THERMAL-HYDRAULIC ANALYSIS OF A BOILING WATER REACTOR CHANNEL OPERATING WITH ANNULAR FUEL

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الملخص

تقدم هذه الورقة البحثية دراسة هيدروحرارية لتصميم جديد من وقود المفاعلات النووية المجوف الذى تم تطويره بمعهد ماسيشوستس للتقنية، MIT، يتم فيه تبريد قضبان وقود المفاعل من الداخل والخارج ويتوقع أن يسمح بتشغيل المفاعلات النووية المبردة بالماء الخفيف بقدرات أكبر (قد تصل إلى 150% من قدرات التشغيل الحالية) وبهوامش أمان أفضل بكثير من تلك السائدة في المفاعلات الحالية التي تستخدم الوقود الاسطواني المصمت المبرد من الخارج فقط عند نفس ظروف التشغيل. استخدم في التحليل برنامج حاسوب كتب في مشروع سابق لتحليل قناة مفاعل ماء مغلي تقليدي يستخدم فيها الوقود من النوع المصمت بعد أن تم تعديله ليلائم الشكل الهندسي للوقود الجديد (المجوف) قيد الدراسة بهدف مقارنة نتائج الوقود القديم بنتائج الوقود الجديد.

من بين أهم النتائج التي تم الحصول عليها مقارنة أقصى درجة حرارة للوقود وكسر الحيود عن الغليان المنتظم في القناة بالنسبة لتصميمي الوقود، حيث كانت أقصى درجة حرارة للوقود المصمت 1800 درجة مئوية بينما كانت أقصى درجة حرارة للوقود المجوف عند نفس ظروف تشغيل المفاعل 1026 درجة مئوية. أما كسر الحيود عن الغليان المنتظم فكان 2.0 في الوقود المصمت و 3.1 في الوقود المجوف. وهذه النتائج توضح أن هوامش الأمان للتصميم الجديد من الوقود (المجوف) أكبر بكثير منها في الوقود القديم (المصمت). كما بينت النتائج أنه يمكن تشغيل المفاعل بهوامش أمان مناسبة باستخدام الوقود المجوف وبقدرة تعادل 130% من قدرة الوقود القديم حيث كانت أقصى درجة حرارة للوقود 1234 درجة مئوية وأصغر كسر حيود عن الغليان المنتظم حوالي 2.6.

ABSTRACT

This paper presents a thermal-hydraulic analysis of a Boiling Water Reactor (BWR) channel operating with a new fuel design concept of an annular shape, developed by the Massachusetts Institute of Technology (MIT). The analysis is carried out using a computer code written in a previous project for the analysis of typical conventional BWR channel using the traditional solid fuel rod design. The code is modified in this work to suit the annular fuel geometry, in which the fuel is cooled from both sides (inside and outside), and is successfully implemented to perform the thermal hydraulic analysis of the new design concept.

The results obtained for the old and new fuel design concepts are presented and compared for the same operating conditions. The results showed that the peak temperature of the annular fuel was 1026°C and the MDNBR was 3.1, while the peak temperature of the solid fuel was 1800°C and the DNBR was 2, indicating considerable improvements of the reactor safety margins of the new fuel design. The results also showed that the annular channel can be operated safely with good safety margins under

130% overpower condition with a peak fuel temperature of 1234^{o}C and a MDNBR of 2.6, while it was not possible to operate the reactor with the old solid fuel at this power level.

KEYWORDS: Boiling Water Reactor (BWR); Solid fuel; Annular fuel, Power density; Peak fuel temperature; Departure from Nucleate Boiling Ratio (DNBR); Reactor safety; Safety Margins.

INTRODUCTION

An internally and externally cooled annular fuel design was proposed by Kazimi and his colleagues at the Massachusetts Institute of Technology (MIT) [1] as part of a new design concept for LWR fuels with the objective of substantially increasing the reactor power density while improving safety margins. Kazimi et al's report presented results for introducing the new annular fuel design to a traditional Pressurized Water Reactor, PWR, core and their results were very encouraging. Based on their results several neutronic and thermal hydraulic analysis of PWR cores using the new annular fuel design were carried out at the Nuclear Engineering Department in the Faculty of Engineering of Tripoli University [2-4] and the results obtained confirm the advantages of using the annular fuel design. Therefore, it was decided to carry out a similar study that analyzes the introduction of the new annular fuel design into a traditional Boiling Water Reactor, BWR, core and this paper presents the results of such study. The geometry of the annular fuel design is shown schematically in Figure (1) where the traditional solid fuel design is also drawn for comparison purposes. It can be noted from the figure that the annular fuel design has significantly larger diameter than the typical solid rods to accommodate the inner coolant channel allowing for sufficient coolant flow.

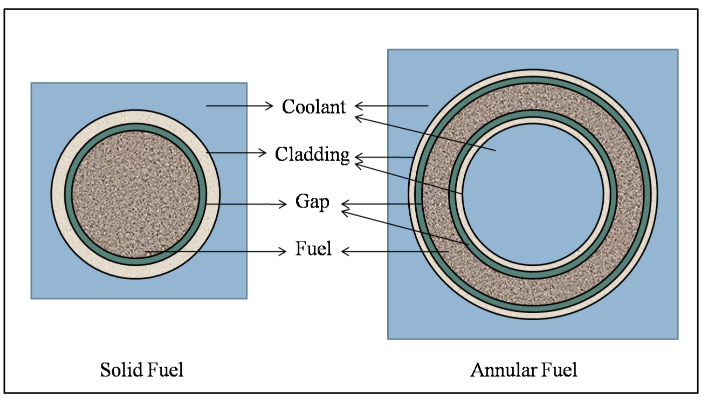

Figure 1: Schematic diagrams of solid and annular fuel rod geometries.

The replacement of the traditional solid geometry with the new annular geometry has two important implications that allow for higher reactor power density, they are:

1. The reduction of the thickness of the heat conduction path (thermal resistance), which allows for better heat transfer from the fuel and hence help increase the

margins between the fuel and clad peak temperatures and their corresponding melting points.

2. The increase of heat transfer area between the clad surface and the coolant, which leads to an increase of the Departure from Nucleate Boiling Ratio (DNBR) margin.

In addition to the lower peak fuel temperature and higher DNBR limits, the new fuel has to satisfy a number of other safety limits and performance constraints. For example, the annular fuel design concept exhibits significantly lower fuel temperature than solid fuel, hence it is expected that fission gas release will be smaller allowing for higher fuel burn-up.

In the present analysis it will be assumed that the new annular fuel is introduced into the same traditional BWR core keeping the number of assemblies and their overall size fixed and distributing the new fuel design equally spaced in the fuel assemblies [5]. The fuel enrichment and composition is also assumed unchanged from the traditional solid fuel. In addition, it is also assumed that the inner diameter of the new annular fuel is the same as the outside diameter of the old solid fuel as can be noted from Figure (2). These constraints necessitate that the new fuel assembly array be 5 by 5 instead of the old 8 by 8 array as shown in Figure (3). This will of course reduce the number of fuel rods in each assembly and hence in the entire core as shown in Table (1).

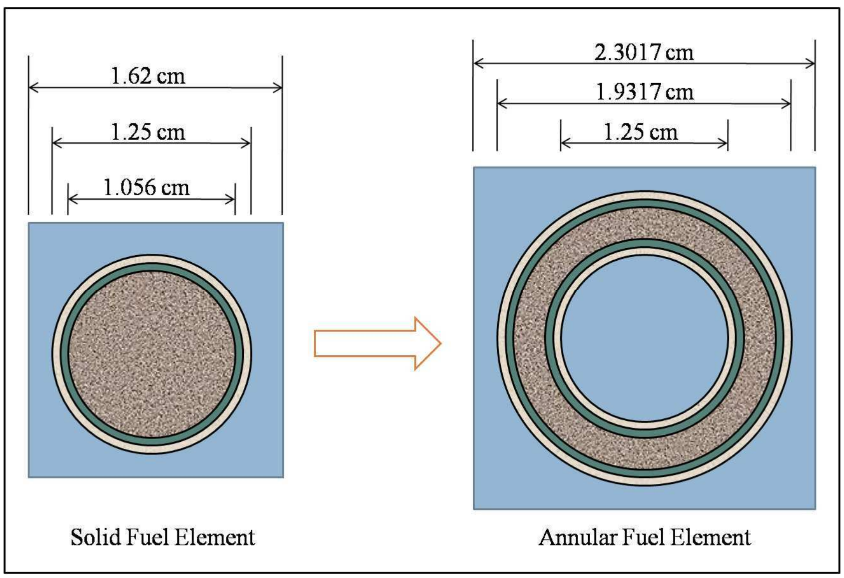

Figure 2: Solid and Annular Fuel Element Dimensions.

Table 1: Comparison of number of fuel assemblies and rods

Parameter	Solid fuel	Annular fuel
Number of assemblies	732	732
Fuel element/assembly	63/ (8x8)	24/ (5x5)
Total number of fuel rod	46,116	17,568

Temperature Distribution Equations for Annular Fuel geometry

The new annular fuel model is defined by five regions as shown in Figure (4) namely: inner cladding, inner gap, fuel meat, outer gap and outer cladding. The fuel is cooled by water which is allowed to flow adjacent to the inner cladding in the inner circular region of the fuel element (inner channel) and on the outside of the outer cladding (outer channel). Therefore, the coolant channel will be divided into two sides, namely, inner and outer. Heat transfer from the annular fuel rod to the cladding surface, in contact with the coolant, is computed by the numerical solution of the1-D heat conduction equation in cylindrical coordinates. The relevant radial temperature distribution equations are derived using appropriate energy balances in the fuel, gap and cladding for both inner and outer sides of the channel and are as follows [5,6] (see nomenclature section for symbol definition):

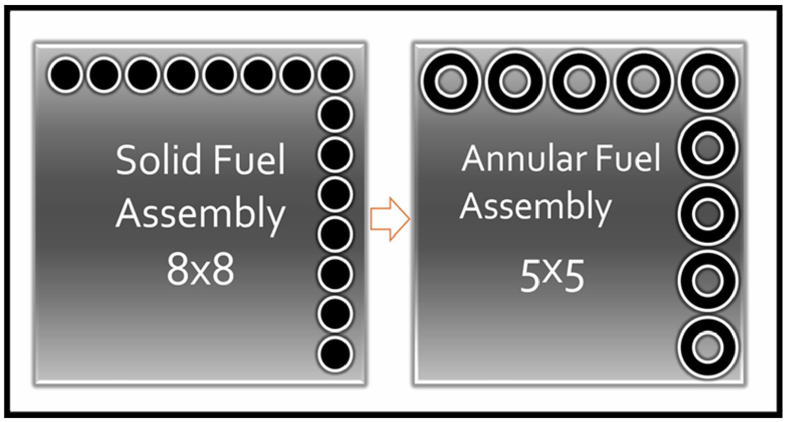

Figure 3: Comparison of solid and annular fuel assemblies.

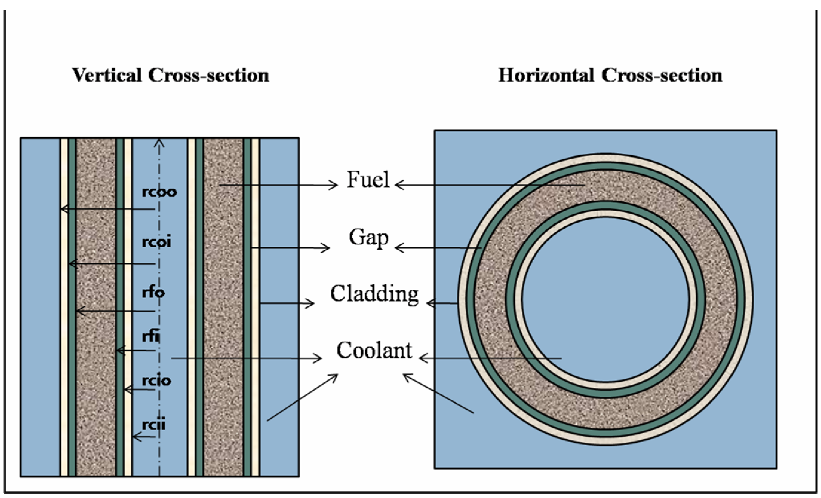

Figure 4: Schematic diagram of annular fuel regions.

- radial temperature distribution equation in the inner fuel:

$$T_{fi} = T_{max} + \frac{q'''(r_{max}^2 - r_{fi}^2)}{4k_f} + \frac{q''' r_{max}^2}{2k_f} \ln\left(\frac{r_{fi}}{r_{max}}\right) \tag{1}$$

- radial temperature distribution equation in the outer fuel:

$$T_{fo} = T_{max} + \frac{q'''(r_{fo}^2 - r_{max}^2)}{4k_f} + \frac{q''' r_{Max}^2}{2k_f} \ln\left(\frac{r_{max}}{r_{fo}}\right) \tag{2}$$

- radial temperature distribution equation in the inner gap:

$$T_{gi} = T_{fi} + \frac{q'''(r_{max}^2 - r_{fi}^2)}{2k_g} \ln\left(\frac{r_{cio}}{r_{fi}}\right) \tag{3}$$

- radial temperature distribution equation in the outer gap:

$$T_{go}(r) = T_{fo} + \frac{q'''(r_{fo}^2 - r_{max}^2)}{2k_{go}} \ln\left(\frac{r_{fo}}{r}\right) \tag{4}$$

- radial temperature distribution equation in the inner cladding:

$$T_{si} = T_{gi} + \frac{q'''(r_{max}^2 - r_{fi}^2)}{2k_c} \ln\left(\frac{r_{cii}}{r_{cio}}\right) \tag{5}$$

- radial temperature distribution equation in the outer cladding:

$$T_{so} = T_{gi} + \frac{q'''(r_{fo}^2 - r_{max\,i}^2)}{2k_{go}} \ln\left(\frac{r_{coi}}{r_{coo}}\right) \tag{6}$$

Reactor Channel Analysis

The analysis of the annular fuel channel is carried out using a modified version of a thermal hydraulic computer code written by Lamya Banoun in a previous project [7] for the thermal hydraulic analysis of a traditional BWR channel that uses solid fuel. The code modification included the channel geometry and the distribution of the channel coolant flow between the inner and outer sides of the annular channel. The computer code is based on the following basic assumptions [8, 9]:

1. The volumetric thermal source strength, q''', is a function of only the axial position and can have either a sinusoidal or bottom peaked profile.
2. Constant thermal conductivity of the coolant.
3. Heat transfer in the axial direction is neglected.
4. Constant slip ratio.

5. The quality is calculated theoretically from the given power distribution.
6. Constant coolant mass flow rate along the flow channel.
7. The pressure drop in the coolant channel is due to friction and acceleration losses only.

The code inputs include geometry parameters such as; internal and external fuel rod diameters (cm) DFI and DFO, internal and external clad diameters (cm) DCII, CIO, DCOI and DCOO, fuel rod pitch S (cm), active core height H (cm), active diameter of the core D (cm), number of fuel assembly FAN (NO.), assembly array size AA and number of fuel rods FRN. The inputs also includes operating parameters such as; total mass flow rate in the core MFRPhr (kg/hr), inlet pressure Pin (Pa), inlet temperature Tin (^{o}C) and total thermal output of the reactor Qt (MW_{th}).

Computer Code operational steps

The computer code is designed to be user friendly and uses input and output files to store data for plotting and analysis. The main operational steps of the code can be summarized as follows [6,7]:

- Reading and printing all input data
- Calculation of mass flow rate and mass velocity for each flow channel.
- Calculations of volumetric thermal source strength for each channel.
- Dividing the channel into a specified number of axial increments and performing the following steps for each increment:
    - Calculation of local coolant bulk temperature at the end of the increment TB(z).
    - calculation of the average temperature average temperature for the increment TBA(i)
    - Calculations of local quality and void fraction x(z), α(z).
    - Calculations of local pressure drop Δp and saturation temperature Tsat.
    - Recalculation of Tsat, TB(z), x(z) and α(z) to include the effect of Δp.
    - Calculation of local heat transfer coefficient h(z).
    - Calculation of cladding surfaces temperature.
    - Calculation of critical heat flux and check for burnout.
    - Calculation of departure from nucleate boiling ratio.
    - Calculation of radial temperature distribution (in clad, fuel and gap) at each axial location.
- Calculation of boiling and non-boiling heights.
- Saving results at each axial location and at core exit for plotting and analysis.

RESULTS AND DISCUSSION

The results presented in this paper are obtained by implementing the old and new versions of the computer code for a typical BWR core using commonly used values of slip ratio (2.23), hot channel factor (1.4) and average channel exit quality factor (20%.).

Axial variation of Quality and Void Fraction:

Figures (5 and 6) show the axial variation of quality and void fraction in typical BWR channels using sold and annular fuel designs. It can be noted from these figures that the behavior of the quality factor and void fraction in the solid fuel exhibit the same general behavior as in the annular fuel. It can also be noted that the non-boiling height of the annular channel is larger than that for the channel in which the solid fuel geometry is used. Near channels exits the values of quality approach the pre-specified design values (20%) and the corresponding void fraction is about 70%.

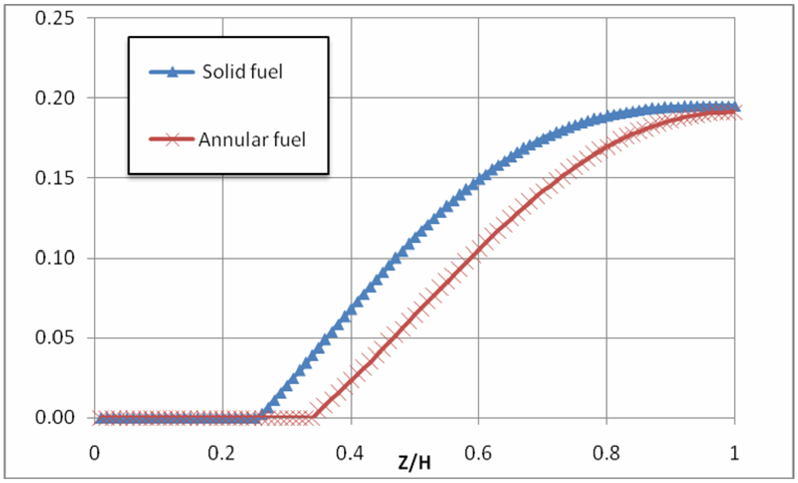

Figure 5: Axial variation of quality in solid and annular channels.

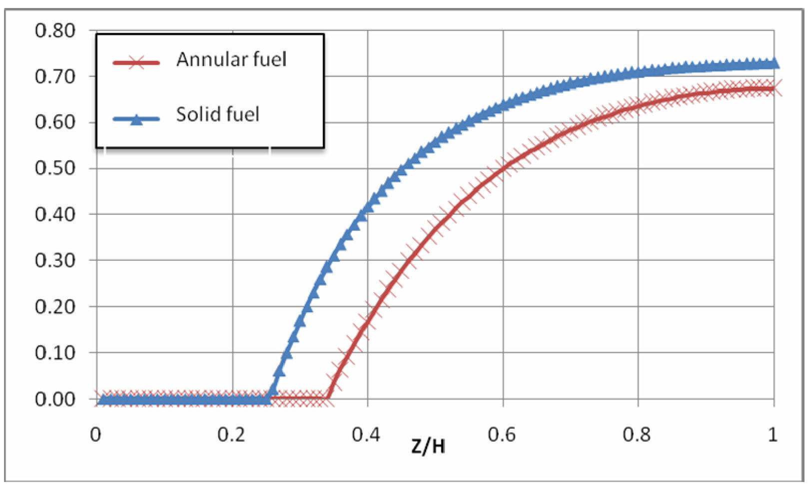

Figure 6: Axial variation of void fraction in solid and annular channels.

Axial variation of coolant and cladding surface temperatures

Figure (7) presents the axial variation of coolant and cladding surfaces (inner and outer) temperatures. It can be seen from the figure that the coolant bulk temperature increases with axial position in the single phase entrance region (non-boiling height) until it reaches the saturation value corresponding to the prevailing system pressure at the start of boiling. In the boiling height, if we ignore the pressure drop, the coolant temperature remains constant and equal to the saturation temperature at the system pressure, but due to the pressure drop the temperature in the boiling height decrease slightly.

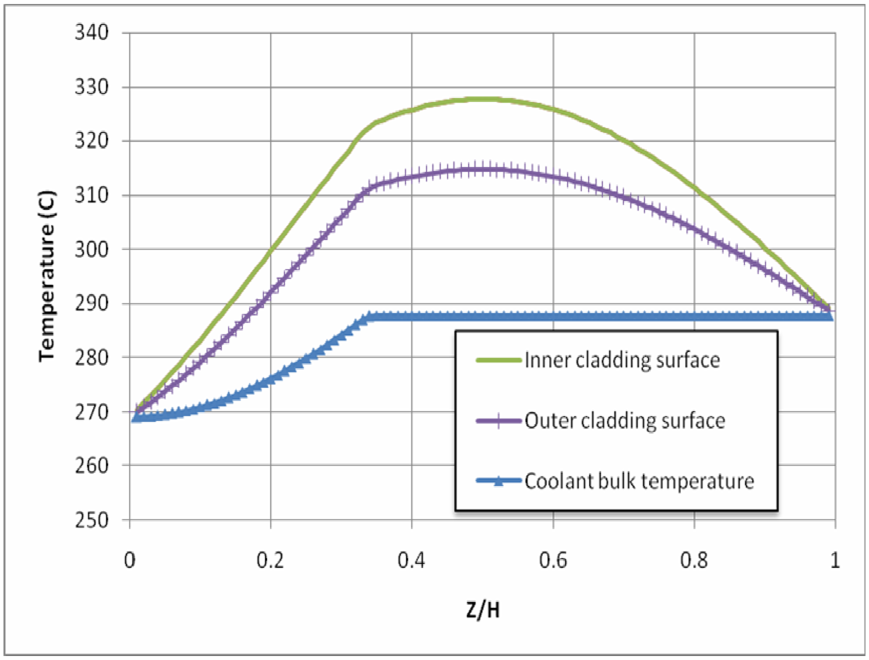

Figure 7: Variations of coolant and cladding surface temperatures for annular fuel

Axial Variation of fuel Temperatures.

Figure (8) shows the axial variations of the fuel maximum T(Max), inner and outer surfaces temperatures of the annular fuel design. It can be seen that the maximum temperature of the fuel is about 1026°C, the inner surface temperature of the fuel is about 794°C, while the maximum temperature of the outer surface of the fuel is about 681°C .

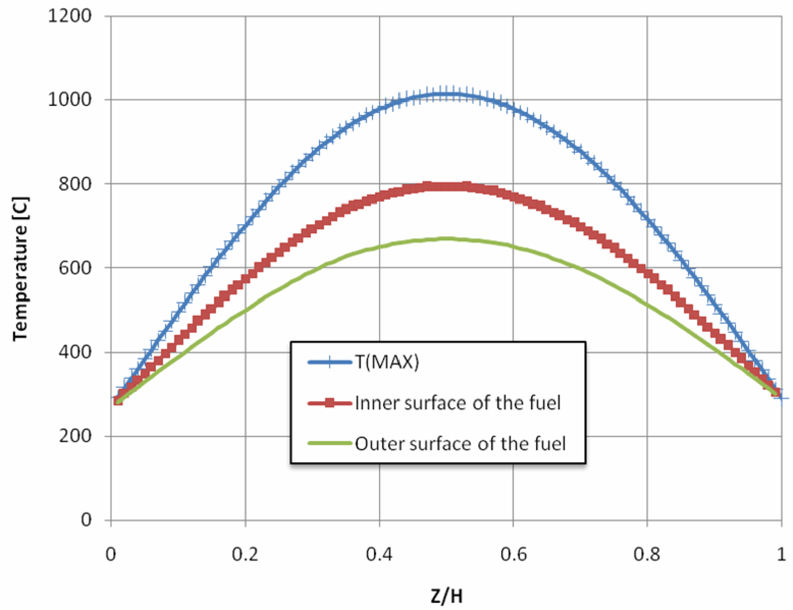

Figure 8: Axial variation of annular fuel temperature.

Axial Variation of Heat Transfer Coefficient

Figure (9) shows the axial variation of heat transfer coefficient along the coolant channel for both the solid and the annular fuel geometries. It can be noted from the figure that the heat transfer coefficient is nearly constant in the non-boiling height of the channel, as expected in single phase flow, and that the non-boiling height is larger for the case of annular channel. In the boiling height nucleate boiling heat transfer process becomes more efficient and it can be seen that the heat transfer coefficient increases more rapidly in the boiling height. Near the end of the channel the rate of heat transfer decreases and exhibits a plateau due to the high void fraction. It can also be observed from the figure that the heat transfer coefficient in solid fuel channel is slightly larger than in the inner side of the annular channel, while the heat transfer coefficient in the outer coolant channel is much larger indicating a more efficient heat transfer process in the larger area outer channel and hence the possibility of operating the reactor at higher power.

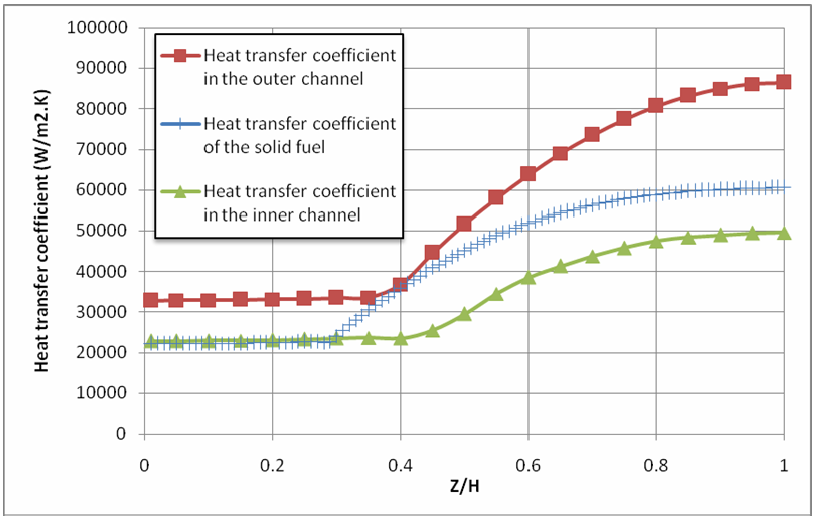

Figure 9: variation of heat transfer coefficient in solid and annular fuel channels.

Axial Variation of Pressure Drop

Figure (10) gives the pressure drop along the flow channel for the solid fuel and the inner and outer sides of the annular channel. It can be seen that in both cases the pressure drop in the non-boiling height is relatively constant and that it increases very rabidly in the boiling height due to the effect of acceleration and the higher friction factor for two-phase flow (two phase friction multiplier). It can also be observed from the figure that the pressure drop in the solid fuel channel lies between the values of inner and outer sides of the annular channel.

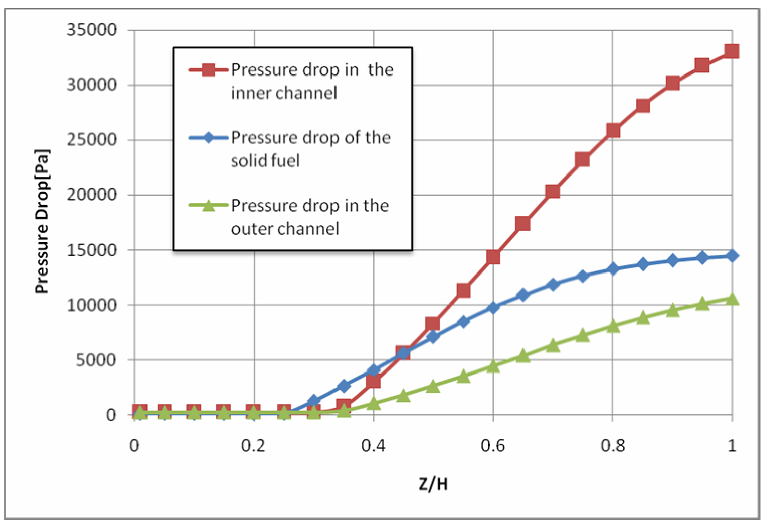

Figure 10: Axial variation of pressure drop in solid and annular channels.

Actual and Critical Heat Fluxes:

Figure (11) illustrate the distribution of the heat flux and the critical heat flux along the annular channel, as shown the range between the flux and the critical heat flux in case of annular design is larger than the range in case of solid design, this gives better margin for safety.

Departure from Nucleate Boiling Ratio, DNBR

Figure (12) indicates that the Departure from Nucleate Boiling ratio (DNBR) decreases along the coolant channel reaching a minimum value then increases. in the solid channel the minimum value was about 2, while the minimum value obtained from annular channel was about 3.1, indicating better safety margin for the annular channel this gives more than the required margin for safety as required in BWR channels.

Redial Temperature Distribution Results

Figure (13) compares the radial fuel temperature profiles for the cases of solid and annular fuel. It can be seen from the figure that maximum temperature of the annular fuel in a 5x5 fuel assembly operating under the same conditions is about 700°C lower than that for a typical solid fuel rod in a 8x8 fuel assembly. It is also shown that a substantial increase in core power by 30% is possible while peak fuel temperature remains less than in the reference case.

Summary of Present Results

The results obtained in this study are summarized and compared in Table (2) which also shows typical values for the traditional sold fuel obtained under the same operating parameters.

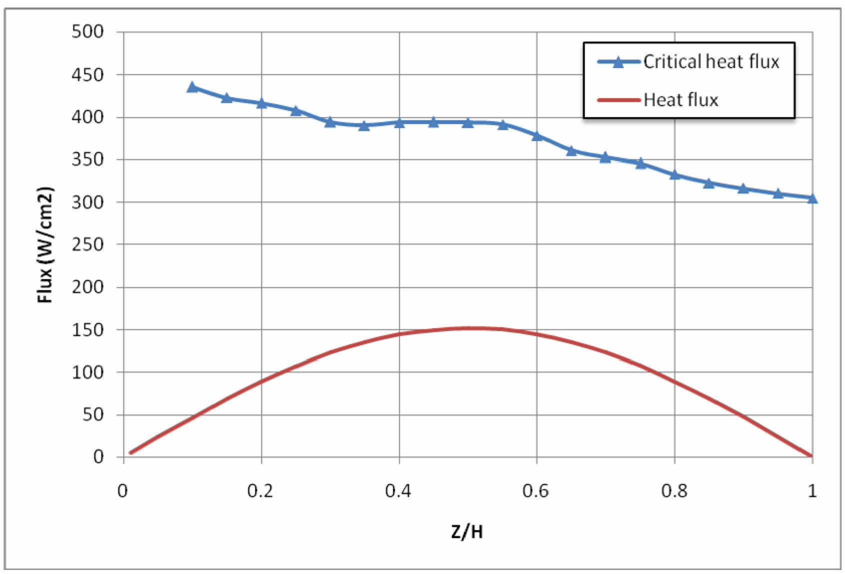

Figure 11: Axial variation of flux and critical heat flux for the annular fuel element.

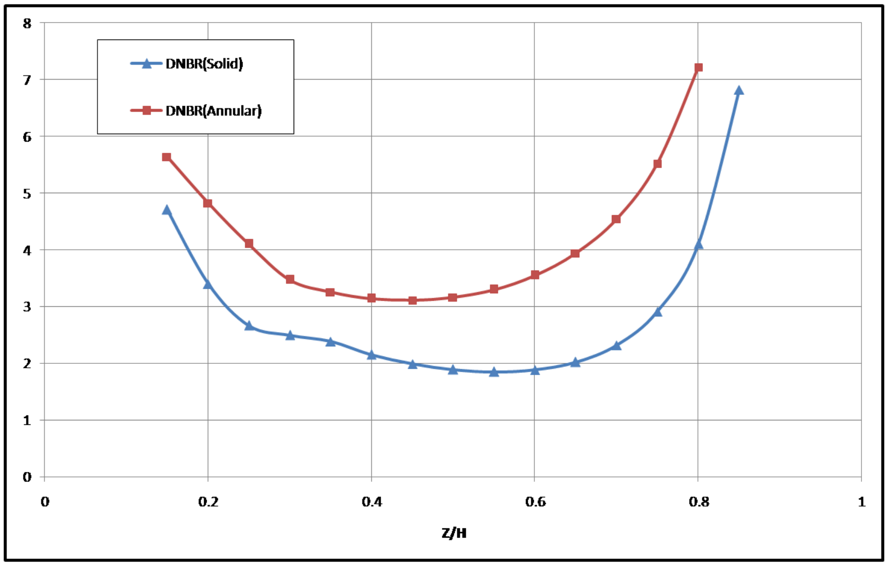

Figure 12: Axial variation of DNBR for solid and annular fuel elements.

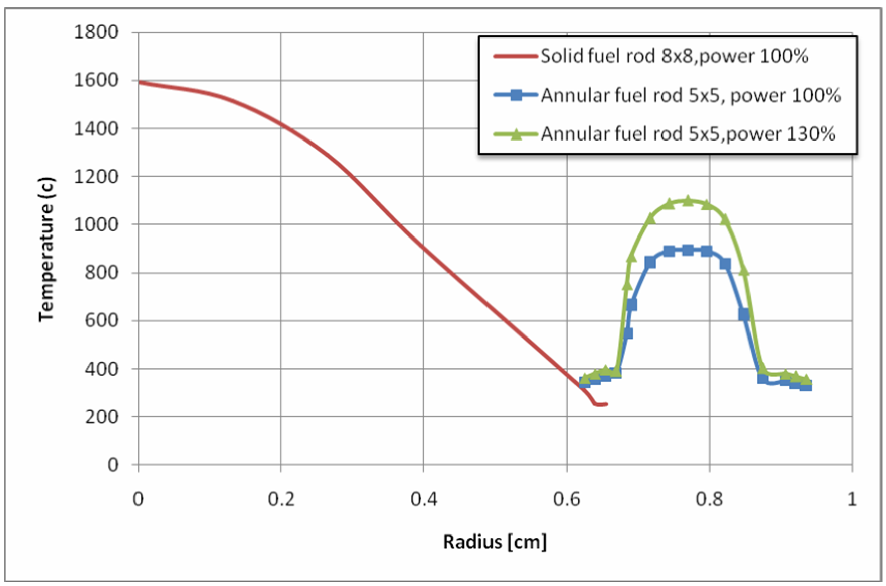

Figure 13: Comparison of radial temperature profiles for solid and annular fuels at channel mid-plane (at Z=H/2).

Table 2: Summary of results for solid and annular fuel designs

Parameter	**Solid fuel (100% power)**	**Annular fuel (100% power)**	**Annular fuel (130% power)**
Peak fuel temperature [^{o}C]	1800	1026	1234
Peak inner cladding temperature [^{o}C]	-	331	378
Peak outer cladding temperature [^{o}C]	-	322	362
Peak inner gap temperature [^{o}C]	-	340	397
Peak outer gap temperature [^{o}C]		332	374
MDNBR	2.0	3.1	2.6

CONCLUSIONS

The results obtained in the present study confirm that it is feasible to operate current BWRs using the new annular fuel design concept suggested by Kazimi and his team with improved safety margins compared to operating with the old solid fuel design, in particular:

- Operating the reactor at 100% power the peak temperature of the solid fuel was 1800^{o}C and the DNBR was 2, while the peak temperature of the annular fuel was 1026 ^{o}C and the DNBR was 3.1.
- With the annular fuel design the reactor can be operated at 130% power with reasonable safety margins. For such case the maximum fuel temperature was about 1234 ^{o}C and the minimum value of DNBR was 2.6.
- The new annular fuel design concept is a very promising option for future nuclear power reactor.

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Nomenclature

k_f : thermal conductivity of the fuel.
k_c : thermal conductivity of the cladding.
k_g : thermal conductivity of the gap.
q''' : volumetric thermal source strength.
r_{fi} : inner radius of the fuel.
r_{fo} : outer radius of the fuel.
r_{cii} : inner radius of inner cladding.
r_{cio} : outer radius of inner cladding
r_{coi} : inner radius of outer cladding.
r_{coo} : outer radius of outer cladding.
T_{fi} : Temperature at the inner surface of the fuel.
T_{fo} : Temperature at the outer surface of the fuel.
T_{max} : Fuel maximum temperature.
T_{gi} : Temperature at the inner gap.
T_{go} : Temperature at the outer gap.
T_{si} : Temperature at the inner surface of the cladding.
T_{so} : Temperature at the outer surface of the cladding.