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Thermo-hydraulic feasibility study for the power-upgraded of Tehran Research Reactor compact core

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HIGHLIGHTS

- Two upgrade strategies: reuse of spent fuel and compact core design for enhanced neutron flux.
- Compact core with 26 fuel assemblies operates safely at 8.5 MW, achieving $> 1.5 \times 10^{14}$ n.cm^{-2}.s^{-1} thermal flux.
- Thermal-hydraulic analysis confirms safety at 8.5 MW with PPF = 2.7 and full compliance at 8.3 MW.
- Conservative analysis (PPF = 3.0) allows safe operation at 7.5 MW.
- Lowering coolant inlet temperature by 3 °C enables safe operation at 8.5 MW and potential for > 9 MW.

ABSTRACT

This study investigates the feasibility of a power upgrade for the Tehran Research Reactor (TRR) to enhance neutron flux for various applications. Two strategies are proposed: the utilization of spent fuel to ensure economic viability, and (2) the adoption of a compact core layout to maximize neutron flux. Neutronic simulations show that the compact core, comprising 26 fuel assemblies, can operate safely at 8.5 MW, yielding a thermal neutron flux exceeding 1.5×10^{14} n.cm^{-2}.s^{-1}. Thermal-hydraulic analysis confirms that, with a power peaking factor (PPF) of 2.7, all safety criteria are met at 8.5 MW. A slight reduction in power to 8.3 MW ensures full compliance with all safety requirements. Under a more conservative approach, assuming a PPF of 3.0, a safe operational power of 7.5 MW remains achievable. Furthermore, lowering the coolant inlet temperature by 3 °C demonstrably improves reactor performance, allowing safe operation at 8.5 MW. This modification also presents the potential to exceed 9 MW at elevated mass flow rates while consistently maintaining adequate safety margins.

KEYWORDS
TRR
MTR_PC
CITVAP code
Compact core
Power peaking factor (PPF)
Coolant inlet temperature

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Abbreviations

IAEA	International Atomic Energy Agency
CFE	Control Fuel Element
FA	Fuel Assembly
IR Box	Irradiation Box
LEU	Low Enriched Uranium
MTR	Material Test Reactor
PPF	Power Peaking Factor
RPPF	Radial Power Peaking Factor
SAR	Safety Analysis Report
SFE	Standard Fuel Element
SRF	Safety Reactivity Factor
SSR	Shim Safety Rod
RR	Research Reactor
TRR	Tehran Research Reactor
DNBR	Departure from Nucleate Boiling Ratio
FE	Fuel Element
GR-Box	Graphite Box
OFI	Onset of Flow Instability
ONB	Onset of Nucleate Boiling
ONBR	Onset of Nucleate Boiling Ratio
CHF	Critical Heat Flux

1 Introduction

Research reactors are advanced devices used for research in particle and nuclear physics, radiochemistry, materials sciences, nuclear power and medicine. They produce high technology commodities like radioactive isotopes and radiation modified materials, nuclear fuel tests and study radiation resistance. RRs also contribute to education and training in science, engineering, and medicine (IAEA, 2014). The operational lifetime of over half of the world's

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RRs exceeds 40 years. With the aging of RRs and the decrease in financial resources available for the establishment of new RRs, the pressure to optimize this category of reactors has increased. The refurbishment process is usually executed as a project or a set of smaller sub-projects to achieve an overarching goal. Therefore, many reported RRs have modified or upgraded their structures and auxiliary equipment during their operational lifetime to facilitate the development and enhancement of new experiments. These reactors are typically of the plate-type fuel. Many new industrial and medical requirements necessitate neutron fluxes in the range of 10^{13} to 10^{14} n.cm^{-2}.s^{-1}, a range achievable in reactors with power outputs in the range of a few megawatts. Upgrading the power of RRs is a practical solution to achieve the desired neutron flux. The power upgrade projects of RRs in several countries in recent years, under the supervision and support of the IAEA, have been operational and successful (Bradley, 2009).

The Budapest RR is a 2 MW pool-type reactor with light water moderation and cooling. It was commissioned in 1967 using a new type of fuel and beryllium reflectors, upgrading from 2 to 5 MW, and later in 1986, it was refurbished to 10 MW. Since then, the reactor has been operational for an average of 3500 hours per year without significant issues (IAEA, 2002). The Pakistan RR was commissioned in 1965 at a power level of 5 MW. Due to changing experimental needs and demand for higher neutron flux, the reactor's power was increased from 5 to 10 MW (Israr et al., 2009).

The TRR is a 5MW pool-type reactor that uses light water as a moderator and coolant. Initially, it was started with HEU fuel. However, in later years, following the approval of the IAEA and the Non-Proliferation Treaty, new LEU fuels were employed. The power upgrade of the TRR can be significant from two perspectives: the first being the refurbishment or expansion of the RR's capabilities, and the second being the increase in reactor applications resulting from the power upgrade.

In the early 1991, INVAP (IAEA, 2014) presented an initial proposal for upgrading the power of the TRR synchronously with the change of the TRR fuel enrichment from HEU to LEU. Two proposed methods were offered to provide the required coolant flow rate to achieve the desired mass flow rate. The first method involved designing a completely new cooling system for the primary and secondary circuits capable of providing 10 MW of power at a mass flow rate of 1300 m^3.h^{-1}. The second method involved designing auxiliary primary and secondary circuits to work in parallel with the existing circuits.

In 2007, a study conducted by (Farhadi and Khakshournia, 2008) examined the possibility of increasing the flux of the TRR. The final results of these studies, taking into account minimal changes in the main systems and the absence of destructive events in the reactor pool, indicated that for a balanced core with of 27 fresh fuel assemblies without any changes in the core's structure and assuming full opening of the control rod, the maximum upgradable power is approximately 7.5 MW. However, with the increase in power and fluid flow rate, certain modifications to the reactor systems and the replacement, modification, and optimization of some equipment are necessary. It is evident that some modifications to the reactor cooling system are required, given the age of the reactor's main systems. Upgrading and replacing instrumentation and control systems, ventilation systems, irradiation pipes, and ultimately strengthening their shielding in accordance with international requirements are also necessary.

In 2019, the results of a study on changing neutron reflectors around the core showed that replacing the graphite neutron reflector with heavy water, beryllium, and beryllium oxide at the TRR cannot significantly increase the thermal neutron flux (Gholamzadeh et al., 2019). In 2021, studies were conducted to increase the thermal neutron flux in the core of the TRR for carrying out domestic fuel tests In this regard, compact core layouts were proposed as a method in this study. A compact core layout containing 19 SFE, 5 CFE, and 6 irradiation compartments in the central core that meets all design criteria. Simulation results using the MCNPX nuclear code show that the thermal neutron flux in the proposed central irradiation compartment of the compact core has increased by 35% compared to the core layout with 33 fuel assemblies. Despite considering conservative conditions and uncertainty in power and ONB relationship, all thermal-hydraulic safety parameters in the proposed compact core are within the safe range (Arshi et al., 2021).

Despite the aforementioned studies, this research has taken a new approach to the power upgrade issue. Since the proposal for fundamental changes in reactor infrastructure and equipment did not yield results in previous research, this study aims to address the power upgrade possibility of the TRR with minimal changes in the reactor core structure and cooling system, utilizing existing facilities. Two new approaches are added to the previous methodologies. The first approach suggests using spent fuel instead of fresh fuel in the design of the proposed core, taking into account economic feasibility and a realistic view of the nuclear fuel cycle. The second approach focuses on compacting the reactor core as much as possible to achieve a higher neutron flux range. Neutronics calculations for this research have been carried out (Aminfarkhani et al., 2023). This research showed that the compact core with 26 fuel assemblies fulfilled all neutronic and operation criteria. Considering thermal hydraulic aspect from previous study (Farhadi and Khakshournia, 2008), and the results of various core configurations showed that TRR can be upgraded to 8.5 MW and consequently the thermal flux larger than 1.5×10^{14} n.cm^{-2}.s^{-1} can be achievable. In this study, simultaneous with the two new approaches, the possibility of power upgrade is investigated for two conservative cases with a maximum PPF of 3 and a realistic case with a maximum power peaking factor of 2.7. The possibility of using a compact cooling system to reduce the coolant temperature has also been studied. Based on the proposed core layout that is capable of meeting the neutronic criteria, the thermal-hydraulic criteria have been examined.

In the first stage, the fluid velocity in the coolant channels is calculated, followed by an analysis of the existing

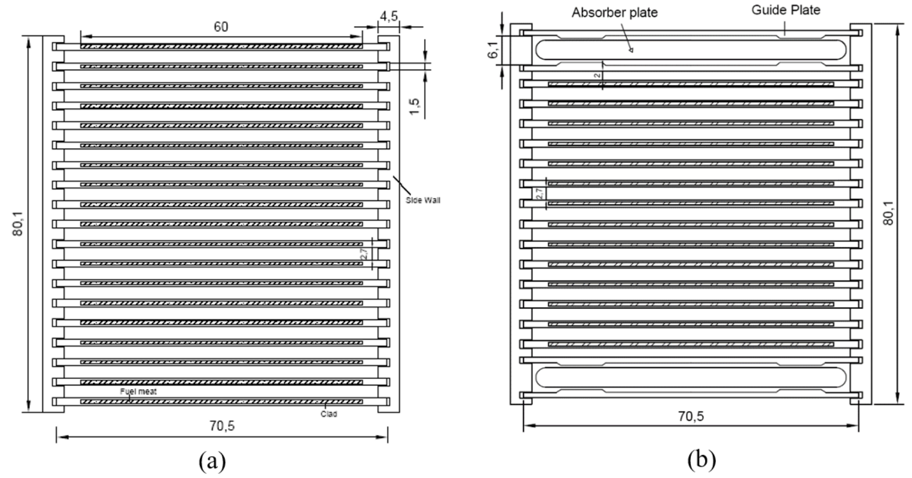

Figure 1: The cross-sectional view of the TRR fuels: (a) SFE, (b) CFE (all dimensions in cm).

temperature conditions for the proposed core in the reactor's steady state. Safety criteria, including maximum clad temperature and departure from nucleate boiling, are compared with the SAR of TRR. Furthermore, the effect of coolant inlet temperature on the level of power upgrade has also been investigate.

2 PROCEDURES

2.1 Description of TRR

The TRR is a 5 MW pool-type RR with heterogeneous solid fuels which is cooled and moderated with light water. TRR uses U3O8-Al MTR fuel. The reactor core is made up of SFEs and CFEs, which consist of 19 and 14 fuel plates, respectively. Figures 1-a and 1-b show a cross-sectional view of SFE and CFE The number and arrangement of the fuel elements in the core may be varied depending on thermal hydraulic and neutronic safety considerations as well as operational requirements. The coolant is gravity driven from pool to hold-up tank and forced flow from hold-up tank through heat exchangers to the reactor pool. Main characteristics of TRR are presented in Table 1. The geometrical data of SFE and CFE is provided in Table 2. The TRR-SAR provide further information on LEU fuel assembly and core characteristics (IAEA, 2001).

2.2 Description of TRR Compact Core

Figure 2 shows the compact core configuration with 26 FEs. Table 3 shows the main neutronic parameters of 26 FEs core configuration. Neutron flux in three thermal, epi-thermal and fast groups in the central and surrounding irradiation channels are shown in Table 4. The results show the average thermal flux in the central channel is higher than 1.5×10^{14} n.cm^{-2}.s^{-1} in 8.5 MW.

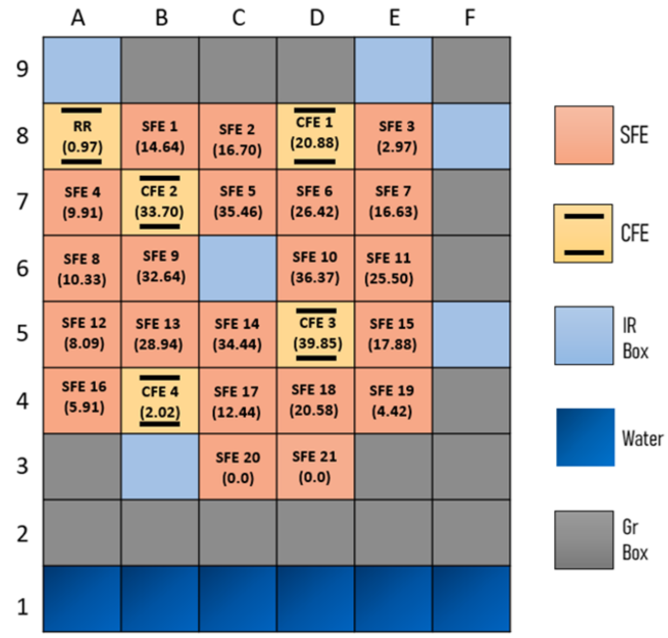

Figure 2: Compact core configuration with 26 fuel assemblies (21 SFE and 5 CFE).

2.3 Thermos-hydraulic Design Criteria

Several key safety limits are defined to ensure the integrity of the fuel plates and prevent critical thermal-hydraulic phenomena. These limits are based on preventing corrosion, fuel blistering, mechanical vibrations, and flow instabilities (INVAP, 2001).

- Maximum Cladding Temperature: A maximum cladding temperature of 105 °C is mandated to prevent corrosion. This translates to a heat flux requirement where the ratio of the heat flux at 105 °C to the maximum heat flux in the "hot channel" (the channel experiencing the highest heat flux) must be greater than 1.0.
- Maximum Fuel Temperature: The maximum allowable fuel temperature is set at 400 °C to avoid fuel blistering.

Table 1: Main Characteristics of Tehran Research Reactor.

Parameters	Values
Fuel elements:	
U-235 per SFE	290 g
U-235 per CFE	214 g
U per fuel plate	76 g
Meat:	
Enriched U_3O_8	20% in weight of U-235
U density	2.9617 g.cm^{-3}
Meat density	4.76 g.cm^{-3}
Void fraction	10.0%
Weight percentage	U-235 12.45%, U-238 49.78%, O 11.18%
Aluminum Meat	Purity 99.6% Density= 2.7 g.cm^{-3}
Frame and covers	Aluminum 6061 Density= 2.7 g.cm^{-3}
Shim and safety rods absorber	Ag-In-Cd Alloy (80,15,5% in weight respectively) Density 10.17 g.cm^{-3}
Control rods' Cladding Material	AISI-316/L stainless steel Density= 7.95 g.cm^{-3}
Gap between absorber and clad	He (1 atm. pressure)
Regulating rod	AISI-316/L stainless steel Density=7.95 g.cm^{-3}
Gird plate	Grid array X -Y Pitch: 7.71 × 8.1 cm
Grid plate material	AL-1100
Grid z thickness G	12.7 cm
Grid passing holes	54 holes diameter: 6.19 cm. Max, 6.17 cm Min 40 holes diameter: 2.222 cm With a reduction to 1.9053 cm
Reflectors	Water/Graphite

Table 2: Geometric Data for SFE & CFE.

Parameter	SFE	CFE
Number of Fuel Plates	19	14
FE External Size, cm	8.01 × 7.7 × 89.7	8.01 × 7.7 × 161.5
Plate Thickness, cm	0.15	0.15
Clad Thickness, cm	0.04	0.04
Water Channel, cm	0.27	0.27
Meat Thickness, cm	0.07	0.07
Meat Width, cm	6.0	6.0
Meat Length, cm	61.5	61.5
Side Walls Thickness, cm	0.45	0.45
Exit channel Length, cm	4.55	4.55
Inner Diameter of Exit Nozzle, cm	5.30	5.30
Outer Diameter of Exit Nozzle, cm	6.16	6.16
Coolant Flow Area, cm2	33.92	25.81
Heat Transfer Area, cm2	14022.0	10332

Table 3: Neutronic Parameters of core configuration with 26 fuel assemblies.

Neutronic Parameters	C&C	HFPX	Safety Criteria
Core Excess Reactivity (pcm)	7134.8	2775.2	
Absolut SDM (pcm)	6261.3	11131.3	>3000
Absolut SDM-1 (pcm)	683.9	5355.8	>500
APPF	1.3	1.3	
RPPF	1.786	1.779	
Total PPF	2.671	2.661	<3.0
RR worth	704	727.9	$<\beta_{eff}$
SRF	1.87	5.01	>1.5
Cycle Length (full power day)	-	24	

- Coolant Velocity: Coolant velocity is strictly controlled to prevent mechanical vibrations of the fuel plates and thermal instability. The maximum velocity is limited to two-thirds of the critical velocity. Conversely, a minimum coolant flow is also required to prevent degraded heat transfer due to vapor blanketing or thermally induced flow reduction. The design ensures that the local heat flux remains below

Table 4: Neutron Flux in Irradiation Channels.

	Irradiation Neutron Flux ($\times 10^{13}$ n.cm^{-2}.s^{-1})					
	A9	E9	F8	C6	F5	A3
State	Fast Epi-thermal Thermal	Fast Epi-thermal Thermal	Fast Epi-thermal Thermal	Fast Epi-thermal Thermal	Fast Epi-thermal Thermal	Fast Epi-thermal Thermal
HFPX	0.79 1.65 4.80	0.93 1.92 5.06	1.02 2.08 5.29	4.80 9.04 16.69	1.68 3.49 8.30	1.94 3.70 8.17

half the value that would lead to these phenomena under nominal conditions.

- Onset of Nucleate Boiling (ONB): While not inherently critical, ONB serves as a conservative steady-state limit. The ratio of the heat flux at ONB to the maximum heat flux in the hot channel must exceed 1.3
- Departure from Nucleate Boiling (DNB): DNB, the transition from nucleate to film boiling, is a far more serious event. To maintain a sufficient safety margin, the ratio of the heat flux at DNB to the maximum hot channel heat flux must be greater than 2.0.

2.4 Codes and simulation methodology

MTR_PC is a computational package developed by INVAP S.E. for performing neutronic, thermal-hydraulic, and shielding calculations for MTR-type reactors. The thermal-hydraulic module of MTR_PC consists of three codes, two of which are used in this study.

2.4.1 CAUDVAP V3.60

The flow behavior within a reactor system plays a crucial role in heat removal from the core and in minimizing dynamic stresses on internal components. CAUDVAP, a computational code within the MTR_PC package developed by INVAP S.E., calculates the velocity distribution in steady-state conditions (Abbate, 2003a). It models multiple parallel channels connected between common inlet and outlet plenums, even if height discrepancies exist between the inlet and exit sections.

CAUDVAP is specifically designed for liquid water flow through reactor channels. The code reads geometrical data from an input file, where users define the lengths, cross-sectional areas, and hydraulic diameters of each section, as well as the number of channels within the system. Using an iterative algorithm, CAUDVAP determines individual channel flows, ensuring a unique pressure drop value for each channel. The code provides two primary calculation modes:

- Flow Distribution Mode: The user specifies the total coolant flow through the core, and CAUDVAP calculates the pressure drop and the corresponding flow distribution across different channels.
- Pressure Drop Mode: The user provides a predefined pressure drop, and CAUDVAP determines the total core flow and its distribution across various channels.

2.4.2 TERMIC

TERMIC is a steady-state thermal-hydraulic code used for designing plate-type fuel assemblies (Abbate, 2003b). It determines the maximum allowable power and heat flux based on key thermal limits, including the ONB, Critical Heat Flux (CHF), and Flow Instability, while accounting for coolant velocity in both upward and downward light-water flow. Although the code primarily operates under single-phase conditions, it incorporates two-phase limits to enhance predictive accuracy.

The TERMIC model treats fuel channels as spaces bounded by flat plates, dividing them into axial cells for energy balance and pressure gradient calculations. It accounts for entrance/exit losses and frictional pressure drops using well-established correlations. Coolant properties are dynamically adjusted based on temperature and pressure variations. In each cell, local heat flux -either analytically determined or based on an external profile- is used to compute interface temperatures via forced convection correlations. Additionally, a one-dimensional (1D) conduction model is employed to determine internal fuel temperatures

The TERMIC V4.1 code utilizes various empirical correlations to predict critical thermal conditions. For the DNBR, the Mirshak correlation is employed for design purposes, while the Sudo-Mishima and Bernath correlations are used for comparison (Kalimullah et al., 2023; Sudo and Kaminaga, 1993; IAEA-TECDOC-233, 1980).

The DNBR is defined by the following relationship:

$$DNBR = \frac{q''_{CHF}}{q''_{actual}} \tag{1}$$

where q''_{CHF} is the critical heat flux, and q''_{actual} is the local surface heat flux.

For the ONB, the Bergles-Rohsenow correlation is applied. A Specifically, the Whittle-Forgan correlation serves as the primary design criterion, while the Saha-Zuber correlation is included for comparative analysis (Whittle and Forgan, 1967). Additionally, the Sudo-Mishima and Mirshak correlations are specifically used to calculate DNBR in the hot channel.

Table 5: Uncertainties of input parameters.

Parameter	Uncertainty
Coolant temperature at core inlet	2 °C
Uranium content per fuel plate	2%
Power measurement	5%
Heat transfer area	5%
Coolant channel width	7%
Coolant velocity in nominal channel	10%
Uranium distribution in fuel plates	8%
Meat thickness	10%
Heat transfer correlation	10%
Pressure	0%
Pool water level	4%
Coolant density	0%
Friction pressure drop formula	10%
Form loss pressure drop formula	10%
ONB correlation	10%
DNB correlation	10%
OFI correlation	6%

Table 6: Initial operational input data for thermal-hydraulic analysis.

Thermal-hydraulic parameters	Physical value
Reactor thermal power (MW)	7.5 - 9.5
Total coolant volumetric flow rate ($m^3.h^{-1}$)	750
Core inlet coolant temperature (°C)	37.8
Inlet pressure at the channel (bar)	1.689
Radial power peaking factor	1.8
Axial power peaking factor	1.3
Best estimate Total PPF	2.7
Conservative Total PPF	3
Average power density in hot channel ($W.cm^{-3}$)	1137.0
Thermal conductivity of fuel ($W.m^{-1}.°C^{-1}$)	10
Thermal conductivity of clad ($W.m^{-1}.°C^{-1}$)	180

Table 7: Properties of SFE and CFE Channels Defined in CAUDVAP.

Channel Name	Length	Area	Hydraulic Diameter	Type
SFE	0.0455	4.98370E-03	7.04990E-02	Rectangular
	0.6550	3.27550E-03	5.22040E-03	Rectangular
	0.0254	4.98370E-03	7.04990E-02	Rectangular
	0.0357	2.64210E-03	5.80000E-02	Circular
	0.1357	2.20620E-03	5.30000E-02	Circular
CFE	0.0869	4.02670E-03	6.33630E-02	Rectangular
	0.6550	2.61970E-03	5.01810E-03	Rectangular
	0.0254	5.04510E-03	7.09080E-02	Rectangular
	0.0357	2.64210E-03	5.80000E-02	Rectangular
	0.1357	2.20620E-03	5.30000E-02	Circular

Table 8: Coolant Velocities for Different Mass Flow Rates ($m.s^{-1}$).

Mass Flow Rate ($m^3.h^{-1}$)	CFE Velocity ($m.s^{-1}$)	SFE Velocity ($m.s^{-1}$)
750	2.294	2.297
700	2.141	2.144
650	1.987	1.991
600	1.834	1.838
550	1.680	1.685
500	1.527	1.532

2.4.3 Uncertainty Analysis in TERMIC

The thermal performance of a nuclear core is strongly influenced by uncertainties in input data and reactor operating conditions. TERMIC evaluates the effects of these uncertainties on: Local coolant temperature, Heat flux distribution, Wall temperature variations, Saturation temperature, Pressure drop, ONB conditions, CHF prediction methods and Flow redistribution correlations. A detailed list of uncertainties in the input parameters used in TERMIC is presented in Table 5 (Abbate, 2003b). To apply all these sub- uncertainties, which must be combined in a specific manner, three methods are proposed. The first method involves the simple multiplication of all sub-factors, leading to a conservative result. The second method is statistical summation, and the third is a combination of the first and second methods. Considering that all uncertainties do not occur at a single time and location, the application of the first method is unrealistic. In the statistical summation of sub-factors (F_i), the following formula is used:

$$F_{total} = \sqrt{\sum_{1}^{n} F_i^2} \tag{2}$$

3 Results and discussions

The CAUDVAP program can calculate the velocity distribution in steady state regime through different channels connected in a parallel array between inlet and outlet common plenums. The geometrical data of the channels is read from a file that are given from Table 2. CAUDVAP, calculates the flows through the channels in a unique pressure drop value for all the channels. In this research the Flow Distribution Mode is activated. The total flow through the core is determine, therefore CAUDVAP calculates, the total pressure drop across the core and the way the total flow is distributed through the different channels. The exact values of some parameters and boundary conditions are given in Table 6. Properties of SFE and CFE channels defined in CAUDVAP are provided in Table 7 for five different zone of fuel assembly from inlet to outlet.

3.1 First Strategy: Increasing the Mass Flow Rate to 750 $m^3.h^{-1}$

The first strategy focuses on increasing the mass flow rate to 750 $m^3.h^{-1}$. This can be achieved by fully opening the output butterfly valve during the initial operating cycle, effectively raising the mass flow rate from 500 to 750 $m^3.h^{-1}$. Table 8 presents the CAUDVAP simulation results, illustrating the coolant velocities for the SFE and CFE fuel channels under mass flow rate variations from 500 to 750 $m^3.h^{-1}$. To meet the design criteria, the coolant velocity within fuel assemblies must remain below 15.3 $m.s^{-1}$. The results in Table 8 confirm that, for the proposed mass flow rate increase, all coolant velocities remain within the acceptable safety limits.

Table 9: Clad temperature in 8.5 MW.

PPF	Mass Flow Rate ($m^3.h^{-1}$)	Temperature (°C)
2.7	750	106.2
	700	109.5
	650	113.6
	600	118.1
	550	123.2
	500	129.6
3.0	750	112.6
	700	116.3
	650	120.9
	600	125.8
	550	131.3
	500	138.3

Table 10: Powers at 105 °C Clad Temperature.

PPF	Mass Flow Rate ($m^3.h^{-1}$)	Power (MW)	Power with Uncertainty (MW)
2.7	750	8.78	8.35
	700	8.29	7.90
	650	7.76	7.39
	600	7.27	6.92
	550	6.77	6.45
	500	6.22	5.92
3.0	750	7.89	7.51
	700	7.46	7.10
	650	6.99	6.66
	600	6.54	6.23
	550	6.09	5.80
	500	5.60	5.33

Table 11: ONB and DNB Rates at 3.8 and 5.7 MW Power.

Power (MW)	Peak Power Factor	Mass Flow Rate ($m^3.h^{-1}$)	DNB	ONB
8.3	2.7	750	4.51	1.31
	3	750	4.02	1.17
7.5	3	750	4.49	1.31

These results demonstrate that all coolant velocities associated with the proposed mass flow rate increase conform to the reactors safety margins. In the next step, in order to find out whether the calculated coolant velocity between fuel plates is enough for effective heat removal from the hot channel in each assumed mass flow, TERMIC code was applied to calculate thermal hydraulic safety parameters in the hot channel of each coolant velocity obtained in the previous step. The first safety studied parameter is the temperature of the fuel clad. To prevent corrosion-related issues in fuel elements, the maximum clad temperature should not exceed 105 °C (the temperature at which the Al_2O_3 oxide layer forms on the cladding, significantly reducing thermal conductivity. Table 9 shows the cladding temperature of the proposed upgrading power 8.5 MW, from neutronic calculations in the different mass flow rates. The calculations performed with two conservative and best estimate approaches. The PPF value in the conservative approach is considered about 3 but in the case of best estimation, it is considered 2.7 that taken from neutronic calculation (Aminfarkhani et al., 2023). According to the results the safety criterion for cladding temperature is not met for the power of 8.5 MW. Therefore, by performing inverse calculations using the TERMIC code and restricting the cladding temperature to 105 °C, the search for an appropriate power is carried out. The results of these calculations can be seen in Table 10.

According to the results reported in Table 10 and taking into account the uncertainties considered in the calculations by the TERMIC code, it is observed that with a mass flow rate of 750 $m^3.h^{-1}$ and a realistic maximum power factor of 2.7 for the proposed neutronic core arrangement, power upgrade to 8.35 MW is feasible. Additionally, under conservative conditions with a maximum power factor of 3, achieving a power of 7.51 MW at a mass flow rate of 750 $m^3.h^{-1}$ will be possible. The second safety condition under review is the ONB and DNB rates that calculated in Table 11.

According to the simulation results, under realistic operating conditions and with maximum mass flow rate of 750 $m^3.h^{-1}$, the clad temperature remains below the safety limit of 105 °C at the of 8.3 MW reactor power. This confirms compliance with the thermo-hydraulic safety criteria. Therefore, a power uprate to 8.3 MW can be considered technically feasible within the validated safety margins, without compromising the integrity of the fuel cladding or core cooling performance.

3.2 *Second Strategy: Reducing the Inlet Coolant Temperature*

Table 12 summarizes the calculation results for two normal and maximum mass flow rates of 500 and 750 $m^3.h^{-1}$, considering both realistic and conservative approaches for various inlet coolant temperatures. These calculations determined the maximum upgradable power for a maximum clad temperature of 105 °C under various conditions. The calculated powers are determined with and without considering the uncertainties used in the TERMIC code. The maximum upgraded power with PPF=3 and mass flow rate of 700 $m^3.h^{-1}$ is predicted to be approximately 7.5 MW for an inlet coolant temperature of 37.8 °C, which is in complete agreement with the reference results (Farhadi and Khakshournia, 2008). The ONB and DNB criteria for the calculated powers are provided in Table 8.

The results in Table 12 confirm that the reduction of the inlet coolant temperature has a positive and direct effect on the possibility of increasing the core power of the TRR. Table 12 shows that the power can be increased from 5.30 MW up to 9.6 MW, based on the coolant inlet temperature, the mass flow rate, and the applied method for power enhancement.

4 Conclusions

The primary objective of this study is to evaluate the thermal-hydraulic safety margins of a proposed core configuration aimed at enhancing the neutron flux and operational power of the TRR. Steady-state simulations predict

Table 12: Power, ONB Rate, and DNB Rate vs. Coolant Temperature at the maximum clad Temperature of 105 °C.

Mass Flow Rate ($m^3.h^{-1}$)	PPF	Inlet Coolant Temperature (°C)	Power (MW)	Power with Uncertainty (MW)	ONB	DNB
500	2.7	37.80	6.19	5.89	1.30	6.80
		35	6.42	6.11	1.29	6.64
		30	6.82	6.50	1.27	6.40
		25	7.21	6.87	1.26	6.19
	3	37.80	5.57	5.30	1.30	6.80
		35	5.77	5.50	1.29	6.64
		30	6.14	5.85	1.27	6.40
		25	6.49	6.18	1.26	6.19
750	2.7	37.80	8.74	8.32	1.31	5.13
		35	9.06	8.63	1.30	5.01
		30	9.63	9.17	1.29	4.83
		25	10.2	9.69	1.27	4.67
	3	37.80	7.87	7.50	1.31	5.13
		35	8.16	7.77	1.30	5.01
		30	8.66	8.25	1.29	4.83
		25	9.16	8.72	1.27	4.67

the coolant velocity and critical safety parameters, including the maximum clad temperature, ONB and DNB limits.

The results indicate that at an operational power of 8.5 MW, some safety criteria are violated. However, reducing the power to 8.3 MW while applying a maximum power peaking factor of 2.7 ensures that all safety requirements are satisfied. This power, according to neutronic calculations, is equivalent to a thermal neutron flux greater than 1.5 of the order of 10^{14}. Under a conservative approach, employing a peaking factor of 3.0, a safe operational power of 7.5 MW is achieved without compromising thermal-hydraulic reliability.

Moreover, sensitivity analyses reveal that lowering the inlet coolant temperature significantly improves the reactor core's operational power potential. A reduction of 3 °C enables safe operation at 8.5 MW, and further temperature decreases allow for power levels above 9 MW under realistic assumptions with high mass flow rate, while ONB margins remain within an acceptable range.

Power uprating of TRR, without any modifications to its existing cooling system, needs that the primary coolant flow rate increase from 500 $m^3.h^{-1}$ to 750 $m^3.h^{-1}$. Consequently, the secondary coolant loop will require pump replacements to increase the flow rate. Furthermore, to effectively remove thermal load, the implementation of a parallel heat exchanger system with the currently installed heat exchanger, is proposed. These thermo-hydraulic modifications are crucial for maintaining core thermal limits and ensuring operational safety under elevated power conditions

Conflict of Interest

The authors declare no potential conflict of interest regarding the publication of this work.

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