Hindawi Publishing Corporation
Science and Technology of Nuclear Installations
Volume 2011, Article ID 194650, 15 pages
doi:10.1155/2011/194650

*Research Article*

# A New Fuel Design for Two Different HW Type Reactors

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Received 26 August 2010; Accepted 6 December 2010

Academic Editor: Juan Pablo Ordóñez

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A new fuel element (called CARA) designed for two different heavy water reactors (HWRs) is presented. CARA could match fuel requirements of both (one CANDU and one unique Siemens's design) Argentine HW reactors. It keeps the heavier fuel mass density and hydraulic flow restriction in both reactors together with improving both thermomechanic and thermalhydraulic, safety margins of present fuels. In addition, the CARA design could be considered as another design line for the next generation of CANDU fuels intended for higher burnup.

## 1. Introduction

Argentina has two pressurized heavy water reactor (PHWR) nuclear power plants (NPPs) in operation (Atucha I and Embalse) since 1974 and 1984, respectively, operated by the same national utility (N.A.S.A.) and has another one under construction projected to be connected to the grid in 2012 (Atucha II). Although both of them are cooled by pressurized heavy water, designed to be fuelled with natural uranium, and moderated with heavy water, they have strongly different designs. Embalse is a standard CANDU-6 [1, 2], horizontal pressure-tubes typical Canadian reactor. Atucha I and II have a unique Siemens' design: vertical fuel channels inside a pressure vessel reactor [3]. Fuels for Atucha I and II have small dimensional differences for the rod diameter and structural spacer grids.

Both nuclear power plants use on-line refuelling, but they differ in the length and number of their fuel elements (FEs). Embalse uses a short FE with a length of 0.5 meter (see Figure 1) [4], and, so, the horizontal 6-meter-long fuel channel is filled with twelve FEs. The vertical channel of Atucha is filled by one FE of 5.3 meters active length (see Figure 2) [3]. Both fuels use 37 fuel rods arranged in a circular cluster array but with different designs of cladding: (1) Atucha has self-supporting rods and one structural rod without fuel, following PWR design [5]; (2) Embalse has collapsible rods, following the well-known CANDU design [4].

The Atucha's fuel uses structural rigid spacer grids at intermediate positions like in PWRs [5]. The fuel of Embalse follows the principle of the CANDU series: a cluster of collapsible rods supported on its extremes by two structural plates (end plates). It uses middle-plane appendages welded on cladding to avoid fretting between contiguous rods (spacers) and between rods to pressure-tube wall (bearing pads). Both reactors use 37 fuel rods of similar diameters (1.0% greater than the Embalse one), and, therefore, they have similar uranium-mass linear density. Their fuel channel diameters are similar, being slightly greater in Atucha (4%) than that in Embalse; and therefore, they have hydraulic similarity too. Unfortunately, their fuel cost is not similar, being the fuel cost in Atucha higher than that in Embalse.

The fuel cost of Atucha I electrical energy was strongly reduced by the use of slightly enriched uranium (SEU) since 1998 up today. The burnup of design (for natural uranium) achieved was around 6,000 MWd/THM, but through this program it was increased up to 11,400 MWd/THM by using an enrichment level of 0.85% $^{235}$U. This program illustrates the efforts pushed by the markedly high fuel costs of Atucha I. This economic performance is mainly due to the unique characteristics of the Siemens' design of Atucha, in which

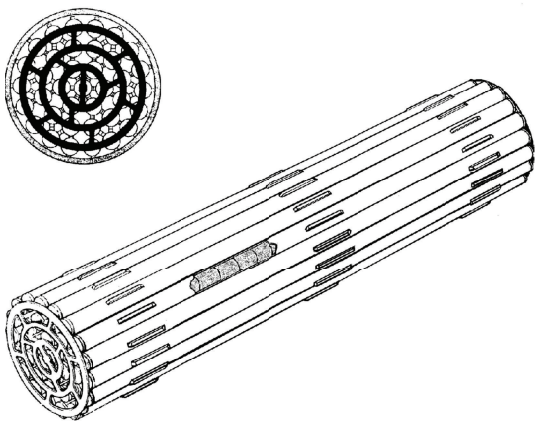

Figure 1: CANDU fuel element.

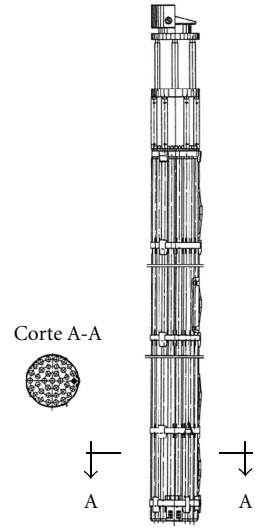


Figure 2: Atucha I fuel element.

a "PWR fuel" is used for a natural-uranium reactor. The low scale of the fuel supplier company (CONUAR), with two different manufacturing lines for feeding only two medium-size reactors, appears like the main drawback of the Argentine nuclear fuel cycle, regarding its economical competitiveness. This performance could be ideally improved by using the same FE on both reactors.

A project was started in 1997 for dealing with this challenge: to design a single FE for fuelling both Argentine reactors and at the same time enhancing their fuel performance, by considering the improvement reached by the use of SEU [6]. So, let us describe both FEs involved. While Embalse's fuel has a robust and simple design, the Atucha's fuel has greater fuel costs due to its more complex mechanical solution related to its design of a long bundle. From this point, the CARA was designed (Advanced Fuel for Argentine Reactors, in Spanish) within "a CANDU concept," that is by using collapsible short rods.

The 37-rod fuel has been the commercial technology for CANDU-6 for the last thirty years [4]. It was designed for natural-uranium low burnup (6,700 MWd/THM). This technology is now evolving towards advanced fuel designs in order to get extended burnup by using SEU [6, 7]. Nowadays, a new generation of FE (CANFLEX) is being developed by AECL jointly with KAERI, expecting to reach higher burnup with higher fuel rod number and consequently lowering the central temperature [8].

By considering the similarities (geometric, hydraulic, and neutronic) between Atucha and Embalse, the feasibility of filling the Atucha fuel channel with ten FE of Embalse will be considered, keeping the uranium mass and the hydraulic similarity and fastening the assembly by means of a circumferential external tube. Then, after having demonstrated this point, a completely new fuel design will be developed under this ideal (but then realistic) scenario.

## 2. Preliminary Feasibility Analysis

In order to carry out a preliminary feasibility assessment of the Embalse-based concept for designing the new fuel element, the behaviour of a CANDU-6 fuel chain into Atucha was studied. This theoretical exercise is useful for understanding the handicaps and drawbacks of CANDU fuel under the Atucha operating conditions.

*2.1. Hydraulic Analysis.* For performing this preliminary hydraulic analysis, a one-dimensional model was used. The hydraulic modelling of the CANDU-6 rod was done by extracting their concentrated and distributed pressure-drop coefficients, obtained from critical-heat flux (CHF) experimental data performed in a Freon loop [9] and from monothermic endurance tests performed in a water loop [10]. Using pressure drop data along the fuel channel [9], the distributed and concentrated pressure drop terms were calculated and checked against published [10] and other data obtained at Argentine test facilities [11]. The concentrated pressure-drop terms are produced on every pair of contiguous end plates and middle planes (with spacers and bearing pads) of every FE and also on the fuel chain inlet and outlet. The distributed pressure-drop term is related to friction along the whole fuel channel.

First, by taking into account the distributed pressure drop along one FE, the Darcy coefficient ($f$) and the equivalent cladding roughness ($\varepsilon$) were calculated by using the flow Reynold number (Re), showing that the distributed pressure drop can be evaluated by means of simple one-dimensional correlation for circular tubes [12]. Subtracting the distributed term from pressure drop measurements between every local restrictions, the hydraulic coefficient of spacer ($K_{sp}$), inlet and outlet channel ($K_i$ and $K_o$), and end-plates junction ($K_{end}$) could be determined, as showed in Table 1. The hydraulic restriction of end-plates junction is a function of its misalignment angle, in accordance with this degree of freedom characteristic of CANDU reactors, in which FEs rest on the channel inner wall placed randomly in the fuel chain.

Table 1: Friction terms of CANDU fuel in Embalse conditions.

| | |
|---|---|
| $F$ | 0.01505 |
| $\varepsilon$ | 2.16 $\mu$m |
| Re | 513,000 |
| $K_{sp}$ | 0.12 |
| $K_i$ | 0.39 |
| $K_o$ | 0.36 |
| $K_{end}$ | 0.34 (full alignment, minimum)<br>0.60 (average misalignment)<br>0.72 (full misalignment, maximum) |

Table 2: Hydraulic parameters of the hottest (design case) fuel channel.

| Reactor Data | | |
|---|---|---|
| | Embalse [13] | CANDU in Atucha I |
| Mass flow | 23.94 Kg/s | 32.90 Kg/s |
| Average liquid density ($\rho$) | 800 Kg/m$^3$ | 832 Kg/m$^3$ |
| Channel diameter | 103.8 | 108.2 |
| Fuel chain length ($L$) | 6 m | 5 m |
| FEs chain number ($N$) | 12 | 10 |
| Results | | |
| Average velocity ($V$) | 8.57 m/s | 9.36 m/s |
| Hydraulic diameter ($D_h$) | 7.56 mm | 9.08 mm |
| Fuel chain pressure drop ($\Delta p$) | 608 KPa | 539 KPa |

Hence, for a given $N$-elements FE chain and a coolant flow, the pressure drop ($\Delta p$) can be estimated from the hydraulic coefficients that characterize the CANDU fuel by using the classic hydraulic one-dimensional model, by(1)

$$\Delta p = \frac{1}{2}\left[K_i + K_o + (N-1) * K_{\text{end}} + N * K_{\text{sp}} + \frac{f * L}{D_h}\right] * \rho * V^2, \tag{1}$$

where $L$, $D_h$, $V$, and $\rho$ are the fuel chain length, hydraulic diameter, average flow velocity, and liquid density, respectively.

The model and the hydraulic parameters of the CANDU FE were validated with the experimental results for a 12-FE chain for the most probable misalignment CANDU-6. The model predictions and experimental data are within a 10% error bandwidth.

By using this model (1) for appropriate flow conditions, and by considering the endplate average misalignment value for $K_{\text{end}}$, the fuel pressure drop was calculated for Embalse and for this preliminary Atucha case study filled with ten CANDU FEs (see Table 2). For the Atucha reactor conditions, a two-phase correction is not necessary since the flow remains in single phase along the whole channel and, so, this model can be directly used.

The estimated pressure drop obtained (by means of this conservative homogeneous model) with ten CANDU FEs is lower than the actual pressure drop of Atucha I (600 KPa) [14]. This pressure drop margin enables us to design a circumferential external tube as the assembly system for Atucha. So, the mechanical compatibility with Atucha's refuelling machine is ensured (note: the Atucha's fuel is hanged up from the top pressure vessel lid, inside the vertical fuel channels).

Table 3: Atucha pressure drop channel for average misalignment or fully aligned FEs, predicted for ten CANDU fuels assembled by means of a circumferential tube.

| Tube thickness (mm) | Velocity (m/s) | $\Delta P$ average (KPa) | $\Delta P$ minimum (KPa) |
|---|---|---|---|
| 0.0 | 9.36 | 539 | 456 |
| 0.5 | 9.75 | 602 | 512 |
| 1.0 | 10.17 | 675 | 576 |

In order to study the hydraulic compatibility of an assembly system, a 1 mm-thick solid tube was adopted (a realistic input value considering its mechanical feasibility) deployed at the maximum external radius, and, so, the diameter (106.2 mm) of channel is reduced. By using this hydraulic model, a new pressure drop value was obtained, which is slightly higher (576 KPa) but still compatible with the reactor conditions, in the case of (minimum pressure drop) fully-aligned FE. In Table 3, the overall fuel channel pressure drop is shown for ten CANDU-37 FEs assembled inside a circumferential tube estimated as function of the tube thickness and the alignment angle (considering average misalignment and fully aligned conditions). It shows that circumferential tube thicknesses up to 0.5 mm are compatible with a chain of randomly misaligned fuels (the simplest mechanical design) but up to about 1.2 mm thickness if fully alignment is imposed, which in turn implies a more complex mechanical design.

Considering now its thermal hydraulic behaviour, this basket reduces the circumferential "water bypass" originated by the greater channel diameter of Atucha I. This water bypass decreases the overall flow restriction, particularly at the end-plates junction, but in turns it reduces the flow within inner subchannels, a bad behaviour for the cooling of rods. This kind of analysis must be quantified on a more detailed study performed by using a subchannel numerical code or by means of experimental data, since it implies momentum and energy balances between coupled subchannel flows. This analysis was performed using the COBRA code, as it will be shown in Section 2.3.

*2.2. Neutronic Analysis.* Both Argentine reactors are designed for natural uranium fuel and heavy water coolant and moderator, having a core built by many channels with similar pitch and length. Besides, their fuels have similar diameters and an equal number of fuel rods with just slightly different diameters and thus have similar linear mass densities (see Table 4). These core and fuel design similarities allow to consider, at this early state of the CARA development, that there could exists a neutronic compatibility between both reactors.

The core extraction burnup could be estimated for continuous refuelling core (like Atucha I and Embalse)

Table 4: Atucha I and CANDU-6 Core data.

| | Atucha I | Embalse |
|---|---|---|
| Thermal power [MW] | 1.179 | 1.992 |
| Fuel channels | 250 | 380 |
| Core length [m] | 5.3 | 5.95 |
| Core diameter [m] | 4.4 | 5.9 |
| Rods per FE | 37 | 37 |
| Fuel rod linear Uranium density [kg$UO_2$/m] | 0.91 | 1.16 |

if the cell calculation is performed with geometrically buckling (including reflector saving to achieve core length and diameter) by calculation of the extraction burnup, as the burnup that equalizes the area between a given excess reactivity for the core and the reactivity calculated with the code [14]. The same code and method have been used in order to obtain the reactivity for each fuel and its reactor. As the burnup depends on the core reactivity value used in the calculation, the value for each reactor was calculated by the present fuel and present extraction burnup, also calculated with the same code, nuclear data, and number of energy group and cell options.

Considering the fuel rod characteristics, the corresponding power densities, dimensions, and geometrical buckling were used as the WIMS D5 input to estimate the CANDU in Atucha I neutronic behaviour (see Table 5) [14–16]. In particular, the radial buckling was not changed, as it is related with the core radii. The change in core length was considered for the axial buckling calculation and the power density was scaled by considering the difference in fuel rods and $UO_2$ mass. The maximum linear power ratio was analyzed in relation to the maximum power peaking factors for the four-pin annulus during the burnup.

By comparing the results shown in Table 5, under Atucha I conditions, the CANDU fuel has higher linear power values than the Atucha natural uranium (NU) (6%) and similar in respect to the CANDU-6 reactor. Moreover, the use of SEU in Atucha I enables the power radial core flattening and the reduction of the maximum linear power ratio.

*2.3. Thermohydraulic Analysis.* COBRA is a well-known subchannel code extensively used for CHF calculations for PWR and BWRs [5, 17–21]. Besides, it has already been demonstrated that COBRA can be applied to PHW reactors [22]. It provides a useful tool for modelling the subchannel geometry, by means of computing mass, momentum, and energy balances for every subchannel at different axial steps by a finite difference scheme in order to solve the axial and cross-flows at every subchannel.

By using the COBRA capabilities, the DNB margin for a CANDU fuel chain filling the Atucha fuel channel was calculated, comparing it with the actual Atucha fuel. It was shown that the geometry of CANDU FE is compatible with the dimension of the Atucha I fuel channel. To avoid the water bypass produced by the smaller fuel diameter, an outer tube is required. Two different conditions were considered, that is, with a 1 mm thickness tube and without any tube, for the CANDU-37 fuel chain. The results are presented in Table 6.

The thermohydraulic margin estimated for a CANDU chain in Atucha I is better than the actual fuel. This margin is increased when an assembly system is placed in the circumferential "water bypass." The COBRA also calculates the channel pressure drop for average end-plate misalignment (see Table 6), and, observing these values, it verifies the results from our simple one-dimensional hydraulic model.

The pressure drop of the Embalse fuel chain is similar to the Atucha I FE. The use of the outer tube increases the pressure drop but increases the DNBR margin. In any case, even in the most conservative conditions (with outer water bypass), this margin is still better than the Atucha condition.

*2.4. Mechanical Analysis.* All CANDU fuels use a high number of weldings (pads and spacers) on the cladding to ensure the clearance between neighbour rods. It implies higher quality assurance costs to certify that at the end of the assembling all the rods and their pads are in the right position.

The contact between CANDU pads limits rod displacement among neighbours, being these ones the only restrictions to rod displacing, and, so, the rods bow under axial load. But due to the vertical position in Atucha, their axial and turbulence loads are higher than Embalse, thus the CANDU mechanical solution becoming inappropriate for Atucha. Therefore, the mechanical requirements of Atucha will be the design base for the new FE. Regarding this point, it is clear that CANDU design does not include a circumferential fastening device. So, the use of the proposed circumferential external tube could help to fit the CANDU fuels in the vertical channels of Atucha. On the other hand, by considering now the fuel elements of PWR (designed for vertical channels), which use spacer grids in order to keep the relative fuel rods positions, there is no need for welding on the clad sheath. These fuels reaches burnup several times higher than CANDU ones, designed for natural uranium [5].

*2.5. Dynamical Analysis.* The most important dynamical mechanism in CANDU-6 and Atucha I are flow-induced vibrations, mainly turbulence [23]. The vibration of the fuel rods could induce fuel rods or assembly failures through several mechanisms, the most important among them being mechanical wear, fretting, and fatigue cracking [24].

The dynamical behaviour of CANDU fuel under both reactor conditions can be studied by comparing their flow-induced excitations, which is proportional to the product $\rho * V$, while its deviation is proportional to the flow turbulence intensity, given by the Reynolds number [23, 24]. Table 7 shows these parameters for the CANDU-37 FE in Embalse, the original Atucha FE in Atucha I, and ten FE chain of CANDU-37 FE with three different assembly tube thicknesses for Atucha I conditions. It can be seen in Table 7 that the original Atucha I has nearly the same $\rho * V$ product as CANDU-37, both in their original reactors, but a significant difference (43% higher) for the Reynolds

Table 5: Neutronic maximum rod power ratio and burnup for Embalse and Atucha I.

| Characteristic | CANDU 37 | Atucha I-NU | Atucha I-SEU (0.85%) | CANDU in Atucha I |
|---|---|---|---|---|
| Burnup [MWd/TonU] | 7.300 | 5.900 | 11.800 | 5.700 |
| Core peak factor | 1.843 | 2.03 | 1.87 | 2.03 |
| Maximum bundle Peak factor | 1.1261 | 1.096 | 1.0996 | 1.1234 |
| Maximum rod Linear power ratio [W/cm] | 595 | 550 | 508 | 586 |

Table 6: Thermohydraulic margin and pressure drop model comparison.

| Fuel element and reactor | DNBR | $\Delta P$ 1-D model (KPa) | $\Delta P$ COBRA (KPa) |
|---|---|---|---|
| Atucha FE in Atucha I | 3.41 | 608 | 601 |
| 10 Embalse FE in Atucha I | 3.88 | 539 | 518 |
| 10 Embalse FE + tube of 1 mm in Atucha I | 4.14 | 675 | 630 |

number. When the CANDU-37 conditions at Atucha I is compared with respect to CANDU-6, the Atucha I flow excitation is appreciably higher ($\rho * V$ increasing up to 123%, Reynolds increasing up to 133%) than the operation condition of CANDU fuel.

*2.6. Thermomechanical Analysis.* The thermomechanical compatibility between both reactor conditions can be studied in a first-order analysis by studying their central pellet temperatures and power history. The steady-state central pellet temperature is proportional to the linear power. In Section 2.2, it was shown that for the CANDU FE inside the Atucha I operating condition, the estimated rod maximum linear power ratios were similar to those in Embalse (CANDU-6 reactor), but the power transient during Atucha refuelling is higher than in Embalse [14, 15]. Therefore, the thermomechanical requirements for fuel rods in Atucha I will be adopted as the design base for the new FE. This implies that a new CANDU fuel must be an enhanced design, thus lowering its linear power density. But this requirement does not match easily with others boundary conditions, as keeping the total hydraulic restriction [14].

*2.7. CANDU Fuel Comparison.* The development of a new fuel requires improving its thermohydraulic, neutronic, mechanical, and thermomechanical behaviour. It should be pointed out the opposite trends between these phenomena. For example, if the rod cluster is more spread out (by using smaller diameters) but keeping the total fuel mass (by increasing the rod number), its hydraulic restriction is increased and, consequently, the coolant flow (and so, thermohydraulic safety margins) would be decreased.

A dimensionless parameter, $Ndg$, can be a useful tool in order to compare the "dispersion grade" of different fuel element designs. This parameter is defined as the heated and fuel cross-section areas ratio, normalized for the heated length per meter of the fuel channel. At higher values of this parameter better, thermohydraulic and thermomechanical behaviours can be obtained. The definition of $Ndg$ is presented in(2)

$$Ndg = \frac{\pi N_b \phi_b L_h}{(\pi/4) N_b \phi_p^2}, \tag{2}$$

where $\phi_b$: rod outside diameter, $\phi_p$: pellet diameter, $L_h$: heated length per channel length unit, and $N_b$: number of fuel rods.

By evaluating the technological evolution of the fuel element series for the CANDU reactors, a continuous growing on $Ndg$ values can be noted, from the first seven-rod (N.D.P. reactor) fuel element up till now, considering the present design (CANFLEX). Table 8 shows this trend, using published data [1, 25].

This trend was also observed by considering several manufacturers for PWR fuel elements. The historical evolution of this technology has also followed an increase in the number of the fuel rods per element, from $14 \times 14$ to $18 \times 18$ squared arrays [5, 25].

## 3. Cara Development

*3.1. Initial Criteria for the New Bundle Design.* The new fuel, called CARA, must keep the same operational conditions for both NPPs. They are the coolant flow, total hydraulic channel pressure drop, and the mechanical compatibility with the refuelling machine of each NPP.

The feasibility of our fuel concept has already been analyzed in previous sections, based on the hydraulic, thermohydraulic, and neutronic compatibilities; the need to enhance their mechanical and thermomechanic performance was also shown. Now, as a starting point for the CARA development the CARA fuel has been designed to improve the major fuel performance of both reactor types. This FE was set up with the following objectives.

(1) Mechanical compatibility with both NPPs.

(2) Hydraulic compatibility (hydraulic pressure drop of each NPP core).

(3) Just one fuel rod diameter.

(4) Higher thermohydraulic safety margins.

(5) Lower fuel pellet-centre temperatures.

(6) Higher linear uranium mass density.

(7) No welding on cladding sheath.

(8) Allowing extended burnup.

(9) Lower energy fuel cycle cost.

Table 7: Flow excitation parameters in both reactors.

| Bundle | Reactor | Tube thickness (mm) | Velocity (m/s) | $\rho * V$ (Kg/m$^2$ s) | Re ($\times 10^5$) |
|---|---|---|---|---|---|
| CANDU-37 | CANDU 6 | — | 8.57 | 6.859 | 5.13 |
| Atucha I | Atucha I | — | 7.78 | 6.477 | 7.34 |
| CANDU-37 | Atucha I | 0.0 | 9.36 | 7.790 | 6.80 |
| | | 0.5 | 9.75 | 8.116 | 6.81 |
| | | 1.0 | 10.17 | 8.466 | 6.83 |

Table 8: Dispersion grade of the CANDU fuel element series.

| Fuel element type | $N_b$ | $Ndg$ |
|---|---|---|
| N.D.P. | 7 | 176 |
| Douglas Pt. | 19 | 295 |
| Pickering | 28 | 302 |
| Bruce | 37 | 354 |
| Canflex | 43 | 377 |

But these objectives go in opposite directions: for example, increasing the number of fuel rods increases the heated perimeter, as a consequence, increases the hydraulic pressure drop due to the distributed friction, and increases the number of welding appendages. Thus, the need to keep similar core pressure drops leads to the CANFLEX solution [26] that looses the possibility of using a single fuel rod diameter, in order to keep the hydraulic cross-section. Moreover, CANFLEX keeps welding pads in the clad and even increases its number, which is not desirable for extending burnup [5]. Hence, it is clear that to simultaneously solve these conditions, the CARA fuel must explore new options.

The key of CARA design is to double the length of present CANDU fuels, eliminating in this way an end-plates junction. This solution is compatible with CANDU refuelling machine (that manages the FE always by pairs) and enables

(i) eliminating the intermediate end-plates and hence their local pressure drop,

(ii) using this handicap to balance the whole hydraulic restriction (#2) at he same time increasing the heated perimeter (#4),

(iii) using spacer grids instead of classical CANDU spacer pads welded on the cladding sheath to eliminate its welding and simplifying the manufacturing process (#7),

(iv) increasing the number of rods by creating a new FE with many thin rods of a single diameter (#3), so that the fuel centre temperature is decreased (#5),

(v) reaching higher burnup can be reached (and so, lower specific fuel cost, #9), due to the lower thermomechanical behaviour (#8).

The mechanical compatibility is obtained by using the slightly greater channel diameter of Atucha I (5 mm greater than Embalse, which is 103 mm), in order to assemble five FEs within a basket assembly compatible with the refuelling machine (#1). The hydraulic compatibility with Atucha is achieved by tuning the assembly pressure drop with the basket geometry and the choosing the angular misalignment between contiguous FEs.

*3.2. Fuel Rod Definitions.* For a given bundle encapsulated cross-section, the wet surface is proportional to the rod number. Since an important pressure drop of the 37-rod CANDU FE is concentrated on the end plates (≈30%) [26], the CARA FE (with double length) reduces the pressure loss ($\Delta p$) by eliminating an intermediate end-plate junction. This handicap could balance its higher friction loss. Besides, this reduction of end plates and plugs increases the available volume to be filled with uranium mass.

By increasing the number of rods, the rod diameter value decreases with the constrain of keeping the linear mass density, but the total external perimeter of fuel rods is increased and thus the pressure drop, so for the condition of keeping pressure drop, the rod diameter must be lower than the value gotten by keeping linear mass density. Clearly both curves decrease for higher rod numbers.

The CARA FE must be compatible with the most restrictive curve for both reactors. Taking into account that Embalse is the FE with higher linear mass, and Atucha I has the higher hydraulic constrain when an external tube is used, the design criteria are the Embalse mass curve and the Atucha $\Delta p$ curve. Having in mind that if a double length bundle is used, an intermediate end-plate junction and plugs can be removed and the uranium mass can be increased. This approach can be checked by plotting two types of curves against the number of rods (see Figure 3), one curve keeping the uranium linear mass density and the other one keeping the hydraulic pressure drop by using very simple analytical models. See Figure 3 where both curves are crossing at about 50 rods by using a one-meter length bundle.

*3.3. Bundle Geometry.* Different bundle geometries were studied by means of symbolic algebra languages, and four final configurations having 48 to 52 fuel rods were selected. Finally the 52-bundle geometry was chosen, for its good symmetry and compactness. This geometry, shown in Figure 4, has rings with 4, 10, 16, and 22 rods. This bundle has one symmetry axe and one mirror symmetry axe. The CARA rod diameter and thickness are similar to the smallest CANFLEX rods [9]; that lets a good referential point for its manufacture feasibility.

In Table 9, there can be observed the bundle characteristics of three CANDU FEs, regarding the total encapsulated

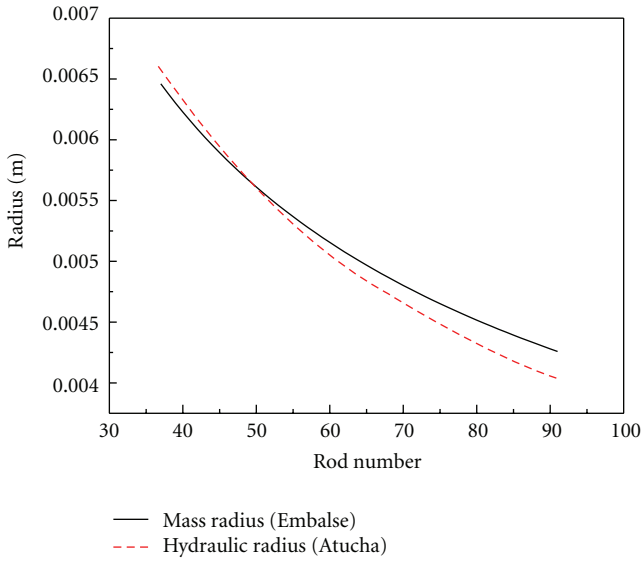


Figure 3: Fuel rod radii for keeping (1) hydraulic similarity and (2) fuel mass similarity.

cross-section filled by uranium. Both CANFLEX and CARA have values 2% smaller than the 37-rod FE.

*3.4. Mechanical Design.* All CANDU fuels use pads welded to the clad sheath in order to ensure the clearance between neighbour rods. The sheath microstructure surrounding the welding zone is modified by the thermal load during the welding process. Despite the complexity inherent to this process and its manufacturer QA, the mechanical integrity margins of this rod can be considered as lower than another one without weldings. In addition, the use of welded pads for cluster geometry implies to deal with different rod types, due to different height of pads needed.

Instead of the use of the standard CANDU approach for ensuring rods position, CARA uses the spacer grid concept, as used in PWRs, adapted to the cluster geometry. This implies that all fuel rods are identical without any welding to the clad sheath and it simplifies the manufacturing process.

The CARA is designed to reach higher extraction burnups by using SEU and keeping the original microstructure to avoid clad failure during irradiation (due to pellet clad interaction by swelling and external cyclic mechanical load due to turbulence).

The CARA FE has 52 diameter fuel rods of the same diameter and of about 1 meter length (see Figure 4) fastened by three self-supported spacer grids (see Figures 5 and 6) and welded to end-plates of low hydraulic restriction (see Figure 7).

Every spacer grid has four concentric rounded strips interconnected by radial strips. It has dimples and springs to fix rod positions and to tightly fast the rod cluster. This is useful to limit rod bending, in order to use them in vertical channels. In PHWR with horizontal fuel channels (like CANDU ones), the CARA fuel laying on the pressure tube by several bearing pads are built on the outer surface

Figure 4: CARA rod bundle.

Figure 5: Cara fuel element.

of the spacer grids, whereas the CANDU bearing pads are welded onto outsider rods. Figure 8 illustrates the geometric comparison of the CARA FE with two 37-rod CANDU fuels.

The assembly system was designed to be loaded by the top side in Atucha and is built in Zircaloy to provide low neutron absorption (see Figures 9 and 10). It has flexible sliding shoes to fix the FE assembly relative position to the channel. By considering that the radial displacement of the assembly system is limited by flexible sliding shoes and the whole systems is hanged by the upper end, the effects of flow induced vibrations in the amplitude of cycling stress will be below the fatigue design limit of the sliding shoes.

*3.5. Preliminary Vibration Analysis.* Considering the CARA under Atucha I and Embalse operating flow conditions, the value of the dimensionless velocity coefficient for the fluid-elastic instability are 0.46 for Atucha I and 0.42 for Embalse.

Table 9: Bundle characteristic of CANDU FEs.

| Bundle type | Rod number | Rod outer diameter (mm) | Clad thickness (mm) | Inner cross-section ($mm^2$) | Relative inner volume |
|---|---|---|---|---|---|
| CANDU 37 | 37 | 13.08 | 0.42 | 4.354 | 1 |
| CARA | 52 | 10.86 | 0.35 | 4.216 | 0.98 |
| CANFLEX | 35 | 11.5 | 0.33 | 4.256 | 0.98 |
| | 8 | 13.5 | 0.36 | | |

Figure 6: CARA spacer grid [27].

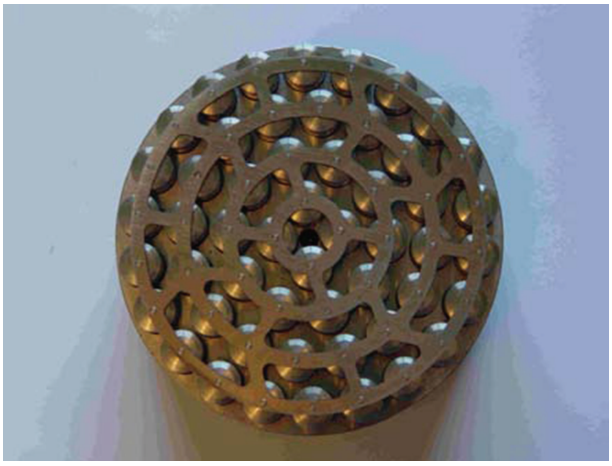

Figure 7: CARA end plate.

These analyses were done considering a conservative case of noncollapsible effects on Young module, and since these values are less than the unit, concerns of fluid-elastic instability are negligible.

The fuel rod natural frequencies mainly depend on mass, length, and cross-section moment of inertia. The CARA fuel cladding was designed to collapse over the fuel pellets at the reactor operational pressure. Therefore, the moment of inertia is related to the shear stress between the cladding and the fuel pellet. To understand this complex behaviour, experimental studies using different metallic pellets inside claddings were performed to simulate collapsible conditions. It was found that Euler-Bernoulli model described the phenomena of collapsible fuel rods by using a Young module 50% higher than the clad value. In this case, the pellet has major contributions to the rod stiffness, which is not the PWR (Atucha) case, having self-supporting cladding.

Table 10: Flow excitation parameters for CARA fuel.

| Reactor | Tube thickness (mm) | Velocity $V$ (m/s) | $\rho * V$ ($Kg/m^2s$) | Reynolds number |
|---|---|---|---|---|
| CANDU | — | 8.21 | 6.567 | 4.51 E5 |
| Atucha I | 0.5 | 9.40 | 7.817 | 5.99 E5 |
| Atucha I | 1.0 | 9.78 | 8.141 | 6.00 E5 |

As was already seen, the CARA mechanical design must fit the flow induced vibration at Atucha I conditions, while CANDU-6 conditions are less demanding (see Table 10). Due to its fuel rods similarities, the CARA FE can be compared with the actual CANDU-37 FE in the CANDU-6 reactor. For this case, the CARA flow excitation is lower (the product $\rho * V$ decreasing up to 95%, Reynolds decreasing up to 88%) than for the CANDU fuel in CANDU 6 reactor. When the CARA FE with an outer tube of 1 mm thickness in Atucha I reactor is compared with the actual CANDU-37 FE in the CANDU-6 reactor, the CARA flow excitation is higher (the product $\rho * V$ increasing up to 119%, Reynolds increasing up to 117%) than the CANDU fuel situation, but not excessively.

For a preliminary analysis, it is useful to do a comparative study including other reactors and their FEs. By comparing Atucha I fuel against CARA, both have three intermediate spacer grids per meter of length, but while Atucha fuel has a single long rod (5.25 meter length) the CARA uses short rods (1 meter length) and, then, its natural frequencies are at least about five times higher than Atucha ones, without considering collapsible effects, using the Euler Bernoulli model for beams [28].

The three spacer-grids of CARA are placed in order to increase the frequencies of its natural transversal vibration modes and bending constrains for mechanical compatibility in horizontal refuelling. One is fixed at the middle, and the others are placed symmetrically at one sixth from each extreme. The distance among spacer grids is 333 mm, similar to PWR [5] and Atucha fuels. This distance is less than the minimum conservative value for mechanical buckling stability without considering collapsible effects.

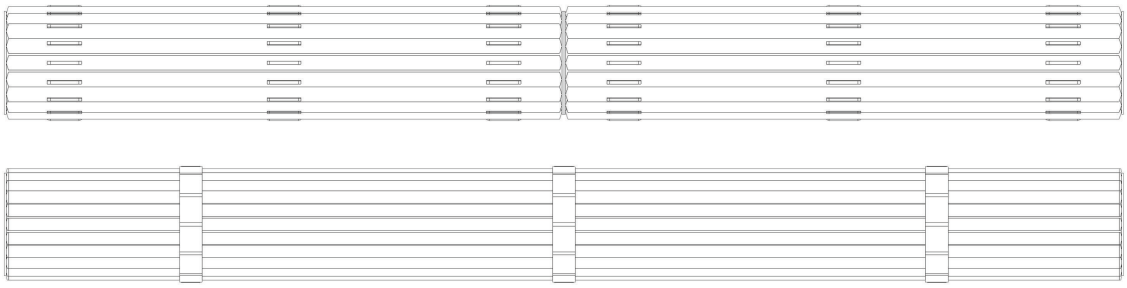

Figure 8: CANDU-CARA and CANDU-37 fuel elements comparison.

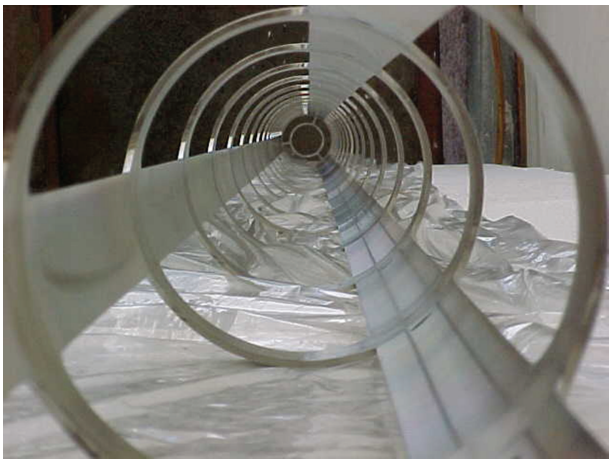

Figure 9: Inner view of an external assembling tube for using in Atucha I.

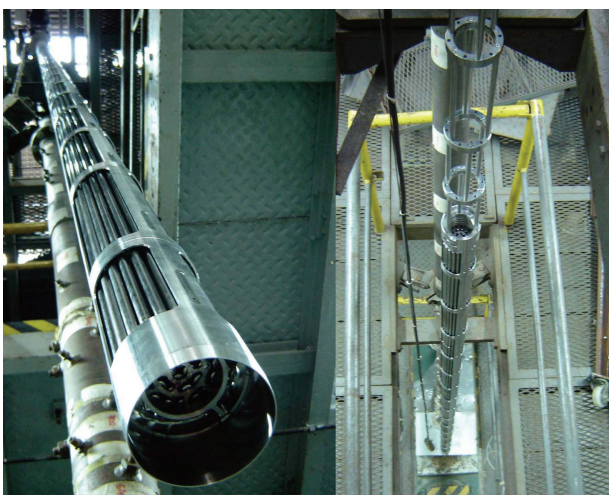

Figure 10: CARA Atucha FE assemblies.

The spacer grids design consider the elastic springs behaviour, especially the residual force at the end of life following the PWR concept (fuel rod always in contact with the spacer grid dimples, see Figure 6). Thus, the clad and spacer-grid interaction (fretting) do not produce any significant wearing effect during irradiation [5]. The designed CARA discharge burnup (about 18,000 MWd/THM) is less than one half of actual PWR (38,000 MWd/THM) burnup and nearly one third of the advanced PWR (55,000 MWd/THM) [5] burnup.

The CARA has fixed extremes (end plates) every 1 m long, and uses collapsible fuel rods, which shows that the CARA rods are more binding than PWR fuel ones and its natural frequencies are at least 5 times higher, together with shorter irradiation time compared with PWR.

A preliminary analysis was carried out without considering the collapsible effects on the stiffness of the fuel rod, which is a conservative assumption. The natural frequencies considering the mechanical constrains, due to spacer grids and end plates, were calculated by a computational code. Considering the Atucha and Embalse operating conditions, the hydrodynamic mass (added mass which increase the weight of vibrating body due to surrounding water) was calculated [29], getting 4 times the water mass in the fuel rod volume. The CARA natural frequency results are: $F1$ = 73.9 Hz, $F2$ = 92.9 Hz, and $F3$ = 212.6 Hz.

The turbulence-induced vibration was estimated in a conservative approach with the Paidoussis formula [23] (without considering the collapsible effect) having for the CARA FE a zero-peak vibration amplitude of 0.155 mm for Atucha and 0.118 mm for Embalse. This is compatible with the maximum acceptance criterion, which is 2% in diameter (0.22 mm).

In accordance with the previous discussions, it was estimated that the mechanical design of CARA could be considered as conservative for CANDU-6 reactors, and as feasible for Atucha ones.

*3.6. Hydraulic Design.* Due to their concepts, the CARA and present CANDU fuels have different balances of concentrated and distributed hydraulic losses. Since only distributed losses are strongly dependent on the flow regime (i.e., Reynolds number), they have different hydraulic performance in reactor conditions (at very high Reynolds numbers) than in low-pressure test facilities (at moderately high Reynolds numbers). Hence, for hydraulic similarity objectives, it is important to model the Reynolds dependence of the fuel hydraulic loss, in order to extrapolate experimental data obtained at low-pressure test facilities.

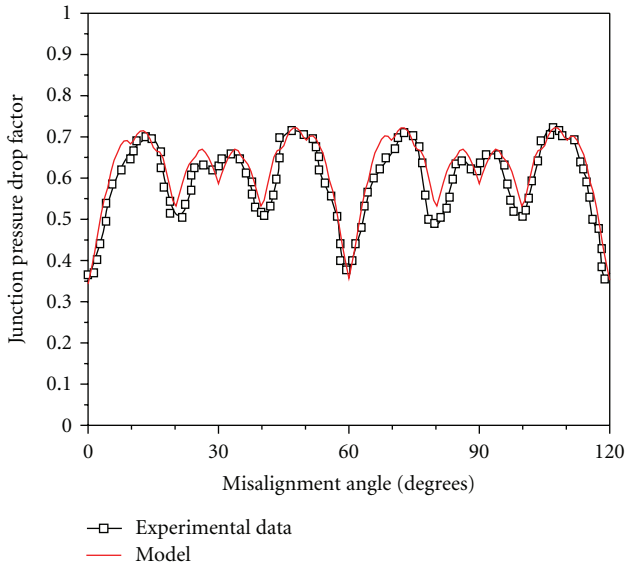


Figure 11: CANDU junction pressure drops.

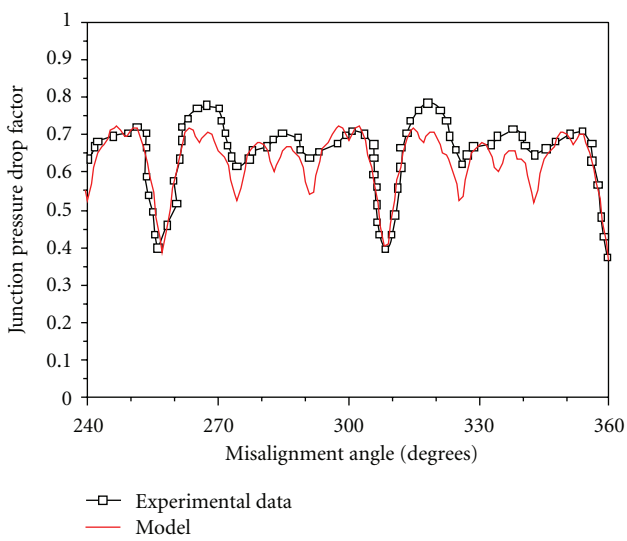


Figure 12: CANFLEX junction pressure drops.

*3.6.1. End Plates Modelling.* In the CANDU reactor, the fuel chain is loaded with random different azimuthal angles. The end plates junction hydraulic loss depends on the misalignment angle. To evaluate the channel average hydraulic pressure drop, it is necessary to measure this dependence. This behaviour can be used to tune the channel pressure drop in the Atucha by fixing their relative angular position with the assembly system.

An analytical model of pressure drop for the misalignment angle of junction between neighbour fuels has been developed and tested using published [28] and CNEA experimental data. The excellent agreement between the model and published experimental data for CANDU 37-rod and CANFLEX fuel elements are shown in Figures 11 and 12, respectively.

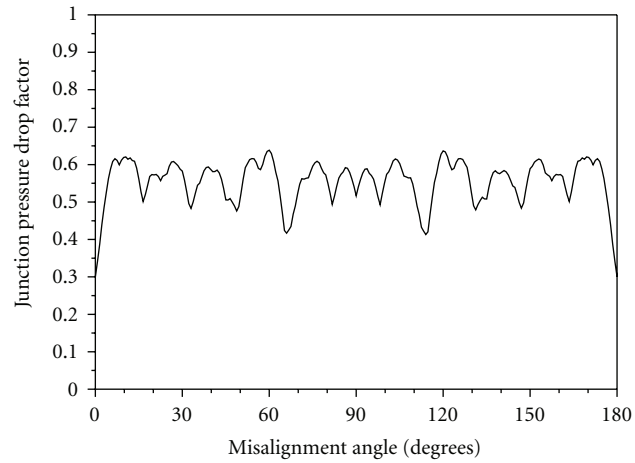


Figure 13: CARA junction pressure drop predicted by model.

The general concept of the CARA end-plate was chosen by following the CANDU 37-rod and CANFLEX 43-rod bundles. The hydraulic pressure drop produced on every contiguous pair of endplates is a function of the misalignment angle, as it can be observed from these two FEs. Then, it is useful to develop a rational base model for this term at the early stage of the CARA design development, for the hydraulic design of the new end-plate geometry. This model provides a useful tool for analyzing the tradeoff between mechanical requirements (that claims for thicker and wider bars) and hydraulic pressure-drop requirements (that claims for the opposite trends).

A simple model was developed for estimating the end-plate hydraulic restriction, based on a detailed calculation of the cross-flow section variation through the conical plugs (gradual expansion and contraction terms) and end-plate width (sudden contraction and expansion terms) [30, 31]. This model was adjusted by using CANDU-37 rod data (Figure 11), and validated against CANFLEX data showing a good accuracy (deviation lower than 10%) as it can be seen in Figure 12. Thus, this model was used for the pressure drop CARA end-plate coefficient prediction, as it is illustrated in Figure 13. The most probable, minimum (fully aligned) and maximum values obtained are 0.60, 0.32, and 0.68 respectively. This model predictions were verified with experimental data obtained in a hydraulic low pressure loop within a 10% error bandwidth.

In order to extrapolate the experimental results to reactor conditions, a sequence of tests were done varying the Reynolds number between $5 \times 10^4$ and $1.6 \times 10^5$, by changing the flow velocity. These tests were useful for end plate modelling as much as grid spacer and friction hydraulic modelling. By considering the experimental findings, it was shown that the Reynolds dependence of end-plate junction is negligible (in agreement with our model), within 5% of accuracy band error and, so, those model predicted values can be considered satisfactory.

*3.6.2. Spacer Grid Modelling.* Several designs of spacer grids could be used. The present design provides a good

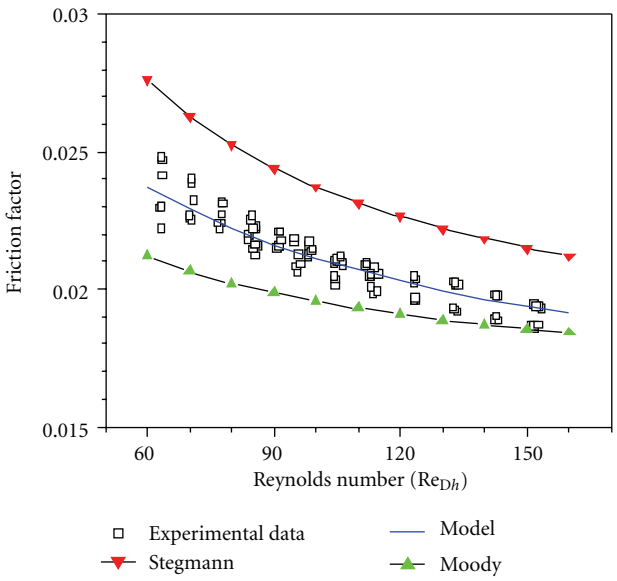


Figure 14: Distributed fiction loss coefficient.

Table 11: Flow parameters for CARA in both reactors.

| Reactor | Re ($\times 10^5$) | $K_{sg}$ |
|---|---|---|
| CANDU-6 | 4.51 | 0.68 |
| Atucha | 6.00 | 0.65 |

performance in both reactors from a hydraulic standpoint. This design, showed in Figure 6, was tested in hydraulic tests loops.

The hydraulic pressure drop of grid spacers depends on flow Reynolds number, as is well known from published models [32, 33] and experimental data, due to friction on spacer wall and changes in flow cross-section. Our experiments confirm this behaviour; thus, the extrapolations for CARA conditions in both reactors are showed in Table 11, where *Ksg* is the hydraulic coefficient for the spacer grid, having considered an external circumferential tube of 1 mm.

*3.6.3. Distributed Friction Modelling.* The Steggeman's correlation [34] is used in order to consider the dependence of the Darcy coefficient with the hydraulic diameter and the flow Reynolds number. By using this, the $f$ factor for CARA fuel in each reactor was estimated and is shown in Table 12, considering an assembly tube of two different thicknesses for Atucha I.

The distributed friction factor from experimental data is compared with the classical well-known Moody correlation [12], and the specific correlation developed for fuel rod PWR arrays [35] in Figure 14, showing good agreement within 10% deviation. A new specific cluster correlation using the experimental data was built with least square fitting. In Figure 15, the total spacer grid loss coefficient was adjusted from the experimental data showing good agreement [11].

Table 12: Flow parameters for CARA in each reactor.

| Reactor | Tube thickness (mm) | Re ($\times 10^5$) | Hydraulic diameter (mm) | $f$ |
|---|---|---|---|---|
| CAND-6 | — | 4.51 | 6.96 | 0.0157 |
| Atucha | 0.5 | 5.99 | 8.01 | 0.0147 |
| | 1.0 | 6.00 | 7.70 | 0.0145 |

Table 13: Estimated CARA fuel channel pressure drop under different configurations.

| Reactor type | Tube thickness (mm) | $K_{end}$ | Pressure drop (KPa) |
|---|---|---|---|
| CANDU-6 | — | Average | 562 |
| Atucha | 0.5 | Average | 510 |
| | | Fully aligned | 472 |
| | 1.0 | Average | 624 |
| | | Fully aligned | 580 |

*3.6.4. Overall Hydraulic Modelling.* Using the previous hydraulic restriction coefficient, the overall fuel channel pressure drop can be calculated by using (1), with the right numbers of end-plate junction and spacer grids in each case, obtaining the results shown in Table 13. These results show that even a 1 mm thickness assembly tube can be acceptable by using the fully aligned configuration. Let us remark that besides this one, an assembly tube with openings has been designed (instead of a solid one as we have consider here), that is expected to produce a still lower pressure drop (see in Figure 10 the first prototype tested on the low-pressure loop test facility of CNEA).

## 4. Fuel Performance Modelling

*4.1. Neutronic Behaviour.* The neutronic behaviour of the CARA fuel element was calculated by using the code WIMS D/4 [26]. Considering the materials of the fuel element and reactor core geometry, the burnup could be estimated by using the cell reactivity evolution, as well as the power peak factor (highest to average power ratio) [36]. The burnup was calculated as the value that equalized the mean core reactivity of an average cell to the required excess reactivity for operation [14].

The beginning of life (BOL) excess reactivity, power peaking factor, and burnup level can be seen in Table 14 and the crown rod power distribution in Table 15, for natural uranium and SEU fuels, respectively, for each reactor. Using the power evolution, burnup level, and peaking factor calculated with WIMS, together with all the geometry and compositions, the complete thermomechanical behaviour could be calculated for the most demanded CARA rods.

*4.2. Thermomechanical Behaviour.* The analyses of the thermomechanical behaviour and the fuel rod design were performed by using the BaCo code [37, 38]. BaCo was developed at CNEA for the simulation of the behaviour

Table 14: WIMS results, neutronic differences between CARA and CANDU and Atucha I.

| Characteristic | CANDU 37 | Atucha I | CARA (in Embalse) | CARA (in Atucha I) |
|---|---|---|---|---|
| Natural uranium-burnup [MW·d/ton·$UO_2$] | 7.500 | 6.100 | 7.529 | 6.368 |
| Peak factor | 1.1261 | 1.0936 | 1.1359 | 1.1483 |
| SEU (0.9%)-burnup [MW·d/ton·$UO_2$] | 14.537 | 13.466 | 14.576 | 14.524 |
| Peak factor | — | — | 1.1484 | 1.1577 |

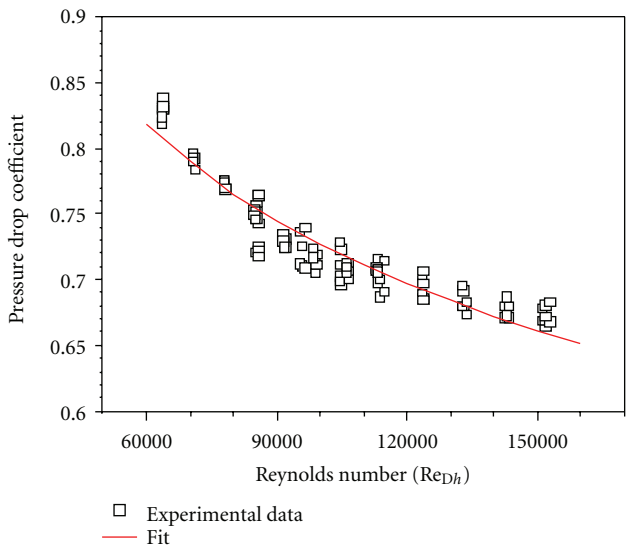


Figure 15: Spacer grid loss coefficient.

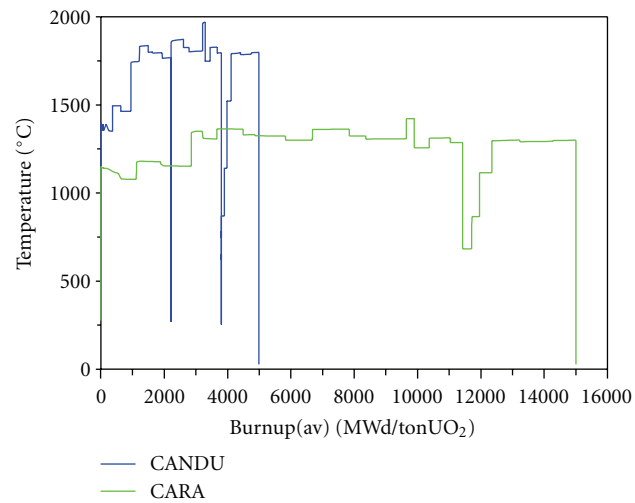


Figure 17: Averaged temperature at the pellet centre of a CANDU fuel rod CARA fuel.

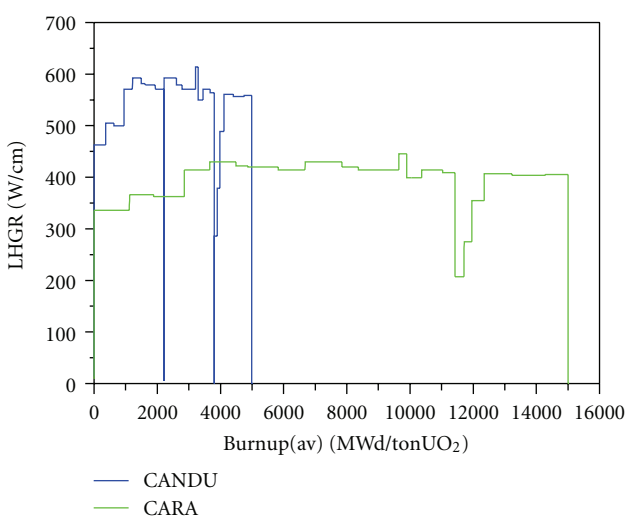


Figure 16: Averaged power history for a CANDU fuel rod and CARA fuel in CANDU.

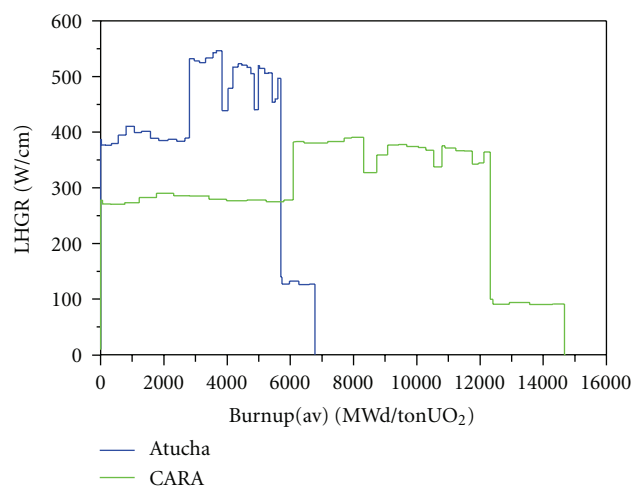


Figure 18: Local power history for the seventh segment of a fuel rod of the Atucha I NPP and a CARA fuel at that axial position in the channel.

of nuclear fuel rods under irradiation. BaCo is a code for the simulation of the thermomechanical and fission gas behaviour of a cylindrical fuel rod under operation. The development of BaCo is focused on PHWR fuels as the CANDU and Atucha ones but it keeps full compatibility with PWR, BWR, WWER, and PHWR MOX fuels, among advanced and experimental fuels. A specific version of BaCo was developed and validated for the CARA fuel. The BaCo present version includes postprocessing tools for statistical improvement [39] and 3D enhancements [40, 41]. BaCo was part of the CRP FUMEX II of the IAEA and at present is part of the CRP FUMEX III of the IAEA in order to continue the validation and experimental support of fuel simulations by means of BaCo [35].

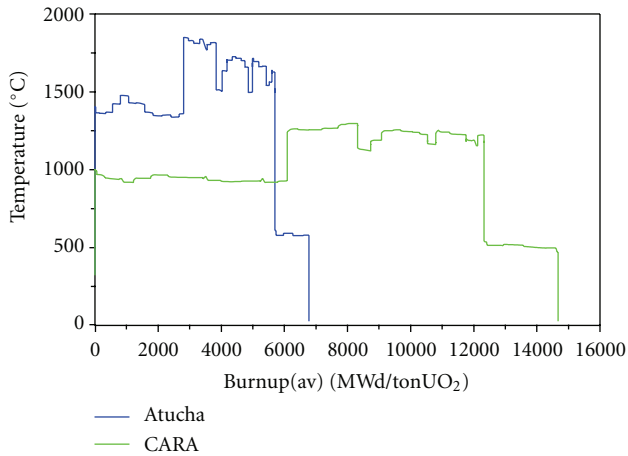


Figure 19: Local temperature in the 7th segment of a fuel rod of the Atucha I NPP and a CARA fuel at the 7th axial position in the channel.

Table 15: WIMS results for the CARA rod power distribution for 0.9% SEU at BOL.

| Crown | CARA (in Embalse) | CARA (in Atucha I) |
|---|---|---|
| 1 | 0.8098 | 0.8045 |
| 2 | 0.8499 | 0.8433 |
| 3 | 0.9474 | 0.9439 |
| 4 | 1.1411 | 1.1476 |

The major changes in the models of the code for CARA were not significant because BaCo was originally designed for Atucha and CANDU fuels. Two specific techniques for fuel design were developed: parametric (or sensibility) analysis and probabilistic (or statistical) analysis among the normal (or standard) analyses and the "extreme cases analysis."

*4.3. CARA Fuel Rod Behaviour.* The power history for a CANDU fuel used for calculation is included in Figure 16. The power history sketched reaches high power (and then high temperature), and it includes two shutdowns with a step-by-step increase in the power level after the second shutdown. This hypothetical, but realistic, power history was defined for real demanding conditions of irradiation for a fuel element and for the BaCo code simulation. Starting with that power history, we extrapolate the respective history for the equivalent CARA fuel conditions in a CANDU reactor correcting by the neutronic cell calculation model. The use of an extra crown of rods, by reducing the rod diameter, produces a decrease in the power level. The extrapolation is based on the burnup extension and the adaptation of linear power levels of the CARA fuel. In order to use a proper power history for a CARA fuel in a CANDU NPP (or in Atucha I), we extend the scale of burnup of a present power history of a CANDU fuel (or Atucha I fuel) keeping the corresponding power level. The extension in burnup is 15,000 MWd/ton$UO_2$ and the linear power is reduced up to 72% of the original value, due to the new geometry of the CARA fuel.

Figure 17 represents the BaCo code output for the centre temperature of the $UO_2$ pellet for a CANDU fuel rod and for the equivalent CARA fuel by using its associated power history (see Figure 16). CARA fuel allows a decrease of 500°C in the temperature at the maximum power level.

Figure 18 represents the local power history of the seventh axial segment of a 5 meter long Atucha I fuel element (numbering from the top of fuel and taking into account ten axial segments). The seventh segment is the most demanded axial section during irradiation as it includes a maximum power level of 547 W/cm. The CARA fuel extrapolation corresponds to the fourth module of a CARA assembly in Atucha I (the fourth CARA module is equivalent with the seventh Atucha segment). The burnup at end of life is 14,750 MWd/ton$UO_2$ and the power level is reduced 73.4% of the original Atucha fuel value.

The maximum calculated pellet temperature for the Atucha fuel is ~1850°C during the maximum power level (see Figure 19). The temperature for the equivalent CARA module is ~1350°C, thus, a decrease of ~500°C respect of the normal Atucha I fuel.

The BaCo code calculations shows: temperature decrease, smaller fission gas release, no restructuring and no central hole, lower thermal expansion, and finally a better tolerance of the dimensional parameters of CARA. This allows to improve the manufacturing tolerance with an improvement in the dishing and shoulder of the pellet, and a smaller plenum.

BaCo code validity is sustained with the participation in international code benchmarking, the Atucha and CANDU experience, irradiation information available in the international literature and our own experimental irradiations.

## 5. Conclusions

The development of the CARA fuel element, intended for use in two different PHWR was presented, showing its design criteria and the way in which they were reached. The mechanical solution proposal by CARA is very innovative (doubling length and using this hydraulic advantage for adding spacer grids and for eliminating weldings on cladding) relative to the evolutionary solution proposal of CANFLEX for CANDU reactors, allowing extended burnup by the use of SEU, and with good thermal hydraulic margins using a single fuel rod diameter. From the point of view of designer, the CARA approach could be considered as another design line for new advanced CANDU fuels intended for higher burnup.

Different CARA fuel elements prototypes were hydraulically tested in a low-pressure loop. The experimentally validated models show the CARA hydraulic similarity with respect to CANDU fuel in Embalse. An additional assembly system enables the use of CARA in the vertical channels of Atucha. The mechanical feasibility for Atucha and Embalse, and hydraulic compatibility were checked, verifying that the CARA fuel can fit the unique Argentine challenge: a single fuel element for two different HWRs. The CARA could comply with all the design requirements, and with

its implementation, SEU fuel element can be used in the Argentine NPPs at competitive values, an essential task for economic production in Argentina.

The mechanical, vibration, neutronic, thermohydraulic, and thermomechanical margins of CARA were verified by means of simple analytical models and numerical state-of-the-art simulation codes, showing perfect behaviour. The CARA fuel rod design was analysed using BaCo parametric and statistical analysis and extreme cases calculations, to know the correct incidence on the manufacturing QA procedure and defining the fabrication uncertainties tolerance limits. All of these analyses are satisfactory for the present stage of development of CARA, with the preliminary engineering finished and the detailed engineering under development. However, it must be noted that full scale tests (thermohydraulic and endurance tests) are still needed in order to actually validate and license this fuel. At present, all of the prototypes needed for these tests have been built.

## Acronyms and Symbols

BOL: Beginning of life;
DNB: Departure of nucleated boiling;
CARA: Combustible Avanzado para reactores Argentinos (advanced fuel for Argentine reactors);
CHF: Critical heat flux;
FE: Fuel element;
NPP: Nuclear power plant;
NU: Natural uranium;
PHWR: Pressurized heavy water reactor;
SEU: Slightly enriched uranium;
$\varepsilon$: Cladding roughness;
$\rho$: Liquid density;
$\Delta p$: Pressure drop;
$\phi_b$: Rod outside diameter;
$\phi_p$: Pellet diameter;
$D_h$: Hydraulic diameter;
$f$: Distributed friction Darcy coefficient;
$K_{\text{end}}$: Hydraulic coefficient of end-plate;
$K_i$: Hydraulic coefficient of inlet;
$K_{\text{sg}}$: Hydraulic coefficient of spacer grid;
$K_{\text{sp}}$: Hydraulic coefficient of spacer;
$K_o$: Hydraulic coefficient of outlet;
$L$: Fuel chain length;
$L_h$: Heated length per channel length unit;
$N$: Number of FE in a chain;
$N_b$: Number of fuel rods;
$V$: Average flow velocity.

## Acknowledgment

The authors acknowledge the following personnel for the support given to the project: *CNEA:* Cirimello R., Florido P., Delmastro D., Daverio H., Gonzalez J., Giorgis M., Pérez E., Gerding R., Bianchi D., Markiewicz M., Pastorini, Martin Ghiselli A. Sacchi M., Olezza R., Belinco C., Galazzo H., Batán V., Converti J., Scheble M., Estévez E., Troiani H., Peregrina J., Riquelme P., Gómez Bastidas C., Banchik D., Vizcaíno P., Rios R., Dell Occhio L., Taboada H., Miceli, M., Fernández J., Yawny A., Eggenschwiller J.C., Tutzauer F., Szegedy E., Muñoz C., Silin N., Masson V., Miceli M., Córdoba R., Notari C., Goya H., M. Sbaffoni, M. Ghio. *CONUAR:* Lamuedra R., Rubio N., Reale F., Lesignana A., Francis H.. Pérez A., Schoeder P., Andino A., Ransemberg B., *N.A.S.A.*: Fink J., Sidelnik J., Fernández R., *INVAP:* Tourn R., Martinez R. The authors also acknowledge Dr. Horacio Troiani for critically reading the manuscript and several suggestions.

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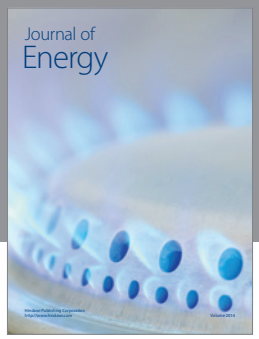

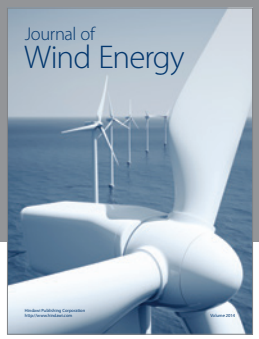

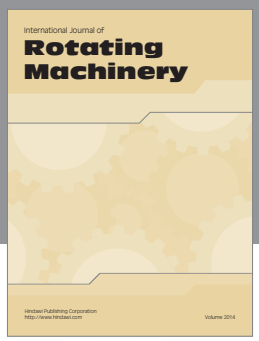

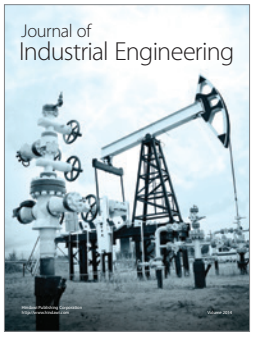


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