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Neutronics and thermal hydraulic analysis of TRIGA Mark II reactor using MCNPX and COOLOD-N2 computer code

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Abstract. The neutronic analysis of TRIGA Mark II reactor has been performed. A detailed model of the reactor core was conducted including standard fuel elements, fuel follower control rods, and irradiation devices. As the approach to safety nuclear design are based on determining the criticality (k_{eff}), reactivity worth, reactivity excess, hot rod power factor and power peaking of the reactor, the MCNPX code had been used to calculate the nuclear parameters for different core configuration designs. The thermal-hydraulic model has been developed using COOLOD-N2 for steady state, using the nuclear parameters and power distribution results from MCNPX calculation. The objective of the thermal-hydraulic model is to determine the thermal safety margin and to ensure that the fuel integrity is maintained during steady state as well as during abnormal condition at full power. The hot channel fuel centerline temperature, fuel surface temperature, cladding surface temperature, the departure from nucleate boiling (DNB) and DNB ratio were determined. The good agreement between experimental data and simulation concerning reactor criticality proves the reliability of the methodology of analysis from neutronic and thermal hydraulic perspective.

1. Introduction

The research reactor (TRIGA- Mark III) at Thailand Institute of Nuclear Technology (TINT) referred to as TRIGA Research Reactor Modified I (TRR-1/M1) is design for steady state and square wave operation up to a power level of 2 MW(th). The three-dimensional model of the TRR-1/M1 reactor was performed using the Monte Carlo code MCNPX.[1] The validation is an essential aspect of developing an accurate reactor physics model. Several studies in neutronic analysis, criticality experiment and MCNP simulation has been conducted for TRIGA Mark II reactor.[2, 3] The integral neutronics parameters, such as core excess reactivity, reactivity worth and integral reactivity curves of the control rods, and criticality value (k_{eff}), were calculated for the TRR-1/M1 and compared with the measurement. The objective of the thermal and hydrodynamic design is to safely remove the heat generated in the fuel without producing stream void formations, excessive fuel temperatures and without approaching the critical heat flux under steady-state operating conditions. The steady-state thermal hydraulic and safety analysis of TRR-1/M1 is performed to ensure that all important thermal hydraulic parameters are within the safety margin according to the operational limit and condition of the reactor. Boulaich Y. et al. provided calculations and the experiments related to the steady state thermal hydraulic analysis of Moroccan TRIGA Mark II reactor by using PARET/ANL and COOLOD-N2 codes.[4] The TRR-1/M1 fuel centerline temperature, cladding temperature, the coolant temperature, the departure from nucleate boiling (DNB) and DNB ratio (DNBR) were calculated with

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the thermal hydraulic codes COOLOD-N2[5] based on the axial power factor distributions obtained by Monte Carlo code MCNPX with the ENDF/B-VI nuclear data libraries. The validation of the calculation was compared with the TRR-1/M1 experimental and operational data for steady-state operations.

2. Computational Methods

2.1. MCNPX Computer Code and Experiments
The TRR-1/M1 lattice was modelled as a hexagonal prism, solids with eight faces (Figure 1). The fuel elements is homogeneous mixture of uranium-zirconium hydride alloy with the uranium-to-zirconium atom ratio of 1.6 to 1.7. The uranium in the uranium-zirconium hydride mixture is enriched to approximately 20% U-235. There are two types of fuel elements loaded in TRR-1/M1 core including 8.5% uranium by weight type and 20% uranium by weight type. The 20 wt.% fuel element is a mixture of uranium-erbium-zirconium-hydride (UErZrH) alloy containing approximately 0.5 wt.% erbium. In this study the core configuration consists of 20 wt.% fuel elements are loaded in the D ring, other locations in the core are loaded with 8.5 wt.% fuel elements, total of 105 fuel elements including fuel follower control rod (FFCRs), 3 locations of neutron detectors (position) and 12 locations of in-core irradiation facilities (including pneumatic transfer system) for in-core utilization

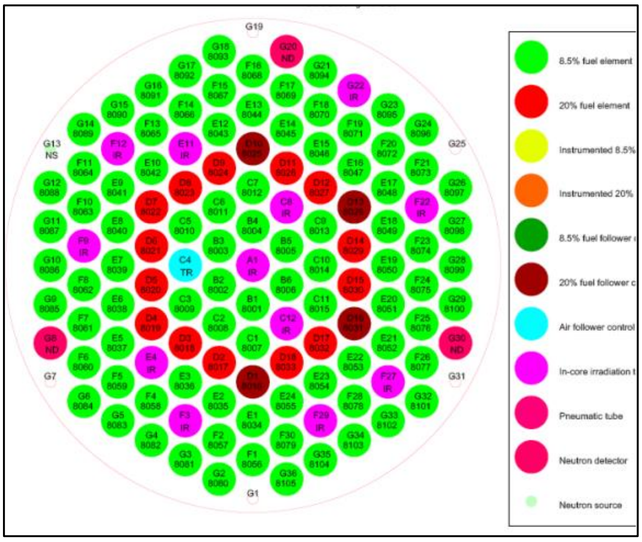
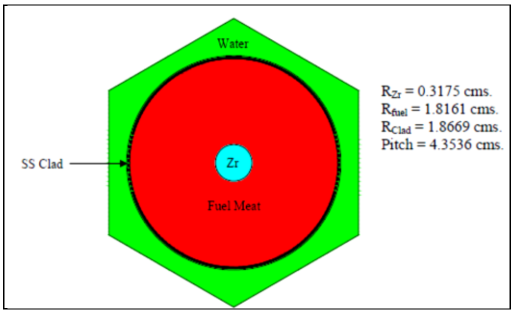


Figure 1. Fuels and FFCRs with 20 wt%. and 8.5wt%. fuel elements.

The fuel element is approximately 3.73 cm in diameter and 73.15 cm in overall length and the active part of the fuel element is 38.1 cm long. The power level of the reactor is controlled with five control rods, a safety rod, a regulating rod, two shim rods and a safety-transient rod. The regulating, shim, and safety rods are sealed 304 stainless steel tubes approximately 109 cm long by 3.43 cm in diameter in which the uppermost 16.5 cm section is an air void and the next 38.1 cm is the neutron absorber (boron carbide in solid form). The fuel elements figures are presented in Figure 2a and 2b.



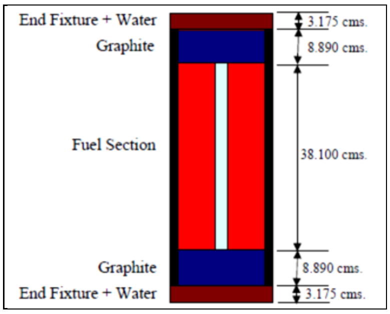


Figure 2a. Fuel element (Horizontal Cross Section) **Figure 2b**. Fuel element (Vertical Cross Section)

In the MCNPX criticality calculation, it simulates a number (N_0) of neutron from their birth at fission to their death by escape of absorption. After that, a number of (N_1) of fission neutrons are selected for simulation in the next cycle, where each particle is weighted by (N_0/N_1).[1] The multiplication factor can be expressed from the k-static Boltzmann equation.[1] The k_{eff} values can be converted into reactivity values using the following equation:

$$\rho = \frac{\frac{k_{eff}-1}{k_{eff}}}{\beta_{eff}} \qquad (1)$$

Where ρ is the reactivity value in units of dollar ($) and β_{eff} is the fraction of effective delayed neutrons (β_{eff} = 0.007 for TRIGA fuel types).[5] The calculation of the control rods reactivity worth simulated explicitly the experiment, which was conducted by the positive period method.[6] The simulation with the control in the critical position calculating the k_{eff0} of the core. Then one of the control rods was withdrawn at a certain position, calculating the new k_{eff}. The control rod worth represented by reactivity ρ for that position was determine by comparing the k_{eff} and k_{eff0} as explained in Eq. 2.[7]

$$\rho = \rho_0 - \rho_1 = \left(1 - \frac{1}{k_{eff0}}\right) - \left(1 - \frac{1}{k_{eff}}\right) = \frac{1}{k_{eff}} - \frac{1}{k_{eff0}} \qquad (2)$$

The MCNPX model is used for the calculation of power peaking parameters including hot rod power peaking factor f_{HR}, axial power peaking factor f_Z and radial power peaking factor f_R. These three factors are important for steady state operation; they determine the maximum total power released by one fuel elements as well as its axial and radial peaking values, which are used as parameters in the thermal hydraulic calculation. The hot rod power peaking factor is defined as the ratio between the maximum power released by one fuel rod P_{rod} and the average power per element in the core $\bar{P}_{core}$

$$f_{HR} = \frac{(P_{Rod})_{max}}{\bar{P}\ Core} \qquad (3) \quad and\ \bar{P}\ Core = \frac{P}{N_{EL}} \qquad (4)$$

Where P is the total power, which is 1.2 MW for TRR-1/M1 and the N_{EL} is the number of fuel elements which is 105 fuel elements (100 fuel elements +5 fuel followers control rods).

2.2. Rod Reactivity Worth and the Criticality Experiments

The control rods reactivity worth experiment was measured by the positive period method.[6] The reactor is made critical at 15 Watts. The test rod is withdrawn a small distance so that the reactor is slightly supercritical and the power starts to increase. The reactor period is determined from the doubling time. The previous procedure is repeated until the rod test has been calibrated along its whole length. From the observed periods the corresponding reactivity are calculated using the inhour equation. Using the rod calibration curve from each control rod, the remaining reactivity value Δρ to the fully withdrawn rod position can be determined.

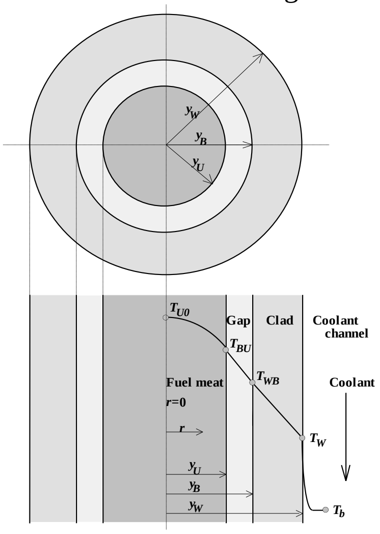
The criticality experiment were conducted based on inverse-multiplication method consisting of loading fuel elements in steps, measuring the count rates on the pulse mode detectors, after each step and plotting 1/M as a function of fuel mass or number of elements. M is the subcritical multiplication as shown in Eq. 5 and 6. Criticality is approached by adding the fuel rods in the core. Inverse multiplication rate is measured. Plot of inverse multiplication rate versus fuel mass is made after each fuel loading step, and the critical mass is estimated by extrapolating the curve to zero.

$$M = \frac{total\ thermal\ neutron\ flux\ (source + fission)}{thermal\ neutron\ flux\ due\ to\ source} \quad (5) \qquad M = \frac{1}{1 - k_{eff}} \quad (6)$$

2.3. COOLOD-N2 Computer Model

The COOLOD-N2 computer code[8] is used in conjunction with the MCNPX to perform the thermal-hydraulic analysis of the steady-state operation of the TRR-1/M1. In the COOLOD-N2 code, a Heat Transfer Package[9] is used for calculate the heat transfer coefficient, DNB heat flux. In the rod type fuel, DNB heat flux is calculated by both the Heat Transfer package and Bernath DNB heat flux correlation.[10]

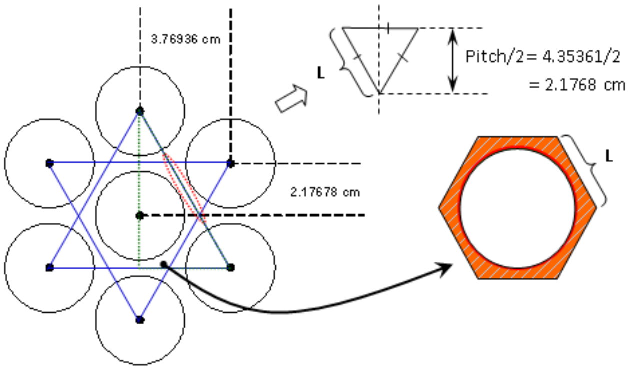
The goal of the thermal and hydraulic design for TRR-1/M1 is to ensure that fuel integrity is maintained during steady-state as well as during those abnormal conditions which might be postulated for reactor operation. During the steady-state operation, fuel integrity is maintained by limiting reactor powers to the values which assure that the heat transfer rate from the fuel meat to the cladding, to the reactor coolant is fast enough such that clad rupture will not occur. In order to assure the fuel element integrity, the maximum temperature in the fuel meat is limited to 600°C which is the designed limit in the operational limits and condition (OLC) for TRR-1/M1 adopted from the fuel temp trip set point from the manufacture.[10] The thermal-hydraulic model of the TRR-1/M1 fuel rod consists of 3 layers: fuel meat (Uranium Zirconium Hydride: UZrH), gap (assumed to be He gas), and cladding (Stainless steel). Heat transfers out radially from the center of the fuel meat through the gap, cladding, and the water coolant, respectively. Figure 3 shows the calculation model used in the code for fuel temperature distribution in the rod. These parameters in Fig. 3 are used to calculate max fuel temperature using 1D heat conduction with heat generation through multiple layers of material.



y_w – distance from rod center to the outer surface of the fuel cladding
y_b – distance from the rod center to the outer surface of the gap
y_u – distance from the rod center to the outer surface of fuel meat
T_{u0} – Temperature at the center of the fuel rod
T_{BU} – Tempeature at the point between fuel meat and gap
T_{WB} – Temperature at the point between gap and inner cladding wall
T_W – Temperature of the cladding wall surface
T_b – Coolant temperature

Figure 3. Fuel rod temperature calculation model

The COOLOD-N2 model for the flow channels between fuel rods of TRR-1/M1 which are not in the form of circular tube, it is necessary to compute the hydraulic diameter in order to implement equations for circular pipe flow to evaluate the thermal hydraulics of the system of interest. Figure 4 is an example of the core flow channel cross section. The cross section of an assumed flow channel is shown as a shaded area. The flow area is the differential area between the area of a hexagonal and a circle (representing a fuel rod). The length L can be calculated from the equilateral triangle. Note that the diameter of the fuel rod is approximately 3.7338 cm.



Number of fuel elements in a core	101
Fuel pallet radius, cm	1.816
Cladding thickness, cm	0.051
Fuel element heated length, cm	38.1
Unheated length of fuel rod at channel inlet, cm	14.351
Unheated length of fuel rod at channel outlet, cm	8.89
Coolant mass flow rate, kg/sec	66.47
Inlet coolant temperature, °C	40.0
Inlet coolant pressure, kg/cm2	1.677
Effective flow area for one fuel rod, cm2	5.465
Fuel rod hydraulic diameter, mm	1.864
Pitch between fuel rods, cm	4.3536
Initial gap thickness between fuel pellet and cladding, cm	0.0023
Cladding thermal conductivity, W/cm-K	0.81 (at 100°C) 1.075 (at 500°C)
Fuel meat thermal conductivity, W/cm-K	0.1758 (at 0°C) 0.1833(at 100°C) 0.1908 (at 200°C) 0.1983 (at 300°C) 0.2133 (at 500°C) 0.2508 (at 1000°C) 0.3257 (at 2000°C)

Figure 4. Cross section view of the flow channel inside the TRR-1/M1 core

3. Results and Analysis

3.1. The criticality (k_{eff}) and core excess reactivity

Approach to criticality experiment was performed on the basis of inverse multiplication method. MCNPX is used to calculate the k_{eff} using KCODE and BURN card. MCNPX approximates k_{eff} by estimating the number of fission neutrons produced per fission neutron started for a given generation. By repeating this process for more than ten thousands of generations MCNPX arrives at a good approximation of a multiplication factor.

Table 1. Results of criticality measurements and calculations for burned core configurations

Core Number	K_{eff} (measured)	K_{eff} (MCNPX calculation)
1	1.00168 ± 0.00015	1.00196 ± 0.0404%
5	1.00185 ± 0.00015	1.00270 ± 0.0365%
7	1.00148 ± 0.00015	1.00225 ± 0.0309%
9	1.00212 ± 0.00015	1.00283 ± 0.0335%

The result shows that the discrepancy is very low for Core 1 since it is a fresh core and the discrepancy increases for the burned core. The calculated values are overestimated than the measurement values. The discrepancy in k_{eff} measurement and calculation may result from the decay time for the fission products poisoning, Xe poisoning, several fission products and actinides produced during the irradiation of the fuel which related to the isotopic compositions of burned fuels.

Figure 5 shows the MCNPX calculated integral reactivity curves for safety, shim I, shim II, and regulating rod, respectively. From these curves, it can be observed that the MCNPX calculated integral reactivity worth of shim I and shim II control rods are consistent with the measurement values. The measured reactivity worth of safety and regulating control rods are slightly below to the calculated results. However the small disagreement remain acceptable related to other TRIGA reactors.[7] In all control rods, the largest differential reactivity worth occurred when the rods position are fully withdrawn. The calculated and measured reactivity worth resulting from the withdrawal of the rods from 90% to 100% are less than 0.1$ and when withdrawal from 40% to 50%, the reactivity worth are less than 0.5$.

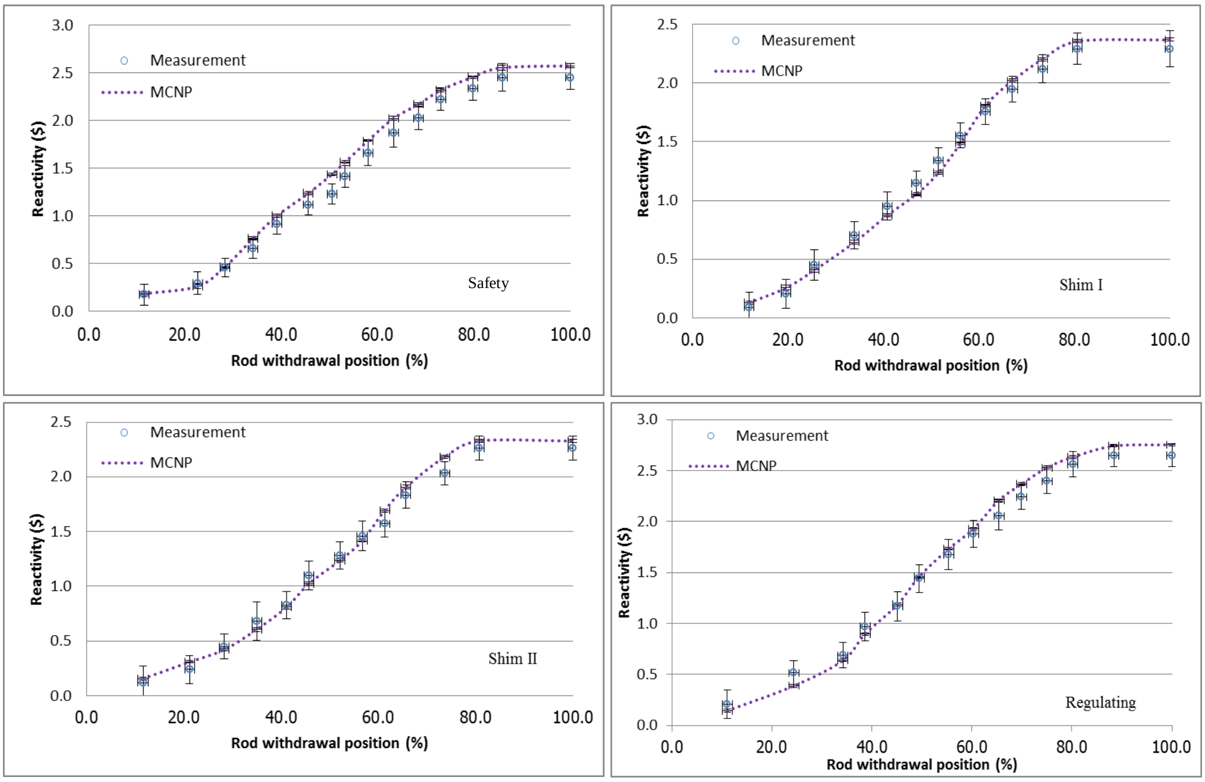


Figure 5. MCNPX calculations and experimental integral rod worth for safety, shim I, shim II and regulating.

3.2. The axial power peaking factor

The axial power distribution within the fuel meat of the hottest fuel elements calculated by MCNPX is shown in Figure 6. The figure shows the cosine shape of axial power distribution with a hot rod power factor (f_{HR}) of 1.894. The axial power peaking factor (f_Z) is 1.268. The value for f_{HR} x f_Z is 2.40. The small increments show in the left and right sides of the axial power profile are resulting from the lower and upper reflectors of TRIGA fuel. The results show that the core excess reactivity (%Δk/k) is 7.07. It is assumed that the power density is directly proportional to the fission density. The maximum power produced in the hottest fuel element is 23.4 kW.

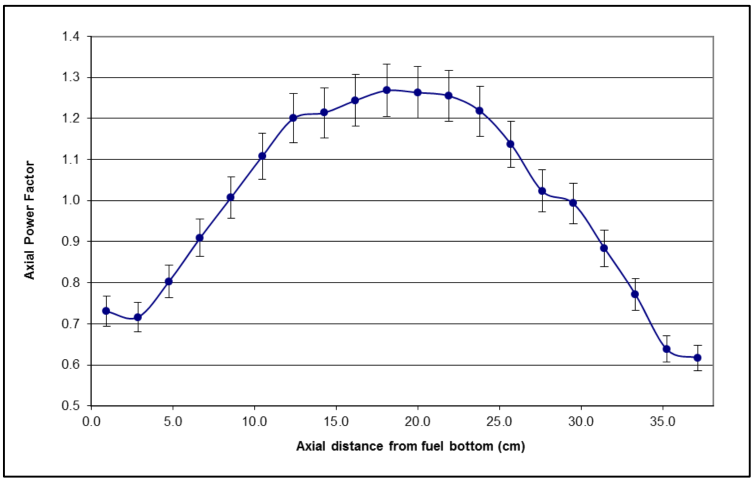


Figure 6. Hot channel fuel axial power factor profile.

3.3. The steady state thermal hydraulic analysis

Considered the case that all control rods are withdrawal at position 400, given the maximum f_{HR} x f_Z value, it is found that based on the temperature criteria, the maximum allowable power of TRR-1/M1 is 1.5 MW. Figure 7 and 8 show temperature distribution of the coolant and the fuel meat along the axial distance of the “hottest” location in the reference reactor core for the maximum allowable power of 1.5 MW (anticipated transient condition) and the normal operating power of 1.3 MW. T_{sat} is the saturated coolant temperature. Axial distance is the relative distance which the maximum is 38.1 cm.

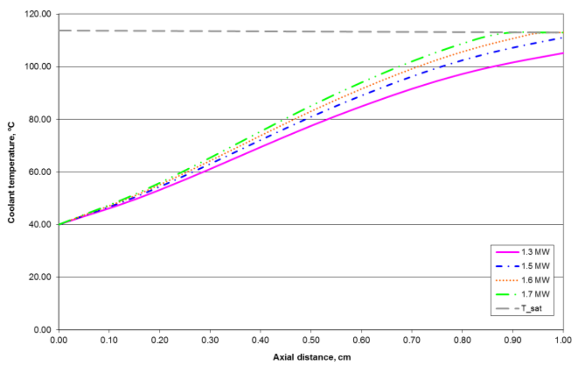


Figure 7. Coolant temperature profiles along axial direction

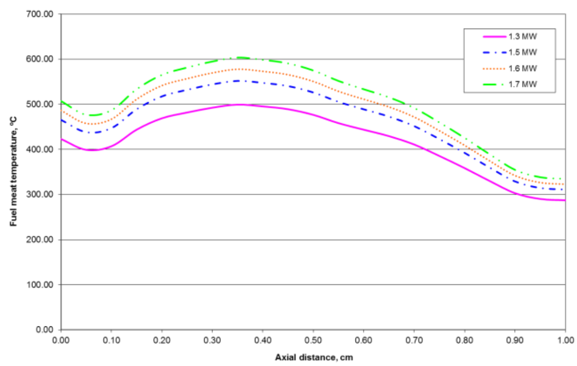


Figure 8. Fuel temperature profiles along the axial direction

From the results, at the maximum allowable power (1.5 MW), the peak of fuel meat temperature is around 552°C which is still below the designed maximum fuel temperature of 600°C. The calculated fuel temperature is far below the fuel swelling temperature limit of 950°C recommended by General Atomic.[11] The outlet coolant temperature is approximately 111°C while the saturated temperature at that point is 113°C. This result infers that there will be no bulk boiling of the water coolant in any flow channel of the reactor core.

3.3.1. DNB ratio (DNBR). The hot channel DNBR evaluated by COOLOD-N2 is depicted in Figure 9 for various operating power. If consider only DNBR, TRR-1/M1 can be operated up to 1.7 MW without breaking the DNBR design criteria which is 2.0. For the maximum allowable power of TRR-1/M1 at 1.5 MW, the maximum fuel temperature is 552°C and the minimum DNBR is 2.43 without bulk boiling in the core and allow approximately 15% thermal safety margin based on the nominal operating power of 1.3 MW. The SCRAM point for reactor power safety channels is set at 1.43 MW (110% of the nominal power) while, for the purpose of testing the reactor steady-state power level scram, the reactor power can be varied up to 1.5 MW.

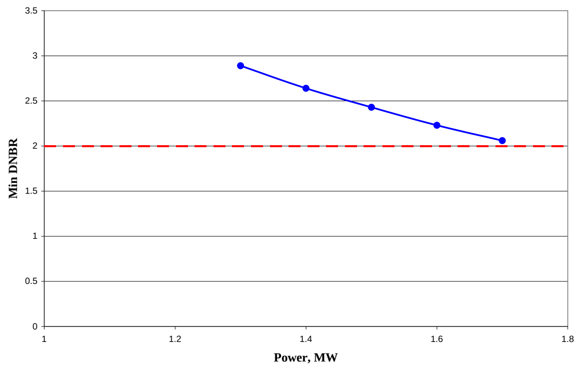


Figure 9. Minimum DNBR in at various power levels.

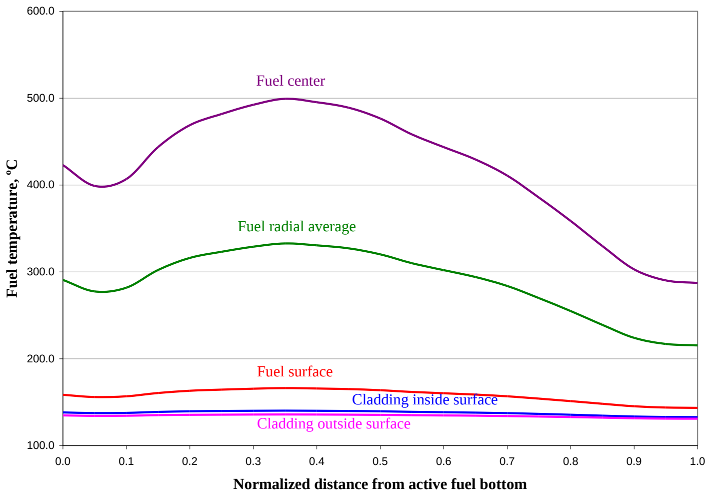
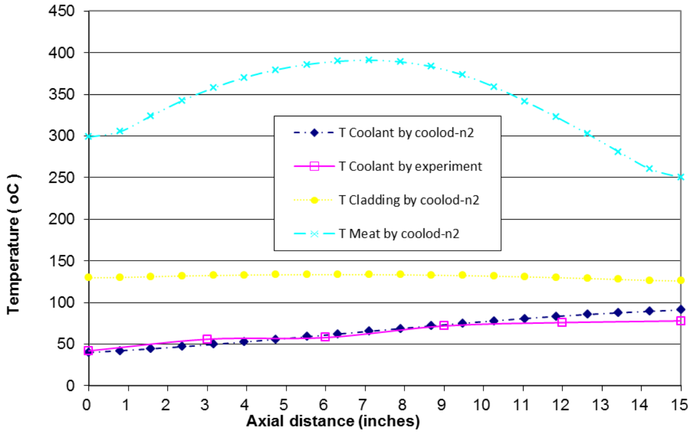


Figure 10. Temperature profiles along the axial direction of the hottest fuel element at 1.3 MW.

3.3.2. Axial temperature profiles. The comparison of axial temperature profile of TRR-1/M1 hot channel using COOLOD-N2 is presented in Fig. 10 with the highest power peaking factor (f_R x f_Z) is 2.40 and all control rods are withdrawal at 400 unit position. It can be seen from the figure that the maximum temperature of fuel centerline, fuel surface, and clad surface are observed at the center of the reactor core. The fuel surface temperature, clad surface temperature is found to be far below the fuel centerline temperature. As the fuel burns up, radiation induced swelling reduces the gap size. This phenomenon enhances the heat transfer at the fuel cladding interface, which will result in lowering the peak temperature at the centerline of the fuel rod. From the calculation, as shown in Fig. 10, the fuel element maximum temperature for steady-state operation at 1.3 MWt at BOL at the location of expected maximum power peaking factor is around 500°C, which is below the operating limit of 600ºC.

3.3.3. The validation of the COOLOD-N2 with the experiments. The results of the COOLOD-N2 were validated with the experimental results measured coolant at 1200 kW of flow channel between C1, C2 and D2 position shown in Fig. 11. The measurement was conducted using K-type thermocouple and temperature processing unit. It can be found that the temperature distribution obtained from the calculation and the experiment are in good agreement with minor discrepancies. With this validation, the thermal hydraulic model using COOLOD-N2 can be used to estimate safety parameter for reactor in a steady state with confidence in some extent.



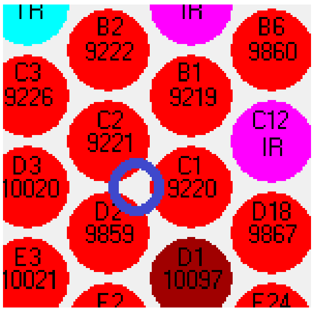


Figure 11. Comparison between the measured hot channel coolant temperature and the predicted values from the COOLOD-N2.

4. Conclusion

The TRR-1/M1 research reactor is model using MCNPX and COOLOD-N2. The practical calculation such as criticality value (k_{eff}), core excess reactivity, total and integral control rod worth and power peaking analysis were performed and compared with the experiments. Good agreement between calculations and experiments ensure the validity and reliability of the model. The validation of the COOLOD-N2 model was performed comparing their calculation results with the experiments. The thermal hydraulic analysis in steady state results shows good agreement between the calculations and the experiments. From this analysis, all safety related thermal hydraulic parameters are within the thermal design limits for steady state operating conditions. Further neutronic study should be conducted to be more consistency and improvement by reducing the discrepancies between the calculation and the measurements. For the thermal hydraulic analysis, the verification of reactivity feedback effects for several transient events should be conducted.

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