TECHNOLOGY DEVELOPMENTS FOR JAPANESE BWR MOX FUEL UTILIZATION

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Abstract

The Long-Term Program for Research, Development and Utilization of Nuclear Energy established by the Atomic Energy Commission of Japan asserts that Japan will promote systematic utilization of MOX fuel in LWRs. Based on this Japanese nuclear energy policy, we have been pushing development of MOX fuel technology aimed at future full scale utilization of this fuel in BWRs. In this paper, the main R & D topics are described from three subject areas, MOX core and fuel design, MOX fuel irradiation behavior, and MOX fuel fabrication technology. For the first area, we explain the compatibility of MOX fuel with UO_2 core, the feasibility of the full MOX core, and the adaptability of MOX design methods based on a mock-up criticality experiment. In the second, we outline the Tsuruga MOX irradiation program and the DOMO program, and suggest that MOX fuel behavior is comparable to ordinary BWR UO_2 fuel behavior. In the third, we examine the development of a fully automated MOX bundle assembling apparatus and its features.

1. INTRODUCTION

The Long-Term Program for Research, Development and Utilization of Nuclear Energy revised by the Atomic Energy Commission of Japan June, 1994 asserts that Japan will promote systematic utilization of MOX fuel in light water reactors (LWRs). This utilization is important from the viewpoint of establishment of the technology and systems needed for nuclear fuel recycling on a practical scale as a prerequisite for future practical use of fast breeder reactors (FBRs). Furthermore in advancing the utilization of MOX fuel in LWRs, the program suggests that for the time up to commercial commissioning of FBRs, it is necessary to undertake use of MOX fuel in LWRs on an appropriate scale which considers the scale of nuclear fuel recycling. Specifically, the program indicates it is appropriate to start using MOX fuel in a few LWRs in the second half of the nineties and to increase the number of such reactors in a planned manner, but with some flexibility, to about ten by the year 2000 and over ten by 2010.

In line with this Japanese nuclear energy policy, we have been pushing development of MOX fuel technology aimed at future full scale utilization of MOX fuel in boiling water reactors (BWRs). The technology developments consist of a wide range of technical areas, such as MOX core and fuel design, MOX fuel irradiation behavior, MOX fuel fabrication technology, design and performance codes for MOX fuel, MOX fuel bundle handling and shipping, and MOX bundle inspection technique, etc. Some of these research and development (R & D) programs have been performed with support from electrical utilities or the Japanese government.

In this paper we select three main areas, the MOX core and fuel design, the MOX fuel irradiation behavior, and the MOX fuel fabrication technology, and describe the main topics of the R & D programs in these areas.

2. DESIGN STUDY OF BWR-MOX CORE AND FUEL

2.1 BASIC FEATURES OF MOX FUEL IN BWR

For the design of MOX fuel and MOX loaded core, good understanding of the basic characteristics of MOX and adequate design considerations are required because of the differences between UO_2 and MOX. The MOX behavior in the LWRs is quite similar to that for UO_2 and just a few design considerations are enough to ensure proper utilization of MOX fuels in BWRs without significant modification in reactor plant designs. Compared to UO_2, the major features of MOX are summarized as follows.

(1) Nuclear characteristics

a. More negative reactivity coefficient in void, Doppler and moderator temperature.
b. Smaller reactivity control worth in the control rod and for burnable poison.
c. Possible higher power peaking in the MOX fuel rod which is adjacent to the soft neutron spectrum area.
d. Shorter prompt neutron lifetime and smaller fraction of delayed neutron.
e. Less reactivity reduction during burnup.
f. Slight differences in He generation and fission yield.

(2) Material properties and fuel irradiation behavior

a. Lower pellet melting point and pellet thermal conductivity as a function of Pu content.
b. Higher pellet creep rate as a function of Pu content.
c. Slightly higher fission product release rate.

2.2 DESIGN CONSIDERATIONS IN MOX FUEL AND CORE

There are several ways to eliminate any effects from the difference between UO_2 and MOX fuels in practical MOX design. For the present use of MOX fuels in Japanese BWRs, MOX fuels are planned as part of reload fuels. Therefore, fuel assembly configurations for MOX fuels are designed to maintain compatibility and exchangeability with those of UO_2 assemblies. That is, fuel rod arrangement, rod dimensions, materials and thicknesses for fuel cladding, and fuel spacers and tie-plates designs are identical to those of UO_2 fuel assemblies. Maximum bundle exposure is designed to stay less than that of UO_2 fuels.

From a nuclear characteristics viewpoint, it is important to sustain the proper void reactivity coefficient to avoid complexity in reactor operation and control. For BWRs, void reactivity coefficient is a major concern. A more negative void reactivity coefficient may result in a smaller thermal margin during a pressure increasing transient. Hence, it is desirable for the MOX fuel to have the same magnitude of reactivity coefficient and the same thermal hydraulic behavior as the UO_2 core, when MOX fuel assemblies make up a substantial part of reload assemblies. In this regard, MOX fuel assemblies for BWRs are identical to the UO_2 fuel assemblies for high burnup, which have a large water rod in place of four fuel rods as shown in Fig. 1. A fuel assembly with a large volume of water rod helps neutron moderation and improves core characteristics of void reactivity and shutdown margin. Until now, most MOX fuel assemblies used for BWRs have been the island type, in which only the central rods in the assembly are MOX fuel rods and others are UO_2 rods. For the coming commercial use of MOX fuel in Japan, we have selected the all-MOX type assemblies rather than the traditional island type assemblies. The all-MOX type assembly can use several types of MOX rods with different Pu enrichments and can contain a large quantity of Pu. Such a design is preferable one using Pu concentrated within a small number of fuel assemblies.

For simplification of MOX fuel fabrication, it is also important to reduce the number of MOX rod types with different Pu enrichments, resulting in a smaller number of MOX assembly production steps. We optimized the number of MOX fuel rod types and quantity of Pu per assembly and developed the

Fig.1 Comparison of fuel assembly configuration for uranium fuel and MOX fuel

all-MOX type assembly with four types of MOX rods of different Pu enrichments and small number of uranium rods at the location of high power peaking. Based on the BWR-MOX design and irradiation experience at Tsuruga Unit - 1 [1], a detailed Pu and Gd distribution design for the commercial BWR-MOX assembly is now in progress. The design is expected to be suitable for the MOX utilization program in BWRs with the UO_2 assemblies for high burnup.

2.3 FEASIBILITY STUDY OF FULL-MOX BWR CORE

It is desirable for BWR to have some flexibility in the quantity of Pu utilization by adding Pu inventory per reactor. It is helpful to adjust Pu surplus from the imbalance of demand and supply. For this reason, Hitachi, Ltd. (Hitachi) studied an Advanced BWR (ABWR) core loaded fully with MOX fuel assemblies. Table I describes the specifications of this MOX core. Preliminary core characteristics evaluation showed satisfactory results in shutdown margin, thermal margin and reactivity coefficient against core design criteria as shown in Figs. 2 and 3. This study also showed that greater moderation of neutrons such as by addition of water rods is a key factor for further improvement of burnup and core characteristics.

2.4 MOX CORE DESIGN METHOD AND ITS VERIFICATION

The BWR core design method for the UO_2 core is based on the three dimensional neutron diffusion analysis of the whole core with a few energy-group nuclear constants from fuel lattice cell neutron transport. This method includes heterogeneity in the BWR core which comes from the void distribution in core, the enrichment distribution and the local gadolinia location in the lattice cell. We planned to use this method for design of the MOX fuel core.

One verification of this method was confirmed by the power profile trace of Tsuruga MOX fuel. The difference in the power profile between TIP (Traversing In-core Probe) measurements and core

Table 1 Specifications of full-MOX BWR core

Reactor core	
Electric power	1356 MWe
Thermal power	3926 MWt
No. of fuel bundles	872
Power density	50.6kW/l
Fuel bundle	
Configuration	8 x 8
Discharge exposure	~33 GWd/t
Puf enrichment (*1)	~2.9 wt%
U enrichment	~1.0 wt%
Core characteristics	
Annual Puf feed	~1.1 ton-year
Shutdown margin (*2)	$\geq$ 2.4 % Δk
MLHGR (*3)	$\leq$12.4 kW/ft
Core stability decay ratio	$\leq$ 0.7

Notes: *1 Puf = fissile Pu
*2 design limit is 1% Δk.
*3 Maximum Linear Heat Generation Rate, operation limit is 13.4 kW/ft.

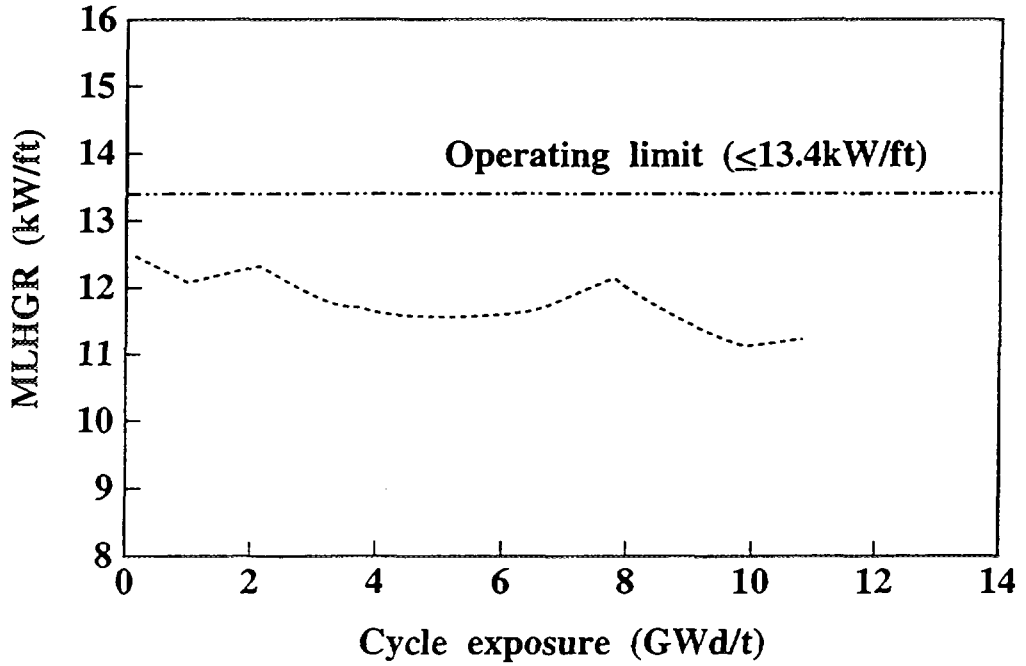


Fig.2 Maximum Linear Heat Generation Rate (MLHGR) for reload full-MOX core design

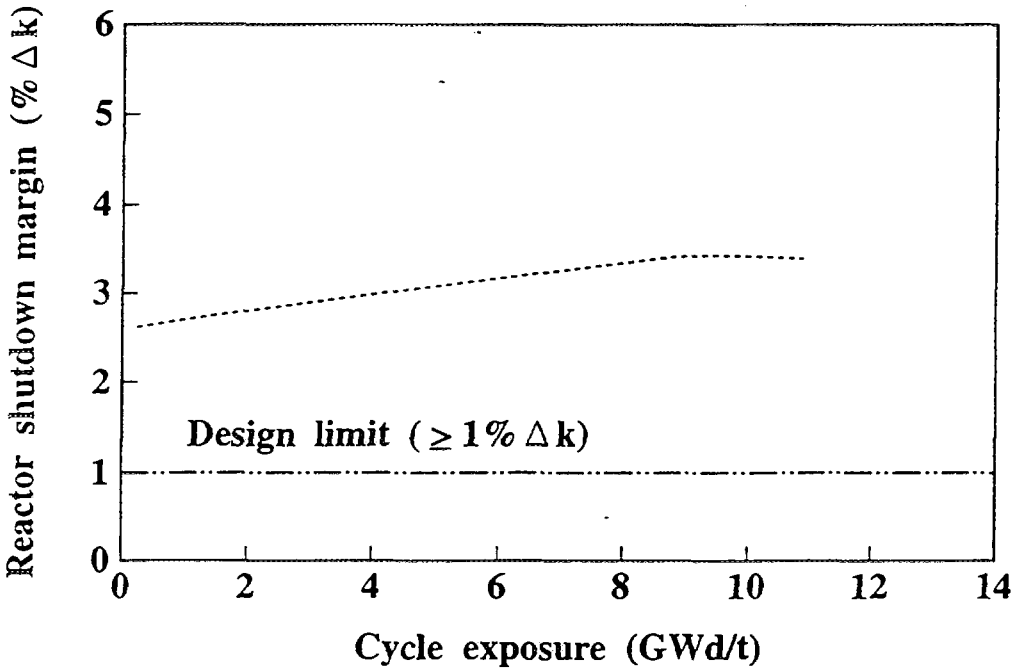


Fig.3 Reactor shutdown margin for reload full-MOX core design

analysis for the MOX fuel assembly was not larger than corresponding results for a UO_2 fuel assembly [1]. Another verification was based on the MOX core mock-up criticality experiment, in which experimental data were compared with the results from a BWR analysis method. Figure 4 compares calculation and measurement for the MOX mock-up experiment. Both eigenvalue and root mean square error in rod power for the MOX mock-up have the same magnitude as for a UO_2 mock-up [2].

An advanced lattice cell analysis method, VMONT, based on Monte Carlo neutron transport and with an isotope depletion capability, has been developed by Hitachi for future sophisticated design work of the core and fuel [3].

3. MOX FUEL IRRADIATION STUDIES

We have been carrying out various irradiation studies on BWR-MOX fuel in order to obtain information on its behavior. In this paper, two irradiation programs are outlined. One is the Tsuruga MOX irradiation program, which was the first MOX fuel irradiation in commercial LWR in Japan, and the other is the DOMO program, which was featured as a high burnup irradiation experiment for BWR-MOX fuels.

3.1 TSURUGA MOX FUEL IRRADIATION PROGRAM

3.1.1 Outline of the program

The program was carried out to demonstrate plutonium usage in BWRs, in which two BWR MOX island type fuel assemblies were irradiated in Tsuruga Unit-1 (BWR) of Japan Atomic Power Company (JAPCO) [1]. Six BWR-operating utilities, Power Reactor and Nuclear Fuel Development Corporation (PNC), and reactor and fuel manufacturers participated in the program. We joined in the fuel design and fabrication, and in the data evaluation. The fuel design work started in 1980, the licensing in 1984, and the fuel fabrication in 1985. Irradiation was begun in July, 1986. This was the first time MOX fuel was irradiated in an LWR in Japan [4, 5]. The fuel rod inspection work included visual observations and fuel rod length measurements which were carried out on the MOX assemblies during the scheduled plant outage after each reactor operating cycle. After completion of three irradiation cycles, two MOX assemblies were shipped to the hot laboratory of Nippon Nuclear Fuel Company (NFD). Pre-scheduled MOX fuel rods and UO_2 fuel rods were retrieved from the assemblies and post irradiation experiments (PIEs) were carried out at PNC on the former and at NFD on the latter.

Average and local peak burnups of the MOX discharged assemblies were about 26 GWd/t and about 37 GWd/t, respectively. The maximum linear heat rate was about 330 W/cm.

3.1.2 Fuel design and fuel rod specifications

The main specifications of the MOX assemblies, which were the identical with the BWR 8 x 8 type UO_2 fuel assembly (STEP-2 assembly) are shown in Table II [1]. The fuel rod arrangement in the MOX assembly was the island type in which 24 MOX fuel rods were placed in the center region and 38 low enrichment UO_2 fuel rods were in the periphery of the assembly. Figure 5 shows the arrangement of the fuel rods [6]. There were three types of MOX fuel rods which differed in plutonium content, low, medium, high, ranging from approximately 2 to 5 %, and four kinds of UO_2 fuel rods having different ^{235}U enrichment in the range of about 2 to 3 %. Fuel pellets for the UO_2 fuel rods had the ordinary solid shape, while the MOX fuel rods adopted hollow pellets in order to secure the needed nuclear-thermal margin by increasing the water to uranium ratio, and to reduce the fuel center temperature [1].

3.1.3 MOX fuel behavior when irradiated at Tsuruga Unit-1

The fuel rod inspections at the reactor site revealed that no abnormalities occurred in the MOX rods during irradiation [1]. As shown in Fig. 6, MOX fuel rod elongation was almost the same as for UO_2

Table 2 Specifications of Tsuruga MOX fuel assembly

Assembly	
Fuel rod array	8 × 8
Average Pu content	~ 3.2 wt%
Average ^{235}U enrichment	~ 2.4 wt%
Number of MOX fuel rods	24
Number of Pu contents	3
Number of UO_2 fuel rods	38
Number of Gd_2O_3 fuel rods	7
Weight of heavy metal	~163 Kg
Weight of Puf	~2 Kg
Fuel Rod	
Active fuel length	366 cm
Filling gas	helium
MOX fuel pellet	
Material	$PuO_2 + UO_2$
Initial enrichment	~ 2.9 wt% (*)
External diameter	~ 1.0 cm
Internal diameter	~ 0.4 cm
Theoretical density	~ 95 %TD

(*); $(^{235}U + ^{239}Pu + ^{241}Pu)/(U + Pu)$

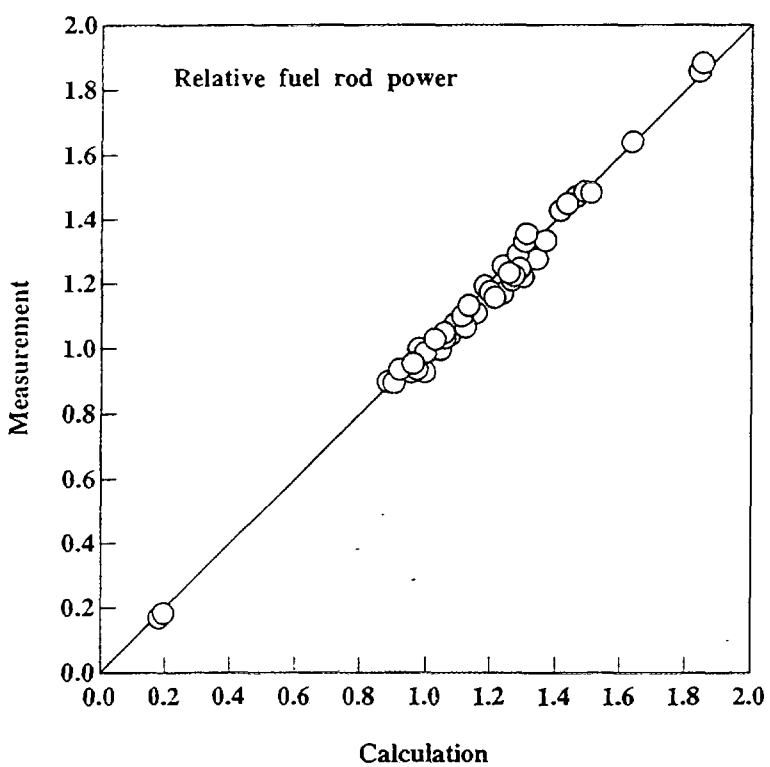
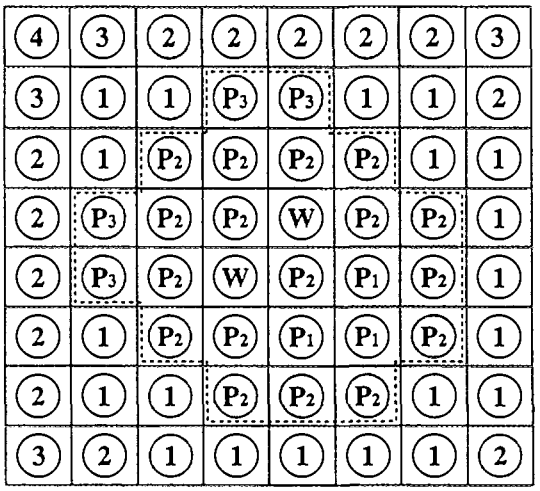


Fig.4 Comparison of calculation and measurement of relative fuel rod power (fission reaction rate) for MOX mock-up criticality experiment



UO_2 fuel rod		MOX fuel rod	
1	high enrichment rod	P_1	high Pu content rod
2	medium-high enrichment rod	P_2	medium Pu content rod
3	medium-low enrichment rod	P_3	low Pu content rod
4	low enrichment rod		

Fig.5 Fuel rod arrangement in Tsuruga MOX fuel assembly

fuel rods [1]. The non-destructive examinations on MOX fuel rods and UO_2 fuel rods were completed and the following marked results were obtained.

(a) Visual inspections, eddy current testing, γ scanning and dimensional measurements confirmed that the MOX fuel rods and the components of the MOX fuel assembly, such as the upper and lower tie-plates, bundle spacers, extension springs, finger springs etc. remained intact during the irradiation [7].

(b) Outer surface oxide thicknesses of the MOX fuel rods were comparable to those of UO_2 fuel rods with Zircaloy-2 (Zry-2) fuel claddings irradiated in various BWRs, [7].

(c) FP gas release rates of the MOX fuel rods were lower than those of UO_2 fuel rods as shown in Fig.7. This was considered due to lower temperature in the hollow MOX fuel pellet [8].

Detailed examinations on MOX fuel pellets and claddings, such as radial distribution of FP elements, melting temperature, thermal diffusivity, oxygen-metal ratio (O/M ratio), and Pu spots were made at PNC and a comprehensive evaluation of them is under way by comparing these data with those of UO_2 fuel pellets and cladding obtained at NFD. A few results from measurements of MOX pellets has been published [9,10]. The results indicated a significant burnup increase in the periphery region of the MOX pellets, and a decrease in pellet melting temperature due to addition of plutonium. However these phenomena are considered too small to affect fuel performance. It can therefore be safely concluded from the data available so far that there is no significant difference in irradiation behavior between MOX fuel and UO_2 fuel.

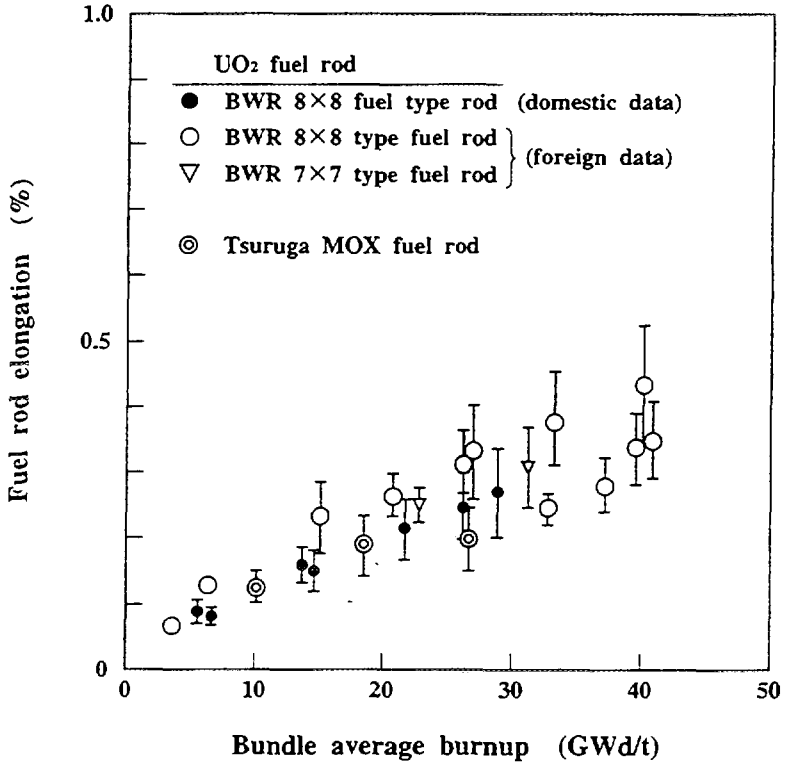


Fig.6 Fuel rod elongation of Tsuruga MOX fuel rods

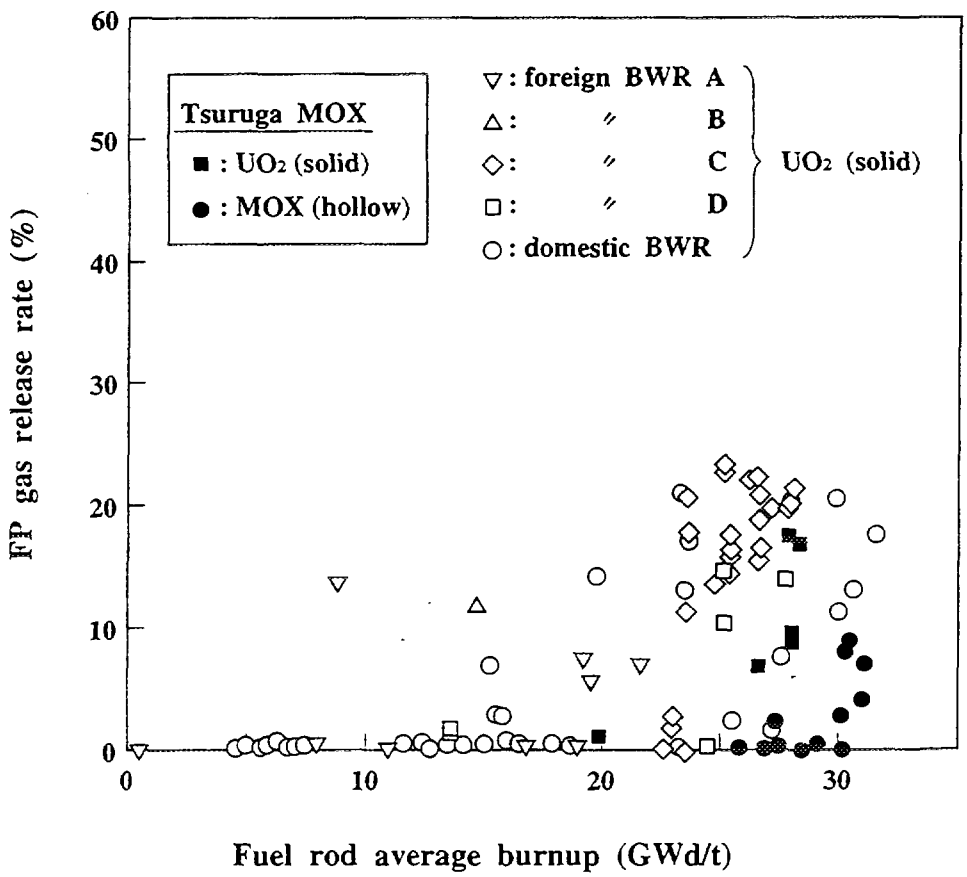


Fig.7 Fission gas release as a function of fuel rod average burnup

3.2 HIGH BURNUP MOX FUEL IRRADIATION BEHAVIOR-DOMO PROGRAM

3.2.1 Outline of the program

The high burnup program for UO_2 fuel has been progressing towards one of its goals of improved economy of BWRs [6]. In order to obtain high burnup data of BWR-MOX fuel the international research program, DOMO, was started in 1986 [11,12]. It has been carried out under the sponsorship of utilities, fuel manufacturers, reactor manufacturers and research organizations in Japan (six BWR-operating utilities, Hitachi, Toshiba, PNC, Japan Nuclear Fuel Co., Ltd (JNF) and NFD), the Netherlands (GKN and KEMA) and Switzerland (PSI) [12]. The director of the program has been Belgonucléaire (BN) of Belgium. In the program, five assemblies containing BWR-MOX fuel rods are irradiated in the Dodewaard BWR in the Netherlands. The MOX fuel rods are individually retrieved after achieving target burnups of 20, 40 and 60 GWd/t and subjected to PIEs. A couple of rods are selected out of the retrieved rods at the respective burnups and ramp tested in the BR2 up to the maximum power of 600 W/cm [12].

3.2.2 Test parameters and fuel rod specifications

The test parameters in the program were mainly related to the MOX fuel pellet fabrication process and cladding material. Some of the MOX fuel pellets were provided by BN and used BN's MIMAS (Micronized Master Blending) powder, and others were provided by PNC and used PNC's MH (Microwave Heating Process) powder. The cladding materials were Zry-2 cladding tube and Zry-2 cladding tube with zirconium-liner (Zr-liner cladding). UO_2 fuel pellets which served as a comparison were fabricated by NFD. Short length fuel rods (segments X,B,J) were fabricated by combining the respective MOX pellets and claddings as described in Table III. Short length rods (segment U) had UO_2 pellets with Zr-liner cladding, and were manufactured at JNF. Figure 8 shows a schematic view of the segment. Four segments of the same combination were mounted into a full length rod. Sets of two MOX full length rods and one UO_2 full length rod were inserted into each of five assemblies.

Table 3 Main specifications of fuel segments and rods

Segment Type			X	B	J	U
Fuel pellet						
$^{235}U/U$	(wt%)		0.239	0.239	0.70	4.95
^{235}U + Puf/U + Pu + ^{241}Am	(wt%)		4.8	4.8	4.81	4.95
Theoretical density		(%TD)	95	95	95	95
Diameter		(mm)	10.35	10.35	10.35	10.35
Height	(mm)		11.5	11.5	10.3	10.3
Manufacturer			BN	BN	PNC	JNF
Zry Cladding						
Zr-liner			yes	no	yes	yes
Inner diameter	(mm)		10.55	10.55	10.55	10.55
Outer diameter	(mm)		12.27	12.27	12.27	12.27
Segment						
Fuel stack length		(mm)	281	281	281	281
Overall length	(mm)		447	447	447	447
He absolute pressure	(atm)		5	5	5	5
Fuel rod						
Overall length	(mm)		2011	2011	2011	2011

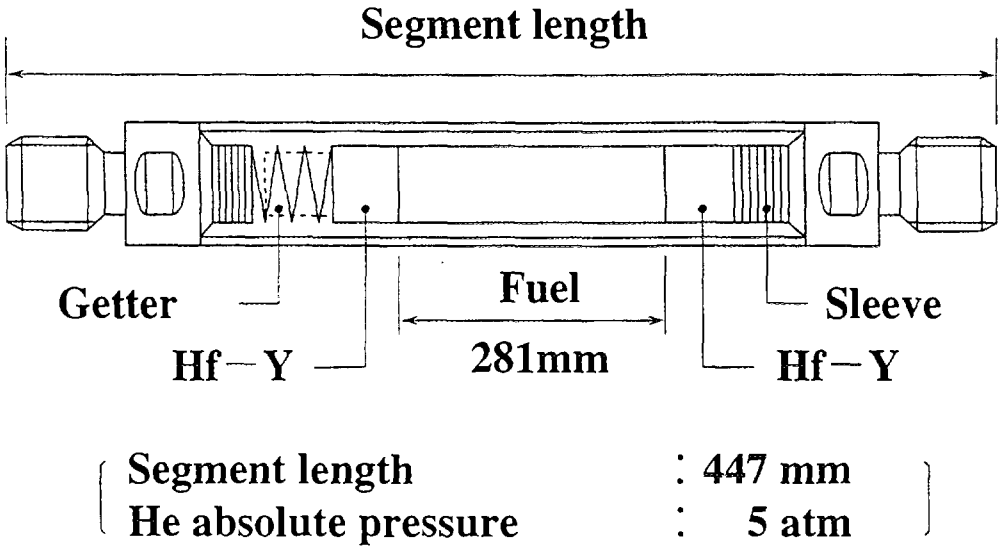


Fig.8 Schematic of segment

3.2.3 Irradiation in Dodewaard and power ramp testing at BR2

Irradiation in the Dodewaard BWR began in February 1988 and continued until January 1993. One MOX assembly was discharged in January 1990 after a two-cycle irradiation. Its segment peak burnup was about 28 GWd/t. Pre-selected segments were removed from the assembly and subjected to PIE. The two more assemblies were discharged in January 1992, after a four-cycle irradiation. Their segment peak burnup was about 50 Gwd/t. Non-destructive examinations have been completed and destructive examinations are almost completed. The remaining two MOX assemblies were retrieved from the reactor after a five-cycle irradiation. Their pellet peak burnup was estimated as about 60 GWd/t. Site inspections were done on the assemblies.

Stepwise power ramp tests were performed in the BR2 by using a pressurized water capsule [13]. The segments were preconditioned for about one day at a power of 300 W/cm. The power was then increased by 50 W/cm during 1-2 minutes, followed by a one-hour hold. The maximum power levels were about 600 W/cm, followed by about a 24-hour hold.

3.2.4 Results

a) **Steady state irradiation behavior.** For all segments irradiated for two, four and five reactor cycles in the Dodewaard, deposits of crud were apparent and nodular corrosion was observed on the outer surfaces of the fuel regions and on the plenum regions, respectively. However no other abnormal features were found during site inspections and gamma scans [12]. Eddy current tests confirmed that all segments irradiated for two and four reactor cycles were intact. No systematic difference in length change was observed between the MOX and UO_2 segments, and between Zry-2 and Zr-liner claddings. There was no apparent correlation between the increase of segment length and the maximum linear power experienced during the irradiations. Therefore it seems that the segment length increase was mainly due to cladding irradiation growth. There was no significant difference in averaged thickness of the cladding outer surface oxide between the MOX segments and UO_2 segments.

The thickness of the inner surface oxide was apt to increase with segment burnup as shown in Fig.9 [12]. However, even when the burnup reached about 55 GWd/t, the inner oxide thickness was thinner than 20 μ m. No distinct difference was observed between Zry-2 claddings with and

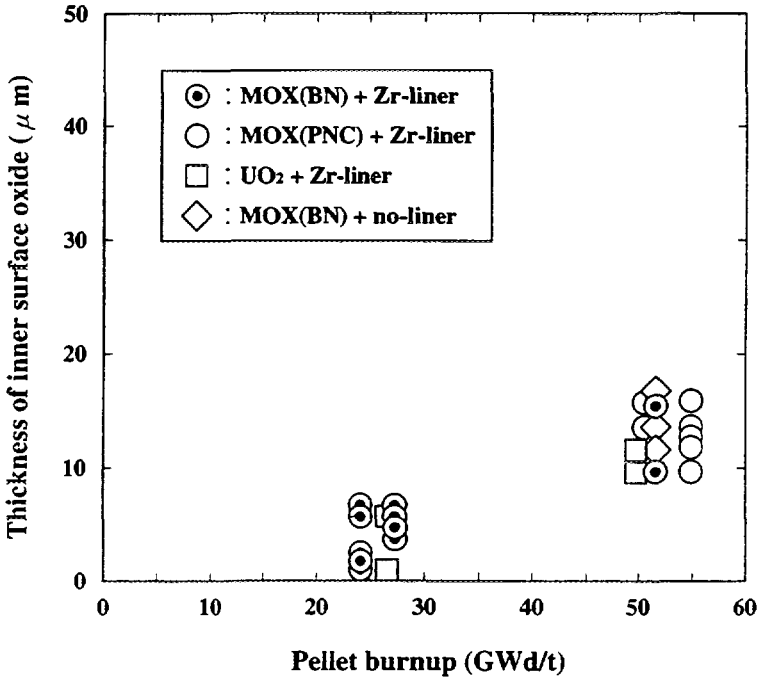


Fig.9 Thickness of inner surface oxide as a function of pellet burnup

without Zr-liner. A notable point was the fact that the maximum inner oxide thicknesses of MOX segments were comparable to those of UO_2 segments [12,13]. The chain yield for ^{235}U and ^{239}Pu shows that in plutonium fission the low-mass number peak is shifted to higher mass numbers compared to uranium fission. This means more metallic elements are produced in the plutonium fission. Therefore, on a simple thermodynamical consideration, the oxygen potential in MOX fuel seems to increase more rapidly at an early stage of burnup compared to UO_2 fuel. Accordingly, the inner surface oxidation of the MOX segments should have been more prominent than those of the UO_2 segments. The oxygen potential of fuel is affected by the chemical state of fission products, i.e. depending on the affinity of the fission products for oxygen. The SIMS (Secondary Ion Mass Spectrometry) analysis on the MOX fuel pellets, and ceramographic observation as well, revealed no evidence for significant oxidation of the metallic elements, such as Mo to MoO_3, but there was clear evidence of metallic precipitates being present in the middle-to-central region of the pellets [12]. These data suggested that free oxygen released by fission may have been consumed mainly in the cladding inner surface oxidation. Since the present thickness data obtained from each cross section were limited to very local information on the oxidation, further information, such as about the extent of the oxidation area along the cladding, is necessary for better understanding of the inner surface oxidation of the MOX fuel.

Fractional fission gas release was calculated using released fission gas quantity obtained by measuring ^{85}Kr non-destructively and by puncturing fuel segments. The fractional fission gas release rose at an average burnup of around 20 GWd/t [12]. At burnup up to about 25 GWd/t the fractional fission gas release of the MOX segments was comparable to that of UO_2 segments, but as burnup drew near 50 GWd/t the fractional fission gas release of the MOX segments seemed to be comparable to, or slightly higher than, the UO_2 segments. Figure 10 shows pellet microstructure of the BN-MOX pellets and the PNC-MOX pellets irradiated for four reactor cycles (pellet burnup: about 50 - 55 GWd/t) [12]. The microstructure in the peripheral regions where temperatures were low enough to prevent grain growth during irradiation retained their as-fabricated microstructure. In spite of a significant difference in the microstructure between the BN-MOX and PNC-MOX fuel pellets, there was no distinct difference in the fractional fission gas release between them [13]. In a comparison at the same maximum power, the fractional fission

gas release of the MOX segments was almost the same as the UO_2 segments with two reactor cycle irradiations, at four reactor cycles, the fractional fission gas release of the MOX segments seemed to be a little higher than for the UO_2 segments [12]. The above results should be confirmed by data from segments irradiated for five cycles (peak burnup: about 60 GWd/t).

b) **Transient behavior.** Five segments, four MOX segments and one UO_2 segment selected from the segments irradiated for two and four cycles, were ramp tested in the BR2. All the ramp tested segments were confirmed by a failure detection system to remain intact during the ramp testing and by non-destructive inspections after ramp testing [12]. For the segments of the two cycle irradiation, no changes in length and diameter were observed. However, additional fission gas release took place. The ceramographic examination revealed that a very large grain growth occurred in the middle-to-central regions of the pellets. For the segments of the four cycle irradiation, only non-destructive examination has been completed on each MOX segment and UO_2 segment. Neutron radiographs suggested the formation of central voids along the fuel columns for the MOX and UO_2 segments.

3.3 CONCLUSION

Results of the BWR-MOX fuel demonstration program in Tsuruga Unit - I and the high burnup MOX fuel irradiation experiments in DOMO (which includes power ramp testing at burnup up to about 50 GWd/t), showed that the BWR-MOX fuel has very high performance and integrity comparable to that of BWR-UO_2 fuel under steady and transient conditions, even though the MOX fuel had a slightly higher fission gas release rate. Further information from the PIEs on the Tsuruga MOX fuels and DOMO fuels is expected to confirm this conclusion.

4. DEVELOPMENT OF MOX FUEL FABRICATION TECHNOLOGY

4.1 EQUIPMENT FOR FUEL FABRICATION

Hitachi has a capability of designing and supplying manufacturing equipment and a process control system for FBR fuel which includes powder handling process, pellet fabricating process, pin (fuel rod) fabricating process, and assembling process.

Hitachi has much experience and a wealth of knowledge on the design and manufacturing of UO_2 fuel fabrication equipment for BWRs. In supplying the fuel manufacturing equipment, we conducted R & D on key components of the apparatus to meet particular design requirements and appropriate to handling of fissile material.

4.2 DEVELOPMENT OF A FULLY AUTOMATED MOX BUNDLE ASSEMBLING APPARATUS

In the late 1980's Hitachi launched development programs on MOX fuel fabrication technology aimed at a future domestic MOX fuel fabrication plant. Major achievements included development of a fully automated MOX bundle assembling apparatus [14] and a conceptual design study for the MOX fuel fabrication plant. These were performed as contract R & D commissioned by MITI (Ministry of International Trade and Industry). In this paper, we include only a simple explanation of the former. In designing a MOX fuel fabrication plant, various factors, such as criticality, confinement, radiation and heat generation of fissile material, should be considered in a much more rigorous manner as compared to designing a UO_2 fuel fabrication plant. Particularly important is radiation level taken into account to avoid excessive radiation exposure for plant operators. For example, if we compare the equivalent dose rates during assembling of a MOX fuel bundle having fissile plutonium contents of 3 - 5 % with those during assembling of a typical BWR UO_2 fuel bundle, the following numbers are obtained:

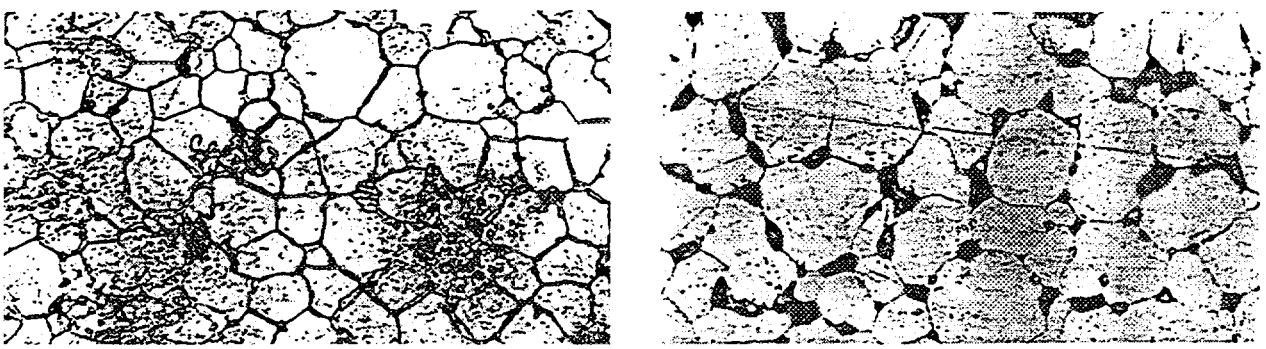
	MOX fuel bundle	UO_2 fuel bundle
assembly surface(mSv/h)	3 - 5	$\sim 2 \times 10^{-2}$
1 m from surface(mSv/h)	2×10^{-1} - 3×10^{-1}	$\sim 7 \times 10^{-4}$

The equivalent dose rates for a MOX fuel bundle are more than 200 times higher than those of the UO_2 fuel bundle. In the MOX bundle assembling process, therefore, it becomes important to automate as many features as possible for the bundle assembling apparatus and to use remote controlled steps. From this viewpoint we set out to develop the fully automated MOX bundle assembling apparatus.

The development, which took about four years, led to the apparatus shown in Fig. 11 [14]. It is composed of a fuel rod transfer system, fuel rod handling system, fuel bundle assembling station, fuel rod insertion/extraction system, and upper tie-plate installation/dismantling system. Fuel rods are assembled into a bundle by fully automated control in the fabrication process sequence; fuel rod transfer → rod number reading → spring insertion → fuel rod insertion → screwing connector rod on lower tie-plate → raising bundle assembling station → upper-tie plate installation → fixing upper-tie plate and perpendicular adjustment by screwing nuts on connector rods → fixing nuts by locking tab washers. The apparatus carries out tasks which require special skills for human workers, such as spring installation onto the upper plug, adjustment of the key thread on the plug after screwing the connector rod down to the lower tie-plate, etc., automatically and reliably through computer control. The main functions of the apparatus are summarized below.

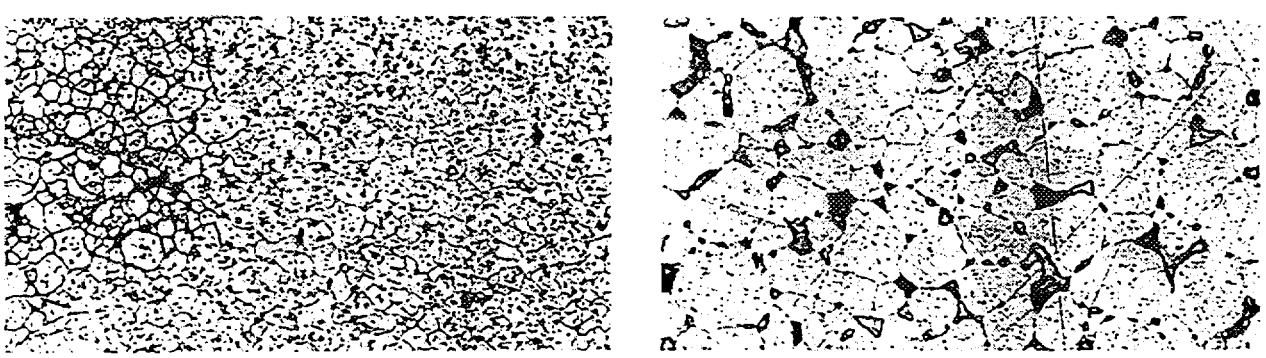
a) Fully automation of the bundle assembling process.
b) High adaptability and flexibility for various types of BWR fuel assembly designs.
c) Capability for disassembling a bundle and retrieving fuel rods to move them to a fuel rod transfer pallet automatically.

If something goes wrong during the assembling process, operators must approach the apparatus to fix it. They would be exposed to high radiation unless the fissile material is removed from the apparatus



Peripheral region ×500 Central region

(a) BN-MOX



Peripheral region ×500 Central region

(b) PNC-MOX

Fig.10 Typical microstructures of the BN-MOX and PNC-MOX pellets irradiated to about 50-55 GWd/t

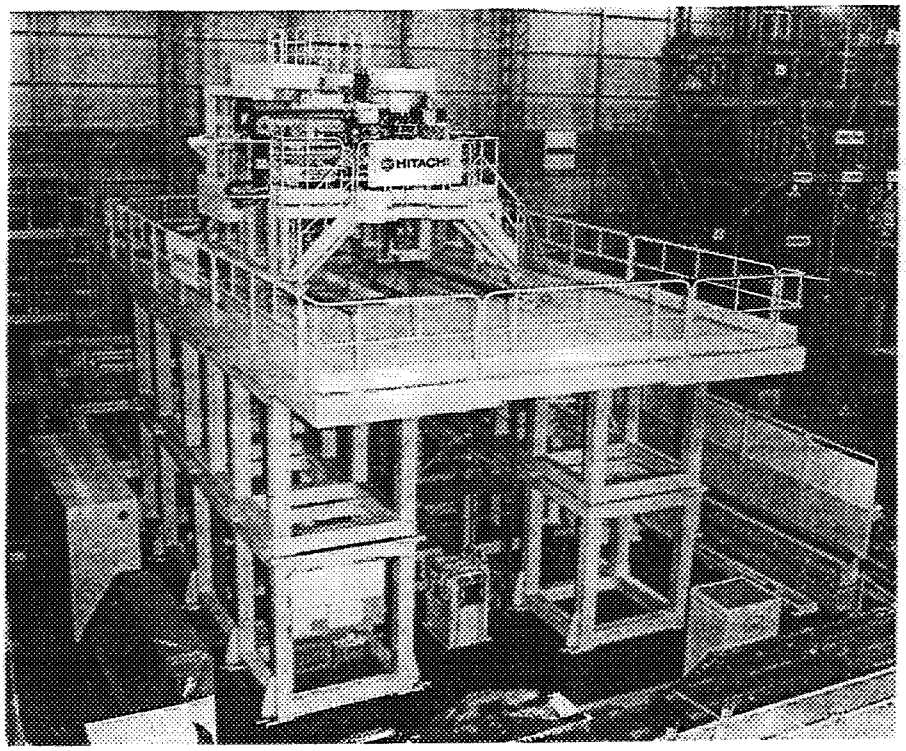


Fig.11 Fully automated MOX bundle assembling apparatus

before the repair can begin. Therefore item c) is essential to the automated apparatus. In fabrication tests using dummy MOX fuel rods, the apparatus demonstrated satisfactory performance [14].

5. CONCLUSION

In its nuclear energy policy the Japanese government is promoting systematic utilization of MOX fuel in LWRs, we have joined in the development of MOX fuel technology to allow future full scale utilization.

In this paper, we selected three subject areas, MOX core and fuel design, MOX fuel irradiation behavior, and MOX fuel fabrication technology, and described the main topics of R & D in them. In the MOX core and fuel design studies, we examined the compatibility of MOX fuel with the UO_2 core, the feasibility of a full MOX core, and the adaptability of MOX design methods based on a mock-up criticality experiment. Regarding MOX fuel irradiation behavior, we outlined the Tsuruga MOX irradiation program and the DOMO program, and showed that MOX fuel behavior was comparable to ordinary BWR UO_2 fuel behavior. In the MOX fuel fabrication technology, we dealt with the development of a fully automated MOX bundle assembling apparatus and explained its features.

REFERENCES

[1] MEGURO, T., et al., 1990 Annual Meeting of At. Energy Soc. Jpn., D38 (1990)

[2] YOSHIZAWA, A. et al., 1994 Fall Meeting of At. Energy Soc. Jpn., G78 (1994)

[3] MORIMOTO, Y. et al., Nucl. Sci. Eng., 103 (1989) 351-355

[4] OGUMA, M., IRISA, Y., "3.4 Pu-thermal fuel", Nuclear Fuel Technology, Present and Future, J. At. Energy Soc. Jpn., (1993)

[5] MEGURO, T., et al., 1990 Annual Meeting of At. Energy Soc. Jpn., D39 (1990)
[6] OGUMA, M., et al., Nucl. Enrg, 31, 1 (1992) 25-39
[7] KATAYAMA, N, et al., 1992 Fall Meeting of At. Energy Soc. Jpn., F53 (1992)
[8] KATAYAMA, N, et al., ibid., F54 (1992)
[9] UNNO, I., et al., 1994 Spring Meeting of At. Energy Soc. Jpn., E29 (1994)
[10] MITSUGI, T, et al., ibid., E30 (1994)
[11] ASAHI, K., et al., 1993 Fall Meeting of At. Energy Soc. Jpn., K17 (1993)
[12] ASAHI, K., et al., ANS 1994 Int. Topical Mtg. on Light Water Reactor Fuel Performance, Proc. (1994) 726-732
[13] ASAHI, K., et al., 1994 Fall Meeting of At. Energy Soc. Jpn., J34 (1994)
[14] OGUMA, M., et al., 1991 Spring Meeting of At. Energy Soc. Jpn., J10 (1991)