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INVESTIGATING THE FLOW INSTABILITY AT THE ITR-4M FUEL OF TNRC’S REACTOR USING PLTEMP/ANL CODE

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دراسة عدم استقرار تدفق المبرد في وقود ITR-4M لمفاعل تاجوراء للأبحاث النووية باستخدام برنامج PLTEMP/ANL

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مؤسسة الطاقة الذرية، طرابلس، ليبيا

الملخص

تُعدّ دراسة عدم إستقرار تدفق المبرد أمرًا بالغ الأهمية في المفاعلات النووية لتفادي حدوث أي اضطراب أثناء حادث افتراضي. فعدم الاستقرار في المبرد من شأنه أن يخفض من هوامش الأمان ضد ظاهرة فيض الحرارة الحرجة، حيث ترتفع درجة حرارة سطح مغلف الوقود بشكلٍ سريع. في هذا البحث، تمّ التحري من استقرارية التدفق في مفاعل مركز تاجوراء للأبحاث النووية، الذي يحتوي على قلب من اليورانيوم منخفض التخصيب (LEU)، ونوع مجموعة الوقود المستخدمة فيه هو IRT-4M. تمّ تقييم عدم إستقرار التدفق في الخلية الساخنة في المفاعل باستخدام برنامج PLTEMP/ANL.

يُحسب هامش الأمان لعدم استقرار التدفق بتشغيل البرنامج عند أقصى قدرة للمفاعل (9.7 ميغاوات) مع تغيير معدل التدفق للمبرد لإيجاد النقطة التي تساوي فيها نسبة عدم الاستقرار FIR واحدًا. كانت درجة حرارة الدخول لسائل التبريد هي 45 درجة مئوية، وفرق ضغط المفاعل 0.066 ميغاباسكال، وضغط الدخول 0.179 ميغاباسكال. في البداية، أُجريت الحسابات عند القدرة القصوى للمفاعل ومعدل تدفق يساوي 8.1 كيلوغرام/ثانية في الخلية الساخنة. فأظهرت النتائج أن أقصى درجة حرارة لسطح المغلف هي 96.971°م وهي أقل من الدرجة المسموح بها (102°م)، وأدنى قيمة لـ FIR، ONBR و DNBR هي 2.034، 1.559، 4.147، على التوالي. ثم أجريت الحسابات بتخفيض قيمة معدل تدفق المبرد للبحث عن بداية حدوث عدم الاستقرار ، فتبيّن أن نسبة عدم استقرار التدفق (FIR) يساوي 1.0 بمعدل تدفق يساوي 3.976 كيلوغرام/ثانية. وبحساب هامش الأمان لعدم استقرار التدفق وجد أنه يبلغ 2.038. وبذلك يُستنتج أن معدل التدفق داخل المفاعل مستقر عند تشغيله بالقدرة القصوى، كما يُستخلص من قيمة هامش الأمان لعدم استقرار التدفق أنها تتوافق مع متطلبات السلامة الحرارية الهيدروليكية للمفاعل.

الكلمات المفتاحية: عدم استقرار التدفق، قلب مفاعل مركز البحوث النووية بتاجوراء، نظام الغليان تحت التبريد، هوامش الأمان، البرنامج PLTEMP/ANL.

ABSTRACT

The examination of the flow instability is crucial in nuclear reactors to prevent the occurrence of flow excursion during a postulated accident. Instabilities can reduce the safety margins against these critical heat flux phenomena, where the fuel cladding surface temperature increases dramatically. In this paper, flow instability is investigated at the reactor of Tajoura Nuclear Research Centre with low enriched uranium (LEU) core, the recent reactor's fuel assembly type is IRT-4M. The flow instability is assessed at the hottest cell of the reactor by implementing PLTEMP/ANL code.

The safety margin to flow instability is evaluated at the maximum power of the reactor (9.7 MW) by running the code at various core flow rates to find the point where FIR equals 1. The inlet coolant temperature is 45°C, the core pressure drop is 0.066 MPa, and the inlet pressure is 0.179 MPa. At the beginning, the calculations have been obtained at the flow rate of 8.1 kg/s. The results show that the maximum cladding surface temperature is 96.971°C which is less than the permissible value (102°C), the minimum FIR, ONBR, and DNBR equals to 2.034, 1.559, and 4.147, respectively. By reducing the core flow rate at the maximum power level, it is determined that FIR equals to 1.0 at the flow rate of 3.976 kg/s. Which means that the safety margin value to flow instability is 2.038. It is concluded from the safety margin value to flow instability that it is in agreement with the thermal-hydraulic safety requirements of TNRC's reactor.

KEYWORDS: Flow instability, TNRC's core, sub-cooled boiling regime, safety margins, PLTEMP/ANL code.

INTRODUCTION

Flow instability is a critical safety issue in nuclear reactors, referring to the unpredictable and oscillatory behaviour of coolant flow within the reactor core. This phenomenon can have significant consequences on reactor performance, safety, and overall power output [1]. Flow instability occurs when the coolant flow rate, pressure, or temperature fluctuates, causing the reactor to deviate from its intended operating conditions and can cause potentially catastrophic accidents [2] and [3]. Flow instability can have significant safety consequences such as: flow stagnation, insufficient coolant, Departure from Nucleate Boiling (DNB) and Onset of Nucleate Boiling (ONB) [2].

Tajoura Nuclear Research Centre's reactor has been converted from a HEU to LEU fuel in 2006 [4]. Activities during the operation of nuclear reactor is subjected to standards of safety [5]. In the reactor core, the coolant flow direction is downward and it currents into subchannels, of which the geometry has a narrow channel with a comparatively wide width and a long height. The thermal-hydraulic and safety analysis are vital for the core. Therefore, it is necessary to predict flow instability in narrow channels. One of the initiating postulated accidents to accomplish the safety analysis is a loss of flow accident [6] and [7]. Flow instability can be considered as a loss of flow accident in the reactor. Flow instability can cause a premature existence of critical heat flux, and ultimately leads to burnout of fuel elements [8]. Therefore, this study has great significance for safety analysis of TNRC's reactor. Numerous studies have been conducted on the flow instability for the narrow rectangular channel and downwards flow [9-15]. The PLTEMP/ANL code was utilized in this study to evaluate the flow instability at a steady state along the hottest cell of the core.

PLTEMP/ANL V4.2 is a FORTRAN program that generates a steady-state temperature and flow rates for a reactor core, or for a fuel assembly [16]. It was designed to analyze the "MTR-type" fuel assemblies. "PLTEMP" codes is developed to PLTEMP/ANL at the Argonne National Laboratory (ANL) and some Russian correlations are included in this version and using of VVR-KN and IRT-4M fuel assemblies which are Russian fuel-types [16].

The PLTEMP/ANL code was used for the thermal-hydraulic analysis of the Kazakhstan WWR-K reactor core which is loaded with LEU fuel, and the results were verified by those of the RELAP5 code, and a reasonable agreement was attained for the cladding surface temperatures [17] and [18]. Moreover, the PLTEMP/ANL code was used for the simulation of both the Uzbekistan WWR-SM and IRT-Sofia reactors which are loaded with LEU IRT-4M fuel [19] and [20]. In addition, the PLTEMP/ANL code was performed for the evaluation of the safety parameters of the hottest cell of 10MW multipurpose research reactor at a steady state using VVR-KN research reactor fuel [21]. Furthermore, the PLTEMP/ANL code was used for simulating of the deposition and transfer of nuclear heat generation in the HEU fuel assembly, and the accomplishment of the safety analysis at the steady-sate of the GHARR-1 reactor with LEU core [22] and [23]. The verification and validation of the PLTEMP/ANL code for thermal-hydraulic analysis of MTR research reactors have been conducted to ensure its accuracy and reliability. The verification includes measurements, experimental data, results from different codes such as RELAP5, MATLAB, Mathematica calculations. This rigorous process ensures that the PLTEMP/ANL code is a reliable tool for thermal hydraulic analysis in research reactors [24].

IRT-4M FUEL TYPE

The recent fuel type of the reactor of TNRC is IRT-4M fuel which is a Russian design with LEU fuel of 19.7 wt% of U^{235}, and the material of the fuel meat is UO_2-AL. The IRT-4M fuel assembly is recognised as high density with LEU fuel. The TNRC's reactor is a pool type reactor and the core contains 36 cells [25], where fuel assemblies occupy 16 cells and the removable beryllium units surround them into 20 cells as shown in Figure (1). Six cells of the fuel assemblies have eight tube fuel assembly (8TFA) and ten cells have six tube fuel assembly (6TFA). The shape of the fuel assemblies is nested thin wall square tube as indicated in Figure (2). The power density distribution of the reactor core is obtained from the neutronic codes such as MCNP and REBUS-PC codes. From the neutronic calculations, the hottest cell of the core is 8TFA and its location is illustrated in Figure (1), where its relative power is 0.083. The maximum operating power of the reactor has been examined in previous studies [26-27] and it is set that it must not to exceed the value of 9.7 MW.

The reactor is normally cooled by operating three pumps of the primary circuit. Each pump is linked with a heat exchanger where heat is transferred to a secondary circuit. Thus, heat is extracted from the secondary circuit to a third circuit. Therefore, heat is ejected from the third circuit to the atmosphere through two cooling towers. The reactor is operated with a downward flow. The number of pumps that are required at each power level are investigated in [28], it is concluded that one pump is enough for a power level less than 3 MW; two pumps are required for 3 MW $< P <$ 6MW; three pumps are needed for power level more than 6 MW.

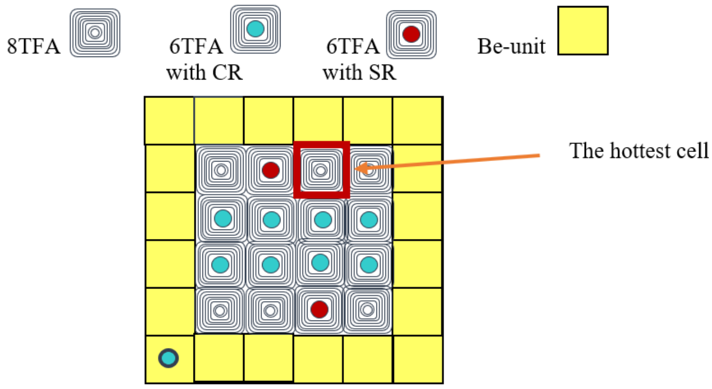


Figure 1: Core configuration of TNRC.

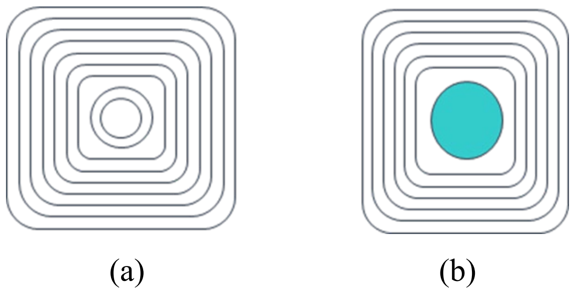


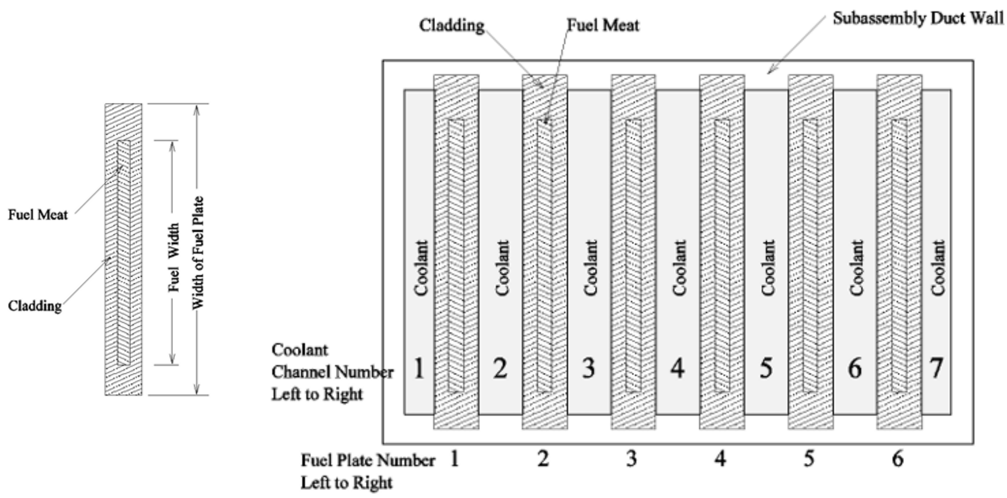
Figure 2: (a) Eight tube fuel assembly (8TFA); (b) Six tube fuel assembly (6TFA).

FLOW INSTABILITY

Flow instability refers to the tendency of a fluid flow to undergo oscillations in space or time when disturbed. The flow is defined to be stable, neutrally stable, or unstable, when the amplitude of the disturbance decreases, remains the same, or grows in time, respectively [2]. The investigation of the flow instability is vital in nuclear reactors to avoid the flow excursion when a postulated accident takes place. Flow instabilities affect the performance of the heat transfer, generate mechanical vibrations, cause thermal stress, decrease the critical heat flux, deteriorate the control system, and eventually reduce the safety margins [1] and [29]. According to the mechanism of flow instabilities, it has been categorized as two forms: static and dynamic [3]. These mechanisms contribute to flow instability in low-pressure, low-flow nuclear research reactors utilizing IRT-4M fuel assemblies. This instability mechanism may have a form of: pressure drop oscillations, oscillations in pressure and flow rate; flashing induced instability**,** instability caused by the fast change of liquid to vapor (flashing) due to pressure or temperature deviations; density-wave oscillations**,** oscillations resulted as a response of flow rate, heat transfer, and coolant density changes due to the boiling [3]. The interaction between these different types of flow instabilities creates strong nonlinear characteristics within the reactor system. Density-wave oscillations triggered by pressure drop fluctuations, can cause the flow of coolant to fully stop in certain channels, which named as flow stagnation. In addition, coolant flow instability can lead to an insufficient cooling of some fuel elements, causing localized temperature increases and potential fuel damage.

PLTEMP/ANL CODE

A steady-state temperature and flow solution are generated for a single fuel assembly, or for the whole nuclear reactor core using PLTEMP/ANL V4.2 code which is a FORTRAN program [16]. It was originally designed to analyze the “MTR-type” fuel assemblies. “PLTEMP” code is developed to PLTEMP/ANL at the Argonne National Laboratory (ANL) and some Russian correlations are included in this version. Modelling of the fuel and bypass regions is performed. The typical fuel assembly modeled by PLTEMP/ANL code may contain plates surrounded by coolant channels as shown in Figure (3), or nested tubes. Thus, the geometry is slab or cylinder. Each fuel plate may include up to five layers of various materials, with its own energy source. The fuel plate may be divided into several longitudinal increments, with its particular axial power distribution. The temperature solution is obtained for two-dimensional coordination. At the entrance to the assembly, a solution of one-dimensional coordination is performed for the coolant channels then for fuel plates or tubes. Hence for the next axial nodes, the temperature solution is repeated through out the length of the fuel assembly up to the exit of the coolant.



(a) (b)

Figure 3: (a) A fuel plate; (b) A typical fuel assembly [16].

Different thermal-hydraulic correlations are presented to obtain safety margins for example, departure from nucleate boiling (DNB), onset of nucleate boiling (ONB), and onset of flow instability (FI). FORTRAN functions include the physical properties of the coolant (light or heavy water). The code is designed for thermal-hydraulic analysis of nuclear research reactors in the regime of the sub-cooled boiling. Either turbulent or laminar flow regimes can be demonstrated. In addition, forced flow or natural circulation can be modelled.

In this study, the Russian correlation is applied to obtain the single-phase heat transfer coefficient, as follows:

$$Nu = 0.021\, Re^{0.8}\, Pr^{0.43}\, [Pr/Pr_w]^{0.25} \qquad (1)$$

and the Russian modified Forster-Greif correlation is adopted for the ONB thermal margin:

$$\Delta T_{sat} = 2.04\ q^{0.35}/P^{0.25} \tag{2}$$

$$T_w = T_{sat} + \Delta T_{sat}$$

where, the unit of the heat flux, q, in kW/m^2.

For evaluating the critical heat flux, the Mirshak-Durant-Towell correlation is implemented:

$$q_c = 1.51\ (1 + 0.1198\ U)(1 + 0.00914\ \Delta T_{sub})(1 + 0.19\ P) \tag{3}$$

Flow Instability Analysis with PLTEMP/ANL Code

The code controls flow excursion instability by means of three approaches: the Flow Excursion Ratio (FER); the Whittle and Forgan Flow Instability Ratio (FIR) correlation; and the criterion anticipated by Babelli and Ishii is presented in a basic form. Babelli and Ishii propose a method on flow excursion instability (downward flows) which is presented in the following equation:

$$\frac{N_{sub}}{N_{zu}} = \left(\frac{L_{nvg}}{L}\right)_{critical} + \frac{A_F}{\zeta_H L}\begin{Bmatrix} 0.0022\ Pe & \text{if} & Pe < 70000 \\ 154 & \text{if} & Pe > 70000 \end{Bmatrix} \tag{4}$$

Regarding this, the code calculates the ratio N_{sub}/N_{zu}. Hence, if the right-hand side of the previous equation is smaller than the ratio N_{sub}/N_{zu} then the flow is considered to be stable. And vice verse, it is unstable if the ratio N_{sub}/N_{zu} is lesser. The simplified criterion for flow instability is that: the flow is clearly stable when the ratio N_{sub}/N_{zu} exceeds 1.36; probably unstable for the value of the ratio from 1.36 to 1.0; and clearly unstable when the ratio is smaller than 1.0 [16].

The dimensionless non-boiling length L_{nvg}/L is computed for editing. The actual coolant flow rate of the channel is divided by the obtainable coolant flow rate to onset of boiling to determine the safety margin to flow instability. PLTEMP/ANL incorporates the Whittle and Forgan formula to predict the onset of flow instability. It is founded from the following formula [15-16]:

$$R = \frac{1}{1+\eta\frac{D_H}{L_H}} \tag{5}$$

To avoid the excursive flow instability, the value of η should be larger than 32.5. The safety margin to flow instability can be evaluated by running the code at different flow rates to determine the point where FIR equals 1, which is the ratio of that flow rate to the flow rate at the reactor's nominal full power. The estimation of the code for flow instability has been validated with experimental data and other codes like RELAP5, and showing good agreement [16].

Equivalent plate fuel type of IRT-4M fuel

The LEU IRT-4M fuel bundle has eight or six coaxial annular tubes (fuel elements) as mentioned previously and the hottest cell is 8TFA based on MCNP results. This shape is converted into parallel plates (MTR-type fuel assemblies) in order to be modelled by PLTEMP/ANL code as shown in Figure (4). This conversion is achieved by preserving the total volume of the fuel and the cladding of the origin shape. For the same dimensions of the pitch (7.15×10^{-2} m), fuel thickness (0.7×10^{-3} m), cladding thickness (0.45×10^{-3} m), and coolant channel gap (1.85×10^{-3} m), nineteen fuel plates are generated surrounded by twenty coolant channels.

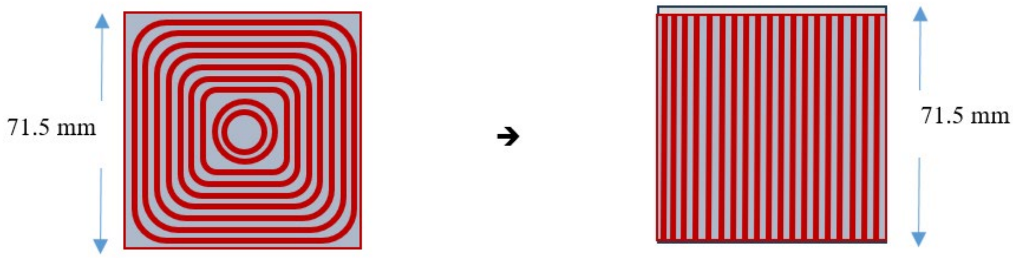


Figure 4: Converting the IRT-4M shape to plates.

RESULTS AND DISCUSSION

Flow instability at the hottest cell is examined by implementing the PLTEMP/ANL code. The calculations are conducted for the inlet coolant temperature of 45°C, and the inlet pressure is 0.179 MPa, and the core pressure drop is 0.066 MPa (0.67 kg_f/cm^2). The relative power and the axial power peaking factor (F_z) of the hottest cell are 0.083 and 1.254, respectively. By simulating the IRT-4M as parallel plates, nineteen fuel plates are created enclosed by twenty coolant channels. The width and the thickness of each coolant channel are 7.0×10^{-2} m and 1.85×10^{-3} m, respectively. The width of the fuel meat is 6.91×10^{-2} m.

Steady-State Thermal Hydraulic Analysis

At the beginning, a steady-state thermal hydraulic analysis is assessed with the reactor nominal power of 9.7 MW and 0.8045 MW at the hottest cell. At this power, the flow rate is 8.1045 kg/s at the hottest cell for the core pressure drop equals to 0.066 MPa with zero bypass flow rate. The temperature distributions along the hottest plate and hottest channel are shown in Figure (5). It is obvious that the maximum cladding surface temperature is 96.97°C and the outlet coolant temperature equal to 68.712°C Moreover, the onset of boiling temperature is above 120°C. Figure (6) demonstrates the heat flux as well as the critical heat flux at the hottest plate. It can be seen from Figure (6) that the heat flux is slighter than the critical heat flux which means that the reactor is safe. In addition, results indicates that the Flow Excursion Heat Flux Ratio FER (ORNL) is 3.9793 and the flow is stable since FER (ORNL) is larger than 1.0. In the hottest channel, Peak to Average Heat Flux Ratio = 1.6405. Thus, the flow is stable based on Babelli-Ishii criterion (N_{sub}/N_{zu}) Ratio which is 2.4576 (greater than 1.36). The resulted minimum FIR equals to 2.034, the minimum ONBR is 1.559, and the minimum DNBR is 4.147. Thus, it is apparent that the flow is stable and the reactor is safe when the reactor operates at the nominal power of 9.7 MW with operating of three pumps.

Safety Margins Evaluation

The code is executed by reducing the flow rate at the maximum power level to investigate the onset of flow instability by inspecting FIR equals to 1. Figure (7) displays the minimum FIR as a result of decreasing the flow rate at the hottest channel. Figure (8) illustrates the minimum ONBR, min DNBR as well as the minimum FIR by reducing the flow rate.

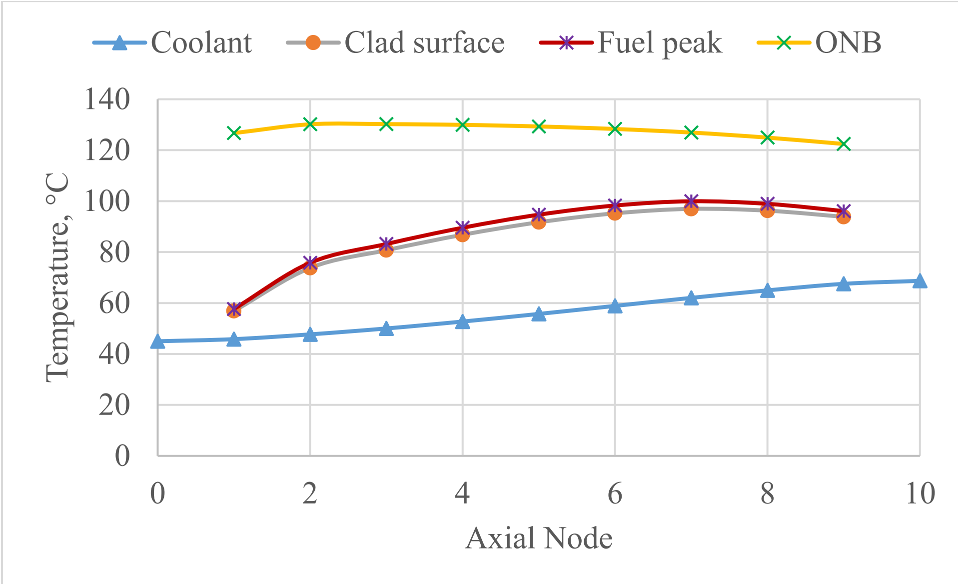


Figure 5: Temperatures along the hottest channel at the power 9.7 MW.

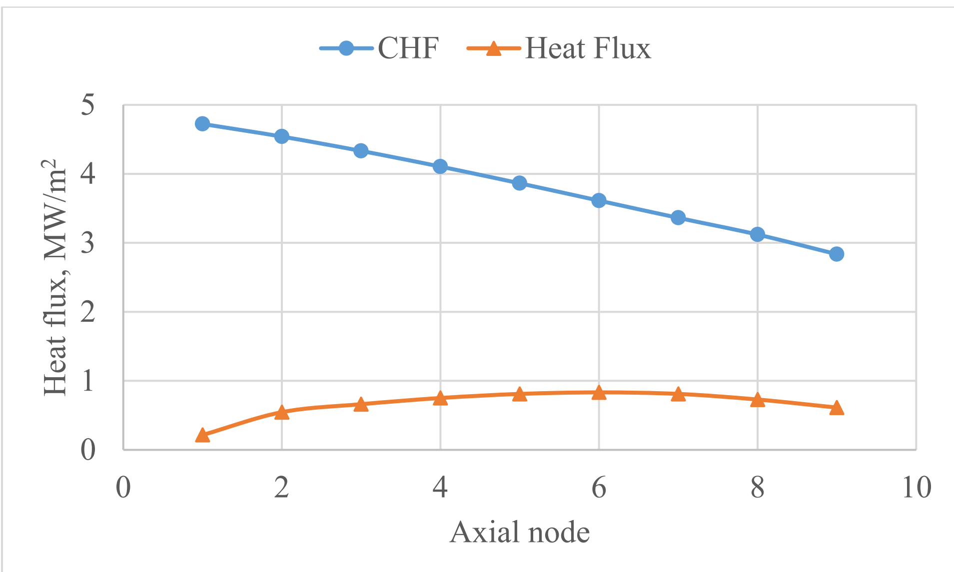


Figure 6: Heat flux and the critical heat flux along the hottest channel at the power 9.7 MW.

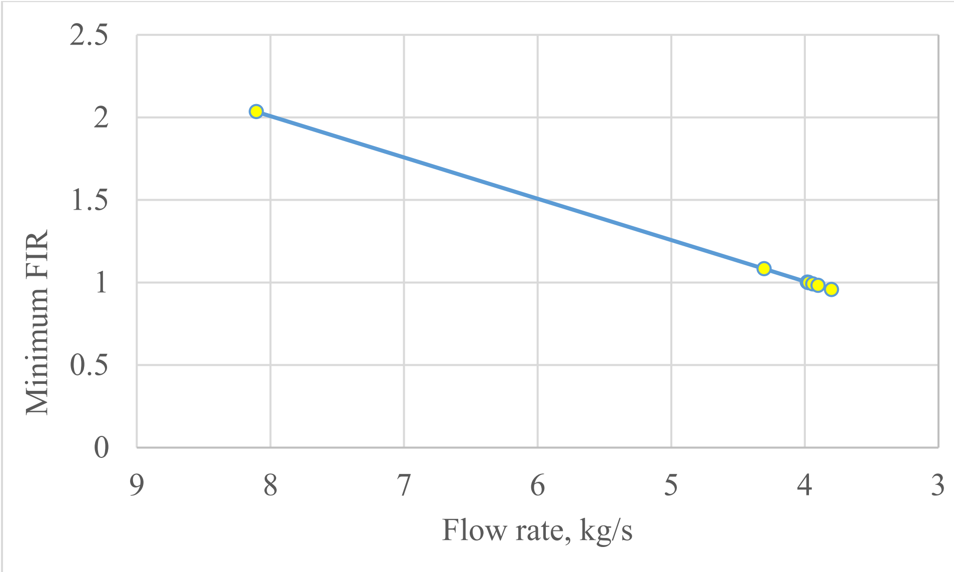


Figure 7: Minimum Flow Instability Ratio by reducing the flow rate at the maximum power level of 9.7 MW.

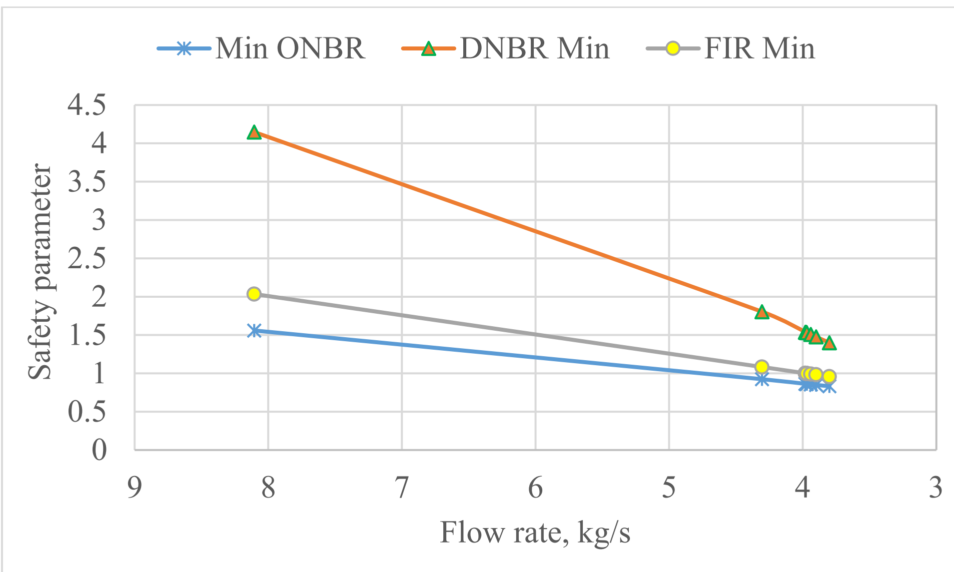


Figure 8: Minimum ratio of ONB, FI, DNB as a result of the reduction of the flow rate (core power = 9.7 MW).

The variation of the maximum cladding surface, fuel, and coolant temperatures are presented in Figure (9). Different parameters for this investigation are shown in Table (1).

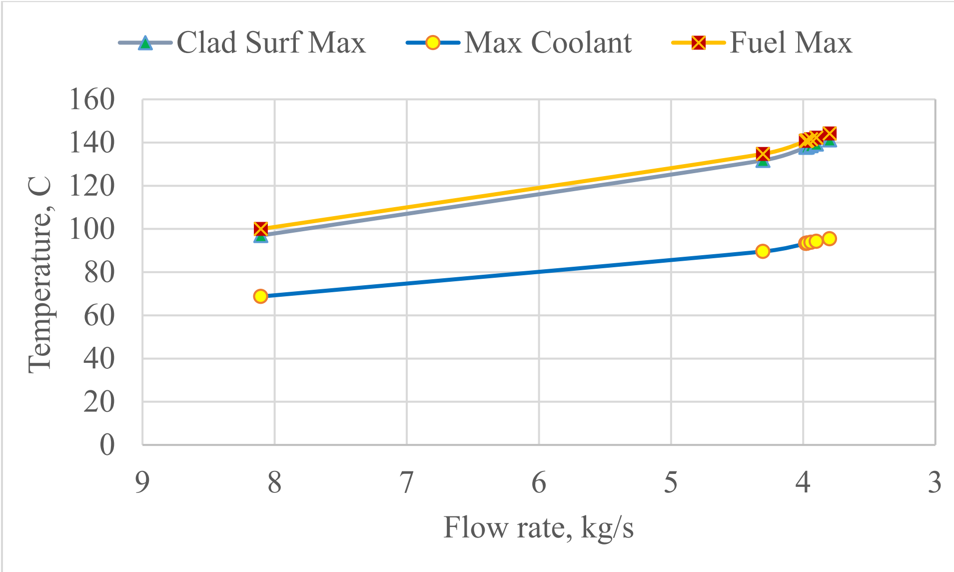


Figure 9: Fuel, maximum cladding surface, and coolant temperatures variation with flow rate (core power = 9.7 MW).

Table 1: Results for reducing the flow rate at the hottest channel for the maximum power level (9.7MW) and core pressure drop (ΔP) of 0.066 MPa.

Flow rate, kg/s	Min ONBR	Min DNBR	Min FIR	Maximum Cladding Surfurface Temperature, °C	Maximum Coolant Temperature, °C	Heat Flux, MW/m^2	Maximum Fuel Temperature, °C
8.1045	1.559	4.147	2.034	96.971	68.712	8.33E-01	99.949
4.3045	0.924	1.804	1.083	131.73	89.537	8.33E-01	134.708
3.98	0.863	1.538	1.001	137.765	93.257	8.49E-01	140.743
3.976	0.863	1.535	1	137.838	93.302	8.48E-01	140.817
3.974	0.862	1.533	0.999	137.874	93.323	8.48E-01	140.853
3.972	0.862	1.532	0.999	137.912	93.347	8.48E-01	140.89
3.97	0.862	1.53	0.998	137.95	93.372	8.48E-01	140.929
3.94	0.856	1.507	0.991	138.509	93.718	8.45E-01	141.488
3.9	0.849	1.476	0.982	139.272	94.194	8.42E-01	142.251
3.8	0.832	1.399	0.957	141.229	95.413	8.33E-01	144.208

It is clear that the minimum FIR is approached to 1 when the flow rate equals to 3.976 kg/s which is approximately half of the normal operating value. At this point, the safety margin value to flow instability is determined and it equals to 2.038. Obviously for this flow rate, the flow is unstable and the maximum value of the cladding surface temperature reaches to 137.8 °C which exceeds the allowable value (102°C). Of course, this situation must be avoided and operate the reactor at the safe conditions. However, it can be deduced that the safety margin to flow instability is good and operating the reactor with normal flow rate is safe.

It is concluded that the results are in agreement with the thermal-hydraulic safety requirements since the safety margin value to flow instability is 2.038. which means that the instability of the flow starts by reducing the flow rate to the half of the normal condition.

CONCLUSION

In summary, the PLTEMP code provides a computational method to assess the margin to flow instability for the IRT-4M fuel, confirming safe operation at specified power and flow rates, but it is not a transient analysis code that simulates the dynamic progression of an instability. The examination of the flow instability is crucial in nuclear reactors to prevent the occurrence of flow excursion throughout accidents. Instabilities can reduce the safety margins against these critical heat flux phenomena, where the fuel cladding surface temperature increases dramatically. Flow instability is investigated at the reactor of Tajoura Nuclear Research Centre which contains low enriched uranium (LEU) core, the recent reactor's fuel assembly type is IRT-4M.

In this paper, the flow instability is assessed at the hottest cell of the reactor by implementing PLTEMP/ANL code. PLTEMP is mainly a steady-state code, which calculates temperature distributions, heat fluxes, and flow rates. In addition, it is used to establish safe operating parameters and design margins for instance, onset of nucleate boiling ratio (ONBR), departure from nucleate boiling ratio (DNBR), and the flow instability ratio (FIR),

The safety margin to flow instability is evaluated at the maximum power level (9.7 MW) by running the code at several flow rates to find the point where FIR equals 1. The calculations have been carried out for the hottest cell (0.0805 MW), the inlet coolant temperature of 45°C, the flow rate is 8.1 kg/s, the core pressure drop is 0.066 MPa, and the inlet pressure is 0.179 MPa. The results indicate that the maximum cladding surface temperature is 96.97°C which is smaller than the maximum permissible value of the cladding surface temperature (102°C), the minimum FIR equals to 2.034, the minimum ONBR is 1.559, and the minimum DNBR is 4.147. By reducing the flow rate and running the code, it is determined that FIR equals to 1.0 at flow rate of 3.976 kg/s at the hottest channel. Which means that flow instability starts by reducing the flow rate to almost half of its original value. At this flow rate, the flow is unstable and the maximum cladding surface temperature is 137.8 °C which exceeds the permissible value (102°C). Accordingly, this situation must be avoided and the reactor must be operated at safe conditions. The safety margin value to flow instability is determined to be 2.038. It is obvious that the flow is stable when the reactor operates at its nominal power of 9.7 MW with the flow rate of three pumps and the reactor is safe. Therefore, it is concluded that the results are reasonable and in good agreement with the thermal-hydraulic safety requirements.

DECLARATION OF CONFLICTING INTERESTS

I declare that there are no potential conflicts of interest with respect to the research, authorship, and/or publication of this article.

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DECLARATION OF GENERATIVE AI AND AI-ASSISTED TECHNOLOGIES IN THE WRITING PROCESS

The author did not use generative AI or AI-assisted technologies in the preparation of this manuscript.

LIST OF SYMBOLS

A_F	Channel flow area, m^2
C_p	Specific heat capacity at constant pressure, J/kg-K
D_H	The heated diameter of the channel,
K	Thermal conductivity, W/m-K
L_H	The heated length,
L	Heated length of channel, m
L_{nvg}	Non-boiling length, m
L_{nvg}/L	The dimensionless non-boiling length
$(L_{nvg}/L)_{critical}$	Critical value of the dimensionless non-boiling length
N_{sub}	The subcooling number for the channel
N_{zu}	The Zuber number, $\frac{q''_w\, \zeta_H\, L}{\rho_{in} V_{in}\, A_F\, h_{fg}} \frac{(\rho_{f,nvg}-\rho_{g,nvg})}{\rho_{g,nvg}}$
P	Coolant absolute pressure, bar
Pe	Peclet number = $\rho_{in}\, C_p V_{in}\, D_H / k$
q''_w	Wall heat flux, W/m^2
U	Coolant velocity, m/s
V_{in}	Inlet coolant velocity, m/s
η	The flow instability factor,
ζ_H	Channel heated perimeter, m
ρ_{in}	Inlet coolant density, kg/m^3
ΔT_{sub}	Coolant subcooling at the axial node of CHF, °C

LIST OF ABBREVIATIONS

ANL	Argonne National Laboratory	LEU	Low Enriched Uranium
CHF	Critical Heat Flux	MCNP	Monte Carlo N-Particle
DNBR	Departure from Nucleate Boiling Ratio	MTR	Material Test Reactor
FER	Flow Excursion Ratio	PLTEMP	Plate Temperature
FIR	Flow Instability Ratio	REBUS-PC	REactor BUrnup System using PC's under WINDOWS or linux
HEU	High Enriched Uranium	RELAP	Reactor Excursion and Leak Analysis Program
ONBR	Onset of Nucleate Boiling Ratio	TNRC	Tajoura Nuclear Research Centre
IRT-4M	High-density, low-enriched fuel	Nvg	Position of net vapor generation

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