EVALUATION OF ADVANCED NUCLEAR FUEL ELEMENTS BEHAVIOUR PROPOSED FOR USE IN CANDU NUCLEAR REACTOR

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Abstract

In this study, different concepts for nuclear heavy water reactors fuel bundles are compared and analysed from the thermo-mechanical point of view by using specific computer codes. These new concepts, with 52 and 61 fuel elements, respectively, were designed by the Atomic Energy Commission of Argentina (CARA52) and the Nuclear Fuel Performance Group from RATEN ICN Pitesti (NU61), and contain natural uranium (0.72% enrichment in U^{235}) fuel. The geometrical characteristics were established in order to meet the requirements of the CANDU fuel channel. Their performances during irradiation were compared with the CANDU standard fuel bundle (37 elements) behaviour in normal operating conditions.

Keywords: *CANDU advanced fuel bundle, Natural uranium, Thermo-mechanical behaviour*

Introduction

CANDU fuel bundle design has evolved from Nuclear Power Demonstration fuel bundle (19 elements), to Pickering fuel bundle (28 elements), Bruce fuel bundle (37 elements) and CANFLEX fuel bundle (43 elements), respectively, [1].

Atomic Energy Commission of Argentina (CNEA) recently developed a new concept for heavy water reactors fuel bundles, named CARA (Combustible Avanzado para Reactores Argentinos), composed by 52 fuel elements, all characterized by the same fuel element diameter. The target of this new concept was to obtain a substantially improvement in fuel burn-up and security margins at reasonable costs, not higher than those for CANDU fuel bundle fabrication costs [2].

Based on this concept, Nuclear Fuel Performances Group from RATEN ICN Pitesti developed a new type of fuel bundle, composed by 61 identical fuel elements, and named NU61 (Natural Uranium). The new fuel bundle was designed using the INVENTOR-3D mechanical design, documentation and simulation tool. A higher number of fuel elements with smaller diameter comparatively with the CANDU standard fuel element diameter, enable to reach higher burn-up and lower fuel central temperature. The natural uranium is considered to be the fuel material inserted in a Zy-4 clad. Geometrical data of NU61 were established in order to ensure the compatibility with CANDU systems [3].

The fuel and cladding performance analyses were carried out in order to establish the irradiation performances of the two advanced concepts, CARA52 and NU61, using different computer codes. For CARA52, similar analyses were carried out in Argentina using BACO code [2].

Irradiation histories were obtained from the DIREN code calculations and the power peaking factors (PPF) resulted from the WIMS code simulations, previously performed [4]. Further on, the irradiation histories were used as input in TRANSURANUS, ELESIM and FEMAXI-6 codes along with geometrical data and a proper selection of the model options.

The paper contains a brief introduction of CARA52 and NU61 bundles provided along with the main features, a short discussion on the fuel computer codes used in simulation, including a description of the relevant modelling aspects involved, a section of results obtained on the fuel elements performance; finally, the main conclusions are drawn.

Fuel bundles general characteristics

Generally, CANDU fuel bundle design represents an optimization issue in which the aim is to achieve an adequate combination between constructive, technological, material and performance aspects.

The characteristic of CARA fuel bundle were set up in order to use the same fuel for both operating Argentine NPPs (Embalse and Atucha). Therefore, CARA bundle main objectives, among others, are: utilization of a single type of fuel element diameter, decreasing the fuel centre temperature, keeping the Uranium content, high fuel burn-up and similar fabrication costs with CANDU standard fuel bundle (CANDU37) [2].

This new configuration with 52 fuel elements was chosen due to the good symmetry and compactness of the array.

Fuel bundle length is 495.3 mm, same as for CANDU standard fuel bundle, CARA bundle geometry consisting in four concentrically rings with 4, 10, 16 and 22 fuel elements per ring, respectively. Outer cladding diameter is 10.8 mm, with 0.35 mm clad thickness. This prototype uses three spacer grids and two end plates at both ends (see Fig.1).

Other relevant characteristics are presented in Table 1. The values were obtained using the INVENTOR Autodesk program simulation.

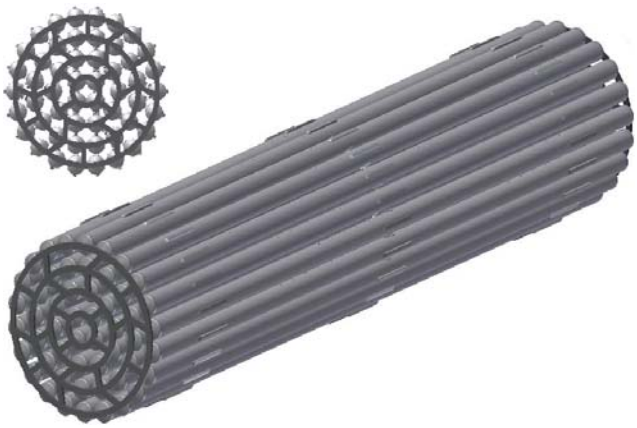


Fig.1. Argentine CARA52 fuel bundle [3]

NU61 fuel bundle, developed at RATEN ICN Pitesti, contains identical fuel elements disposed as follows: one central fuel element and four concentrically rings with 6, 12, 18 and 24 fuel elements, respectively. The outer diameter is 10 mm, with 0.35 mm clad thickness (see Fig.2). Each fuel element contains 39 fuel pellets with a diameter of 9 mm and natural uranium as fuel (0.72% enrichment in U^{235}). No CANLUB layer is considered.

The geometric dimensions of NU61 fuel bundle are illustrated in Table 1. This prototype [3] was designed using INVENTOR Autodesk program.

The geometry of NU61 bundle it is expected to lead to a better flattening of power distribution inside the bundle.

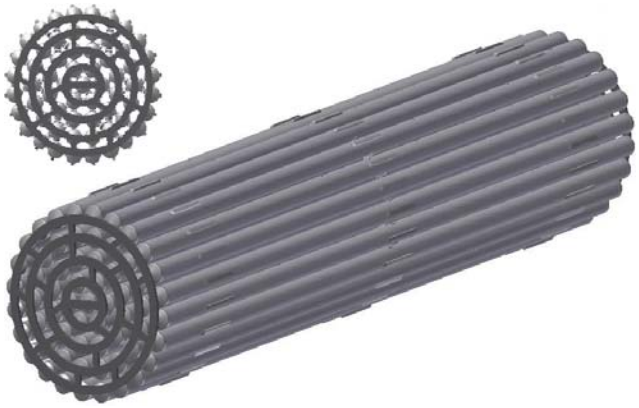


Fig.2. Romanian NU61fuel bundle [3]

Table 1. Geometrical characteristics of new bundle concepts

	CANDU37	**CARA52**	**NU61**
Fuel bundle	37 fuel elements	52 fuel elements	61 fuel elements
Fuel bundle lenght [mm]	495.3 ± 0.76	495.3 ± 0.76	495.3 ± 0.76
Fuel	UO_2 natural	UO_2 natural	UO_2 natural
Element lenght [mm]*	491.1	491.5	491.7
Clad	Zy-4	Zy-4	Zy-4
Axial gap [mm]	1.75	1.65	2
Diametral gap [mm]	0.09	0.09	0.09
Clad outer diameter [mm]	13.1	10.8	10
Clad thickness [mm]	0.40	0.35	0.35
Fuel stack lenght [mm]	480	481	480
Pellets number	29 + 2 end pellets	35 + 2 end pellets	37 + 2 end pellets
Pellet outer diameter [mm]	12.2	10	9
Pellet lenght [mm]	15.5	13	12
Pellet density [g/cm^3]	10.7	10.7	10.7
UO_2 mass [kg]	22.1	21	22
Total bundle mass [kg]	24.3	23	24

* over end caps

Fuel performance modelling

The main objectives of the Fuel Performance Group from RATEN ICN Pitesti regard the nuclear fuel performance simulations and the design of new fuel concepts with improved performances. In order to meet these objectives, several specific computer codes are used, as follows: INVENTOR (a 3D mechanical design, documentation and simulation tool) and ELESIM, FEMAXI-6 and TRANSURANUS (thermo-mechanical fuel behaviour analyses codes).

Fuel performance codes represents a key component in developing and designing advanced fuel concepts by giving a better understanding of nuclear fuel behaviour under normal and off-normal conditions. In this study, all three computer programs mentioned before were used to describe CARA52 and NU61 fuel behaviour to irradiation.

These fuel performance codes treat a single fuel element and the surrounding structure. In general, input files contain information about fuel element geometrical data, power histories and different models used to evaluate specific quantities, such as: thermal conductivity, fuel swelling, fission gas release, gap conductance, heat transfer coefficient between clad and coolant, etc.

Power histories for CARA52 and NU61 fuel bundles were obtained from previously performed DIREN calculations. Similar simulations were previously carried out for the CANDU standard fuel bundle with 37 elements (in the following will be mentioned as CANDU37) in order to compare the new fuel bundle prototypes performance with and the CANDU37 fuel bundle one. Applying the PPF (obtained in previously performed WIMS simulations) [4], the linear power for an outer fuel element from CANDU37, CARA52 and NU61 bundles was defined, as represented in Fig.3. For ELESIM calculation it is necessary to introduce as input data a power history with a burn-up step of 8-12 MWh/kgU. Similar data were used in TRANSURANUS and FEMAXI-6 input data files.

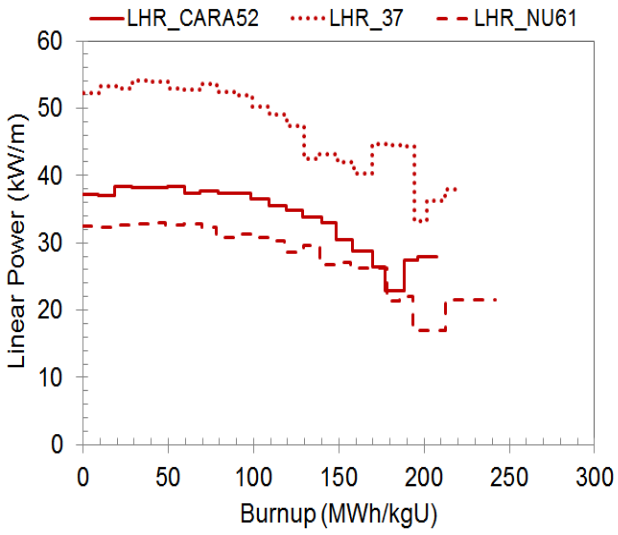


Fig.3. Power histories for an outer fuel element from CANDU37, CARA52 and NU61 fuel bundles

The power histories for NU61 and CARA52 fuel bundles are similar; the maximum values obtained were 32.8 kW/m and 38.2 kW/m, respectively. The lowest linear power levels were obtained for NU61 fuel element. The maximum linear power on an outer element for CANDU37 fuel bundle was 54 kW/m. Therefore, the utilisation of these new bundle concepts offers the possibility for the nuclear reactor to operate at slightly higher powers compared with CANDU standard fuel bundle.

Besides the linear power, some other relevant irradiation conditions should be defined in the input data files, such as: fast neutron flux, coolant pressure and temperature, and the heat transfer coefficient between clad and coolant (HTC). Some of these quantities can be prescribed, (e.g. coolant pressure and temperature) or internally calculated by codes subprograms (e.g. HTC). In order to calculate the heat transfer coefficient between clad and coolant, several quantities are required, such as the hydraulic diameter and coolant area [5].

Using power evolution together with geometrical data, complete thermo-mechanical behaviour could be calculated for CANDU37, CARA52 and NU61 fuel elements.

Results

In this section, the thermo-mechanical behaviour of CARA52 and NU61 fuel elements is shown with a comparison between ELESIM, FEMAXI–6 and TRANSURANUS codes predictions. The evaluation of the new concepts of fuel bundle was compared with the CANDU standard fuel element behaviour, used as reference in our study.

Fig.4 shows the fuel center temperature results obtained with ELESIM, FEMAXI-6 and TRANSURANUS codes for all studied fuel elements.

In the case of CARA52 and NU61 fuel bundles, the temperature reaches a maximum value of 1240 °C and 1080 °C, respectively, which are well below the values obtained for CANDU37 (~ 1800 °C). Therefore, the advance fuel elements behaviour under normal operating conditions allows a temperature decrease of ~500 °C for CARA52 and ~700 °C for NU61, respectively, compared with CANDU37. These maximum values were obtained with FEMAXI-6 code. TRANSURANUS code predictions are very similar with those provided by FEMAXI-6 code (only small differences can be observed), while ELESIM code under predicted both codes results with differences in the range of 150 °C – 180 °C (see Fig.4).

A group of researchers from CNEA Argentina performed similar calculations with BACO code for CARA52 fuel element, using a power history with a maximum level of 40 kW/m. BACO calculations show a maximum value of 1340 °C for fuel center temperature (using its associated power history) [6].

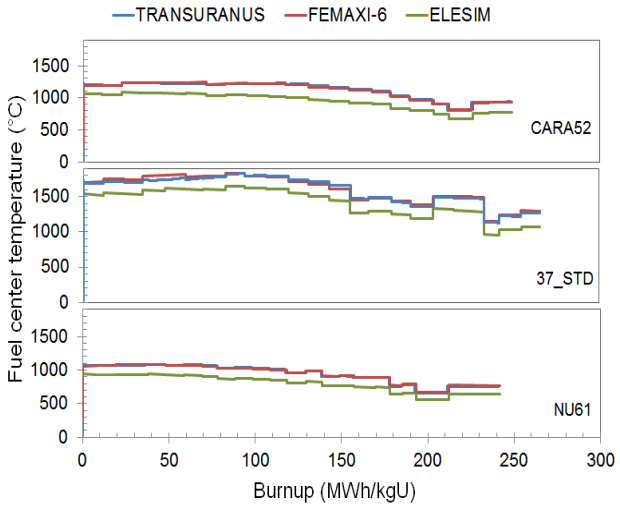


Fig.4. Fuel center temperature evolution for CARA52, NU61 and CANDU37 fuel bundles

All codes results show lower values for CARA52 and NU61 internal pressures compared with the CANDU37 ones (see Fig.5). In both cases, the maximum values does not exceed 1 MPa, while for CANDU37 a maximum value of 6 MPa can be observed (maximum value resulted from ELESIM predictions).

The differences between fuel performance codes predictions can be assigned to the different models used for evaluation of some important quantities, but this is not the object of our study, therefore we do not insist on this matter. However, the lowest values registered for CARA52 and NU61 bundles show an improvement in their behaviour, allowing also an improvement in the dimensional parameters, for e.g. small plenum, improved pellet dimensions (dish, land).

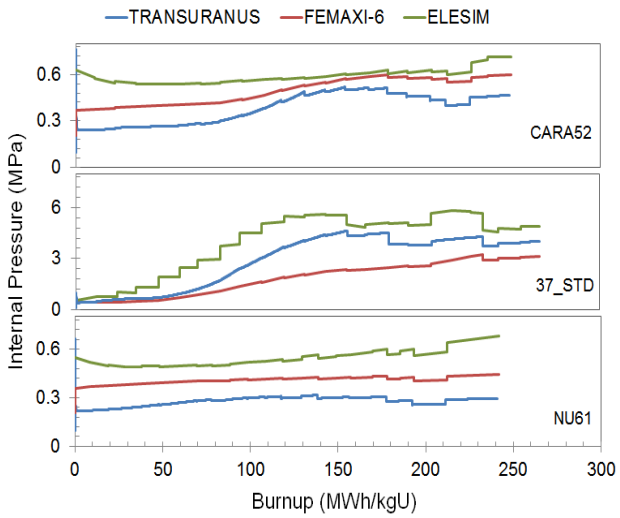


Fig.5. Internal pressure evolution for CARA52, NU61 and CANDU37 fuel bundles

Fission gas release (FGR) results evolution for the analyzed fuel bundles is shown in Fig.6. As it was expected, the lower linear power induces smallest fuel temperature and, consequently, lowers FGR.

For CARA52 bundle, FGR reaches almost 1 cm^3, and for NU61 bundle FGR values are very small, at about 0.6 cm^3. ELESIM predictions do not register a significant release, compared with other two fuel codes used in analyses, but FEMAXI–6 and TRANSURANUS results are in a good agreement.

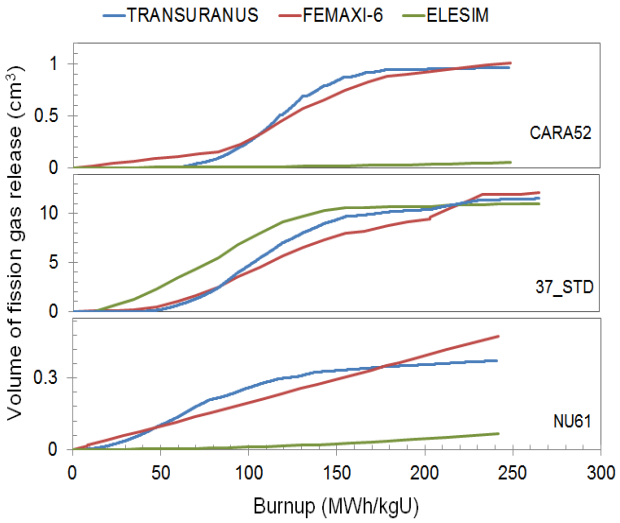


Fig.6. Fission gas release evolution for CARA52, NU61 and CANDU37 fuel bundles

Conclusions

In this paper, ELESIM, FEMAXI-6 and TRANSURANUS fuel performance codes have been used in CARA52 and NU61 fuel elements analysis.

Fuel centre temperature, internal pressure and fission gas release have been investigated in the light of CANDU standard fuel behaviour.

As the main results of the analyses performed, the advantages of the CARA52 and NU61 fuel elements in normal operating conditions were the following: significant benefit in terms of fuel temperature reduction, smaller fission gas release and internal pressure.

Given these satisfactory results, one can conclude that both advanced fuel bundle concepts can be safely used in CANDU nuclear reactor.

CARA52 design already attracted the interest of the nuclear power utility in Argentina (NASA) and the fuel manufacturing company (CONUAR). In Romania, NU61 design is one of the best options to move from ahead the actual natural uranium CANDU fuel behavior to a next stage with a lot of benefits.

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