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Safety Analysis of MNSR Reactor during Reactivity Insertion Accident Using the Validated Code PARET

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Abstract

In the frame work of the IAEA's CRP project (J7.10.10) on "Safety significance of postulated initiating events for various types of research reactors and assessment of analytical tools" the Syrian team contributed in the assessment of computational codes related to the safety analysis of research reactors [1]. During the project implementation the codes PARET and has been tested, modified and verified regarding specific phenomena related to safety analysis of research reactors [2]. In the framework of this contribution the code PARET has been applied to model the core of the Syrian MNSR reactor. The code analysis includes the simulation of steady state operation and two selected reactivity insertion accidents (RIA) representing step reactivity insertion of 1 mk and the design basis accidents of inserting the maximum available excess reactivity of fresh and cold reactor core of about 3.6 mk. The calculation results ensures the high inherently safety features of MNSR as in all phases of the transients the operation limits have not been exceeded.

1. Description of MNSR

MNSR (Miniature Neutron Source Reactor) is a compact, thermal and low power research reactor. It adopts tank-in-pool type structure, fuel element of high-enriched uranium, and light water as moderator, coolant and shield. The core is surrounded by annulus, bottom and top beryllium reflectors that reduce the critical mass and provide neutron flux peak inside the reflectors where the irradiation sites are situated (Figure 1). The thermal neutron flux achieves, at the nominal power of 30 kW, a maximum of about $1 \cdot 10^{12}\, n.cm^{-2}.s^{-1}$ in the inner irradiation channels of the annular reflector. There is only one control rod located at the centre of the core to regulate power, compensate reactivity, and safely shut down the reactor. The reactor is designed by the Chinese Institute of Atomic Energy for the purposes of neutron activation analysis (NAA), some short-lived radioisotope preparation, and personnel training in nuclear technique applications [3].

The core consists of about 350 fuel rods that are cooled by natural convection. Due to the physical core design characterized by under-moderated system, a large negative temperature feedback coefficient of reactivity is achieved. Although this effect is of great benefit to the reactor safety, it causes in connection with the continuously rise of average core temperature, the rapidly consumption of the available excess reactivity and shortens thereafter the reactor operation time. The maximum continuous operation time amounts to 6 hours for the new core. Since the fuel management of MNSR considers only the replacement of the complete reactor core after its useful life, also after many years of operation, the maximum continuous operation time decreases continuously due to the permanent reduction of excess reactivity as results of fuel burn up. This time becomes insufficient even for some NAA applications. Thus, long-term reactivity control is exercised by periodically increasing the thickness of the top beryllium reflector to compensate for the reactivity loss caused by fuel burn up and samarium poison.

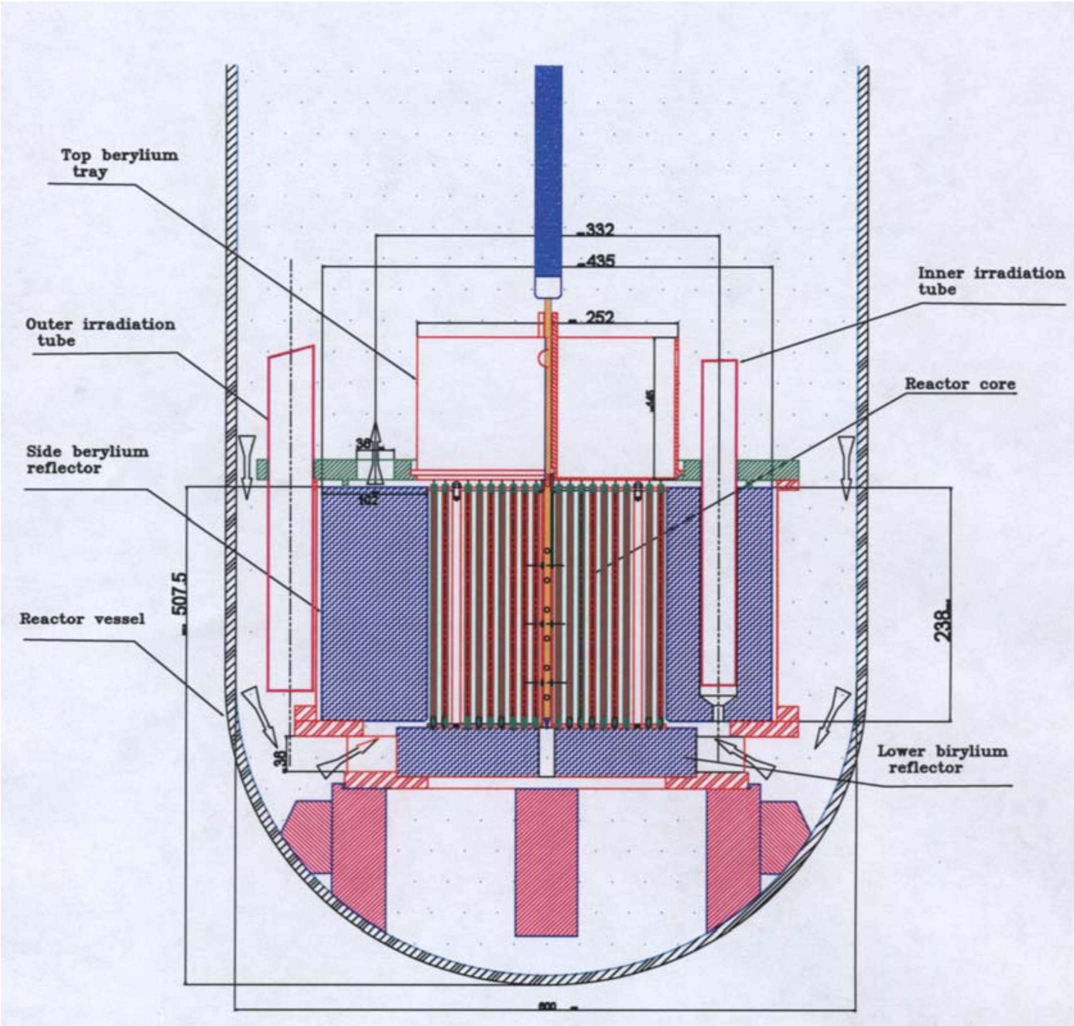

Fig.1. Schematic longitudinal section of MNSR vessel with fuel elements, control rod and beryllium reflector.

2. PARET Model of MNSR Reactor

The PARET input model for the core of MNSR reactor has been developed enabling the simulation of neutron kinetic and thermal hydraulic of reactor core including reactivity feedback effects. The neutron kinetic model depends on the point kinetic with 15 groups of delayed neutrons including photo neutrons of beryllium reflector. In this regard the effect of photo neutron on the dynamic behaviour has been analysed through two additional calculations. In the first the yield of photo neutrons was neglected completely and in the second its share was added to the sixth group of delayed neutrons. In the thermal hydraulic model the fuel elements with their cooling channels were distributed to 4 different groups with various radial power factors. The pressure loss factors for friction, flow direction change, expansion and contraction were estimated using suitable approaches (Figure 2)

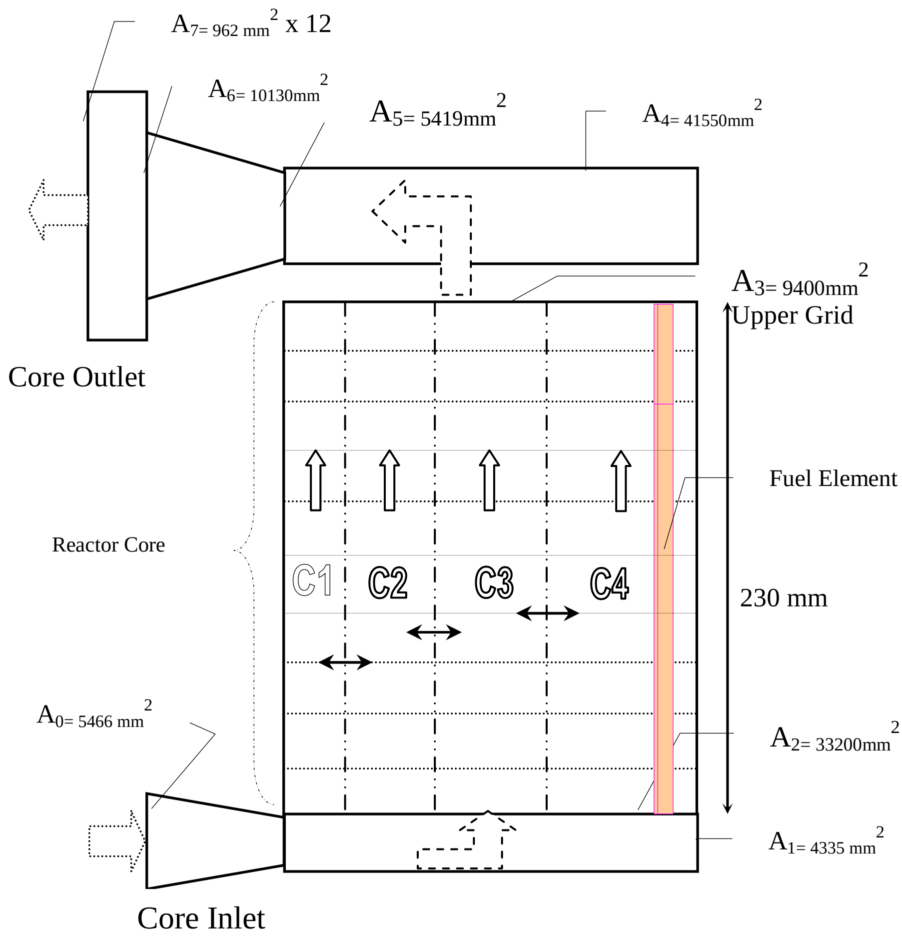

Fig.2. PARET Nodalization Model for the Core of MNSR reactor

3. Simulation of Step Reactivity Insertion

The simulation of reactivity transients supports the verification of the used model regarding the neutron dynamic behaviour and its coupling with the thermo hydraulic. For this purpose an experiment dealing with step insertion of positive reactivity of 1.07 mk has been post calculated [4]. The measurement was performed using calibrated cadmium capsule with a negative reactivity worth of 1.07 mk. The cadmium sample was inserted in one of the inner irradiation sites (i.e. in the annular beryllium reflector) of the reactor using the pneumatic rabbit system. The reactor was operated at low power level of 20% of nominal power (30 kW). The power recording was started and then the cadmium sample was ejected out of the reactor with continued data recording. Figure 3 presents the evolution of relative reactor power after the step reactivity change starting from 20% of reactor nominal power.

PARET simulation was carried out with the initial condition of 20 oC for the coolant temperature and 20% on nominal power. The negative temperature feedback coefficient $\alpha_T(mk/C)$ amounts to 0.14 mk/^{o}C according to experimental measurement on MNSR in the temperature range of 20-30 ^{o}C. However, the total average temperature coefficient of moderator is given to be - 0.13 mk/°C in the Safety Analysis Report [4]. The good agreement between calculation and measurement showing in Figure 3 indicates the qualification of PARET code to simulate reactivity insertion transients.

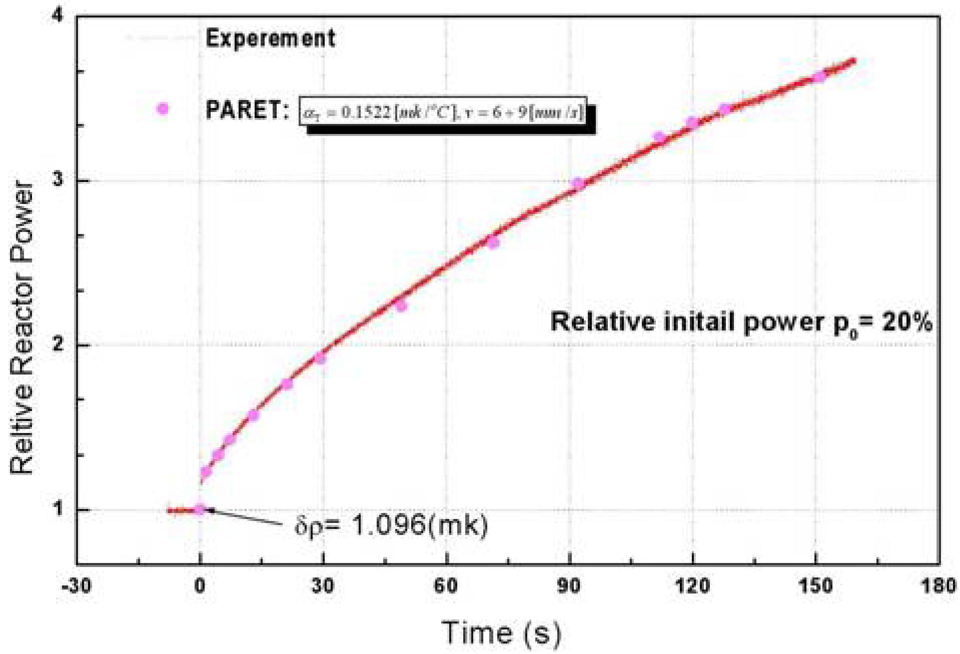

Fig.3. Relative power distribution after a step reactivity change starting from the initial power level of 20% of nominal reactor power.

4. Simulation of Reactivity Insertion Accident (RIA)

The reactivity insertion accident (RIA) simulated here is considered as design basis accident for MNSR. The simulation of this accident aims to demonstrate and insure the self-limiting power excursion feature in case of full cold excess reactivity insertion. The control and protection system of MNSR has been designed using fail-safe principle. If however, failure occurs which does not result in reactor shut down, the limiting consequence is total withdrawal of the control rod resulting in inserting the maximum available excess reactivity of fresh and cold reactor core of about 3.6 mk. This available excess reactivity is less than $1/2\beta_{eff}$ ensuring delayed criticality with high safety margin to prompt criticality.

Starting from the initial conditions of 19 ^{o}C for core temperature and 0.01% of nominal for the power, a ramp reactivity of 3.6 mk was inserted during 15 sec. The average temperature feedback coefficient of reactivity amounts to ca. 0.14 mk/^{o}C.

Figure 4 and Figure 5 show the development of calculated relative reactor power and average core temperature compared with the experimental data. The comparison depicts good agreement between measurements and calculation. Starting from the initial value the power increases to about 3.3 of nominal power (i.e. 99 kW). During this phase the average core temperature escalates to about 2.3 of its initial (i.e. 42 ^{o}C). This behaviour stimulates the the negative reactivity feedback effect that is able at this point to consume the entire available excess reactivity forcing the power to decrease and demonstrating the inherently safety features of MNSR reactor.

No measurement of clad or fuel temperature are possible at MNSR. Thus, the claculated temperatures are used to evaluate the consequences of the accident. During the accident the calculated maximum fuel and clad temperature of the central channel amount to 77.5 oC and 76 oC respectively. Thus, the maximum clad temperature is far from the saturation temperature of 113 oC, ensuring that no sub-cooled boiling can occur. Hence, the transient indicates that the accomplished power peak of about 100 kW doesn't affect the fuel elements, since no DNB is to be expected under these conditions.

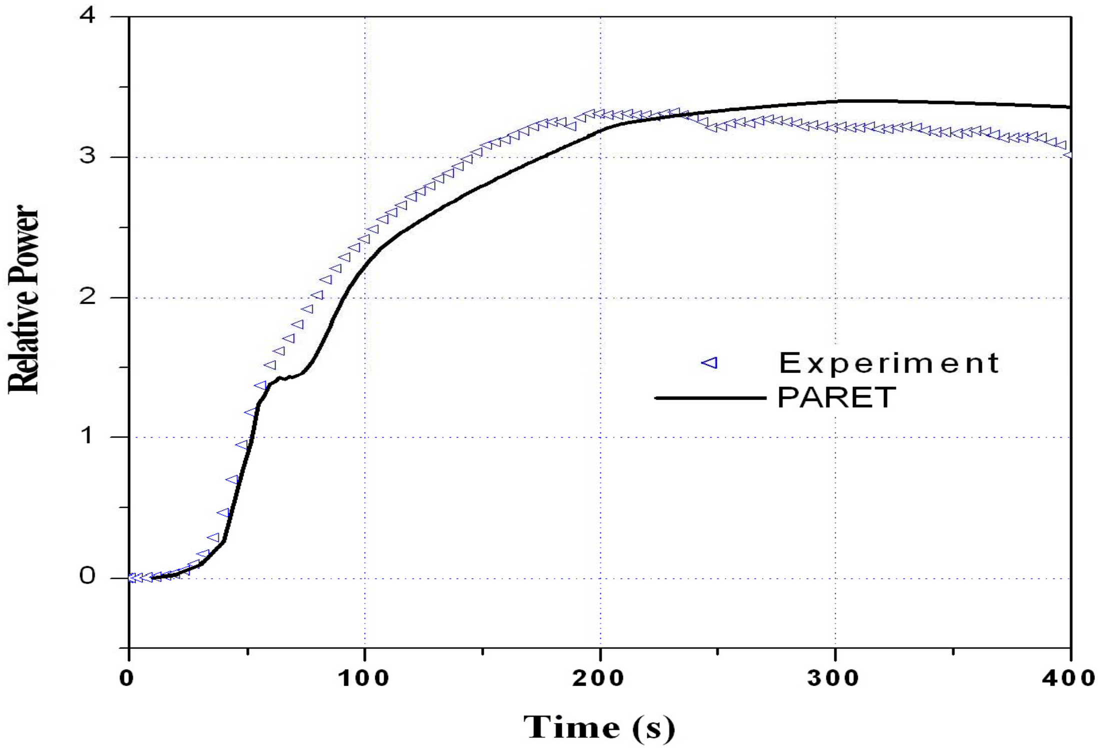

Fig.4.Evolution of relative reactor power following a complete withdraw of reactor control rode.

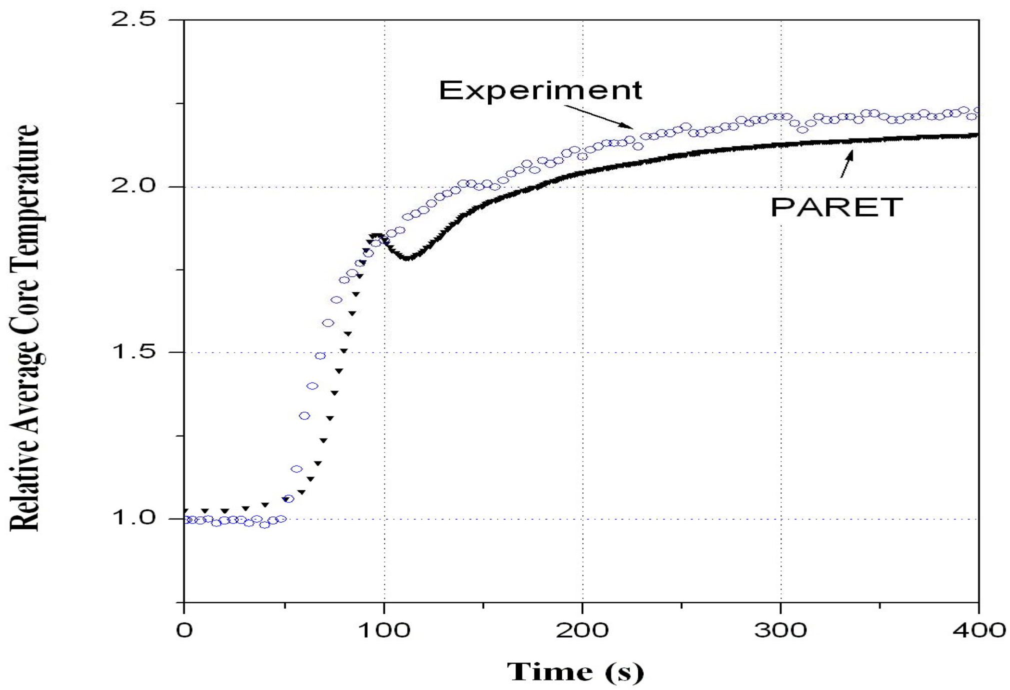

Fig.5.Evolution of relative average core temperature following a complete withdraw of reactor control rode.

5. Conclusion

The performed reactivity insertion transient shows that the code PARET possesses good ability to model the expected thermal hydraulic and neutron dynamic phenomena. The simulated design basis accident full control rod withdraw indicates the inherently safety features of MNSR reactor as the safety limits have been not exceeded resulting from the strong reactivity feedback effects of coolant temperature under natural circulation conditions.

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[2] Hainoun, A., Gazi, N., Alhabit, F. 2007. Modification and Validation of the Natural Heat Convection and Subcooled Void Formation Models in the Code PARET, Annals of Nuclear Energy.

[3] The Syrian MNSR Safety Report. 1992. internal report.

[4] Hainon, A., Khamis, I. 2000. Determination of Neutron Generation Time in Miniature Neutron Source Reactor by Measurement of Neutronics Transfer Function, Nuclear Engineering and Design.