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Thermal-hydraulic Simulation of a WWR-S Reactor for Reactivity Insertion Accident

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ABSTRACT

In order to conduct a deterministic safety assessment of a WWR-S research reactor, a thermal-hydraulic transient model by RELAP5 code is developed and used to simulate the Reactivity Insertion Accident (RIA). The reactor power is 2 MW with downward flow direction and different types of fuel bundles with different power densities and different coolant flow-rates. The model is verified by comparing its results with TPRTHA program results for steady-state normal operation. RIA is simulated for four different scenarios including inadvertent withdrawal of a control rod at full power operation regime, inadvertent withdrawal of a control rod at start-up operation regime, control rod ejection accident at full power operation regime and control rod ejection accident at start-up operation regime. Except the last scenario, the reactor is shutting down safely in the first three scenarios without any damage to the fuel bundles integrity. In the fourth scenario where a control rod is ejected at start-up operation regime, both the fuel and clad are melted for the hot channel calculations. The model results for the four scenarios are analyzed and discussed

1. INTRODUCTION

Reactivity Insertion Accidents (RIA) are critical events during nuclear reactor operations, characterized by an unanticipated increase in reactivity that results in a sudden spike in power output. Due to the significance of RIA in assessing reactor safety, many researchers have concentrated their studies on this type of accident to understand reactor behavior and safety implications such as the following studies, where it has been demonstrated across different reactors and various scenarios. Where, Dawahra et al. [1] have demonstrated the inherent safety features of the Miniature Neutron Source Reactor (MNSR) by simulating various of reactivity insertion scenarios and analyzing the resulting behavior of reactor parameters, including power, fuel temperature, and coolant temperature. Also, PARK et al. [2] presented an initial accident analysis for a LOFA and RIAs of a conceptual 10 MW MRR (multi-purpose research reactor) under design study using the RELAP5/MOD3 code. The purpose of their analysis was to provide a preliminary evaluation of the safety margins and offer design insights for the MRR. However, Hamidouche et al. [3] focused on evaluating a model integrated into a computer code, specifically examining the neutron kinetics model within the widely used RELAP5/Mod3 code. The evaluation was based on positive reactivity insertion transients, considering calculations that include thermal-hydraulic feedback and transients without feedback effects. investigated reactivity insertion accidents at the 20-MW D2O-moderated research reactor (NBSR) at the National Institute of Standards and Technology (NIST). The analysis covered both highly enriched uranium fuel and a proposed equilibrium core using low-enriched uranium fuel. Where, Hossain et al. [5] Conducted a series of experiments at the 3 MW BAEC TRIGA Mark-II Research Reactor to examine the effects of positive and negative step reactivity insertions. They measured and analyzed the reactor power and fuel temperature responses. Additionally, the reactor period was calculated, and comparisons were made to evaluate how the initial power level influences the reactor period. Chatzidakis and Ikonomopoulos [6] proposed a systematic approach utilizing the RELAP5/MOD3 code to conduct a comprehensive reactivity insertion analysis in research reactors. The primary goal was to introduce a methodical process for determining the maximum reactivity insertion in a research reactor facility. Khater et al. [7] developed a dynamic model for the thermal-hydraulic analysis of an MTR research reactor during a reactivity insertion accident. This model was formulated to couple reactor

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kinetics with feedback reactivity and the reactor core's thermal hydraulics. It was employed to simulate the uncontrolled withdrawal of a control plate in the ETRR-2 reactor, analyzing both scenarios: a transient with an overpower scram trip and a self-limited transient. Also, Fengrui et al. [8] simulated the performance of plate-type fuel during a RIA under various burnup conditions using a 0D point kinetic model combined with a 1D coolant model and a 3D fuel coupling scheme integrated into the multi-physics code BEEs-Plates. The coupling scheme and RIA results were verified against the MTR benchmark, showing that both mechanical and burnup effects have minimal impact on neutronic–thermal hydraulics outcomes when using constant thermal properties. At last, a thermal-mechanical analysis was conducted for the fuel plate at different burnup levels during the RIA. In this study, the RELAP5 code is utilized to simulate the thermal-hydraulic behavior of a WWR-S type research reactor during a reactivity insertion accident. RELAP5 (Reactor Excursion and Leak Analysis Program) is an advanced computer code developed by the Idaho National Laboratory (INL). It is extensively used in the nuclear industry and research institutions to analyze the thermal-hydraulic behavior of nuclear reactors under various operational and accident conditions.

2. REACTOR DESCRIPTION

The reactor under study operates at a power of 2 MW, utilizing light water as both a coolant and moderator. The fuel used in this reactor is uranium dioxide, housed in EK-10 type fuel rods with aluminum cladding. The layout of the reactor core is depicted in Fig.1, where positions 1 through 6 are occupied by fuel bundles, and position 7 contains empty bundles. To regulate the coolant flow through the core cooling channels, throttle nozzles are installed in the lower grid of the reactor core [9].

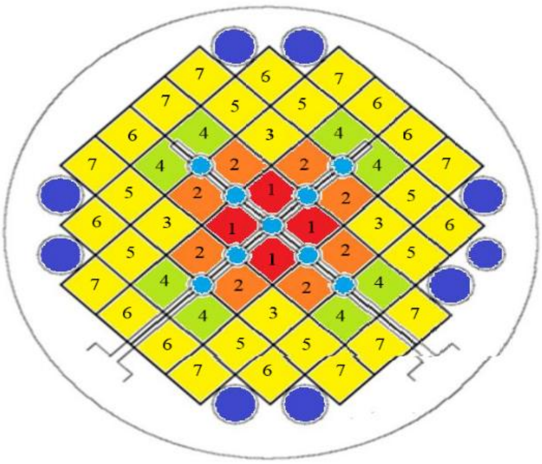

Fig. (1): Reactor core configuration

The core is loaded with four distinct types of fuel bundles, each varying in cross-sectional shape depending on its position within the core. Specifically, bundles in groups 3, 5, and 6 have a square cross-section, group 4 bundles have one cut corner, group 2 bundles have two cut corners, and group 1 bundles have three cut corners. Figure 2 provides a schematic representation of a fuel bundle along with the four types of bundle groups. Table 1 lists the parameters of the reactor core that were used in the RELAP5 model for simulating the thermal-hydraulic behavior of the reactor under the accident [9][10].

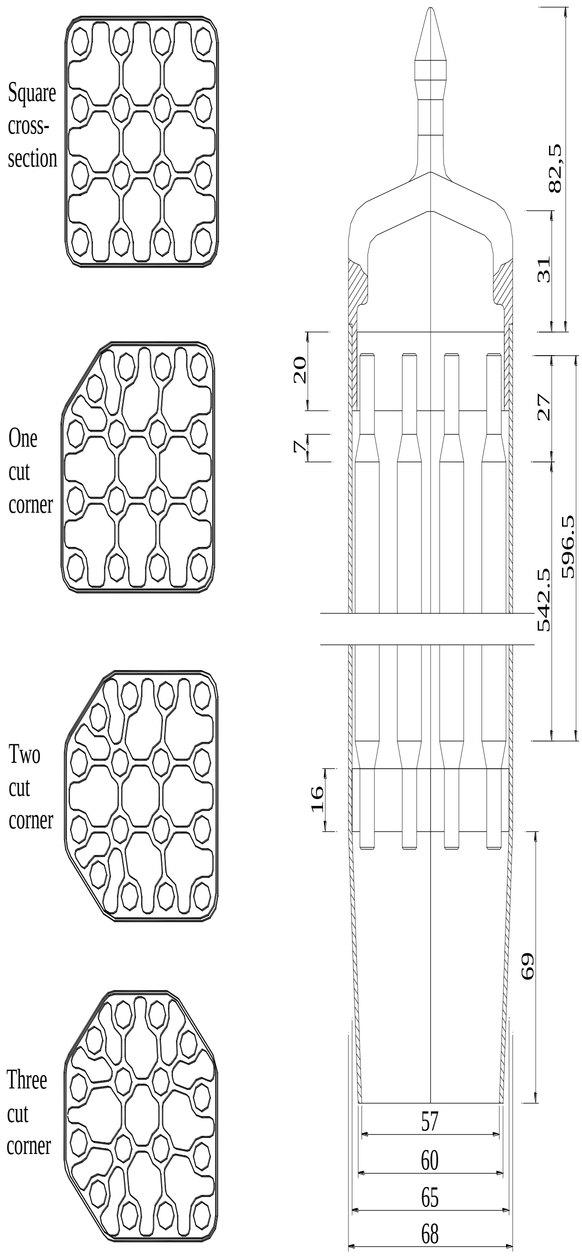

Fig. (2): Fuel bundles scheme [11].

Table (1): Reactor core specifications [11].

Design thermal power, MW	2
Water temperature at core inlet, °C	34
Core inlet pressure, bar	1.5
Nominal core flow kg/sec	240
Number of fuel bundles	41
Number of fuel bundles in the startup period	36
Number of fuel rods per fuel bundle	16
Reactor equivalent radius, cm	24.3
Radial peaking factor	1.54
Axial peaking factor	1.3
Active length, cm	50.0
Fuel rod diameter, mm	10
Clad thickness, mm	1.5
Extrapolated addendum, cm	7.0
Clad thermal conductivity, W/m°C	170
Fuel thermal conductivity, W/m°C	25
Clad specific heat, J/kg°C	900
Fuel specific heat, J/kg°C	234.5
Clad density, kg/m^3	2700
Fuel density, kg/m^3	5775

3. METHODOLOGY AND VERIFICATION

3.1 Model description

The thermal-hydraulic behaviour of the reactor during normal operation and under a reactivity insertion accident was simulated using the RELAP5 code. The simulation employed a straightforward model encompassing the reactor core, pool, and both upper and lower plenums, as shown in Fig. 3. This model is deemed sufficient provided the boundary conditions are applied appropriately. The RELAP5 model categorizes the reactor core into six different fuel bundle group types based on their proximity to the core centre, as illustrated in Fig.1; with bundle group type 7 being empty. The hottest rod is represented as part of a fuel bundle from group type 1. Coolant channels within the bundles are segmented into control volumes connected by junctions, and the fuel rods are divided into radial nodes [13].

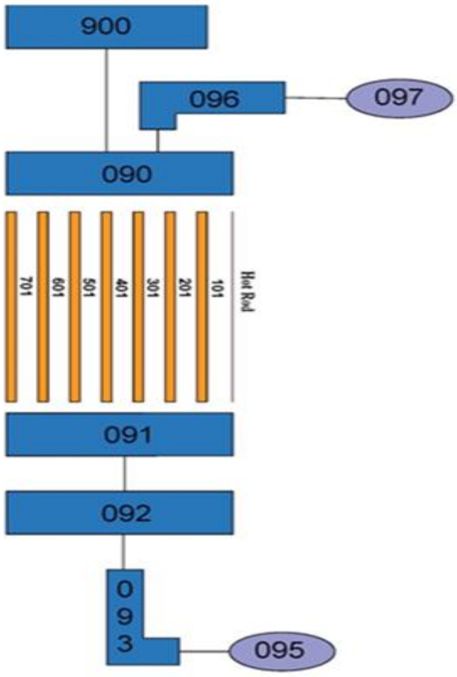

Fig. (3): System nodalizations for RELAP code[13].

3.2 Heat Structures

The fission power P_i generated in the fuel bundles of group type i at a radial distance ri from the reactor core center is given by [11]

$$P_i = \frac{P}{N_{bs}} \times PPR \times J_o\left(\frac{2.405\, r_i}{R+\delta}\right) \qquad \textbf{(1)}$$

Where: i represent the fuel bundle group types (1, 2, 3, 4, 5, and 6), P is the reactor power level, and N_{bs} denotes the number of fuel bundles during the startup period. Additionally, PPR is the radial power peaking factor, J_0 is the Bessel function of zero order, R represents the radius of the reactor core, and δ is the extrapolated addendum. The axial heat flux distribution along the core is modeled as a chopped cosine shape. [11]

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

And the average heat flux for fuel bundles group *i* is [11]

$$\phi_0\Big|_a = \frac{P_i}{2\, N_r\, d L_p \sin\left(\frac{\pi H)}{2\, H_p}\right)} \qquad \textbf{(3)}$$

The heat flux along the hottest rod

$$\phi_0\big|_h = PPT \times P_1 / N_r \, \pi \, d \, H \quad (4)$$

N_r is the number of fuel rods per bundle, H is the core active length, H_p is the extrapolated length, $H_P = H + 2e$, and e is is the extrapolated distance. PPT is the core total power peaking factor = $PPR \times PPA$, where PPA is the axial peaking factor.

3.3 Reactor Kinetics

In this study, the point kinetics solution is used to determine the reactor's power level during transients with RELAP5, instead of relying on a power table. Unlike the power table, which provides predefined power levels over time, the point kinetics method dynamically calculates the reactor power by accounting for factors such as the moderator void fraction coefficient, fuel temperature coefficient, and reactor kinetics parameters, as shows in Tables 2, 3, and 4, respectively. This approach offers a more accurate representation of the reactor's behavior by considering real-time changes in reactivity during transient conditions.

Table (2): Moderator density coefficient

Moderator density, kg/m^3	Reactivity, \$
495.0	-49.120
740.0	-14.868
955.0	0.100079

Table (3): Fuel temperature coefficient

Temperature, K	Reactivity, \$
300.0	0.292
343.0	0.100079
400.0	-0.011783
500.0	-0.289227

Table (4): Reactor kinetic parameters

Group	precursor yield ration	decay constant (s^{-1})
1	0.032307692	0.0124
2	0.218461538	0.0305
3	0.196923077	0.111
4	0.395384615	0.301
5	0.115384615	0.140
6	0.041538462	3.010

3.4 Model Verification

Model verification involves ensuring that a computational model accurately represents the conceptual model and its underlying mathematical and physical principles. It is a critical step to confirm that the model is correctly implemented and produces reliable results [12]. One verification method, known as code-to-code verification, compares the results of one computational model with those produced by another independently developed model to assess accuracy and reliability. In this study, the results from the RELAP5 code for the reactor's steady-state operations were compared with those from the TPRTHA code [11], focusing on the reactor parameters listed in Table 1. The first six groups are fueled, while the seventh group contains empty rods; therefore, the comparison is conducted for the six fueled group bundles. Table 5 provides a comparison of the predicted maximum temperature values from the RELAP5 code with the TPRTHA code under steady-state normal operation. The RELAP5 results demonstrate strong consistency. Additionally, Table 5 shows that the maximum coolant bulk temperature in all core coolant channels is well below the water saturation temperature as shows in Fig. 4.

Table (5): Maximum temperature values during normal operation [13].

Temperature		H.R	Group1	Group2	Group3	Group4	Group5	Group6
Coolant-bulk	TPRTHA	-----	37.2	37.1	36.6	36.7	36.2	35.2
	RELAP5	-----	37.4	37.1	36.6	36.7	36.2	35.2
Clad-surface	TPRTHA	65.5	56.6	56.8	54.6	56.2	51.9	44.0
	RELAP5	65.2	56.0	56.1	54.0	55.5	51.4	43.7
Fuel-center	TPRTHA	137.5	107.3	99.9	90.8	85.7	75.8	57.3
	RELAP5	140.8	108.2	100.7	91.5	86.1	76.1	57.4

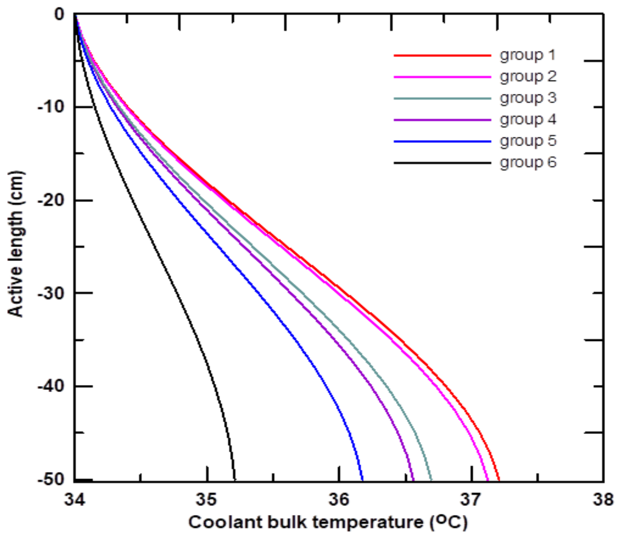

Fig. (4): Coolant bulk temperature profile for the core coolant channels.

Also the RELAP5 analysis shows that the maximum clad-surface temperature of 65.2°C is significantly lower than the onset of nucleate boiling temperature of 111.4°C at a reactor pressure of 1.5 bars as shown in table 5. This large temperature margin ensures that the coolant remains in a subcooled state, with efficient heat removal from the fuel rods and no risk of nucleate boiling or departure from nucleate boiling (DNB). The coolant-bulk temperatures, ranging from 35.2°C to 37.4°C, further confirm that the reactor operates well within safe thermal limits, maintaining stable and effective cooling of the core. The axial temperature distribution in the clad-surface temperature and fuel-center temperature for the six fuel-bundle groups and the hottest fuel rod is shows in Fig.5 and Fig.6 respectively

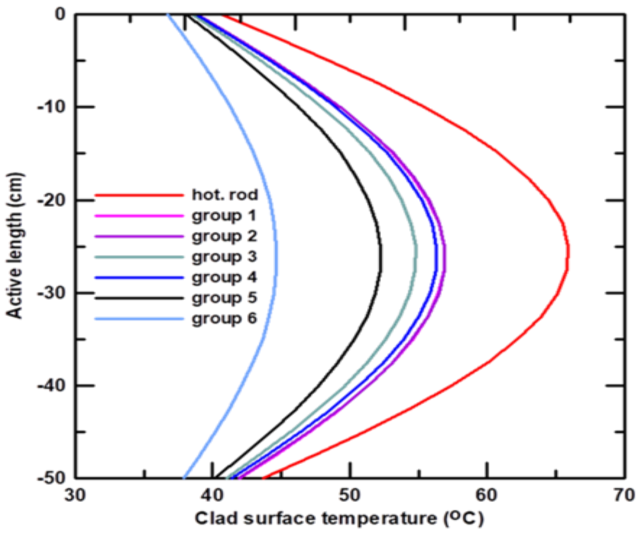

Fig. (5): Axial Temperature Distribution of Clad-surface during Steady-State Operation

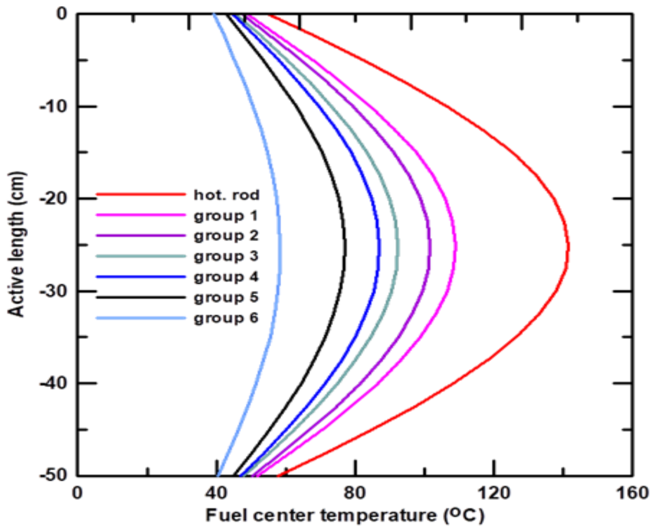

Fig. (6): Axial Temperature Distribution of Fuel-Center during Steady-State Operation

After validating the RELAP5 model to ensure its reliability and accuracy, the simulation of the transient operation was conducted to analyze the reactor's behavior under such conditions. This involved initiating the RIA simulation to observe and quantify the thermal-hydraulic responses of the WWR-S type research reactor during the event. The model was used to simulate the reactor core's response to four different initiating events that could lead to an RIA. These events include the inadvertent withdrawal of a control rod at full power, inadvertent withdrawal of a control rod during the start-up operation regime, a control rod ejection accident at full power, and a control rod ejection accident during the start-up operation regime. In all scenarios, the reactor is assumed to be tripped by a high reactor power signal at 2.4 MW, with a trip delay time of 0.2 seconds.

4. RESULTS AND DISCUSSION

The reactor core is divided into seven bundle groups, as illustrated in Figure 1. The first six groups contain fuel, with each bundle consisting of 16 fuel rods, while the seventh group contains empty rods. The power generated per bundle at a reactor power level of 2 MW is calculated and shown in Table 6, where the bundles closer to the core center exhibit higher power output. The mass fluxes for each group, calculated using the RELAP5 code at a nominal core flow rate of 240 kg/s, are also presented in Table 6. These results indicate that higher coolant flow rates are directed toward the central bundles, which are equipped with wider nozzles to handle the increased flow.

Table (6): Mass flux and Power distribution through the bundles group types.

Group type	Number of bundles	Power (kW) per bundle	Mass flux (Kg/m^{2}s)
1	4	81.84	.104
2	8	69.643	3.38
3	4	58.497	2.89
4	8	47.788	1.8
5	8	38.553	1.67
6	9	21.461	1.67
7	10	0.0	1.67

4.1 Inadvertent withdrawal of a control rod at full power operation regime

In this case; the reactor is operating at full power condition with nominal core coolant flow rate and a control rod is assumed to be withdrawn leading to a reactivity insertion of 1.0$ in 2 s. The reactor power variation and reactivity behavior during the transient are shown in Fig.5. The high power trip signal of 2.4MW is generated upon insertion of 0.244 $ at 1.49 s. Due to a trip scram signal delay of 0.2 s, the reactor power reaches 2.69 MW as maximum value at 1.69 s before sharply decreases due to the insertion of 5 $ negative shutdown reactivity. The net reactivity behavior as result well as the initiating event, the shutdown and feedback reactivity are shown in Fig .7 where the net reactivity reached a maximum value of 0.2763$ at 1.68 s, then decreased sharply due to negative reactivity insertion to reach a minimum value of -4.7 $ directly after the scram. Then the net reactivity is increased gradually to reach a steady state value of -3.917$. Figure 8 depicts the coolant, clad-surface and fuel-centerline temperature behaviors at the hot channel during the transient. The maximum clad-surface is 75.5°C which much lower than the onset of nucleate boiling temperature and the fuel-centerline temperature is 178.8°C which is much lower than the fuel melting temperature.

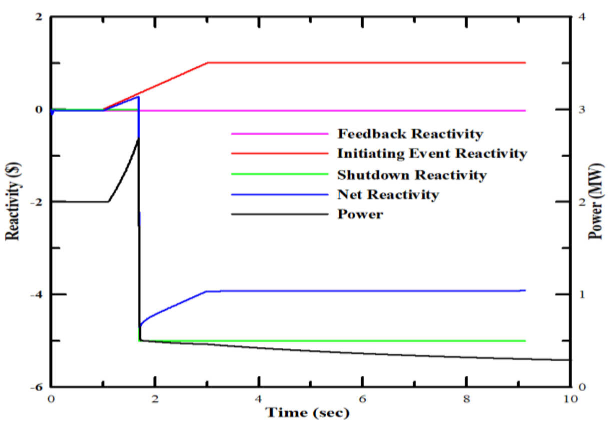

Fig. (7): Reactivity and power variations for inadvertent withdrawal of a control rod at full power

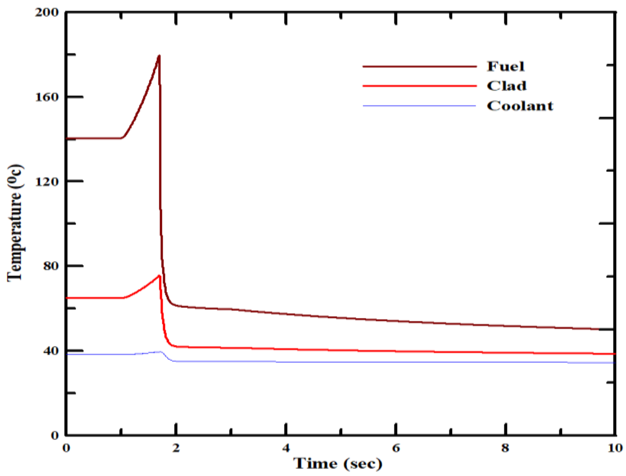

Fig. (8): Coolant, clad-surface and fuel-centerline temperature variations at the hot channel for inadvertent withdrawal of a control rod at full power

4.2 Inadvertent withdrawal of a control rod at start-up operation regime

In this case; the reactor is reaching 1 % full power under start-up operation regime with nominal core coolant flow and a control rod is assumed to be withdrawn resulting in a reactivity insertion of 2.5$ in 4 s. Figure 9 presents the variations of the reactor core reactivity and power during an adverse withdrawal of a control rod event. As the net reactivity reaches 1.21$, the reactor power reaches 2.4 MW leading to a power trip signal. However, due to actual scram delay of 0.2s, the reactor power continues increasing up to 7.4 MW at 3.15s, then decreases sharply as scram is affected by inserting 5 $ negative reactivity due to the insertion of the control rods. The net reactivity reached a maximum value of 1.1092 $ at 2.92 s then decreased gradually as a result of feedback effect before decreasing sharply due to scram negative reactivity insertion to reach a minimum value of -3.9919 $. Then it increases gradually after scram to reach a steady state value of -2.5022 $.

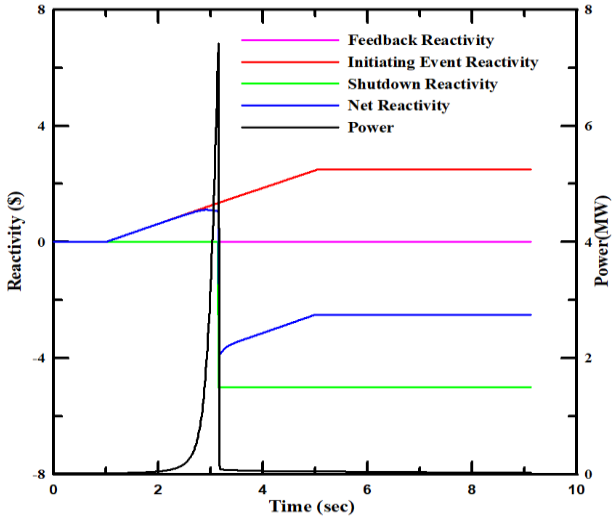

Fig. (9): Reactivity and power variations for inadvertent withdrawal of a control rod at start-up

The coolant, cld-surface and fuel-centerline temperature behaviors at the hot channel are shown in Fig. 10. It shows that, the maximum clad surface exceeds the onset of nucleate boiling temperature for about 0.5 s but the coolant temperature still subcooled by a vast margin where the maximum coolant temperature was 44.6°C and so some vapor bubbles may predicted at the clad-surface for about 0.5 s but there is no chance for bulk boiling. On the other hand, the maximum fuel centerline temperature predicted is about 400°C which is still much lower than the fuel melting temperature (2200°C).

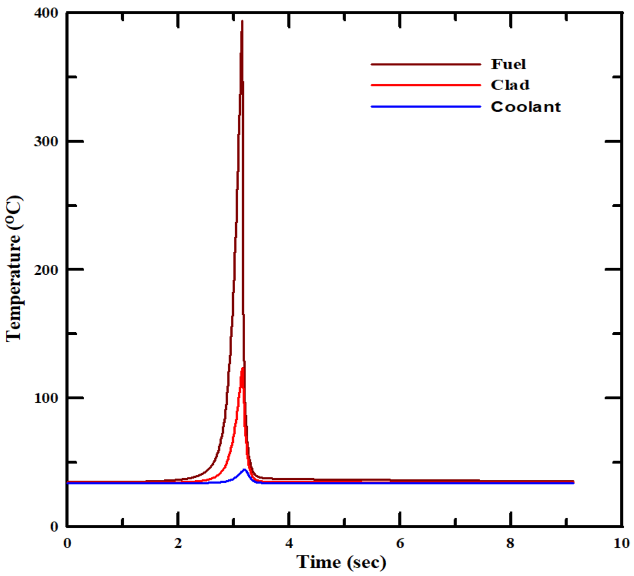

Fig. (10): Coolant, clad-surface and fuel-centerline temperature variations at the hot channel for inadvertent withdrawal of a control rod at start-up

4.3 Control rod ejection accident at full power operation regime

In this case; the reactor is operating at full power operating conditions and a step reactivity of 1.0 $ is instantly inserted in the core due to control rod ejection. Figure 11 presents the variation of the reactor core reactivity and power. As soon as the step reactivity inserted, the reactor power jumped to exceed the safety limit 2.4 MW at 1.01s leading to a high power trip signal. However, the reactor power increases up to 8.56 MW at .116 s, then decreases gradually due to feedback reactivity to reach 8.36 MW just before scram where 5 $ negative reactivity is inserted to shut down the reactor at 1.21 s. The maximum and minimum values predicted for the net reactivity are 0.976$ and -3.92$ respectively. The coolant, clad-surface and fuel-centerline temperature behaviors during transient are shown in Fig.12 where the maximum clad temperature value exceeds the onset of nucleate boiling temperature for about 0.5 s but the coolant temperature still subcooled by a vast margin where the maximum coolant temperature was 50.3°C and so some vapor bubbles may predicted at the clad-surface for about 0.5 s but there is no chance for bulk boiling. On the other hand, the maximum fuel centerline temperature predicted is 473.9℃ which is still much lower than the fuel melting temperature.

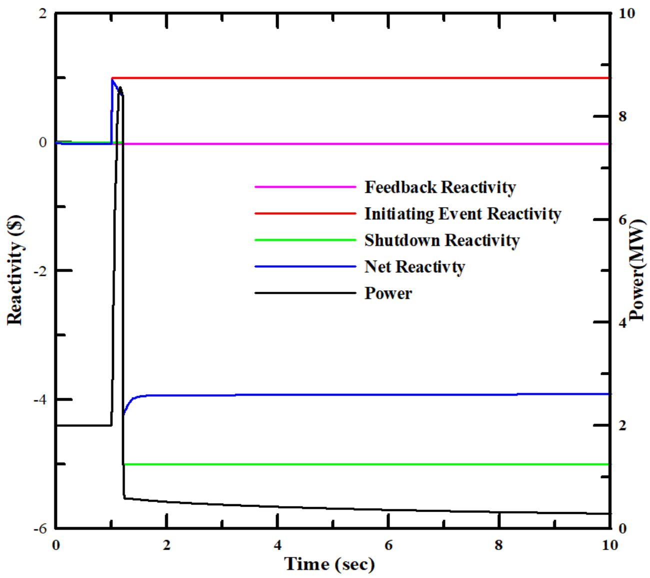

Fig. (11): Reactivity and power variations for control rod ejection at full power

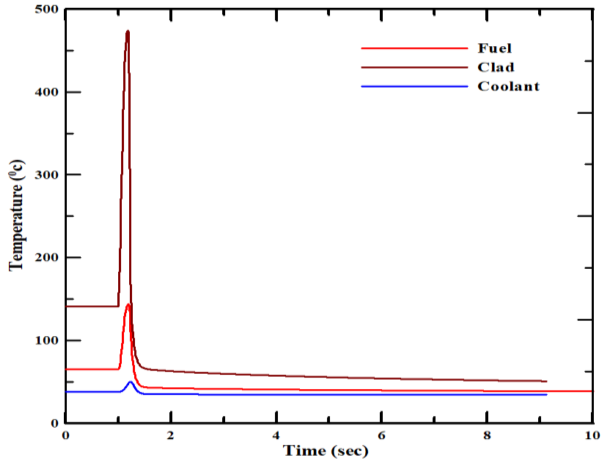

Fig. (12): Coolant, clad-surface and fuel-centerline temperature variations at the hot channel for control rod ejection at full power

4.4 Control rod ejection accident at start-up operation regime

This last case is for the reactor power reaching 1 % full power under start-up operation regime with nominal core coolant flow and a control rod is ejected leading to a step reactivity of 2.5 $. The reactor power variation during the transient shows in Fig. 11 where it shows that, the power level in this accident reached 90.0 MW in 0.14 s. The increase in the power value is much faster than being controlled by the safety systems that need a time period of 0.2 s to affect scram shows in Fig. 13. This leads to a rapid huge amount of fission energy in the core resulting in a sharp increase in the clad and fuel temperatures. Therefore; the clad-surface temperature exceeded the melting point in 0.145 seconds, and the fuel temperature exceeded the melting point in 0.155 seconds while the coolant temperature was still below 43°C as shown in Fig. 14. Table 7 details the temperature variations of the coolant, clad-surface, and fuel-centerline across the core coolant channels during the control rod ejection at start-up. The data shows that the fuel has melted in all bundle types except bundle group type 1, and the cladding has also melted in all bundle types except for types 4, 5, and 6.

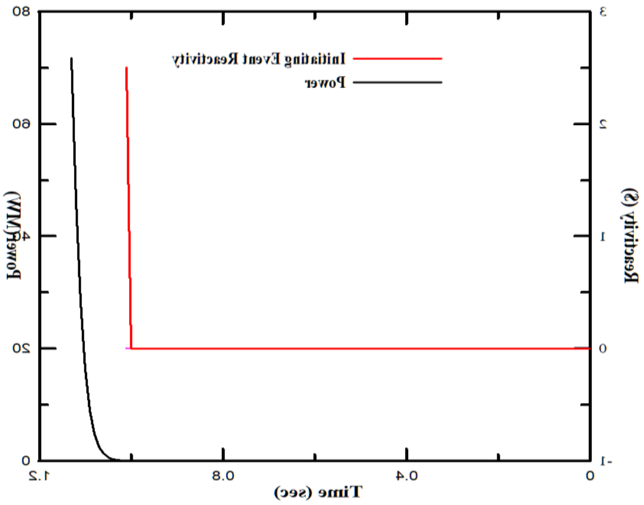

Fig. (13): Reactivity and power variations for control rod ejection at start-up

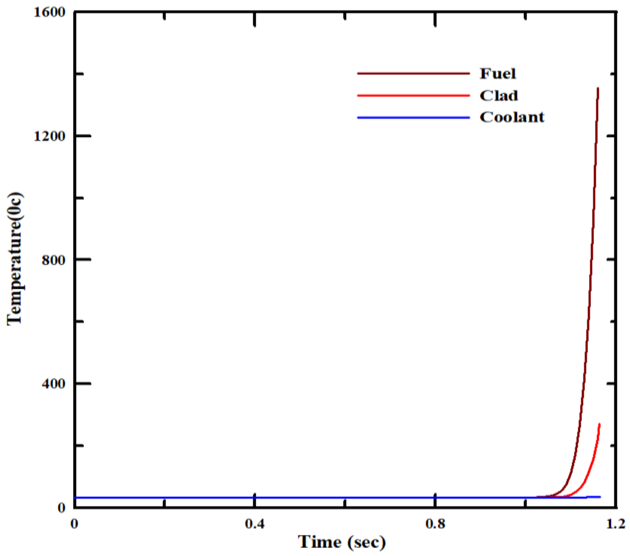

Fig. (14): Coolant, clad-surface and fuel-centerline temperature variations at the hot channel for control rod ejection at start-up

Table (7): Coolant, clad-surface and fuel-centerline temperature (°C) variations in the core during the control rod ejection at start-up

Temperature / Coolant Ch.	Coolant temp.(°C)	Clad-surface temp. (°C)	Fuel-center temp. (°C)
Hot channel	42.85	1355.496	7308.30
Bundles group 1	41.04	932.41	5088.84
Bundles group 2	39.45	798.44	4395.48
Bundles group 3	39.87	671.26	3663.35
Bundles group 4	38.85	544.37	2990.89
Bundles group 5	37.92	430.44	2408.58
Bundles group 6	35.76	224.95	1351.79

5. CONCLUSION

A thermal-hydraulic model is developed by RELAP5/Mod3.3 to conduct a safety assessment for a WWR-S type research reactor. The model’s accuracy was first verified by comparing its results with those from the TPRTHA code under steady-state normal operation conditions. Following verification, the model was used to simulate a Reactivity Insertion Accident (RIA) under four different scenarios, each with varying initial reactivity values and conditions. These four scenarios are inadvertent withdrawal of a control rod at full power operation regime, inadvertent withdrawal of a control rod at start-up operation regime, control rod ejection accident at full power operation regime and control rod ejection accident at start-up operation regime. It has been observed that in the first three cases, despite a delay in the response time to safety limits where the power increased by 120% of the specified level the reactor managed to shut down in time, thereby preserving the integrity of the fuel bundles. However, in the final scenario, both the fuel and cladding reached their melting points across the majority of the core's coolant channels before the reactor's safety systems were able to respond effectively.

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