CARA FUEL BUNDLE: A NEW CONCEPT FOR HWR PRESENT SITUATION

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ABSTRACT

A new concept for HWR fuel bundles, namely CARA, is presented. The CARA design allows to improve all the major performances in the PHWR fuel technology. Among others, it reaches higher burnup and thermohydraulic safety margins, together with lower fuel pellet temperatures and Zry/HM mass ratio. Moreover, it keeps the fuel mass content per unit length and the channel pressure drop by using a single diameter of fuel rods.

1. INTRODUCTION

In Argentina there are two operating NPPs (Atucha and Embalse). Both of them are cooled by pressurized heavy water and are fueled with natural uranium, but they have very different design for the primary system. Embalse is a standard CANDU 6 reactor (horizontal pressure tube, see figure 1), and Atucha I have vertical fuel channels inside a pressure vessel reactor (Siemens' design, see figure 2). Therefore, their nuclear fuel elements are strongly different (see figures 3 and 4). Embalse uses a national developed CANDU 37 rods fuel element, and Atucha uses a long bundle similar to PWR rod type, both supplied by a private owned fuel-manufacturing company namely CONUAR. This diversified scenario leads to several complications from the point of view of the production at commercial scale, especially when the competitiveness is a main task in electricity generation costs.

Nowadays the CANDU fuel design is a very active area, with a new bundle generation (CANFLEX) [1] following the present LWR trends, enabling to reach higher burnup with smaller rod diameter and, consequently lower central temperature, linear and surface heat flux. Taking into account that the Argentinean electric system dispatch for fuel marginal cost, it is reasonable to have a new generation of advanced nuclear fuels. These fuels must lead, the most ambitious goals described, with an additional one: one fuel element for both types of NPPs in order to achieve the smallest costs at small-scale commercial production.

2. INITIAL CRITERIA FOR A NEW BUNDLE CONCEPT

Instead of small changes in fuel element improvements, we analyze the feasibility study of a completely new fuel element, for both types of NPPs, namely CARA (Advanced Fuel for Argentinean Reactors).

This fuel element was set up with the following objectives:

1 - To fit for both NPPs.
2 - Increasing the heated perimeter.
3 - Using a single type of fuel rod diameter.
4 - Decreasing the fuel center temperature.
5 - Decreasing the Zry/Uranium mass ratio.
6 - Keep the higher uranium mass per fuel length unit.
7 - Do not change the hydraulic pressure drop of each NPP core.
8 - Burnup extended using SEU (Slightly Enriched Uranium)
9 - Do not exceed the fabrication cost of the CANDU fuel.

As in the CANFLEX design, which has many of the same objectives using two different rod diameters, the CARA fuel must explore new choices. Increasing the fuel rod number, the distributed friction is increased, related to the smaller hydraulic diameter. To keep constant the core pressure drop leads to the CANFLEX solution, loosing a single rod diameter (condition number 3), or use one type of rod diameter loosing uranium mass (condition number 6). The key in the CARA solution is a fuel element length twice the present CANDU fuel substituting a couple of elements for a single one in the refueling machine in order to decrease the concentrated pressure drop and using this handicap to create a new bundle of many thin fuel rods.

At present, the Atomic Energy Commission of Argentina (CNEA) is developing the CARA together with the fuel manufacturing company (CONUAR) and the interest of the nuclear power operator utility (NASA) in Argentina. The scope of the present project is to develop the CARA with until the commercial production of CARA bundle using 0.9 % enrichment fuel.

3. GENERAL CHARACTERISTICS

3.1. Fuel Rods Definitions

Analyzing the pressure drop of an Embalse fuel channel [2], and subtracting the distributed pressure drop [3, 4], we could find the concentrated pressure drop of one CANDU bundle (end plates and spacers). In reference [5] several changes in the CANDU fuel are studied, and using these data, pressure drop of the end plates and the spacers in present CANDU 37 rods bundle could be calculated and checked with our own data set.

An important pressure drop is concentrated on the end plates, and if we consider a bundle with twice of length of the usual CANDU fuel (without compatibly problems with CANDU 6 refueling machine), we could decrease the hydraulic diameter, increasing the number of rods.

To check this approach, two type of curves were developed for a given fuel channel, in which the radius was changed in order to keep constant the uranium mass (mass constant curve) or hydraulic pressure drop (Δp constant curve) for different number of rods. Clearly both curves monotonally decrease for higher rod diameters, and if standard CANDU 6 bundle is taken as a basis, both curves have the same radius at 37 rods number. At higher rod number, if the Δp is keep constant uranium mass must decrease (mass constant curve above the hydraulic pressure drop curve).

But if a twice length bundle is used, both curves cross at 66 rods number. The Δp gain for the end plates number reduction, gives a Δp credit for the distributed friction together with a new concept for spacer function with only 33 % pressure drop reduction. As CARA fuel element must be compatible with two types of fuel channels, the most restrictive curve must be used for mass and Δp curves. Atucha has the most restrictive Δp requirements, and Embalse has the most restrictive mass requirements. Using both curves, with 1 meter long bundle, approximately 50 rods could be used for both NPPs, as is shown in figure 5.

3.2. Bundle Geometry

For the study of different bundle geometries, symbolic algebra languages enable very simple and fast evaluation; including rod ring rotation and different central rod number. Using this type of approach, 4 final geometries were studied, with 48 to 52 fuel rods. Finally a 52-bundle geometry was selected, for the good symmetry and compactness of the array. This geometry, shown in figure 6, has 4, 10, 16 a 22 rods per ring. The reduction of number or plugs and end plates gives a uranium credit that could be used to increase the bundle uranium mass.

The corresponding pellet diameter is very similar to present smallest CANFLEX one, but with slightly thinner clad thickness and present gap clearance of CANDU 37 rods bundle.

3.3. Mechanical Concept

General dimensions of the first prototype series are based in simple engineering solutions, in order to perform different tests and produce an extensive engineering feedback in the design concept. Using an outside diameter of 10.86 mm and 0.35 of clad thickness, the bundle has a double total length of the CANDU fuel.

Present prototypes use three spacer grids and two dismountable end plates at both ends, both shown in figure 7 and 8. Each fuel rod is fixed in its position for one elastic spring and two fixed dimples. The rods end caps could be easily dismounted from the end plate using specially designed screw. Each loaded springs and fixed supports are built in the spacer itself, in order to drastically reduce the number of welds over the cladding length, with only four rods welded to the spacer grids in order to fixe its axial position.

External pads in the spacers fixed the bundle in the CANDU pressure tube for Embalse, and in an Zircalloy alloy external metal jacket or basket in order to build using 5 bundles an Atucha type fuel element. This external basket reduce the effect of the external water bypass due to the small differences in both fuel channels (the Atucha channel have an internal diameter slightly greater than Embalse).

The dismountable end plates able to easily dismount each rods after the different loop tests in order to measure any type of dimensional change and the effect of different elastic spring loads. A general view of one of the first prototypes presently tested in the hydraulic low-pressure test loop is shown in figure 9.

4. FUEL PERFOMANCE MODELING

4.1. Neutronic Behavior

The neutronic fuel element behavior was estimated using WIMS D/4 [6] as a neutronic cell code. Differences in pitch and pressure tube thickness were considered in the equivalent cell. Burnup could be estimated using cell reactivity evolution, and an adecuated criteria for continuum refueling NPPs, (figure 16). Power peaking factors for the four pin annulus were also determined as a function of burnup (figure 16)[7]. Considering the fuel rod definitions in section 3.1, the proper power densities, dimensions and geometrical bucklings are calculated and inserted in the WIMS D4 input to estimate the CARA fuel element neutronic behavior. These points are detailed in the following subsections.

4.1.1 Average cell

The geometric fuel dimensions of CARA fuel element are the following:

Fuel element geometry	:	Cluster
Number of fuel rods per fuel element	:	52
Number of uranium fuel rods	:	52
Total length	:	990.6 mm
Fuel element radii	:	51.285 mm
Fuel rod length	:	981.5 mm
Clad external radii	:	5.43 mm
Clad thickness	:	0.34 mm
Pellet radii	:	5.045 mm
Pellet length	:	10.78 mm

The input cell for WIMS-D4 code was built using the dimensions detailed above, as the following figures show (figures 17 a and b):

4.1.2 Core burnup

The radial buckling will remain constant for each NPP, as it is related with the core radii. However, for CNA the axial buckling will change as the total fuel element length will be 5 meters instead of 5.3 meters, and in this condition the new buckling will be 2.7778 10^{-5} cm^{-2}.

For the CNA power density, the value used for the original fuel element was scaled considering the UO_2 mass rate between this fuel element and CARA. The original power density was d = 29.644 MW/TonU. Considering an UO_2 mass of 211.37 kg for five CARA elements, the new power density will be: 24.431 MW/TonU.

Finally to estimate these results, the core burnup could be calculated as the burnup that equalized the mean reactivity of core of an average cell to the required excess reactivity for operation.

To apply these criteria, old working reactivity values from both NPP data were used. For instance, in CNA NPP for a core reactivity burnup of 6100 MWd/TonU, a working reactivity k_w = 1.019950 resulted, and was used to obtain the core burnup with CARA fuel element, as figure 16 show.

Table 1 shows fuel burnup an power peaking factors for both NPP, with both fuel elements, and for natural and 0.9 enriched uranium.

TABLE 1. WIMS RESULTS, NEUTRONIC DIFFERENCES BETWEEN THE CARA FUEL ELEMENT FOR CANDU (CNE) AND ATUCHA I (CNA I).

Characteristic	CANDU 37	Atucha I	CARA (in the CNE)	CARA (in the CNA I)
Natural Uranium Burnup [MWd/THU]	7500	6100	7529	6368
Peak Factor [a.U.]	1.1261	1.0936	1.1359	1.1483
Uranium .9% Burnup [MWd/THU]	14537	13466	14576	14524
Peak Factor [a.U.]	-	-	1.1484	1.1577

The BOL excess reactivity, power peaking factor and burnup level could be seen in table 1, for natural uranium or SEU rods, for both types of Argentinean NPPs.

Using the power evolution, burnup level and peaking factor calculated with WIMS, together with all the geometric and compositions, a complete thermomechanical behavior could be calculated for the most restrictive CARA rods.

4.2. Fuel Rod Thermomechanical Behaviour

The power history for a CANDU fuel used for calculation is included in the Figure 10. 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 with demand conditions of irradiation for a real fuel element and for the BACO code simulation [8]. 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 extrapolation is based on the burnup extension and the adaptation of linear power levels of the CARA fuel. The extension in burnup is 15000 MWd/tonUO2 and the linear power is reduced up to a 72 % of the original value.

The Figure 11 represents the BACO code output for the temperature center of the UO_2 pellet for a CANDU fuel rod and for the equivalent CARA fuel using its associated power history (Figure 10). CARA fuel allows a decrement of temperature of 500°C at the maximum power level. The Figure 12 represents the local power history of the seventh axial segment of a 5 meter long Atucha I fuel element (from the top of fuel and taking account ten axial segments). The seventh segment is the axial section most demanded during irradiation, it include a maximum power level of 547 W/cm. The CARA fuel extrapolation corresponds with the forth module of a CARA assembly in Atucha I (the forth CARA module is equivalent with the seventh Atucha segment). The burnup at end of life is 14750 MWd/tonUO_2 and the power level is reduced a 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 13). The temperature for the equivalent CARA module is 1340°C, conversely, a decrease of 510°C.

The BACO code calculations shows: temperature decreasing, smallest fission gas release, no restructuring and no central hole, lowest thermal expansion, and finally a best tolerance of the CARA's dimensional parameters. This allows to best manufacturing tolerance with an improvement in the dishing and shoulder of the pellet, and small plenum.

BACO code validity is sustained with the participation in international code benchmarking, the Atucha and CANDU experience, irradiation provided by international literature and own experimental irradiation [9].

4.3. Hydraulic Pressure Drop

The hydraulic pressure drop along a fuel channel is due to two general components: concentrated and distributed friction losses. A complete analytical hydraulic pressure drop model was developed. It considers the contribution of the end plates, grid spacers and fuel rods to friction loss.

For the case of grid spacer, it was considered local obstructions relates to strap, dimples and springs, following previous approach.[11,12]

The particular cluster geometry of the CARA bundle need experimental validation of this model. Experiments were carry out in a low pressure loop with two prototypes, finding good agreement with theoretical predictions. After that the spacer grid design was improved considering the thermohydraulic performance of the fuel element.

An analytical model of pressure drop for the misalignment angle of junction between neighbors fuels has been developed [10] and tested using published [5] and CNEA measured experimental data. The excellent agreement between the model and published experimental data for CANDU have been shown in figure 14.

Using this model, the CARA coefficient pressure drop has been calculated for both Embalse and Atucha I power plants, and present calculated values are 18 to 6 % lower than the respective maximum pressure drop. Figure 15 illustrate this behavior for Embalse. For the Atucha I fuel channel, the theoretical prediction of pressure drop, taking in to account the assembling device (by coupling five CARA fuel elements in order to fill the vertical channel), shows good agreement with the core pressure drop.

The critical heat flux assessment of the CARA fuel is a qualify test, rather than a developing test, due to improved heated surface (18% greater than CANDU one) with a single rod diameter homogeneous bundle.

5. CARA DEVELOPMENT PROJECT

The CARA design, attracted the interest of the nuclear power operator utility in Argentina (NASA), and the fuel element manufacturing company (CONUAR). Then a new project is right now under planning with the cooperation of three partners (CNEA - NASA - CONUAR) in order complete the whole development program in the shortest time, finishing in the commercial production of CARA fuel bundle for both type of reactors.

The strong economics advantages of the new fuel, together with the excellent experience for the close to commercial SEU Atucha program, put strong incentives for the fastest fuel development up to commercial level.

Present CARA project, including an Atucha and Embalse program irradiation and post irradiation analysis, looks for ambitious tasks of 4 years of time span.

The following projects milestones have been achieved:

- Conceptual design of fuel bundle
- Basic design of fuel bundle
- First demonstration bundle
- First CARA bundle
- Low pressure loop test
- Detailed design of fuel bundle
- Detailed design of coupling system for Atucha I
- Agreement for fuel rod irradiation tests

6. CONCLUSIONS

The feasibility study of an advanced SEU fuel element, compatible with CANDU 6 and Atucha type reactor have been successfully done, using a single rod diameter, as an essential task for economic production in Argentina.

The condition of the present project is to develop the CARA fuel element at the shortest time, finishing with the commercial production of CARA bundles.

7. REFERENCES

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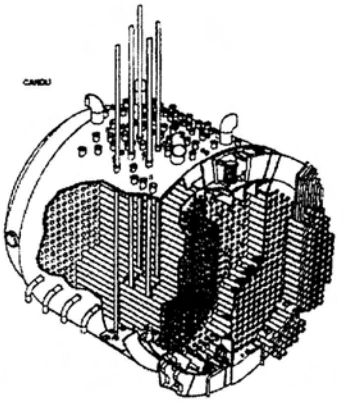

FIGURE 1: CANDU REACTOR.

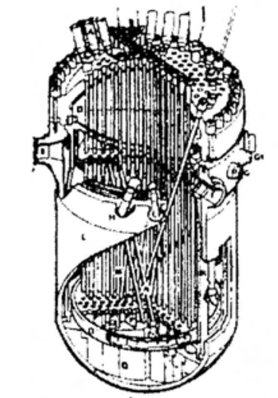

FIGURE 2: ATUCHA I PRESSURE VESSEL TYPE REACTOR.

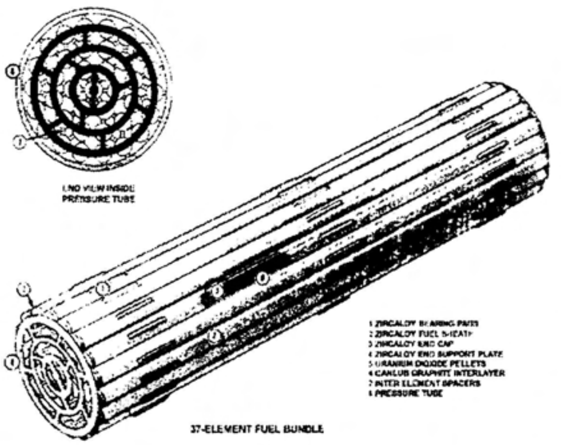

FIGURE 3: CANDU FUEL ELEMENT

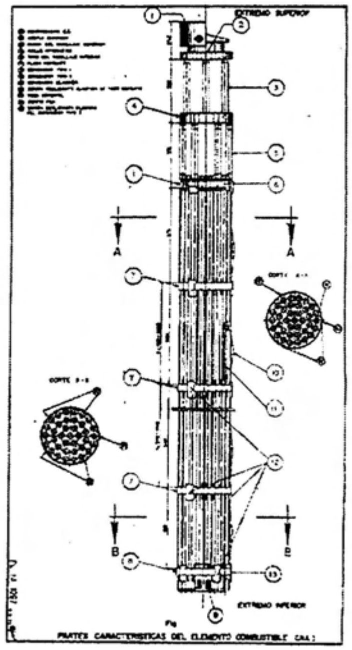

FIGURE 4: ATUCHA I FUEL ELEMENT.

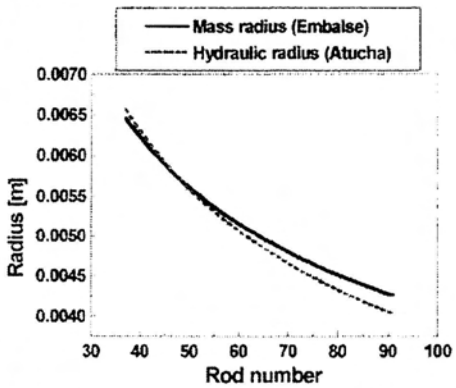

FIGURE 5: CURVE WITH THE “MASS” AND “HYDRAULIC” RADIUS OF CANDU AND ATUCHA.

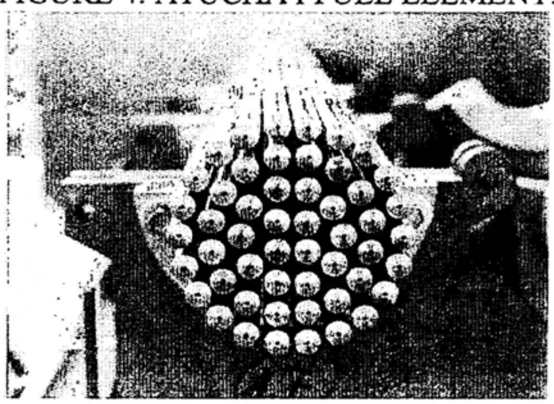

FIGURE 6: GEOMETRY OF THE CARA BUNDLE WITH 52 FUEL ROD.

FIGURE 7: CARA BUNDLE SPACER GRID WITH 52 FUEL RODS.

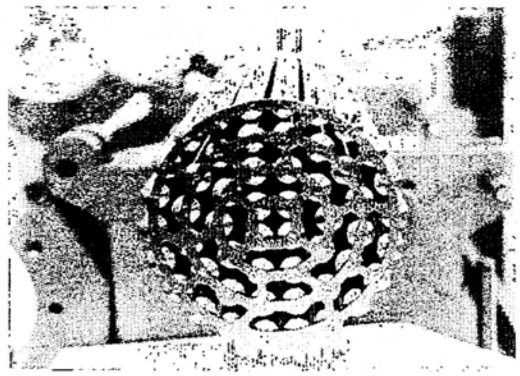

FIGURE 8: DISMOUNTABLE END PLATES OF THE CARA BUNDLE WITH 52 FUEL RODS.

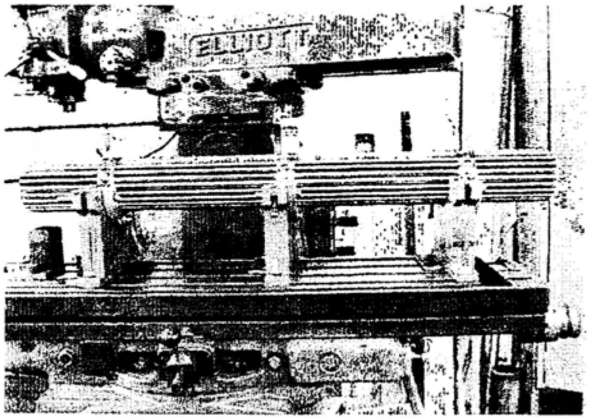

FIGURE 9: EXTERNAL VIEW OF ONE OF THE PROTOTYPES OF CARA BUNDLE WITH 52 FUEL RODS.

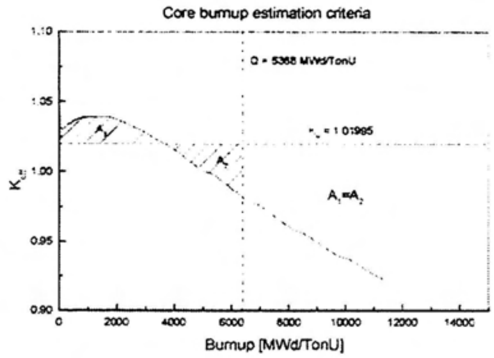

FIGURE 16: CORE BURNUP ESTIMATION CRITERIA.

FIGURE 17: AVERAGE CELL FOR CNA NPP (A), AND FOR CNE NPP (B)

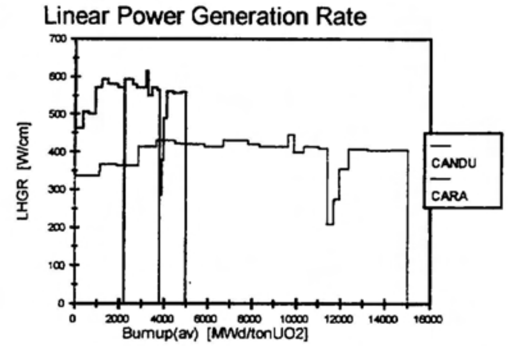

FIGURE 10: AVERAGED POWER HISTORY FOR A CANDU FUEL ROD AND CARA FUEL IN CANDU NPP.

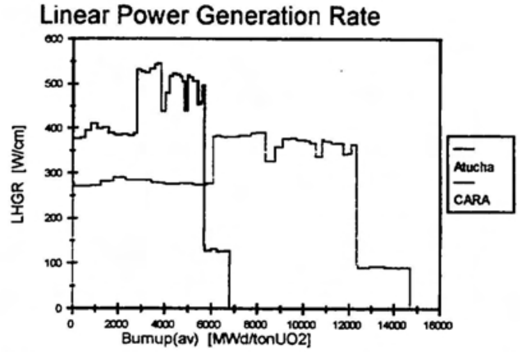

FIGURE 12: LOCAL POWER HISTORY FOR THE SEVENTH SEGMENTS OF A FUEL ROD OF THE ATUCHA I NPP AND CARA FUEL.

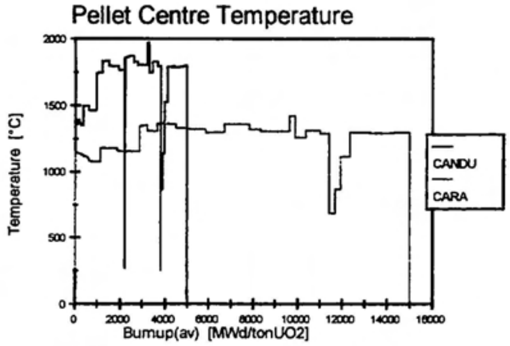

FIGURE 11: AVERAGED TEMPERATURE AT THE PELLET CENTER OF A CANDU FUEL ROD CARA FUEL.

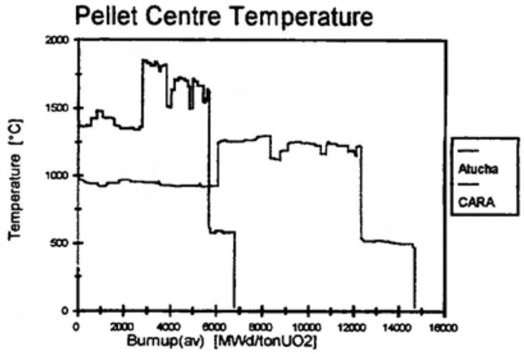

FIGURE 13: LOCAL TEMPERATURE IN THE SEVENTH SEGMENTS OF A FUEL ROD FOR ATUCHA I CARA FUEL.

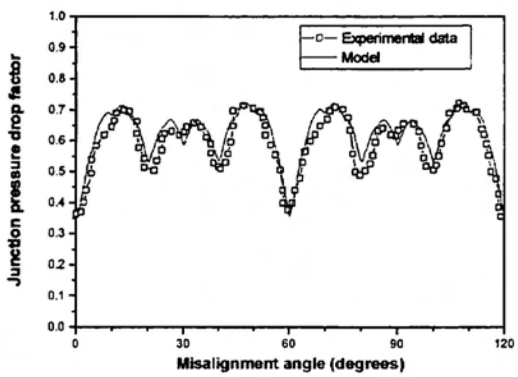

FIGURE 14: COMPARISON OF CANDU END PLATE FRICTION FACTOR COEFFICIENT BETWEEN EXPERIMENTAL AND THE DEVELOPED MODEL.

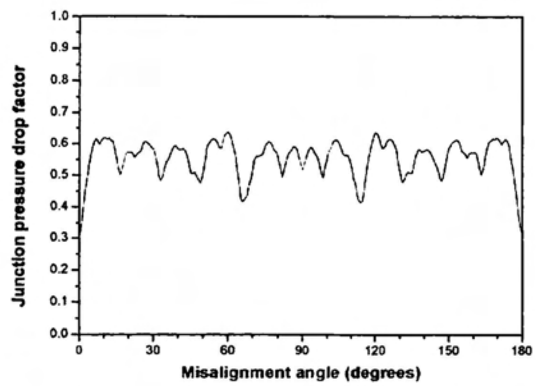

FIGURE 15: CALCULATED END PLATE FRICTION FACTOR COEFFICIENT FOR CARA FUEL USING THE DEVELOPED MODEL.