Compilation of Measurement and Analysis Results of Isotopic Inventories of Spent BWR Fuels

Toru YAMAMOTO
Japan Nuclear Energy Safety Organization
TOKYU REIT Toranomon Bldg., 3-17-1, Toranomon, Minato-ku, Tokyo, 105-0001, Japan
Tel:+81 3 4511 1762, Fax:+81 3 4511 1897, Email: yamamoto-toru@jnes.go.jp

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Abstract

The current spent fuel isotopic composition database SFCOMPO (http://www.nea.fr/sfcompo/) contains the isotopic inventory data of the fuel samples taken from the BWR fuel assemblies irradiated in JPDR (6x6, UO_2), Tsuruga-1 (7x7, UO_2), Fukushima-Daiichi-3 (8x8-1, UO_2) and Fukushima-Daini-2 (8x8-2, UO_2). This report presents the additional measurement data of the fuel pellet samples taken from the fuel assemblies irradiated in the Japanese commercial BWR plants. They are 8x8-4, 9x9-9, and 9x9-7 UO_2 fuel assemblies, and 8x8-2 MOX fuel assemblies. The measured data are mainly on U, Pu, and Nd isotopes. The burnups of the samples were evaluated based on the Nd-148 method. The specifications of the cores and fuel assemblies are presented, which were taken from published documents. The report also presents the analysis results for part of the samples based on infinite assembly calculations with a deterministic analysis code SRAC and a Monte Carlo burnup calculation code MVP-BURN.

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1. Introduction

The current spent fuel isotopic composition database SFCOMPO (http://www.nea.fr/sfcompo/) contains the isotopic inventory data of the irradiated BWR UO_2 nuclear fuels discharged from JPDR, Tsuruga-1, Fukushima-Daiichi-3 (Unit 3 of Fukushima Power Station 1 (1F3)), and Fukushima-Daini-2 (Unit 2 of Fukushima Power Station 2 (2F2)). In addition to these data, isotopic inventory data are available from the measurement that has been implemented on the other irradiated BWR fuels as part of fuel mechanical integrity studies in Japan. They are

(1) BWR 8x8-4 UO_2 Fuel of Unit 2 of Fukushima Power Station 2 (2F2),

(2) BWR 9x9-9 UO_2 Fuel of Unit 1 of Fukushima Power Station 2 (2F1),

(3) BWR 9x9-7 UO_2 Fuel of Unit 1 of Fukushima Power Station 2 (2F1),

(4) BWR 8x8-2 MOX Fuel of Tsuruga-1.

The measured data are mainly on U and Pu, and Nd isotopes for determining the burnups.

Part of the data have been also analyzed using lattice analysis codes and compared with the measured data to validate the analysis methods.[1,2] This report presents the measured data and analysis results.

2. BWR 8x8-4 UO_2 Fuel of Unit 2 of Fukushima Power Station 2

2.1 Fuel Assembly and Irradiation History of BWR 8x8-4 UO_2 Fuel

Isotopic inventories were obtained for four BWR 8x8-4 UO_2 assemblies, which were irradiated as lead-use fuel assemblies of high burnup fuel in Unit 2 of Fukushima Power Station 2 (2F2) and each discharged after one, two, three and five cycles of irradiation. The assemblies are referred to as 2F2D1, 2F2D2, 2F2D3, and 2F2D8, respectively. The major core parameters of 2F2[3,4] and the major characteristics of the BWR 8x8-4 UO_2 fuel assembly[3,5] are shown in **Table 2.1.1** and **2.1.2**. **Figure 2.1.1** shows the enrichment distribution among fuel rods in the assembly.[3,6] The data of irradiation cycles and assembly- and rod-average burnups of the four assemblies[6] are shown in **Table 2.1.3**. **Figure 2.1.2** shows the locations of the assemblies in the core from the first through the fifth cycle.[6] **Figure 2.1.3** shows the power histories of the assembly 2F2D8 and the fuel rods of the assembly.[6]

A core average void fraction, which is defined as volume fraction of the steam void in the in-channel volume, is 43% for the cores fully loaded with the BWR 8x8-4 UO_2 fuel assemblies.[7] Core-average axial void distribution was estimated from the axial void distribution in the hottest channel given in Reference 7 so that the axially-averaged value coincides with 43%. **Table 2.1.4** shows the estimated in-channel void fraction for each axial node. The length of one node is about 15.5 cm.

Table 2.1.1 Major core parameters of 2F1 and 2F2[3,4)]

Reactor thermal power	3,293 MW
Core flow	48.3×10^3 t/h*
Inlet sub-cooling	11.4 kcal/kg
Outlet coolant temperature	286 ℃*
Core pressure	71.7 kg/cm^2 a*
Core	
Number of fuel assemblies	764
Control rod pitch (cm)	305*
Fuel assembly pitch (cm)	15.2*
Inside of channel box (cm)	13.4*
Effective fuel length	3.71 m*
Equivalent core diameter	4.75 m*
Power density	50.0 kW/l

*Round number

Table 2.1.2 Major characteristics of BWR 8x8-4 UO_2 fuel assembly[3,5)]

1. Fuel assembly	
Lattice	8x8
Fuel rod pitch (mm)	16.3
Number of fuel rods	60
Average enrichment (wt%)	3.0*
2. Fuel rod	
Out diameter (mm)	12.3*
Thickness of cladding (mm)	0.86* (include Zr lining 0.1*mm)
Cladding material	Zry-2 (Zr lining)
Pellet diameter (mm)	10.4*
Pellet-cladding gap (mm)	0.20*
Pellet density (%TD)	97*
He pressure (MPa)	0.5*
Temperature of outside of cladding (℃)	310
Content of Gd_2O_3 (wt%)	≦6
3. Water rod	
Number of water rod	1
Material	Zry
Outer diameter (mm)	34.0*

*Round number

Table 2.1.3 Irradiation periods and burnups of BWR 8x8-4 fuel assemblies[6)]

Irradiation cycle (Operating cycle)	1 (5th)	2 (6th)	3 (7th)	4 (9th)	5 (10th)
Start and end of cycle	1989/1/14 -1990/3/8	1990/7/4 -1991/8/24	1991/11/14 -1992/11/14	1994/7/14 -1995/9/5	1995/12/12 -1997/1/30
Assembly ID discharged for PIE	2F2D1	2F2D2	2F2D3	-	2F2D8
Assembly average burnups (GWd/t)	12.6	24.6	34.6	44.3	47.9
Fuel rod average burn-up (GWd/t)	9.6-14.1	21.4-27.1	30.5-38.0	39.8-47.4	43.2-51.0

Table 2.1.4 Axial void distribution of BWR fuel assemblies

(Unit: %)

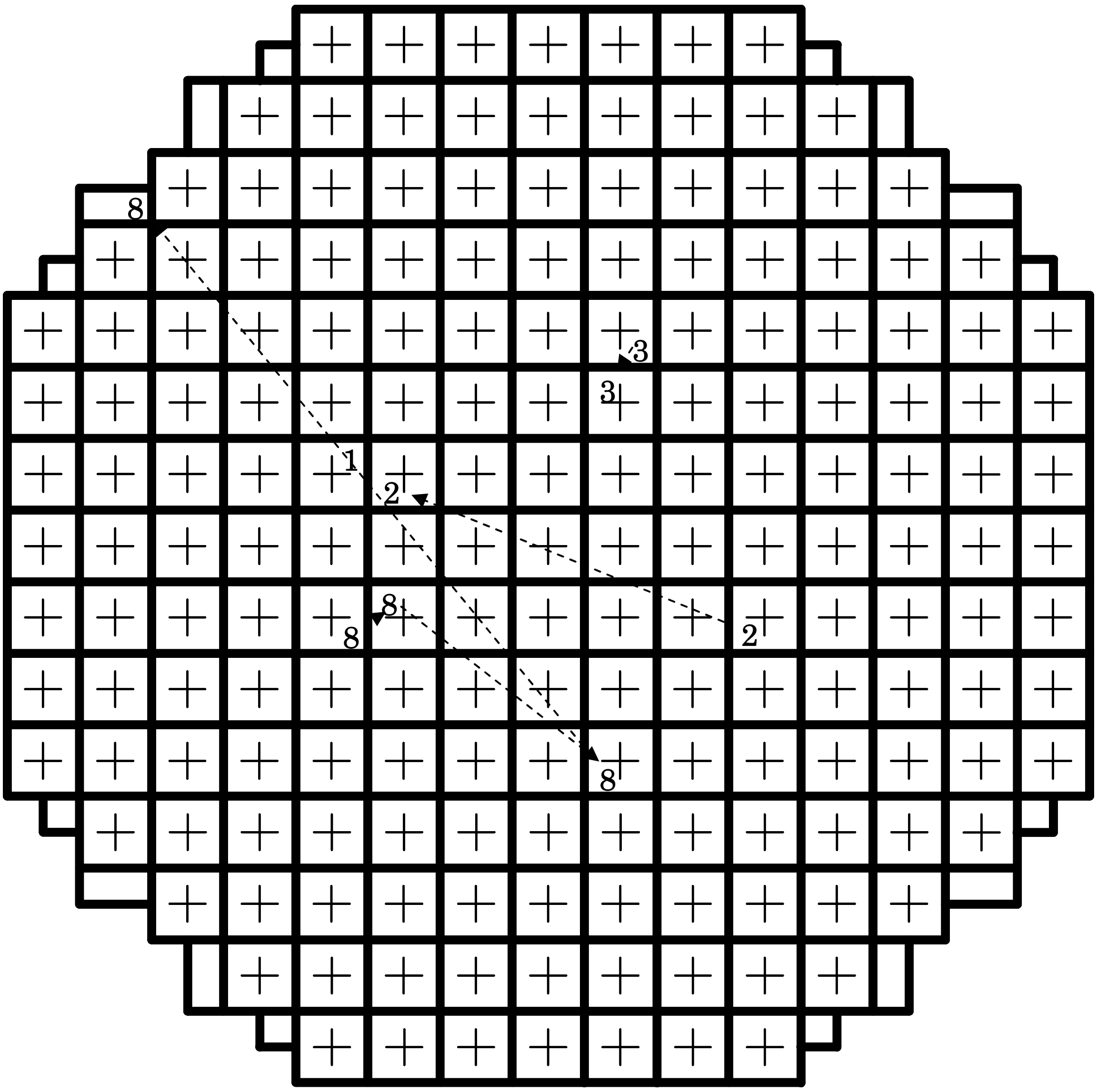
Node	2F2 BWR8x8-4 UO2	2F1 BWR9x9-9 UO2	2F1 BWR9x9-7 UO2	Turuga-1 BWR8x8-2 MOX
1	0	0	0	0
2	1	1	1	0
3	4	4	5	0
4	9	9	11	0
5	15	16	17	2
6	23	23	26	5
7	29	29	31	9
8	34	35	36	14
9	40	40	41	19
10	43	44	45	24
11	46	47	48	29
12	49	50	50	34
13	52	53	53	38
14	54	55	55	41
15	55	57	57	45
16	58	59	58	48
17	58	60	59	50
18	60	62	61	52
19	61	62	62	53
20	63	64	62	54
21	63	65	63	55
22	64	65	63	56
23	64	66	63	57
24	64	66	64	58
Average	43	43	43	31

	A	B	C	D	E	F	G	H
1	5	4	3	3	3	3	4	5
2	4	2	G	2	2	G	2	4
3	3	G	1	4	4	1	G	3
4	3	2	4	W		4	2	3
5	3	2	4			4	2	3
6	3	G	1	4	4	1	G	3
7	4	2	G	2	2	G	2	4
8	5	4	3	3	3	3	4	5

1 to 5: order of enrichment (higher to lower)

W: water rod

Fig. 2.1.1 Enrichment distribution among fuel rods of BWR 8x8-4 fuel assembly[3,6]



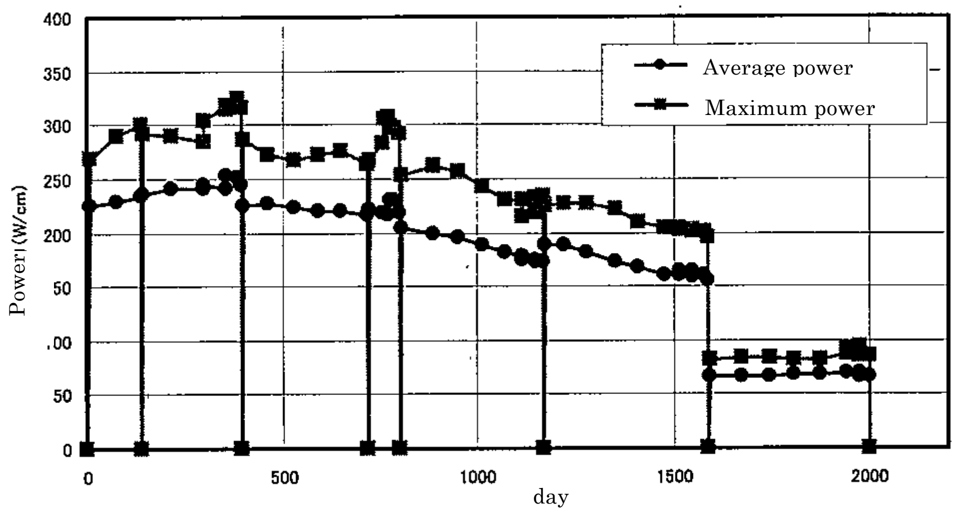
1: 2F2D1

2: 2F2D2

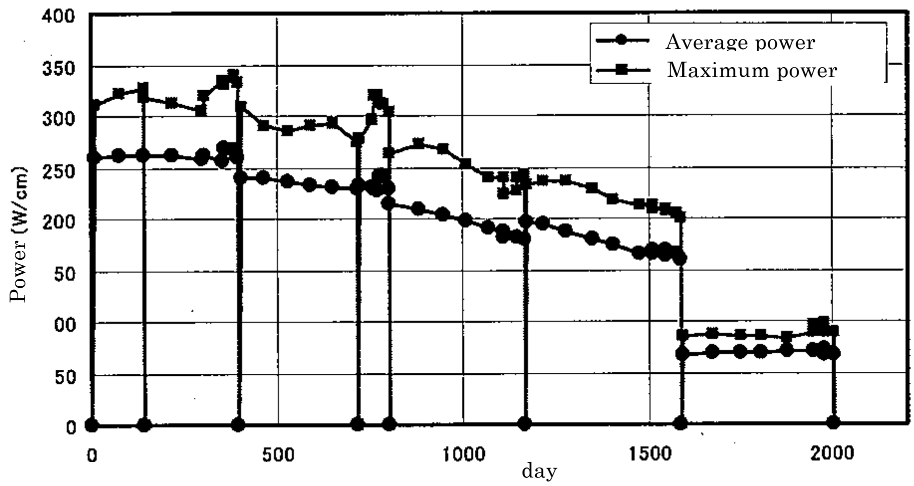
3: 2F2D3 (note: same location for 2nd and 3rd cycle)

8: 2F2D8 (note: same location fro 2nd and 3rd cycle)

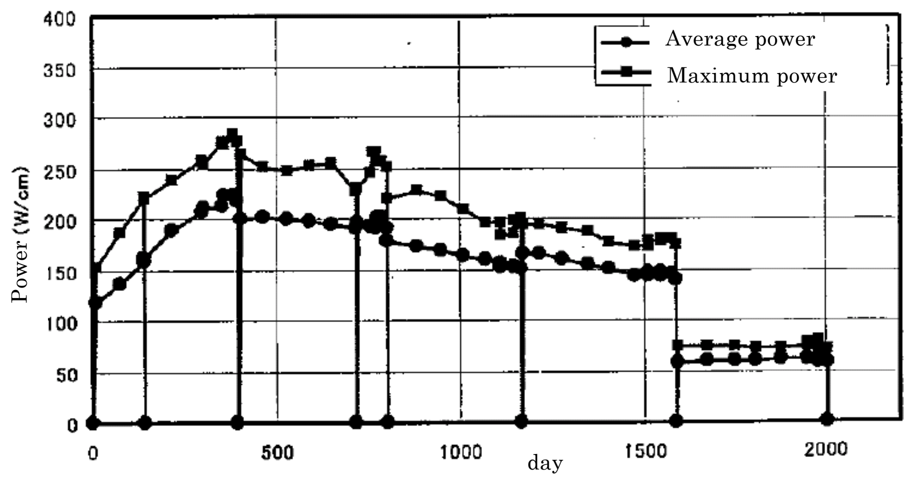
Fig. 2.1.2 Locations of irradiation of BWR 8x8-4 fuel assemblies in 2F2[6)]



(a) Assembly power history of 2F2D8



(b) Fuel rod power of A4 and H5 of 2F2D8



(c) Fuel rod of B3 of 2F2D8

Fig. 2.1.3 Power histories of fuel assembly and rods of 2F2D8[5)]

2.2 Measured Isotopic Inventories of BWR 8x8-4 UO_2 Fuel

Isotopic inventories were measured for the samples of fuel pellet-cross-sections (pellet-average sample) and those of fuel pellet local areas (pellet-local sample) with about 1 x 1 mm at r/r_0 0.0 (C), 0.48 (M) and 0.87 (O), where r and r_0 are the radial distances of the sample from the pellet center and the pellet radius 5.4 mm, respectively. **Table 2.2.1** listed the number of samples from the assemblies.

2.2.1 Measurement Method

The isotopic inventories of the pellet samples of the BWR 8x8-4 fuel assemblies were measured in the Nippon Nuclear Fuel Development, and the isotopic dilution mass spectrometry (IDM) was applied to Nd and U isotopes with spikes in the forms of ^{150}Nd and natural U, respectively, the alpha spectrometry and mass spectrometry to Pu isotopes, and the alpha spectrometry to Am and Cm isotopes.

The measurement errors of isotopic ratios in atomic percent are less than 1 to 2% for major U and Pu isotopes except ^{234}U and those of inventories in a mass ratio to total uranium (mg/gU(t) or µg/gU(t)) at the measurement date are 1.5 to 2% for U isotopes, 3 to 4% for Pu isotopes, 3% for total Pu, 4 to 8% for Am isotopes, and 3 to 5% for Cm isotopes.

2.2.2 Measured Burnups and Isotopic Inventories of Samples

The burnups of the samples were determined by the Nd-148 method[8)] with the inventory data of U, Pu and ^{148}Nd, and using the representative values of the effective fission yield of ^{148}Nd, and energy release per fission, 1.70% and 204.6 MeV/fission, respectively, which were analyzed by assembly burnup calculations on the 9x9-7 assembly of average enrichments 3.4 wt% using a continuous energy Monte Carlo burnup calculation code.[2)] The errors of the burnups are estimated to be about 6.4%,[9)] which include the errors related to adding the spikes, the mass spectrometry, the alpha spectrometry, the effective fission yield of ^{148}Nd and the effective energy release per fission.

The measured burnups and the isotopic inventories in the unit of the weight ratio to total U at the measurement date are shown in **Table 2.2.2** for the pellet-average samples and **Table 2.2.3** for the pellet-local samples. The data of Pu isotopes shown in Table 2.2.2(3) was ones measured using the isotopic dilution mass spectrometry (IDM). The errors of measurements are shown in Chapter 4.

Table 2.2.1 Number of samples for BWR 8x8-4 fuel assemblies

Assembly	Pellet-average sample	Pellet-local sample*		
		C	M	O
2F2D1	6	-	2	2
2F2D2	5	-	2	2
2F2D3	7	-	3	3
2F2D8	8	2	2	2

*r/r_0: 0.0 (C), 0.48 (M), 0.87 (O) (see text.)

Table 2.2.2(1) Measured isotopic inventories and burnups of pellet-average samples for BWR 8x8-4 fuel assemblies (1/3)

Sample ID		TU101	TU102	TU103	TU104	TU105	TU106	TU201	TU202	TU203	TU204	TU205
Assembly		2F2D1	2F2D1	2F2D1	2F2D1	2F2D1	2F2D1	2F2D2	2F2D2	2F2D2	2F2D2	2F2D2
Rod location		F6	F6	B3	B3	B3	F6	F6	F6	B3	B3	B3
Type of pellet		UO_2	UO_2	$(U,Gd)O_2$	$(U,Gd)O_2$	$(U,Gd)O_2$	UO_2	UO_2	UO_2	$(U,Gd)O_2$	$(U,Gd)O_2$	$(U,Gd)O_2$
Enrichment (wt%)		4.5	4.5	3.4	3.4	3.4	4.5	4.5	4.5	3.4	3.4	3.4
G_2O_3 (wt%)		-	-	4.5	4.5	4.5	-	-	-	4.5	4.5	4.5
Axial height (mm)*1		665～700	3401～3436	700～735	1300～1335	3303～3338	1354～1389	865～900	3567～3597	865～900	1451～1486	3465～3500
U isotope (atomic%)	U-234	0.03	0.04	0.03	0.03	0.03	0.03	0.03	0.03	0.02	0.02	0.02
	U-235	3.44	2.87	2.58	2.40	2.47	3.04	2.17	1.65	1.62	1.50	1.29
	U-236	0.26	0.30	0.18	0.20	0.21	0.32	0.47	0.51	0.36	0.36	0.37
	U-238	96.27	96.79	97.21	97.37	97.29	96.61	97.33	97.81	98.00	98.12	98.32
U isotope (mg/gU(t))*2	U-234	0.34	0.35	0.25	0.26	0.26	0.32	0.28	0.28	0.20	0.21	0.21
	U-235	33.94	28.31	25.49	23.67	24.44	30.04	21.39	16.33	16.04	14.83	12.75
	U-236	2.62	3.03	1.80	2.03	2.04	3.14	4.68	5.04	3.57	3.54	3.68
	U-238	963.10	968.31	972.46	974.04	973.25	966.51	973.64	978.35	980.19	981.42	983.35
Pu assay		α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum
Pu isotope (atomic%)	Pu-238	0.26	0.31	0.21	0.29	0.22	0.39	1.07	1.16	0.90	1.01	0.92
	Pu-239	82.27	76.62	82.36	79.31	77.01	78.38	67.45	59.56	66.40	64.9	59.84
	Pu-240	12.22	16.57	12.40	14.15	15.89	14.24	19.33	25.10	20.33	21.33	25.63
	Pu-241	4.77	5.76	4.6	5.57	6.21	6.26	10.34	10.69	10.17	10.09	10.07
	Pu-242	0.48	0.75	0.42	0.68	0.67	0.78	1.82	3.49	2.20	2.67	3.54
Pu isotope (μg/gU(t))*2	Pu-238	20	20	10	20	10	30	110	90	90	100	70
	Pu-239	4860	4140	4810	4610	4640	5770	6700	4670	6730	6520	4320
	Pu-240	730	900	730	830	960	1050	1930	1980	2070	2150	1860
	Pu-241	280	310	270	330	380	470	1040	850	1040	1020	730
	Pu-242	30	40	20	40	40	60	180	280	230	270	260
Total Pu (μg/gU(t))		5916	5410	5847	5817	6031	7373	9947	7858	10162	10066	7233
Nd isotope (mg/gU(t))*2	Nd-148	0.1577	0.2062	0.1121	0.1051	0.1384	0.1818	0.3342	0.3787	0.2791	0.2677	0.2592
Am isotope (μg/gU(t))*2	Am-241	59.4	69.9	58.3	70.8	69.5	102	156	143	161	179	129
	Am-243	1.73	1.63	1.53	1.87	1.53	1.82	12.9	19.2	20.3	35.2	23.5
Cm isotope (μg/gU(t))*2	Cm-242	0.0005	0.0007	0.0004	0.0006	0.0007	0.0011	0.066	0.062	0.068	0.075	0.053
	Cm-243	0.046	0.095	0.135	0.181	0.061	0.148	0.36		0.89	1.21	0.38
	Cm-244	0.121	0.170	0.115	0.255	0.138	0.458	6.32	5.42	6.75	8.81	4.64
%FIMA for Nd-148 Y(%)=1.7		1.46	1.90	1.04	0.98	1.28	1.68	3.03	3.43	2.55	2.45	2.38
Burnups (GWd/t) for GWd/t/%FIMA=9.6		14.0	18.2	10.0	9.4	12.3	16.1	29.1	32.9	24.5	23.5	22.8
Measurement date of TRU except Cm-242		1994/9/5	1994/8/31	1994/10/31	1994/10/31	1994/10/31	1994/10/31	1994/10/31	1994/10/31	1994/8/31	1994/10/31	1994/10/31
Date of normalization of Cm-242		1994/9/30	1994/9/30	1994/9/30	1994/9/30	1994/9/30	1994/9/30	1994/9/30	1994/9/30	1994/9/30	1994/9/30	1994/9/30

*1: distance from the shoulder of the upper-end plug of fuel rods which is about 350 mm higher than the top of an effective fuel part
*2: weight ratio to uranium at measurements

Table 2.2.2(2) Measured isotopic inventories and burnups of pellet-average samples for BWR 8x8-4 fuel assemblies (2/3)

Sample ID		TU301	TU302	TU304	TU306	TU308	TU309	TU311	TU501	TU502	TU503	TU505	TU506	TU510	TU511
Assembly		2F2D3	2F2D3	2F2D3	2F2D3	2F2D3	2F2D3	2F2D3	2F2D8	2F2D8	2F2D8	2F2D8	2F2D8	2F2D8	2F2D8
Rod location		H5	H5	A4	A4	B3	B3	B3	H5	H5	H5	A4	A4	B3	B3
Type of pellet		UO_2	UO_2	UO_2	UO_2	$(U,Gd)O_2$	$(U,Gd)O_2$	$(U,Gd)O_2$	UO_2	UO_2	UO_2	UO_2	UO_2	$(U,Gd)O_2$	$(U,Gd)O_2$
Enrichment (wt%)		3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4
G_2O_3 (wt%)		-	-	-	-	4.5	4.5	4.5	-	-	-	-	-	4.5	4.5
Axial height (mm)*1		1250～1285	3620～3655	1187～1222	3595～3630	801～836	1263～1298	3500～3535	847.6～867.6	1597.2～1617.2	3246.6～3267.1	1820.6～1840.6	3200～3220	1098～1118	3380～3400
U isotope (atomic%)	U-234	0.02	0.02	0.02	0.02	0.02	0.02	0.02	0.01	0.02	0.01	0.01	0.01	0.02	0.02
	U-235	0.69	0.66	0.66	0.63	1.13	0.95	0.73	0.40	0.25	0.15	0.25	0.16	0.49	0.32
	U-236	0.49	0.47	0.48	0.47	0.44	0.48	0.48	0.53	0.53	0.52	0.53	0.51	0.55	0.55
	U-238	98.81	98.85	98.84	98.88	98.41	98.55	98.77	99.05	99.20	99.31	99.21	99.31	98.94	99.12
U isotope (mg/gU(t))*2	U-234	0.17	0.17	0.17	0.19	0.22	0.21	0.22	0.15	0.15	0.13	0.15	0.14	0.15	0.16
	U-235	6.78	6.55	6.54	6.22	11.13	9.32	7.15	3.97	2.50	1.50	2.45	1.62	4.83	3.13
	U-236	4.84	4.66	4.77	4.65	4.40	4.75	4.78	5.24	5.28	5.20	5.24	5.04	5.47	5.44
	U-238	988.21	988.62	988.52	988.93	984.26	985.72	987.85	990.64	992.07	993.16	992.17	993.20	989.55	991.27
Pu assay		α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum
Pu isotope (atomic%)	Pu-238	2.49	1.78	2.54	1.84	1.69	2.08	1.79	4.30	4.66	4.09	4.56	3.83	3.95	3.43
	Pu-239	52.05	50.37	51.93	49.84	59.15	55.14	49.71	45.15	41.06	38.43	41.25	38.38	45.09	40.6
	Pu-240	26.31	30.02	26.21	29.99	24.65	26.13	30.31	28.56	29.26	31.48	29.76	32.36	30.3	32.69
	Pu-241	12.26	10.84	12.29	10.95	10.55	11.26	11.00	12.37	12.82	11.14	11.78	10.74	11.76	10.76
	Pu-242	6.89	6.99	7.03	7.38	3.97	5.38	7.18	9.62	12.20	14.86	12.65	14.69	8.91	12.52
Pu isotope (μg/gU(t))*2	Pu-238	280	130	270	140	170	290	140	480	510	340	570	370	540	330
	Pu-239	5870	3670	5620	3750	6070	7600	3920	5090	4530	3180	5200	3680	6160	3870
	Pu-240	2980	2200	2850	2260	2540	3620	2400	3230	3240	2610	3770	3120	4150	3130
	Pu-241	1400	800	1340	830	1090	1570	870	1410	1430	930	1500	1040	1620	1030
	Pu-242	790	520	770	560	410	750	570	1100	1360	1240	1610	1430	1230	1210
Total Pu (μg/gU(t))		11318	7309	10850	7537	10285	13825	7901	11312	11071	8296	12652	9628	13696	9578
Nd isotope (mg/gU(t))*2	Nd-148	0.3993	0.3598	0.4386	0.3702	0.3474	0.4037	0.3858	0.6273	0.6991	0.6547	0.7023	0.6793	0.6265	0.563
Am isotope (μg/gU(t))*2	Am-241	249	142	220	153	198	270	156	196	261	151	281	239	298	240
	Am-243	148	55	148	75	72	107	63	223	329	268	383	270	265	223
Cm isotope (μg/gU(t))*2	Cm-242	0.287	0.155	0.280	0.167	0.176	0.297	0.174	2.25	2.34	1.37	0.46	0.29	0.42	0.27
	Cm-243	2.46	0.33	2.48	1.43	6.78	1.03	0.64	37.1	12.7	96.3	0.62	3.83	21.33	1.14
	Cm-244	70.8	17.8	70.0	20.4	24.0	55.6	21.6	157	234	144	241	139	153	90
%FIMA for Nd-148 Y(%)=1.7		3.60	3.27	3.94	3.36	3.15	3.63	3.49	5.54	6.14	5.79	*6.16*	*5.99*	*5.53*	*5.01*
Burnups (GWd/t) for GWd/t/%FIMA=9.6		34.6	31.4	37.8	32.3	30.2	34.8	33.5	53.2	58.9	55.6	*59.1*	*57.5*	*53.1*	*48.1*
Measurement date of TRU except Cm-242		*1995/8/1*	*1995/8/1*	1995/8/1	1995/8/1	1995/8/1	1995/8/1	1995/8/1	*1999/7/6*	*1999/7/6*	*1999/7/6*	1999/7/6	1999/7/8	1999/8/2	1999/8/4
Date of normalization of Cm-242		1995/9/30	1995/9/30	1995/9/30	1995/9/30	1995/9/30	1995/9/30	1995/9/30	1998/10/1	1998/10/1	1998/10/1	1999/10/1	1999/10/1	1999/10/1	1999/10/1

*1: distance from the shoulder of the upper-end plug of fuel rods which is about 350 mm higher than the top of an effective fuel part

*2: weight ratio to uranium at measurements

Italic-typed information and data: estimated data from the similar samples because the information or data are missing.

Table 2.2.2(3) Measured isotopic inventories and burnups of pellet-average sample for BWR 8x8-4 fuel assembly (3/3)

Sample ID		MS1
Assembly		2F2D8
Rod location		H4
Type of pellet		UO2
Enrichment (wt%)		3.4
G_2O_3 (wt%)		-
Axial height (mm)*1		1830-1850
U isotope (atomic%)	U-234	0.01
	U-235	0.23
	U-236	0.52
	U-238	99.23
U isotope (mg/gU(t))*2	U-234	0.15
	U-235	2.25
	U-236	5.20
	U-238	992.41
Pu assay		IDM
Pu isotope (atomic%)	Pu-238	4.91
	Pu-239	41.22
	Pu-240	30.76
	Pu-241	9.76
	Pu-242	13.35
Pu isotope (μg/gU(t))*2	Pu-238	582
	Pu-239	4908
	Pu-240	3678
	Pu-241	1172
	Pu-242	1610
Total Pu (μg/gU(t))		11950
Nd isotope (atomic%)	Nd-142	1.05
	Nd-143	13.31
	Nd-144	38.13
	Nd-145	14.81
	Nd-146	18.86
	Nd-148	9.19
	Nd-150	4.65
Nd isotope (mg/gU(t))*2	Nd-142	0.0766
	Nd-143	0.9774
	Nd-144	2.8196
	Nd-145	1.1028
	Nd-146	1.4141
	Nd-148	0.6985
	Nd-150	0.3582
Am isotope (μg/gU(t))*2	Am-241	408
	Am-243	316
Cm isotope (μg/gU(t))*2	Cm-242	0.004
	Cm-243	1.2
	Cm-244	205
Np isotope (μg/gU(t))*2	Np-237	890
%FIMA for Nd-148 Y(%)=1.7		6.13
Burnups (GWd/t) for GWd/t/%FIMA=9.6		58.8
Date of normalization of data except Am and Cm		2004/01/01
Date of normalization of Am and Cm		2004/02/01

*1: distance from the shoulder of the upper-end plug of fuel rods which is about 350 mm higher than the top of an effective fuel part

*2: weight ratio to uranium at measurements

Table 2.2.3(1) Measured isotopic inventories and burnups of pellet-local samples for BWR 8x8-4 fuel assemblies (1/2)

Sample ID		TU101M	TU101O	TU102M	TU102O	TU201M	TU201O	TU202M	TU202O
Assembly		2F2D1	2F2D1	2F2D1	2F2D1	2F2D2	2F2D2	2F2D2	2F2D2
Rod location		F6	F6	B3	B3	F6	F6	B3	B3
Type of pellet		UO_2	UO_2	$(U,Gd)O_2$	$(U,Gd)O_2$	UO_2	UO_2	$(U,Gd)O_2$	$(U,Gd)O_2$
Enrichment (wt%)		4.5	4.5	3.4	3.4	4.5	4.5	3.4	3.4
G_2O_3 (wt%)		-	-	4.5	4.5	-	-	4.5	4.5
Axial height (mm)*1		2247-2249	2247-2249	2214-2216	2214-2216	1943-1945	1943-1945	2226-2228	2226-2228
U isotope (atomic%)	U-234	0.075	0.085	0.040	0.040	0.101	0.138	0.037	0.037
	U-235	3.28	3.46	2.72	2.39	2.22	2.31	1.55	1.53
	U-236	0.38	0.44	0.26	0.28	0.75	0.78	0.44	0.50
	U-238	96.27	96.02	96.98	97.29	96.93	96.77	97.97	97.93
U isotope (mg/gU(t))*2	U-234	0.74	0.84	0.39	0.39	0.99	1.36	0.36	0.36
	U-235	32.40	34.18	26.87	23.61	21.93	22.82	15.31	15.11
	U-236	3.77	4.36	2.58	2.78	7.44	7.74	4.36	4.96
	U-238	963.09	960.62	970.16	973.22	969.64	968.09	979.96	979.57
Pu assay		α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum
Pu isotope (atomic%)	Pu-238	-	-	-	-	-	-	-	-
	Pu-239	78.28	78.60	80.43	76.77	66.16	65.57	63.56	63.98
	Pu-240	15.06	15.20	13.63	16.09	21.96	21.84	23.81	23.08
	Pu-241	5.59	5.46	5.32	5.78	9.06	9.81	9.71	9.85
	Pu-242	1.07	0.74	0.62	1.36	2.82	2.78	2.93	3.10
Pu isotope (μg/gU(t))*2	Pu-238								
	Pu-239	5090	5912	4652	4531	6636	7329	6405	6798
	Pu-240	983	1148	792	954	2212	2451	2409	2463
	Pu-241	367	414	310	344	916	1106	987	1055
	Pu-242	70	56	36	81	286	315	299	334
Total Pu (μg/gU(t))		6510	7530	5790	5910	10050	11200	10100	10650
Nd isotope (mg/gU(t))*2	Nd-148	0.1474	0.1637	0.0990	0.1040	0.2921	0.3102	0.2515	0.2902
Am isotope (μg/gU(t))*2	Am-241	78	88	45	55	162	155	122	135
	Am-243	-	-	-	-	-	-	-	-
Cm isotope (μg/gU(t))*2	Cm-242	-	-	-	-	-	-	-	-
	Cm-243	-	-	-	-	-	-	-	-
	Cm-244	0.2	0.4	0.1	0.1	4.1	5.0	6.1	7.5
%FIMA for Nd-148 Y(%)=1.7		*1.37*	*1.51*	*0.92*	*0.97*	*2.66*	*2.82*	*2.30*	*2.65*
Burnups (GWd/t) for GWd/t/%FIMA=9.6		*13.20*	*14.50*	*8.80*	*9.30*	*25.50*	*27.10*	*22.10*	*25.40*
Measurement date of TRU except Cm-242		*1994/9/5*	*1994/9/5*	*1994/8/31*	*1994/8/31*	*1994/10/31*	*1994/10/31*	*1994/10/31*	*1994/10/31*
Date of normalization of Cm-242									

*1: distance from the shoulder of the upper-end plug of fuel rods which is about 350 mm higher than the top of an effective fuel part

*2: weight ratio to uranium at measurements

Italic-typed information and data: estimated data from the similar samples because the information or data are missing.

Table 2.2.3(2) Measured isotopic inventories and burnups of pellet-local samples for BWR 8x8-4 fuel assemblies (2/2)

Sample ID		TU303M	TU303O	TU305M	TU305O	TU310M	TU310O	TU5041/C	TU5042/M	TU5043/O	TU507/C	TU508/M	TU509/O
Assembly		2F2D3	2F2D3	2F2D3	2F2D3	2F2D3	2F2D3	2F2D8	2F2D8	2F2D8	2F2D8	2F2D8	2F2D8
Rod location		H5	H5	A4	A4	B3	B3	H5	H5	H5	B3	B3	B3
Type of pellet		UO_2	UO_2	UO_2	UO_2	$(U,Gd)O_2$	$(U,Gd)O_2$	UO_2	UO_2	UO_2	$(U,Gd)O_2$	$(U,Gd)O_2$	$(U,Gd)O_2$
Enrichment (wt%)		3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4
G_2O_3 (wt%)		-	-	-	-	4.5	4.5	-	-	-	4.5	4.5	4.5
Axial height (mm)*1		2902-2904	2902-2904	3132-3133	3119-3121	3083-3085	3083-3085	1622-1627	1622-1627	1622-1627	1078-1098	1078-1098	1078-1098
U isotope (atomic%)	U-234	0.016	0.018	0.017	0.017	0.022	0.020	0.014	0.014	0.013	0.02	0.02	0.02
	U-235	0.43	0.46	0.52	0.48	0.80	0.65	0.28	0.29	0.21	0.58	0.52	0.44
	U-236	0.50	0.50	0.50	0.50	0.48	0.49	0.55	0.54	0.52	0.57	0.54	0.54
	U-238	99.06	99.02	98.96	99.00	98.71	98.84	99.15	99.16	99.26	98.83	98.92	99.00
U isotope (mg/gU(t))*2	U-234	0.16	0.18	0.17	0.17	0.22	0.20	0.14	0.14	0.13	0.21	0.16	0.16
	U-235	4.25	4.54	5.14	4.74	7.90	6.42	2.81	2.90	2.06	5.70	5.09	4.36
	U-236	4.96	4.96	4.96	4.96	4.76	4.86	5.44	5.31	5.13	5.63	5.40	5.32
	U-238	990.64	990.32	989.74	990.13	987.12	988.53	991.61	991.65	992.68	988.46	989.36	990.16
Pu assay		α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum	α spectrum
Pu isotope (atomic%)	Pu-238	2.84	2.97	2.60	2.67	2.15	1.71	4.73	4.70	5.18	3.81	3.89	4.05
	Pu-239	45.82	45.82	46.79	46.25	50.38	49.74	40.50	40.72	40.67	45.12	44.35	44.91
	Pu-240	29.50	31.02	31.60	30.90	30.55	27.27	32.95	33.00	28.72	32.93	32.33	29.82
	Pu-241	11.86	11.24	10.87	11.18	10.55	12.67	11.34	11.35	12.47	10.64	10.99	11.84
	Pu-242	9.98	9.17	8.14	9.00	6.37	8.61	10.48	10.23	12.95	7.51	8.44	9.39
Pu isotope (μg/gU(t))*2	Pu-238	231	236	202	205	157	205	450	550	610	510	460	510
	Pu-239	3748	3634	3654	3568	3691	5986	3890	4770	4790	6020	5290	5710
	Pu-240	2423	2483	2478	2394	2247	3295	3180	3880	3400	4410	3870	3810
	Pu-241	978	904	856	870	780	1538	1100	1340	1480	1430	1320	1520
	Pu-242	827	740	644	703	472	1050	1020	1210	1550	1010	1020	1210
Total Pu (μg/gU(t))		8207	7997	7834	7740	7347	12074	9640	11742	11831	13387	11959	12762
Nd isotope (mg/gU(t))*2	Nd-148	0.3208	0.4269	0.3969	0.3947	0.3227	0.4200	0.7125	0.6124	0.7585	0.6126	0.543	0.6314
Am isotope (μg/gU(t))*2	Am-241	120	61	116	127	104	268	235	197	340	273	280	377
	Am-243	108	51	84	98	55	171	192	194	348	235	234	261
Cm isotope (μg/gU(t))*2	Cm-242	0.230	0.204	0.187	0.174	0.154	0.328	1.76	1.94	2.73	0.38	0.37	0.44
	Cm-243	0.80	3.88	1.72	12.79	0.32	3.04	99.1	132	18.5	15.81	6.56	5.36
	Cm-244	54.9	45.9	35.5	34.1	21.3	52.1	167	185	286	115	116	159
%FIMA for Nd-148 Y(%)=1.7		2.92	3.85	3.59	3.57	2.94	3.78	6.26	5.42	6.63	*5.41*	*4.83*	*5.57*
Burnups (GWd/t) for GWd/t/%FIMA=9.6		28.0	37.0	34.5	34.3	28.2	36.3	60.1	52.0	63.6	*51.9*	*46.4*	*53.5*
Measurement date of TRU except Cm-242		*1995/8/1*	*1995/8/1*	*1995/8/1*	*1995/8/1*	*1995/8/1*	*1995/8/1*	*1999/7/6*	*1999/7/6*	*1999/7/6*	*1999/7/6*	*1999/7/6*	*1999/7/6*
Date of normalization of Cm-242		1995/9/30	1995/9/30	1995/9/30	1995/9/30	1995/9/30	1995/9/30	1998/10/1	1998/10/1	1998/10/1	1999/10/1	1999/10/1	1999/10/1

*1: distance from the shoulder of the upper-end plug of fuel rods which is about 350 mm higher than the top of an effective fuel part
*2: weight ratio to uranium at measurements
Italic-typed information and data: estimated data from the similar samples because the information or data are missing.

2.3 Analysis of Burnups and Isotopic Inventories

Theoretical analysis has been performed mainly for the pellet-average samples and details of the analysis procedures are shown in Reference 2. In the analysis of the burnups and isotopic inventories of the pellet-average samples, an interpolation analysis[2)] was mainly adopted taking into account the irradiation history of the samples. The burnup calculations with an infinite assembly model were carried out using a collision probability module (PIJ) of 107 neuron energy groups of a general purpose nuclear calculation code system SRAC,[10)] and a continuous energy Monte Carlo burnup calculation code MVP-BURN[11)] based on the nuclear data library, JENDL-3.2.[12)] The geometrical dimensions and atomic number densities of the assembly were taken from the design base specifications of the fuel vender. The temperatures of the pellets were postulated to be 627 ℃. The temperatures of the fuel cladding and the moderator were set to be 286 ℃.

Table 2.3.1 shows the results of the burnups and inventories in mg/gU or μg/gU for the pellet-average samples by the interpolation analysis of SRAC and the comparison with the measurements in the calculated / measured result (C/E).[2)] **Table 2.3.2** also shows the calculated results and comparisons in C/E for the isotopic ratios in atomic % of U and Pu isotopes.[2)]

Figure 2.3.1 schematically shows the calculated burnups against the measurements.[2)] Most of calculated values are in the range of ±10%.

Figure 2.3.2 shows a tend of the C/Es of the inventories of ^{235}U and all Pu isotopes (totPu) against the C/E of the burnups. Biases of around +5% for ^{235}U and -10% for totPu seem to exist between the calculations and measurements at the C/Es of burnups equal to 1.0. The observations of Figure 2.3.2 and other information on C/Es in similar studies on the inventories of pellet-average samples taken from 8x8-2 assembly[13)] implies that the measured inventories for part of the samples of the BWR 8x8-4 fuel assemblies have systematic errors other than the reported errors.

The results of MVP-BURN are in **Table 2.3.3**, and the trend of C/Es is almost same as those of SRAC, which indicate that the trend of C/E is not related to the modeling of burnup calculations.

In order to review the modeling uncertainty in the interpolation analysis, further analyses were performed by tracking the node power and the in-channel void fraction using SRAC calculations for the samples TU505 and TU506 of the five-cycle irradiation assembly, 2F2D8. **Table 2.3.4** shows the results with the C/Es of the interpolation analysis shown in Table 2.3.1. The differences in the C/Es between the tracking and interpolation analysis are discussed in Reference 2.

Table 2.3.1 Analysis results of interpolation analysis with SRAC and comparison with measurements (C/E) on burnups and isotopic inventories for BWR 8x8-4 fuel assemblies[2)]

(Unit: mg/gU(t) except for *µg/gU(t))

Assembly	Sample ID, Rod Node	Pellet burnup GWd/t	U234	U235	U236	U238	Pu238	Pu239	Pu240	Pu241	Pu242	* totPu	* Am241	* Am243	* Cm242	* Cm244
		14.28	0.35	31.49	2.69	965.47	0.01	4.44	0.74	0.26	0.03	5488	70.5	2.06	0.0006	0.2
	TU101 F6	14.0	0.34	33.94	2.62	963.10	0.02	4.86	0.73	0.28	0.03	5916	59.6	1.81	0.0008	0.1
	22	1.02	1.02	0.93	1.03	1.00	0.98	0.91	1.02	0.93	1.01	0.93	1.18	1.14	0.72	1.69
		15.41	0.36	29.85	2.75	967.04	0.01	3.47	0.72	0.21	0.03	4432	56.0	1.37	0.0005	0.081
	TU102 F6	18.2	0.35	28.31	3.03	968.31	0.02	4.14	0.90	0.31	0.04	5410	70.1	1.64	0.0007	0.093
	5	0.85	1.02	1.05	0.91	1.00	0.60	0.84	0.80	0.67	0.68	0.82	0.80	0.83	0.67	0.87
		10.28	0.26	25.04	1.86	972.85	0.01	4.22	0.71	0.24	0.03	5203	65.2	1.71	0.0005	0.130
	TU103 B3	10.0	0.25	25.49	1.80	972.46	0.01	4.81	0.73	0.27	0.03	5847	58.5	1.54	0.0006	0.058
2F2D1	22	1.03	1.02	0.98	1.03	1.00	0.90	0.88	0.98	0.87	1.05	0.89	1.11	1.11	0.83	2.25
		10.48	0.25	25.17	1.89	972.69	0.01	4.56	0.78	0.27	0.03	5651	74	2.07	0.0006	0.168
	TU104 B3	9.4	0.26	23.67	2.03	974.04	0.02	4.61	0.83	0.33	0.04	5817	71	1.87	0.0007	0.158
	18	1.11	0.97	1.06	0.93	1.00	0.77	0.99	0.95	0.82	0.76	0.97	1.05	1.11	0.92	1.06
		9.90	0.26	24.98	1.73	973.02	0.01	3.26	0.65	0.17	0.02	4104	47.4	0.92	0.0004	0.048
	TU105 B3	12.3	0.26	24.44	2.04	973.25	0.01	4.64	0.96	0.38	0.04	6031	69.7	1.54	0.0009	0.094
	5	0.80	1.01	1.02	0.85	1.00	0.51	0.70	0.67	0.46	0.53	0.68	0.68	0.60	0.39	0.51
		15.42	0.34	30.67	2.86	966.13	0.02	4.76	0.84	0.31	0.04	5959	86	2.87	0.0008	0.259
	TU106 F6	16.1	0.32	30.04	3.14	966.51	0.03	5.77	1.05	0.47	0.06	7373	102	1.82	0.0012	0.372
	18	0.96	1.07	1.02	0.91	1.00	0.67	0.82	0.80	0.66	0.65	0.81	0.85	1.58	0.67	0.70
		27.33	0.30	21.80	4.45	973.45	0.08	5.97	1.71	0.77	0.17	8705	160	24.1	0.045	4.39
	TU201 F6	29.1	0.28	21.39	4.68	973.64	0.11	6.70	1.93	1.04	0.18	9947	156	12.9	0.099	5.65
	21	0.94	1.07	1.02	0.95	1.00	0.74	0.89	0.89	0.75	0.95	0.88	1.03	1.87	0.45	0.78
		27.84	0.32	19.56	4.44	975.69	0.05	4.02	1.51	0.55	0.16	6285	112	14.2	0.032	1.72
	TU202 F6	32.9	0.28	16.33	5.04	978.35	0.09	4.67	1.98	0.85	0.28	7858	143	19.2	0.067	5.41
	4	0.85	1.13	1.20	0.88	1.00	0.52	0.86	0.77	0.65	0.57	0.80	0.79	0.74	0.48	0.32
		22.07	0.22	16.96	3.31	979.51	0.06	5.59	1.70	0.74	0.17	8263	146	23.4	0.043	4.22
2F2D2	TU203 B3	24.5	0.20	16.04	3.57	980.19	0.09	6.73	2.07	1.04	0.23	10162	163	20.3	0.099	5.89
	21	0.90	1.09	1.06	0.93	1.00	0.68	0.83	0.82	0.71	0.76	0.81	0.89	1.15	0.43	0.72
		23.55	0.21	16.37	3.45	979.96	0.07	5.69	1.87	0.81	0.21	8646	170	29.4	0.053	5.55
	TU204 B3	23.5	0.21	14.83	3.54	981.42	0.10	6.52	2.15	1.02	0.27	10066	179	35.3	0.078	8.09
	17	1.00	1.02	1.10	0.97	1.00	0.72	0.87	0.87	0.79	0.77	0.86	0.95	0.83	0.68	0.69
		21.26	0.23	16.03	3.20	980.54	0.03	3.78	1.46	0.51	0.15	5928	103	12.6	0.029	1.44
	TU205 B3	22.8	0.21	12.75	3.68	983.35	0.07	4.32	1.86	0.73	0.26	7233	129	23.6	0.052	3.58
	4	0.93	1.11	1.26	0.87	1.00	0.51	0.88	0.78	0.69	0.57	0.82	0.80	0.53	0.55	0.40
		43.25	0.17	6.59	4.67	988.57	0.27	6.10	3.06	1.44	0.79	11650	278	177	0.26	64.1
	TU304 A4	37.8	0.17	6.54	4.77	988.52	0.27	5.62	2.85	1.34	0.77	10850	221	155	0.28	68.9
	19	1.14	0.97	1.01	0.98	1.00	0.98	1.09	1.07	1.07	1.02	1.07	1.26	1.14	0.93	0.93
		32.63	0.20	7.90	4.25	987.65	0.09	3.70	2.09	0.75	0.40	7020	136	49	0.106	8.9
	TU306 A4	32.3	0.19	6.22	4.65	988.93	0.14	3.75	2.26	0.83	0.56	7537	153	82	0.167	19.7
	3	1.01	1.06	1.27	0.91	1.00	0.63	0.99	0.92	0.90	0.72	0.93	0.89	0.60	0.63	0.45
		33.51	0.19	11.00	4.27	984.54	0.16	6.09	2.59	1.16	0.44	10440	222	86	0.162	25.1
2F2D3	TU308 B3	30.2	0.22	11.13	4.40	984.26	0.17	6.07	2.54	1.09	0.41	10285	199	76	0.176	20.3
	21	1.11	0.85	0.99	0.97	1.00	0.95	1.00	1.02	1.06	1.07	1.02	1.11	1.13	0.92	1.24
		36.02	0.18	10.18	4.44	985.21	0.19	6.09	2.79	1.25	0.53	10847	242	107	0.194	33.4
	TU309 B3	34.8	0.21	9.32	4.75	985.72	0.29	7.60	3.62	1.57	0.75	13825	270	141	0.296	55.1
	18	1.04	0.84	1.09	0.93	1.00	0.68	0.80	0.77	0.80	0.70	0.78	0.90	0.76	0.66	0.61
		31.22	0.20	9.81	4.15	985.84	0.09	3.85	2.11	0.77	0.39	7197	141	49	0.109	9.1
	TU311 B3	33.5	0.22	7.15	4.78	987.85	0.14	3.92	2.40	0.87	0.57	7901	157	85	0.174	21.3
	4	0.93	0.91	1.37	0.87	1.00	0.62	0.98	0.88	0.88	0.67	0.91	0.90	0.57	0.63	0.43
		58.78	0.13	2.44	4.93	992.50	0.46	5.44	3.58	1.55	1.47	12495	274	389	0.52	205
	TU505 A4	59.1	0.15	2.45	5.24	992.17	0.57	5.20	3.77	1.50	1.61	12652	281	383	0.46	241
	15	0.99	0.88	0.99	0.94	1.00	0.80	1.05	0.95	1.03	0.91	0.99	0.98	1.02	1.14	0.85
		53.71	0.14	1.78	4.90	993.18	0.30	3.84	3.04	1.10	1.34	9606	184	292	0.39	122
	TU506 A4	57.5	0.14	1.62	5.04	993.20	0.37	3.68	3.12	1.04	1.43	9628	239	270	0.29	139
2F2D8	6	0.93	1.00	1.10	0.97	1.00	0.82	1.04	0.97	1.06	0.94	1.00	0.77	1.08	1.33	0.88
		50.20	0.15	5.34	5.03	989.48	0.39	6.10	3.59	1.56	1.03	12664	293	261	0.46	121
	TU510 B3	53.1	0.15	4.83	5.47	989.55	0.54	6.16	4.15	1.62	1.23	13696	298	265	0.42	153
	20	0.95	0.99	1.11	0.92	1.00	0.72	0.99	0.86	0.96	0.83	0.92	0.98	0.98	1.10	0.79
		47.49	0.16	3.44	4.96	991.45	0.24	3.79	2.89	1.03	1.03	8969	179	202	0.34	70
	TU511 B3	48.1	0.16	3.13	5.44	991.27	0.33	3.88	3.13	1.04	1.21	9578	240	223	0.27	90
	5	0.99	0.98	1.10	0.91	1.00	0.72	0.98	0.92	0.99	0.85	0.94	0.75	0.91	1.24	0.78
Average of C/E		0.97	1.00	1.09	0.93	1.00	0.72	0.92	0.89	0.84	0.80	0.89	0.93	0.98	0.77	0.85
S.D. of C/E		0.09	0.08	0.11	0.05	0.00	0.14	0.10	0.10	0.16	0.16	0.09	0.15	0.32	0.27	0.44

Upper: Calculation
Middle: Measurement
Bottom: C/E

Table 2.3.2 Analysis results of interpolation analysis of SRAC and comparison with measurements (C/E) of isotopic ratios for BWR 8x8-4 fuel assemblies2)

(Unit: at%)

Assembly	Sample ID, Rod Node	U234	U235	U236	U238	Pu238	Pu239	Pu240	Pu241	Pu242
		0.04	3.19	0.27	96.51	0.27	80.96	13.48	4.76	0.53
	TU101 F6	0.03	3.44	0.26	96.27	0.26	82.26	12.22	4.77	0.49
	22	1.04	0.93	1.04	1.00	1.03	0.98	1.10	1.00	1.08
		0.04	3.02	0.28	96.66	0.23	78.30	16.16	4.69	0.63
	TU102 F6	0.04	2.87	0.31	96.79	0.31	76.62	16.56	5.77	0.75
	5	1.01	1.05	0.90	1.00	0.75	1.02	0.98	0.81	0.83
		0.03	2.53	0.19	97.25	0.21	81.14	13.64	4.51	0.50
	TU103 B3	0.03	2.58	0.18	97.21	0.21	82.36	12.40	4.61	0.42
	22	1.04	0.98	1.04	1.00	0.99	0.99	1.10	0.98	1.18
		0.03	2.55	0.19	97.24	0.23	80.76	13.77	4.70	0.53
2F2D1	TU104 B3	0.03	2.40	0.21	97.37	0.29	79.32	14.15	5.57	0.68
	18	0.99	1.06	0.91	1.00	0.80	1.02	0.97	0.84	0.78
		0.03	2.53	0.17	97.27	0.16	79.46	15.69	4.17	0.52
	TU105 B3	0.03	2.48	0.21	97.29	0.22	77.01	15.89	6.22	0.66
	5	0.99	1.02	0.83	1.00	0.73	1.03	0.99	0.67	0.79
		0.03	3.10	0.29	96.57	0.31	79.89	14.03	5.14	0.62
	TU106 F6	0.03	3.04	0.32	96.61	0.39	78.38	14.25	6.20	0.79
	18	1.09	1.02	0.90	1.00	0.81	1.02	0.98	0.83	0.79
		0.03	2.21	0.45	97.31	0.91	68.67	19.65	8.80	1.97
	TU201 F6	0.03	2.17	0.47	97.33	1.07	67.45	19.33	10.34	1.82
	21	1.05	1.02	0.95	1.00	0.85	1.02	1.02	0.85	1.08
		0.03	1.98	0.45	97.54	0.75	64.08	24.02	8.66	2.48
	TU202 F6	0.03	1.65	0.51	97.81	1.15	59.58	25.10	10.69	3.48
	4	1.11	1.20	0.88	1.00	0.65	1.08	0.96	0.81	0.71
		0.02	1.72	0.33	97.93	0.75	67.80	20.50	8.90	2.06
	TU203 B3	0.02	1.62	0.36	98.00	0.90	66.41	20.33	10.17	2.19
2F2D2	21	1.11	1.06	0.93	1.00	0.83	1.02	1.01	0.88	0.94
		0.02	1.66	0.35	97.97	0.85	65.91	21.54	9.31	2.39
	TU204 B3	0.02	1.50	0.36	98.12	1.01	64.91	21.33	10.09	2.66
	17	1.03	1.11	0.97	1.00	0.84	1.02	1.01	0.92	0.90
		0.02	1.62	0.32	98.03	0.57	63.91	24.53	8.53	2.46
	TU205 B3	0.02	1.29	0.37	98.32	0.91	59.85	25.63	10.07	3.53
	4	1.08	1.26	0.87	1.00	0.63	1.07	0.96	0.85	0.70
		0.02	0.67	0.47	98.84	2.32	52.49	26.24	12.26	6.68
	TU304 A4	0.02	0.66	0.48	98.84	2.54	51.93	26.22	12.30	7.01
	19	0.99	1.01	0.98	1.00	0.91	1.01	1.00	1.00	0.95
		0.02	0.80	0.43	98.75	1.24	52.83	29.66	10.60	5.66
	TU306 A4	0.02	0.63	0.47	98.88	1.84	49.83	29.99	10.95	7.39
	3	1.03	1.27	0.91	1.00	0.68	1.06	0.99	0.97	0.77
		0.02	1.11	0.43	98.44	1.58	58.43	24.77	11.05	4.17
2F2D3	TU308 B3	0.02	1.13	0.44	98.41	1.69	59.15	24.64	10.54	3.98
	21	0.86	0.99	0.98	1.00	0.94	0.99	1.01	1.05	1.05
		0.02	1.03	0.45	98.50	1.79	56.25	25.69	11.43	4.83
	TU309 B3	0.02	0.95	0.48	98.55	2.09	55.14	26.13	11.27	5.37
	18	0.83	1.08	0.93	1.00	0.86	1.02	0.98	1.01	0.90
		0.02	0.99	0.42	98.57	1.21	53.58	29.26	10.64	5.31
	TU311 B3	0.02	0.73	0.48	98.77	1.80	49.70	30.31	11.00	7.19
	4	0.90	1.36	0.87	1.00	0.67	1.08	0.97	0.97	0.74
		0.01	0.25	0.50	99.24	3.69	43.73	28.61	12.32	11.66
	TU505 A4	0.02	0.25	0.53	99.21	4.56	41.25	29.76	11.78	12.65
	15	0.90	0.99	0.94	1.00	0.81	1.06	0.96	1.05	0.92
		0.01	0.18	0.49	99.31	3.15	40.10	31.59	11.36	13.79
	TU506 A4	0.02	0.16	0.51	99.31	3.83	38.38	32.36	10.74	14.69
2F2D8	6	0.95	1.13	0.97	1.00	0.82	1.04	0.98	1.06	0.94
		0.02	0.54	0.51	98.94	3.07	48.34	28.33	12.24	8.03
	TU510 B3	0.02	0.49	0.55	98.94	3.95	45.09	30.30	11.76	8.91
	20	1.01	1.10	0.92	1.00	0.78	1.07	0.93	1.04	0.90
		0.02	0.35	0.50	99.14	2.64	42.39	32.23	11.42	11.33
	TU511 B3	0.02	0.32	0.55	99.12	3.43	40.60	32.69	10.76	12.52
	5	1.00	1.09	0.91	1.00	0.77	1.04	0.99	1.06	0.91
Average of C/E		1.00	1.09	0.93	1.00	0.81	1.03	0.99	0.93	0.89
S.D. of C/E		0.08	0.11	0.05	0.00	0.11	0.03	0.04	0.11	0.13

Upper: Calculation
Middle: Measurement
Bottom: C/E

Table 2.3.3 Analysis results of interpolation analysis with MVP-BURN and comparison with measurements (C/E) of burnups and isotopic inventories for BWR 8×8-4 fuel assemblies[2)]

(Unit: mg/gU(t) except for *μg/gU(t))

Assembly	Sample ID, Rod Node	Pellet burnup GWd/t	U234	U235	U236	U238	Pu238	Pu239	Pu240	Pu241	Pu242	* totPu	* Am241	* Am243	* Cm242	* Cm244
		14.32	0.35	31.41	2.71	965.53	0.02	4.41	0.74	0.27	0.03	5471	72.3	2.29	0.0006	0.2
	TU101 F6	14.00	0.34	33.94	2.62	963.10	0.02	4.86	0.73	0.28	0.03	5916	59.6	1.81	0.0008	0.1
	22	1.02	1.02	0.93	1.03	1.00	1.00	0.91	1.02	0.96	1.06	0.92	1.21	1.27	0.77	2.02
		15.20	0.36	29.97	2.73	966.94	0.01	3.38	0.70	0.21	0.03	4324	56.0	1.43	0.0005	0.087
	TU102 F6	18.20	0.35	28.31	3.03	968.31	0.02	4.14	0.90	0.31	0.04	5410	70.1	1.64	0.0007	0.093
	5	0.84	1.02	1.06	0.90	1.00	0.59	0.82	0.77	0.67	0.69	0.80	0.80	0.87	0.69	0.93
		10.59	0.25	24.70	1.92	973.12	0.01	4.30	0.74	0.25	0.03	5329	68.8	1.99	0.0005	0.165
	TU103 B3	10.00	0.25	25.49	1.80	972.46	0.01	4.81	0.73	0.27	0.03	5847	58.5	1.54	0.0006	0.058
2F2D1	22	1.06	1.02	0.97	1.07	1.00	0.94	0.89	1.01	0.92	1.13	0.91	1.18	1.29	0.91	2.84
		10.84	0.25	24.79	1.96	973.00	0.01	4.61	0.81	0.28	0.03	5753	78.6	2.48	0.0007	0.214
	TU104 B3	9.40	0.26	23.67	2.03	974.04	0.02	4.61	0.83	0.33	0.04	5817	71.0	1.87	0.0007	0.158
	18	1.15	0.97	1.05	0.96	1.00	0.80	1.00	0.98	0.87	0.84	0.99	1.11	1.33	1.02	1.35
		10.15	0.26	24.66	1.78	973.29	0.01	3.30	0.66	0.18	0.02	4167	50.2	1.06	0.0004	0.058
	TU105 B3	12.30	0.26	24.44	2.04	973.25	0.01	4.64	0.96	0.38	0.04	6031	69.7	1.54	0.0009	0.094
	5	0.83	1.01	1.01	0.87	1.00	0.51	0.71	0.69	0.49	0.57	0.69	0.72	0.69	0.43	0.62
		15.42	0.34	30.60	2.88	966.17	0.02	4.70	0.84	0.31	0.04	5908	87	3.16	0.0008	0.302
	TU106 F6	16.10	0.32	30.04	3.14	966.51	0.03	5.77	1.05	0.47	0.06	7373	102	1.82	0.0012	0.372
	18	0.96	1.07	1.02	0.92	1.00	0.68	0.81	0.80	0.67	0.68	0.80	0.86	1.74	0.71	0.81
		27.40	0.30	21.64	4.51	973.55	0.08	5.87	1.68	0.79	0.18	8599	163	26.5	0.047	5.16
	TU201 F6	29.10	0.28	21.39	4.68	973.64	0.11	6.70	1.93	1.04	0.18	9947	156	12.9	0.099	5.65
	21	0.94	1.06	1.01	0.96	1.00	0.73	0.88	0.87	0.77	0.98	0.86	1.04	2.06	0.47	0.91
		27.51	0.32	19.74	4.42	975.52	0.05	3.95	1.45	0.55	0.16	6161	112	14.8	0.033	1.88
	TU202 F6	32.90	0.28	16.33	5.04	978.35	0.09	4.67	1.98	0.85	0.28	7858	143	19.2	0.067	5.41
	4	0.84	1.13	1.21	0.88	1.00	0.51	0.85	0.74	0.65	0.57	0.78	0.79	0.77	0.49	0.35
		22.47	0.22	16.62	3.37	979.79	0.06	5.68	1.73	0.77	0.18	8419	150	26.0	0.045	5.06
2F2D2	TU203 B3	24.50	0.20	16.04	3.57	980.19	0.09	6.73	2.07	1.04	0.23	10162	163	20.3	0.099	5.89
	21	0.92	1.08	1.04	0.95	1.00	0.70	0.84	0.84	0.74	0.79	0.83	0.92	1.28	0.46	0.86
		23.96	0.21	16.04	3.52	980.22	0.08	5.80	1.88	0.85	0.22	8828	177	32.9	0.057	6.55
	TU204 B3	23.50	0.21	14.83	3.54	981.42	0.10	6.52	2.15	1.02	0.27	10066	179	35.3	0.078	8.09
	17	1.02	1.01	1.08	0.99	1.00	0.74	0.89	0.87	0.84	0.81	0.88	0.99	0.93	0.73	0.81
		21.47	0.23	15.81	3.24	980.71	0.03	3.85	1.46	0.53	0.15	6026	108	13.6	0.031	1.63
	TU205 B3	22.80	0.21	12.75	3.68	983.35	0.07	4.32	1.86	0.73	0.26	7233	129	23.6	0.052	3.58
	4	0.94	1.10	1.24	0.88	1.00	0.52	0.89	0.78	0.73	0.60	0.83	0.83	0.58	0.59	0.46
		43.05	0.16	6.69	4.70	988.44	0.27	6.18	3.01	1.46	0.77	11693	278	181	0.267	68.9
	TU304 A4	37.80	0.17	6.54	4.77	988.52	0.27	5.62	2.85	1.34	0.77	10850	221	155	0.280	68.9
	19	1.14	0.95	1.02	0.98	1.00	0.97	1.10	1.06	1.09	1.01	1.08	1.26	1.17	0.95	1.00
		32.54	0.20	7.92	4.26	987.61	0.09	3.71	2.04	0.76	0.40	6990	136	51	0.106	9.5
	TU306 A4	32.30	0.19	6.22	4.65	988.93	0.14	3.75	2.26	0.83	0.56	7537	153	82	0.167	19.7
	3	1.01	1.06	1.27	0.92	1.00	0.62	0.99	0.90	0.91	0.71	0.93	0.89	0.62	0.64	0.48
		33.97	0.18	10.68	4.33	984.80	0.17	6.17	2.61	1.20	0.45	10597	226	94	0.170	29.0
2F2D3	TU308 B3	30.20	0.22	11.13	4.40	984.26	0.17	6.07	2.54	1.09	0.41	10285	199	76	0.176	20.3
	21	1.12	0.84	0.96	0.98	1.00	0.97	1.02	1.03	1.10	1.10	1.03	1.14	1.24	0.97	1.43
		36.50	0.18	9.90	4.50	985.42	0.20	6.20	2.80	1.28	0.55	11026	246	116	0.205	38.3
	TU309 B3	34.80	0.21	9.32	4.75	985.72	0.29	7.60	3.62	1.57	0.75	13825	270	141	0.296	55.1
	18	1.05	0.83	1.06	0.95	1.00	0.69	0.81	0.77	0.82	0.73	0.80	0.91	0.82	0.69	0.70
		31.44	0.20	9.64	4.18	985.97	0.09	3.92	2.10	0.80	0.40	7298	145	52	0.114	10.2
	TU311 B3	33.50	0.22	7.15	4.78	987.85	0.14	3.92	2.40	0.87	0.57	7901	157	85	0.174	21.3
	4	0.94	0.91	1.35	0.88	1.00	0.62	1.00	0.87	0.92	0.70	0.92	0.92	0.62	0.66	0.48
		58.56	0.13	2.53	4.98	992.35	0.45	5.54	3.51	1.59	1.44	12532	277	393	0.54	217
	TU505 A4	59.10	0.15	2.45	5.24	992.17	0.57	5.20	3.77	1.50	1.61	12652	281	383	0.46	241
	15	0.99	0.86	1.03	0.95	1.00	0.80	1.07	0.93	1.06	0.89	0.99	0.99	1.03	1.17	0.90
		53.52	0.14	1.81	4.94	993.10	0.29	3.84	2.97	1.10	1.32	9529	183	295	0.39	128
	TU506 A4	57.50	0.14	1.62	5.04	993.20	0.37	3.68	3.12	1.04	1.43	9628	239	270	0.29	139
2F2D8	6	0.93	0.99	1.12	0.98	1.00	0.80	1.04	0.95	1.06	0.92	0.99	0.76	1.09	1.35	0.92
		50.74	0.14	5.22	5.07	989.56	0.39	6.27	3.59	1.62	1.04	12919	299	276	0.48	136
	TU510 B3	53.10	0.15	4.83	5.47	989.55	0.54	6.16	4.15	1.62	1.23	13696	298	265	0.42	153
	20	0.96	0.96	1.08	0.93	1.00	0.73	1.02	0.86	1.00	0.85	0.94	1.00	1.04	1.15	0.89
		48.11	0.15	3.29	5.00	991.56	0.24	3.87	2.89	1.06	1.06	9113	182	217	0.35	80
	TU511 B3	48.10	0.16	3.13	5.44	991.27	0.33	3.88	3.13	1.04	1.21	9578	240	223	0.27	90
	5	1.00	0.97	1.05	0.92	1.00	0.73	1.00	0.92	1.02	0.87	0.95	0.76	0.97	1.30	0.89
Average of C/E		0.98	0.99	1.08	0.94	1.00	0.73	0.93	0.88	0.86	0.82	0.90	0.95	1.07	0.81	0.98
S.D. of C/E		0.09	0.08	0.11	0.05	0.00	0.15	0.10	0.10	0.17	0.17	0.09	0.16	0.37	0.27	0.56

Upper: Calculation
Middle: Measurement
Bottom: C/E

Table 2.3.4 Analysis results of tracking calculations with SRAC and comparison with measurements (C/E) for 2F1D8[2)]

Sample ID		TU505		TU506	
Analysis code		SRAC	C/E	SRAC	C/E
Burnup(GWd/t)		53.78	1.00 (0.99)	58.98	0.94 (0.93)
U isotopic ratio (at%)	U234	0.014	0.93 (0.90)	0.015	0.98 (0.95)
	U235	0.25	1.01 (0.99)	0.18	1.15 (1.13)
	U236	0.50	0.95 (0.94)	0.50	0.98 (0.97)
	U238	99.23	1.00 (1.00)	99.30	1.00 (1.00)
U isotopic weight (mg/gU)	U234	0.14	0.92 (0.88)	0.14	1.03 (1.00)
	U235	2.50	1.02 (0.99)	1.82	1.13 (1.10)
	U236	4.99	0.95 (0.94)	4.94	0.98 (0.97)
	U238	992.37	1.00 (1.00)	993.10	1.00 (1.00)
Pu isotopic ratio (at%)	Pu238	4.13	0.91 (0.81)	3.59	0.94 0.82)
	Pu239	41.35	1.00 (1.06)	38.80	1.01 (1.04)
	Pu240	30.21	1.02 (0.96)	32.56	1.01 (0.98)
	Pu241	11.91	1.01 (1.05)	10.87	1.01 (1.06)
	Pu242	12.39	0.98 (0.92)	14.18	0.97 (0.94)
Pu isotopic weight (mg/gU)	Pu238	0.49	0.86 (0.80)	0.34	0.92 (0.82)
	Pu239	4.97	0.96 (1.05)	3.66	1.00 (1.04)
	Pu240	3.65	0.97 (0.95)	3.09	0.99 (0.97)
	Pu241	1.44	0.96 (1.03)	1.04	1.00 (1.06)
	Pu242	1.51	0.93 (0.91)	1.36	0.95 (0.94)
Total Pu(μg/gU)		12070	0.95 (0.99)	9480	0.98 (1.00)
Am241(μg/gU)		322	1.14 (0.98)	228	0.96 (0.77)
Am243(μg/gU)		387	1.01 (1.02)	286	1.06 (1.08)
Cm242(μg/gU)		0.48	1.04 (1.14)	0.34	1.18 (1.33)
Cm244(μg/gU)		192	0.80 (0.85)	113	0.81 (0.88)

( ): C/Es of interpolation analysis

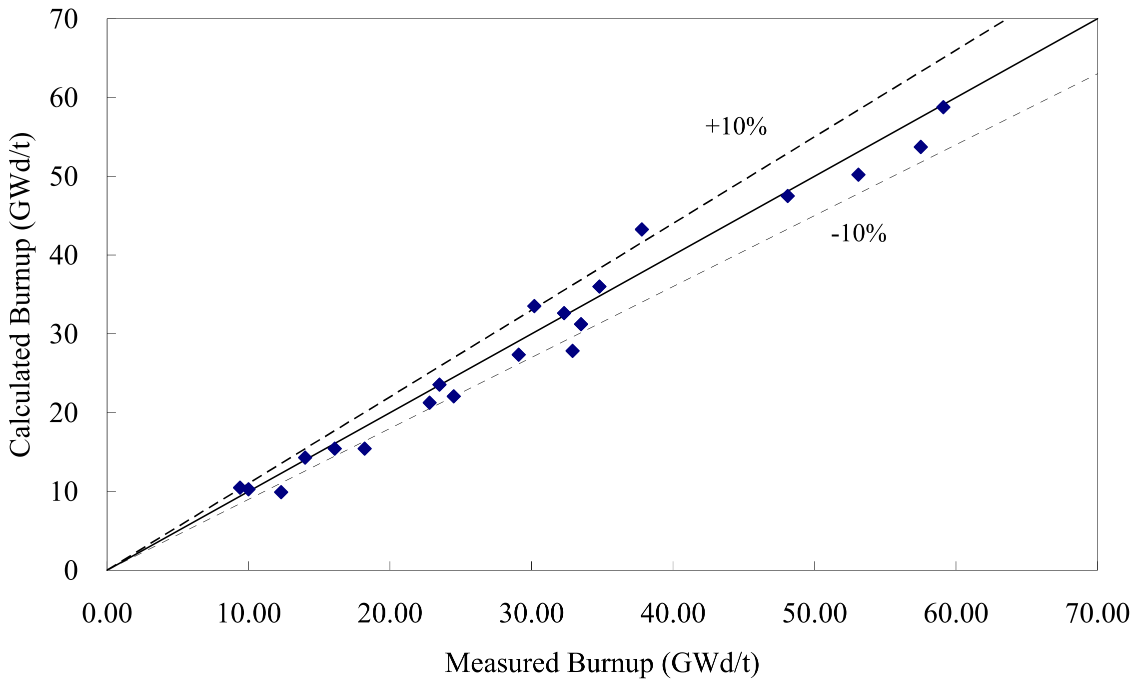


Fig. 2.3.1 Calculated burnups against measurements in SRAC calculations for BWR 8x8-4 fuel assemblies[2)]

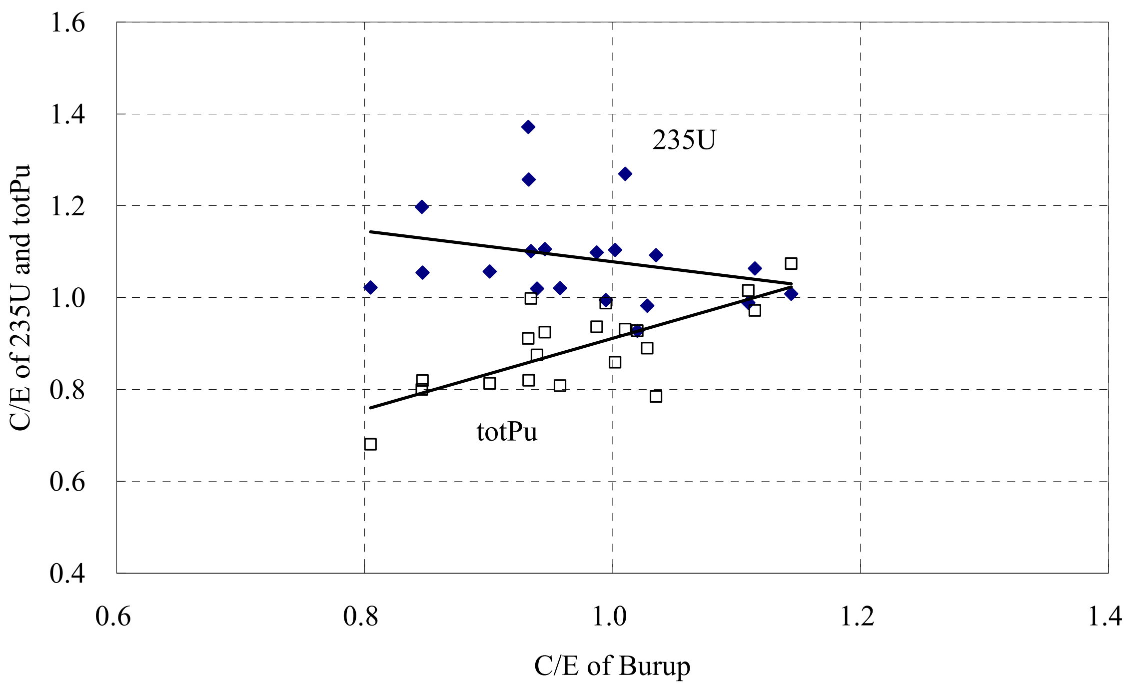


Fig. 2.3.2 C/Es and fitted curves of ^{235}U and total Pu (totPu) inventories against those of burnups for BWR 8×8-4 fuel assemblies[2)]

3. BWR 9x9-9 UO_2 Fuel of Unit 1 of Fukushima Power Station 2

3.1 Fuel Assembly and Irradiation History of BWR 9x9-9 UO_2 Fuel

Isotopic inventories were measured for the samples taken from two BWR 9x9-9 UO_2 assemblies, which were inserted as lead-use fuel assemblies of high burnup 9x9 fuel in Unit 1 of Fukushima Power Station 2 (2F1) in July 1996. One of them were discharged in May 2000 after three cycles and the other in January 2003 after five cycles for postirradiation examination.[14,15] The assemblies are referred to as 2F1ZN2 and 2F1ZN3, respectively. The major core parameters of 2F1 are shown in Table 2.1.1, and the major characteristics of the BWR 9x9-9 UO_2 fuel assembly[15,16,17] are shown in **Table 3.1.1**. **Figure 3.1.1** shows the enrichment distribution among fuel rods in the assembly.[1,16] The data of irradiation periods and assembly average burnups of the two assemblies[15] are shown in **Table 3.1.2**. The assembly average burnups of 2F1ZN2 and 2F1ZN3 are 35.6 and 53.5 GWd/t, respectively. **Figure 3.1.2** shows the locations of the assemblies in the core.[14,15] **Figure 3.1.3** and **3.1.4** show the power histories of the assemblies 2F1ZN2 and 2F1ZN3 and the fuel rods of the assemblies.[15]

A core average void fraction is 43% for the cores fully loaded with the BWR 9x9-7 UO_2 fuel assemblies.[18] Core-average axial void distribution was estimated from the axial void distribution in the hottest channel given in Reference 18 so that the axially-averaged value coincides with 43%. Table 2.1.4 shows the estimated in-channel void fraction for each axial node. The length of one node is about 15.5 cm.

Table 3.1.1 Major specifications of BWR 9x9-9 UO_2 fuel assembly[1,15,16,17)]

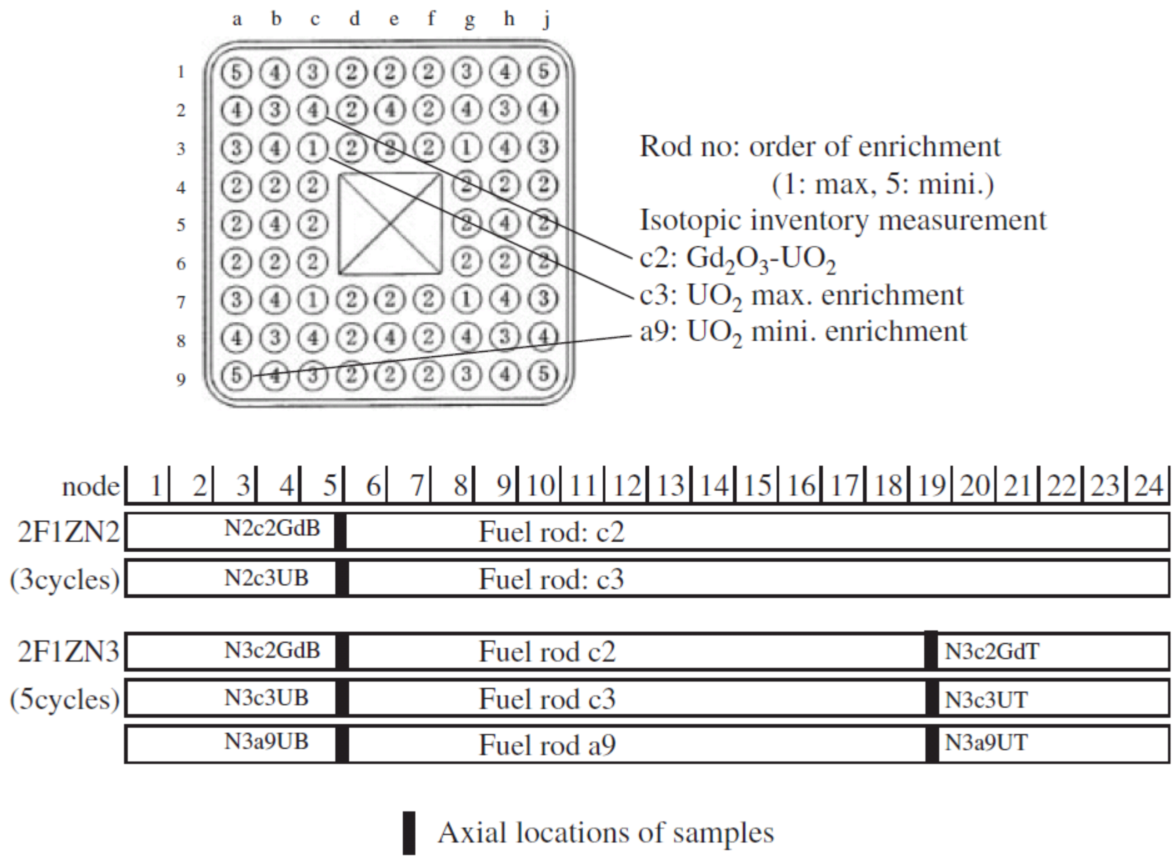
Fuel assembly	
Lattice	9x9
Fuel rod pitch (cm)	1.45*
Number of fuel rods	72
Assembly av. ^{235}U enrichment (wt%)	3.4*
Fuel Rod	
Outer diameter (mm)	11.0*
Cladding thickness (mm)	0.70* (include Zr lining 0.1*mm)
Cladding material	Zry-2 (Zr lining)
Pellet diameter (mm)	9.4*
Pellet-cladding gap (mm)	0.20*
Pellet density (%TD)	97*
Pellet material	UO_2, UO_2-Gd_2O_3
Pellet ^{235}U enrichment (wt%)	2.1 to 4.9*
Temperature of outside of cladding (℃)	340
Content of Gd_2O_3 (wt%)	5.0*
Water channel	Square
Dimension of outside	38.5 mm*
Material	Zry
Number	1

* Rounded number

Table 3.1.2 Irradiation periods and burnups of BWR 9x9-9 and -7 fuel assemblies[15)]

Irradiation cycle (Operating cycle)	1 (12th)	2 (13th)	3 (14th)	4 (15th)	5 (16th)
Start and end of cycle	1996/7/26 -1997/9/23	1997/10/31 -1998/12/17	1999/3/12 -2000/5/9	2000/6/15 -2001/8/13	2001/11/9 -2003/1/7
BWR 9x9-9 fue assembly ID	-	-	2F1ZN2 (35.6)*	-	2F1ZN3 (53.5)*
BWR 9x9-7 fue assembly ID	-	-	2F1Z3 (35.0)*	-	2F1Z2 (53.0)*

*Assembly average burnups (GWd/t)



Note: N2c3UB means N2c3UB, N2c3UB(O), (M) and (C) in Table 3.2.1

Fig. 3.1.1 Enrichment distribution of BWR 9x9-9 fuel assemblies,[1,16] and radial and axial locations of samples for inventory measurements

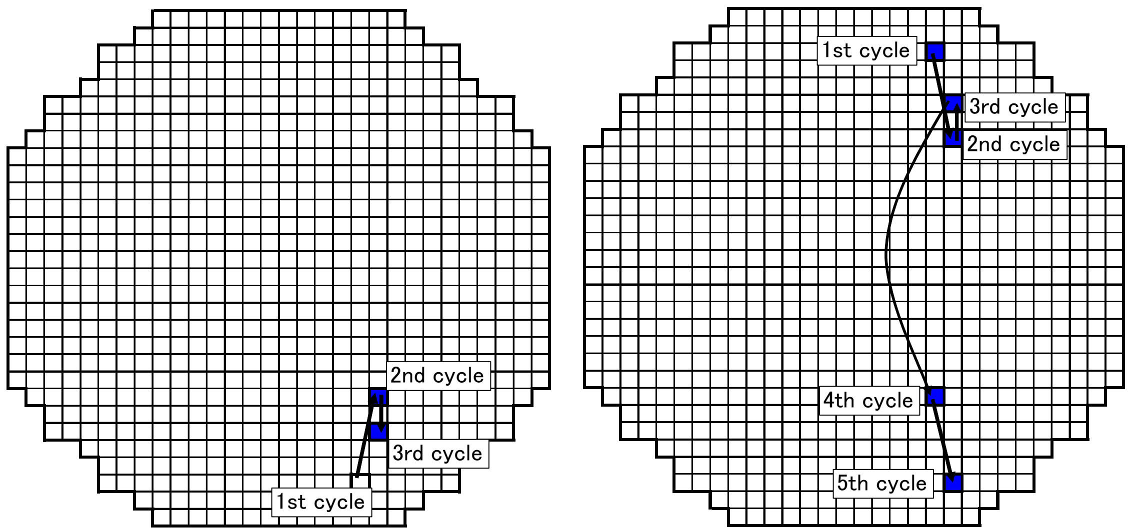


Fig. 3.12 Loading locations of 9x9-9 BWR fuel assemblies, 2F1ZN2 and 2F1ZN3[14,15]

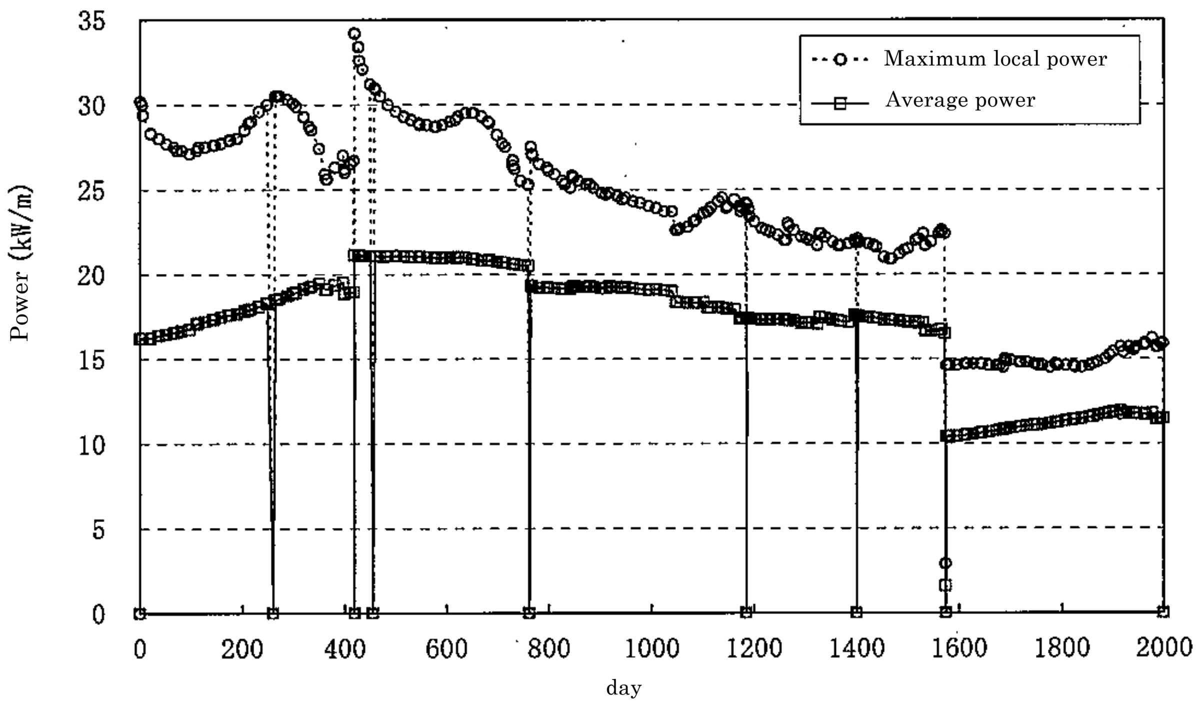
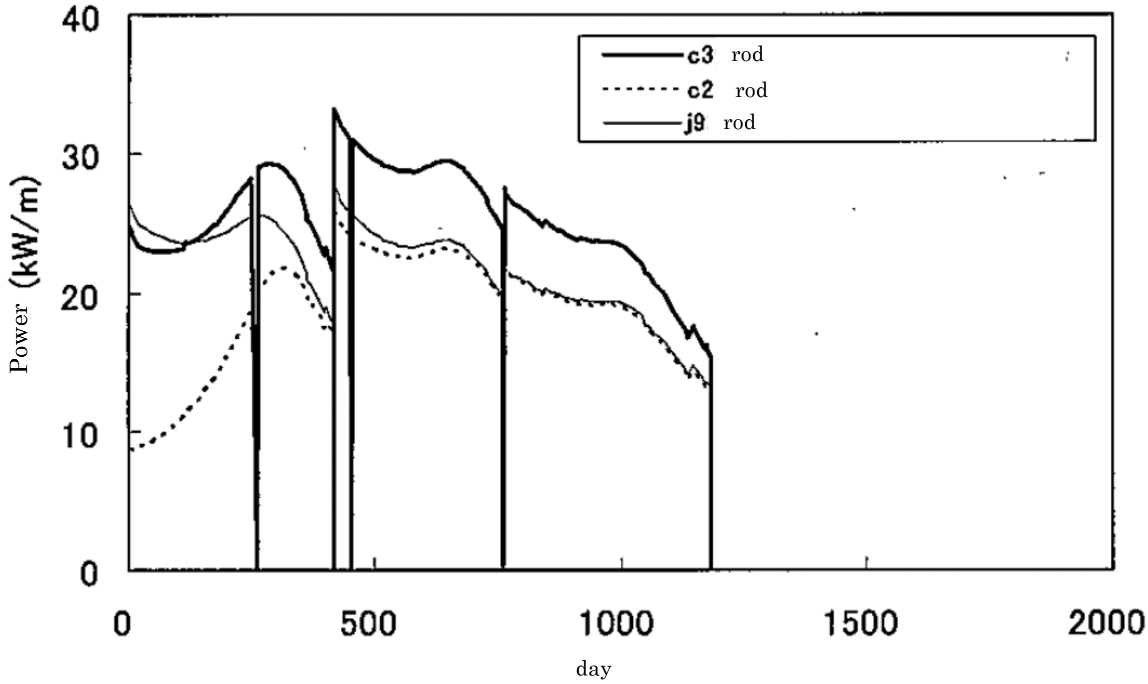
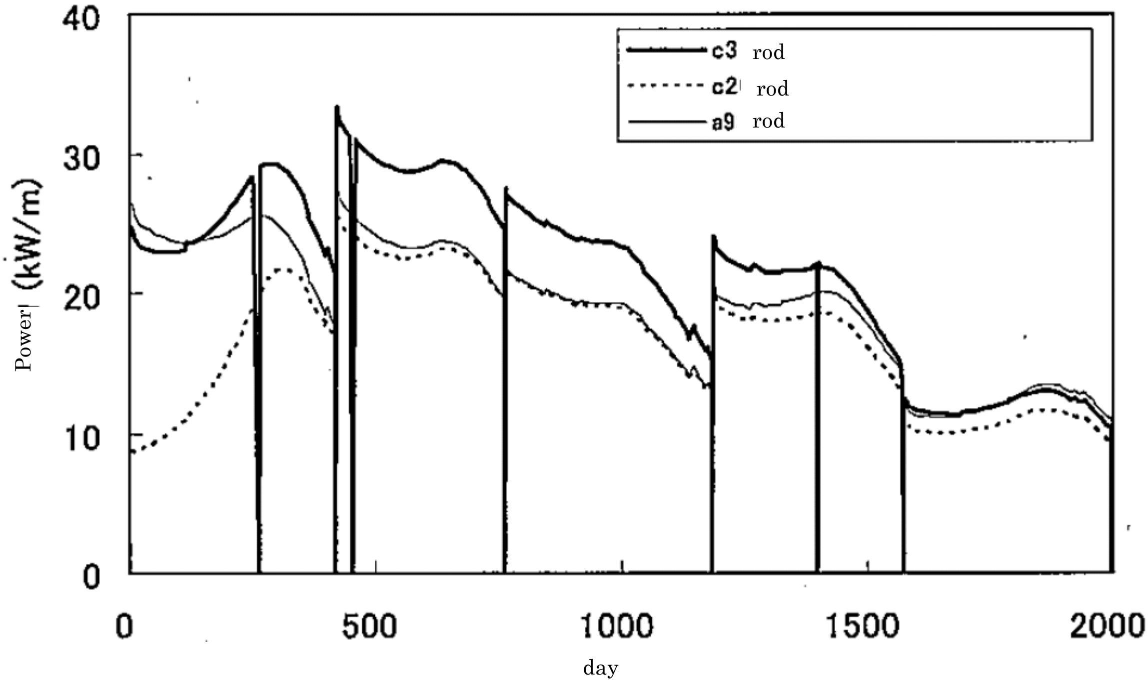


Fig. 3.1.3 Power History of 2F1ZN3[15)]



(a) Second span (around 7 node) of fuel rods of 2F2ZN2



(b) Second span (around 7 node) of fuel rods of 2F1ZN3

Fig. 3.1.4 Power histories of fuel rods of 2F1ZN2 and 2F1ZN3[15)]

3.2 Measured Isotopic Inventories of BWR 9x9-9 UO_2 Fuel

Isotopic inventories were measured for pellet-average samples and pellet-local samples at the radial distance 0.0 (C), 0.22 (M) and 0.38 (O) mm. **Table 3.2.1** listed the pellet-average samples of for the BWR 9x9-9 fuel assemblies.

3.2.1 Measurement Method

The isotopic inventories of the samples of the BWR 9x9-9 fuel assemblies was measured in the Japan Atomic Energy Agency, and the isotopic dilution mass spectrometry (IDM) was applied to Nd, U, and Pu isotopes with spikes in the forms of ^{150}Nd, ^{233}U and ^{242}Pu, respectively. The measurement errors[13)] are less than 0.1% for neodymium isotopes, ^{235}U and ^{238}U, less than 1% for ^{234}U, less than 2% for ^{236}U, less than 0.3% for $^{239, 240, 241, 242}$Pu and less than 0.5% for ^{238}Pu.

3.2.2 Measured Burnups and Isotopic Inventories of Samples

(1) Pellet-Average Samples

The burnups of the samples were determined by the Nd-148 method[8)] with the inventory data of U, Pu and ^{148}Nd, and using the effective values of fission yield of ^{148}Nd and effective energy release per fission, which were determined for each sample by the assembly burnup calculations on the 9x9-9 assemblies using the continuous energy Monte Carlo burnup calculation code.[1)] The effect of neutron absorptions of ^{147}Nd and ^{148}Nd on the determination of the effective fission yields of ^{148}Nd was also taken into account. The details of the burnup determination are given in the Reference 1. **Table 3.2.2** shows the determined burnups.

The measured isotopic inventories in the unit of atoms per total U atoms are shown in **Tables 3.2.3, 3.2.4** and **3.2.5** for the pellet-average samples.

(2) Pellet-Local Sample

The burnups of the pellet-local samples were determined by the Nd-148 method[8)] with the inventory data of U, Pu and ^{148}Nd, and using the effective fission yield of ^{148}Nd = 1.724 (=1.03 (correction factor) x 1.67366 (for thermal fission of ^{235}U))%, and GWd/t/%FIMA = 9.6.[19)] The determined burnups are shown in **Table 3.2.6**.

The measured isotopic ratio (at%) of U, Pu and Nd are shown in **Tables 3.2.7, 3.2.8** and **3.2.9**, respectively.

Table 3.2.1 Pellet-average samples and its characteristics for BWR 9x9-9 fuel assemblies[1)]

Sample ID	Assembly	Irradiation cycles	Fuel rod (fuel type)	Axial height(cm)	Axial node*	Measurement date
N2c3UB	2F1Z-N2	3 cycles (2000/5/9)**	c3(UO_2)	79.3-79.5	5.1-5.2	2002/10/10
N2c2GdB			c2(Gd_2O_3-UO_2)	76.4-76.6	4.9-5.0	2002/10/10
N2c3UB(O)			c3(UO_2)	79.5-79.8	5.1-5.2	2002/10/10
N2c3BU(M)			c3(UO_2)	79.5-79.8	5.1-5.2	2002/10/10
N2c3BU(C)			c3(UO_2)	79.5-79.8	5.1-5.2	2002/10/10
N3c3UB	2F1Z-N3	5 cycles (2003/1/7)**	c3(UO_2)	78.6-82.6	5.1-5.4	2005/8/26
N3c2GdB			c2(Gd_2O_3-UO_2)	78.6-82.6	5.1-5.4	2005/8/26
N3a9UB			a9(UO_2)	77.4-81.4	5.0-5.3	2005/10/4
N3c3UT			c3(UO_2)	285.4-289.4	18.5-18.7	2005/8/26
N3c2GdT			c2(Gd_2O_3-UO_2)	285.4-289.4	18.5-18.7	2005/8/26
N3a9UT			a9(UO_2)	285.4-288.8	18.5-18.7	2005/10/4

* One axial node=fuel effective length(about 3.71m)/24

** Date of end of cycle

Table 3.2.2 Burnups of samples for BWR 9x9-9 fuel assemblies compared with calculations[1)]

Sample ID	Assembly	Fuel rod	Axial node	Burnup (GWd/t)		
				Measurement	SRAC (C/E)	MVP-BURN (C/E)
N2c3UB	2F1ZN2	c3	5.1-5.2	47.49	46.51 (0.979)	46.10 (0.971)
N2c2GdB		c2	4.9-5.0	35.56	34.25 (0.963)	34.43 (0.968)
N3c3UB	2F1ZN3	c3	5.1-5.4	68.25	66.81 (0.979)	66.19 (0.970)
N3c2GdB		c2	5.1-5.4	54.56	52.27 (0.958)	52.59 (0.964)
N3a9UB		a9	5.0-5.3	61.20	56.18 (0.918)	57.24 (0.935)
N3c3UT		c3	18.5-18.7	59.05	62.47 (1.058)	62.05 (1.051)
N3c2GdT		c2	18.5-18.7	46.32	48.98 (1.057)	49.37 (1.066)
N3a9UT		a9	18.5-18.7	56.32	57.16 (1.015)	57.64 (1.023)

Table 3.2.3 Measured isotopic inventories (atoms/total U atoms) of U, Pu and Nd compared with analysis of SRAC for N2c3UB and N2c2GdB of BWR 9x9-9 fuel assembly[1)]

	Fuel rod	U-234	U-235	U-236	U-238
Measured	c3	3.000E-04	1.030E-02	7.000E-03	9.824E-01
	c2	2.000E-04	6.200E-03	4.300E-03	9.893E-01
Calculated	c3	2.837E-04	1.035E-02	6.876E-03	9.825E-01
	c2	1.663E-04	6.181E-03	4.225E-03	9.894E-01
C/E	c3	0.946	1.005	0.982	1.000
	c2	0.831	0.997	0.983	1.000

	Fuel rod	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	totPu
Meaured	c3	2.195E-04	4.272E-03	2.557E-03	1.002E-03	6.246E-04	8.675E-03
	c2	1.637E-04	4.175E-03	2.560E-03	9.873E-04	6.394E-04	8.526E-03
Calculated	c3	1.998E-04	4.281E-03	2.589E-03	9.887E-04	6.115E-04	8.670E-03
	c2	1.426E-04	4.067E-03	2.571E-03	9.540E-04	6.267E-04	8.362E-03
C/E	c3	0.910	1.002	1.012	0.987	0.979	0.999
	c2	0.871	0.974	1.004	0.966	0.980	0.981

	Fuel rod	Nd-142	Nd-143	Nd-144	Nd-145	Nd-146	Nd-148	Nd-150
Meaured	c3	5.670E-05	1.846E-03	3.506E-03	1.688E-03	1.734E-03	8.921E-04	4.040E-04
	c2	4.000E-05	1.325E-03	2.468E-03	1.216E-03	1.249E-03	6.604E-04	3.149E-04
Calculated	c3	5.721E-05	1.882E-03	3.485E-03	1.721E-03	1.722E-03	9.025E-04	4.076E-04
	c2	4.047E-05	1.346E-03	2.461E-03	1.239E-03	1.243E-03	6.688E-04	3.183E-04
C/E	c3	1.009	1.019	0.994	1.020	0.993	1.012	1.009
	c2	1.012	1.016	0.997	1.019	0.996	1.013	1.011

Table 3.2.4 Measured isotopic inventories (atoms/total U atoms) of U, Pu and Nd compared with analysis of SRAC for N3c3UB, N3c2GdB and N3a9UB of BWR 9x9-9 fuel assembly[1)]

	Fuel rod	U-234	U-235	U-236	U-238
Measured	c3	2.000E-04	2.400E-03	7.800E-03	9.896E-01
	c2	1.000E-04	1.400E-03	4.700E-03	9.938E-01
	a9	1.000E-04	1.000E-04	2.900E-03	9.969E-01
Calculated	c3	2.095E-04	2.613E-03	7.670E-03	9.895E-01
	c2	1.214E-04	1.415E-03	4.676E-03	9.938E-01
	a9	6.020E-05	1.287E-04	2.876E-03	9.969E-01
C/E	c3	1.047	1.089	0.983	1.000
	c2	1.214	1.011	0.995	1.000
	a9	0.602	1.287	0.992	1.000

	Fuel rod	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	totPu
Measured	c3	4.823E-04	3.803E-03	3.163E-03	1.113E-03	1.528E-03	1.009E-02
	c2	3.559E-04	3.663E-03	3.023E-03	1.068E-03	1.484E-03	9.593E-03
	a9	2.941E-04	3.115E-03	2.665E-03	9.185E-04	2.084E-03	9.076E-03
Calculated	c3	4.511E-04	3.865E-03	3.298E-03	1.104E-03	1.492E-03	1.021E-02
	c2	3.243E-04	3.692E-03	3.220E-03	1.057E-03	1.509E-03	9.803E-03
	a9	2.899E-04	3.355E-03	3.078E-03	9.919E-04	2.318E-03	1.003E-02
C/E	c3	0.935	1.016	1.043	0.992	0.976	1.012
	c2	0.911	1.008	1.065	0.990	1.017	1.022
	a9	0.986	1.077	1.155	1.080	1.113	1.105

	Fuel rod	Nd-142	Nd-143	Nd-144	Nd-145	Nd-146	Nd-148	Nd-150
Measured	c3	1.625E-04	1.683E-03	5.926E-03	2.184E-03	2.750E-03	1.307E-03	6.235E-04
	c2	1.192E-04	1.309E-03	4.398E-03	1.675E-03	2.093E-03	1.031E-03	5.181E-04
	a9	2.008E-04	9.224E-04	5.157E-03	1.662E-03	2.453E-03	1.162E-03	6.109E-04
Calculated	c3	1.594E-04	1.784E-03	5.876E-03	2.263E-03	2.710E-03	1.324E-03	6.298E-04
	c2	1.224E-04	1.356E-03	4.416E-03	1.729E-03	2.082E-03	1.045E-03	5.242E-04
	a9	1.996E-04	9.844E-04	5.180E-03	1.732E-03	2.439E-03	1.186E-03	6.233E-04
C/E	c3	0.981	1.060	0.992	1.036	0.985	1.013	1.010
	c2	1.026	1.036	1.004	1.032	0.995	1.014	1.012
	a9	0.994	1.067	1.005	1.042	0.994	1.021	1.020

Table 3.2.5 Measured isotopic inventories (atoms/total U atoms) of U, Pu and Nd compared with analysis of SRAC for N3c3UT, N3c2GdT and N3a9UT of BWR 9x9-9 fuel assembly[1)]

	Fuel rod	U-234	U-235	U-236	U-238
Measured	c3	2.000E-04	7.300E-03	7.700E-03	9.848E-01
	c2	1.000E-04	4.500E-03	4.700E-03	9.907E-01
	a9	1.000E-04	7.000E-04	3.100E-03	9.961E-01
Calculated	c3	2.379E-04	7.663E-03	7.605E-03	9.845E-01
	c2	1.403E-04	4.602E-03	4.664E-03	9.906E-01
	a9	7.449E-05	6.973E-04	3.026E-03	9.962E-01
C/E	c3	1.190	1.050	0.988	1.000
	c2	1.403	1.023	0.992	1.000
	a9	0.745	0.996	0.976	1.000

	Fuel rod	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	totPu
Measured	c3	5.232E-04	5.586E-03	3.493E-03	1.456E-03	1.025E-03	1.208E-02
	c2	4.042E-04	5.737E-03	3.626E-03	1.496E-03	1.061E-03	1.232E-02
	a9	4.081E-04	4.342E-03	3.128E-03	1.310E-03	1.754E-03	1.094E-02
Calculated	c3	4.651E-04	5.556E-03	3.550E-03	1.424E-03	1.014E-03	1.201E-02
	c2	3.376E-04	5.219E-03	3.495E-03	1.353E-03	1.008E-03	1.141E-02
	a9	3.834E-04	4.492E-03	3.453E-03	1.364E-03	1.919E-03	1.161E-02
C/E	c3	0.889	0.995	1.016	0.978	0.989	0.994
	c2	0.835	0.910	0.964	0.904	0.950	0.926
	a9	0.939	1.035	1.104	1.041	1.094	1.061

	Fuel rod	Nd-142	Nd-143	Nd-144	Nd-145	Nd-146	Nd-148	Nd-150
Measured	c3	1.006E-04	2.081E-03	4.486E-03	1.958E-03	2.290E-03	1.122E-03	5.369E-04
	c2	7.098E-05	1.582E-03	3.298E-03	1.485E-03	1.722E-03	8.698E-04	4.363E-04
	a9	1.609E-04	1.310E-03	4.352E-03	1.621E-03	2.177E-03	1.064E-03	5.659E-04
Calculated	c3	9.671E-05	2.166E-03	4.458E-03	2.026E-03	2.253E-03	1.134E-03	5.397E-04
	c2	7.111E-05	1.624E-03	3.280E-03	1.526E-03	1.697E-03	8.785E-04	4.404E-04
	a9	1.318E-04	1.360E-03	4.337E-03	1.673E-03	2.143E-03	1.079E-03	5.687E-04
C/E	c3	0.961	1.041	0.994	1.035	0.984	1.011	1.005
	c2	1.002	1.027	0.995	1.028	0.986	1.010	1.009
	a9	0.819	1.038	0.997	1.032	0.985	1.014	1.005

Table 3.2.6 Measured burnups of pellet-local samples for BWR 9x9-9 fuel assembly

Sample ID	U (atoms*)	Pu (atoms*)	Nd-148 (atoms*)	%FIMA**	Burnups*** (GWd/t)
N2c3UB(O)	4.026×10^{16}	3.498×10^{14}	3.647×10^{13}	4.95	47.5
N2c3BU(M)	2.142×10^{16}	1.652×10^{14}	1.822×10^{13}	4.67	44.8
N2c3BU(C)	3.563×10^{16}	2.723×10^{14}	2.949×10^{13}	4.55	43.7

*Atoms in solution sample

**Atomic percent fission based on the Nd-148 method with fission yield of 148Nd = 1.724 (=1.03 (correction factor) x 1.67366 (for thermal fission of ^{235}U)) %

***based on GWd/t/%FIMA = 9.6

Table 3.2.7 Measured isotopic ratio (at%) of U of pellet samples for BWR 9x9-9 fuel assembly

Sample ID	U-234	U-235	U-236	U-238
N2c3UB(O)	0.03	0.96	0.70	98.31
N2c3BU(M)	0.03	1.07	0.69	98.21
N2c3BU(C)	0.03	1.12	0.69	98.16

Table 3.2.8 Measured isotopic ratio (at%) of Pu of pellet samples for BWR 9x9-9 fuel assembly

Sample ID	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242
N2c3UB(O)	2.68	48.99	28.74	11.93	7.67
N2c3BU(M)	2.65	49.25	30.54	10.97	6.60
N2c3BU(C)	2.58	49.56	31.05	10.61	6.21

Table 3.2.9 Measured isotopic ratio (at%) of Nd of pellet samples for BWR 9x9-9 fuel assembly

Sample ID	Nd-142	Nd-143	Nd-144	Nd-145	Nd-146	Nd-148	Nd-150
N2c3UB(O)	0.58	17.89	34.94	16.59	17.18	8.82	4.00
N2c3BU(M)	0.55	18.30	34.47	16.71	17.14	8.84	3.99
N2c3BU(C)	0.54	18.65	34.30	16.76	17.05	8.77	3.93

3.3 Analysis of burnups and isotopic inventories

The analysis of the burnups and isotopic inventories of the pellet-average samples was performed by tracking the irradiation history of node power and in-channel void fraction, which were provides by the plant utility as the data of a core monitoring system. The burnup calculations with a infinite assembly model were carried out using the collision probability module (PIJ) of SRAC,[10] and MVP-BURN[11] based on the nuclear data library JENDL-3.3.[20] The geometrical dimensions and atomic number densities of the asembly were taken from the design base specifications of the fuel vender. The temperatures of the fuel cladding and the moderator were set to be 286 ℃.

Table 3.2.2 shows the results of the burnups of the pellet-average samples by the analysis of SRAC and MVP-BURN compared with the measurements.[1] **Figure 3.3.1** shows schematically the calculated burnups of SRAC against the measurements.[1] Most of calculated values are in the range of ±6%.

In order to eliminate the errors in isotopic inventories caused by the difference of the calculated sample burnups from the measurements for obtaining reliable information on the accuracy of lattice calculations, analyses have been performed so that, modifying the power history, the calculated sample burnups coincide with the measurement for each sample. The calculated inventories of SRAC are shown in Tables 3.2.3, 3.2.4 and 3.2.5 and compared with the measurement. The discussion on the C/Es are mentioned in Reference 1. Though the results of MVP-BURN are not shown here, the C/Es of the inventories of SRAC and MVP-BURN calculations are 1.00 to 1.09 for ^{235}U, 0.91 to 1.04 for ^{239}Pu, 0.96 to 1.07 for ^{240}Pu and 0.90 to 1.02 for ^{241}Pu for the six pellet-average samples excluding those of the corner fuel rod a9. The behavior of a9 is different from those of c3 and c2 in the way that the calculations overestimate the total Pu inventory up to 10%.

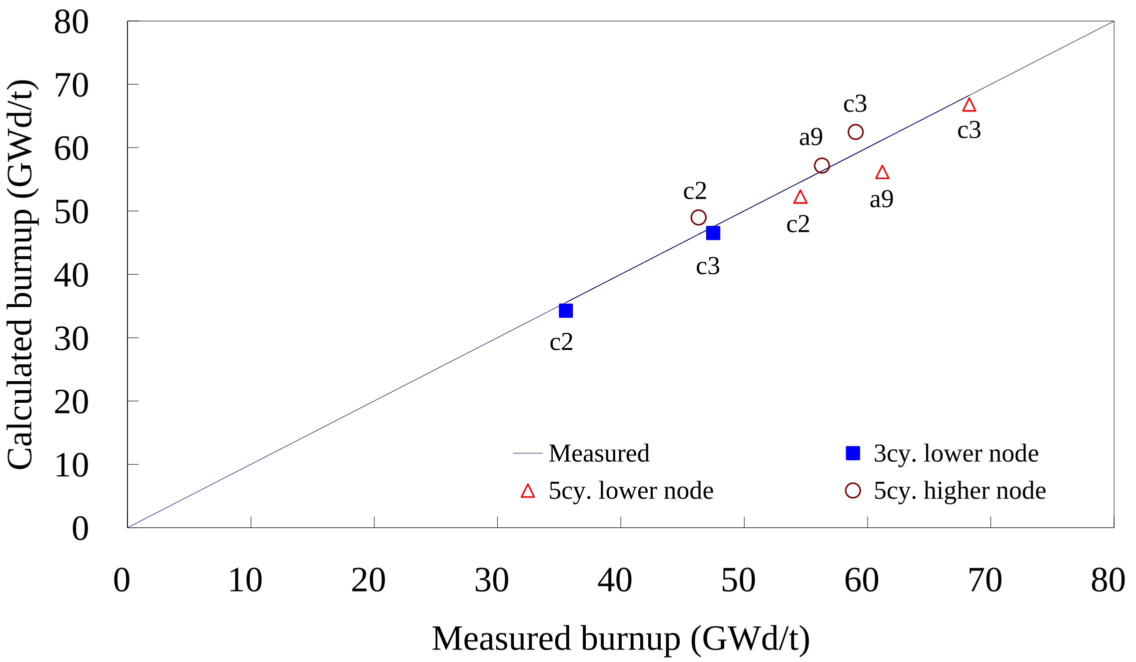


Fig. 3.3.1 Comparison of measured burnups and SRAC calculations for BWR 9x9-9 fuel assemblies[1)]

4. BWR 9x9-7 UO_2 Fuel of Unit 1 of Fukushima Power Station 2

4.1 Fuel Assembly and Irradiation History of BWR 9x9-9 UO_2 Fuel

Isotopic inventories were measured for samples taken from two BWR 9x9-9 UO2 assemblies, which were inserted as lead-use fuel assemblies of high burnup 9x9 fuel in Unit 1 of Fukushima Power Station 2 (2F1) in July 1996. One of them was discharged in May 2000 after three cycles and the other in January 2003 after five cycles for post irradiation examination.[14,15] The assemblies are referred to as 2F1Z3 and 2F1Z2, respectively. The major core parameters of 2F1 are shown in Table 2.1.1, and the major characteristics of the BWR 9x9-7 UO_2 fuel assembly[5,15,16,21] are shown in **Table 4.1.1**. **Figure 4.1.1** shows the enrichment distribution among fuel rods in the assembly.[15,16] The irradiation periods and assembly average burnups of the two assemblies are shown in Table 3.1.2.[15] The assembly average burnups of 2F1Z3 and 2F1Z2 are 35.0 and 53.0 GWd/t, respectively. **Figure 4.1.2** shows the locations of the assemblies in the core.[15] **Figures 4.1.3**, **4.1.4**, and **4.1.5** show the power histories of the assemblies 2F1Z2 and the fuel rods of the assembly.[15]

A core average void fraction is 43% for the cores fully loaded with the BWR 9x9-7 UO_2 fuel assemblies.[18] Core-average axial void distribution was estimated from the axial void distribution in the hottest channel given in the reference[18] so that the axially-averaged value coincides with 43%. Table 2.1.4 shows the estimated in-channel void fraction for each axial node. The length of one node is about 15.5 cm.

Table 4.1.1 Major specifications of BWR 9x9-7 UO_2 fuel assembly[5,15,16,21)]

Fuel assembly	
Lattice	9x9
Fuel rod pitch (mm)	14.4*
Number of fuel rods	74 (include 8 part-length rods)
Effective fuel rod length (cm)	371* (normal rod) 216* (part-length rod)
Assembly av. ^{235}U enrichment (wt%)	3.4*
Fuel Rod	
Outer diameter (mm)	11.2*
Cladding thickness (mm)	0.71* (include Zr lining 0.1*mm)
Cladding material	Zry-2 (Zr lining)
Pellet diameter (mm)	9.6*
Pellet-cladding gap (mm)	0.20*
Pellet density (%TD)	97*
Pellet material	UO_2, UO_2-Gd_2O_3
Pellet ^{235}U enrichment (wt%)	2.5 to 4.9 wt%*
Gd_2O_3 density	≦6
Water rod	
Outer diameter (mm)	24.9 *
Material	Zry
Number	2

* Rounded number

	A	B	C	D	E	F	G	H	J
1	9	8	5	3	5	3	5	8	9
2	8	4P	7G	6	2P	7G	6	4P	8
3	5	7G	8	1	6	3	7G	6	5
4	3	6	1	8	W		3	7G	3
5	5	2P	6				6	2P	5
6	3	7G	3	W		8	1	6	3
7	5	6	7G	3	6	1	8	7G	5
8	8	4P	6	7G	2P	6	7G	4P	8
9	9	8	5	3	5	3	5	8	9

Rod number: order of enrichment for each assembly in descending order,

G: Gd_2O_3-UO_2 fuel rod, P: part length rod, W: water channel or rod

Fig. 4.1.1 Enrichment distribution of BWR 9x9-7 fuel assemblies[15,16)]

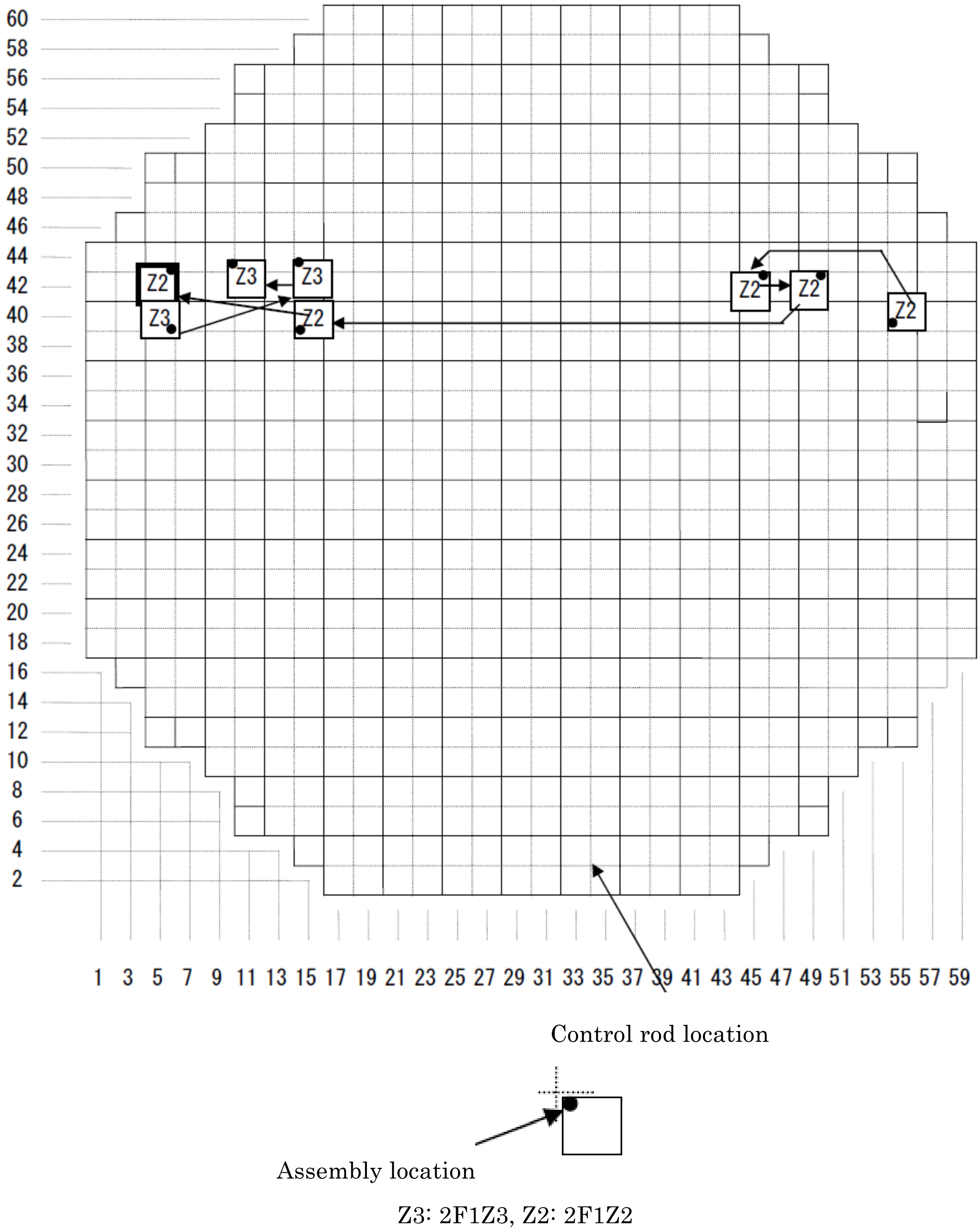


Fig. 4.1.2 Loading locations of 9x9-7 BWR fuel assemblies, 2F1Z3 and 2F1Z2[15)]

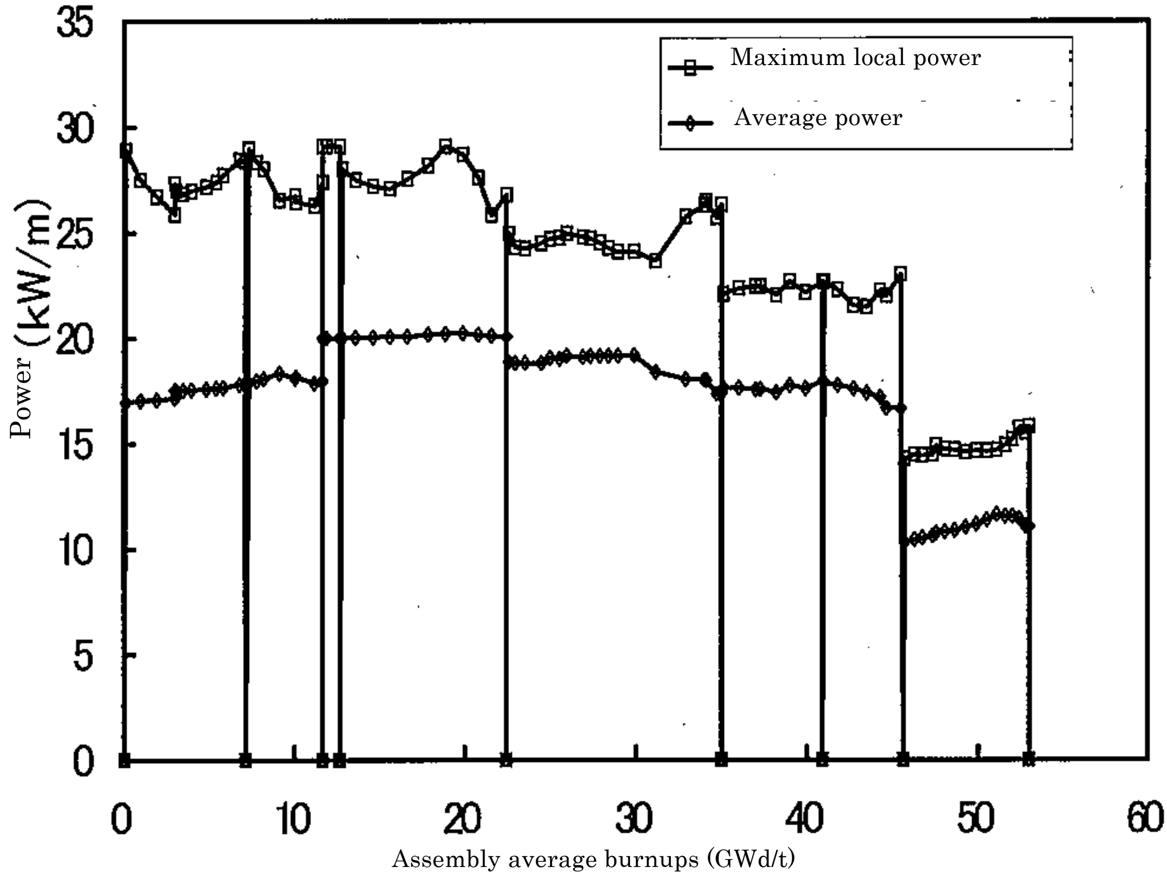


Fig. 4.1.3 Power histories of 2F1Z2[15)]

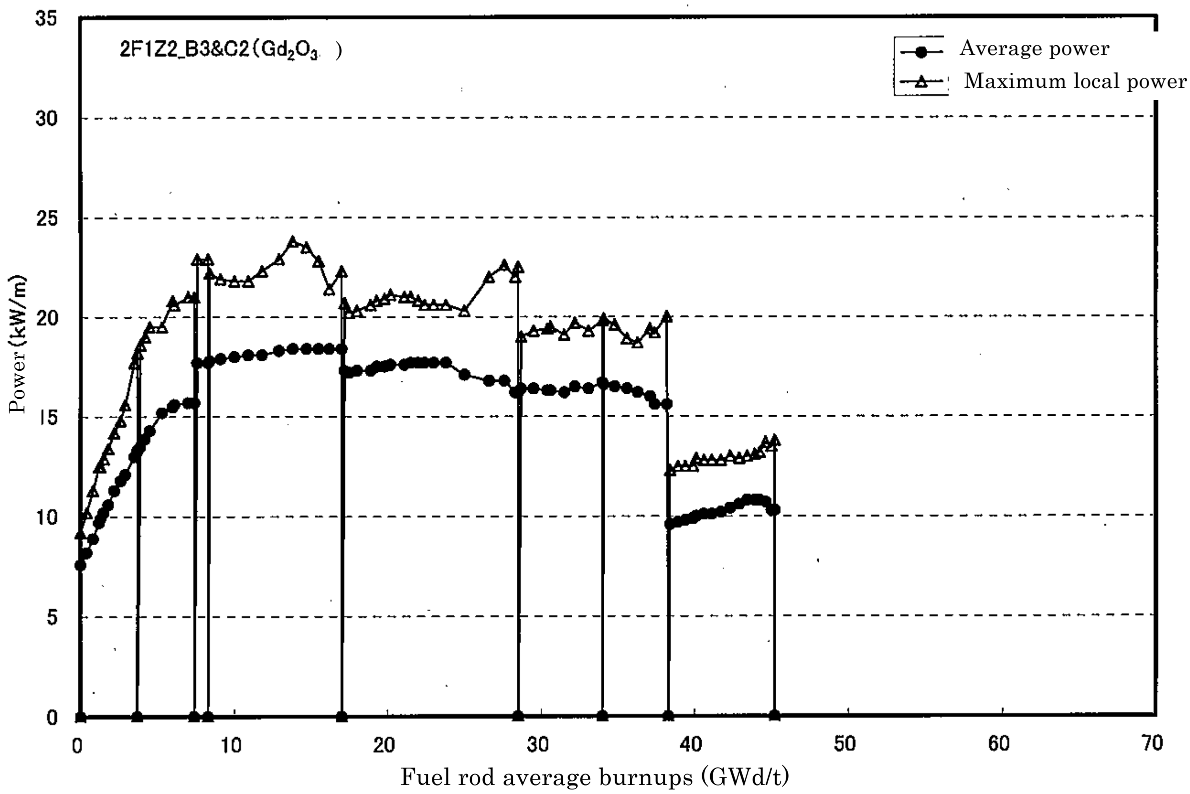
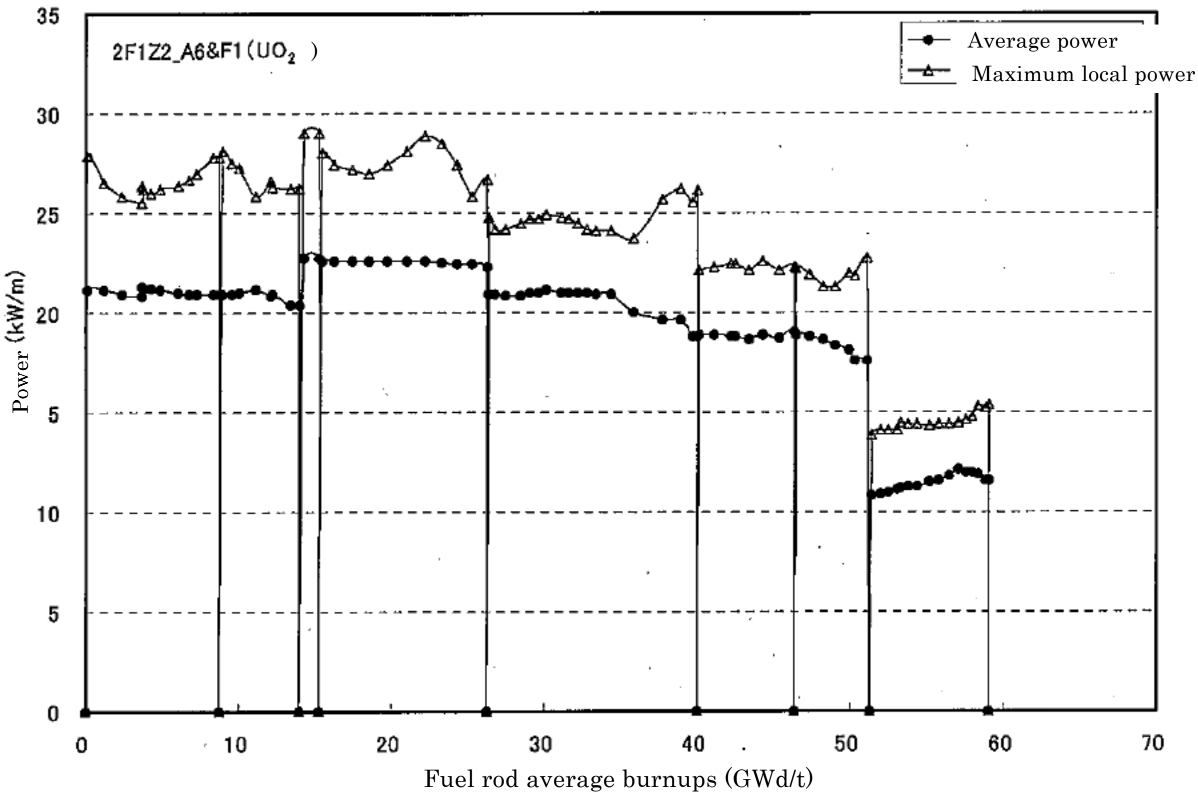
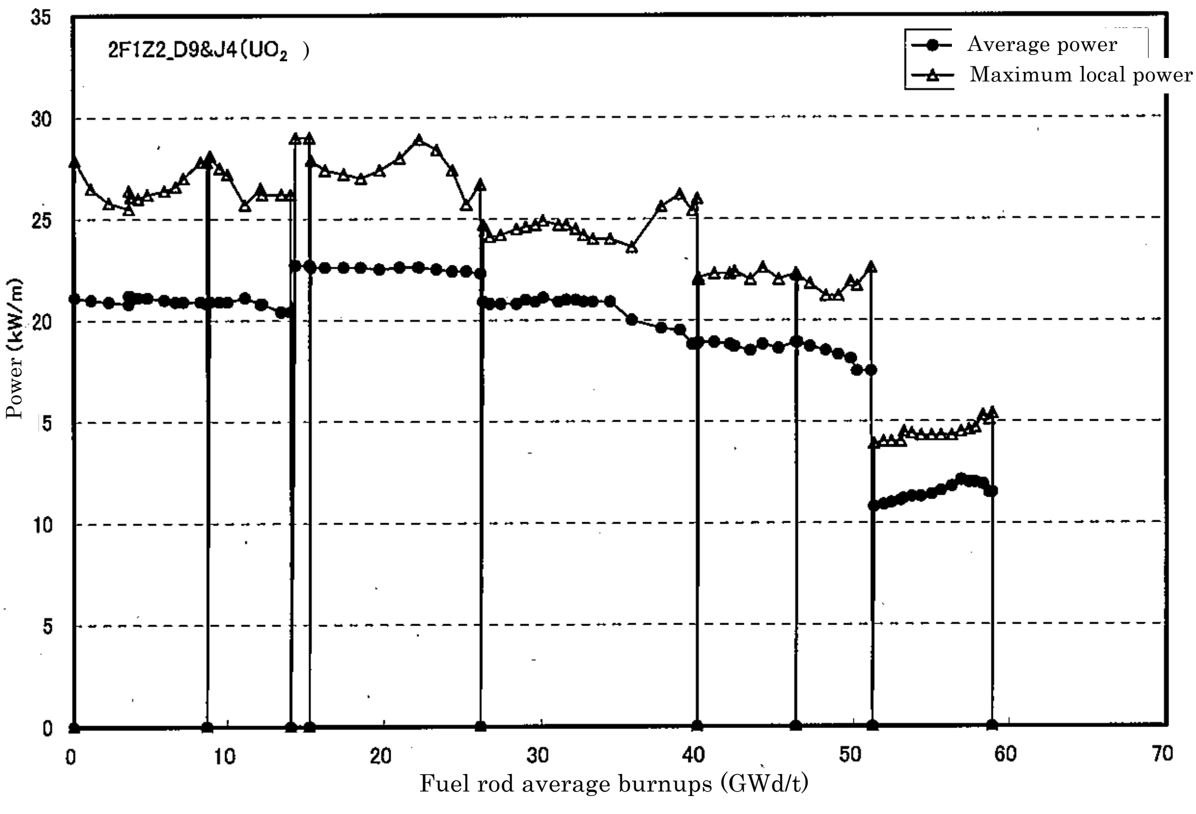


Fig. 4.1.4 Power histories of C2 rod of 2F1Z2[15)]



(a) A6 and F1



(b) D9

Fig. 4.1.5 Power histories of A6, F1, and D9 of 2F1Z2[15)]

4.2 Measured Isotopic Inventories of BWR 9x9-9 UO_2 Fuel

4.2.1 Measurement Method

Isotopic inventories were measured for pellet-average samples and pellet-local samples. The pellet-local samples were taken from a part of the pellet cross section in the form shown in **Fig. 4.2.1**. The isotopic inventories of the samples of the BWR 9x9-7 fuel assemblies was measured in the Nippon Nuclear Fuel Development,[15] and the isotopic dilution mass spectrometry (IDM) was applied to Nd, U, and Pu isotopes with spikes in the forms of ^{150}Nd, natural U, and ^{242}Pu, respectively, and the alpha spectrometry for Am and Cm isotopes.

The measurement errors[22] are shown in **Table 4.2.1**, which is based on that the errors in the atomic numbers of ^{235}U, ^{150}Nd, ^{242}Pu in the spikes added to the solution samples are 2%, and the reproducible errors of the mass spectrometry is 0.3%.

4.2.2 Measured Burnups and Isotopic Inventories of Samples

The burnups of the samples were determined by the Nd-148 method[8] with the inventory data of U, Pu and ^{148}Nd, and using the representative values of the effective fission yield of ^{148}Nd, and energy release per fission, 1.70% and 204.6 MeV/fission, respectively, which were determined by the assembly burnup calculations on the 9x9-7 assembly using the continuous energy Monte Carlo burnup calculation code.[2] The errors of burnups are estimated to be about 6.4%,[9] which include the errors related to adding the spikes, the mass spectrometry, the alpha spectrometry, the effective fission yield of ^{148}Nd and the effective energy release per fission.

The burnups of the samples were also determined using the values of the effective fission yield of ^{148}Nd, and energy release per fission calculated for each sample. The effect of neutron absorptions of ^{147}Nd and ^{148}Nd on the determination of the effective fission yields of ^{148}Nd was also taken into account. The method of the burnup determination are given in Reference 1. **Table 4.2.2** and **4.2.3** show the obtained burnups.

The measured isotopic inventories in the unit of the weight ratio to total U at the measurement date are shown in **Tables 4.2.2** and **4.2.3** for BWR 9x9-7 fuel assemblies.

Table 4.2.1 Measurement errors for BWR 9x9-7 UO_2 and Tsuruga MOX fuel assemblies in percent (one sigma)[22)]

Measurement method	Isotope	UO2	MOX
IDM	Nd-146	2.5	3.6
	Nd-148	2.6	3.7
	U-236	2.2	2.3
	total U	2.2	2.3
	Pu-239	2.0	2.0
	total Pu	2.0	2.0
Alpha spectrometry	Cm-244	2.8	2.8
	Am-243	3.4	3.4

Table 4.2.2 Measured isotopic inventories and burnups of pellet-average samples for BWR 9x9-7 fuel assemblies

Sample ID		A04F1BU01	A04F2BU01	A04F2BU02	A04F2TU01	A04F2TU02	A04F2TU03	A04F2TU04
Assembly		2F1Z3	2F1Z2	2F1Z2	2F1Z2	2F1Z2	2F1Z2	2F1Z2
Rod location		A6	F1	F1	C2	C2	D9	D9
Type of pellet		UO2	UO_2 +Al-Si-O	UO_2 +Al-Si-O	$(U,Gd)O_2$	$(U,Gd)O_2$	UO_2	UO_2
Enrichment (wt%)		4.4	4.4	4.4	3.4	3.4	4.4	4.4
G_2O_3 (wt%)		-	-	-	5.0	5.0	-	-
Axial height (mm)*1		1300-1320	1280-1300	3301-3321	1244-1264	3228-3249	1280-1300	3300-3320
U isotope (atomic%)	U-234	0.02	0.02	0.02	0.01	0.01	0.02	0.02
	U-235	1.20	0.45	0.15	0.49	0.23	0.42	0.18
	U-236	0.60	0.69	0.68	0.54	0.54	0.70	0.69
	U-238	98.17	98.83	99.15	98.95	99.21	98.87	99.12
U isotope (mg/gU(t))*2	U-234	0.24	0.19	0.17	0.15	0.14	0.19	0.18
	U-235	11.82	4.49	1.53	4.87	2.31	4.10	1.75
	U-236	6.00	6.85	6.78	5.34	5.38	6.89	6.82
	U-238	981.94	988.46	991.53	989.64	992.17	988.81	991.25
Pu assay		IDM	IDM	IDM	IDM	IDM	IDM	IDM
Pu isotope (atomic%)	Pu-238	2.28	4.76	5.02	3.75	3.85	4.71	4.61
	Pu-239	55.90	44.81	38.49	46.66	40.52	42.49	37.75
	Pu-240	23.87	27.61	29.74	29.02	31.13	29.40	31.17
	Pu-241	12.67	12.09	10.69	11.68	11.01	11.62	10.35
	Pu-242	5.28	10.72	16.06	8.89	13.49	11.78	16.11
Pu isotope (μg/gU(t))*2	Pu-238	232	552	484	383	338	500	401
	Pu-239	5711	5218	3724	4786	3572	4528	3295
	Pu-240	2449	3228	2888	2989	2755	3146	2732
	Pu-241	1305	1419	1043	1208	978	1249	911
	Pu-242	546	1264	1573	924	1204	1271	1424
Total Pu (μg/gU(t))		10243	11683	9712	10290	8847	10695	8763
Nd isotope (atomic%)	Nd-142	—	1.26	1.55	1.98	1.95	1.66	2.03
	Nd-143	20.14	14.91	10.55	16.27	12.95	14.32	10.79
	Nd-144	32.15	36.86	40.95	34.83	38.12	37.27	40.59
	Nd-145	16.81	15.00	14.45	15.29	14.96	14.96	14.61
	Nd-146	17.59	18.59	19.12	18.09	18.45	18.48	18.75
	Nd-148	9.07	8.98	8.99	9.01	9.03	8.92	8.90
	Nd-150	4.25	4.41	4.39	4.54	4.54	4.38	4.33
Nd isotope (mg/gU(t))*2	Nd-142	-	0.099	0.136	0.123	0.133	0.133	0.169
	Nd-143	1.01	1.18	0.932	1.02	0.893	1.15	0.906
	Nd-144	1.63	2.95	3.64	2.21	2.65	3.03	3.43
	Nd-145	0.856	1.21	1.29	0.975	1.05	1.22	1.24
	Nd-146	0.902	1.51	1.72	1.16	1.30	1.52	1.61
	Nd-148	0.472	0.738	0.822	0.586	0.644	0.745	0.773
	Nd-150	0.224	0.367	0.407	0.299	0.328	0.370	0.381
Am isotope (μg/gU(t))*2	Am-241	-	-	-	216	264	296	214
	Am-243	-	-	-	158	234	256	303
Cm isotope (μg/gU(t))*2	Cm-242	-	-	-	0.071	0.032	0.078	0.056
	Cm-243	-	-	-	0.65	0.26	0.77	0.31
	Cm-244	-	-	-	114	145	172	174
%FIMA for Nd-148 Y(%)=1.7		4.23	6.46	7.15	5.21	5.70	6.52	6.76
Burnups (GWd/t) for GWd/t/%FIMA=9.6		40.6	62.0	68.6	50.0	54.7	62.6	64.9
Burnups (GWd/t) *3		-	62.4	69.0	50.2	54.9	63.0	65.3

*1: distance from the shoulder of the upper-end plug of fuel rods which is about 350 mm higher than the top of an effective fuel part
*2: weight ratio to uranium at measurements
*3: burnups determined taking into account effective parameters for each sample (see text)
Analysis of U :2001/12/19, Pu:2001/12/18 to 2001/12/20 for A04F1BU01
IDM of U, Pu for A04F2TU01-04,BU01～02: 2006/11/13 to 2006/11/15
Normarization of Am, Cm: 2006/12/1
Data for Cm-242, Cm-243 are tentative

Date of analysis	A04F1BU01: 2001/12/10
	A04F2TU01-TU03: 2006/10/20
	A04F2TU04, A04F2BU01-02: 2006/10/23

Table 4.2.3 Measured isotopic inventories and burnups of pellet-local samples for BWR 9x9-7 fuel assembly

Sample ID		A03F1EP06	A03F1EP08
Assembly		2F1Z2	2F1Z2
Rod location		A6	A6
Type of pellet		UO2	UO2
Enrichment (wt%)		4.4	4.4
G_2O_3 (wt%)		-	-
Axial height (mm)*1		586-587	1414-1415
U isotope (atomic%)	U-234	0.03	0.02
	U-235	1.55	0.45
	U-236	0.55	0.69
	U-238	97.87	98.84
U isotope (mg/gU(t))*2	U-234	0.27	0.19
	U-235	15.30	4.45
	U-236	5.46	6.85
	U-238	978.97	988.52
Pu assay		IDM	IDM
Pu isotope (atomic%)	Pu-238	1.97	4.98
	Pu-239	59.06	43.15
	Pu-240	25.74	29.77
	Pu-241	9.50	11.68
	Pu-242	3.72	10.42
Pu isotope (μg/gU(t))*2	Pu-238	150	536
	Pu-239	4528	4662
	Pu-240	1982	3231
	Pu-241	735	1273
	Pu-242	289	1140
Total Pu (μg/gU(t))		7684	10842
Nd isotope (atomic%)	Nd-142	0.53	1.03
	Nd-143	21.10	14.83
	Nd-144	32.05	37.13
	Nd-145	17.03	15.09
	Nd-146	16.61	18.60
	Nd-148	8.69	8.96
	Nd-150	3.98	4.36
Nd isotope (mg/gU(t))*2	Nd-142	0.0206	0.0697
	Nd-143	0.8241	1.0101
	Nd-144	1.2606	2.5467
	Nd-145	0.6745	1.0422
	Nd-146	0.6624	1.2935
	Nd-148	0.3513	0.6317
	Nd-150	0.1631	0.3115
Am isotope (μg/gU(t))*2	Am-241	-	-
	Am-243	-	-
Cm isotope (μg/gU(t))*2	Cm-242	-	-
	Cm-243	-	-
	Cm-244	-	-
Np isotope (μg/gU(t))*2	Np-237	-	-
%FIMA for Nd-148 Y(%)=1.7		3.19	5.60
Burnups (GWd/t) for GWd/t/%FIMA=9.6		30.6	53.8
Date of normalization of data except Am and Cm		2006/1/1	2006/1/1
Date of normalization of Am and Cm		-	-

*1: distance from the shoulder of the upper-end plug of fuel rods which is about 350 mm higher than the top of an effective fuel part

*2: weight ratio to uranium at measurements

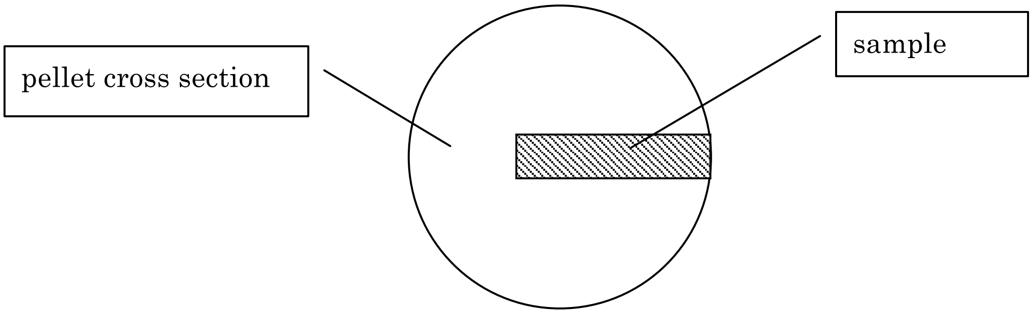


Fig. 4.2.1 Sample geometry of pellet-local sample of 9x9-7 fuel assembly

4.3 Analysis of Burnups and Isotopic Inventories

The analysis of the burnups and isotopic inventories of the pellet-average samples taken from 2F1Z2 was performed by tracking the irradiation history of node power and in-channel void fraction, which were provides by the plant utility from a core monitoring system. The burnup calculations with the infinite assembly model were carried out using the collision probability module (PIJ) of 107 neuron energy groups of the general purpose nuclear calculation code system SRAC,[8] and the continuous energy Monte Carlo burnup calculation code MVP-BURN[9] based on the nuclear data library, JENDL-3.3.[15] The geometrical dimensions and atomic number densities of the assembly were taken from the design base specifications of the fuel vender. The temperatures of the fuel cladding and the moderator were set to be 286 ℃.

Table 4.3.1 shows the results of the burnups of the pellet-average samples for the five-cycle irradiated BWR 9x9-7 assembly by the analysis of SRAC and MVP-BURN compared with the measurements (Burnups*3 in Table 4.2.2). **Figure 4.3.1** schematically shows the calculated burnups of SRAC against the measurements. Most of calculated values are in the range of ±7%.

In order to eliminate the errors in isotopic inventories caused by the difference of the calculated sample burnups from the measurement for obtaining reliable information on the accuracy of lattice calculations, analyses have been performed so that, modifying the power history, the calculated sample burnups coincide with the measurement for each sample. The calculated inventories of MVP-BURN are shown in **Tables 4.3.2** compared with the measurements. The calculated inventories of Pu isotopes generally overestimate the measurements. Table 4.3.2 also shows the comparison of Pu isotopes in the unit of an atomic percent (at%). The general trend of the C/Es seen in at% base is commonly known in the references.[1, 2] **Figure 4.3.2** schematically shows the changes of the C/Es in the samples, which indicate a systematic trend that the C/Es of ^{239}Pu and ^{241}Pu decrease with increase of those of ^{240}Pu and ^{242}Pu. This is typically observed in the isotopic compositions of Pu when the neutron spectrum change to be soft Therefore it is likely that the deviations of the C/Es from 1.0 in the isotopic inventories are due to the neutron spectrum calculations, which are related to the calculation conditions such as the in-channel void fractions, the infinite assembly model with neglecting the effect from neighboring fuel assemblies and others. It is noted that the fuel rods, from which the samples were taken, located in the periphery of the fuel bundle and easy to be affected from the neighboring fuel assemblies during the irradiation. Further study on the calculations and the measurements is necessary to inquiry the observed discrepancies between the calculated and measured inventories.

Table 4.3.1 Burnups of pellet-average samples for BWR 9x9-7 fuel assemblies compared with calculations

(Unit: GWd/t)

Sample ID	Rod-node	Measurement	SRAC	SRAC (C/E)	MVP	MVP (C/E)
A04F2TU04	D9-5/6	65.3	66.3	1.015	66.1	1.012
A04F2TU03	D9-18/19	63.0	66.5	1.056	66.4	1.054
A04F2BU02	F1-5/6	69.0	66.3	0.961	66.0	0.957
A04F2BU01	F1-18/19	62.4	66.5	1.066	66.4	1.064
A04F2TU02	C2-5/6	54.9	53.5	0.974	53.7	0.978
A04F2TU01	C2-18/19	50.2	53.7	1.070	54.1	1.078

Table 4.3.2(1) Calculated inventories of MVP-BURN and comparison with measurements for five-cycle irradiated BWR 9x9-7 fuel assembly (1/2)

Sample ID	A04F2TU04			A04F2TU03			A04F2BU02			A04F2BU01		
Rod-node	D9-5/6			D9-18/19			F1-5/6			F1-18/19		
Isotope	Calculation (mg/gU(t)	C/E	C/E (at% base)	Calculation (mg/gU(t)	C/E	C/E (at%)	Calculation (mg/gU(t)	C/E	C/E (at% base)	Calculation (mg/gU(t)	C/E	C/E (at% base)
U-234	0.18	1.00		0.19	1.00		0.17	1.00		0.19	1.01	
U-235	2.01	1.15		4.13	1.01		1.51	0.99		4.26	0.95	
U-236	6.74	0.99		6.86	1.00		6.74	0.99		6.86	1.00	
U-238	991.07	1.00		988.82	1.00		991.57	1.00		988.69	1.00	
Pu-238	0.4315	1.08	0.91	0.51	1.02	0.91	0.4580	0.95	0.87	0.50	0.90	0.89
Pu-239	4.0696	1.24	1.05	5.12	1.13	1.01	4.0558	1.09	1.00	5.11	0.98	0.96
Pu-240	3.1805	1.16	0.99	3.56	1.13	1.01	3.2578	1.13	1.04	3.54	1.10	1.08
Pu-241	1.1069	1.22	1.03	1.42	1.14	1.02	1.1019	1.06	0.97	1.40	0.99	0.97
Pu-242	1.5484	1.09	0.92	1.38	1.09	0.97	1.6912	1.08	0.99	1.35	1.07	1.05
Pu-Total	10.3370	1.18		12.00	1.12		10.5647	1.09		11.90	1.02	
Am-241	0.2899	1.35		0.3859	1.30		0.2843			0.3814		
Am-243	0.3807	1.26		0.3594	1.40		0.4518			0.3551		
Cm-242	7.04E-05	1.17		8.87E-05	1.11		7.19E-05			8.81E-05		
Cm-243	6.91E-04	2.23		8.80E-04	1.14		7.31E-04			8.55E-04		
Cm-244	0.1737	1.00		0.1685	0.98		0.2263			0.1650		
Nd-142	0.095	0.56		0.076	0.57		0.109	0.80		0.074	0.75	
Nd-143	0.991	1.09		1.199	1.04		0.963	1.03		1.198	1.02	
Nd-144	3.407	0.99		3.028	1.00		3.647	1.00		2.992	1.01	
Nd-145	1.293	1.04		1.265	1.04		1.334	1.03		1.256	1.04	
Nd-146	1.597	0.99		1.510	0.99		1.709	0.99		1.494	0.99	
Nd-148	0.787	1.02		0.757	1.02		0.831	1.01		0.750	1.02	
Nd-150	0.384	1.01		0.373	1.01		0.409	1.00		0.369	1.00	

Table 4.3.2(2) Calculated inventories of MVP-BURN and comparison with measurements for five-cycle irradiated BWR 9x9-7 fuel assembly (2/2)

Sample ID	A04F2TU02			A04F2TU01		
Rod-node	C2-5/6			C2-18/19		
Isotope	Calculation (mg/gU(t))	C/E	C/E (at% base)	Calculation (mg/gU(t))	C/E	C/E (at% base)
U-234	0.14	1.02		0.15	1.03	
U-235	2.37	1.03		4.81	0.99	
U-236	5.33	0.99		5.30	0.99	
U-238	992.16	1.00		989.74	1.00	
Pu-238	0.35	1.03	0.89	0.37	0.96	0.86
Pu-239	4.18	1.17	1.01	5.24	1.10	0.98
Pu-240	3.24	1.18	1.01	3.47	1.16	1.04
Pu-241	1.12	1.14	0.98	1.34	1.11	0.99
Pu-242	1.38	1.14	0.98	1.08	1.17	1.05
Pu-Total	10.26	1.16		11.50	1.12	
Am-241	0.2965	1.12		0.3692	1.71	
Am-243	0.3369	1.44		0.2619	1.66	
Cm-242	7.01E-05	2.34		7.74E-05	1.11	
Cm-243	6.73E-04	2.59		7.25E-04	1.12	
Cm-244	0.1422	0.98		0.1115	0.98	
Nd-142	0.07	0.51		0.05	0.39	
Nd-143	0.93	1.04		1.04	1.02	
Nd-144	2.66	1.00		2.21	1.00	
Nd-145	1.09	1.04		1.01	1.03	
Nd-146	1.29	0.99		1.14	0.98	
Nd-148	0.66	1.02		0.59	1.01	
Nd-150	0.33	1.00		0.30	0.99	

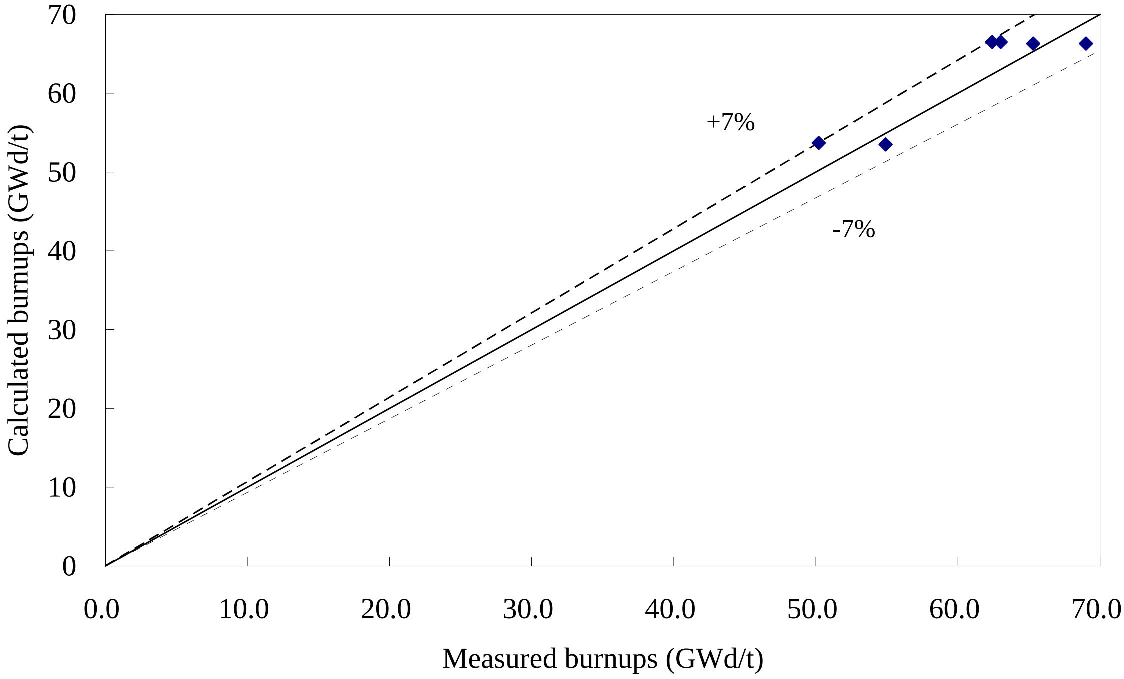


Fig. 4.3.1 Comparison of measured burnups and SRAC calculations for BWR 9x9-9 fuel assemblies

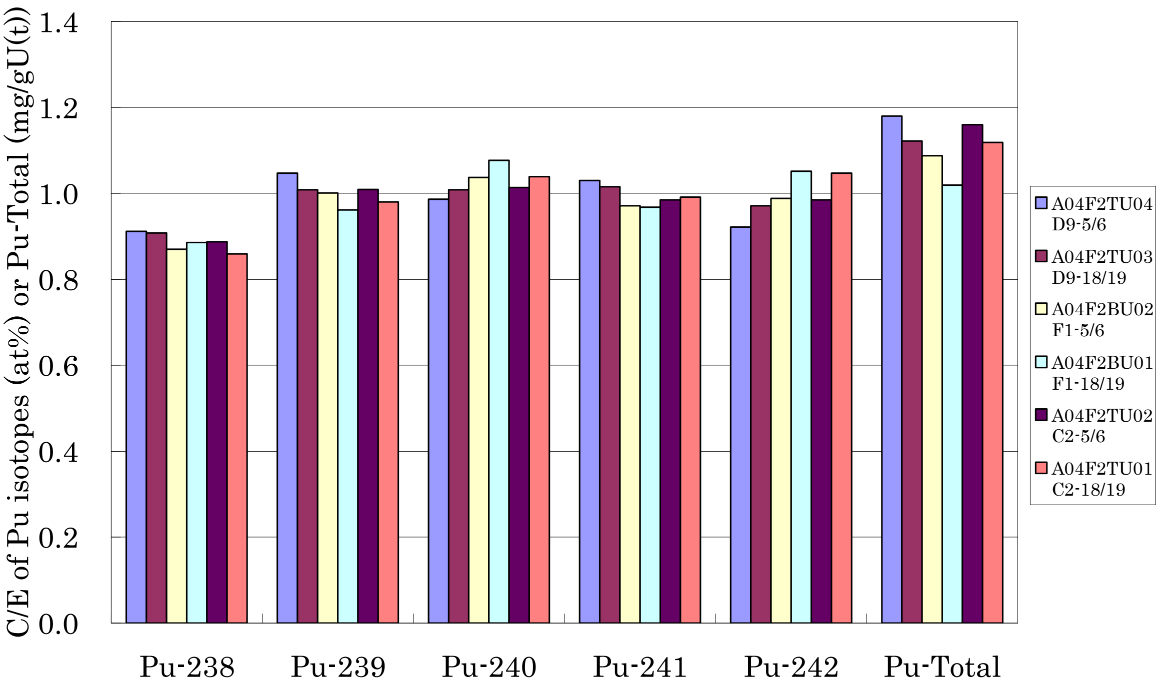


Fig. 4.3.2 Comparison of Pu isotopic inventories calculated by MVP-BURN with the measurements in atomic percents for Pu isotopes and mg/gU(t) for Pu- total on BWR 9x9-7 fuel assembly

5. BWR 8x8-2 MOX Fuel of Tsuruga-1

5.1 Fuel Assembly and Irradiation History of BWR 8x8-2 MOX Fuel

Isotopic inventories were measured for samples taken from a BWR 8x8-2 MOX assembly. The assembly was irradiated for three cycles from 1986 through 1990 in Tsuruga-1, which is a BWR operated by Japan Atomic Power Company (JAPC), for a demonstration program of utilization of MOX fuel.[23] The major core parameters of Tsuruga-1[24] are shown in **Table 5.1.1**, and the major characteristics of the BWR 8x8-2 MOX fuel assembly[24,25] are shown in **Table 5.1.1 Figure 5.1.1** shows the enrichment distribution among fuel rods in the assembly.[23] The irradiation periods and assembly average burnups of the assembly[22,23] are shown in **Table 5.1.2**. The assembly average burnups of JARX02 is 26.4 GWd/t. **Figure 5.1.2** shows the location of the assembly in the core.[23]

A core average void fraction is 31% for the cores fully loaded with the BWR 8x8-2 UO_2 fuel assemblies.[24] Core-average axial void distribution was estimated from the axial void distribution in the hottest channel given in the reference[24] so that the axially-averaged value coincides with 31%. Table 2.1.4 shows the estimated in-channel void fraction for each axial node. The length of one node is about 15.3 cm.

Table 5.1.1 Major core parameters of Turuga-1[4,24]

Reactor thermal power	1,064 MW
Core flow	17.7×10^3 t/h
Outlet coolant temperature	285 ℃
Core pressure	71.3 kg/cm^2 a
Core	
Number of fuel assemblies	308
Control rod pitch (cm)	30.5
Fuel assembly pitch (cm)*	15.7** (W-W), 14.8* (N-N)
Inside of channel box (cm)	13.4**
Effective fuel length	3.66 m
Equivalent core diameter	3.02 m
Power density	40.6 kW/l

*W-W: control-rod insertion side, N-N: opposite side of control-rod insertion side,

** Round number

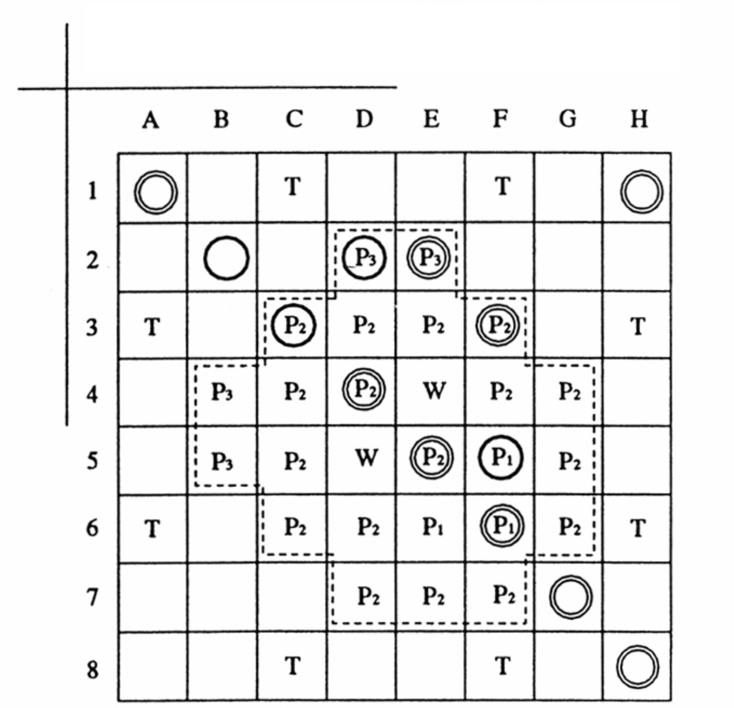
Table 5.1.2 Major characteristics of BWR 8x8-2 MOX fuel assembly[24,25]

1. Fuel assembly	
Lattice	8x8
Fuel rod pitch (mm)	16.3
Number of fuel rods	62
UO_2 fuel rod	38
MOX fuel rod	24
Assembly av. ^{235}U enrichment (wt%)	2.9*
2. Fuel rod	
Out diameter (mm)	12.3*
Thickness of cladding (mm)	0.86*
Cladding material	Zry-2
Pellet outer diameter (mm)	10.3*
Pellet inner diameter (mm)	
UO_2 fuel rod (mm)	0 (solid)
MOX fuel rod (mm)	3.6*
Pellet-cladding gap (mm)	0.24*
Pellet density (%TD)	95*
He pressure (at.)	
UO_2 fuel rod	3*
MOX fuel rod	1
3. Water rod	
Number of water rod	2
Material	Zry
Diameter (mm)	15.0*

*Round number

Table 5.1.3 Irradiation periods and burnups of BWR 8x8-2 MOX fuel assembly[22,23)]

Irradiation cycle (Operating cycle)	1 (16th)	2 (17th)	3 (18th)
Start and end of cycle	1986/7/28 -1987/7/22	1987/10/2 -1988/10/10	1989/3/3 -1990/1/10
Assembly ID discharged for PIE	-	-	JARX02
Assembly average burnups (GWd/t)	-	-	26.4



T: tie rod, W: water rod, Circled rod: under PIE

MOX rod	Fissile wt%	Total Pu wt%	Fissile Pu wt%	Number of rods
P_1	5.31	6.2	4.64	3
P_2	3.82	4.2	3.14	17
P_3	2.94	3.0	2.25	4

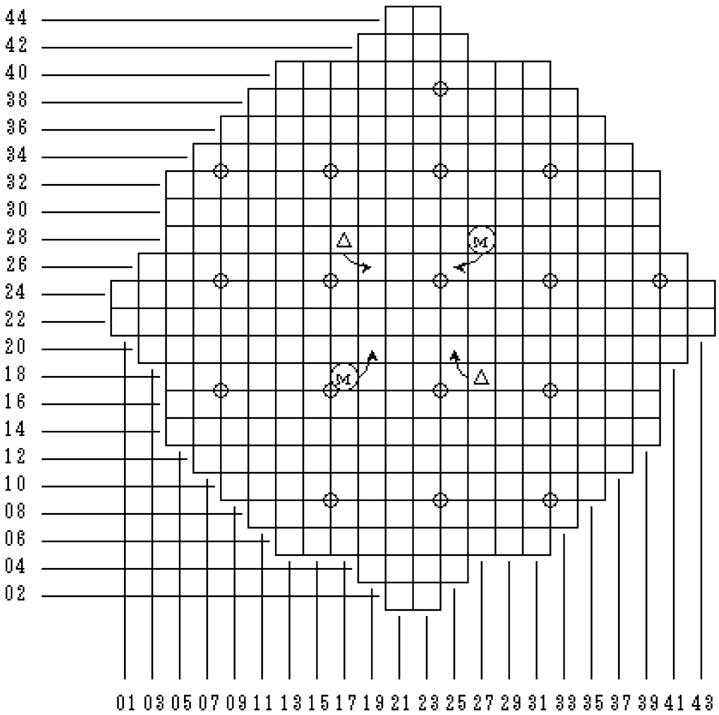
Fissile wt% = $(^{239}Pu+^{241}Pu+^{235}U)/(Pu+U) \times 100$

Total Pu wt% = $Pu/(Pu+U) \times 100$

Fissile Pu wt% = $(^{239}Pu+^{241}Pu)/(Pu+U) \times 100$

Date of Pu composition: July 1986

Fig. 5.1.1 Enrichment distribution of BWR 8x8-2 MOX fuel assembly [23)]



M: MOX fuel assembly of 16th and 17th cycles

→: MOX fuel assembly of 18th cycle

△: UO_2 fuel at symmetric locations of MOX fuel assembly

○: local power monitor

Fig. 5.1.1 Loading location of BWR 8x8-2 MOX fuel assembly[23)]

5.2 Measured Isotopic Inventories of BWR 8x8-2 MOX fuel

5.2.1 Measurement Method

Isotopic inventories were measured for pellet-average samples and pellet local samples. The measurements were performed in the Nippon Nuclear Fuel Development,[22)] and the isotopic dilution mass spectrometry (IDM) was applied to Nd, U, and Pu isotopes with spikes in the forms of ^{150}Nd, natural U, and ^{242}Pu, respectively, and the alpha spectrometry for Am and Cm isotopes.

The measurement errors[22)] are shown in Table 4.2.1 which is based on that the errors in the atomic numbers of ^{235}U, ^{150}Nd, ^{242}Pu in the spikes to be added to the solution samples are 2%, and the reproducible errors of the mass spectrometry is 0.3%.

5.2.2 Measured Burnups and Isotopic Inventories of Samples

The burnups of the samples were determined by the Nd-148 method[6)] with the inventory data of U, Pu and ^{148}Nd, and using the representative values of the effective fission yield of ^{148}Nd, and energy release per fission, 1.73% and 210.0 MeV/fission, respectively, which were determined by the assembly burnup calculations on the 8x8-2 MOX assembly using the continuous energy Monte Carlo burnup calculation code.[9)] The errors of burnups are estimated to be about 6.4%,[9)] which include the errors related to adding the spikes, the mass spectrometry, the alpha spectrometry, the effective fission yield of ^{148}Nd and the effective energy release per fission.

The burnups of the samples were also determined using the values of the effective fission yield of ^{148}Nd, and energy release per fission calculated for each sample. The effect of the neutron absorptions of ^{147}Nd and ^{148}Nd on the determination of the effective fission yields of ^{148}Nd were also taken into account. The method of the burnup determination are given in Reference 1. **Table 5.2.1** shows the determined burnups. The sample A03F1EP05(EP05) is a pellet-local samples taken from a part of the pellet cross section, and the burnups in the table are the corrected ones for pellet-average burnups.

The measured isotopic inventories in the unit of atoms per total U atoms are shown in Tables 5.2.1 for the BWR 8x8-2 MOX fuel assembly.

Table 5.2.1 Measured isotopic inventories and burnups of pellet-average samples for BWR 8x8-2 MOX fuel assemblies

Sample ID		MS1	A03F1EP05(EP05)	No.1	No.2	No.3
Assembly		JARX02	JARX02	JARX02	JARX02	JARX02
Rod location		C3	F6	F5	F5	F5
Type of pellet		MOX	MOX	MOX	MOX	MOX
Enrichment (wt%)		0.7	0.7	0.7	0.7	0.7
Fissile (wt %)*1		3.82	5.31	5.31	5.31	5.31
Fissile Pu (wt%)*2		3.14	4.64	4.64	4.64	4.64
Total Pu (wt%)*3		4.2	6.2	6.2	6.2	6.2
Axial height (mm)*4 (node)		2800-2900 (7.4-6.6)	3135-3136 (5.3-5.2)	About 500 (22.8)	About 1300 (17.6)	About 2800 (7.7)
U isotope (atomic%)	U-234	0.02	0.02	0.02	0.02	0.02
	U-235	0.39	0.36	0.57	0.44	0.37
	U-236	0.07	0.07	0.03	0.07	0.07
	U-238	99.53	99.54	99.38	99.47	99.54
U isotope (mg/gU(t))*5	U-234	0.15	0.24	0.17	0.18	0.18
	U-235	3.83	3.59	5.58	4.36	3.62
	U-236	0.68	0.74	0.30	0.69	0.71
	U-238	995.34	995.43	993.95	994.77	995.49
Pu assay		IDM	IDM	spectrum	spectrum	spectrum
Pu isotope (atomic%)	Pu-238	3.07	4.21	1.78	3.00	3.35
	Pu-239	42.26	36.00	57.88	43.63	35.48
	Pu-240	38.39	43.14	26.69	32.69	37.73
	Pu-241	7.80	7.08	9.96	14.47	15.10
	Pu-242	8.48	9.57	3.70	6.21	8.35
Pu isotope (μg/gU(t))*5	Pu-238	898.4	1483	1178	1076	1202
	Pu-239	12432.3	12731	38465	15704	12779
	Pu-240	11339.2	15318	17822	11811	13685
	Pu-241	2314.4	2525	6677	5249	5468
	Pu-242	2526.2	3426	2490	2265	3046
Total Pu (μg/gU(t))		29510.5	35483	66632	36105	36180
Nd isotope (atomic%)	Nd-142	0.40	0.33	-	-	-
	Nd-143	22.45	22.29	-	-	-
	Nd-144	27.60	27.73	-	-	-
	Nd-145	17.18	17.15	-	-	-
	Nd-146	16.51	16.55	-	-	-
	Nd-148	10.03	10.09	-	-	-
	Nd-150	5.84	5.85	-	-	-
Nd isotope (mg/gU(t))*5	Nd-142	0.0123	0.0131	-	-	-
	Nd-143	0.6940	0.8912	-	-	-
	Nd-144	0.8592	1.1165	-	-	-
	Nd-145	0.5385	0.6953	-	-	-
	Nd-146	0.5211	0.6756	-	-	-
	Nd-148	0.3209	0.4176	0.2092	0.3408	0.4182
	Nd-150	0.1894	0.2454	-	-	-
Am isotope (μg/gU(t))*5	Am-241	2302	-	-	-	-
	Am-243	387	-	-	-	-
Cm isotope (μg/gU(t))*5	Cm-242	0.03	-	-	-	-
	Cm-243	1.9	-	-	-	-
	Cm-244	115	-	-	-	-
Np isotope (μg/gU(t))*5	Np-237	410	-	-	-	-
%FIMA for Nd-148 Y(%)=1.73		2.82	3.62	1.79	2.97	3.62
Burnups (GWd/t) for GWd/t/%FIMA=9.85		27.8	35.3	17.6	29.3	35.7
Burnups (GWd/t) *6		27.7	35.4	-	-	-
Measurement date		2004/2/1	2006/1/1	1990/1*7	1990/1*7	1990/1*7

*1: (Pu239+Pu241+Pu235)/(Pu+U)*100
*2: (Pu239+Pu241)/(Pu+U)*100
*3: Pu/(Pu+U)*100
*4: distance from the shoulder of the upper-end plug of fuel rods
*5: weight ratio to uranium at measurements
*6: burnups determined taking into account effective parameters for each sample (see text)
*7: Norimalization date

5.3 Analysis of burnups and isotopic inventories

The analysis of the burnups and isotopic inventories of part of the samples were performed by tracking the irradiation history of node power and in-channel void fraction which were provides by the plant utility from a core monitoring system. The burnup calculations with the infinite assembly model were carried out using the collision probability module (PIJ) of 107 neuron energy groups of the general purpose nuclear calculation code system SRAC,[10)] and the continuous energy Monte Carlo burnup calculation code MVP-BURN[11)] based on the nuclear data library, JENDL-3.3.[20)] The geometrical dimensions and atomic number densities of the assembly were taken from the design base specifications of the fuel vender. The temperatures of the fuel cladding and the moderator were set to be 285 ℃.

Table 5.3.1 shows the results of the burnups of the pellet-average samples for the five-cycle irradiated BWR 8x8-2 MOX assembly by the analysis of SRAC and MVP-BURN compared with the measurements (Burnups*6 in Table 5.2.1). The calculated systematically overestimate the measurements.

In order to eliminate the errors in isotopic inventories caused by the difference of the calculated sample burnups from the measurement for obtaining reliable information on the accuracy of lattice calculations, analyses have been performed so that, modifying power history, the calculated sample burnups coincide with the measurement for each sample. The calculated inventories of MVP-BURN are shown in **Tables 5.3.2** compared with the measurements. The table also shows the C/Es of Pu isotopes in the unit of at%. **Figure 5.3.1** schematically shows the C/Es in atomic percents for Pu isotopes and mg/gU(t) for Pu-total. The two sample shows an opposite trend in the C/Es. Further study on the calculations and the measurements is necessary to inquiry the observed discrepancies between the calculated and measured inventories.

Table 5.3.1 Burnups of pellet-average samples for BWR 8x8-2 MOX fuel assembly compared with calculations

(Unit: GWd/t)

Sample ID	Rod-node	Measurement	SRAC	SRAC (C/E)	MVP	MVP (C/E)
MS1	C3	27.7	36.6	1.321	36.9	1.332
A03F1EP05(EP05)	F6	35.4	39.4	1.113	37.5	1.059

Table 5.3.2 Calculated inventories of MVP-BURN and comparison with the measurements for BWR 8x8-2 MOX fuel assembly

Sample ID Rod-node	MS1 C3-7			A03F1EP05(EP05) F6-7		
Isotope	Calculation (mg/gU(t)	C/E	C/E (at% base)	Calculation (mg/gU(t)	C/E	C/E (at% base)
U-234	0.12	0.78		0.17	0.71	
U-235	3.79	0.99		4.09	1.14	
U-236	0.64	0.94		0.65	0.88	
U-238	995.45	1.00		995.08	1.00	
Pu-238	0.5952	0.66	0.71	0.89	0.60	0.50
Pu-239	10.6265	0.85	0.91	18.50	1.45	1.19
Pu-240	11.6289	1.03	1.09	17.20	1.12	0.92
Pu-241	2.2250	0.96	1.02	3.14	1.25	1.02
Pu-242	2.6307	1.04	1.11	3.48	1.01	0.83
Pu-Total	27.7063	0.94		43.22	1.22	
Am-241	2.6691	1.07		-	-	
Am-243	0.4778	1.13		-	-	
Cm-242	2.07E-05	0.63		-	-	
Cm-243	1.74E-03	0.87		-	-	
Cm-244	0.0760	0.61		-	-	
Nd-142	0.008	0.65		0.010	0.57	
Nd-143	0.717	1.03		0.937	1.04	
Nd-144	0.869	1.01		1.075	1.00	
Nd-145	0.558	1.04		0.707	1.04	
Nd-146	0.521	1.00		0.669	0.99	
Nd-148	0.328	1.02		0.420	1.02	
Nd-150	0.191	1.01		0.247	1.01	

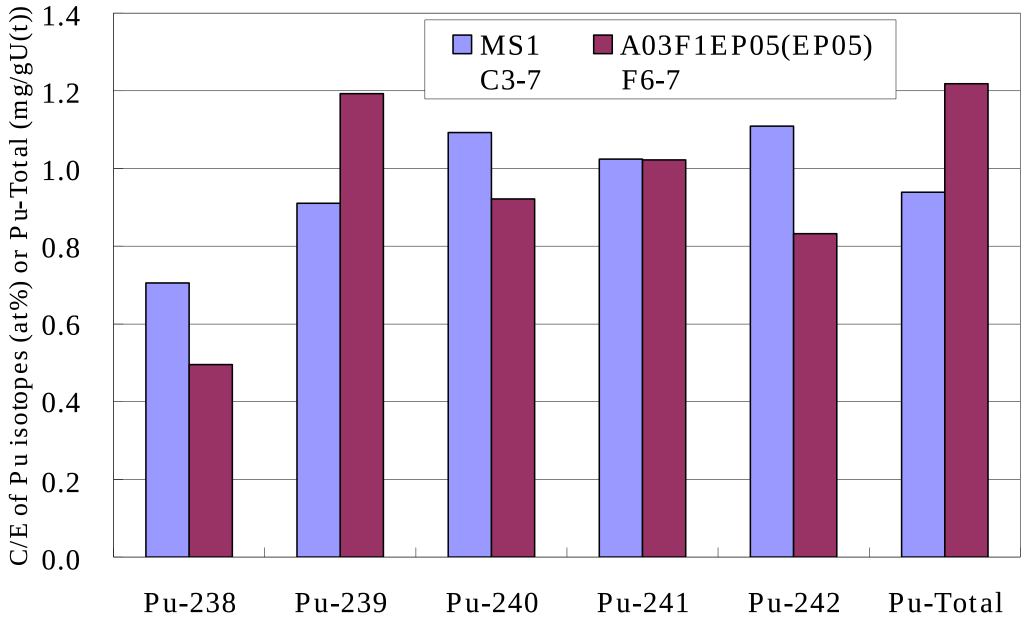


Fig. 5.3.1 Comparison of Pu isotopic inventories calculated by MVP-BURN with the measurements in atomic percents for Pu isotopes and mg/gU(t) for Pu-total on 8x8-2 MOX fuel assembly

6. Summary

The report presents the measurement data of the isotopic inventories of the fuel pellet samples taken from the 8x8-4, 9x9-9, and 9x9-7 UO_2 fuel assemblies, and 8x8-2 MOX fuel assembly which were irradiated in the Japanese commercial BWR plants. The measured data are mainly on U, Pu, and Nd isotopes. The burnups of the samples were obtained based on the Nd-148 method and are shown with the inventory data.

The specifications of the cores and fuel assemblies are also presented with the irradiation histories of the fuel assemblies and the typical axial distributions of in-channel void fractions.

The report also presents the analysis results based on the infinite assembly calculations with the nuclear analysis codes SRAC and MVP-BURN, which have been developed by the Japan Atomic Energy Agency and are publicly available.

7. Reference

1) T. Yamamoto, Y. Kanayama, "Lattice Physics Analysis of Burnups and Isotope Inventories of U, Pu, and Nd of Irradiated BWR 9×9-9 UO_2 Fuel Assemblies," *J. Nucl. Sci. and Technol.*, **45**[6], 547 (2008).

2) T. Yamamoto, Y. Kanayama, "Nuclear Analysis of PIE Data of Irradiated BWR 8×8-2 and 8×8-4 UO_2 Fuel Assemblies," *J. Nucl. Sci. and Technol.*, **45**[11], 1193 (2008).

3) *Application for Establishment Permit of Modification of Unit 2 of Fukushima power station 2*, Tokyo Electric Power Company, Sept. 29, 1986 (1986), [in Japanese].

4) *Boiling Water Reactors, Geometry of Cores and Fuel Assemblies*, TLR-056, Toshiba corporation (1998), [in Japanese].

5) *Boiling Water Reactors, 9x9 Fuel*, GLR-001, Global Nuclear Fuel - Japan (2003).

6) *Safety Experiments of High Burnup Fuel (BWR High Burnup Fuel, Comprehensive Evaluation 2)*, Nuclear Power Engineering Corporation, Mar., 2002 (2002), [in Japanese].

7) *Application for Establishment Permit of Modification of Unit 2 of Fukushima power station 2*, Tokyo Electric Power Company, May 9, 1991 (1991), [in Japanese].

8) *Standard Test Method for Atom Percent Fission in Uranium and Plutonium Fuel (Neodymium-148 Method)*, ASTM E321-96 (2005).

9) *Reevaluation of measured burnups of irradiated BWR fuel*, JNES-SS-0807, Japan Nuclear Energy Safety Organization (JNES), (2008), [in Japanese].

10) K. Okumura, K. Kaneko, K. Tsuchihashi, *SRAC95: General Purpose Neutronics Code System*, JAERI-Data/Code 96-015 (JAERI), (1996), [in Japanese].

11) K. Okumura, T. Mori, M. Nakagawa *et al.*, "Validation of a continuous-energy Monte Carlo burn-up code MVP-BURN and its application to analysis of post irradiation experiment," *J. Nucl. Sci. Technol.*, 37[2], 128 (2000).

12) T. Nakagawa, K. Shibata, S. Chiba *et al.*, "Japanese evaluated nuclear data library version 3 revision-2," *J. Nucl. Sci. Technol.*, **32**, 1259 (1995).

13) Y. Nakahara, K. Suyama, J. Inagawa et al., "Nuclide Composition Benchmark Data Set for Verifying Burnup Codes on Spent Light Water Reactor Fuel," *Nucl. Technol.*, **137**, 111 (2002).

14) Y. Hirano, Y. Mozumi, K. Kamimura *et al.*, "Irradiation Characteristics of BWR High Burn Up 9×9 Lead Use Assemblies," *Proc. Int. Conf. on International Topical Meeting on Light Water Reactor Fuel Performance*, Kyoto, Japan, October 2-6, 2005 (2005).

15) *Demonstration Program on Reliability of High Burnup 9x9 Fuel*, Japan Nuclear Energy Safety Organization, December, 2007 (2007), [in Japanese].

16) *Application for Establishment Permit of Modification of Unit 1 and 2 of Fukushima power station 2*, Tokyo Electric Power Company, May 26, 1994 (1994), [in Japanese].

17) *Boiling Water Reactors, 9x9 Fuel*, NLR-15 (Rev. 3), Nuclear Fuel Industries, (2003).

18) *Application for Establishment Permit of Modification of Unit 1 and 2 of Fukushima power station 2*, Tokyo Electric Power Company, Feb. 28, 1997 (1997), [in Japanese].

19) Y. Nakahara, R. Nagaishi, N. Kohno *et al.*, "Evaluation of Effective Fission Yield of ^{148}Nd," *The Atomic Energy Society of Japan, 1996 Fall Meeting*, F23, Sept. 23-25, 1996 (1996), [in Japanese].

20) K. Shibata, T. Kawano, T. Nakagawa *et al.*, "Japanese Evaluated Nuclear Data Library Version 3 revision-3: JENDL3.3," *J. Nucl. Sci. Technol.*, **39**[11], 1125 (2002).

21) *Demonstration Program on Reliability (High Burnup 9x9 Fuel)*, Nuclear Power Engineering Corperation, October, 2003 (2003), [in Japanese].

22) *Demonstration Program on Reliability of MOX Fuel for One-Third MOX Lorded Cores*, Japan Nuclear Energy Safety Organization, June, 2004 (2004), [in Japanese].

23) M. Ichikawa, S. Tottori, K. Yamate *et al.*, "Results of Demonstration Program on Utilization and Post-irradiation Examination of MOX Fuel in Japan," *J. At. Energy Soc. Japan*, **39**[2], 93(1997), [Japanese].

24) *Application for Establishment Permit of Modification of Unit 1 of Tsuruga power station*, Japan Atomic Power Company, Dec. 1, 1982 (1982), [in Japanese].

25) *Boiling Water Reactors, High Burnup 8x8 Fuel*, HLR-041, Hitach, Ltd., (1989).