INERT MATRIX FUEL ASSEMBLY AS AN OPTION FOR THE LAGUNA VERDE NPP FUEL RELOADS

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ABSTRACT
The availability of large amounts of reactor and weapons grade plutonium in the world shows the necessity of anticipating situations for the use and disposition of it. Because Light Water Reactors (LWRs) prevail on the stage of electric energy generation by nuclear power, it is important to take into account the potential of these reactors to reduce the plutonium inventory. Several studies performed in Pressurized Water Reactors (PWRs) show that reactor and weapons grade plutonium can effectively be burned in these reactors, in assemblies with fertile-free fuel, and maintaining reactivity control and other safety issues at least comparable to those related to the standard fuel normally used.

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The Instituto Nacional de Investigaciones Nucleares, currently carries out research on diverse alternatives to use Inert Matrix Fuel (IMF) as an option to fuel reloads for the two BWR/5 Units at the Laguna Verde Nuclear Power Plant. This work presents first the neutronic analysis of a fuel assembly conceptual design, which contains a combination of plutonium oxide (in an inert matrix) fuel rods, uranium oxide fuel rods, and uranium oxide with gadolinia fuel rods. Then, simulations for three different fuel assembly reload options were performed for Unit 1. Results of reactor operation from the different reload options are presented.

The results obtained with reload fuel using inert matrix fuel assemblies observe a decrease in the length of operation cycle in the plant. However, the mass of uranium used is minor to require for make all fuel assemblies.

1. INTRODUCTION
The availability of large amounts of reactor and weapons grade plutonium in the world shows the necessity of anticipating situations for the use and disposition of plutonium. Because light water reactors (LWRs) prevail on the stage of electric energy generation by nuclear power, it is important to take into account the potential of these reactors to reduce the plutonium inventory. Several studies performed in Pressurized Water Reactors (PWRs) show that reactor and weapons grade plutonium can effectively be burned in these reactors, in assemblies with fertile-free fuel, and maintaining reactivity control and other safety issues at least comparable to those related to the standard fuel normally used.

Full cores of mixed oxide (MOX) fuels have been proposed for PWRs. In these studies, degradation of some nuclear parameters has been found, for example delayed neutron fraction. This problem leads to issues related to operation and control. However, other studies [1,2] show that some modifications in control design and operation made possible the use of 100% MOX cores. Highly-moderated reactors consume more plutonium, while producing less actinides. Since the two main actinides, plutonium and americium, present significant resonances at thermal energies, the moderation ratio (water volume to fuel volume) becomes an important parameter for the core design of PWRs with 100% MOX fuel [3,7].

By increasing the moderation ratio, more neutrons have the chance of reaching thermal energies and increasing the probability of escaping the actinides’ resonance region. Some kinetics parameters also show favorable performance in high moderation ratio reactors.
Regarding Boiling Water Reactors (BWRs), new fuel assembly designs present a decrease in the geometric dimensions of fuel rods. As consequence, an increase in the moderation ratio is achieved. Some of these new designs also consider the possibility of using plutonium oxide as fuel material. Studies are then performed to determine the amount of power generated and if safety constraints are all satisfied.

In addition to existing MOX fuel designs for plutonium stockpile reduction, alternative fuel designs are being and studied for possible use in power reactors. In particular, for LWRs, Inert Matrix Fuel (IMF) and Rock-like Oxide (ROX) fuel concepts have been shown as a viable alternative for plutonium consumption in the future.

The homogeneous solid solution of PuO2 in stabilized ZrO2, with Erbium as burnable poison, is the most studied concept for plutonium burning in LWRs. This IMF concept has important physical and economical advantages. However, the PuO2-ZrO2 IMF has a lower thermal conductivity than the UO2 matrix. Thus, other types of IMFs have been proposed because of their higher thermal conductivity, as MgAl2O4, CeO2, Y2Al5O12 and SiC for cermets fuels, and Zr and SS for cermet fuels [10].

Although all these material compounds still present physical and chemical issues, the fuel assembly design and simulation of neutronic behaviour during a fuel cycle can help identify potential problems for power reactor operation.

A more realistic option for use of IMFs in power reactors requires that fuel assemblies contain both UO2 fuel rods and other fuel rods with some of the IMFs mentioned above. For use in PWRs, design concepts of a fuel assembly have been proposed. Both full [11] and partially loaded cores have been studied. Heterogeneous fuel assembly designs contain UO2 and PuO2-MgAl2O4 [12] or PuO2-CeO2 fuel rods [13]. Based on fuel assembly level computations, several different concentrations of plutonium, burnable poisons and IMF have been proposed [14,15], to find the smallest reactivity variation with relation to burnup and temperature, as well as the best control strategy using the current reactor equipment. For BWR fuel assembly, experimental studies have been performed, showing that some measured parameters, reactivity worth, for example [16], still present noticeable differences with respect to those computed.

Computations of the dynamics of a reactor core loaded with traditional UO2 or MOX fuel can be performed with high confidence with the current analysis techniques. On the contrary, simulation of power reactor operation is still not fully acceptable when IMFs are introduced to the reactor core. This is so because of the quite different neutronic characteristics of the IMF in a LWR, when compared to those of UO2 or MOX fuels.

A numerical benchmark study was performed to compare results from different computational methods and data sets for analysis of a PWR with IMFs [17]. The results show that some parameters agreed well, for example kinf, for all computer codes employed, but significant differences were also observed in others, as fuel temperature coefficients for End of Life (EOL) conditions. Therefore, much care still needs to be taken when analyzing the results of reactor state simulations.

The Instituto Nacional de Investigaciones Nucleares (ININ, National Institute of Nuclear Research) of Mexico, at its Departamento de Sistemas Nucleares (DSN, Nuclear Systems Department), currently carries out research on diverse alternatives to use IMF as an option to fuel reloads for the two BWR/5 Units at the Laguna Verde Nuclear Power Plant (LVNPP). This work presents first the neutronic analysis of a fuel assembly conceptual design, which contains a combination of plutonium oxide (in an inert matrix) fuel rods, uranium oxide fuel rods, and uranium oxide with gadolinite fuel rods. Then, simulations for three different fuel assembly reload options were performed for the Cycle 11 of the Unit 1. Results of reactor operation from the different reload options are presented.

2. FUEL ASSEMBLY DESIGN

Some studies have shown that one of the best candidates to become the inert matrix, in a uranium free fuel, is the oxide of cerium. One reason is that the CeO2 crystalline structure is similar to that of the PuO2. Other reason is its very low microscopic cross section of neutron capture at thermal energies (0.13 barns at 0.625 eV). Another important advantage of a PuO2-CeO2 fuel matrix constitutes a solid solution no soluble in hot water. This type of IMF was therefore chosen to design fuel assemblies for different options for a BWR fuel reload.

Geometrically, the inert matrix fuel rods have smaller dimensions than those of UO2, but the rod to rod pitch was kept the same. In this way, the moderation ratio was increased for the inert matrix fuel rods. Figure 1 shows the dimensions of fuel pellet and cladding for both the standard uranium oxide pin and the inert matrix fuel pin.
The fuel assembly base design is a 10×10 geometric array, which contains 42 PuO₂-CeO₂ fuel rods with weapons-grade plutonium, 34 fuel rods of UO₂, and 16 fuel rods contain fuel and burnable poison, UO₂-Gd₂O₃. The isotopic concentrations for each type of fuel pin used in the cell calculations are shown in the Table 1, and the distribution of the pins in the fuel assembly is shown in Figure 2. The fuel lattice neutronic design and nuclear data bank generation were carried out with the HELIOS code [18].

Table I. Fuel rod isotopic composition of inert matrix fuel assembly

<table>
<thead>
<tr>
<th>Pin</th>
<th># of pins</th>
<th>235U [%]</th>
<th>238U [%]</th>
<th>239Pu [%]</th>
<th>Gd₂O₃ [%]</th>
<th>CeO₂ [%]</th>
</tr>
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<tbody>
<tr>
<td>1</td>
<td>3</td>
<td>2.00</td>
<td>98.00</td>
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<td>-</td>
<td>-</td>
</tr>
<tr>
<td>2</td>
<td>1</td>
<td>2.40</td>
<td>97.60</td>
<td>-</td>
<td>-</td>
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</tr>
<tr>
<td>3</td>
<td>2</td>
<td>2.80</td>
<td>97.20</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>4</td>
<td>4</td>
<td>3.20</td>
<td>96.80</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>5</td>
<td>6</td>
<td>3.60</td>
<td>96.40</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>6</td>
<td>18</td>
<td>3.95</td>
<td>96.05</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>IMF</td>
<td>42</td>
<td>-</td>
<td>-</td>
<td>32.60</td>
<td>-</td>
<td>67.40</td>
</tr>
<tr>
<td>G</td>
<td>16</td>
<td>3.95</td>
<td>96.05</td>
<td>-</td>
<td>4.00</td>
<td>-</td>
</tr>
</tbody>
</table>

The second pattern replaced 104 UO₂ spent fuel assemblies (26 fuel assemblies per quarter of core, as shown in Figure 3a) are exchanged for the full core reload. The other pattern replaced 104 UO₂ spent fuel assemblies with 32 IMF and 72 UO₂ fuel assemblies, and quarter core symmetry was employed. Figure 3 shows the four reload schemes proposed in this study for Cycle 11.

In the first case, the base reload pattern, 104 new UO₂ fuel assemblies were placed instead of spent UO₂ fuel assemblies (Figure 3c). In the last reload scheme, 104 IMF assemblies were placed instead of spent UO₂ fuel assemblies (Figure 3d).

The simulation of the three-dimensional, steady state, neutronic core behavior was performed with the simulation codes that compose the Fuel Management System (FMS). HELIOS code was used to generate the nuclear data bank, and the simulation of the reactor operation in steady state was performed with the CM-PRESTO code [19]. For the simulation, the same plant operational conditions are used for the UO₂-only and Pu₂-CeO₂ fuel assemblies, and quarter core symmetry was employed. Figure 3 shows the four reload schemes proposed in this study for Cycle 11.
4. RESULTS

Figure 4 shows a comparison of the infinite multiplication factor behavior when using the new IMF lattice design and the standard uranium fuel lattice proposed, base case, for cycle 11 of LVNPP, at three different void fractions. Note, in the figure, that up to 2000 MWD/TM, $k_{\text{inf}}$ is greater for the standard UO$_2$ lattice, at the three void fractions (0%, 40% and 70%), but after this burnup the IMF lattice presents an infinite multiplication factor value greater than that for the uranium fuel lattice. The figure also shows that the behavior of $k_{\text{inf}}$ varies quite slowly for the IMF lattice during the first 3000 MWD/TM, quite different from the behavior of $k_{\text{inf}}$ for the UO$_2$ lattice.

![Figure 4. Infinite multiplication factor comparison from the IMF and UO$_2$ lattices at three void fractions](image)

The local peaking factor behavior as function of burnup shows always greater values for the IMF lattice at all three void fractions, as shown in Figure 5, compared with the base case uranium fuel lattice. It can also be noted in Figure 5 that the local peak factor values from the IMF lattice at high burnup (70,000 MWD/TM) almost reach again their initial values, in opposition to the noticeable decrease in value shown by the local peak factor for the UO$_2$ lattice.

![Figure 5. Local peaking factor comparison from the IMF and UO$_2$ lattices at three void fractions](image)

Also, the maximum relative power occurs at node 7 (106.68 cm) for all reload schemes. Note also that for all BOC and EOC cases, the fuel reloads with IMF assemblies generates more power at the lower part of the core, while at the upper region of the core the fuel reload options with less IMF assemblies generate more power.

Figure 7 shows the average axial burnup distribution of the reactor core from the four reload schemes. The burnup distribution at the BOC is the same for all cases, so only the case for the OU$_2$ is shown in the figure. The burnup profile is also similar at EOC for the four reload options. The figure also shows that when using more IMF assemblies there is a small decrease in the burnup reached during the cycle.

![Figure 6. Axial Relative power average for core for four fuel reload schemes analyzed](image)

![Figure 7. Reactor core average burnup from each fuel reload scheme proposed, both at BOC and EOC](image)
Although the average axial burnup distribution is quite similar for all fuel reload options analyzed, the burnup at BOC presents noticeable discrepancies. One major reason for the difference comes from the modeling in the core simulator CM-PRESTO. When the burnup is evaluated, the code considers only the current uranium content, but not plutonium or other fissile materials present. That is, the plutonium produced in the assemblies introduced during the previous cycles is not taken into account for burnup evaluation at the beginning of the new cycle, nor the additional plutonium content of the IMF assemblies. The differences on the uranium contents in the reactor core are shown in Table II, as well as the cycle length and burnup, for each reload option. It can be noted from Table II that the more the uranium content at BOC the less burnup level of the core.

The average lineal heat generation rate (LHGR) of the reactor core from the four reload schemes is shown in Figure 8. At BOC, the four reload schemes present a similar behavior. The maximum value of the LHGR (210 W/cm) occurs for the fuel reload consisting of IMF assemblies only (104 assemblies). The maximum LHGR is located either at node 6 (91.44 cm) or at node 7 (106.68 cm) for all fuel reload options. However, at EOC, the location where the maximum LHGR takes place changes depending on the number of IMF assemblies inserted. The mixed reload consisting of 72 UO₂ and 32 IMF assemblies and the full UO₂ reload show their maximum LHGR at node 18 (274.32 cm), while the other two options still continue presenting their maximum LHGR at node 7 (106.68 cm). For the EOC case, the maximum LHGR value is 180 W/cm.

Figure 9, shows the axial distribution of the lineal heat generation rate for the hottest fuel assembly in each reload scheme analyzed. The maximum value occurs for the reload scheme consisting of a mix of 32 IMF and 72 UO₂ assemblies. The maximum value of the lineal heat generation rate (MLHGR) values both at beginning and end of cycle from the four reload fuel schemes analyzed. Note, in Table III, that the full UO₂ reload option has its MLHGR occurring in the same channel, but this does not happen for the other three schemes, which contain the PuO₂-CeO₂ IMF.

Finally, the important results regarding the fuel cycle length for each reload option can be seen in Table II. The less number of IMF assemblies inserted the closer the cycle length is to the base case, full 104 UO₂ assemblies reload.

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<tbody>
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<td>104</td>
<td>0</td>
<td>80.350</td>
<td>0</td>
<td>406.98</td>
<td>21323</td>
<td>31589</td>
</tr>
<tr>
<td>72</td>
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<td>21889</td>
<td>32155</td>
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<tr>
<td>32</td>
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<td>75.674</td>
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<tr>
<td>0</td>
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<td>73.596</td>
<td>1.551</td>
<td>372.77</td>
<td>23279</td>
<td>33545</td>
</tr>
</tbody>
</table>
5. CONCLUSIONS

As can observe in results, the use of IMF assemblies as reload elements during simulation of operation cycle 11 of LVNPP present a behavior similar in axial power distribution and LHGR average in core, independently in number of fuel assemblies used in reload scheme. Also, take account an analysis HALING the LHGR is not exceeded limit.

Also, observed the amount of uranium required decrease for the cycle. However, has imply a decrease in length of operation cycle until in 34 days if reload scheme with 104 inert matrix fuel assemblies is considered. But the benefit is considered that can be obtained from one reduced amount of plutonium highly enriched, the use of this type of fuel can be considered like alternative to eliminate the plutonium inventories that are had in the world

ACKNOWLEDGMENTS

The authors wish to thank both CONACYT (grant: SEP-2004-C01-46694) and ININ (grant: CA-610) for their support to this research study.

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