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A THERMAL HYDRAULIC MODEL FOR POWER UPGRADING OF WWR-S RESEARCH REACTOR

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ABSTRACT

This study presents a developed steady-state thermal-hydraulic model (called the SSTH-RR10 model) for upgrading the WWR-S reactor, from its basic power of 2 MWth to a level of 10 MWth, using two different types of fuels. The SSTH-RR10 model is capable to calculate thermal-hydraulic parameters. Two different fuel types were investigated: the reference fuel EK-10 rod type, and the MTR plate type. For each fuel type, the central fuel, clad, and coolant temperatures profiles for average and hot channels were predicted in the axial direction. Power distributions and pressure gradients were predicted as well. Moreover, the program calculates the safety limits and margins against the critical phenomena encountered such as the Onset of Nucleate Boiling (ONB), Departure from Nucleate Boiling (DNB), and the Onset of Flow Instability (OFI). Results of the SSTH-RR10 program for the benchmark of 2 and 10 MWth are verified by comparing it with IAEA published results, and those published for other programs such as PARET code, and very good agreement is found. The safety margins against ONB and DNB were evaluated in which the minimum DNB ratio was found to be about 3.698, which gives a sufficient margin against the DNB. The present work gives confidence in the model results and applications.

Keywords: Steady state, Thermal hydraulic, WWR-S research reactor, SSTH-RR10 model, Power upgrading.

نموذج هيدروحراري لرفع قدرة مفاعل بحثي من النوع WWR-S

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الملخص:

تقدم هذه الدراسة نموذجا هيدروحراريا مطور والذى تم تسميته (SSTH-RR10) لرفع قدرة مفاعل من النوع الذي يستخدم فيه الماء كمبرد ومهدئ (WWR-S) من قدرتة الأساسية المقدرة ب 2 ميجاوات إلى قدرة 10 ميجاوات، باستخدام نوعين مختلفين من الوقود .النموذج الهيدروحراري SSTH-RR10 قادر على حساب البراميترات الهيدروحرارية. تمت الدراسة علي نوعين مختلفين من الوقود :الوقود المرجعي من نوع القضيب (EK-10 rod type) و الاخر من نوع الوقود اللوحي (MTR plate type). لكل نوع من أنواع الوقود ، تم التنبؤ وحساب توزيع درجات الحرارة علي امتداد محور الوقود و لكل من الغطاء الخارجي للوقود وماء التبريد الموجود داخل القنوات سواء كانت القنوات المتوسطة أوالساخنة في الاتجاه المحوري. كذلك تم التنبؤ بتوزيع القدرة الحرارية و الضغط . علاوة على ذلك ، يحسب البرنامج حدود الأمان ضد الظواهر الحرجة التي تمت مواجهتها مثل نقطة بداية غليان النواة (ONB)، والخروج من الغليان (DNB)، وبدء عدم استقرار التدفق(OFI). كذلك وقد تم التحقق من النتائج التي تم الحصول عليها باستخدام البرنامج المستخدم في الحل SSTH-RR10 عند القدرة الاساسية 2 ميجاوات و بعد رفع القدرة ل 10 ميجاوات و ذلك من خلال مقارنة النتائج التي تم حسابها مع النتائج المنشورة للوكالة الدولية للطاقة الذرية وتلك المنشورة لبرامج أخرى مثل PARET Code، وقد وجد توافق جيد في النتائج. تم تقييم حدود الأمان ضد ONB و DNB حيث وجد أن الحد الأدنى لنسبة DNB بلغ حوالي 3.698 ، مما يعطي هامشًا كافيًا للأمان من DNB. وقد حقق البحث الحالي اسهاما عاليا في نتائج النموذج الذى تم استنباطه.

1. Introduction

Nuclear Research Reactors (RRs) have played an important role in the development of nuclear science and technology since the first artificial, self-sustaining, nuclear chain reaction was initiated. In pool type RRs, the core is a cluster of fuel elements located in a large pool of water. There are control rods and empty channels for experimental materials and probes among the fuel elements. Each element comprises several curved aluminum clad fuel plates in a vertical box. Light water doubles as moderator and coolant for the reactor [1].

WWR-S research reactor is a Tank-in-Pool one, which is cooled and moderated by light water. It is a soviet–designed reactor with an original thermal power of 2 MW_{th}. Many such RRs were built and brought into operation during the 60s. Since then, most countries already upgraded their original WWR-S reactors, and some others did not do yet [2].

Egypt is one of the countries that still has its own WWR-S research reactor with its basic designed thermal power of 2 MW_{th}; the power of the reactor is not upgraded yet. After more than 50 years of operation of this research reactor, the power upgrading is one of the most considered priority options to extend and increase its utilization and related applications. It is a cylindrical tank-in-pool type RR, with a maximum thermal neutron flux of 2×10^13 n/cm^{2}s. The reactor uses 10% enriched UO_2 fuel rods enclosed in aluminum clad (EK-10 type). Demineralized light water is used to moderate, cool, and shield the reactor. The reactor core is placed 5 m below the surface of the reactor tank; the tank diameter is 2.3 m, and its height is 5.7 m. The layout of the core cooling system is shown in Fig. 1, in which cooling water is driven downward with a nominal flow rate of 980 m^3/h, and core inlet and outlet temperatures of 34 and 36 ˚C, respectively [3,4].

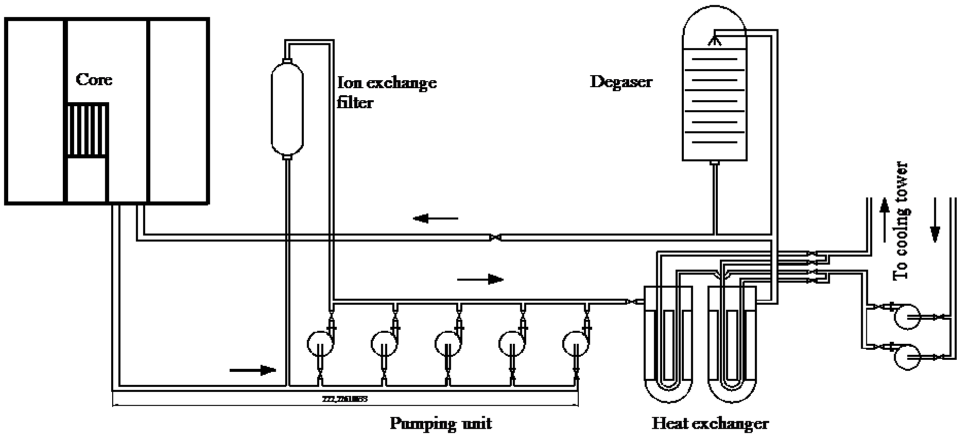


Fig. 1 Layout of WWR-S Core Cooling System [3].

Thermal hydraulic (TH) analysis of the original WWR-S (2 MW_{th}, EK-10 fuel) had been fully studied and covered during its long operational time [5-9]. A steady state thermal-hydraulic model TPRTHA has been developed to simulate the steady-state operation in such type of research reactor. The model was used to simulate the 2 MW_{th} research reactors with downward flow direction and EK-10 fuel bundles. The best-estimate thermal-hydraulic safety margins were determined, and the model results are analyzed and discussed [10].

As for power upgrading, a study was conducted for the Egyptian first Research Reactor (ETRR-1) using the original EK-10 fuel basket by increasing the number of fuel rods in the basket from 16 to 25 rods with the same core configuration [11]. The study concluded that the reactor power can be upgraded safely up to 4 MW with the original 4×4 EK-10 fuel basket and up to 5 MW_{th} with the proposed 5×5 EK-10 fuel basket with the same coolant flow rate of 900 m3/ h [11].

Due to the fact that the original EK-10 fuel type is not commercially manufactured anymore, so, the MTR plate fuel type is used in the present study. Figure 2 depicts two different fuel types, currently in use, which is the MTR plate fuel, in addition to the basic obsolete EK-10 fuel rod type as a reference fuel. The plate type Fuel Element (FE) is usually fabricated by assembling a number of fuel plates together and fastening them to two side plates. Between each two plates there is a channel for cooling purposes [12].

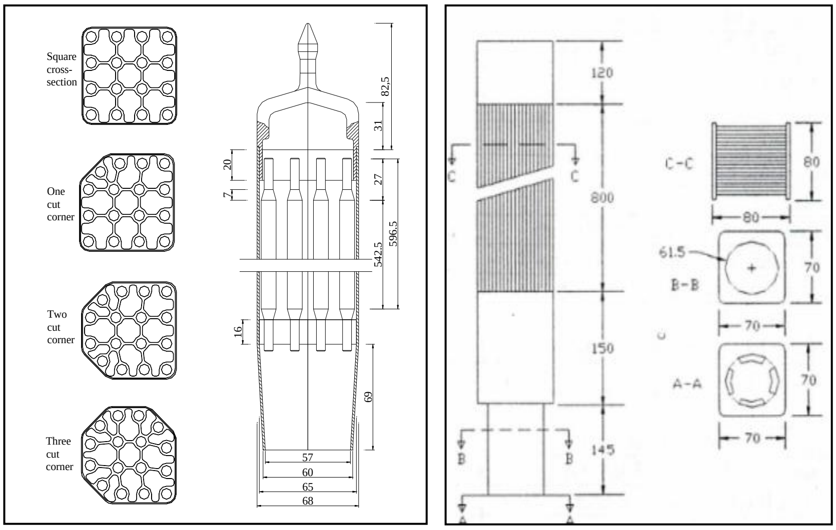


Fig. 2 Different fuel types: a) EK-10 Rod type, b) MTR plate type.

The present research is conducted to overcome the deficiencies declared above. The objective is to study the thermal hydraulic analysis at steady state conditions for possible power upgrading of the WWR-S research reactor from 2 to 10 MW_{th} using two different types of fuels, and to develop a model to satisfy the purpose of the study. The developed model which we called SSTH-RR10 predicts the core thermal hydraulic (TH) parameters for the upgraded research reactor (RR) to 10 MW_{th} as well as the related safety margins of ONBR, DNBR, and OFIR. The calculations were performed for two different fuel types and two different fuel lengths 50cm and 60 cm to compare between them. Due to the fact that the original rod fuel type of the used fuel for that reactor (EK-10 type) is no longer manufactured, the other type of fuel with different active fuel lengths of 50 and 60cm were used and fully studied. From the TH analysis point of view and based on the obtained results, all the proposed upgrading cases are applicable for use except for the case of the MTR plate type with active length of 50 cm.

There are many significant applied benefits relevant to the present work. In addition to aging and degradation problems of the WWR-S research reactor and the unavailability of its original nuclear fuel that prompt the necessity of power upgrading of such reactors, there are also many scientific, research, and commercial benefits that could be gained from such upgrading. By increasing the reactor power, the neutron flux is increased to a level that enhances the productivity of the radioisotopes needed for medical and industrial applications specially those which need higher neutron fluxes. There are wide ranges of research and utilization activates in the fields of materials science, condensed matter, nuclear safety, shielding.

2. Structure and methodology of calculations of the steady state model

For upgrading the reactor power level from 2 to 10 MW_{th}, a complete mathematical model was created to calculate thermal hydraulic parameters and related safety margins at steady state operation. As three different types of fuels with different shapes are used, i.e., rod, and plate types, the main model comprises two different sub models for each fuel type. The model can predict and calculate the distribution of fuel, clad and coolant temperatures as well as heat fluxes for both the average and hot channels. The hot channel represents the hottest one in the core and the average channel represents all other core channels. Best estimation for the related safety margins could be predicted as well.

The main assumptions of the model are:

1. Steady state single-phase one-dimensional heat transfer, i.e., in the radial direction from fuel zone to clad zone and then to coolant.
2. Nodal calculations in the axial direction are up to 50 nodes.
3. The model focuses on TH calculations rather than on neutronics ones in which the predefined neutronic values for each fuel type and each core configuration are used when needed.
4. Constant thermal properties of clad and fuel materials i.e., thermal conductivity and density.

The SSTH-RR10 model is constructed and built using the Engineering Equation Solver (EES) computer program software package to work on PC computers. The main advantage of EES is that it has a large and accurate library of thermal properties for a wide range of fluids as a function of operating pressures and temperatures that could be used during calculations.

Table 1 presents fuel and core technical design data for both basic or reference design and the studied new upgrading design. The basic design data for the two reference reactors are used for the purpose of model validation. The data for the two reference reactors are used as input values to the developed model EES-THRR-10 and its related TH output results are used to validate the model by comparing these values with the well-known TH values for reference reactors.

Table 1: Reactor core design parameters.

Fuel Type	Basic Design		Upgrading Design	
	EK-10	MTR	EK-10 50-60	MTR-50-60
Flow direction	Downward	Upward	Downward	Downward
Design thermal power, MW_{th}	2	22	10	10
Water temperature at core inlet, °C	34	40	35	35
Core inlet pressure, bar	1.37	2.7	1.37	1.37
Nominal core flow, m^3/h	860	1900	1000	1000
Number of fuel assemblies	41	29	41	29
Number of fuel rods/plates/tubes per fuel assembly	16 Rods	19 Plates	16 Rods	17 Plates
Reactor equivalent radius, cm	24.3	24.3	24.3	24.3
Enrichment, %	10	19.7	10	19.7
Radial peaking factor	1.54	2.22	1.54	2.22
Axial peaking factor	1.3	1.35	1.3	1.35
Active length, cm	50	80	50/60	50/60
Core lattice "Grid" size (x*y) mm	71.5*71.5	80*80	71.5*71.5	71.5*71.5
Extrapolated addendum, mm	70	70	70	70
Coolant channel thickness, mm	1.5	2.7	1.5	2.7
Clad thickness, mm	1.5	0.4	1.5	0.4
Clad thermal conductivity, W/m°C	170	180	170	180
Clad specific heat, J/kg °C	900	892	900	892
Clad density, kg/m3	2700	2700	2700	2700
Fuel meat thickness, mm	Φ 7	0.7	Φ 7	0.7
Fuel thermal conductivity, W/m°C	25	53.6	25	53.6
Fuel specific heat, J/kg °C	234.5	728	234.5	728
Fuel density, kg/m^3	5775	4450	5775	4450

3. Mathematical equations for studied fuel types

3.1 EK-10 rod fuel type

As mentioned above, this type of fuel is the basic design fuel used in the reference WWR-S reactor in which the reference core configuration is utilized for modeling that type. Based on the distance of each fuel bundle from the center of the core, six different groups of fuel bundles are considered in this model [3,10]. The power released from the fuel assemblies of group g, P(g) at an equivalent radial distance, r(g) from the core center is given by

$$P(g) = \frac{P}{N_{bs}} \times PPF_R \times J_o\left(\frac{2.405 * r(g)}{R+\delta}\right) \quad (1)$$

Where: 'g' is the group number of fuel assembly i.e., 1- 6, P is the total thermal power level of the reactor, N_{bs} is the number fuel assemblies at the startup period, PPF_R is the radial power peaking factor, J_o is the Bessel function of zero order, R is the equivalent radius of the reactor, and δ is the value of extrapolated addendum [13].

The axial heat flux distribution along the core is a cosine shape as follows:

$$\phi(z) = \phi_0 \cos\left(\frac{\pi(z-H/2)}{H_p}\right) \tag{2}$$

For average and hottest rod heat fluxes of an assembly group 'g'

$$\phi_0|_a = \frac{P(g)}{2N_r d H_p \sin\left(\frac{\pi H}{2H_p}\right)} \tag{3}$$

$$\phi_0|_h = PPF_T \times P(g)/N_r \pi dH \tag{4}$$

Where: Nr is the number of fuel rods per assembly, Hp= H+2*e where H is the active length, Hp is the extrapolated length, e is the extrapolated distance, PPF_T=PPF_A*PPF_R, where, PPF_A, PPF_R and PPF_T are the axial, radial, and total peaking factors, respectively.$\phi(z)$ is the heat flux at specified node in axial direction (z), ϕ_0 is the maximum heat flux, and $\phi_0|_a$, and $\phi_0|_h$ are the maximum heat flux at average and hot channel, respectively.

The generated heat is absorbed and removed by the coolant. The coolant moves along the fuel length and hence, its temperature is gradually increased. To calculate the bulk coolant temperature at an elevation z, from the channel inlet $T_{co}(z)$, apply an energy balance up to the height, z from channel inlet gives

$$T_{co}(z) = T_{co_{in}} + \frac{N_r dH_p \phi_0|_a}{GAC_p}\left[\sin\frac{\pi(z-H/2)}{H_p} + \sin\frac{\pi H}{2H_p}\right] \tag{5}$$

The distribution of clad surface and central fuel temperatures along the fuel length are given by:

$$T_{cl,o}(z) = T_{co}(z) + \frac{\phi(z)}{h(z)} \tag{6}$$

$$T_{cl,i}(z) = T_{cl,o}(z) + \frac{\phi(z)}{k_c} R_c \ln\left(\frac{R_c}{R_f}\right) \tag{7}$$

$$T_{c,f}(z) = T_{cl,i}(z) + \frac{q(R_f^2)}{4k_f} \tag{8}$$

where: $T_{cl,o}(z)$, $T_{cl,i}(z)$, and $T_{c,f}(z)$ are the outside, inside, and central fuel temperatures. The Rf, and Rc are the fuel and clad radii. The k_c and k_f are the clad and fuel thermal conductivities.

3.2 MTR plate fuel type

Following the same procedure, the coolant channel is divided into a specified axial region while the fuel plate is divided into a specified radial node, then a nodal thermal hydraulic calculation for both average and hot channels is performed with cosine heat generation flux.

$$q(z) = q_m \cdot \cos\frac{\pi z}{2z_e} \tag{9}$$

Where: q_m is the maximum volumetric heat generation, z_e is the extrapolated distance (distance from the center of the axial core to the point where neutron flux is nil).

Fig. 4 illustrates a single channel for proposed MTR plate fuel type.

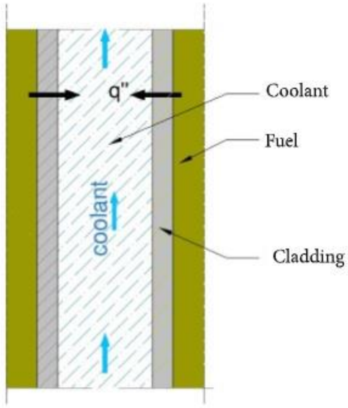


Fig. 4 Single channel of MTR plate fuel geometry [14].

In order to determine the maximum heat generation, it is necessary to calculate the average volumetric heat generation in the core (q). In this case, it is known that the average heat flux density is estimated by

$$\phi = \frac{P}{A_f \cdot N_{fc} \cdot N_{fp}} \tag{10}$$

and the average volumetric heat generation is calculated by

$$q = \frac{\phi}{x_f} \tag{11}$$

Where: q is the volumetric heat generation, P is the average thermal reactor power, A_f is the heating surface of each fuel plate, N_{fe} is the number of fuel elements in the reactor, N_{fp} is the number of fuel plates per fuel element, φ is the local heat flux based on cosine shape, and x_f is half the fuel plate thickness. The value of maximum heat generation will be, q_{max}=q*PPF, where PPF is the Power Peaking Factor. The heat flux distribution along the channel is evaluated by

$$\varphi(z) = \varphi_{max} \cdot cos\frac{\pi.z}{2.z_c} \tag{12}$$

$$\varphi_{max} = \varphi.PPF \tag{13}$$

Where: φ is the average heat flux, φ_{max} is the maximum admissible heat flux, and z_c is the active core height in which coolant temperature can be determined by

$$T_{co}(z) = T_{co,in} + \frac{q_{max} \cdot A_{th}}{w \cdot c_p} \cdot \frac{2 \cdot z_c}{\pi}\left[sin\frac{\pi \cdot z}{2 \cdot z_c}\right]_{-z_c}^{z_e} \tag{14}$$

Where: z_e=z_c+e, e is the extrapolated length, A_{th} is the cross sectional area of the fuel, q_{max} is the maximum volumetric heat generation (at the core center), w is the coolant mass flow rate, $T_{co,in}$ is the coolant inlet temperature, and z is the direction of fuel element height.

The outer and inner clad-surface temperatures are predicted from the following heat conduction equations

$$T_{c_{l,o}}(z) = T_{co}(z) + \frac{\varphi(z)}{h_{conv}} \quad (15)$$

$$T_{cl,o} - T_{cl,i} = -\frac{\varphi(z)}{k_{cl}} \cdot x_{cl} \quad (16)$$

Where: h_{conv} is the convective heat transfer coefficient, $T_{cl,o}$ (z) is the outer clad-surface temperature, T_{co} is the local temperature of coolant, K_{cl} is the clad thermal conductivity, and x_{cl} is the clad thickness.

Considering only one half of the plane for the heat flow in the + x direction and a thin layer of thickness, $2x_f$ at distance, x_f from the mid-plane, a heat balance for the case of steady state heat transfer can be derived. The central fuel temperature distribution is calculated as

$$T_{c,f}(z) = \frac{q(z)}{2.k_f} \cdot x_f^2 + T_{c1,i}(z) \quad (17)$$

4. Safety margins

4.1 Onset of nucleate boiling

Boiling is initiated when the coolant channel wall temperature is equal to the onset of nucleate boiling temperature, T_{ONB}, where:

$$T_{ONB} = T_{sat} + (\Delta T_{sat})_{ONB} \quad (18)$$

Where: $(\Delta T_{sat})_{ONB}$ is given by Bergles and Rohsenow correlation [15] which is valid for water only over the pressure range 1-138 bar, and is given as

$$(\Delta T_{sat})_{ONB} = 0.556 * \left[\frac{\phi_{ONB}}{1082\, p^{1.156}}\right]^{0.463\, p^{0.0234}} \quad (19)$$

Where: P is the local pressure in bar and Φ_{ONB} is in W/m^2. The heat flux leading to ONB is found out by iteration. Critical Heat Flux was calculated and used by Bernath correlation [16] as follows

$$\phi_{cri} = h_{cri}(T_w - T_b) \quad (20)$$

4.2 Onset of flow instability (OFI)

Flow instabilities must be avoided in heated channels as flow oscillations affect the local heat transfer characteristics and may induce a premature burnout. The burnout heat flux occurring under unstable flow conditions was well below q''_{DNB} for the same channel under stable flow conditions. For practical purposes in plate-type fuel design, q''_{DNB} that leads to the onset of flow instability is more limiting than the heat flux for stable burnout. The most common flow instabilities encountered in heated channels with forced convection are the flow excursion and density wave oscillation types. In low-pressure sub cooled boiling systems, the criterion for the onset of flow instability has been obtained for rectangular channels by [17] as

$$R = \frac{T_{out} - T_{in}}{T_{sat} - T_{in}} = \frac{1}{1 + \eta \frac{D_{he}}{L}} \tag{21}$$

Where: the bubble detachment parameter η, is assumed to be 25, L is the fuel heated length, and D_{he} is the heated equivalent diameter of the channel given by

$$D_{he} = \frac{4 * \text{Channel flow area}}{\text{Channel heated perimeter}} \tag{22}$$

4.3 Departure from nucleate Boiling (DNB)

For reactor design purposes, an acceptable prediction method for burnout heat flux is needed since Departure from Nucleate Boiling (DNB) is potentially a limiting design constraint. The heat flux leading to this situation is named DNB flux, q''_{DNB}.

In order to maintain the cladding integrity at any point in the reactor core due to the onset of nucleate boiling, the reactor must be designed so that heat flux, q'' is always below q''_{DNB}. For this purpose, it is convenient to define departure from nucleate boiling ratio (DNBR) as

$$DNBR = \frac{q''_{DNB}}{q''_{actual}} \tag{23}$$

Where: q''_{actual} is the actual surface heat flux at the same position of the channel. For research reactors in steady state conditions, a literature survey of DNB correlations applicable to low pressure and low temperature narrow rectangular channel concluded that Mirshak and Labuntsov correlations are recommended for assessing the DNBR [18]. The Mirshak correlations are given as:

$$q''_{DNB} = 151 \times (1 + 0.1198 v_{ch})(1 + 0.00914 \Delta T_{sub})(1 + 0.19P) \tag{24}$$

$$q''_{DNB} = 145.4 \vartheta(P) \left(1 + \frac{2.5 v_{ch}^2}{\vartheta(P)}\right)^{1/4} \left(1 + \frac{15.1 c_p \Delta T_{sub}}{\lambda \sqrt{P}}\right) \tag{25}$$

$$\vartheta(P) = 0.99531 P^{1/3} (1 - P/P_{crit})^{4/3} \tag{26}$$

Where: P_{crit} is the water critical pressure, and l is the water latent heat. The water exit sub-cooling (ΔT_{sub}) is given by

$$\Delta T_{sub} = T_{sat} - T_{in} - \frac{20 \times l_f w_p q''_{DNB}}{\rho C_p t_w w_{ch} F_A v_{ch}} \tag{27}$$

5. Results and Discussion

From the thermal hydraulic core design point of view, the crucial goal is to preserve the integrity of fuel element to prevent and exclude any radioactive release. The most important constriction is that the core temperatures remain below values inducing the clad rupture as well. The endorsed adopted criteria for the maximum fuel centerline, fuel clad, and core outlet temperatures must remain within the fuel manufacturer and IAEA recommended permissible values [19]. Steady state thermal hydraulic calculations were performed using the present developed SSTH-RR10 for the two types of fuel and the corresponding reference and upgraded reactor powers. For the aim of upgrading, all TH calculations are obtained at nominal downward coolant flow rate of 1000 m3/hr, inlet pressure of 1.5 bar, and inlet coolant temperature of 35°C. The calculations for the studied WWR-S research reactor were carried out based on the axial and radial Power Peaking Factors obtained by Wims-4B and CITATION neutronic codes based on the recent study of Ref. [20]. In the following sections, the results for the reference reactors as well as the upgraded ones are presented and discussed.

5.1 Verification of the present steady state SSTH-RR10 developed code

Here, the data of the two basic design reference reactors shown previously in Table 2 are used as input data to assess and validate the developed model. For the WWR-S with the basic design EK-10 fuel rod**,** the model is used for the steady-state thermal calculations at a nominal reactor power of 2 MW_{th}, and nominal core downward flow rate of 860 m^3/h with inlet coolant temperature of 34 °C. The reactor core is divided into six fueled bundle groups [3,10]. The results for steady state average and hot heat fluxes, coolant, surface clad, and central fuel temperatures for each group are depicted in Figs. 7, 8, 9, and 10, respectively. It can be seen that heat flux distributions for each bundle group depends mainly on its distance from the core center, i.e., the radius of bundle group r(g) in Eq. 6, in which increasing r(g) results in a decrease of the power of bundle as indicated in Fig. 7.

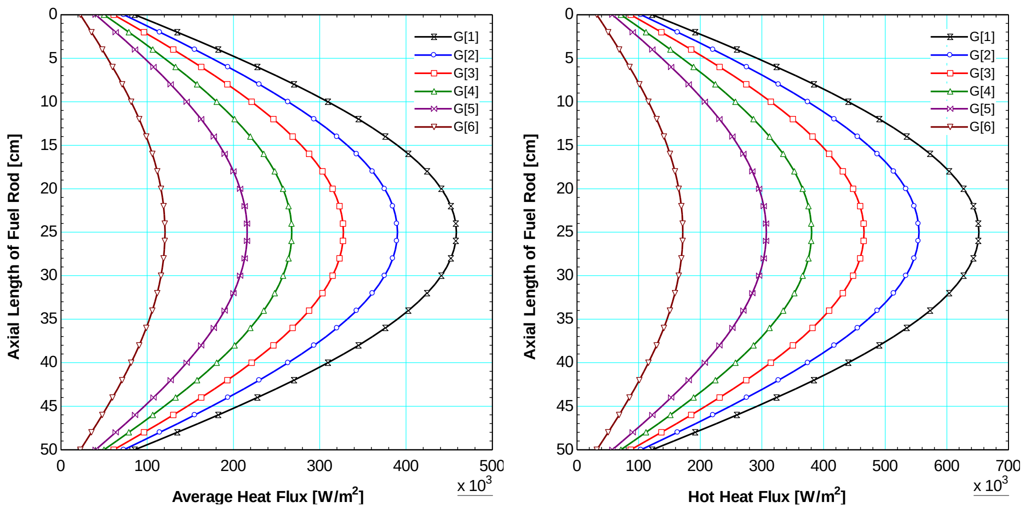


Fig. 7 Heat flux distribution for the average and hottest rod of fuel bundles.

Contrarily, coolant and surface clad temperatures are strongly dependent on coolant flow rate and velocity through each bundle group. Figs. 8 and 9 show that maximum coolant and clad temperatures are obtained to bundle group 4 which is attributed to the lower coolant flow rate values for it compared to the higher values for bundle group 1 which has the highest power value. The temperature profiles at the center of average and hottest fuel rods for the six group bundles are plotted in Fig. 10, where maximum fuel-center temperature values are for higher power fuel bundles.

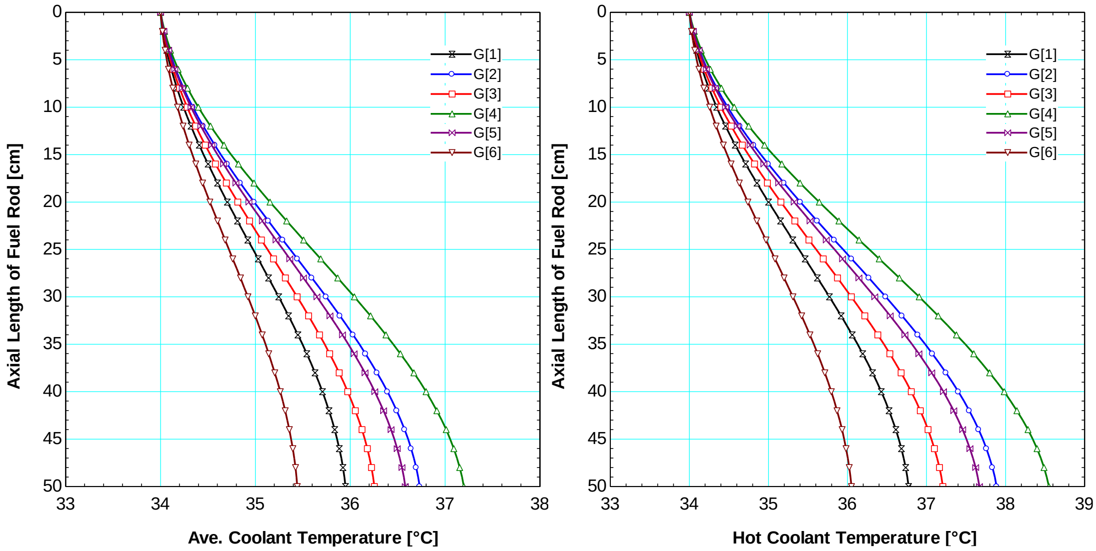


Fig. 8 Coolant temperature distribution for the average and hottest rod of fuel bundles.

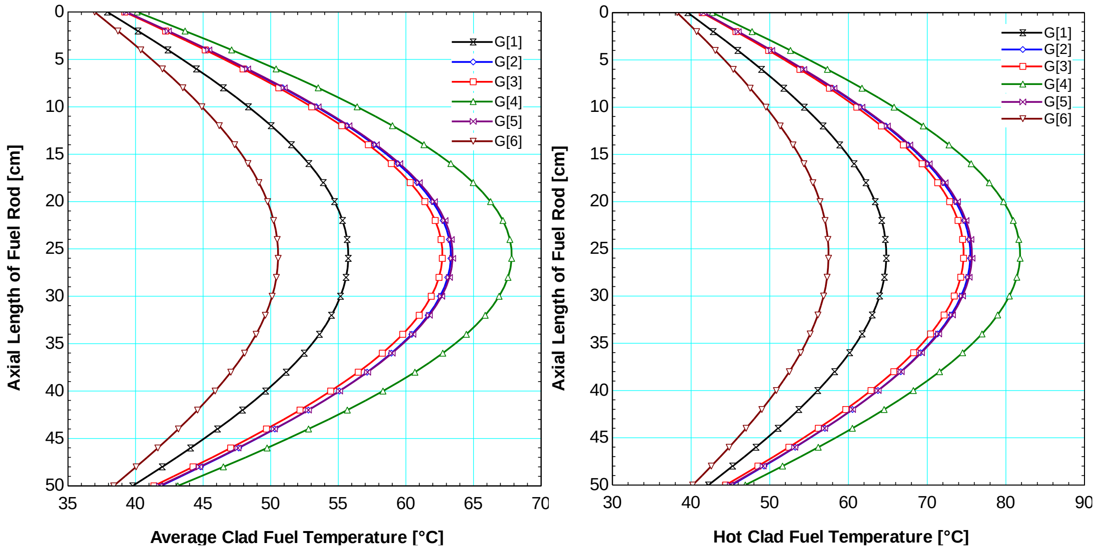


Fig. 9 Clad temperature distribution for the average and hottest rod of fuel bundles.

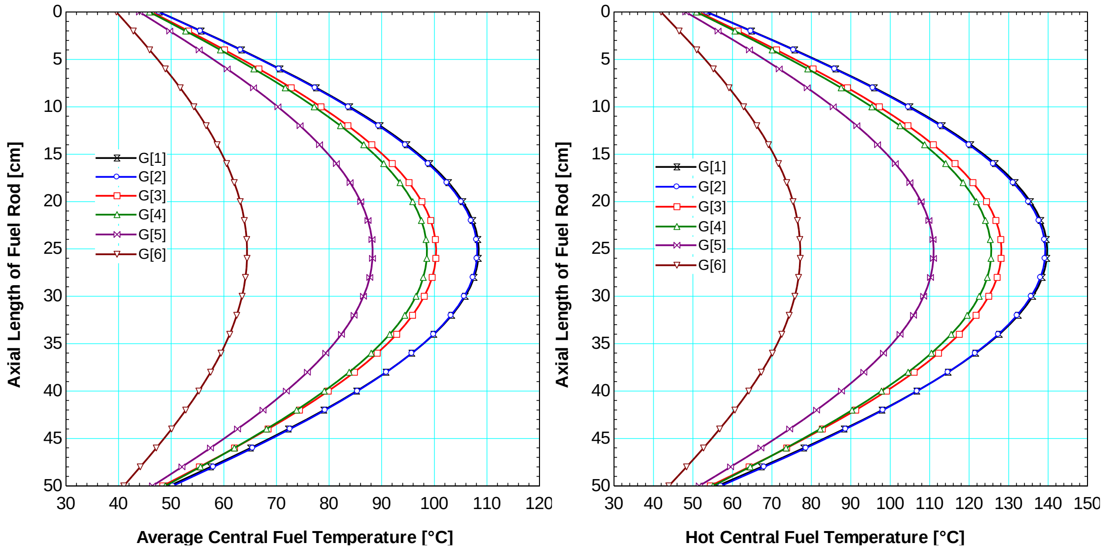


Fig. 10 Central fuel temperature distribution for average and hottest rod of fuel bundles.

For comparison and model validation, Table 2 summarizes the predicted maximum temperature values and the estimated minimum safety margins by the SSTH-RR10 model for steady-state normal operation compared with the reference basic values [3] and TPATHA model values [10]. Very good quantitative agreements are obtained.

Table 2: Comparison of present SSTH-RR10 code with other codes.

Parameter	Code	G 1	G 2	G 3	G 4	G 5	G 6
$T_{co,ave}$ (°C)	SSTH-RR10	35.96	36.73	36.25	37.20	36.56	35.44
	TPTHA [10], SAR [3]	35.96	36.75	36.27	37.22	36.60	35.45
	Error percentage (E %)	0.00	0.05	0.06	0.05	0.11	0.03
$T_{cl\text{-}s,hot}$ (°C)	SSTH-RR10	64.79	75.47	74.63	81.80	75.68	57.46
	TPTHA [10], SAR [3]	64.30	74.80	74.10	81.20	75.10	57.00
	Error percentage (E %)	0.76	0.90	0.72	0.74	0.77	0.81
$T_{f\text{-}c,hot}$ (°C)	SSTH-RR10	139.80	139.30	128.20	125.60	111.00	77.20
	TPTHA [10], SAR [3]	139.30	138.60	127.70	124.90	110.40	76.70
	Error percentage (E %)	0.36	0.51	0.39	0.56	0.54	0.65
ONBR	SSTH-RR10	4.10	2.22	2.23	1.86	1.40	1.81
	TPTHA [10], SAR [3]	4.50	2.73	2.32	1.57	1.45	1.45
	Error percentage (E %)	8.9	7.7	3.9	0.6	3.4	4.8
DNBR	SSTH-RR10	8.65	8.45	9.58	9.58	10.18	10.06
	TPTHA [10], SAR [3]	7.89	8.39	9.70	8.89	9.59	9.60
	Error percentage (E %)	9.6	0.7	1.2	7.8	6.2	4.8

Where, G is group number of bundles, and SAR is Safety Analysis Report.

On the other hand, for the second MTR basic design and reference one in which MTR plate type fuel is used, the present model is used for the steady-state thermal calculations at a nominal reactor power of 22 MW_{th} and nominal core upward flow rate of 1900 m^3/h with inlet coolant temperature of 40 ˚C.

In the same fashion, the predicted TH values of the SSTH-RR10 model are presented and compared with those obtained from the PARET code [12], as illustrated in Figs. 11 and 12. For instance, Fig. 11 shows the predicted values of clad, coolant, and central fuel temperatures for the average and hottest fuel plate compared with those obtained by PARET code. Qualitative comparison of results seen in the figures, show good agreement with an estimated quantitative deviation of less than 4%.

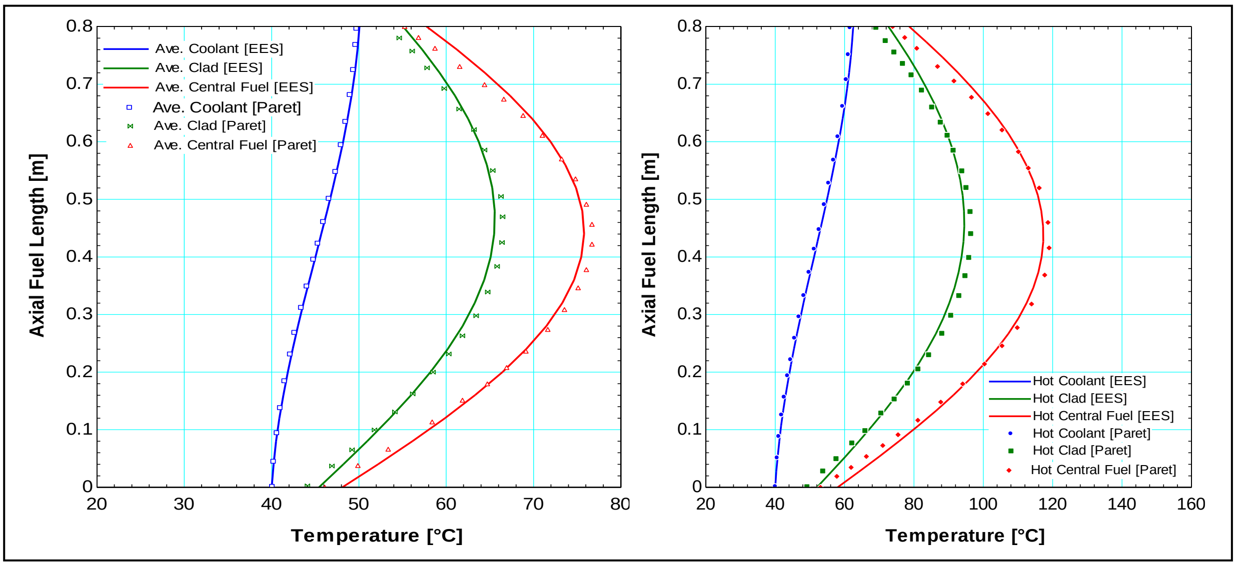


Fig. 11 Comparison between predicted axial average and hot temperatures of MTR basic design reactor at 22 MW_{th}.

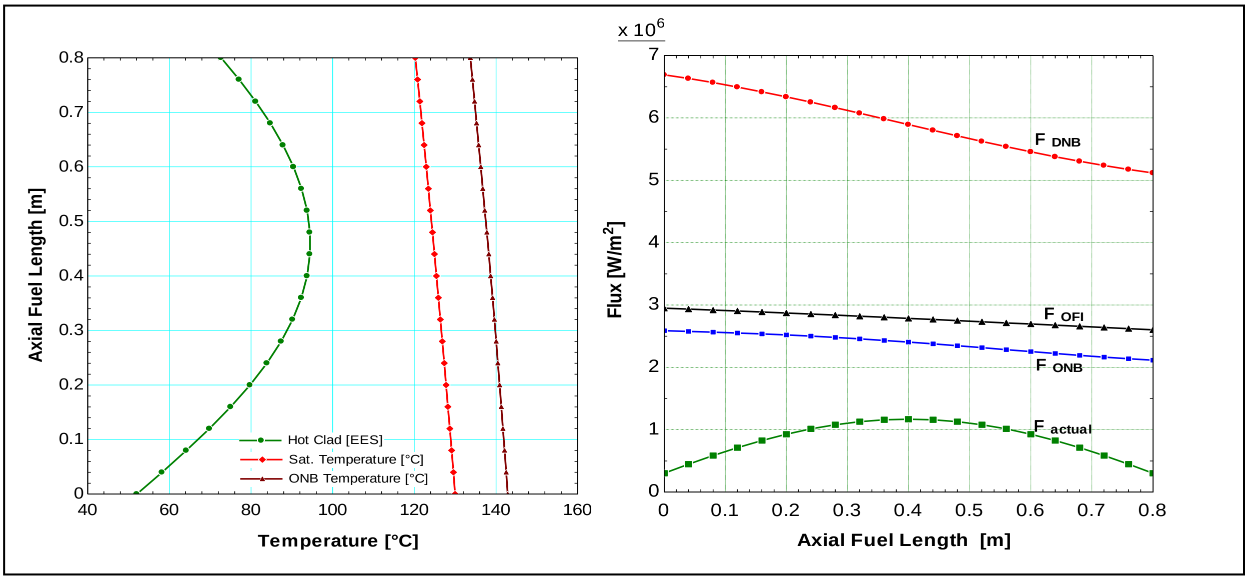


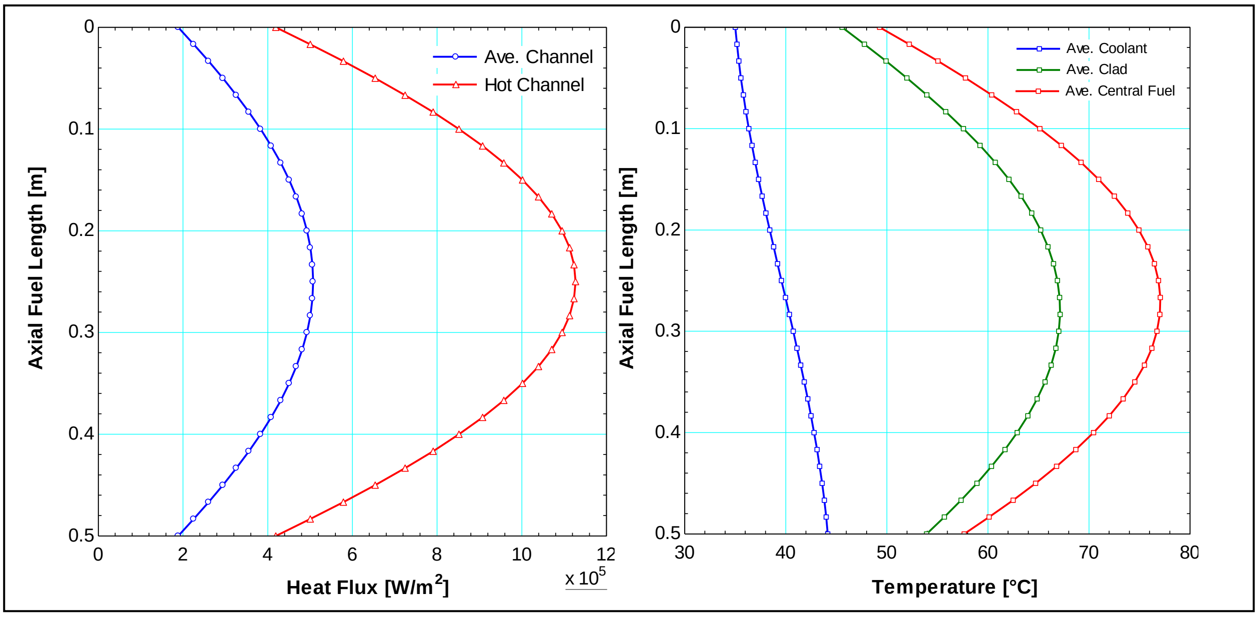
Fig. 12 Predicted axial peak temperatures and fluxes margins MTR basic design reactor at 22 MW_{th}.

5.2 Upgrading results of 10 MW_{th} power level

For upgrading, the study focused on MTR plate fuel. Due to the fact that the original EK-10 fuel type is not commercially manufactured anymore. Technical data of these types (MTR-50 and MTR-60) are summarized in Table 1. Two different fuel plate lengths (50 and 60 cm) for MTR plate fuel types are considered to get a comparative study. For abbreviations, MTR-50 and MTR-60 are MTR plate fuel types with active lengths of 50 cm, and 60 cm respectively. All TH present results are obtained at steady state for all cases at an upgraded power of 10 MW_{th}, downward coolant flow rate of 1000 m^3/h, inlet pressure of 1.37 bar, and inlet coolant temperature of 35 °C. Calculations were carried out based on the axial and radial power peaking factors obtained by Wims-4B and CITATION neutronic codes based on the predefined recent study [20].

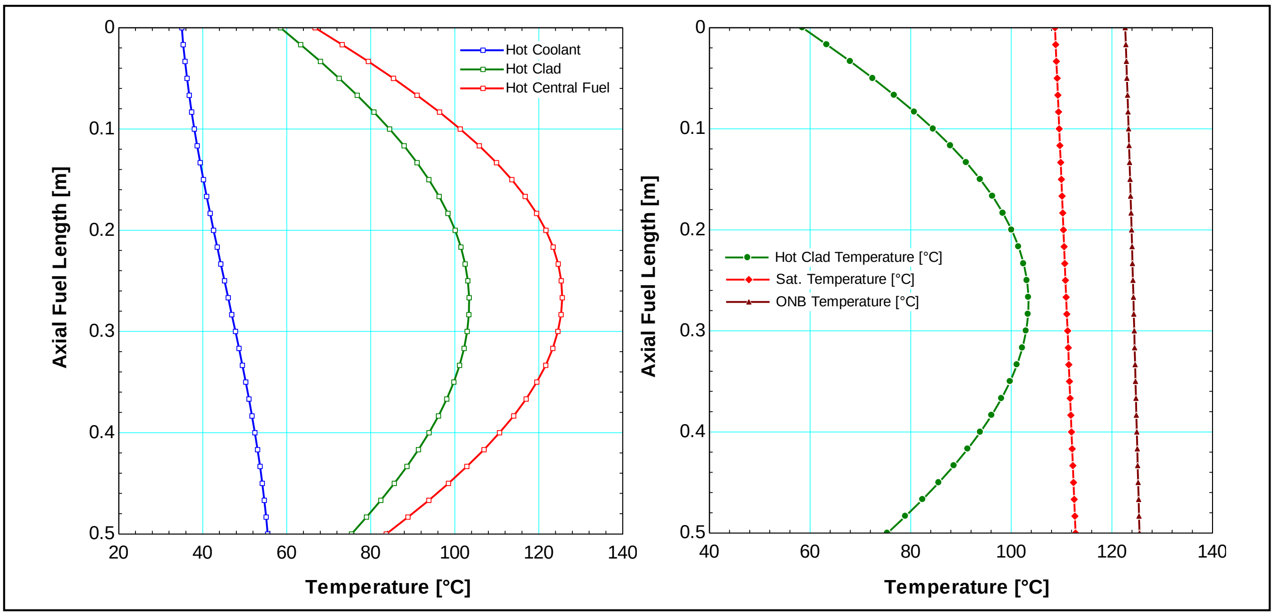
5.2.1 MTR plate fuel type

For MTR plate fuel type, two different upgrading cases were considered and presented in Figs. 13 and 14, for MTR-50, and MTR-60 plate fuels, respectively, for 10 MW_{th} power. As an illustrative case, Fig. 13 gives the predicted TH values distribution for the 10 MW_{th} upgraded WWR-S research reactor using the MTR-50 plate fuel. Fig. 13a shows the distribution of the average and hot channel fluxes with respect to the axial distance from top of the core for the upgraded reactor, in which maximum heat flux for the hot channel is 1123 kW/m^2.



a) Axial heat fluxes of average and hot channels of MTR-50 at 10 MW

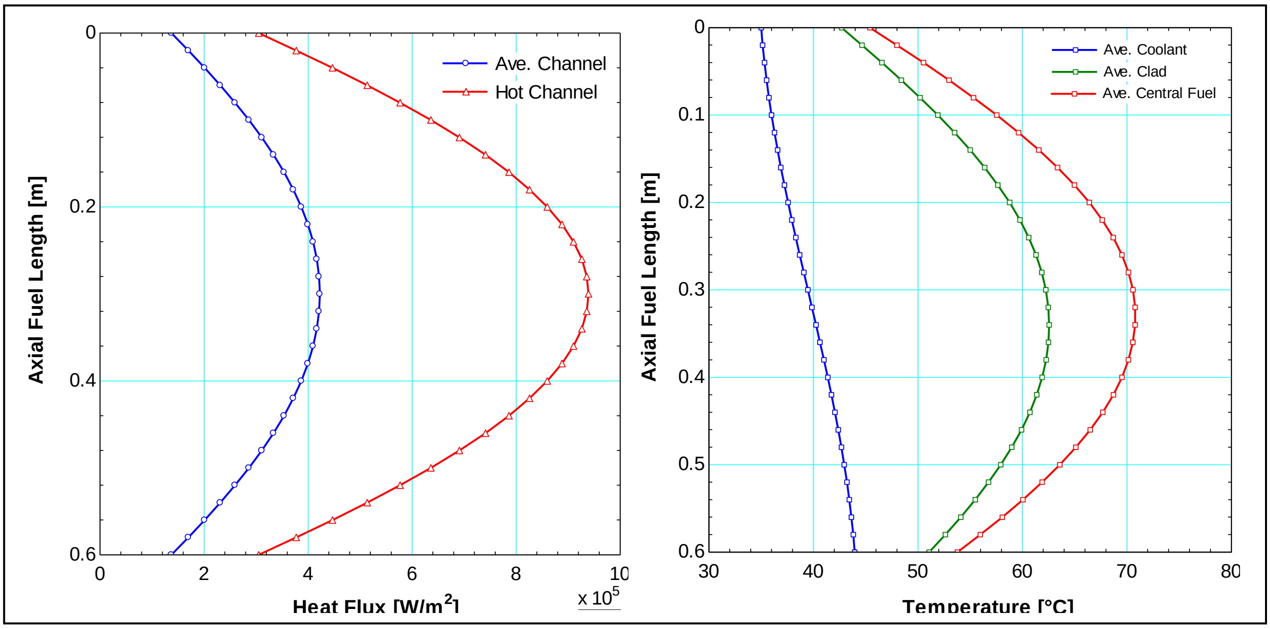
b) Axial average coolant, clad and central fuel temperatures of MTR-50 at 10 MW



c) Axial hot coolant, clad and central fuel temperatures of MTR-50 at 10 MW.

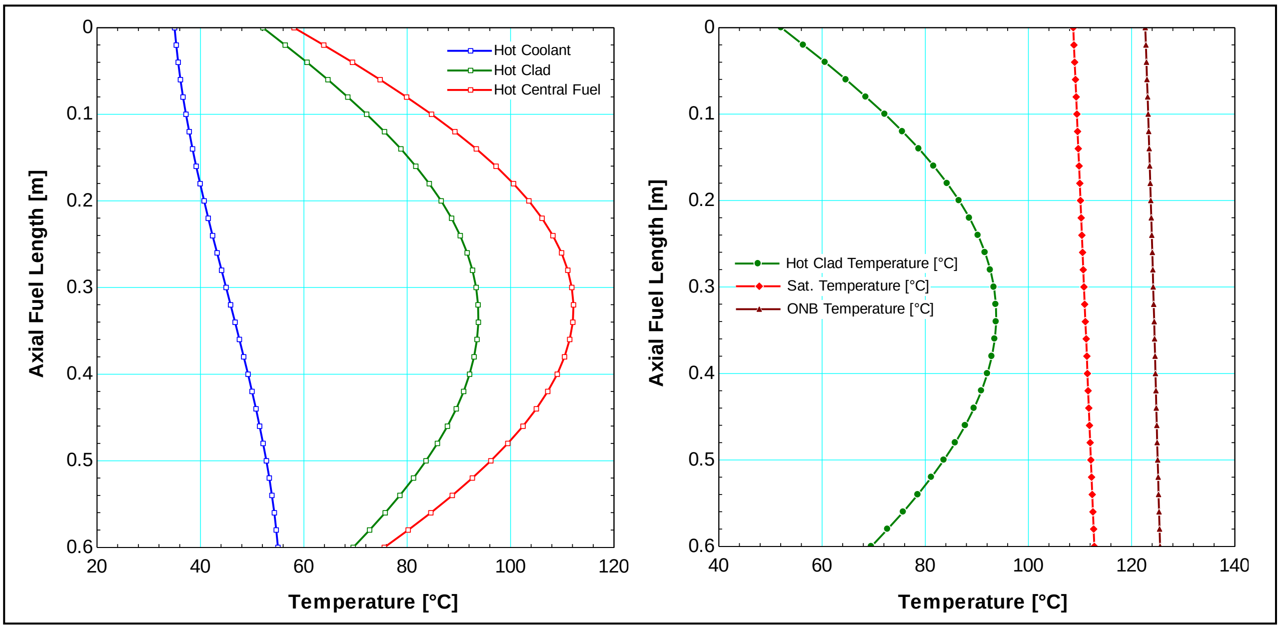
d) Axial peak clad, saturation, ONB temperatures of MTR-50 at 10 MW

Fig. 13 Predicted Steady state TH and safety margins for WWR-S upgrading at 10 MW for MTR-50 plate fuel.



a) Axial heat fluxes of average and hot channels of MTR-60 at 10 MW

b) Axial average coolant, clad and central fuel temperatures of MTR-60 at 10 MW



c) Axial hot coolant, clad and central fuel temperatures of MTR-60 at 10 MW.

d) Axial peak clad, saturation, ONB temperatures of MTR-60 at 10 MW

Fig. 14 Predicted Steady state TH and safety margins for WWR-S upgrading at 10 MW for MTR-60 plate fuel.

The temperature distributions of fuel centerline, cladding surface, and bulk coolant along the hot and average WWR-S fuel coolant channels as a function of axial distance are demonstrated in Figs. 13b and 13c, respectively, for the same case. The maximum clad temperature is about 103.4 °C which is far from saturation and ONB temperatures as indicated in Fig. 13d. The peak clad temperature compared to saturation and ONB temperatures for the MTR-50 plate fuel for upgraded level of 10 MW_{th} WWR-S is presented in Fig. 13d. It is shown from the figure that the maximum clad surface temperature at the steady-state power level of 10 MW_{th} would be 8.7 °C below saturation temperature. As illustrated for all two cases for the MTR plate type, it is noticed that by increasing the length of fuel plate the corresponding fuel clad and coolant temperatures are decreased for the same level of upgraded power. For the case of using the MTR-50, it is found that the value of clad temperature for the hottest plate is nearly 103.4 °C. This value is close to the maximum allowed peak value of 105 °C, therefore the MTR-50 is not recommended for the purpose of upgrading to 10 MW_{th}. The other of MTR-60 could be safely used from TH analyses point of view.

Table 3: Summary of WWR-S upgrading with MTR plate fuel type.

Fuel type and its length	Maximum Temperatures (°C)			Safety Margins		
	Coolant	Surface Clad	Central Fuel	ONBR	OFIR	DNBR
MTR-50	44.15	103.4	125.6	1.286	2.183	3.698
MTR-60	43.95	93.75	112.2	1.549	2.271	4.451
Reference values [12]		**< 105**	**< 150**	**> 1.3**	**> 1.5**	**> 2**

6. Conclusions

The main conclusions from the present research are:

1. A steady state thermal hydraulic analysis has been performed for power upgrading of the WWR-S research reactor from its original power of 2 MW_{th} to a level of 10 MW_{th} using plate type of nuclear fuel.
2. In addition to the MTR plate fuel types with different active lengths of 50, and 60 cm have been considered and studied.
3. A steady state thermal hydraulic model has been developed to compute all the related thermal hydraulic parameters and safety margins for the original 2 MW_{th} and the upgraded 10 MW_{th} power values of the WWR-S reactor core using this type of nuclear fuel.
4. The developed model is called Steady-State Thermal Hydraulic Research Reactor at 10 MW_{th} (SSTH-RR10). It is modeled and constructed using an Engineering Equation Solver (EES) numerical software computer program.
5. The SSTH-RR10 model is capable to predict and calculate with good accuracy all the thermal hydraulic parameters for the defined core configuration for each type of fuel used at a level of 10 MW_{th}. The model is not limited for calculations of the 10 MW_{th} case, but it can be used for any desired power value. The model predicted the axial distribution of central fuel, clad and coolant temperatures in the average and hot channels for each fuel at the considered fuel active heights. Power distributions and pressure gradients were predicted as well. Moreover, the program calculates the safety limits and margins against the critical phenomena encountered such as the Onset of Nucleate Boiling (ONB) not to allow the nucleate boiling anywhere in the core and against the Departure from Nucleate Boiling (DNB), and the Onset of Flow Instability (OFI).

6. The model has been applied for all cases of the simulated upgraded core with EK-10 rod, and MTR fuels at the same coolant flow rate of 1000 m^3/h and inlet temperature of 35 °C. The Results of the model for benchmark of 2 and 10 MW_{th} were verified by comparing it with IAEA published results and with those using the PARET code, and very good agreement was found. This supports confidence in the present model and in its predictions.
7. For all the studied upgrading cases, safety margin results against ONB and DNB gave a minimum DNBR ratio of about 3.69, which gives a sufficient margin against DNB to take place.
8. The maximum clad temperatures remain below the sub cooled boiling condition with sufficient vast margins for all cases except for the case of MTR fuel type with active length of 50 cm in which the maximum predicted clad temperature is nearly 103.4 °C while the maximum allowable one is 105 °C.
9. An extension for this model is underway to cover cases beyond steady state operation, i.e., transient TH calculations.

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Nomenclature

A	channel flow area, m^2
Cp	specific heat at constant pressure, J/kg K
d	fuel rod diameter, m
De	bundle equivalent hydraulic diameter, m
H	active length, m
G	mass flux, $kg/m^2 \cdot s$
h	heat transfer coefficient, $W/m^2.$℃
I	enthalpy, J/kg
k	thermal conductivity, W/m.℃
Nu	Nusselt number = h*De/k
Pr	Pradentl number = m*Cp/k
Re	Reynolds number = G*De/m
q	volumetric heat generation, W/m3
r	distance in radial direction, m
T	temperature C
u	coolant velocity, m/s
z	distance in axial direction, m

Greek symbols

a	thermal diffusivity, m2/s
ρ	density, kg/m3
μ	dynamic viscosity, kg/m.s
ϕ	surface heat flux, W/m2

Subscripts

co	coolant bulk
co_{in}	coolant bulk inlet
c	clad
f	fuel
ONB	onset of nucleate boiling
DNB	departure from nucleate boiling
OFI	onset of flow instability