Experimental Results and Analysis of Core Physics Experiments, FUBILA, for High Burn-up BWR Full MOX Cores

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Abstract

JNES has been performing MOX core physics experiments, FUBILA, in the EOLE critical facility of the CEA Cadarache center with collaboration of a French Consortium (CEA and COGEMA). The experiments have been designed to obtain the core physics data of operating conditions of full MOX BWR cores consisting of high burn up BWR MOX assemblies. The experiments consisting of seven different core configurations started from January 2005 and will be completed by August 2006. Theoretical analysis of the experimental data has been also carried out using a deterministic code, SRAC, and a continuous energy Monte Carlo calculation code, MVP, with major nuclear data libraries, JENDL-3.3, 3.2, ENDF/B-VI and JEFF-3.1 for the first critical core.

KEYWORDS: BWR, full MOX core, high burn up, critical experiments, core analysis, JENDL-3.3

1. Introduction

Japan Nuclear Energy Safety Organization (JNES) has been conducting an experimental program that is aimed to obtain a comprehensive data base for validation of core analysis methods applied to the full MOX ABWR [1] and also for high burn up MOX fuel expected in the future. As a part of this program, JNES has been performing a MOX core physics experimental program, FUBILA[2], with collaboration of a French Consortium (CEA and COGEMA). The experiments has been designed to obtain the core physics data of operating conditions of full MOX BWR cores consisting of high burn up BWR MOX assemblies. The experiments started from January 2005 in the EOLE critical facility of the CEA Cadarache center in France and will be completed by August 2006. Theoretical analysis of the experimental data has been also carried out with a continuous energy Monte Carlo calculation and a deterministic method with major nuclear data libraries. This paper presents a preliminary results of a part of the experiments and analysis about the first critical core.

2. Outline of FUBILA Experimental Program [2]

2.1 MOX Fuel Rods and Assemblies

BWR type MOX rods were newly fabricated for the program in a MOX fabrication plant of

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COGEMA, MELOX. They are composed of MOX pellets of Pu enrichments of 3.0, 5.0, 8.5 and 11.5 wt% with Pu compositions: 238Pu/239Pu/240Pu/241Pu/242Pu/241Am: 2/56/25/9/7/1 (January 2005) in a depleted uranium (0.25 wt% 235U) matrix. The outer diameter of the MOX pellet is 9.4 mm and the outer diameter of the cladding (Zry-2) is 11.0 mm and the thickness 0.7 mm. The effective length of the MOX rods is 800 mm. Those rods are sealed by over-claddings of AG3 for adjusting core moderation ratio and protecting the rods in handling. In addition to these BWR type MOX rods, the PWR type MOX rods that were used for the French EPICURE program are used in a driver region of the experimental cores. These MOX rods are composed of MOX pellets of Pu enrichments of 7.0 wt% with Pu compositions: 238Pu/239Pu/240Pu/241Pu/242Pu/241Am: 1/58/25/5/5/6 (January 2005) in a depleted uranium (0.24 wt% 235U) matrix. The outer diameter of the MOX pellet is 8.2 mm and the outer diameter of the cladding (Zry-4) is 9.5 mm and the thickness 0.6 mm.

2.2 Core Configurations

A test region is composed of four full MOX BWR assemblies consisting of the BWR type MOX rods and placed in the center of the EOLE core tank. That is surrounded by the assemblies of the EPICURE MOX rods. The test region is arranged in seven different configurations and the number of fuel rods in the driver region is adjusted to compensate the change of reactivity in the test region correspondingly. The core configurations are (1) 9x9 reference cores, (2) 40 % in-channel void cores, (3) 70 % in-channel void cores, (4) Axial void cores, (5) Control blade cores, (6) Burnable poison cores and (7) 10 x 10 cores. The dimensions of the lattices and the cores are shown in Reference [2].

2.3 Measurement Parameters

Measurement parameters are (1) Critical mass, (2) Core radial and axial fission rate distribution, (3) Core axial flux distribution, (4) Spectral indices and conversion factor, (5) Reactivity effects of increasing void fraction, insertion of a control blade and burnable poison rods.

2.4 Experimental Schedule

The first criticality of the 9x9 reference core was reached in January 6, 2005 and the experiments are in progress. The Fig. 1 shows the latest schedule of the experiments.

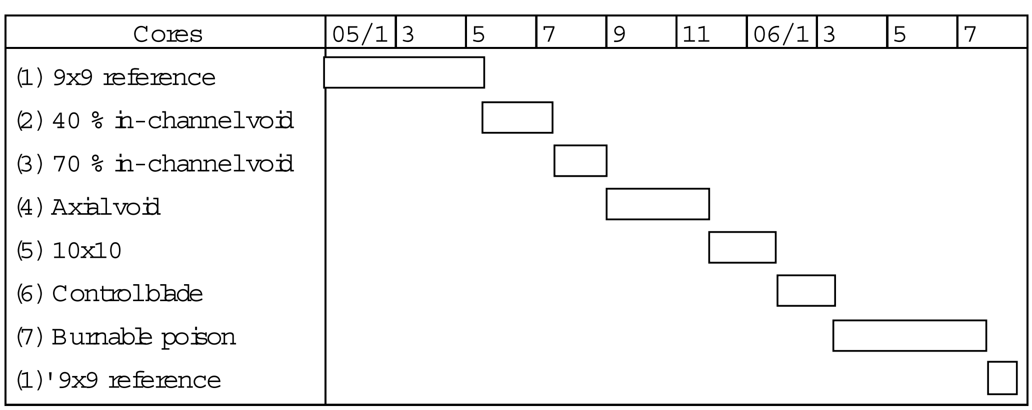


Figure 1: 1 Experimental schedule of FUBILA

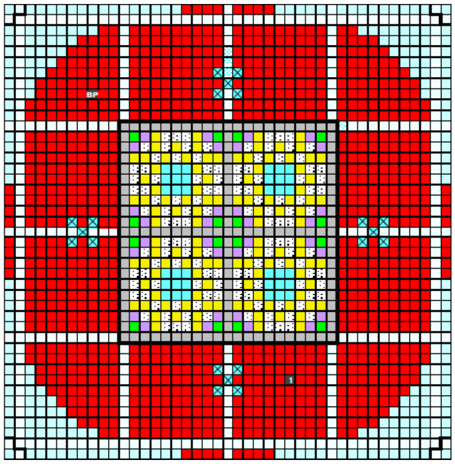
3. Preliminary Experimental Results

3.1 Critical mass

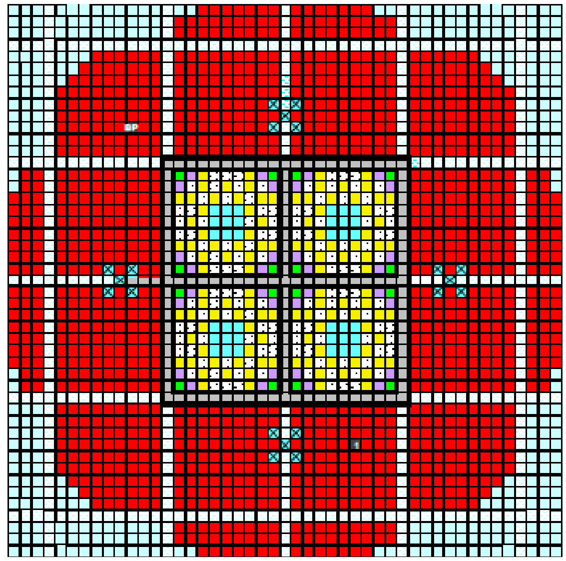
Table 1 shows the date of the measurement, core configuration, core temperature and doubling time of the 9x9 reference (0% in-channel core), 40% in-channel void, 70% in-channel void, Axial void and 10x10 cores when the critical masses were measured or the first criticality was achieved. The critical core configurations of the void cores are schematically shown in Figure. 2.

Table 1: Summary of core conditions at critical mass measurements (including preliminary data)

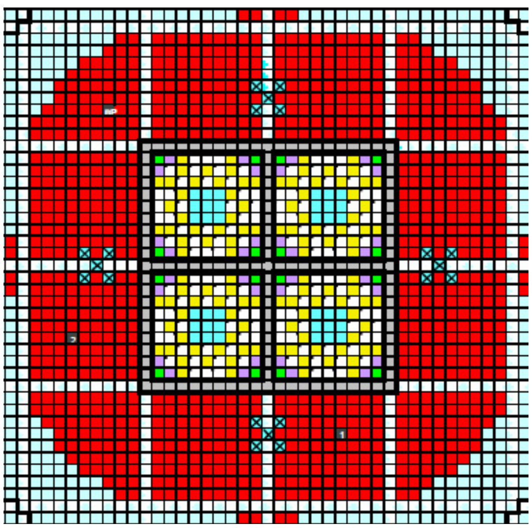
	9x9 Ref.	40% void	70% void	Axial void	10x10
Date	2005/2/14	2005/5/31	2005/8/5	2005/9/23	2005/12/8
Core configuration					
(Test region)					
MOX 3wt%	16(4x4)	16(4x4)	16(4x4)	16(4x4)	-
MOX 5wt%	32(4x8)	32(4x8)	32(4x8)	32(4x8)	-
MOX 8.5wt%	112(4x28)	112(4x28)	112(4x28)	112(4x28)	112(4x28)
MOX 11.5wt%	128(4x32)	128(4x32)	128(4x32)	128(4x32)	256(4x64)
Guide tube for water channel)	36(4x9)	36(4x9)	36(4x9)	36(4x9)	32(4x8)
AG3 micro rod	-	400(4x100)	-	-	-
AG3 block	-	-	4	4(Lower)	-
AG3 rod for out-channel	117	117	117	117	129
(Driver region)					
MOX 7wt%	598	894	1130	870	617
Guide tube for control cluster	20	20	20	20	20
Guide tube for pilot rod	1	1	1	1	1
Instrument. tube	1	1	1	1	2
Core temperature (oC)	19.8	20.0	20.0	20.0	
Doubling time (s)	39.5	34.9	12.4	37	
Excess reactivity ($)	0.1426	0.1553	0.221	0.148	



40% void core



70% void core



Axial void core

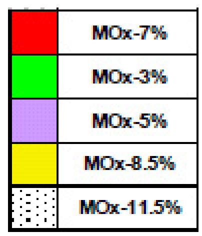


Figure2: Critical core configuration of void cores

3.2 Fission Rate Distribution

Core radial and axial fission rate distributions were measured for each critical core with integral gamma scanning. Fig. 3 shows a preliminary relative fission rate distribution in the diagonal direction of the test region for the 9x9 Ref (0% in-channel void), the 40% Void and the 70% Void cores. Fig. 4 also shows preliminary results of normalized fission rate distribution in the left-bottom assembly of the test region for the 9x9 Ref core.

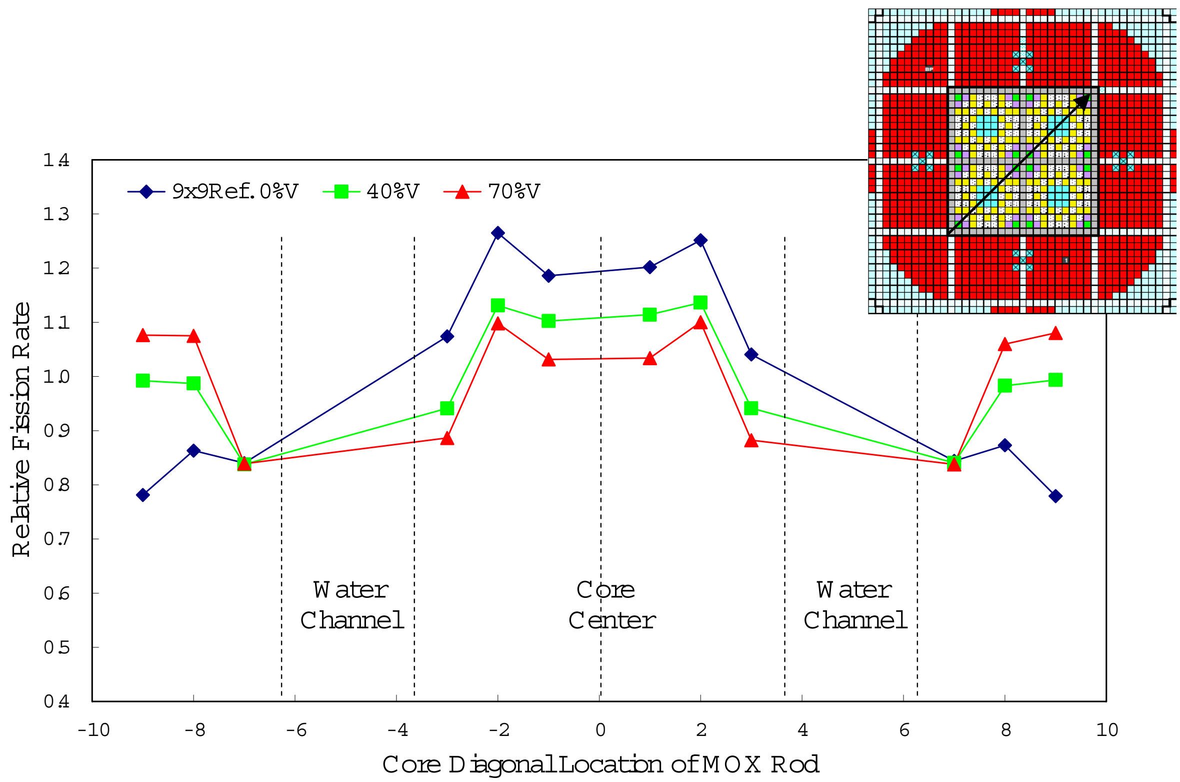


Figure 3: Normalized relative fission rate distributions along the diagonal direction in the test region

Core Center

0.9539	0.9957	1.1538	1.2545	1.2655	1.3198	1.2813	1.1918	1.1734
0.9362	1.0462	0.7721	0.9183	0.8153	0.9687	0.8590	1.2515	1.1787
1.0295	0.7329	0.9256	1.0493	1.3976	1.0896	1.0627	0.8729	1.3060
1.0747	0.8222	1.0056				1.1253	0.9857	1.3276
1.0175	0.6924	1.3065				1.3946	0.8038	1.2810
1.0016	0.7888	0.9860				1.0497	0.9077	1.2469
0.9035	0.6352	0.8314	0.9721	1.2863	1.0092	0.9275	0.7601	1.1506
0.7787	0.8539	0.6490	0.7709	0.6860	0.8311	0.7365	1.0645	0.9877
0.7729	0.7971	0.9114	1.0043	1.0373	1.0805	1.0459	0.9502	0.9469

Figure 4: Normalized fission rate distribution in the left-bottom assembly in the test region of the 9x9 Ref core

4. Preliminary Analysis of 9x9 Ref Core

Preliminary analysis of the experimental results is in progress using SRAC [3, 4] and MVP [5] code systems. The former system performs pin cell and core calculations with deterministic methods and the later system does a core calculation with the continuous energy Monte Carlo method. Following results are preliminary results about the 9x9 Ref core.

4.1 Analysis with SRAC

The three libraries equipped with the SRAC code system were used in the analyses. These libraries correspond to the fast, thermal and resonance energy ranges and were generated from the nuclear data files, JENDL-3.3[6]. The resonance library was used in a PEACO module of SRAC to treat resonance shielding in detail. The boundary energies of the resonance library in this analyses are 1.8554eV and 961.12eV that were determined by parameter surveys with a single cell model of the 7% enrichment MOX fuel pin (7%-MOX pin).

Collision probability calculations (Pij-calculation) for each specific cell configuration were performed by SRAC to prepare 107 energy group cell averaged cross sections and coarse-group macro cross sections for diffusion and discrete ordinate transport core calculations. These core calculations were performed using CITATION and TWOTRAN modules in the SRAC code system. Two dimensional XY calculation with 16 energy group structure was employed for the core analyses with leakage of axial direction calculated by the measured axial buckling. The effective delayed neutron fraction was also analyzed by SRAC-CITATION with three-dimensional XYZ geometry with 21 energy group.

4.2 Analysis with MVP

Three dimensional continues energy Monte Carlo calculations were performed by the MVP code with detail treatment of geometry and neutron energy using MVP's library processed from JENDL-3.3 and 3.2, ENDF/B-VI and JEFF-3.1. A number of simulated particles was 10,000 per batch, and 2,000 batches calculation was performed. Therefore, the total history number was twenty millions.

4.3 Criticality

Core calculations with above methods were applied to the 9x9 Ref core (see Table 1 and Fig. 2) and the excess reactivity was subtracted from the values of the effective keff to obtain the critical keff. The measured excess reactivity of 9x9 Ref core in unit $ is converted to be 49.7 pcm using the calculated delayed neutron fraction of 348.6 pcm with SRAC-CITATION (JENDL-3.3). Table 2 summarize the result of the critical keff.

Table 2: Critical keff of FUBILA 9x9 Ref core

SRAC		MVP			
CITATION	TWOTRAN				
JENDL-3.3	JENDL-3.3	JENDL-3.3	JENDL-3.2	ENDF/B-VI	JEFF-3.1
0.9975	0.9998	1.0029[a]	1.0033[a]	0.9996[a]	1.0019[a]

[a] Statistical error (1 sigma) is 0.0002 dk.

The difference in the critical keff between SRAC-CITATION and MVP with JENDL-3.3 is -0.0054 dk, which shows an analysis error of the diffusion calculation model applied in this study. The difference in the critical keff between SRAC-TWOTRAN and SRAC-CITATION is 0.0023 dk, which shows transport effect in this core analysis. The difference in the critical keff of MVP shows specific characteristics in the difference nuclear data libraries. The difference between JENDL-3.3 and 3.2 is small; that between JENDL 3.2 and ENDF/B-VI is rather large.

Table 3 shows comparison to the preceding BWR full MOX experiments of BASALA [7, 8]. The main difference of the core configurations of FUBILA 9x9 Ref from Core 1 Ref of BASALA is (1) the BWR assemblies consisting of new BWR type MOX fuel (Pu isotopic compositions) and higher Pu enrichment in the test region. The Core 2 Ref of BASALA has the H/HM (atomic ratio of hydrogen to heavy metal) of 9 representing cold conditions of BWR. On the other hand, Core 1 Ref of BASALA and FUBILA 9x9 Ref has the H/HM of 5 representing hot operating conditions. The decrease of the critical keff for FUBILA 9x9 Ref. would be due to the difference of Pu composition of the assemblies of test region: lower composition of 241Am and higher 241Pu for FUBILA and higher 241Am and lower 241Pu for BASALA.

Table 3: Comparison of critical keff of MVP analysis [b] for 9x9 Ref core

	FUBILA	BASALA	
	9x9 Ref	Core 1 Ref.	Core 2 Ref
JENDL-3.3	1.0029	1.0074	1.0066
JENDL-3.2	1.0033	1.0086	1.0086

[b] Statistical error (1 sigma) is 0.0002 dk

4.4 Radial Fission Rate Distribution

The calculated relative fission rate distributions are shown in Fig. 4 compared with the measured ones. The statistical error of the MVP analysis is less than 0.5 % (relative error) and the experimental error is 1.5%. The relative difference among the calculated values is less than 1% and small enough. The relative difference between the measured data and the MVP analysis is almost less than 1.5% with maximum error of 3.7% at the location, -3.

The calculated fission rate distributions in the left-bottom assembly of the test region were compared with the measured. Fig. 5 shows the values of (calculated - measured)/measured in % for SRAC-CITATION, -TWOTRAN and MVP with JENDL-3.3. The values of root mean square (RMS) is also shown in the figure. This comparison indicates that the RMS of SRAC-TWOTRAN and MVP is same level as the experimental error and that of SRAC-CITATION is a little larger than the experimental error.

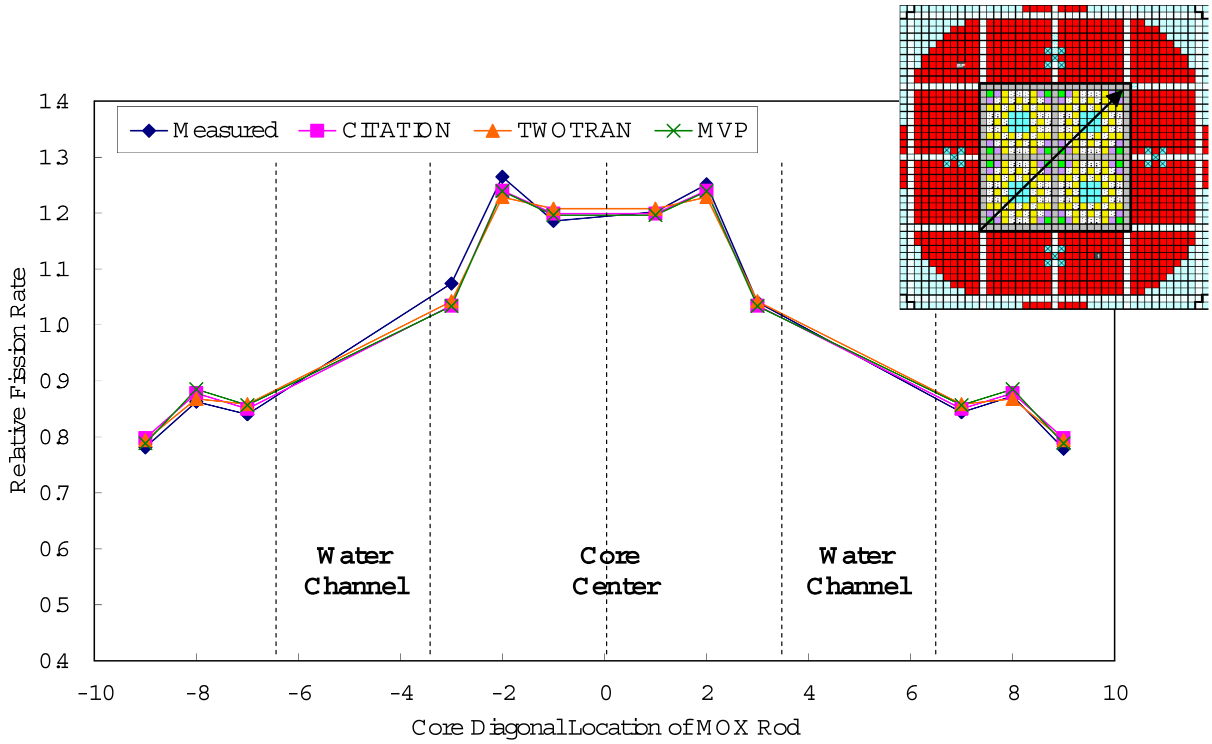


Figure 4: Comparison of relative fission rate distribution along the core diagonal locations for 9x9 Ref core

5. Conclusion

JNES has been performing the MOX core physics experiments, FUBILA, in the EOLE critical facility of the CEA Cadarache center with the collaboration of a French Consortium (CEA and COGEMA). The critical experiments have been designed to obtain the core physics data of operating conditions of full MOX BWR cores consisting of high burn up BWR MOX assemblies. The experiments consisting of seven different core configurations started from January 2005 and will be completed by August 2006. The theoretical analysis of the experimental data has been also carried out using the deterministic code, SRAC, and the continuous energy Monte Carlo calculation code, MVP with the major nuclear data libraries, JENDL-3.3, 3.2, ENDF/B-VI and JEFF-3.1 for the first critical core

The preliminary analysis results of the 9x9 Ref critical core indicate that (1) the critical keff is 0.9975 to 1.0033, (2) the critical keff of diffusion calculation gives 0.54 %dk under estimate and that of the transport calculation 0.31 %dk underestimate compared with Monte Carlo calculation, (3) the critical keff is smaller than the preceding experiments, BASALA, which would be due to the different isotopic composition of 241Pu and 241Am of the MOX fuel, (4) the measured radial fission rate distributions are generally well reproduced by the diffusion, the transport and the Monte Carlo calculations, (5) the nuclear data library used in this study does not make any difference on the analysis of the fission rate distributions.

Core Center

2.45	0.71	-2.01	-1.69	-1.77	-0.75	-0.55	-0.36	1.42
3.18	-0.87	1.44	1.37	0.61	1.52	2.20	-1.63	0.75
0.55	1.41	0.51	-0.43	-2.54	0.02	-3.32	0.57	-2.44
-0.65	2.26	0.05				-3.16	-0.23	-1.33
-0.13	1.76	-3.19				-2.33	2.05	-2.96
0.38	1.04	-2.03				-0.47	2.56	-1.09
1.47	4.51	1.53	-0.63	-1.67	-0.31	0.30	3.04	-1.74
4.02	2.07	2.29	3.39	2.71	1.17	0.92	-2.58	1.53
2.44	1.62	0.60	0.11	-2.04	-1.19	-1.03	1.66	3.20

RMS = 1.90%

(a) SRAC-CITATION

Core Center

2.19	0.39	-1.40	-0.52	-0.52	0.41	0.12	-0.32	1.99
2.62	-2.16	-0.03	-0.30	-1.71	-0.21	0.67	-2.76	0.79
0.77	-0.16	1.16	0.31	-0.71	0.71	-2.82	-0.93	-1.78
0.19	0.43	0.82				-2.49	-1.93	-0.18
0.74	-0.81	-1.35				-0.50	-0.30	-1.72
1.26	-0.79	-1.25				0.27	0.87	0.09
1.74	2.95	2.32	0.16	0.20	0.46	0.95	1.55	-1.13
3.35	0.72	0.76	1.52	0.11	-0.65	-0.64	-3.85	1.20
1.79	0.97	0.87	0.99	-1.19	-0.34	-0.81	1.11	2.94

RMS = 1.41%

(b) SRAC-TWOTRAN

Core Center

0.89	-0.20	-0.90	-0.37	-0.13	0.69	-0.18	-0.90	0.12
2.51	-1.67	0.23	0.55	-1.16	0.71	1.58	-2.76	0.20
0.16	0.70	0.78	0.40	-1.77	0.78	-4.46	-0.04	-2.07
0.27	0.70	1.37				-2.42	-1.03	0.10
0.70	0.39	-2.51				-1.56	0.25	-1.34
1.68	0.44	-0.81				0.36	1.72	0.24
1.72	3.20	1.19	0.61	-0.97	1.02	0.57	1.81	-0.63
3.25	1.83	1.00	2.78	1.32	-0.38	0.22	-3.36	0.61
0.18	0.88	0.85	1.41	-1.23	-0.27	-1.42	1.00	1.63

RMS = 1.44%

(c) MVP

Figure 5: (Calculated - measured)/measured in % for the calculated values of the normalized fission rate distributions of SRAC-CITATION, -TWOTRAN and MVP for 9x9 Ref core

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