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MOX fuel use in a BWR with extended power up-rate

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ABSTRACT

Although MOX fuel coming from reprocessed depleted uranium fuels has been used as a recycling strategy by countries like France and Japan it is not a common policy in the 30 countries that uses nuclear power, nowadays it seems to be a more direct alternative to reduce the depleted fuel interim storage. Previously, the spent fuel pools of Laguna Verde Nuclear Power plant were redesigned to host the total operating life depleted fuel under its original nominal power condition, however the plant has been up-rated to 120% of its original nominal power increasing the number of depleted fuel forecasted. This new situation makes necessary the analysis of alternatives, being one of them recycling. The current paper assesses the viability of using MOX fuel in the up-rated Power Plant; the design of the boiling water reactor MOX fuel addresses the two main constraints of its use: shutdown margin and reactor stability. Fuel design proposed sets the appropriate MOX enrichment and the maximum MOX fuel batch reload that does not imply any modification to the reactor control systems to avoid an extra economical cost due to its use.

1. Introduction

The amount of spent nuclear fuel in nuclear power stations grows more each time becoming a hard problem; however through the reprocessing of spent fuel it is possible to recover the unused uranium and plutonium in the used fuel elements. So, the plutonium and uranium recovered can be used for mixed oxide (MOX) fuel manufacturing and recycled in nuclear reactors contributing in this way to solve storage and proliferation concerns. Nowadays the technology for reprocessing and recycling is considered mature and there is enough experience on MOX manufacturing (MOX fuel cycle technologies for medium and long term deployment, IAEA, 1999).

Up to now there is no a global policy about recycling depleted fuel, from a technical point of view it seems that this alternative can help to alleviate the depleted fuel storage problem; meanwhile it recovers valuable energy making nuclear energy sustainable. Mexico has not defined any policy on this regard, thus assessing this alternative will provide with information about MOX performance in nuclear reactors cores to have elements to make an informed decision.

Much work has been done in learning about the physics of PWR MOX fuel assemblies and its use has been documented mainly in France and Japan. However, there is not much information about BWR MOX fuel, but recently to gain insight, a Benchmark study was conducted by the Nuclear Energy Agency (NEA–OECD Physics of plutonium recycling BWR MOX benchmark, 2003). In addition to that, Demaziére (2002) set two of the main constraints in the use of BWR MOX fuel: the core shutdown margin and the stability of the core.

This study will focus in the design of a BWR MOX fuel assembly based on the geometric design of a 10×10 fuel assembly. The objective is to find out if this design used in a mixed core along with UO_2 fuel can achieve the target energy requirements, thermal margins and reactivity margins for an equilibrium cycle under extended power up-rate conditions (120% nominal original power and 85% nominal coolant flow).

The MOX fuel yields a lower thermal flux than does UO_2 fuel, which causes thermal absorbers to have lower worth. This limits the amount of MOX that can be loaded without major modifications to control systems.

The study will show, for the equilibrium cycle, what is the maximum number of MOX fuels to be loaded in a core to get a nuclear reactor having an acceptable shutdown margin. Also, the resulting core will be tested under BWR startup operation and in-side the instability region of the flow-power map.

2. Fuel design objectives and constraints

The mechanical design of MOX fuel assembly is based on the 10×10 fuel assembly design of the corresponding UO_2 assemblies. This assumption is supported by the fact that MOX fuel assemblies

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that currently exist in the international market are geometrically similar to the conventional uranium fuel assemblies.

The main neutronic design criterion for the MOX fuel assembly is that the burnup at discharge will be the same for both MOX and enriched uranium fuels. Furthermore, the MOX fuel assemblies will be compatible with the enriched uranium fuel assemblies regarding reload strategies, the MOX assemblies in the core will not add constraints to the reactor operation, the cycle length for a mixed core will be the same as for enriched uranium fuel, and the thermal limits will not exceed the limits currently established for uranium fuel. These requirements must be met without modifying the shutdown and reactor control systems (Zannoti et al., 1999).

The core analyzed will be the up-rated equilibrium cycle that has been met after nine transition cycles; before the power up-rate of the power plant, the nuclear core was composed by fuel assemblies of a 10×10 type.

The UO_2 equilibrium cycle is a 444 fuel assemblies array composed by two different 10×10 fuel assemblies with corresponding initial enrichment of 4.001 w/o of U-235, that will be called type A and 3.994 w/o of U-235 that will be called type B, it works in 18-months cycles and has a total (up-rate) power of 770 MWe. The equilibrium reload will comprise 56 assemblies type A and 68 type B, these assemblies will remain in the core during four 18-months cycles.

The codes used in this analysis is the CMS package from Studsvik Scandpower, it comprises the codes INTERPIN (Studsvik Scandpower, 2004), CASMO-4 (Rhodes and Edenius, 2004), SIMULATE-3 (Studsvik Scandpower, 2005), and SIMULATE-3 K (Studsvik Scandpower, 2002, 2006a,b), these codes has been tested for the analysis of BWR MOX–UO_2 mixed cores showing reliable results (Noel et al., 2008).

Plutonium is obtained from UO_2 fuel irradiated in power reactors, so the plutonium isotopic composition depends on the initial enrichment of the UO_2, reactor type, discharge burnup, and storage time of the spent fuel (this is known as reactor-grade plutonium) (IAEA TRS-415, 2003). To reduce the number of variables in this study, we use a typical plutonium isotopic concentration from spent LWR fuel (NEA/OECD Benchmark, 2003). This fissile plutonium content is shown in Table 1.

The design principle in the MOX fuel assembly is to find a MOX assembly equivalent to the UO_2 Fuel, Starting with a fuel containing the same amount of fissile material than the corresponding UO_2 fuel assembly, thus from Table 1, there are a 59.6% of plutonium fissile content, using the following relations the scaling factor for the MOX enrichment can be obtained.

$$X = \text{Plutonium Quality} = \text{fissile Pu/total Pu}$$
$$\text{Scale factor} = X^{-1}$$
$$\%\ \text{w of Total Pu} = (\text{Scale factor})(\text{fissile uranium})$$

Thus, in this case 1.678 of total plutonium is leveling one unit of fissile uranium. Using the previous relation each uranium concentration in each cell assembly will be scaled to total plutonium and the fissile Pu concentration will match at least in the amount of fissile material than the one that has the uranium fuel pin.

In this analysis, type A and B fuel assembly will be replaced by the corresponding MOX fuel assembly design; it will be called type C. Both, type A and B, fuel assemblies are composed by seven axial regions called cell assemblies. Top and bottom cell assemblies are composed only by natural uranium; these cell assemblies will remain the same in the MOX fuel assembly.

Table 1
Plutonium isotopic composition.

Isotope	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242
% w	2.20	46.20	29.40	13.40	8.80

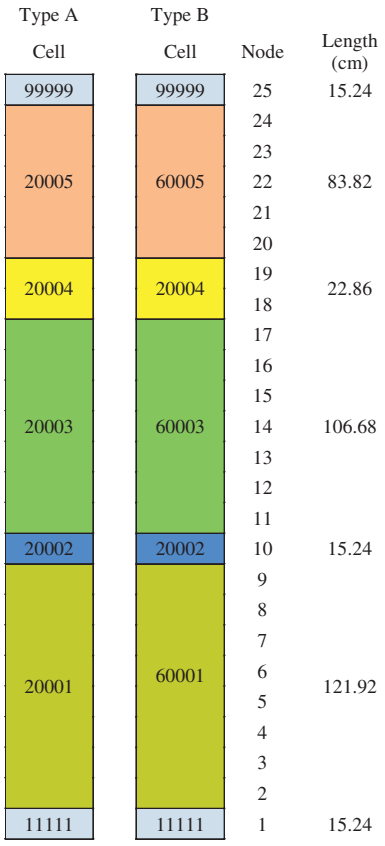

Fig. 1. Axial structure for the fuel assembly.

There are five cell assemblies that will be designed. MOX cell assemblies will not have MOX pins with gadolinium because the licensing problems that will arise by the use of MOX pins with gadolinium instead UO_2 fuel pins with gadolinium will be used, due to this constraint, the gadolinium will worth more in these pin fuels due to a greater thermal spectrum coming from the UO_2. Fig. 1 shows the axial distribution of the MOX fuel assembly (type C) and Figs. 2–8 shows the radial distribution of the seven MOX cell assemblies.

The MOX fuel rods have a density of 9.921 g/cm^3 (Benchmark Specifications, NEA/OECD), while the UO_2/Gd_2O_3 rods have a density of 9.791 g/cm^3 (with 4.0% w of Gd) and 9.758 g/cm^3 (with 5.0% w of Gd), and an enrichment of 4.90% w into the uranium matrix, the last densities were calculated with the equation

$$\rho_C = \rho_T * (\%\rho_{TC}) - (\%\ wGd) * 0.03245$$

3. Core simulation

The first step to analyze the equilibrium fuel cycle using MOX fuel as part of the BWR fuel reloads, is to obtain the cell assembly homogenized cross sections and another nuclear parameters that will be calculated using CASMO-4 (Tamer Bahadir, 1997). This simulation starts with the use of INTERPIN code (Studsvik Scandpower, 2004), to obtain MOX fuel temperatures under full power (hot condition), so it is necessary to know all the geometric parameters for fuel rods and pellets as well as physical properties for them. The general data for the core are: