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MIXED RELOAD DESIGN USING MOX AND UOX FUEL ASSEMBLIES

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ABSTRACT

As part of the studies involved in plutonium utilization assessment for a Boiling Water Reactor, a conceptual design of MOX fuel was developed, this design is mechanically the same design of 10X10 BWR fuel assemblies but different fisil material. Several plutonium and gadolinium concentrations were tested to match the 18 months cycle length which is the current cycle length of LVNPP, a reference UO_2 assembly was modeled to have a full cycle length to compare results, an effective value of 0.97 for the multiplication factor was set as target for 470 Effective Full Power days for both cycles, here the gadolinium concentration was a key to find an average fisil plutonium content of 6.55% in the assembly. A reload of 124 fuel assemblies was assumed to simulate the complete core, several load fractions of MOX fuel mixed with UO_2 fresh fuel were tested to verify the shutdown margin, the UO_2 fuel meets the shutdown margin when 124 fuel assemblies are loaded into the core, but it does not happen when those 124 assemblies are replaced with MOX fuel assemblies, so the fraction of MOX was reduced step by step up to find a mixed load that meets both length cycle and shutdown margin. Finally the conclusion is that control rods losses some of their worth in presence of plutonium due to a more hardened neutron spectrum in MOX fuel and this fact limits the load of MOX fuel assemblies in the core, this results are shown in this paper.

INTRODUCTION

In prior years, the use of MOX fuel in thermal reactors was considered only as an alternative back-end policy option. However the plutonium recycling and MOX fuel technology has evolved to industrial level and currently several countries have established the recycling as integral part of their fuel cycle policy. Countries such as Belgium, France, Germany, Japan and Switzerland are using MOX in a considerable number of power reactors (PWRs and BWRs) Of their 40 licensed reactors, 33 have MOX fuel loaded or have applied for a license to use MOX fuel at levels up to 30% of the reactor core [1]. In Mexico there is an interest on MOX technology. Currently, there are studies to determine if the Mexican BWR power reactors can use MOX fuel and if so, determine how this technology can be implemented. This paper presents the results of a part of these studies that have been conducted at ININ (National Institute for Nuclear Research).

NOMENCLATURE

MOX	Mixed Oxide Fuel
BWR	Boiling Water Reactor
PWR	Pressurized Water Reactor
EFPD	Effective Full Power Days
FA	Fuel Assembly
FMR	Fissile Material Ratio

1.0 MOX FUEL ASSEMBLY

The MOX fuel assemblies that currently exists at the international market are geometrically similar to the conventional uranium fuel assemblies. The mechanical design of the MOX fuel assembly is exactly the same as the mechanical design as the enriched fuel assembly. The fissile material, of course does change.

The main neutronic design criteria for the MOX fuel assembly is that the burn-up at discharge should be the same burn-up as the enriched uranium fuel assembly at discharge. This design criteria is complicated by other more general requirements, for example, 1) The MOX fuel assemblies should be compatible to the enriched uranium fuel assemblies with regard to reload strategies. 2) The assembly cycle in the core should not add constrains for the reactor operation, 3) The cycle length for a mixed core should be the same as for enriched uranium fuel, and 4) The thermal limits should not exceed the limits currently established for uranium fuel. These requirements must be met without modifying to the shutdown and reactor control systems[2].

The MOX fuel assembly design and the core design are not independent process. The assembly design procedure starts by defining some average plutonium content for the MOX fuel assemblies. Next, core design calculations are used to determine if this average plutonium concentration for the MOX fuel assemblies meet the design goals [1].

2.0 PLUTONIUM COMPOSITION

The plutonium is obtained from UOX fuel irradiated in power reactors, so the isotopic composition depends on the initial enrichment, reactor type in which the fuel was irradiated, discharge burn-up of the fuel and storage time of the spent fuel. The isotopic composition is called the plutonium vector. For the calculations showed here, the plutonium isotopic concentration from BWR reactor fuel with 30 000 MWd/t burn-up was used. The composition of this plutonium vector is shown in Table 1.

Table 1. Plutonium from BWR fuel at 30 GWd/t

Isotope	Pu-239	Pu-240	Pu-241	Pu-242
%(weight)	56.3	25.5	13.4	4.8

The plutonium quality is defined as the relation between fissile isotopes and the total of plutonium isotopes, on this way the quality of plutonium from the table is 0.697.

3.0 FUEL MATRIX

It is possible to use uranium in several different forms with the plutonium in the fuel matrix. It could be in the form of depleted uranium coming from enrichment tails, natural uranium, or recovered uranium from fuel reprocessing. The main difference is in the enrichments. Natural uranium has 0.71% U-235 while depleted uranium could contain between 0.2 to 0.3% of U-235 depending on the source of the uranium [3]. The uranium from reprocessing contains around of 0.8% U-235 that is a function of the burn-up the fuel reaches during irradiation in the reactor. For calculation purposes here, depleted uranium coming from enrichment tails was used. The enrichment of this uranium was set to 0.2% U-235.

4.0 MOX CELLS

To model the MOX fuel assembly, it is necessary to construct a fuel cell for each axial zone, in this case the BWR fuel was represented for only 4 cells, 2 for top and bottom and 2 to represent partial length rods and no partial length rods zones, thus the axial enrichment is the same along the active length except for top and bottom end-pellets that are of natural uranium. The assembly has 14 partial length rods, so the assembly has some additional empty space at the top. These partial length fuel rods are selectively located in the lattice to maximize fuel weight, reduce pressure drop and increase cold shutdown reactivity margins. The partial length rods extends just two thirds of total active length in the fuel assembly. The two main cells are shown in Figures 1 and 2.

1	5	7	8	9	9	9	6	5	2
5	3S	7	4S	G1	8	4S	G1	4S	5
7	7	9	9	9	9	9	8	G1	6
8	4S	9	G1	9	W	W	G1	4S	9
9	G1	9	9	4S	W	W	9	8	9
9	8	9	W	W	4S	9	9	G1	9
9	4S	9	W	W	9	9	9	4S	9
6	G1	8	G1	9	9	9	9	G1	8
5	4S	G1	4S	8	G1	4S	G1	4S	5
2	5	6	9	9	9	9	8	5	2

Fig. 1 Rod positions BWR 10x10 assembly (Cell 1)

1	5	7	8	9	9	9	6	5	2
5		7		G1	8		G1		5
7	7	9	9	9	9	9	8	G1	6
8		9	G1	9	W	W	G1		9
9	G1	9	9		W	W	9	8	9
9	8	9	W	W		9	9	G1	9
9		9	W	W	9	9	9		9
6	G1	8	G1	9	9	9	9	G1	8
5		G1		8	G1		G1		5
2	5	6	9	9	9	9	8	5	2

Fig. 2 Rod positions BWR 10x10 assembly (Cell 2)

The table 2 shows all types of rods for the MOX assembly, this assembly has 14 short rods those type of rods has an active length of two thirds of a normal rod.

Table 2 Fissile material for each type of rod

Type	Num.	% Puf	% U^{235}	% Gd

1	1	3.96	0.1886	
2	3	4.68	0.1865	
3S	1	7.11	0.1795	
4S	13	7.92	0.1772	
5	8	5.40	0.1845	
6	4	6.12	0.1824	
7	4	6.48	0.1814	
8	10	7.11	0.1795	
9	35	7.92	0.1772	
G1	13		3.95	2.0
W	8			
Total	100			

5.0 Calculation Procedures and Results

The FMS system (Fuel Management System) from Scandpower, Inc., was used for the calculational procedures. This system contains the three integrated codes: AURORA, HELIOS and ZENITH that are used for cell calculations. The AURORA code is a preprocessor of input data file [4]. HELIOS is a neutron and gamma transport code used for lattice burn-up calculations in general two-dimensional geometry [5]. The ZENITH code permits the user to handle the basic data generated by HELIOS, and personalize the data processing for specific needs [6].

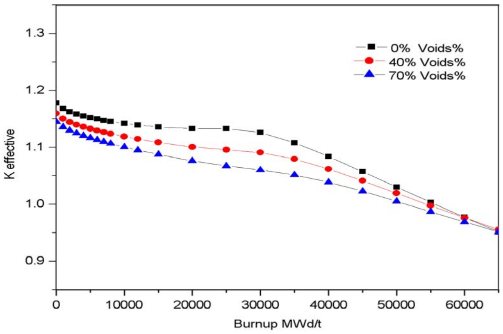
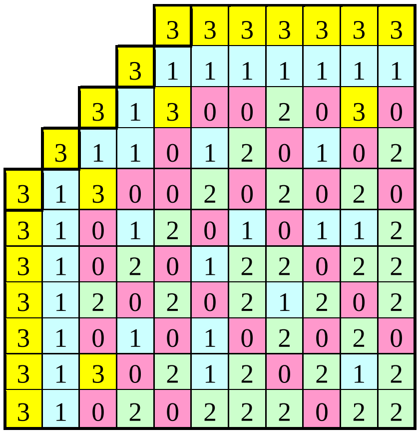


Fig. 3 Helios results for the fuel assembly proposed

The reactor simulation calculations was carried out using the CORE-MASTER PRESTO code in order to calculate the beginning of cycle, end of cycle and shutdown margin. For these calculations a reload scheme of 124 new uranium fuel assemblies was used where the average enrichment was 3.66%. This reload was the reference for comparison with the MOX fuel partial loads. The reactor core was configured taking into account one of the current operation cycles for a BWR reactor. The core contains fuel from one burn cycle, two burn cycles, three burn cycles, and 124 fresh fuel assemblies. Figure 3 shows a quarter of core corresponding to the beginning of cycle where the numbers indicate the number of irradiation cycles that the fuel has been in the core. The assemblies have had 0, 1, 2 or 3 irradiation cycles.



				3	3	3	3	3	3	3
			3	1	1	1	1	1	1	1
		3	1	3	0	0	2	0	3	0
	3	1	1	0	1	2	0	1	0	2
3	1	3	0	0	2	0	2	0	2	0
3	1	0	1	2	0	1	0	1	1	2
3	1	0	2	0	1	2	2	0	2	2
3	1	2	0	2	0	2	1	2	0	2
3	1	0	1	0	1	0	2	0	2	0
3	1	3	0	2	1	2	0	2	1	2
3	1	0	2	0	2	2	2	0	2	2

Fig.4 Quarter core reload map for 124 fresh fuel assemblies

To obtain the optimum fissile plutonium concentration for the MOX fuel assembly, models were developed in that several of the uranium pin cells were replaced with a mixture of uranium and plutonium oxides. Several concentrations of fissile plutonium material were modeled based on the fissile concentration in the uranium cells.

Once a plutonium concentration was defined for the assembly, core calculations were made. In these calculations the fresh uranium oxide fuel was replaced with MOX fuel[7], the calculation of BOC and Haling was made in order to determine the cycle length. This process was repeated for different fissile material ratios (FMR) and for different gadolinium concentrations so as to match the cycle length for uranium fuel obtained in the reference calculation, Table 4 gives a summary of the results.

Table 4. Cycle length for different fissile material ratios

Folder	CELLS	FMR	Gd 2%	Gd 3%	Gd 5%
			Cycle (days)	Cycle (days)	Cycle (days)
01	103-104	1		270.41	198.21
02	105-106	1.362		328.11	266.85
03	107-108	1.489	426.01	375.74	321.71
04	109-110	1.8	469.33	427.31	379.51
05	111-112	1.9		443.81	397.41
06	113-114	2.02		459.98	415.22
07	115-116	2.1		472.78	431.99

Once the cycle length was matched (within an accuracy of one day), the reload fraction of MOX fuel was changed from 124, 40, 32, 28, and 26 MOX fuel assemblies. The remaining fraction of the reload used was the reference uranium fuel. The results for all cases are shown in Table 5.

Table 5. Cycle length for different loads of MOX fuel

MOX FA	Cell	FMR	UOX FA	Cycle days
124	109-110	1.8	0	468.91
26	109-110	1.8	98	468.73
28	109-110	1.8	96	468.66
32	109-110	1.8	92	468.82
40	109-110	1.8	84	469.11

As may be noted in Table 5, the cycle length for all MOX fuel fractions in the reload was not significantly affected.

To complete the process the calculation of shutdown margin was performed on each burn-up step in order to insure that the reactor could be shutdown at any time during the cycle. The results of those calculations are shown in Figure 5. The value of the multiplication factor is greater than one for the core with the highest worth rod extracted for MOX reload fractions higher than those corresponding to 24 fuel assemblies, the factor is higher than 1, so there is not shutdown margin for those fractions, thus, there is not enough shutdown margin for those fractions. The reactor will not shutdown with all rods inserted in those burn-up points where the factor is greater than 1., It is observed that this occurs at the beginning of the cycle, however this results depends on where the MOX fuel is placed in the reactor.

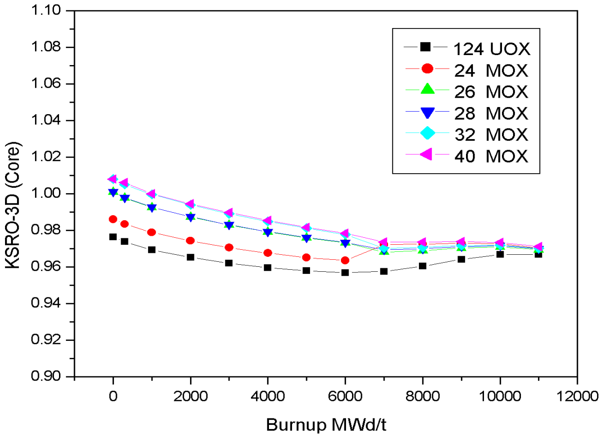


Fig. 5 Multiplication factor with the strongest rod fully withdrawn for partial loads of MOX and the reference.

Conclusions

The results obtained in study shows that the plutonium fuel reduces the effectiveness of control rods, it affects the rod worth and shutdown margin, this limits the number of MOX fuel assemblies that may be loaded into a BWR core without modifying the shutdown and control systems. Based on this study, the maximum number of MOX fuel assemblies in a reload is 24. However this number can be different depending on the position where the fuel is placed in the reactor. The multiplication factors observed in Figure 5 for some of the proposed reloads exceeds the limit at the beginning of cycle this means that an optimization of gadolinium content in the fuel assembly would be required.

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KEYWORDS

Plutonium-fuel, MOX-fuel, Fuel cycle, Power-Reactors, Nuclear Fuel.

REFERENCES

[1] K. Fukuda, J.S. Choi, R. Shani, L. Van der Durpel, E. Bertel, E. Sartori, “MOX FUEL USE AS A BACK END OPTION: TRENDS MAIN ISSUES AND IMPACTS ON FUEL CYCLE MANAGEMENT”, MOX Fuel Cycle Technologies for Médium and Long Term Deployment, IAEA-SM-358-I, Vienna, Austria, 1999.

[2] Laura Zanotti, Jacques Porta, Gilbert Rouviere, “Influence of Neutron Spectrum Modifications on Recycled MOX Fuel in Boiling Water Reactors”, 7th International Conference on Nuclear Engineering, Tokio, Japan 1999.

[3] “Plutonium Fuel: an Assessment”, Report by an Expert Group, NEA-OECD 1989, p. 21.

[4] AURORA, User Manual, SCANDPOWER, December 1995.

[5] HELIOS, User Manual, SCANDPOWER, December 1995.

[6] ZENITH, User Manual, SCANDPOWER, July 1996.

[7] D. Porsch, A. Charlier, G. Meier, J. C. Mougniot, K. Tsuda., “Overview of neutronic fuel assembly design and in core fuel management”, MOX Fuel Cycle Technologies for Médium and Long Term Deployment, IAEA-SM-358-I, Vienna, Austria, 1999. pp 345.