MOX FUEL DESIGN AND DEVELOPMENT CONSIDERATION

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Abstract

Pu thermal utilization in Japan will be realized in several plants in late 1990's, and will be expanded gradually. For this target, adequacy of methods for MOX fuel design, nuclear design, and safety analysis has been evaluated by the committee of competent authorities organized by government in advance of the licensing application. There is no big difference of physical properties and irradiation behaviors between MOX fuel and UO_2 fuel, because Pu content of MOX fuel for Pu thermal utilization is low. The fuel design code for UO_2 fuel will be applied with some modifications, taking into account of characteristic of MOX fuel. For nuclear design, new code system is to be applied to treat the heterogeneity in MOX fuel assembly and the neutron spectrum interaction with UO_2 fuel more accurately. For 1/3 MOX fueled core in three loop plant, it was confirmed that the fuel rod mechanical design could meet the design criteria, with slight reduction of initial back-filling pressure, and with appropriate fuel loading patterns in the core to match power with UO_2 fuel. With the increase of MOX fuel fraction in the core, control rod worth and boron worth decrease. Compensating the decrease by adding control rod and utilizing enriched B-10 in safety injection system, 100% MOX fueled core could be possible. Up to 1/3 MOX fueled core in three loop plant, no such modification of the plant is necessary. The fraction of MOX fuel in PWR is designed to be less than 1/3 in the present program. In order to improve Pu thermal utilization in future, various R & D program on fuel design and nuclear design are being performed, such as the irradiation program of MOX fuel manufactured through new process to the extent of high burnup.

1. INTRODUCTION

Japanese government announced a long-term plan concerning nuclear utilization in June, 1994, where the prospect for Pu thermal utilization is also described; i.e., Pu thermal utilization will be realized in several plants in late 1990's, and will be expanded to approximately ten plants around 2000. It is foreseen more than ten plants of Pu thermal utilization in the next period from 2000 to 2010.

For this target, adequacy of methods for MOX fuel mechanical design, core nuclear design, and safety analysis has been evaluated by the committee of competent authorities organized by government in advance of the licensing application.

There is no big difference of physical properties and irradiation behaviors between MOX fuel and UO_2 fuel, because Pu content of MOX fuel for Pu thermal utilization is low compared to MOX fuel for fast reactor. Taking into account of characteristic of MOX fuel, the fuel mechanical design code for UO_2 fuel will be applied with some modification. For nuclear design, the new code system will be used to treat the heterogeneity in MOX fuel assembly and the neutron spectrum interaction with UO_2 fuel more accurately.

This paper describes the status and future plan of fuel design, core design, and research and development for MOX fuel.

2. OUTLINE OF PWR MOX FUEL ASSEMBLY

All fuel rods within a PWR MOX fuel assembly are made of MOX, with the distribution of Pu content of fuel rod to reduce local power peaking at the periphery of the assembly. An example of 17 x 17 type fuel is shown in Fig.1.

The average Pu fissile content of MOX fuel assembly is designed so that the reactivity of MOX fuel is equivalent to that of current standard UO_2 fuel for Japanese PWR's whose enrichment is 4.1wt%. Typical plutonium fissile and total content as a function of fissile isotope fraction is shown in Fig.2. The Pu fissile content is evaluated approximately 6wt% for the Pu isotopic composition described in Fig.2, which is considered typical of Pu recovered from PWR spent fuel. Maximum assembly average burnup of UO_2 fuel is 48GWd/t, while that of MOX fuel assembly is designed lower to be 45GWd/t. It is due to large peak to average burnup ratio in MOX fuel assembly.

The other structure of MOX fuel assembly is almost same as that of UO_2 fuel assembly to be compatible with UO_2 fuel assemblies in the core. And also, from a view point of keeping reliability as

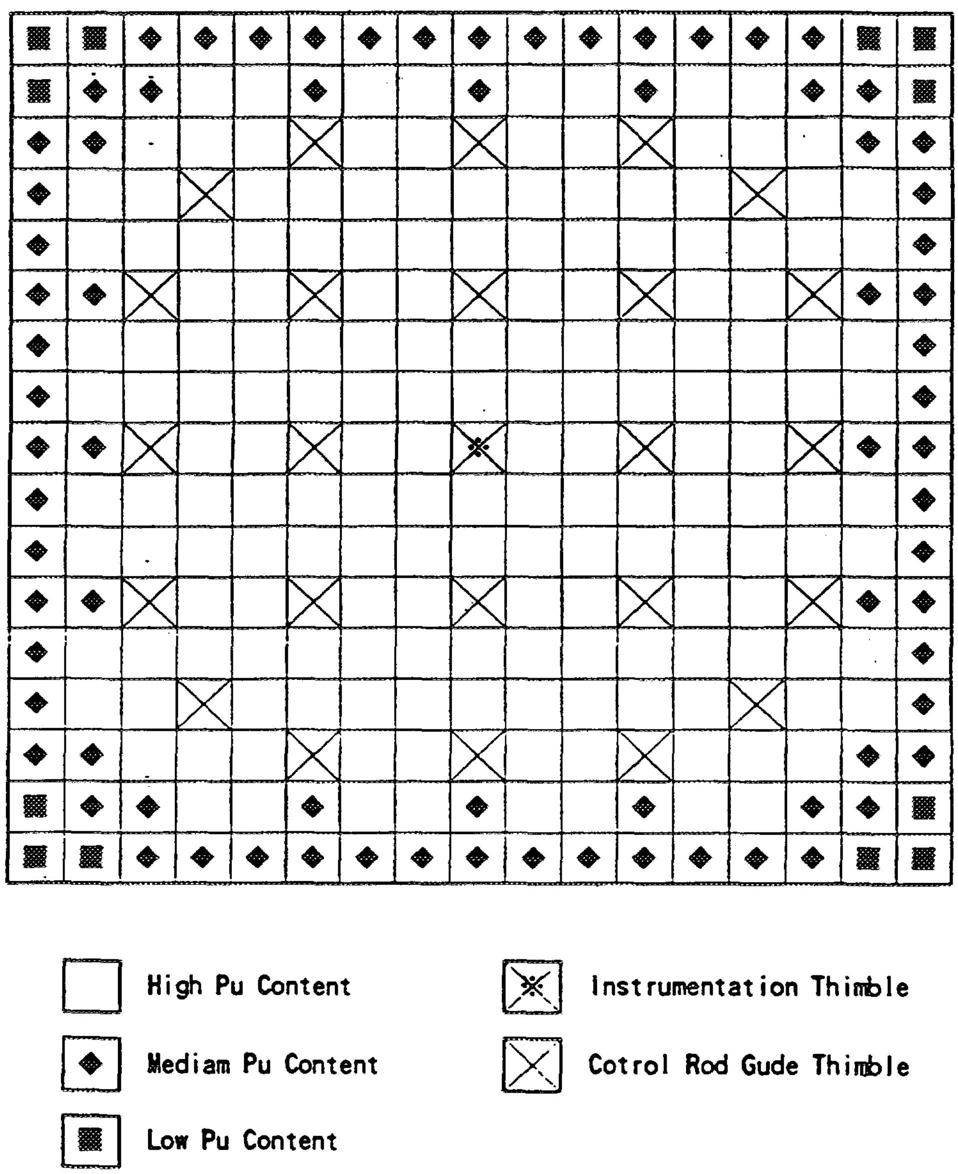

FIG. 1. Pu Content Distribution in PWR MOX Fuel Assembly

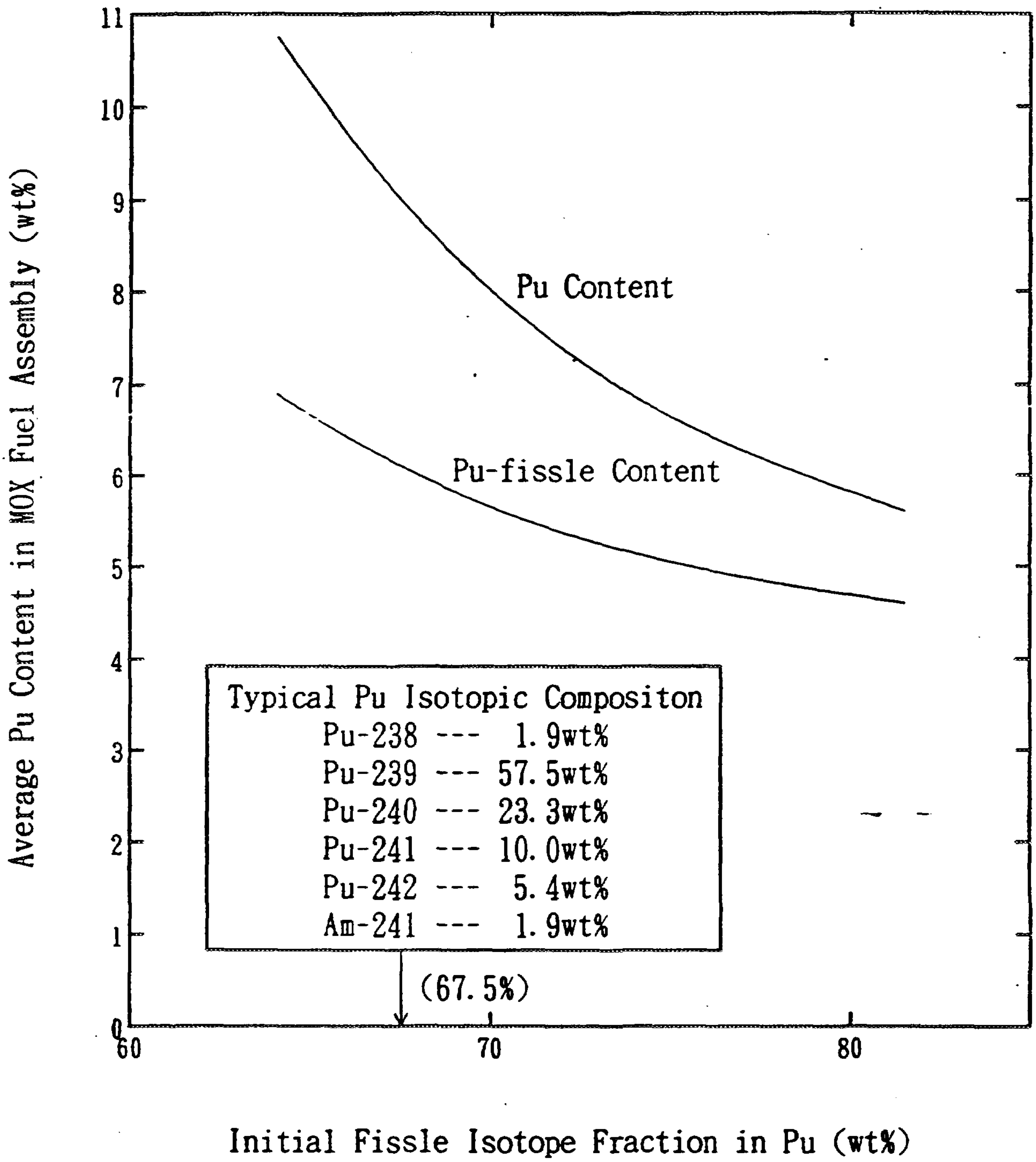

FIG. 2. Typical Pu Content vs Fissile Isotope Fraction

high as at present, it is planned that all components except for MOX pellets will be supplied by domestic vendors to overseas manufactures, who will fabricate MOX fuels for certain time period. The basic description of PWR MOX fuel is shown in TABLE 1.

3. FUEL MECHANICAL DESIGN OF MOX FUEL

3.1 MOX CHARACTERISTICS AND DESIGN CONSIDERATIONS

PuO_2 and UO_2 form solid solution, and material properties of MOX are very similar to those of UO_2 to approximately 13wt% of Pu content. Two of the most important properties, melting point and thermal conductivity are reduced by only several percent [1]. Thermal conductivity is illustrated in Fig. 3 with UO_2 design model, and small reduction appears in MOX fuel. Thermal expansion of MOX fuel is very close to that of UO_2.

TABLE I **Fuel Description of PWR**
(Typical Specification for 17 x 17 Fuel Assemblies)

		UO_2 Fuel	MOX Fuel
1. Fuel Assemblies			
Rod Array		17 x 17	17 x 17
Active Fuel Height,	m	~3.6	~3.6
Rods per Assemblies		264	264
U-235 Enrichment,	wt%	4.1 (2.6 in Gd fuel)	~0.2
Average Pu-fissile Content,	wt%	--	~6.1
Maximum burnup,	GWd/t	48	45
2. Fuel Rods			
Fuel Pellets Diameter,	mm	8.19	8.19
Fuel pellets Material		UO_2 UO_2 - Gd_2O_3	UO_2 - PuO_2
Clad Outside Diameter,	mm	9.50	9.50
Clad Thickness,	mm	0.57	0.57
Clad Material		Zircaloy-4	Zircaloy-4
Pellet--Clad Diameter Gap,	mm	0.17	0.17

Fission gas release data are illustrated in Fig.4 [2,3,4,5]. It is said that fission gas release of MOX fuel is larger than that of UO_2 fuel due to inhomogeneity and possibly microstructure differences. Particularly, MOX pellet which were manufactured through old process have considerable amount of Pu enriched spots. These spots burn faster than surrounding MOX matrix, and the temperature of these area is also higher and therefore more fission gas is released. However fission gas release of current MOX fuel which were manufactured through advanced process reduced to similar level of UO_2 fuel. In fuel rod design, it is conservatively assumed that fission gas release from MOX fuel is higher than that of UO_2 in order to bound old data also. These data were analyzed by fuel design code FINE [6], which is one of Japanese codes developed by Mitsubishi. Comparison of predictions with measurement are illustrated in Fig. 5 and shows good agreement.

Fuel stack length changes are illustrated in Fig. 6 [2,4] which shows that MOX fuel data stay within variation of UO_2 fuel data. From these data, it can be said that addition of PuO_2 hardly affects on pellet densification and swelling, and the behavior is rather dependent on UO_2 characteristics.

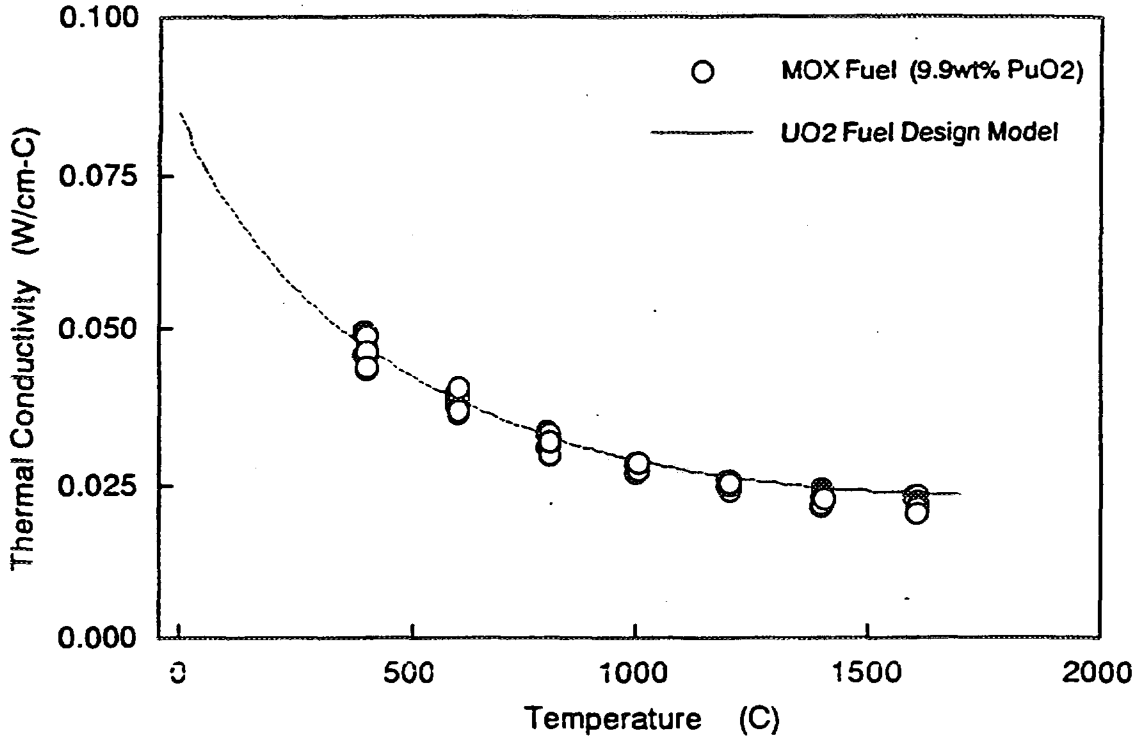

FIG. 3. Thermal Conductivity of MOX Pellet

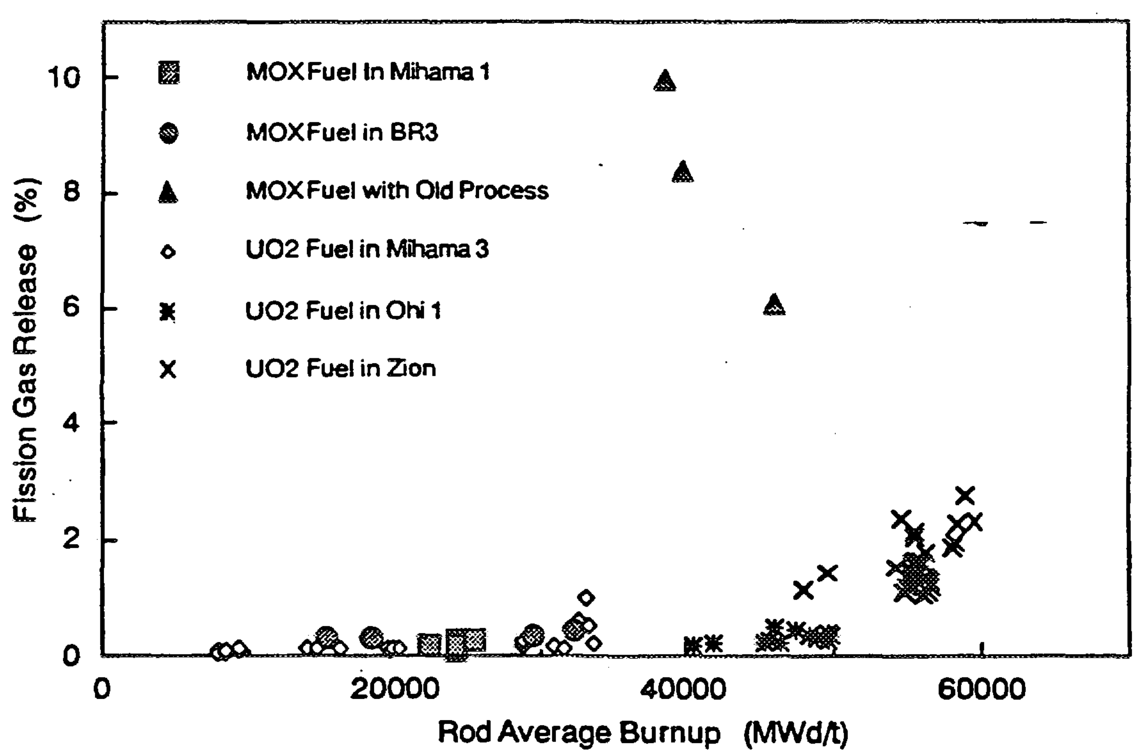

FIG. 4. Fission Gas Release

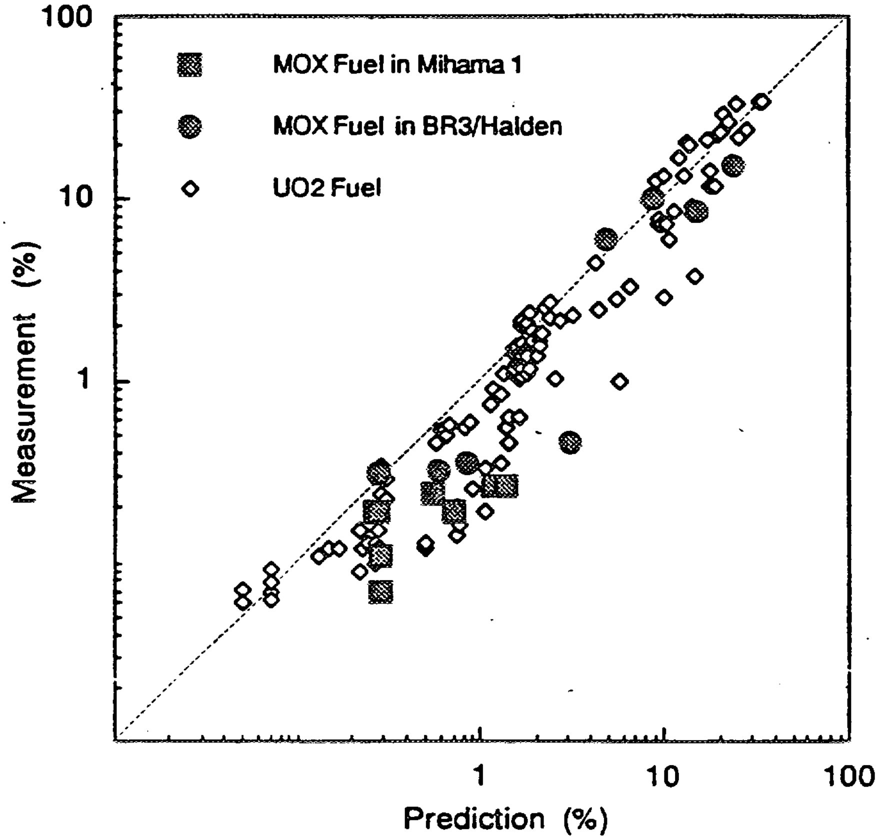

FIG. 5. Comparison of Fission Gas Release

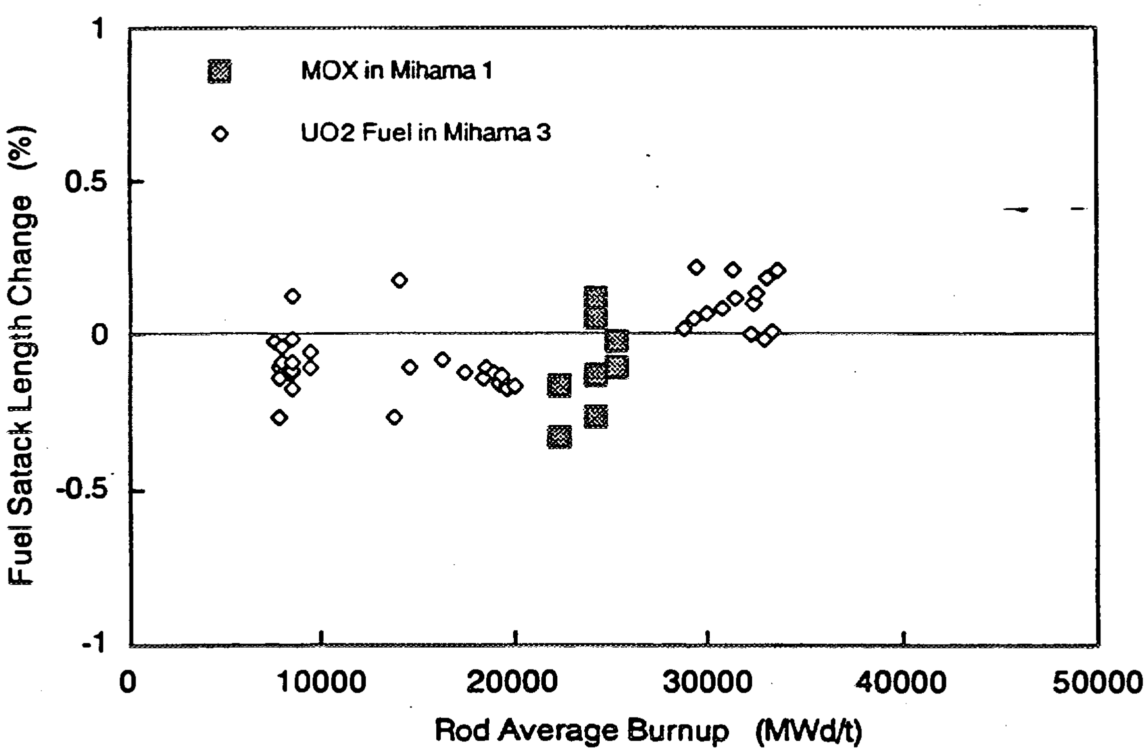

FIG. 6. Fuel Stack Length of Changes

More Helium is produced in MOX fuel through α-decay. This Helium production and release from pellet increase fuel rod internal pressure. This is modelled in the fuel design code.

Based on these considerations, the fuel design code can fairly predict fuel rod internal pressure and fuel temperature as illustrated in Fig. 7 and Fig. 8, and both of them show good agreement.

3.2 FUEL DESIGN FEATURES

Nuclear characteristics of MOX fuel is slightly different from that of UO_2. Reactivity of MOX fuel decreases slower than that of UO_2, and then power of MOX fuel remains higher at high burnup. This also causes higher internal pressure in MOX fuel. Considering also the characteristics of MOX fuel as described in the last section, fuel rod internal pressure is important. As the results of the analysis for 1/3 MOX in three loop plant, taking into account the fuel and nuclear characteristics of MOX, it was confirmed that fuel rod internal pressure could meet the design criteria with slight reduction of initial back-filling pressure, and with appropriate fuel assembly loading patterns in the core to match power with UO_2 fuel.

4. NUCLEAR DESIGN OF MOX FUEL

4.1 METHOD OF NUCLEAR DESIGN

For nuclear design of MOX fueled core, the new code system has been used. PHOENIX--P/ANC [7, 8] is one of the Japanese code systems applied by Mitsubishi. The accuracy of the new code system has been verified by comparing the calculation results with the measurement of critical experiments and core operating data.

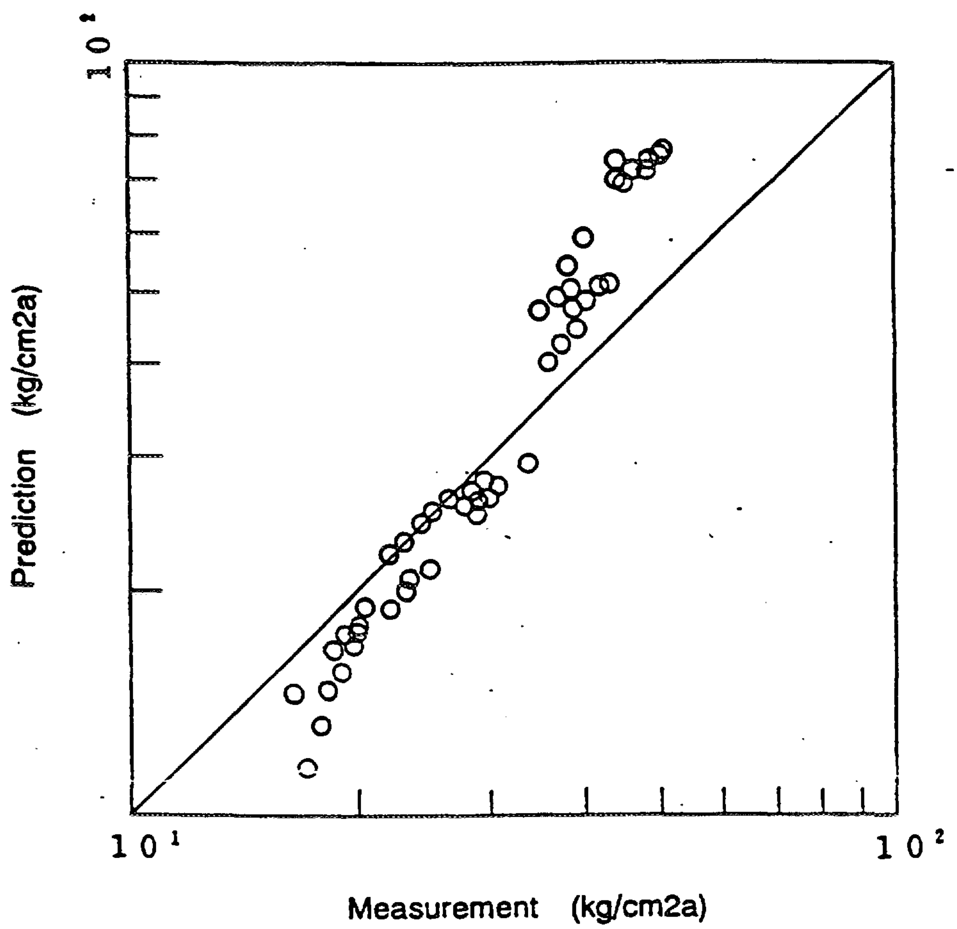

FIG. 7. Comparison of Rod Internal Pressure

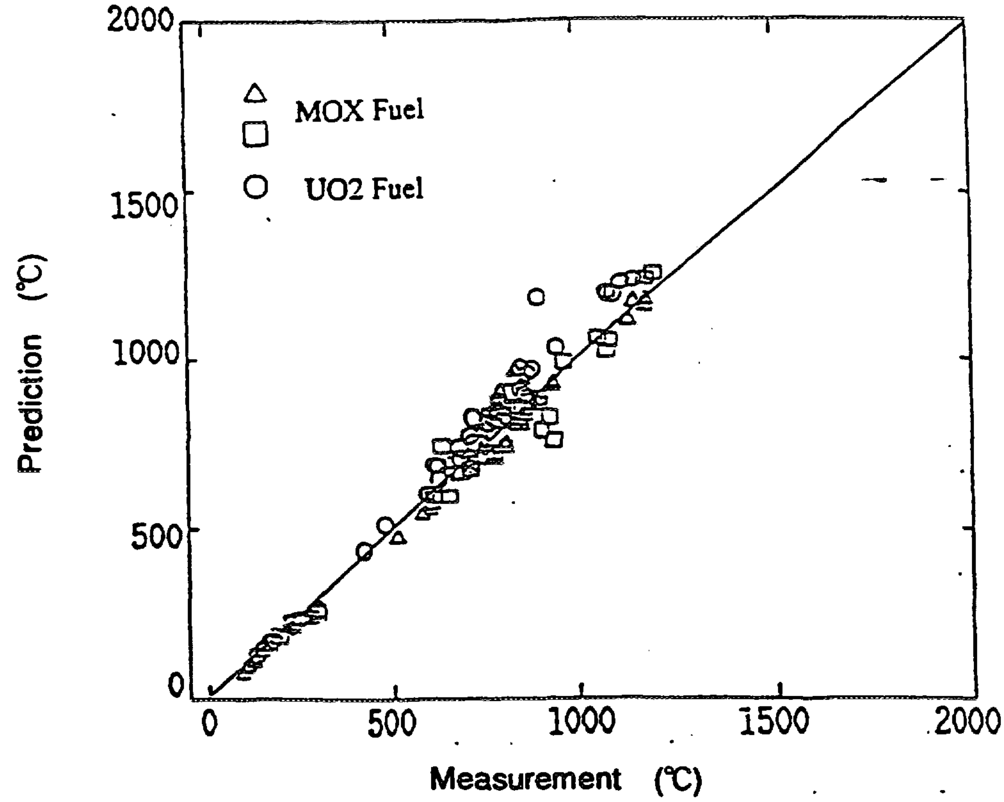

FIG. 8. *Comparison of Fuel Centre Temperature*

4.2 CHARACTERISTICS OF MOX FUELED CORES

With the increase of MOX fuel fraction, some core physics parameters change from UO_2 fueled core. Typical variation of these parameters are shown in Fig. 9. The reactivity effect of control rod and boron in the coolant decrease with MOX fuel fraction. To 1/3 MOX fuel fraction, degradation of control rod worth can be mitigated by loading MOX fuel avoiding the control rod position. To increase MOX fuel fraction more than 1/3, it will be necessary to increase the number of control rods or change the rod absorber material from Ag-In-Cd. Decrease of boron worth can be compensated by increasing the boron concentration in safety injection system to 1/3 MOX. To increase MOX fuel fraction more than 1/3, it will be necessary to utilize enriched B-10 in safety injection system. Moderator temperature coefficient becomes more negative, causing larger reactivity insertion during cool down event, but that can be overcome by strengthening the ability of control rod and safety injection system as described above. Delayed neutron fraction becomes smaller with the increase of MOX fuel fraction. But the decrease of control rod worth and power peaking factor will mitigate the results of reactivity insertion event.

Comparison of typical core design parameters of UO_2 fueled core, 1/3MOX, and 100% MOX fueled core for three loop PWR plant is shown in Table II. For 1/3MOX fuel core, fuel assemblies with three plutonium fissile contents as shown in Fig.1 are used to match with adjacent UO_2 fuel and reduce power peaking factor in the assembly. On the other hand, single plutonium fissile content in the fuel assembly can be used for 100% MOX core. Many burnable poison rods are necessary to adjust power distribution especially for fuel rod internal pressure in 1/3MOX fueled core, while no burnable poison is needed in 100% MOX fueled core. Addition of control rods and utilization of enriched B-10 in refueling water storage tank and boron injection tank, will make 100% MOX fueled core possible.

Up to 1/3MOX fueled core in three loop plant, no such modification of the plant is necessary. Consequently, the fraction of MOX fuel in PWR is designed to be less than 113 in the present program.

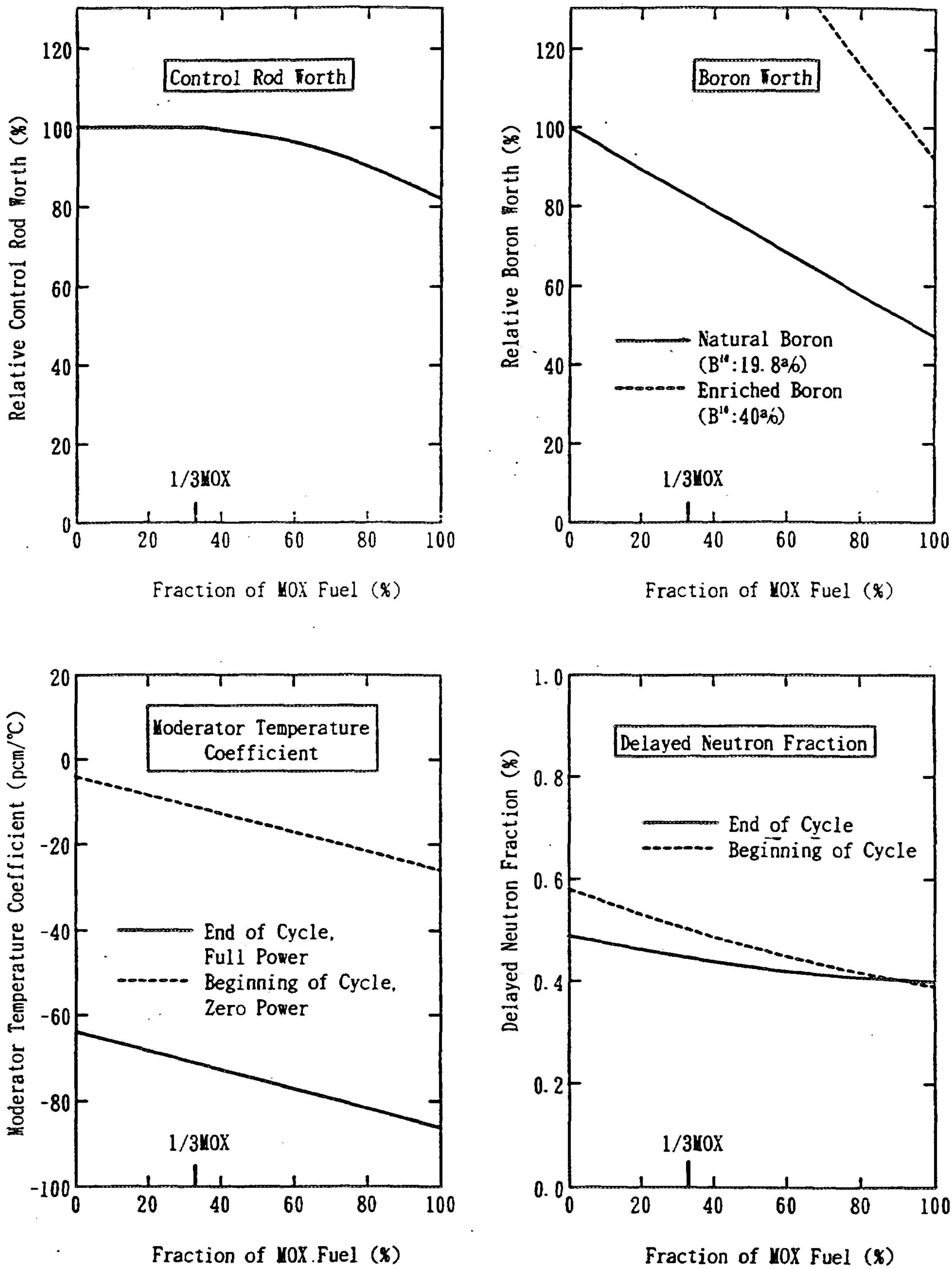

FIG. 9. Dependence of Nuclear Design Parameters on Fraction of MOX Fuel in the Core

TABLE II Characteristics of MOX Fueled Core and $U0_2$ Fueled Core

	$U0_2$Core	1/3 MOX Core	1/1 MOX Core
Number of total fuel assemblies	157	157	157
Number of fresh fuel assemblies			
UO_2 Fuel	24	28	0
UO_2 Fuel with gadolinia	36	16	0
MOX Fuel	0	16	60
Total	60	60	60
Number of burnable poison assemblies	0	52	0
Number of Pu fissile content in MOX fuel assemblies	--	3	1
Cycle burnup [GWd/t]	15.2	15.2	15.2
Maximum burnup of fuel assemblies			
$U0_2$ fuel [Gwd/t]	48	48	--
MOX fuel [Gwd/t]	--	45	48
Number of control rod clusters			
In the case of Ag-In-Cd rod	48	48	~68
In the case of B4C rod with 90% B10	--	--	~60
Boron concentration in refueling water storage tank			
In the case of natural boron [ppm]	2200	~3000	~6000
In the case of 60% B10 [ppm]	--	--	~2000
Boron concentration in boron injection tank			
In the case of natural boron [ppm]	20000	-21000	~60000
In the case of 60% B10 [ppm]	--	--	~20000

5. FUTURE DESIGN CONSIDERATION OF PWR MOX FUEL

It is considered very important to make further efforts on MOX fuel R & D in order to enhance Pu thermal utilization in more flexible and economical design in future. The following design considerations are now being investigated and planed.

5.1 REVISION OF FISSION GAS RELEASE (FGR) MODEL

The present data base of MOX fuels includes the data of MOX fuels manufactured through the old mechanical blending processes whose homogeneity of Pu and U is poor and FGR is large.

This fact causes high FGR on the design of MOX fuel. However, the MOX fuels to be used in Japanese PWR's will be manufactured through new processes, which gives the improved homogeneity of Pu and U and are expected to have lower FGR. The irradiation data of MOX fuels manufactured through the new processes are now being accumulated to revise FGR model.

It is hereby important to obtain the data at high bumup and high power, considering the design conditions for the further MOX fuel utilization. The data for the peak pellet burnup of 60 to 70 GWd/t and the data of the irradiation power level of 300W/cm will be accumulated. For example, two program of MOX fuel irradiation is now in progress in Halden reactor. One is the irradiation program of small diameter of MOX fuel to achieve high burnup data in near future. And the another is the irradiation program of Japanese specification MOX fuel to demonstrate the integrity of Japanese MOX fuel.

5.2 FURTHER REDUCTION OF INITIAL He PRESSURE

Further reduction of initial He pressure will be investigated from a view point of the influence on fuel rod elongation and bowing.

5.3 INCREASE OF PLENUM VOLUME

The increase of plenum volume is an another countermeasure to decrease the fuel rod internal pressure. The additionnal length of plenum will be determined taking into account of local power peaking at the end of fuel stack due to thermal neutron current from surrounding UO_2 region.

5.4 ACCUMULATION OF MOX RAMPING DATA

Although PCI resistance of MOX fuel is expected to be better than that of UO_2, considering available ramping test results and its characteristics of high creep rate, it is still necessary to enhance the ramping test data of MOX fuel, especially at high burnup region.

5.5 LARGE GRAIN MOX FUEL PELLET

In the bumup extension of UO_2 fuels, large grain pellet to reduce fission gas release has been developed. The investigation of large grain MOX pellet will be considered in future,

5.6 GADOLINIA MOX FUEL ASSEMBLY

If gadolinia fuel rods are implemented in MOX fuel assemblies, the flexibility of MOX core design will be increased. There are two designs of gadolinia MOX fuel assembly. One uses UO_2 - Gd_2O_3 fuel rods in a MOX fuel assembly, while the other uses MOX - Gd_2O_3 fuel rods. The nuclear characteristics of these design and core design as well as the possibility of the manufacturing of MOX - Gd_2O_3 pellet will be studied.

5.7 REDUCTION OF PU CONTENT VARIETIES IN ASSEMBLY

It is very effective to decrease the variety of Pu content to reduce the fabrication cost of MOX fuel. The feasibility will be considered in future within the limit of the present fuel and core design.

6. CONCLUSION

The adequacy of the methods for MOX fuel mechanical design, nuclear design, and safety evaluation was authorized recently for the practical Pu thermal utilization in PWR's. Up to 1/3 MOX fueled core, modification of the plant is not necessary. Consequently, the fraction of MOX fuel in PWR is designed to be less than 1/3 in the present program. For further improvement of MOX fuel in PWR, Japanese PWR utilities and vendors are now continuing R & D works of MOX fuel irradiation programs at experimental reactors to study the MOX fuel performance such as the fission gas release behavior and PCI resistance.

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