Neutron Fluence Database of Experimental Fast Reactor JOYO for Fuel and Structural Material Irradiation Tests

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ABSTRACT

The experimental fast reactor JOYO operated as an irradiation test facility for 18 years with its MK-II core. During this time, an extensive study was conducted to characterize the neutron field in order to assure the accuracy and reliability of neutron dose. Neutron flux for individual irradiation tests was calculated using core management code system based on three-dimensional diffusion theory and these results were corrected with the adjusted neutron spectrum by means of the multiple foil activation method. The neutron fluence calculation accuracy in the core region was within 5%. At positions away from the core region or within irradiation test subassemblies with heterogeneous internal structure, the neutron flux distribution was precisely analyzed with two-dimensional transport code or Monte Carlo code. Neutron flux and fluence from the core management calculations are compiled as a database with related irradiation information and fuel subassembly material compositions. This data is expected to be used for the development of fast reactor fuels, structural materials and future core designs.

1. INTRODUCTION

The experimental fast reactor JOYO is located at the Oarai Engineering Center of the Japan Nuclear Cycle Development Institute. At JOYO, various irradiation tests have been carried out to develop the fuels and materials for future FBRs and to support surveillance tests. Neutron fluence including spectral information is a key parameter in the post irradiation test analysis, therefore their specified accuracies are set for irradiation tests as shown in Table 1.

For high performance fuel irradiation tests, linear heat rate and burn-up of fuel pin need to be evaluated within 5%. Structural material irradiation tests, designed to develop low swelling and high creep strength materials, evaluate the displacement per atom (DPA) and helium production. The relevant neutron fluence accuracies of fast energy (E>0.1MeV) and lower energy (E>0.4eV) neutrons are required to be within 10% and 20% respectively. Surveillance tests check reactor component lifetimes and evaluate the same variables as the structural tests. The relevant neutron fluence accuracies of fast energy (E>0.1MeV) and lower energy (E>0.4eV) neutrons are required to be within 10% and 50% respectively.

Therefore, a number of reactor dosimetry measurements, based on the multiple foil activation method, were conducted in addition to the nuclear calculations to meet these specifications. This paper describes the method and accuracy of the JOYO MK-II irradiation test neutron fluence evaluations. The material composition calculations include atomic number density changes due to fuel burn-up. The core management calculations of neutron flux and material compositions described in this paper are stored as a reference database for future development.

2. JOYO PLANT DESCRIPTION AND OPERATING HISTORY

JOYO is a sodium cooled fast reactor with mixed oxide (MOX) fuel. The main reactor parameters of the MK-II irradiation bed core are shown in Table 2, which compares the MK-II with the future MK-III core. The MK-II driver fuel plutonium content is about 30wt%. Initially ^{235}U enrichment was about 12wt%, however this was increased up to 18wt% in 1987 to provide enough excess reactivity so that the core burn-up was increased. Consequently, the operational period was extended from 45 days to 70 days and the plant availability increased. Since 1998, some of the MK-III driver fuel subassemblies, which have the same specification as the MK-II except a shorter fuel stack length, were loaded in the outer region of the core. Figure 1 shows an example of the MK-II core configuration.

JOYO has two primary sodium loops, two secondary loops and an auxiliary cooling system. Approximately 200 tons of sodium is used for the cooling system. In the MK-II core, sodium enters the core at 370 °C at a flow rate of 1,100 tons/h/loop and exits the reactor vessel at 500 °C. The maximum outlet temperature of a fuel subassembly is about 570 °C. An intermediate heat exchanger (IHX) separates radioactive sodium in the primary system from non-radioactive sodium in the secondary system. The secondary sodium loops transport the reactor heat from the IHX to the air-cooled dump heat exchanger (DHX). A flow diagram of the cooling system of MK-II core is shown in Fig. 2.

Figure 3 shows the JOYO MK-II operating history. Since 1983, the MK-II operations have included thirty-five duty cycle operations and thirteen special test operations. The reactor operated for 48,000 hours and the integrated power was 4,400GWh. During this time, 382 driver fuel subassemblies and approximately 47,000 fuel pins were irradiated. A peak burn-up value of the MK-II driver fuel reached 86.0 GWd/t without any fuel pin failures.

3. CHARACTERIZATION OF NEUTRON FIELD IN JOYO

3.1 Evaluation System Outline

Figure 4 illustrates the system outline for characterizing JOYO's neutron field. The MAGI [1] core management code system calculates the neutron flux distribution. The precise calculations are conducted using the DORT [2] two-dimensional transport code or the MCNP[3] Monte Carlo code. The MAGI neutron source distribution calculation is used in these two calculations. To confirm the calculation accuracy, a series of foil activation method dosimetry tests have been conducted on the reactor. The neutron flux was adjusted with the measured reaction rates using the NEUPAC[4] J1-Unfolding code and the results were used to correct the MAGI calculation values. These neutron spectrum analysis methods are described below.

3.2 Core Management Calculation

The MAGI core management code system was developed for core and fuel management as well as operational planning purposes. The MAGI calculation flow is illustrated in Fig. 5. MAGI is a neutronic and thermo-hydraulic coupling code system to calculate core characteristics. The nuclear calculation is based on diffusion theory with seven energy groups. The calculating model uses three-dimensional-hexagonal-Z geometry. In the horizontal cross section of the calculation model, there are 331 meshes with one mesh per assembly. The assembly pitch is 8.15cm. In the vertical direction, the calculation region is 140cm high and divided into 20 meshes. The 55 cm high fuel region is divided into 11 meshes. The calculation accuracy for the core characteristics like criticality is within 0.1%Δk/kk'. MAGI also calculates the neutron flux and fluence across the core. The core region neutron fluence calculation accuracy is within 5%.

The MAGI burn-up calculation treats every refueling and operating record of each cycle and calculates the atomic number densities for all fuel assemblies. These atomic number densities are inputs for the detailed DORT and MCNP calculations and were verified by Post Irradiation Examination (PIE). Figure 6 shows the measured burn-up ratios in comparison with the calculated values. The close correspondence between calculated and measured values indicates an accurate MAGI burn-up calculation.

The thermal hydraulic calculation provides confirmation that the temperatures of the fuel, cladding and coolant are below the thermal design limit. For this calculation, MAGI calculates the temperatures of the cladding and fuel of all the fuel assemblies, coolant flow rate distribution, and maximum coolant temperature. This temperature data is also used as an irradiation condition for the post irradiation test analysis.

3.3 Reactor Dosimetry and Neutron Spectrum Adjustment

The MAGI calculations are corrected by adjusting the neutron flux using reaction rates measured with a set of dosimeters (Fe, Sc, Co, Cu, Ti, Ni, Nb, ^{235}U, ^{238}U, ^{237}Np and others). These dosimeters cover the relevant JOYO MK-II neutron spectrum energy range from approximately 0.1eV to 20MeV. The dosimeter materials used are listed in Table 3. A Helium Accumulation Fluence Monitor (HAFM) has been developed and used for fast reactor dosimetry in addition to the activation and fission foils. Figure 7 shows a typical neutron spectrum at the core center position and a 90% confidence level for each reaction rate. Each set of dosimeters is encapsulated into dosimeter capsules that are placed at several locations in the irradiation test subassembly.

The reaction rate was determined by analyzing the gamma ray or X-ray spectra as measured from the irradiated dosimeters. The measurements were performed with a high purity Germanium (Ge) gamma-ray detector and a Low Energy Photon Spectrometer (LEPS). Both detectors were calibrated with the standard gamma-ray sources whose energies spanned those of the activated nuclides. The accuracy of gamma-ray measurement system had been confirmed within 3% through the integral tests in a fast reactor neutron field in the YAYOI fast neutron source reactor at the University of Tokyo and the reactor dosimetry intercomparison study between JOYO and EBR-II[5].

The neutron spectrum at each dosimeter position is adjusted by the measured reaction rates using NEUPAC, the J1 type spectrum unfolding code package, as shown in Fig. 8. NEUPAC uses the 103-group cross-section: the error covariances are processed from either the ENDF/B-V cross-section library or the JENDL-3 dosimetry file. Using NEUPAC code, the neutron dose and relevant irradiation parameters such as dpa (NRT) and helium production were evaluated from the adjusted whole neutron spectrum.

An example of the neutron spectrum adjustment is illustrated in Fig. 9. The test fuel subassembly (C3M) was initially irradiated in the third row (core address [3E1]) for 639 EFPDs (Effective Full Power Days). It was then moved to the second row (core address [2D2]) and continued irradiation test for another 197 EFPDs. The maximum displacement per atom reached approximately 70 dpa (NRT) at the end of irradiation. The core configuration in Fig. 9 shows the C3M subassembly adjacent to the control rod. The B_4C control rod is enriched with 90% ^{10}B.

Table 4 compares the fast neutron fluence, E>0.1 MeV, at three dosimeter positions between the MAGI calculation and the NEUPAC adjustment. The C/E (Ratio of Calculation to Experiment) value was approximately 1.2 at the core mid plane, and a slightly larger discrepancy was observed at the upper and lower elevations. The neutron flux distribution around the C3M subassembly including the control rod shows there was a large gradient due to the neutron capture reaction by the control rod. Usually the MAGI calculations were within 5% of the NEUPAC adjustments in the fuel region. However, near the control rod, MAGI overestimated the local neutron flux distribution. In this case, it is essential to correct the calculated neutron fluence with dosimetry measurements.

3.4 Transport Calculation

At positions away from the core center MAGI appears to have large uncertainty in the neutron flux due to its high gradient and significant spectral change. In these cases, the neutron flux distribution was calculated using DORT in two-dimensional RZ or XY-Rθ geometry. The transport calculation flow is shown in Fig. 10. The RADHEAT-V3[6] system computes the 100 energy group macroscopic cross sections using JSDJ2/JFTJ2[7] set.

The transport equation is solved as a fixed source problem where the neutron source distribution is obtained from MAGI. The two dimensional RZ calculation model is shown in Fig. 11. The calculation region includes the radial carbon shield and pedestal concrete: the diameter and the height are 420cm and 570cm respectively. In the two-dimensional XY-Rθ calculation, neutron leakage in the axial direction was considered as a pseudo-absorption to calculate the neutron flux distribution in the core center plane. Figure 12 shows the radial neutron flux distribution.

Fission products (FPs) were not considered in conventional fast reactor shielding analyses like DORT that were developed predominantly in clean core experiments like the JASPER program [8, 9]. In a recent study [10], the FP cross-sections for a fast reactor shielding calculation were computed and compiled with the shielding constants. As a result, the effect of FPs on the shielding analysis of JOYO was about 2%. With these, the results by conventional fast reactor shielding analysis were to be corrected by FPs' effects about 2%.

The DORT calculation for the core center spectrum, where the spectrum is the hardest, is shown in Fig. 7. The typical neutron spectra of the other irradiation fields, the reflector, the upper core structure and the ex-vessel are shown in Fig. 13. These spectra are softer than the core center. These 100 neutron energy spectra calculated by DORT are used as an initial guess of spectrum unfolding by the NEUPAC code.

3.5 Monte Carlo Calculation

Deterministic calculations such as MAGI and DORT do have limitations. As the compositions of fuels and structures are homogenized in the calculation mesh the neutron flux distribution and the fine spectrum change inside the complex and heterogeneous structural subassembly cannot be calculated. Since the as the complex and heterogeneous structure can be modeled in MCNP calculation, the continuous energy MCNP Monte Carlo code can more accurately evaluate the neutron flux distribution inside the subassembly.

The Monte Carlo calculation does not provide the burn-up fuel composition so the core neutron source distribution is obtained from MAGI. Figure 14 shows an example of a heterogeneous structured irradiation subassembly. The B5D-2 irradiation test subassembly was used for a fuel performance test that was conducted to optimize the thermal analysis method in the MOX fuel pin design. The fuel pellets of B5D-2 were irradiated at high linear heat rate so that the center of each pellet could be melted.

The MCNP calculation accuracy was verified by comparing calculated fission rates with measured values from the PIE[11]. The calculation accuracy was confirmed using fission rate based on the ^{148}Nd production measurement method. ^{148}Nd is one of the stable fission products and its fission yield is highly reliable, therefore the ^{148}Nd production obtained by the destructive examination has been commonly used as a burn-up index. In the PIE, ^{148}Nd production was measured by means of isotopic dilution mass spectrometry[12] that has an accuracy of approximately 1%. The comparison of measured fission rates and MCNP calculations is shown in Figure 15. The average C/E value was 0.955 ± 0.020. This result indicated that MCNP calculates the fission rates precisely.

4. NEUTRON FLUENCE DATABASE

The MAGI calculation results like the burn-up fuel compositions and fission neutron sources were used for the detail analyses. Therefore, they were compiled as a database in a consistent format and recorded on CD-ROM for user convenience. The CD-ROM contains three types of data, Configuration Data, Assembly Library Data and Output Data. The Configuration Data includes the fuel exchange and core arrangement history for each reactor duty cycle. The Assembly Library Data includes the atomic number density, neutron fluence, burn-up and integral power for about 300 fuel assemblies, and 60 irradiation subassemblies. The Output Data includes the neutron fluxes with seven group spectra, gamma fluxes, power density, linear heat rates, coolant and fuel temperature of all the assemblies at the beginning and end of each cycle. The neutron fluence and spectrum data are also served for each irradiation tests as a database together with the detailed methods.

Examples of the burn-up and the neutron fluence data are shown in Fig. 16 and neutron flux distribution and the spectrum data are shown in Fig. 17. Users can browse and edit this data with their personal computers and analyze the irradiation conditions. By comparing the evaluated values with the measured PIE data, the users can use these estimations for the development of the fuels and structural materials.

5. CONCLUSION

Neutron fluence and spectra of the JOYO MK-II core are evaluated by core management calculations, transport calculations, Monte Carlo calculations and spectrum unfolding calculation. These results, along with the measured reaction rates are available for the post analysis of the irradiation tests. The evaluation accuracy of the core neutron fluence and the flux distributions inside the irradiation subassemblies were less than 5% in the core region. The irradiation data and evaluation methods have been compiled in a database and can be applied in various FBR fields like fuel and structural material development and core design.

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Table 1 Irradiation Tests in JOYO and Specified Accuracy for Neutron Fluence

Test	Purpose	Characteristics	Fluence Accuracy
Fuel Irradiation	High Power and Burn-up Fuels Development	Linear Heat Rate Burn-up	5% for Total
Cladding and Component Material Irradiation	Low Swelling and High Creep Strength Materials Development	DPA He Production	5~10% for E>0.1MeV and 20% for E>0.4eV
Surveillance	Life Evaluation of Reactor Component	DPA He Production	10% for E>0.1MeV and 50% for E>0.4eV

Table 2 Main Core Parameters of JOYO

Items			MK-II	MK-III
Reactor Thermal Output		(MWt)	100	140
Max. No. of Test Irradiation S/A			9	21
Core Diameter		(cm)	73	80
Core Height		(cm)	55	50
^{235}U Enrichment		(wt%)	12(J1)/18(J2)	18
Pu Content		(wt%)	≦30	≦30
Pu fissile Content (Inner/Outer Core)		(wt%)	~20	~16/21
Neutron Flux	Total	(n/cm^2/s)	4.9×10^{15}(J2)	5.7×10^{15}
	Fast(>0.1MeV)	(n/cm^2/s)	3.2×10^{15}(J2)	4.0×10^{15}
Primary Coolant Temp. (Inlet/Outlet		(℃)	370/500	350/500
Operation Period		(days/cycle)	45(J1)/70(J2)	60
Reflector/Shielding			SUS/SUS	SUS/B_4C
Max. Excess Reactivity (at 100℃)		%Δk/kk'	5.5	4.5
Control Rod Worth		%Δk/kk'	≧9	≧7.6

Table 3 Standard Dosimeter Set of Joyo

Monitor Material	Form	Reaction: Non-Threshold	Reaction: Threshold
Co	Wire (Co-V or Co-Al)	^{59}Co(n, γ)	
Sc	Vanadium Capsuled (Sc_2O_3)	^{46}Sc(n, γ)	
Ti	Wire		^{46}Ti(n,p)
Fe	Wire	^{58}Fe(n, γ)	^{54}Fe(n,p)
Ni	Wire		^{58}Ni(n,p)
Cu	Wire		^{63}Cu(n, α)
Ta	Wire (Ta-V or Ta-Al)	^{181}Ta(n, γ)	
Nb	Thin Foil		^{93}Nb(n,n')
^{237}Np	Vanadium Capsuled (NpO_2)		^{237}U(n,f)
^{235}U	Vanadium Capsuled (UO_2)	^{235}U(n,f)	
^{238}U	Vanadium Capsuled (UO_2)		^{238}U(n,f)
^{232}Th	Vanadium Capsuled (Th)	^{232}Th(n, γ)	^{232}Th(n, γ)

Table 4 Result of Fast Neutron Fluence and DPA Analysis

Dosimeter Position	Item	Fast Neutron Fluence and DPA		MAGI / NEUPAC
		NEUPAC	MAGI	
Z = +27.5cm	$\Phi_{E>0.1MeV}$	$(6.03\pm0.57)\times10^{22}$	7.93×10^{22}	1.32
	DPA	30.7 ± 2.0	------	------
Z = $\pm$0cm	$\Phi_{E>0.1MeV}$	(13.8 ± 1.2)	16.5×10^{22}	1.20
	DPA	69.6 ± 4.4	------	------
Z = -27.5cm	$\Phi_{E>0.1MeV}$	(7.28 ± 0.73)	9.13×10^{22}	1.25
	DPA	36.2 ± 2.5	------	------

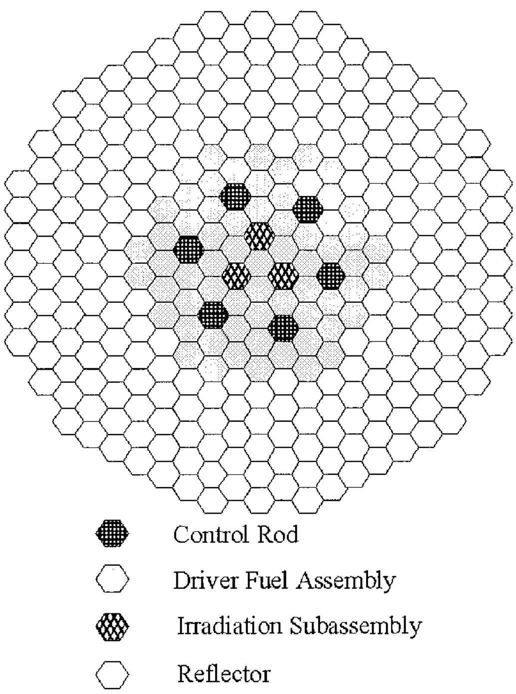


Fig.1 JOYO MK-II Core Arrangement

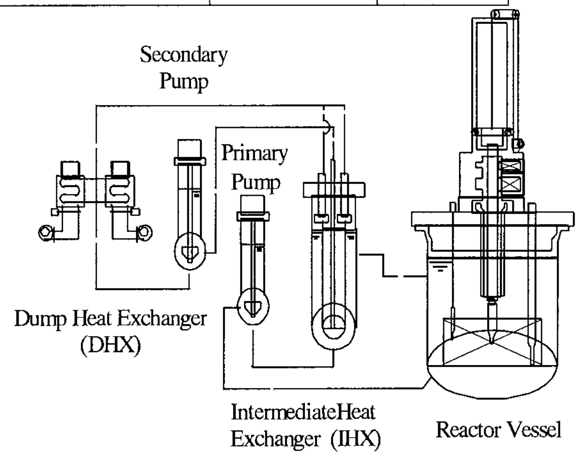


Fig.2 JOYO Cooling System Diagram

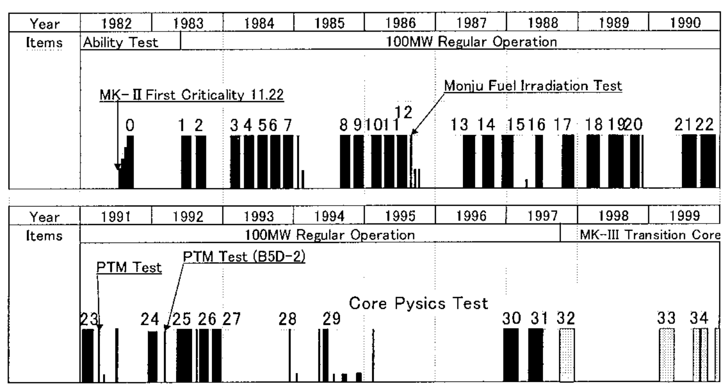


Fig.3 JOYO MK-II Operational History

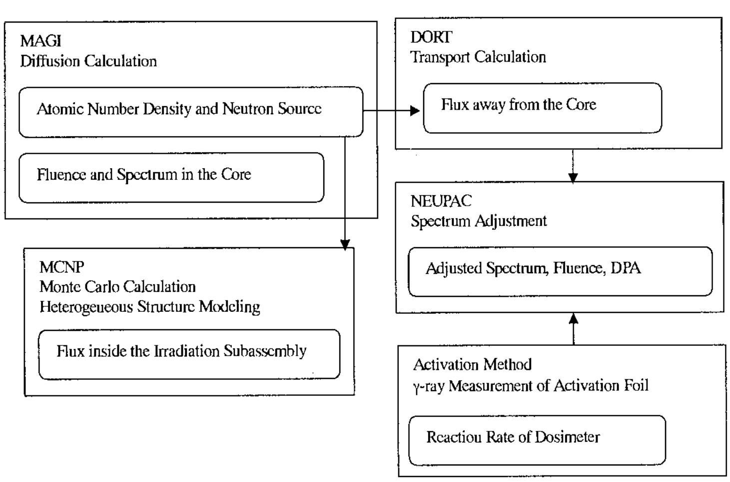


Fig.4 JOYO Dosimetry System

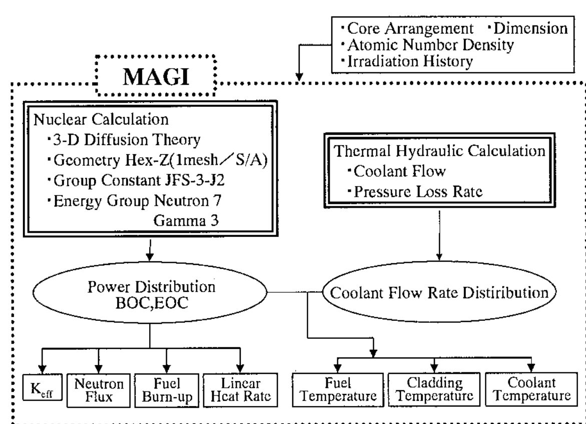


Fig. 5 JOYO MK-II Core Management Code System MAGI

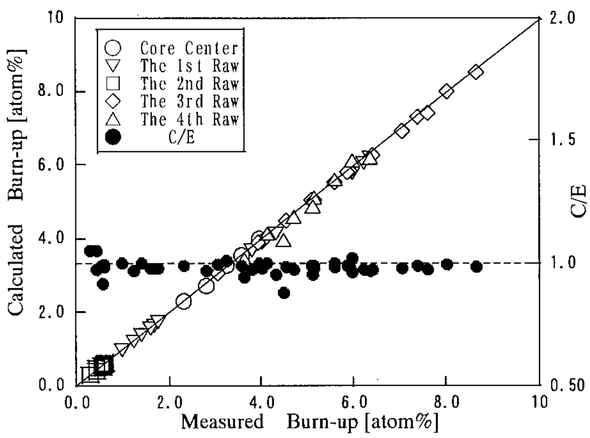


Fig.6 C/E of Burn-up Ratio

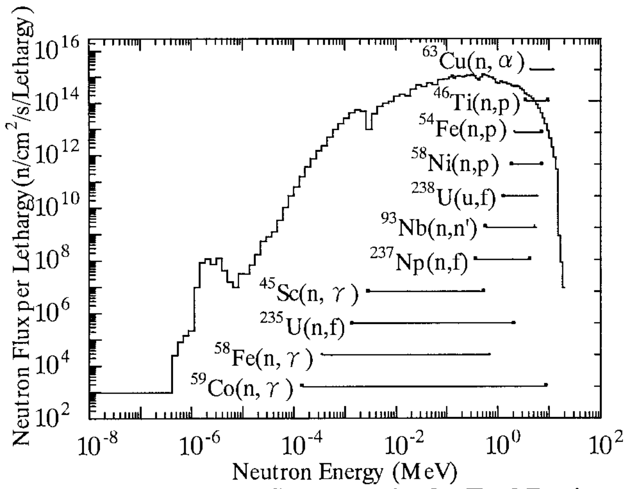


Fig. 7 Typical Neutron Spectrum in the Fuel Region and Standard Dosimeter Set of JOYO MK-II

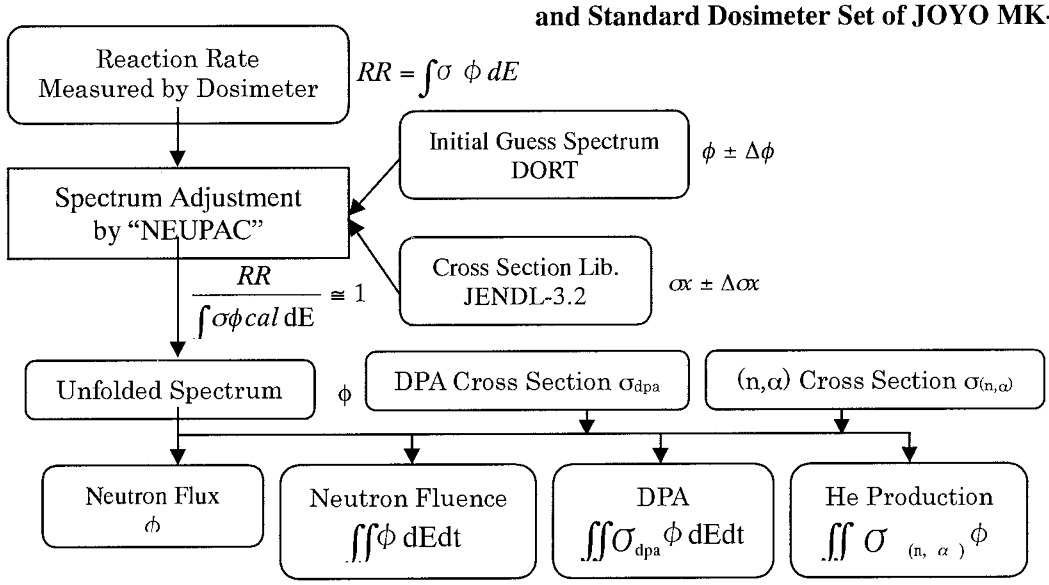
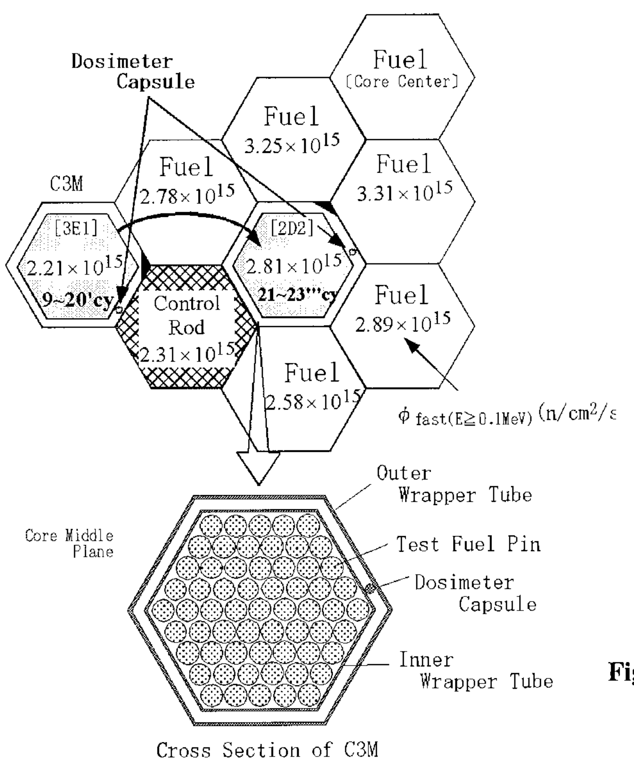


Fig.8 Calculation Flow of Neutron Adjustment by NEUPAC



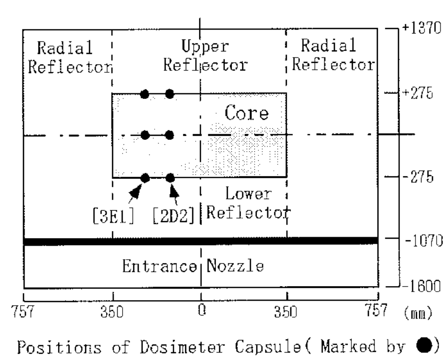


Fig.9 Core Configuration and Dosimeter Positions in C3M Irradiation Fuel Subassembly

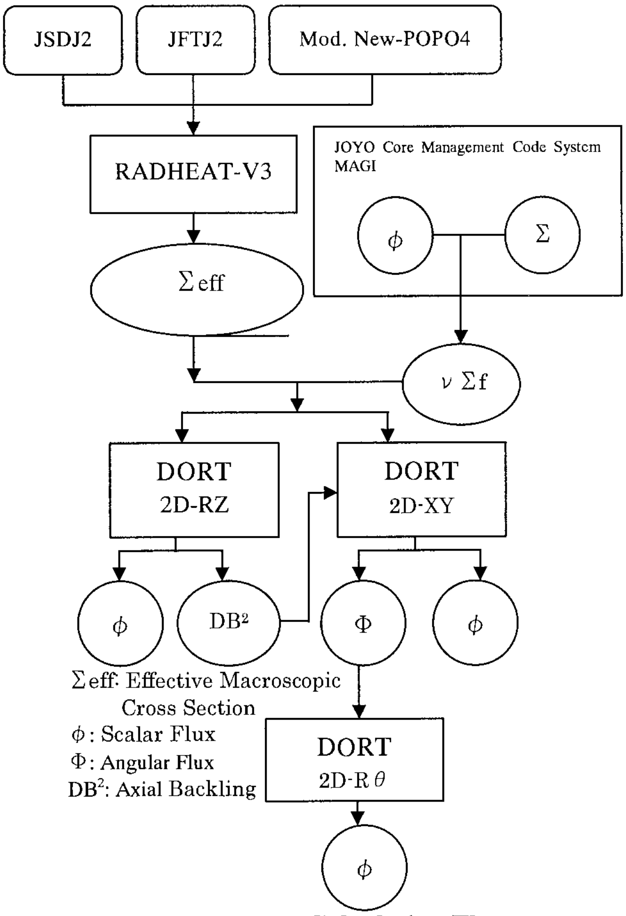


Fig.10 Transport Calculation Flow

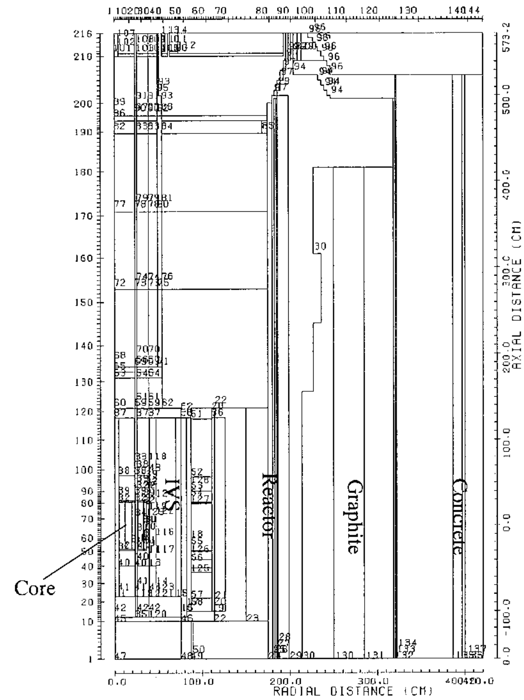


Fig.11 2D-RZ Transport Calculation Model

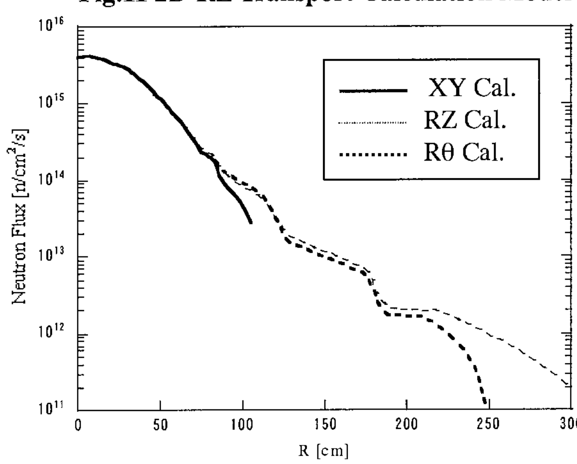


Fig.12 Radial Flux Distribution by DORT

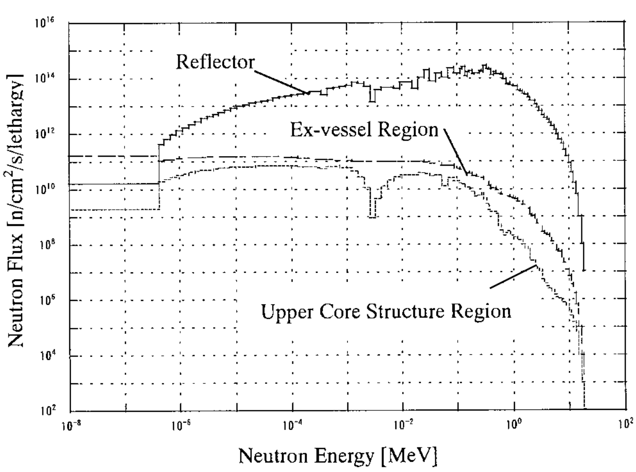


Fig.13 The typical neutron spectra by DORT

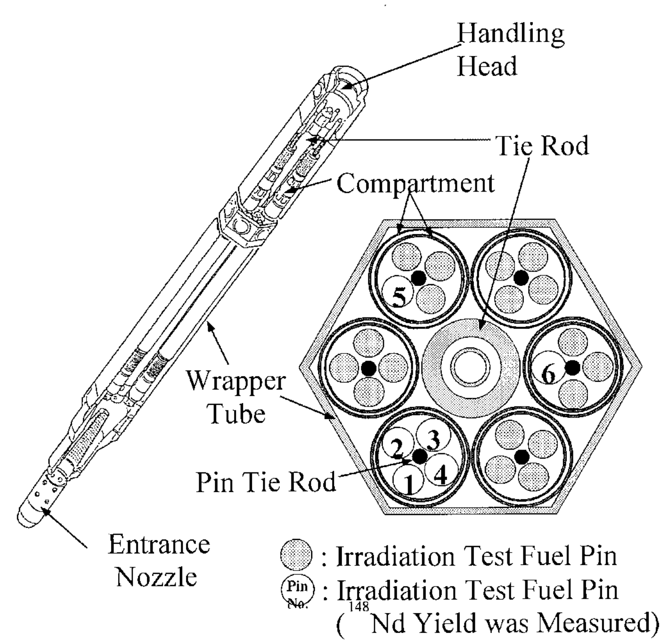


Fig.14 Structure of B5D-2 Irradiation Fuel Subassembly

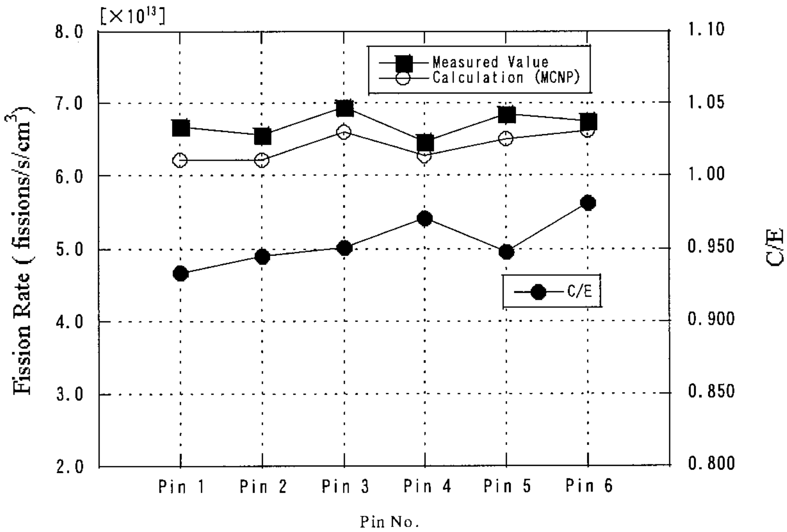


Fig.15 Comparison of MCNP Calculationswith Measurements

Burn-up (MWd/t) Burn-up ratio (atom%) Fast Neutron Fluence (≧0.1MeV) Fast Neutron Fluence (≧1.0MeV) Total Fluence Integrated Power

```
0
      BUNDLE ID= PFD512      CALCU .DATE= 1997.   5.  19.  18.  55.     LOC=    1  ==>  000       TYPE NO=   1
0
                                                  UNIT N/D     : N.10**24/C.C.
/                                                     FLUENCE : NVT              ; CUM.POWER: MWD
```

	MWD/T	A/O	>=0.1MEV	>=1.0MEV	TOTAL	CUM.POWER
1	0.000E+00	0.000E+00	3.973E+21	2.973E+20	8.913E+21	9.888E-02
2	0.000E+00	0.000E+00	8.031E+21	8.079E+20	1.624E+22	2.730E-01
3	0.000E+00	0.000E+00	1.550E+22	2.182E+21	2.758E+22	4.866E-01
4	4.972E+03	2.668E-01	2.657E+22	5.489E+21	4.133E+22	1.165E+00
5	3.263E+04	3.560E+00	3.837E+22	9.719E+21	5.455E+22	3.187E+01
6	4.006E+04	4.369E+00	4.935E+22	1.294E+22	6.843E+22	3.912E+01
7	4.664E+04	5.103E+00	5.864E+22	1.557E+22	8.076E+22	4.555E+01
8	5.198E+04	5.701E+00	6.566E+22	1.746E+22	9.046E+22	5.076E+01
9	5.549E+04	6.096E+00	7.026E+22	1.867E+22	9.691E+22	5.419E+01

Fig16 Neutron Fluence and Burn-up Data

Energy Group

Linear Heat Rate

```
S/A ID. NAME>>>(PFD512  ) : S/A TYPE>>>( DRIVER ) NO.= 1 : S/A LOACTION >>>( 000 ) NO.=  1
***R-6  NUCLEAR CHAR. 3RD REC. (FLUX BOC)          NEUTRON FLUX
```

NODE	1G	2G	3G	4G	5G	6G	7G	:	TOTAL(1-3G)	TOTAL(1-7G)	LINER HEAT RATE
1	6.96473E+12	5.39380E+13	1.36790E+14	1.27021E+14	4.83874E+13	3.62416E+13	1.38090E+13	:	1.97693E+14	4.23151E+14	3.10557E-01
2	2.23457E+13	1.20916E+14	2.56822E+14	2.22210E+14	8.10405E+13	5.51012E+13	1.85598E+13	:	4.00083E+14	7.76994E+14	8.67696E-01
3	7.04012E+13	2.52787E+14	4.48715E+14	3.54541E+14	1.15961E+14	6.32047E+13	1.71864E+13	:	7.71903E+14	1.32280E+15	1.55593E+00
4	1.88513E+14	4.45922E+14	6.89897E+14	4.94952E+14	1.22957E+14	4.05054E+13	6.46692E+12	:	1.32433E+15	1.98921E+15	8.42689E+00
5	3.45488E+14	6.45950E+14	9.18434E+14	5.96573E+14	1.02223E+14	1.56808E+13	7.76838E+11	:	1.90987E+15	2.62512E+15	2.01461E+02
6	4.64717E+14	8.35123E+14	1.15338E+15	7.12647E+14	1.05919E+14	9.81404E+12	1.24070E+11	:	2.45322E+15	3.28172E+15	2.43995E+02
7	5.58428E+14	9.93582E+14	1.35768E+15	8.22953E+14	1.17025E+14	8.83052E+12	4.78548E+10	:	2.90969E+15	3.85854E+15	2.80255E+02
8	6.28020E+14	1.11600E+15	1.51905E+15	9.15242E+14	1.28959E+14	9.08535E+12	3.78027E+10	:	3.26308E+15	4.31640E+15	3.09445E+02
9	6.71034E+14	1.19716E+15	1.62798E+15	9.80496E+14	1.38857E+14	9.19465E+12	1.29420E+10	:	3.49617E+15	4.62473E+15	3.28530E+02

CORE AVE. FLUX(NODE NO 5-15)   3.95927E+15

Neutron Flux
(n/cm²/s)

Fig17 Neutron Flux and Linear Heat Rate Data