VOID REACTIVITY STUDIES FOR THE ATUCHA-II PHWR REACTOR. PRELIMINARY DESIGN OF A LOWER VOID REACTIVITY FUEL

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Abstract

Atucha-II is a 745 MWe (2160 MWt) nuclear station, D_2O moderated and cooled, of German (Siemens) pressure-vessel design, located in Argentina. It has 451 vertical coolant channels and the fuel assemblies (FA) are clusters of 37 natural UO_2 rods with an active length of 530 cm. The coolant and moderator have average temperatures of 296 ºC and 177 ºC. The Atucha-II reactor has a positive coolant void reactivity (CVR) of about 10 mk. This paper presents a study that includes comparisons of WIMS-D5 and MCNP5 results of cell and reactor CVR calculations and a preliminary fuel design change oriented to reduce the CVR to less than the delayed neutron fraction.

1. Introduction

In 2005 the Argentine Government took the decision to complete the construction of Atucha-II, which is a 745 MWe (2160 MWt) nuclear power plant, D_2O moderated and cooled, of German (Siemens) pressure-vessel design. It has 451 vertical coolant channels and the fuel assemblies (FA) are clusters of 37 natural UO_2 rods with an active length of 530 cm. Power regulation is performed through absorber control rods made of hafnium and steel tubes which are located between rows of channels and enter the core slightly obliquely. The coolant and moderator have average temperatures of 296 ºC and 177 ºC.

The Atucha-II reactor has a positive coolant void reactivity (CVR) of about 10 mk. This paper presents a study that includes comparisons of results of cell and reactor CVR calculations with MCNP5 [1]. The cell calculations have been performed with WIMS-D5 [2], the incremental cross sections have been calculated with DRAGON 3.04o [3], and the core calculations were obtained with the 3-D diffusion reactor code PUMA [4]. In addition, a preliminary design oriented to reduce the CVR to less than the delayed neutron fraction using dysprosium in the central rod has also been performed. Results of some parametric studies that lead to the final proposal are presented.

2. Void reactivity in HWR reactors

The heavy water nuclear power plants operating at the present time (both the CANDU pressure tube design and the Siemens pressure vessel design) have a positive coolant void reactivity coefficient. In a possible accident with a break in the heat-transport system and subsequent depressurization, in the initial period of the event the coolant vaporizes rapidly while the moderator conditions prior to the event are maintained for a much longer time. The

voiding of the coolant increases the reactor multiplication factor and reactivity which produces an increase in reactor power until the reactor is shut down by the action of the safety systems. The rise in the multiplication factor is mainly due to a small reduction in the resonance absorption and a small increase in the U^{238} fast fissions.

3. Technical possibilities to reduce CVR in HWRs

The coolant void reactivity can be reduced in HWR cluster type fuels by using in the central rod (or in the central seven rods) neutron absorbers or depleted fuel. In principle, different neutron absorbers might be used. Gadolinium (Gd), Dysprosium (Dy) and Erbium (Er) are some possibilities considered. Gd has a higher absorption cross section than Dy and burns up more rapidly with irradiation. In Canada, Dy was selected as the fuel absorber to reduce CVR for the Bruce reactors [5]. For this reason, Dy was also chosen for this preliminary study for Atucha-II, without doing any comparative analysis with other possible materials.

4. Basic assumptions for this study

The main idea was to include neutron absorbers in the central or in the seven central rods of the fuel assembly and using in them natural or depleted uranium.

Basically, the simplified approach used for this preliminary study was to use different fractions of Dy in the central rod (and in a few cases in the seven central rods) and evaluate the effect on the fuel exit burnup, the CVR, the relative power of the outer fuel rods, and the delayed neutron fraction.

In the Atucha-I reactor a program of gradual introduction of slightly enriched uranium (SEU), 0.85-wt% U^{235}, was successfully implemented between 1995 and 2000, which increased the exit burnup from 5900 to 11000 MWd/tU. In Embalse, a CANDU-6 nuclear reactor, a joint Nucleoeléctrica Argentina S. A. (NASA) - AECL study to evaluate a program of gradual introduction of SEU fuel with 0.9-wt% enrichment was finished in 2004 but a decision to carry it out was not taken yet. Due to the successful results of the SEU program in Atucha-I, the possibility of using slightly enriched fuel for Atucha-II was always considered (without burnable absorbers), not for the first core, but after the first period of operation as the programs carried out for Atucha-I and proposed for Embalse. The level of enrichment prior to this study was conservatively considered as 0.85% (as in Atucha-I).

To compensate for the burnup loss caused by the poison addition, slightly enriched uranium fuel was initially used with 1.2-wt% U^{235} enrichment. After the beginning of the study, the Fuel Procurement Department proposed to use enrichments not higher than 1-wt% U^{235}, because this level of enrichment might have less stringent requirements for transportation.

5 Description of the methodology and computer programs used for this study

5.1 Computers programs used

5.1.1. WIMS-D5

The code WIMS-D version 5b is a widely used multi-group energy lattice spectrum calculation code that provides a solution to the neutron transport equation in a fuel lattice cell. The main results given by the code are the multiplication factors, multi-group cross sections and fluxes in the cell regions. The description of the input data and some explanation of the calculation methods used by the code can be seen in [2]. The library used is a recently developed library (IAEA.LIB) based on ENDF/B-VI data in the WIMS Library Update Project (IAEA Coordinated Research Program) [6].

5.1.2 DRAGON

DRAGON is a versatile transport calculation program, developed by l'École Polytechnique of the Université de Montréal that can treat a variety of problems in one, two and three dimensions in different geometries using first-flight collision-probability methods. DRAGON has the advantage that can use different formats of cross section libraries including WIMS format. For the DRAGON calculations the IAEA.LIB library was also used.

The description of the collision probability calculation methods used in DRAGON can be seen in [7]. The use of the program is described in [3].

5.1.3 PUMA

The code PUMA 4 [5] is a 3-D, multi-group, diffusion reactor code that solves the diffusion equation by the finite difference method, using centre mesh finite differences discretization and can use response matrices for boundary conditions. It has the capability for fuel management, space dependent xenon, thermo-hydraulic feedback and space dependent kinetics.

Recently a cell heterogeneity correction was implemented to consider the variation of the flux inside the cell introducing in the finite difference equations cell heterogeneity discontinuity factors. This correction improved significantly the agreement of channel powers with MCNP in core benchmarks based on the Atucha-I core [8].

The reactor calculations were done with PUMA 4 using a 3-D model of the Atucha-II whole core with 2-group cross sections obtained from WIMS. The model has 16 mesh volumes per lattice cell and a detailed grid in X, Y and Z with 105, 121 and 31 planes respectively, with a boundary condition of flux 0 at the reflector outer radius plus 2-group extrapolated distances of $0.7104/\Sigma tr$. Guide tubes were included in the model.

5.1.4. MCNP5

The general-purpose Monte Carlo code MCNP5 version 1.20 [1] was used to model the whole Atucha-II reactor core, allowing the representation of each rod of each fuel cluster and the oblique control rods with great geometrical precision.

Because of this great flexibility for geometrical modelling, the "quasi-continuous" cross-section libraries and the use of probability distributions of the physical processes of individual neutron histories, MCNP was chosen as the reference program to benchmark the results of the other neutronic programs. The nuclear data library used was based on ENDF/B-VI, Release 6.

5.2. Description of the approach used for the calculations

Briefly, the preliminary studies done can be described as follows:

a) Comparison between results of slightly modified cell cases obtained with WIMS and DRAGON and reactor cases obtained with PUMA of Atucha-II with results of MCNP5, taken as benchmark comparisons.

b) Approximate evaluation of the CVRs of different fuel assemblies types (varying the enrichment, wt% Dy, poison in 1 or 7 rods) using cell calculations (see 5.2.2).

c) Selection of three alternatives to do reactor "Time Average" calculations and to determine the CVR. No adjustment of the fuelling strategy was done for this work, neither the incremental cross sections of the reactivity devices (control rods and guide tubes) were recalculated, assuming that the impact on void reactivity is small. The ones used correspond to the values available for NU fuel.

5.2.1 Atucha-II idealized cell for void reactivity comparisons with MCNP5

To benchmark the calculations of the CVR the variation of k-infinity with coolant voiding obtained with WIMS and DRAGON were compared with MCNP5 for a fresh fuel cell of Atucha-II slightly modified at full power conditions and without boron in coolant-moderator.

Pure Zr was utilized instead of Zircalloy-4 to reduce the processing time of MCNP5 and all temperatures were taken as 293.6 K = 20.45 °C, to avoid generating new MCNP5 thermal libraries for light and heavy water. However, the densities of the moderator and the coolant were taken as the values corresponding to full power conditions.

5.2.2 Atucha-II cell calculations with Dy to reduce CVR

These calculations were divided in three series. The main differences between them are the methodology of solving the transport equation and the fuel type. The first and second series were done with a simplified WIMSD5 model, using PIJ up to the isolation tube with 140 lines and 13 angles and the 1D method Perseus for the rest of the cell to reduce the processing time. The final series of calculations used PIJ up to cell boundary, with 400 lines and 29 angles. The geometrical description of the cell and the other material data are the same for all cases.

The burnup weighted cell average reactivity $\rho_{ex}(NU)$ of the NU cell is calculated with the expression:

$$\rho_{ex}(NU) = \frac{\int_0^{be(NU)} \rho(b,NU) b\, db}{\int_0^{be(NU)} b\, db} \quad (1)$$

where $be(NU)$ represents the exit burnup assumed for Atucha-II NU core, 7800 MWd/tU, and $\rho_{ex}(NU)$ represents the average reactivity worth of control rods and structural material not included in the cell calculations. Afterwards, with the assumption that the average reactivity worth of control rods and structural material is approximately maintained when changing to fuel type *Fi*, the exit burnup $be(Fi)$ for *Fi* fuel is estimated as the value of burnup that satisfies:

$$\rho_{ex}(NU) = \frac{\int_0^{be(Fi)} \rho(b,Fi) b\, db}{\int_0^{be(Fi)} b\, db} \quad (2)$$

For the initial comparison, reactor values of the void reactivity $\rho_v^R(Fi)$ and the delayed neutron fraction $\beta_{Nd}^R(Fi)$ for fuel type *Fi* are estimated as the average $\overline{\rho_v(Fi)}$ and $\overline{\beta_{Nd}(Fi)}$ of the values (obtained by cell calculations) between 0 and the cell estimated exit burnup *be(Fi)*,

$$\rho_v^R(Fi) \approx \overline{\rho_v(Fi)} = \frac{\int_0^{be(Fi)} \rho_v(b,Fi) b\, db}{\int_0^{be(Fi)} b\, db} \quad (3)$$

$$\beta_{Nd}^R(Fi) \approx \overline{\beta_{Nd}(Fi)} = \frac{\int_0^{be(Fi)} \beta_{Nd}(b,Fi) b\, db}{\int_0^{be(Fi)} b\, db}. \quad (4)$$

This assumption allows the comparison of the CVR values of the different fuel types only with the cell calculations, and the reactor calculations are done only for the cases where the CVR approaches the desired value (one dollar).

The small contribution of the delayed photoneutrons to the delayed neutron fraction was not considered as a conservative assumption.

5.2.3 Atucha-II core calculations ("Time Average")

The core "Time Average" calculations implemented in PUMA for Atucha type reactors are a simple way to obtain in one reactor calculation global core parameters averaged during a long time period when a given fuel movement strategy is used.

More specifically, the Atucha reactors have on power refuelling, and the fuel channels of the core are divided in paths C_k and each path is formed by two regions R_{kj}, which are sets of channels with the same average and exit burnup. For a given refuelling operation, first the path C_k is chosen, usually the one with the region R_{k1} that contains the FA with the highest burnup, and this FA is taken out of the reactor. Then the FA with highest burnup in R_{k2} is moved to fill the empty position in R_{k1}, and the empty channel in R_{k2} is filled with a fresh FA.

The "Time Average" core calculations receive as data the core geometry, the distribution of paths and regions of each path, and the exit burnup of the fuel in each path. The results are the core reactivity, the "Time Average" distributions of channel and linear power, the average burnup at which the fuel is transferred from R_{k2} to R_{k1}, and the burnup of each axial sector at the beginning and at the end of the period of residence. The "Time Average" channel or linear powers do not contain fuelling ripple, which has to be obtained through detailed refuelling simulations.

Four "Time Average" calculations were done, the NU reference case and three SEU 1.00-wt% U^{235} with NU and Dy_2O_3 in the central rod, with 3, 3.5 and 4-wt% of Dy.

5.2.4 Idealized Atucha-II core for reactor void reactivity comparison with MCNP5

To benchmark the calculations of reactor void reactivity the variation of k-infinity with coolant voiding obtained with WIMS-PUMA were compared with MCNP5 for a reactor model based on the cell of Atucha-II slightly modified mentioned above (see 5.2.1).

Only fresh fuel was considered, the rod insertions were the nominal ones, and without boron in the coolant-moderator.

6 Results

6.1 Comparison of cell void reactivity coefficient calculations of WIMS and DRAGON against MCNP

Table 1 shows the infinite multiplication constants of the reference and the void cases, and the void reactivity coefficients obtained with MCNP, WIMS and DRAGON with 69 and 172 group libraries. WIMS calculations of the void reactivity coefficients are the closest to MCNP values. WIMS and DRAGON void reactivity coefficients with the 172 energy group library seem to show better agreement with MCNP than the ones with the 69 energy group library.

	MCNP	WIMSD5 k-infinite		DRAGON k-infinite	
	k-infinite	69- groups	172-groups	69- groups	172-groups
Reference	1.14029 $\sigma = 0.00009$	1.13780	1.14108	1.13880	1.14074
Void	1.15406 $\sigma = 0.00007$	1.15227	1.15531	1.15044	1.15383
Δk-inf / k-inf$_{ref}$ (mk)	12.08 ± 0.11 [a]	12.71	12.47	10.22	11.48
Delta rho (mk)	10.46 ± 0.10 [b]	11.03	10.79	8.88	9.95

[a] $\Delta_{\Delta k} = \sqrt{(\sigma_{ref}^2 + \sigma_{void}^2)} \cdot \frac{k_{ref}}{k_{void}} \cdot 1000$ [b] $\Delta_{\Delta\rho} = \sqrt{\left(\frac{\sigma_{ref}^2}{k_{ref}^2} + \frac{\sigma_{void}^2}{k_{void}^2}\right)} \cdot 1000$ (68.27% confidence level)

Table 1: Comparison between cell coolant void reactivity coefficients obtained with different codes with MCNP5.

6.2 Cell calculations of different alternatives

Initially, some preliminary calculations, denominated as first series, were done with a simplified cell model, using PIJ up to the isolation tube in the WIMSD5 model and the 1D method Perseus for the rest of the cell to reduce the processing time.

Table 2 shows the main results of the NU reference case and several SEU (0.9 to 1.2 %w U^{235}) cases with different fuel compositions in the 7 central fuel rods. The maximum relative powers of the outer fuel rods are between 6.5 and 12.9 % higher than the NU case.

Simplified WIMS model (PIJ up to PT)	Fuel enrich. (wt%)	Dy in 7 central fuel rods (wt%)	Fuel enrich. in 7 central rods (wt%)	Average fuel exit burnup (MWd/tU)	Average void reactivity (mk)	Relative power of outer fuel rods	ΔAFB / ΔCVR [1] (MWd/tU / mk)	Average delayed neutron fraction
NU	0.711	--	0.711	**7800**	**9.63**	1.109	--	0.00524
SEU 1.20 %	1.20	--	1.20	**22833**	**9.55**	1.134	--	0.00493
SEU; 0.5% U^{235}	1.20	--	0.5	**19566**	**8.73**	1.221	3991	
SEU and 0.5% U^{235}, 0.3% Dy	1.20	0.3	0.5	**16485**	**6.90**	1.238	2398	
SEU and NU with 0.3 % Dy	1.20	0.3	0.711	**17540**	**7.23**	1.210	2281	
SEU and 0.5 % U^{235}, 0.6 % Dy	1.20	0.6	0.5	**13400**	**4.79**	1.252	1983	
SEU 0.95 %	0.95	--	0.95	**16126**	**9.57**	1.120	--	0.00496
SEU; 0.5% U^{235}	0.95	--	0.5	**13683**	**8.87**	1.189	3490	
SEU 0.90 %	0.90	--	0.90	**14620**	**9.45**	1.117	--	0.00498
SEU; 0.5% U^{235}	0.90	--	0.5	**12345**	**8.78**	1.181	1978	

[1] ΔAFB / $\Delta\rho$V = rate of the delta average fuel exit burnup to the delta average void reactivity (MWd/tU / mk).

Table 2: First series of cell calculations: Cases with SEU fuel, and Dy and depleted or natural U in 7 central fuel rods.

Table 3 shows the results of cases with only the central rod loaded with depleted fuel. The enrichment of the other rods varies between 1.00-wt% and 1.20-wt% of U^{235}. The percent of Dy in the central rod is 5-wt%, and two cases have 5-wt% of Dy as Dy_2O_3. In these cases with Dy, the average void reactivity is between 41 and 48 % of the reference case. The maximum relative powers of the outer fuel rods are up to 5.3 % higher than the NU case.

Simplified WIMS model (PIJ up to PT)	Fuel enrich. (wt%)	Dy in central fuel rod (wt%)	Fuel enrich. in central rod (wt%)	Average fuel exit burnup (MWd/tU)	Average void reactivity (mk)	Relative power of outer fuel rods	ΔAFB / ΔCVR [1] (MWd/tU / mk)	Average delayed neutron fraction
NU	0.711	--	0.711	**7800**	**9.63**	1.109	--	0.00524
SEU 1.20 %	1.20	--	1.20	**22833**	**9.55**	1.134	--	0.00493
SEU; 0.5% U^{235}	1.20	--	0.5	**22383**	**9.39**	1.143	2813	0.00493
SEU; 0.5% U^{235} with 5% Dy	1.20	5	0.5	**16860**	**4.63**	1.168	1214	
SEU 1.10 %	1.10	--	1.10	**20278**	**9.55**	1.128	--	0.00493
SEU; 0.5% U^{235} with 5% Dy	1.10	5	0.5	**14025**	**4.36**	1.162	1205	0.00529
SEU; 0.5% U^{235} with $Dy(5\%)_2O_3$	1.10	5	0.5	**14118**	**4.44**	1.162	1205	
SEU 1.00 %	1.00	--	1.00	**17561**	**9.56**	1.122	--	0.00494
SEU; 0.5% U^{235}	1.00	--	0.5	**17204**	**9.42**	1.130	2550	0.00495
SEU; 0.3% U^{235}	1.00	--	0.3	**17059**	**9.37**	1.134	2642	0.00495
SEU; 0.5% U^{235} with 5% Dy	1.00	5	0.5	**10800**	**3.95**	1.155	1205	

[1] $\Delta AFB / \Delta \rho V$ = rate of the delta average fuel exit burnup to the delta average void reactivity (MWd/tU / mk).

Table 3: First series of cell calculations: Cases with SEU fuel and Dy and 0.5-wt% depleted U in central rod.

The CVRs and the average fuel exit burnup of table 2 and 3 were compared to determine the advantages of using 1 or 7 rods with Dy. As the ratios of loss of exit burnup per mk of CVR reduction with Dy in the 7 central fuel rods are 1.5 times or higher than the ratios of the cases with only the central fuel rod with Dy it is more convenient to use Dy only in the central fuel rod. Another disadvantage of the use of 7 central rods with Dy is that the maximum relative powers of the outer rods are higher than the ones with Dy only in the central rod.

The use of depleted fuel (without Dy) in the central rod has a much smaller effect and also has a higher ratio of loss of burnup per mk of CVR reduction so it is not a useful alternative.

The second series of calculations are the cell cases with 1-wt% (the level of enrichment chosen to finish the study) SEU fuel, with Dy_2O_3 in the central rod, the only different fuel rod.

Table 4 shows the results of 1.00-wt% cases. The central fuel rod has NU, 0.5-wt% or 0.3-wt% depleted U^{235} fuel and Dy_2O_3, with 3, 4 or 5-wt% of Dy. The case with lower average void reactivity, 3.94 mk, is the one with 0.3-wt% of U^{235} and 5-wt% of Dy in the central fuel rod, and this represents 41.2% of the void reactivity of the 1.00-wt% U^{235} reference case. The case with the highest void reactivity, 6.27 mk, is the one with NU and 3-wt% of Dy in the central fuel rod, which is 65.6 % of the 1.00-wt% U^{235} reference case. The maximum relative powers of the outer fuel rods are up to 4.4 % higher than the NU reference case.

Simplified WIMS model (PIJ up to PT)	Fuel enrich. (wt%)	Dy in central fuel rod (wt%)	Fuel enrich. in central rod (wt%)	Average fuel exit burnup (MWd/tU)	Average void reactivity (mk)	Relative power of outer fuel rods	ΔAFB / ΔCVR [1] (MWd/tU / mk)	Average delayed neutron fraction
NU	0.711	--	0.711	**7800**	**9.63**	1.109	--	0.00524
SEU 1.00 %	1.00	--	1.00	**17561**	**9.56**	1.122	--	0.00494
SEU; 0.5 % U^{235}	1.00	--	0.5	**17204**	**9.42**	1.130	2550	0.00495
SEU; 0.3 % U^{235}	1.00	--	0.3	**17059**	**9.37**	1.134	2642	0.00495
SEU and NU with $Dy(3\%)_2O_3$	1.00	3	0.711	**13329**	**6.27**	1.146	1286	0.00519
SEU; 0.5 % U^{235} with $Dy(3\%)_2O_3$	1.00	3	0.5	**13168**	**6.18**	1.149	1300	0.00520
SEU; 0.3 % U^{235} with $Dy(3\%)_2O_3$	1.00	3	0.3	**13014**	**6.10**	1.152	1314	0.00520
SEU; 0.5 % U^{235} with $Dy(4\%)_2O_3$	1.00	4	0.5	**12004**	**5.11**	1.152	1249	0.00528
SEU; 0.3 % U^{235} with $Dy(4\%)_2O_3$	1.00	4	0.3	**11846**	**5.01**	1.155	1256	0.00529
SEU; 0.5 % U^{235} with $Dy(5\%)_2O_3$	1.00	5	0.5	**10905**	**4.05**	1.155	1208	0.00537
SEU; 0.3 % U^{235} with $Dy(5\%)_2O_3$	1.00	5	0.3	**10744**	**3.94**	1.157	1213	0.00538

[1] $\Delta AFB / \Delta \rho V$ = rate of the delta average fuel exit burnup to the delta average void reactivity (MWd/tU / mk).

Table 4: Second series of cell calculations: Cases with 1.00-wt% SEU fuel and Dy_2O_3 and depleted or natural U in central rod.

For the final series of cell calculations the more detailed cell model (PIJ up to cell boundary, 400 lines and 29 angles) was used with 1-wt% SEU fuel and NU with Dy_2O_3 (3, 3.5 or 4-wt% Dy) in the central rod, and can be seen in table 5. The NU reference case has an average void reactivity of 9.83 mk for 7800 MWd/tU and an average delayed neutron fraction (without photoneutrons) of 0.00525. The 1.00-wt% U^{235} reference case, with the same excess reactivity, has an exit burnup of 17433 MWd/tU and an average void reactivity of 10.03 mk.

The 1-wt% SEU case with NU and $Dy(3\text{-wt}\%)_2O_3$ in the central fuel rod has an exit burnup of 13275 MWd/tU, an average void reactivity of 6.46 mk and an average delayed neutron fraction (without photoneutrons) of 0.00520. The same case but with 3.5-wt% of Dy in the central fuel rod has an exit burnup of 12690 MWd/tU, an average void reactivity of 5.89 mk and an average delayed neutron fraction of 0.00525. The case with 4-wt% of Dy has an exit burnup of 12121 MWd/tU, an average void reactivity of 5.32 mk and an average delayed

neutron fraction of 0.00529. The maximum relative powers of the outer fuel rods are 3.2, 3.4 and 3.6 % higher than the NU reference case. The maximum relative powers rises when the Dy fraction in the central rod or the SEU enrichment of the others fuel rods are increased.

Detailed WIMS model (PIJ up to cell boundary)	Fuel enrich. (wt%)	Dy in central fuel rod (wt%)	Fuel enrich. in central rod (wt%)	Average fuel exit burnup (MWd/tU)	Average void reactivity (mk)	Relative power of outer fuel rods	ΔAFB / ΔCVR [1] (MWd/tU / mk)	Average delayed neutron fraction
NU	0.711	--	0.711	**7800**	**9.83**	1.107	--	0.00525
SEU 1.00 %	1.00	--	1.00	**17433**	**10.03**	1.120	--	0.00496
SEU and NU with $Dy(3\%)_2O_3$	1.00	3	0.711	**13275**	**6.46**	1.143	1165	0.00520
SEU and NU with $Dy(3.5\%)_2O_3$	1.00	3.5	0.711	**12690**	**5.89**	1.145	1146	0.00525
SEU and NU with $Dy(4\%)_2O_3$	1.00	4	0.711	**12121**	**5.32**	1.147	1128	0.00529

[1] ΔAFB / ΔρV = rate of the delta average fuel exit burnup to the delta average void reactivity (MWd/tU / mk).

Table 5: Final series of cell calculations: Cases with 1-wt% SEU fuel and Dy_2O_3 and depleted or natural U in central rod, detailed model.

6.3 "Time Average" core calculations

Table 6 shows the main results of the core calculations. For a "Time Average" NU core with an exit burnup of 7800 MWd/tU, the reactor void reactivity coefficient is 10.29 mk and the maximum channel power (MCP) is 6.55 MW.

PUMA Code	Reference k-effective	Void k-effective	Void delta k-eff (mk) [a]	Void delta rho eff (mk) [b]	Exit burnup	MCP Ref (MW)	MCP Void (MW)
NU	1.002567	1.013024	10.32	**10.29**	**7800**	6.55	6.48
SEU 1.00% and NU with $Dy(3\%)_2O_3$ in central fuel rod	1.002471	1.008712	6.19	**6.17**	**13000**	7.39	7.18
SEU 1.00% and NU with $Dy(3.5\%)_2O_3$ in central fuel rod	1.002023	1.007767	5.70	**5.69**	**12500**	7.25	7.04
SEU 1.00% and NU with $Dy(4\%)_2O_3$ in central fuel rod	1.002554	1.007815	5.25	**5.21**	**11900**	7.09	6.89

[a] Computed as $\Delta k = (k_{void} - k_{ref})/k_{ref} \cdot 1000$

[b] Computed as $\Delta\rho = (k_{void} - k_{ref})/(k_{ref} \cdot k_{void}) \cdot 1000$

Table 6: "Time Average" Atucha-II core calculations of void reactivity.

The reactor case with SEU 1.00-wt% U^{235} and NU with Dy(3-wt%)$_2O_3$ in the central fuel rod, with an exit burnup of 13000 MWd/tU has a reactor void coefficient of 6.17 mk, but the MCP increases until 7.39 MW. The reactor case with 3.5-wt% of Dy and 12500 MWd/tU has a reactor void coefficient of 5.69 mk.

The reactor case with the amount of Dy increased up to 4-wt% and an exit burnup of 11900 MWd/tU has a reactor void coefficient of 5.21 mk, which is about the delayed neutron fraction. The MCP is 7.09 MW. An adjustment in the fuel management strategy (path and regions) are required to increase the flattening of the power distribution and have maximum channel powers below 6.93 MW, which is the channel power limit for the central region. However, the adjustment of the fuel strategy is considered to have a small effect on the void reactivity and the optimization of the fuel strategy will not be studied in this paper.

6.4 Comparison of reactor void reactivity coefficient calculations of WIMS-PUMA against MCNP

Table 7 shows the comparison of the k-effective and the reactor void coefficient obtained with PUMA/WIMS and MCNP5. The void reactivity calculated by PUMA is very close to the MCNP5 values. The difference of the void reactivity coefficient between PUMA (9.62 mk) and MCNP (9.38 mk) is less than 0.3 mk.

Reactor code	Reference k-effective	Void k-effective	Void delta k-eff (mk)	Void delta rho eff (mk)
PUMA	1.102856	1.114684	10.61	9.62
MCNP5	1.10381 σ = 0.00005	1.11536 σ = 0.00005	10.46 [a] ± 0.071 [c]	9.38 [b] ± 0.063 [d]

[a] Computed as $\Delta k = (k_{void} - k_{ref})/k_{ref} \cdot 1000$ [b] Computed as $\Delta\rho = (k_{void} - k_{ref})/(k_{ref} \cdot k_{void}) \cdot 1000$

[c] Computed as $\Delta_{\Delta k} = \sqrt{(\sigma_{ref}^2 + \sigma_{void}^2)} \cdot k_{ref}/k_{void} \cdot 1000$ (68.27% confidence level)

[d] Computed as $\Delta_{\Delta\rho} = \sqrt{\left(\sigma_{ref}^2/k_{ref}^2 + \sigma_{void}^2/k_{void}^2\right)} \cdot 1000$ (68.27% confidence level)

Table 7: Idealized Atucha-II fresh core. Comparison between PUMA core calculations with coolant and coolant voided with MCNP5.

7. Conclusions

This report presents a preliminary study to modify the Atucha-II fuel assembly to reduce the CVR by using SEU fuel in all the rods except for the central or seven central rods, and natural or depleted uranium with dysprosium in these rods. It also presents some comparisons of cell and core void reactivity calculations of a slightly modified Atucha-II cell with Monte Carlo calculations with MCNP5.

The main conclusions are the following:

a) The losses of exit burnup per mk of reduction of void coefficient for the cases with the 7 central rods with Dy are 1.5 or higher than the ratios of the cases with only the central fuel rod with Dy. In addition, the maximum relative powers of the outer rods of the former cases are higher than the ones with only the central fuel rod with Dy. Therefore, it is more convenient to use Dy in only the central fuel rod. Using depleted U for the central rod without Dy does not have a significant effect on CVR.

b) The average delayed neutron fraction (without photoneutrons) rises gradually with the increase of the percentage of Dy and decreases slightly with the increase of the enrichment of the fuel. In the cases chosen for the reactor calculations, the average delayed neutron fractions are around the one of the NU reference case.

c) The relative powers of the outer fuel rods of the cases used in the reactor calculations increases up to 3.61 % (1.147) higher than the NU reference case (1.107).

d) The core (Time Average) calculation results indicate that using SEU (1-wt% U^{235}) fuel and NU with 4-wt% of Dy in Dy oxide, the CVR can be reduced to about from 10.3 to 5.25 mk, with an estimated exit burnup of 11900 MWd/tU. However, the case with SEU (1-wt% U^{235}) fuel without Dy has an exit burnup of 17400 MWd/tU, which implies a burnup loss of about 1100 MWd/tU per mk of CVR reduction.

e) Comparisons with MCNP5 of CVRs obtained with cell calculations with WIMS and core calculations with PUMA using a modified Atucha-II cell with MCNP5 to benchmark the calculation methodology showed very good agreement.

8. References

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