Nuclear data library in design calculation

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Abstract

In core design calculation, nuclear data takes part as multi group cross section library during the assembly calculation, which is the first stage of a core design calculation. This report summarizes the multi group cross section libraries used in assembly calculations and also presents the methods adopted for resonance and assembly calculation.

1. Nuclear data library in light-water reactor core design calculation

The current core design calculation does not treat whole core heterogeneous geometry of a reactor directly (for instance, the actual shape of a fuel assembly). Instead, the calculation is first done at an assembly level, and then to a reactor level. The assembly calculation is executed using multi group cross section data prepared beforehand from the nuclear data library, which then provides a few group cross section library to be used for the reactor calculation. The reactor calculation is then performed using this few group library to evaluate core characteristics such as eigenvalue or flux distribution.

Since a few group cross section library used in the reactor calculation consists of collapsed energy groups and homogenized spatial geometry, the characteristics of nuclear data library in reactor calculation is less obvious compared to that of assembly calculations which utilizes finer multi group cross section and more detailed geometry information. Therefore, this paper will focus on the assembly calculation and will present base library and neutronic calculation methods adapted in its calculation.

2. Present Design Assembly Code

The assembly codes were first introduced into Japan by nuclear fuel vendors such as WH (Westinghouse) or GE (General Electric) at the same time when light-water reactors were introduced. These assembly codes are based on ENDF/B-4/5 library, which have been constantly improved and updated. They are widely used now for a commercial core design.

Some of the features of assembly calculation codes for a core design calculation are noted as follows.

2.1 PHENIX-P(PWR) [1],[2]

PHENIX-P has been used for PWR fuel assemblies at MHI (Mitsubishi Heavy Industries). It was developed with WH and was released for design use in 1987. It employs a 42-group cross-section library based on ENDF/B-V. The resonance calculation method is based on IR approximation. The assembly calculation employs S_4 using energy-condensed 6-group cross-section data as a transport solver. The 6-group cross-section data is collapsed in energy and space by using 42 energy groups flux distribution which is obtained by Node Joined Method. The heterogeneous neutron flux distribution of the fuel cells in the assembly is evaluated by combining the heterogeneous flux of the fuel cells obtained by the Node Joined Method with the homogeneous flux of the fuel cells obtained by the assembly calculation.

2.2 Improved NULIF(PWR) [3],[4]

Improved NULIF(PWR) has been used for PWR fuel assemblies at NFI (Nuclear Fuel Industries). It was developed by NFI and was released for design use in 1988. It employs a 99-group cross-section library based on ENDF/B-V. The resonance calculation method is based on NR approximation and NRIA (Narrow Resonance Infinite Absorber). The assembly calculation is based on diffusion theory, and uses energy-condensed 3-group cross-section data. One of the remarks for Improved NULIF is very short run time, which makes the generation of few-group cross-section library easy.

2.3 CASMO(PWR/BWR) [5]

CASMO was developed by Studsvik of America (now Studsvik Scandpower), and has been used as a nuclear design code for PWR fuel assemblies at NEL (Nuclear Engineering Ltd.) and for BWR fuel assemblies at TEPSYS (Tepco Systems). It usually employs a 70-group cross-section library based on ENDF/B-IV and V. The resonance calculation method is based on IR approximation. Method of characteristic (MOC) is applied with energy-condensed 7-group cross-section data as a neutron transport solver. The explicit geometry of an assembly can be treated.

2.4 TGBLA(BWR) [6],[7],[8]

TGBLA has been used for BWR fuel assemblies at GNF-J (Global Nuclear Fuel Japan). It was developed by GE and Toshiba and was released for design use in 1982. It employs a 98-group cross-section library based on ENDF/B-4 and 5. The resonance calculation method is based on IR approximation and IRCM [9] code. The assembly calculation is based on diffusion theory, and uses energy-condensed 3-group cross -section data.

2.5 HINES(BWR)

HINES has been used for BWR fuel assemblies at GNF-J. It was developed by Hitachi and was released for design use in 1982. It employs a 98-group cross-section library based on ENDF/B-IV and V. The resonance calculation method is based on IR approximation. The assembly calculation is based on diffusion theory, and uses energy -condensed 3-group cross-section data.

2.6 NEUPHYS(BWR)[10]

NEUPHYS has been used for BWR fuel assemblies at NFI. It was developed by NFI and was released for design use in 1985. It employs a 98-group cross-section library based on ENDF/B-IV. The resonance calculation is based on IR approximation. The assembly calculation is based on diffusion theory, and uses energy-condensed 3-group cross-section data.

３）Next Generation Design Assembly Code

Decades have passed since the introduction of assembly codes, and domestic vendors are now developing next generation assembly codes. The next generation assembly codes feature improved accuracy with more explicit treatment of geometry and more energy groups. Some domestic venders have started employing JENDL series (3.2/3.3) as one of the main nuclear data besides ENDF series.

3-1) PARAGON(PWR)[11]

PARAGON has been in development at WH and MHI for PWR fuel assemblies. For Core design calculation, it usually employs a 70-group cross-section library based on ENDF/B-VI. For research purposes, some multi-group cross-section libraries such as a 187-group based on JENDL-3.3 can be applied optionally. The resonance calculation is based on IR approximation with spatially dependent Dancoff method [12]. Current-coupling collision probability methods (CCCP methods)[13] with 70-group cross-section data is utilized in PARAGON as a neutron transport solver.

3-2) AEGIS(PWR)[14]

AEGIS has been in development at NEL for PWR fuel assemblies. It employs three 172-group cross-section libraries made from ENDF/B-VI, ENDF/B-VII and JENDL 3.3. The resonance calculation is done in super fine groups. MOC is currently applied with 172-group cross-section data as a neutron transport solver. It can treat the large-scale calculation with explicit geometry, higher order anisotopic scattering, and resonance shielding effect.

3-3) LANCER(BWR)[15]

LANCER has already been developed for BWR fuel assemblies at GNF-J. It was developed by GNF-J and was released for design use in 2005. It employs a 190-group cross-section library based on ENDF/B-VI. The resonance calculation method is based on F-table. CCCP methods used energy-condensed 35-group cross-section data is utilized in LANCER as a neutron transport solver in the assembly calculation.

3-4) Improved NEUPHYS(BWR)[16]

Improved NEUPHYS has been in development at NFI for BWR fuel assemblies. It employs a 98-group cross-section library based on JENDL 3.2. The resonance calculation is based on

IR approximation. MOC with energy-condensed 20-group cross- section data is utilized in Improved NEUPHYS as a neutron transport solver. It can treat the large-scale calculation with explicit geometry.

４）Summary

The features of assembly codes are summarized in Table 1. The present assembly codes mainly employ the library based on ENDF/B-IV or V. On the other hand, some assembly codes have started employing JENDL series (3.2/3.3) as one of the main nuclide data besides ENDF series. The assembly codes are providing more accurate results, which makes the role of the nuclear library in the calculation more prominent.

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Table 1 Summary of the Features of Assembly Codes

		Code Name	Base Library	Year	Developer	Group Structure	Resonance Calculation	Assembly Calculation	Remarks
Present Assemby Codes	PWR	PHOENIX-P	ENDF/B-5	1987	WH/MHI	42	IR	6G Transport (S4)	Hetero-42G Flux
		Improved NULIF	ENDF/B-5	1988	NFI	99	NR/NRIA	5G Diffusion	Short Run Time
		CASMO	ENDF/B-4 and 5	—	SSP	70	IR	7G Transport (MOC)	Explicit Geometry
	BWR	TGBLA	ENDF/B-4 and 5	1982	GE/Toshiba	98	IR/RICM	3G Diffusion	Short Run Time
		HINES	ENDF/B-4 and 5	1982	Hitachi	98	IR	3G Diffusion	Trabsport Correction
		LANCER	ENDF/B-6	2005	GNF-J	190	F-table	35G Transport (CCCP)	Explicit Geometry
		NEUPHYS	ENDF/B-4	1985	NFI	98	IR	3G Diffusion	Short Run Time
		CASMO	ENDF/B-4 and 5	—	SSP	70	IR	7G Transport (MOC)	Explicit Geometry
Next Generation Assembly Code	PWR	PARAGON	ENDF/B-6	—	WH/MHI	70	IR+SDDM	70G Transport (CCCP)	2D-Full Core, Inner Rod γ-Transport
		AEGIS	ENDF/B-6,7 and JENDL-3.3	—	NEL	172	Super Fine Group	172G Transport (MOC)	Large-Scale Calculation with Explicit Geometry
	BWR	LANCER	ENDF/B-6	2005	GNF-J	190	F-table	35G Transport (CCCP)	38 Fuel Nuclides and 136 Fission Product
		Improved NEUPHYS	JENDL-3.2	—	NFI	96	IR	20G Transport (MOC)	Large-Scale Calculation with Explicit Geometry