4-loop with 17×17 type 193 assemblies.

The increase in the burnup of fuel is efficient for reducing the fuel cycle cost. In 1990 to achieve the maximum assembly burnup of 48GWd/t, the 4.1% U enriched fuel was loaded to TAKAHAMA-3 reactor.

The increase in the plant capacity factor improves plant efficiency because it reduces power generation costs. The cycle length of ~9 months in the 1970's is now ~13 months, and is planned to be increased further.

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The cluster-type burnup absorber (BA; Pyrex or Alumina-B4C) had been mainly used until ~1990. Since then, uranium fuel pellets with Gd_2O_3 have mainly been adopted to reduce radioactive waste and reactivity penalties. The Gd bearing fuel with Gd_2O_3 content of ~6% was used with the 4.1% U enrichment fuel assembly.

The second step of the high burnup is the extension of maximum assembly burnup to 55 GWd/t. The fuel pellet density will be increased to about 97%. The zircaloy grids instead of inconell grids will be also adopted to reduce neutron absorption. To enhance the capacity of core reactivity, the content of Gd_2O_3 will be increase to about 10% from 6%.

The MOX fuel is planned to be loaded to the KEPCO 3 loop with 17x17 assembly PWRs. The average plutonium fissile content of assembly is about 6%.

APWR has been developed to improve the fuel economy. The APWR has a large-scale core with 257 fuel assemblies of an advanced 17x17 type.

Neutronics code systems currently used in Japan for fast and thermal reactor applications are reviewed and the features of the code systems are described.

2. Neutronics Code for LWR analysis

First, let us review the codes for LWRs.

2.1 BWR Analysis Codes

In recent boiling water reactors, high-burnup fuel, many different enriched fuel, MOX fuel have been utilized or are expected to be utilized. Due to the requirement of the assembly burn-up extension, the enrichment of Gd_2O_3 and the number of Gd rods have been increased. Consequently, the heterogeneity of the assembly becomes much strong.

Therefore, 2-dimensional, few group assembly calculations are required as shown in Fig.1.

Now, as assembly analysis codes, TGBLA[1], NEUPHYS[2] and CASMO[3] are utilized(See Table1).

In TGBLA, the RICM method has been adopted to solve the ultra-fine group slowing down equations in heterogeneous system, and can calculate the resonance shielding effect in good accuracy.

In CASMO code, the method of characteristics can be applied to arbitrary models, and can be extended to whole core models.

Also the useful use of Gd rods leads to asymmetry of the rods loading in a fuel assembly. Thus the regional division of Gd rods is required to treat the flux distribution within Gd rods. TGBLA and CASMO can treat the effect.

For BWR core simulators, LOGOS[4], COS3D[5], PANACEA, SIMULATE[6] and AETNA are currently utilized for BWR core analysis.

The BWR core simulators have been improved by using the few groups nodal method with flux discontinuity factor between assemblies from the conventional modified one-group method, because the interference effect between adjacent assemblies becomes important in recent core designs. LOGOS, COS3D and PANACEA are based on the modified one-group calculation model. However, they have been improved their models as new concept fuels are introduced.

SIMULATE uses the 2-group analytic or polynomial nodal method and AETNA uses the 3 group analytic polynomial nodal method. In addition to the utilization of nodal method, the spectrum interface effect is incorporated. The code system currently

used in GNF-J is TGBLA-LOGOS, and is transferring to LANCER[7]-AETNA.

For the analysis of MOX cores, the reactivity loss due to Pu-241 β decay has to be considered, in the shut down cooling (SDC) model. SIMULATE estimates the reactivity loss by using macroscopic cross section table as a parameter of shut down period. AETNA calculates the number densities of Pu-240,241 and Am-241 by microscopic burn up calculations, and predicts the number densities before and after the shut down to modify the cross section.

For the verification of these assembly and core analysis codes for UO₂/MOX fueled cores, data of UO₂/MOX plant operations and the critical experiments were utilized.

The share of usage of 3D core simulators as core management and/or core design of BWR plants in JAPAN is as follows:

CASMO/SIMULATE	~45%
TGBLA/LOGOS	~41%
NEUPHYS/COS3D	~7%
TGBLA/PANACEA	~7%

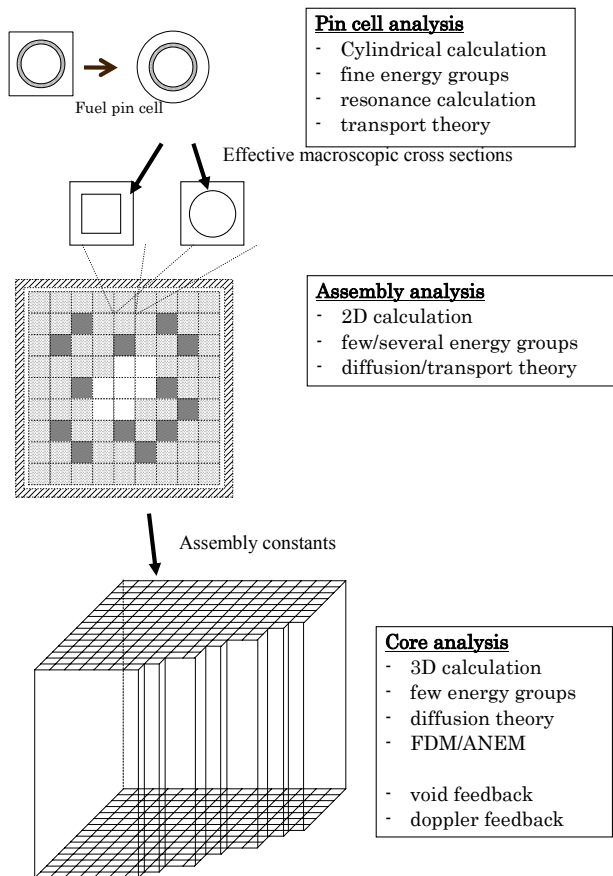


Fig.1 Assembly and Core Analysis Flow for BWR

2.2 PWR Analysis Codes

In PWR cores, the maximum assembly-averaged burnup is going to be extended from 48 to 55 GWd/t as described in Introduction. The MOX fuel was tested and it is expected to be used. The heterogeneity of PWR cores is generally small. However, the use of high content Gd bearing fuel rods and MOX fuel rods increases the heterogeneity in fuel assemblies, and the use of MOX fuel assemblies loaded partly in PWR increases

heterogeneity of cores. 3D design method is to be used for such PWR cores instead of 1D/2D design method as shown in Figure 2.

Conventionally, pin-cell calculations have been used instead of assembly calculations because of the small heterogeneity. However, due to the increase of heterogeneity of fuel assemblies, the assembly calculation codes are recently introduced. PHOENIX-P[8,9], CASMO, improved NULIF[10] codes are currently used for assembly calculations.

PHOENIX-P uses response matrix method and Sn method to calculate flux distribution in a two-dimensional heterogeneous assembly model. The accuracy of the code for MOX fuel was verified. Introduction of the CCCP to PHOENIX-P is planned for future use[11,12].

CASMO can perform heterogeneous two-dimensional transport calculation using the method of characteristics. It is also capable of performing full-core heterogeneous transport and depletion calculation and treating the actual full scope fuel shuffling. In the improved NULIF, the burn-up chain are extended and the many groups are utilized to calculate the neutron spectrum.

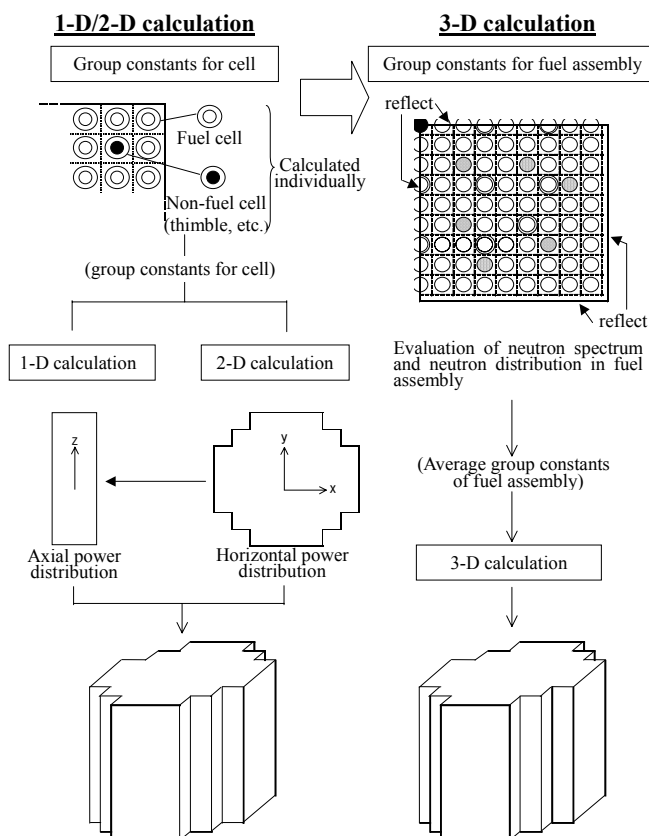


Fig.2 Progress of nuclear calculation method for PWR (typical example)

In PWR core simulators, the 3D modeling has been extended. ANC[9,13], SIMULATE-3, SHARP codes have been developed. In SIMULATE-3 the semi-analytic nodal model, the two group form function, and the spectral interaction model were introduced for the treatment of MOX loaded cores. In ANC, the spectrum interference effect is treated by the multi-assembly model. The micro depletion method was also

developed for 9 actinide nuclides so that the reactivity effect of local spectrum and power history for MOX and high burnup cores can be considered instead of the multi-assembly model. The micro depletion model has been expanded to 12 nuclides[14]. Furthermore, a pin-power recovery method was also developed. In SHARP, the spectrum interference effect is treated by the multi-assembly model. The pin powers are calculated by combination of 3D- coarse mesh calculations and 2D- fine mesh calculations.

The validity of these core calculation codes was checked by analysis of critical experiments and data of power reactors loaded with UO₂/MOX fuel.

Unlikely BWR code shares, it is not proper to show the code shares of PWR because two or three codes are applied to a same plant. The PWR codes described above are to be used as following number of plants.

PHOENIX-P/ANC 23/23*

CASMO/SIMULATE 12/23

Improved NULIF/SHARP 22/23*

(*: based on the current share of 1D/2D codes)

3. Neutronics Codes for FR analysis

Finally let us review the code system for fast reactors. The code systems have been developed for analyses of fast critical assemblies and real power reactors. The JUPITER was the PNC-DOE cooperative experimental program using the ZPPR facility to obtain the neutronic performance of large fast reactors. For such critical assemblies the effective cross sections are calculated by SLAROM or CASUP in 70 groups. The 70-group cross sections are collapsed to 10~20 groups, and the collapsed cross sections are used for the whole core diffusion and transport calculations by CITATION and TRITAC[15]. TRITAC solves the 3-D transport equation using the Sn method. The diffusion synthetic acceleration technique is adopted. TRITAC was effectively applied to a series of critical assemblies of the JUPITER program. The transport corrections to Keff, reaction rate ratio, reaction rate distributions were estimated. The transport correction of the heterogeneous fast critical assembly ZPPR-13 was remarkably large.

For the core calculations of power reactors with 3-D hexagonal-z geometry, CITATION and NSHEX[16] codes are utilized for diffusion and transport calculations, respectively. The NSHEX solves the 3-D transport equation based on the nodal Sn method.

Recently it was found that the previous 70-group cross sections were not enough to produce accurate calculations of core performance parameters. JNC proposed the fine-group cross section set. Furthermore, to treat the heterogeneity of fuel assemblies, particularly special irradiation assemblies, BACH[17] has been developed based on the method of characteristics. BACH can predict pin by pin flux distribution. So it is not necessary to homogenize cross sections over an assembly. From numerical calculations it was found that BACH results agree well the Monte-Carlo results, and can accurately calculate reaction rate even when the Monte-Carlo method produces large standard deviation.

For fast reactor application, the sensitivity analysis code (SAGEP) and cross section adjustment code (ABLE) have been developed. SAGEP calculates the sensitivity coefficient of neutronic performance parameters with respect to micronic cross section

change based on the generalized perturbation theory. The coefficient is obtained based on diffusion theory in 2-D and 3-D geometry.

ABLE predicts the uncertainty of neutronics performance parameters using the sensitivity coefficient obtained by SAGEP. Furthermore, using measured data ABLE can make the cross section adjustment based on Base' theory. The method uncertainty can be incorporated in the adjustment. Using the experimental data obtained from the ZPPR, FCA and other experimental facilities, an adjusted cross section set was produced.

The cross section has been utilized for the calculation of core parameters of MONJU and a large fast reactor.

4. General Analysis Codes

In addition to the above LWR and FR codes developed for design, applications the Monte-Carlo(MC)codes were developed. VMONT[18] is a multi-group MC code for detailed burn up calculations based on the fanton-scattering theory. MVP[19,20] was extended to consider the double heterogeneity and burnup.

A continuous-energy Monte Carlo code MVP for neutron and photon transport calculation, together with its multi-group version GMVP has been developed since 80s at JAERI. The first version of the code was released for domestic use in 1994. These two codes were designed for vector supercomputers at first stage. After that, functions have been extended for 1)parallelization with standard libraries (PVM, MPI), 2)continuous-energy calculation at arbitrary temperature points, 3)geometrical description capability for randomly distributed fuel particles with statistical geometry model, 4)capability of burn-up calculation, 5)perturbation calculation for eigenvalue problems, 6)function of reactor noise analysis used for simulation of the Feynman- α experiment, and so on. The revised MVP/GMVP codes are widely used in Japan, especially in the field of reactor physics analyses.

The SRAC system[21,22] is designed to permit neutronics calculation for various types of thermal reactors. The system covers production of effective microscopic and macroscopic group cross-sections, and static cell and core calculations including burn-up analyses. The effective cross-sections by the conventional table look-up method based on the NR approximation can be replaced by those obtained with more rigorous method PEACO which solves a multi-region cell problem by the collision probability method using a hyper-fine energy group structure for the resonance energy range. The system integrates five elementary codes including imported ones for neutron transport and diffusion calculation, they are, PIJ based on the collision probability method applicable to 16 types of lattice geometries, S_N transport codes ANISN and TWONTRAN, diffusion codes TUD and CITATION. In 1996 the revised SRAC system which works on many machines with UNIX or similar ones was released, and it has been widely used in Japan mainly for experimental analyses of critical assemblies and conceptual design study of future reactors.

5. Future Plan and Conclusions

In future LWR calculations multi-group transport theory method with explicit heterogeneous geometry will be utilized in assembly calculations. Detailed treatment of

neutron anisotropic scattering will be also treated[23]. In core calculations, the improvement of spectrum mismatch effect, nodal or pin-by-pin transport method will be utilized. For example, three dimensional, multi-group, cell averaged pin-by-pin transport calculation code, named SCOPE2[24], has been developed and is ready to apply for real commercial cores. Furthermore, the boundary between assembly and core calculations will disappear. For example, the capability of CASMO-4 heterogeneous full-core transport calculation can be applied to actual core design calculations by means of the Hybrid Core Calculation System[25]. As reported in Ref.[12], PARAGON code has the capability to perform two-dimensional core calculation, too.

PARAGON also has the capability to generate power distribution within a fuel rod in an assembly. Such micro-nuclear-physics will be also applied to design code system[26] in near future.

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Table 1. Codes Used for LWR and FBR Applications in Japan

Core Type	Assembly Analysis Code			Core Analysis Code		
	Code Name	Developer	Users	Code Name	Developer	Users
BWR	TGBLA	GNF-J	GNF-J	LOGOS	GNF-J	GNF-J
			GIS	PANACEA	GNF-J	GIS
	LANCER	GNF-J	GNF-J	AETNA	GNF-J	GNF-J
	NEUPHYS	NFI	CTI	COS3D	NFI	CTI
	CASMO	SSP	TEPSYS	SIMULATE	SSP	TEPSYS
PWR	PHOENIX-P	MHI	MHI	ANC	MHI	MHI
	CASMO	SSP	NEL	SIMULATE	SSP	NEL
	Improved NULIF	NFI	NFI	SHARP	NFI	NFI
FBR	SLAROM	JAERI	JAERI	CITATION	ORNL	JNC
			JNC	TRITAC	Osaka Univ.	JNC
	CASUP	Osaka Univ.	JNC Osaka Univ.	NSHEX	Osaka Univ.	JNC Osaka Univ.
General	VMONT-Monte Carlo Code (HITACHI)					
	MVP-Monte Carlo Code (JAERI)					
	SRAC (JAERI)					