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PRESSURE DROP VARIATION AS A FUNCTION OF AXIAL AND RADIAL POWER DISTRIBUTION IN CANDU FUEL CHANNEL WITH STANDARD AND CANFLEX 43 BUNDLES

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CANDU 600 nuclear reactors are usually fuelled with STANDARD (STD), 37 rods fuel bundles. Natural uranium (NU) dioxide (UO_2), is used as fuel composition. A new fuel bundle geometry called CANFLEX (CFX) with 43 rods is proposed and some new fuel composition are considered. Flexibility is the key word for the attempt to use some different fuel geometries and compositions for CANDU 600 nuclear reactors as well as for innovative ACR-700/1000 nuclear reactors.

The fuel bundle considered in this paper is CFX-RU-0.90 that encodes the CANFLEX geometry, recycled dioxide uranium (RU) with 0.90% enrichment. The goal of this proposal is ambitious: a higher average discharge burn-up up to 14000MWd/tU and, for the same amount of generated electric power, reduction in nuclear fuel fabrication, reduction of spent nuclear fuel radioactive waste and reduction of refueling operational work by using fewer bundles.

An improved sub-channel approach for thermal-hydraulic analysis is used in this paper to compute some flow parameters, mainly the pressure drop along the CANDU 600 fuel channel when STD or CFX-RU-0.90 fuel bundles. Also an intermediate CFX-NU fuel bundle are used, for gradual comparison. For CFX-RU_0.90 four fuel bundle shift refueling scheme is used instead of eight, that will determine different axial power distributions. In the same time radial power distribution is affected by the geometry and by the fuel composition of fuel bundle type used. Some other thermal-hydraulic flow parameters will be influenced, too. One of the most important parameter is pressure drop (PD) along the fuel channel because of its importance in drag force evaluation. We start with an axial power distribution, which is characteristic for a refueling scheme of eight or four fuel bundles on a shift. Comparative results are presented between STD37 , CFX-NU CFX-RU-0.90 fuel bundles in a CANDU nuclear reactor operating conditions. Neutron flux distribution analysis shows that four bundle shift scheme is suited for CANFLEX-RU-0.90 fuel bundles while eight are suited for STD37 andCFX-NU.

Some other thermal-hydraulic parameters like critical heat flux, density distribution, void distribution, velocities, are computed and some briefly presented.

Keywords: CANDU, pressure drop, CANFLEX, recycled Uranium, sub-channel thermal-hydraulics, waste minimization.

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1. Introduction

Two CANDU 600 nuclear reactor type are to be operated in Romania in late 2007. Natural Uranium (NU) dioxide (UO_2) is used in STANDARD (STD37) fuel bundles. CANDU reactor was designed to use NU as fuel in specific fuel bundle geometry. The use of other geometries and fuel compositions could be considered. An alternative is slightly enriched uranium (SEU) obtained by recycling LWR discharged uranium (RU). The use of RU brings significant benefits in terms of residence time of the fuel elements and the average discharge burn-up. New geometries and fuel compositions permit a new refueling scheme which could lead to a better axial and radial power distribution in fuel channel. A solution is CANFLEX (CFX) geometry with 43 rods, filled with different SEU fuel that in this paper is supposed to be recycled uranium (RU) with 0.90% U235 enrichment. Enrichment fractions up to 1.2% could be considered for use in CANDU 600 nuclear reactors. In this paper NU is used in STD37 fuel bundles; NU and RU-0.90 in advanced CFX fuel bundles, in normal operating conditions.

The main objective of analysis presented in this paper is to determine the pressure drop (PD) and other relevant thermal-hydraulic parameters for a fuel channel of an equilibrium core fuelled with STD or CFX fuel bundles, with NU and 0.90% compositions, and to make some comparisons. The bulk of PD is determined mainly by four factors: friction, gravitation, acceleration forces and geometrical local factors like bundle junctions, spacers plane and bearing pads planes [1]. Methodology for geometrical local factors prediction is given in [1] and briefly presented in chapter 3 of this paper. Once we have fixed these PD factors we proceed to a sensitivity analysis having in mind the question: how the use of CANFLEX geometry, a new fuel composition and therefore a new refuelling scheme will influence pressure drop (PD) and other thermal-hydraulic parameters compared with those obtained for STANDARD fuel.

The axial channel heat flux distribution of a channel with CFX-RU is quite different from that of STD37-element NU channel, but there are differences also between STD37 and CFX-NU on one hand and between CFX-NU and CFX-RU fuel types. Radial channel heat flux distribution is influenced by bundle geometry, fuel enrichment and average discharge fuel burn-up for these fuel bundle types.

The sub-channel analysis of pressure drop in a CANDU 600 fuel channel was made with a sub-channel COBRA class computer code [2]. Some analysis were made for these fuel types in other countries [3, 4]. Valuable neutron data like RFD and AFD are taken from works conducted in RAAN-SCN Pitesti [5, 6, 7] and valuable thermal-hydraulic information from Canadian work [8] and Romanian [9]. Previous works and papers of the authors [10, 11, 12,13,14] are a basis for the core of this paper. The results of this analysis are presented in a condensed graphical format which is briefly explained in chapter 2.

2. Pressure drop in a CANDU channel with CANFLEX-RU-0.90

2.1. Generalities

The CANDU structure permits flexible fuel cycles. Besides the current natural uranium (NU) dioxide standard fuel cycle, an alternate option is recycled uranium (RU) from LWR nuclear reactors with different U^{235} enrichments.

The use of low-enrichment uranium in actual and enhanced CANDU or in ACR 700/1000 nuclear reactors will enable reduced capital cost, increased output extended fuel burn-up and in the same time waste and refueling minimization.

An alternate geometry to the STD (Fig.1) fuel namely CFX geometry (see Fig. 2) is proposed for the above mentioned fuel compositions. The resulting new fuel bundle is called in this work either: CFX-NU for NU fuel composition, CFX-RU-0.90. CFX-RU-0.96, CFX -SEU-1.1 for RU fuel with 0.90%, 0.96% and 1.1% U235 enrichment fractions are further option to be considered.

2.2. Thermal-hydraulic analysis for pressure drop

The thermal-hydraulics analysis of fuel channel is of a key importance in nuclear reactor safety with impact on environmental issues. In this paper the variation of pressure drop along fuel channel is analyzed when different fuel bundle geometries and fuel compositions are used. The operating conditions are those nominal for a CANDU nuclear reactor. Fuel bundles components dimensions, positions and shapes have a special meaning for pressure drop. The pressure drop will influence the drag force that is exercised on fuel bundles and on other devices. Pressure drop is determined by components of fuel bundle string: fuel rod array, end plates, spacer pads, bearing pads. A methodology for pressure loss coefficients evaluation is given in [1, 2] and is presented briefly in chapter 3 of this paper. Some other parameters influence the amount of pressure loss: void fraction, fluid temperature and turbulence. These parameters are influenced by axial heat flux distribution (AFD) and radial flux distribution (RFD) which in turn are direct consequence of refueling strategy which varies with fuel type. Fuel geometry, fuel composition, burn-up and reactivity control devices have also impact on the shape of axial power distribution and on radial channel power distributions. In Fig. 3 are presented some axial heat flux distributions (AFD) both theoretical real cos and specific skewed cos distributions. In Table 1 are given channel radial heat flux distributions (RFD) for STD37 bundles and in Table 2 those for CFX_NU and CFX_RU_0.90 fuel bundles respectively. With all operating conditions, geometric and physical fixed, series of analysis were made (at least nine) for pressure drop variation along fuel channel. The results for STD37, CFX-NU and CFX_RU_0.90 fuel bundles computed for different axial heat flux distributions are given in Fig. 4 and details in Fig. 5.

The pressure drop (PD) for STD37 fuel bundle considering real cos axial heat flux distribution is 654.951 kPa while PD for channel exit skewed cos shape is 645.016 kPa. By simple changing the geometry of bundle but not the fuel composition, that is CFX-NU fuel bundle, and using the same axial distributions we have 694.970 kPa drop for real cos and 684.535 for the same skewed cos shape. Regarding CFX-RU-0.90 we considered only a channel inlet skewed cos which characterize four bundle refueling scheme with a PD of 716.45 for 8000 MWd/tU discharge burn-up and 717.482 for 12000 MWd/tU discharge burn-up. Since the PD for CFX-RU-0.90 and real cos is 693.801 kPa near that of CFX-NU (694.970 kPa) we do not overload the figures with this result. Synthetically we have:

STD_R	STD-SK	CFX-NU	CFX-SK	CFX-RU-8000	CFX-RU-12000
654.951	645.016	694.970	684.535	716.450	717.482.

We observe that the ratio of pressure drop due to geometry, that is CANFLEX geometry versus STANARD geometry gives us an increase with 6.1% in pressure drop on CANDU fuel channel. Also keeping the same CANFLEX but changing the fuel composition, and therefore refueling scheme and consequently channel axial heat flux distribution we have another 4.7% increase in PD. Finally between two most probable situations which are STD37 and CFX-RU-0.90 fuel bundle types the difference in almost 11.1 % PD increase which is significant.

The goal of this analysis is to provide valuable information regarding the use of new fuel types (geometry and composition) to see the behaviour of some thermal-hydraulic parameters that are significant for a safe CANDU nuclear reactor operation. While it is almost certain that from economic and environmental point of view, that is more generated electric power and less radioactive waste for are desired benefits, at least the conservation of actual degree of safety must be preserved when new fuel bundles are used in a CANDU 600 nuclear reactor such as those which are operated at Cernavoda, Romania.

3. Pressure loss due to local geometry forms

The general form of equation is:

$$\Delta P_{local} = K_{local} \frac{G^2}{2\rho_l}. \tag{1}$$

$K_{junction}$ for aligned or miss-aligned fuel bundles is:

$$K_{junctiun} = C_{ecc}\left(K_{imp} + K_{sep}\right) \tag{2}$$

$K_{spacers}$ for fuel bundles is:

$$K_{spacer} = C_{ecc} C_{shape} K_{imp} \tag{3}$$

are presented in [1, 10].

4. Tables

Radial power distribution for Standard 37 fuel bundle expressed using power peaking factors by rods type or relative power density in each burnable material.

Table 1

Radial power distribution for STANDARD 37 fuel bundle (U235, 0.71%)

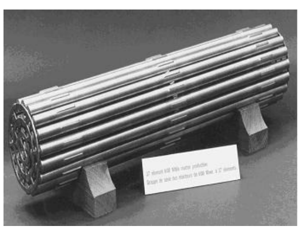
RING	STD37 (0 MWd/tU)	STD37 (4000 MWd/tU)	STD (8000 MWd/tU)	RODS NR.	RODS DIAM [CM]
1	0.757	0.751	0.763	1	1.308
2	0.798	0.793	0.804	6	1.308
3	0.909	0.907	0.910	12	1.308
4	1.141	1.145	1.139	18	1.308

Table 2

Radial power distribution for CFX_NU and CFX_RU_0.90 fuel bundle

RING	CFX_NU (0 MWd/tU)	CFX _NU 8000 MWd/tU	CFX_RU 0.90 U235 (0 MWd/tU)	CFX_RU 0.90 U235 (8000 MWd/tU)	RODS NR.	RODS DIAM [CM]
1	0.912	0.943	0.860	0.940	1	1.365
2	0.970	0.970	0.915	0.970	7	1.365
3	0.900	0.900	0.880	0.905	14	1.146
4	1.085	1.085	1.125	1.075	21	1.146

5. Figures



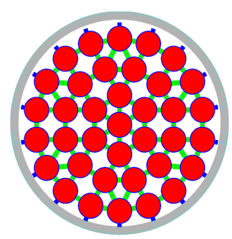


Fig.1. CANDU 600 Standard 37 fuel rods bundle. Side view of PT, rods, spacers and bearing pads.

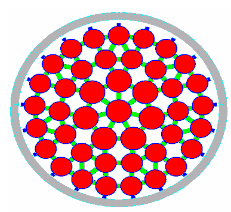


Fig. 2. CANFLEX 43 fuel rods bundle. Side view of pressure tube, rods, spacers and bearing pads.

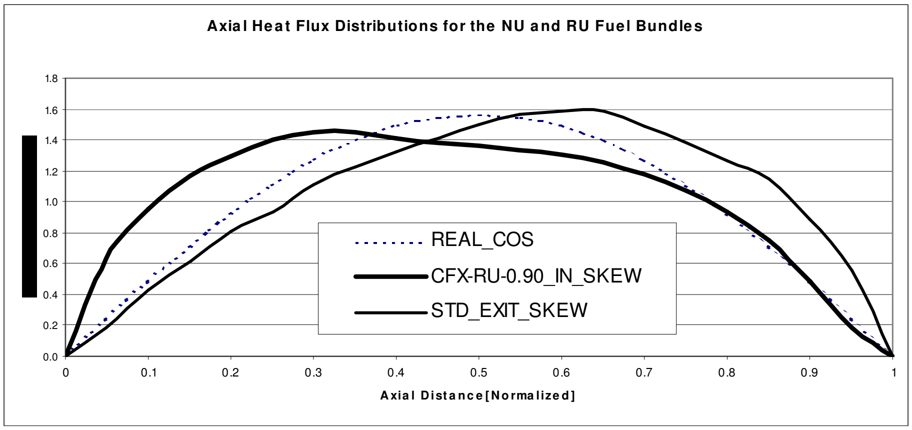


Fig. 3. Axial heat flux distributions for STD37, CFX-NU, CFX-RU-0.90 and Real cosine.

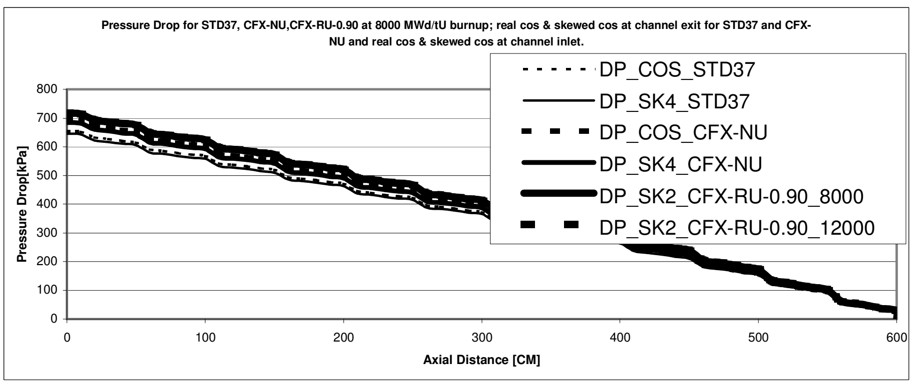


Fig. 4. Pressure Drop for STD37, CFX-NU, CFX-RU-0.90.

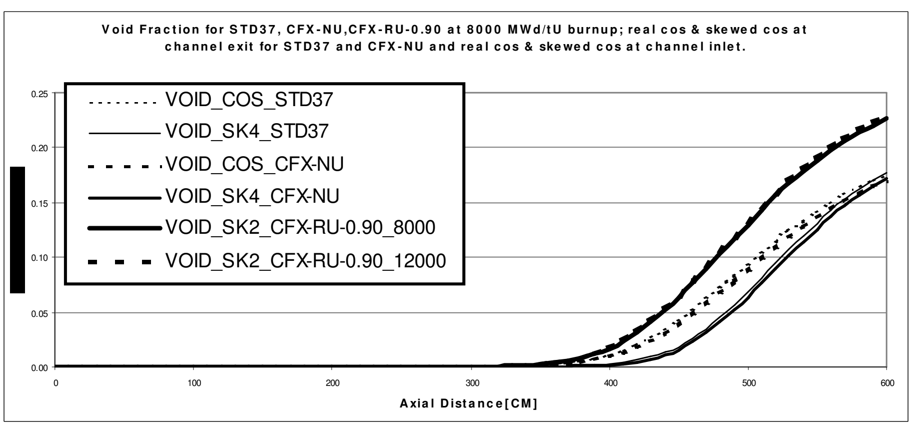


Fig. 5. Void Fraction for STD37, CFX-NU, CFX-RU-0.90.

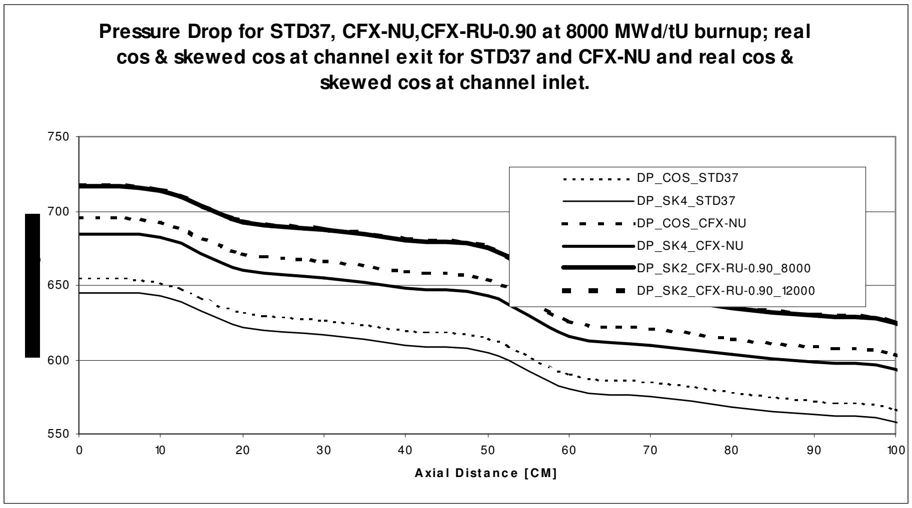


Fig. 6. Pressure Drop (Details for STD37, CFX-NU, CFX-RU-0.90; real cos & skewed cos at channel exit for STD37 and CFX-NU and real cos & skewed cos at channel inlet).

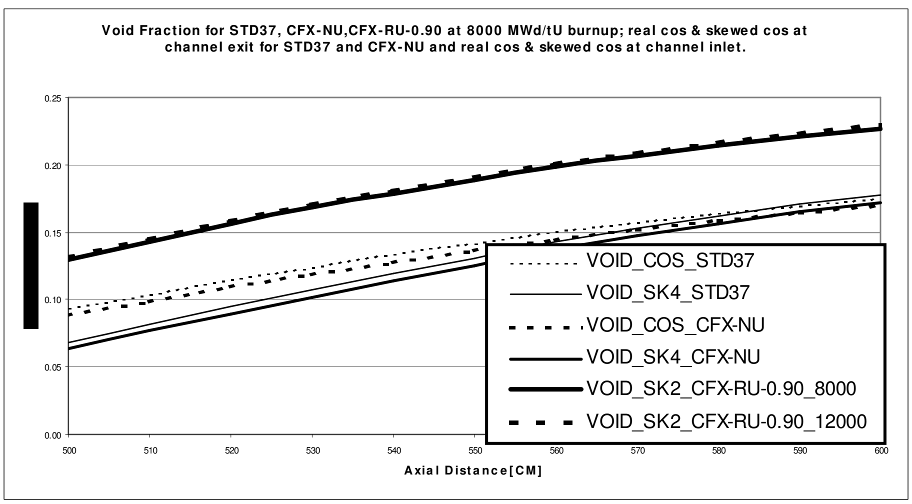


Fig. 7. Void Fractions (Details for STD37, CFX-NU, CFX-RU-0.90; real cos & skewed cos at channel exit for STD37 and CFX-NU and real cos & skewed cos at channel inlet)

6. Conclusions

Main conclusion for analysis in this paper is obvious from the Fig. 3 to Fig. 7: Pressure Drop (PD) for CFX-NU is higher than PD for STD37 fuel types both for theoretical real cos axial heat flux distribution and for channel exit skewed cos axial heat flux distribution specific for 8 bundle shift with 6.11%.

Further PD for CFX-RU-0.90 using a channel input skewed cos axial heat flux distribution specific for 4 bundle shift is higher than PD for CFX-NU with 4.67%, and with 11.1% higher than PD for STD37. Channel average void fraction for CFX-NU is smallest while that for CFX-RU is highest. The benefits expected by using the new CFX-RU fuel bundles have a price both in pressure drop and in void fraction. These parameters have increased values for CFX-RU but differences are not dramatic. Further specialized evaluation have to be made for drag force and for the consequences that void fraction differences could have.

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