Prediction of Peak Temperatures of Nigeria Research Reactor Core Components under Several Reactivity Accident Tests

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Abstract— The Nigeria Research Reactor-1 (NIRR-1) consists of small water cooled square cylindrical core of 23cm in diameter and 23cm high. The small dimension of the core of this reactor facilitated our choice of PARET to perform reactivity accident analysis for NIRR-1 system. Our goal in this work is to predict the peak temperature of some important Nigeria Research Reactor (NIRR-1) core components under several reactivity accident tests. At power levels below 80kW, there were no significant differences between the peak fuel centerline temperatures, the peak fuel surface temperature and the peak clad surface temperature in the hot channel as well as in the average channel. The result from the reactivity accident test shows that power can never rise to an uncontrollable level in the core of NIRR-1 under ramp or step insertion of up to 4mk of reactivity. The calculated temperature of the important core components (e.g. fuel and clad) in the two channels (during this reactivity accident test) were far below their melting point temperatures. Boiling of any kind was not observed during this reactivity accident test. Therefore, NIRR-1 can be operated safely even if there is an inadvertent addition of up to 4mk of positive reactivity

Keywords— Reactivity, Power, PARET, NIRR-1, Temperature, Transient, Accident, Core

1 Introduction

The Nigeria Research Reactor 1 (NIRR-1) was constructed by China Institute of Atomic Energy, specifically to serve as neutron source for research purpose, short-lived radio isotopes production and training of nuclear engineers/technician (Ibrahim et al, 2013 and Jonah et al, 2009). It is located at the Center for Energy Research and Training of Ahmadu Bello University Zaria. This reactor consists of a small water cooled square cylindrical core of 23cm in diameter and 23 cm high (FSAR, 2005 and Salawu, 2013). There are 347 cylindrical active fuel pins in the core of NIRR-1 with each pin containing 90% enriched U-Al alloy metal in aluminum cladding (FSAR, 2005). Detail description of this reactor has already been published elsewhere in several literature materials (Jonah et al, 2007, FSAR, 2005 and Salawu, 2013). The small dimension of the core of this reactor facilitated our choice of PARET to perform reactivity accident analysis for the present NIRR-1 system. PARET is a computer code designed for use in predicting the cause and consequences of reactivity accident in small reactor cores (Obenchain, 1969). It is widely applied for safety analysis of research reactors (Hainoun et al, 2008).

The loss of flow accident condition that normally occurred due to loss of coolant pumping power in reactors (Anthony, 2002) was not considered in this work because the Nigeria Research Reactor-1 (NIRR-1) is cooled by natural convection (FSAR, 2005). Our goal in this work is to predict the peak temperature of some important NIRR-1 core components in the hot and average channels under several reactivity accident tests. Such reactivity accidents can occur in research reactors that become inadvertently supercritical by addition of positive reactivity with a corresponding increase in power above the designed steady state power level of the system. Under this reactivity accident condition, the temperature of the fuel and or the cladding material in the core can rise up to the point of melting (Anthony, 2002).

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The three major categories of reactivity transients studied in this work include step change in reactivity within 0.1 and 0.2 seconds, a fast ramp insertion of reactivity at 0.6s and a slow ramp insertion of reactivity at the normal travel speed of the control rod in 26.4 seconds. The short period of time selected for the step change in reactivity was made in order to study the worst possible reactivity accident condition. If the system can survive such accident scenarios, it will be able to survive similar reactivity addition with much longer insertion time. Studying the worst possible reactivity accident scenarios is the best in ensuring system safety under normal and off normal operating conditions. A rapid insertion of reactivity in the system can occur due to operator error, electrical malfunctioning and failures or failure of the reactor component (Anthony, 2002). In addition, a ramp reactivity insertion can occur from the control rod withdrawal during system start up or a pressure build up within the core forcing the rod out of the core during normal operation or when the system is in a shutdown mode.

2 Methodology

The reactivity accident analysis for the present NIRR-1 system was performed in this work using the Argonne National Laboratory version of PARET code called PARET-ANL. The total active region of NIRR-1 core was divided into two regions (or channels), one representing region with the hottest pin and the other representing the rest of the reactor core, the average channel. The axial dimension of these channels was divided into 20 equally spaced regions called the nodes. The radial heat conduction in the fuel/clad region of the fuel pin was treated with 11 different nodes, six for the active fuel regions and four nodes for the clad.

Since NIRR-1 system has a cylindrical core with a cylindrical fuel pin (FSAR, 2005), a cylindrical geometry was selected for this computer code calculation. Reactivity specified computational mode was selected for the reactivity transient cases after performing series of power specified run to establish a profile of steady state flow versus power for the system. A number of externally inserted reactivity as a function of time was selected for a short/long transient time. There are no delay neutron groups in the power specified run as compared to the reactivity transient cases and the initial reactor power was specified in megawatt in all the cases run.

The total volume of the fuel was obtained by multiplying the active fuel volume per pin by the total number of active fuel pins in the NIRR-1 core. The core inlet operating pressure in natural convection was obtained using atmospheric pressure and the pressure due to the height of the water level above the core in the reactor vessel. The end sections of the fuel of about 9mm long that contain no active fuel were modeled along with the active fuel region. The effective delayed neutron fraction as well as the prompt neutron generation time for the system was obtained from literature material (Dunn et al, 2007, Bokhari and Pervez, 2010). The PARET code requires the reactivity to be provided in the unit of dollars and this conversion was achieved by simply dividing the reactivity by the effective delay neutron fraction. Note that 0.6 second was used for the fast reactivity transient, 1.2 seconds for the step change in reactivity and 26.4 seconds for the control rod withdrawal transients. The moderator inlet density at the initial reactor conditions was set to about 1kg/m3 (i.e. 997kg/m3) in all cases. Since a linear model was assumed in this work for solving the fuel temperature feedback equation by the PARET code, the single coefficient required for the calculation of reactivity feedback from the fuel temperature changes were obtained from literature material (Dunn et al, 2007). Over power trip point was set to 1.2MW. This was selected as the worst case scenario for an increase in power level in a typical Miniature Neutron Source Reactor (MNSR). This trip point controls the time at which the control rods begin to move into the reactor core due to high increase in reactor power. If the system is capable of withstanding power up to 1.2MW for a 31 kW system, it will be able to survive any reactivity accident with low power increase.

The previous operating time was set to 30 days for the decay heat level calculation after scram. This value is essentially infinite as compared to the time it takes for xenon to get to equilibrium level in the core of NIRR-1 (Salawu et al, 2015). The control rod speed was set to 8.7mm/s as found in the 2005 FSAR report. During scram, the rod move faster into the core in about 4-7 seconds as the weight attached to the rod makes it to fall freely when the current through the electromagnet that controls its speed is cut off and de-energizes it. The expressions for volumetric heat capacity as well as the thermal conductivity used in the development of PARET consist of 5 different coefficients (Obenchain, 1969). In natural convection mode, thermal conductivity was considered to be independent of temperature while the volumetric heat capacity was assumed to vary linearly with temperature. With these two assumptions, one or two of these coefficients is sufficient to describe the heat capacity as well as thermal conductivity required in PARET input file preparation. The actual values used for these coefficients were obtained from literature materials (Dunn et al, 2007 and Bokhari and Pervez, 2010). A number of steady state power specified cases were run to determine the steady state mass flow rate at different power level and this value corresponding to a particular energy level, was used in the reactivity transient analysis. The distance from the center of the active fuel region to the center of the moderator region in the fuel cell was used as the radial distance from the center of the water channel to the center of the fuel.

Also needed in the PARET input is the reactivity feedback weighting factors for the two channels. The ratio of the fuel in a single fuel pin to the total volume of the fuel in the core was used as the reactivity feedback factor for the hot channel and the difference of this value from unity was used as the reactivity feedback weighting factor for the average channel. The required axial power profile for the average channel in the preparation of PARET input file was taking from the result of a neutronic calculation for the system using VENTURE model.

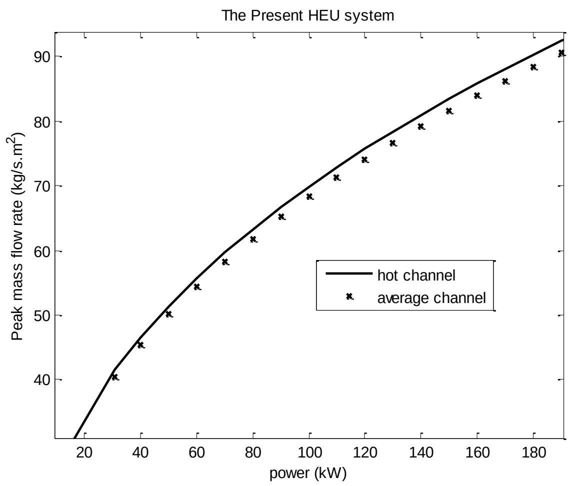

Fig.1: The peak mass flow rate for the HEU core of NIRR-1

3 Result and Discussion

In natural convection mode, the Nigeria Research Reactor-1 (NIRR-1) was licensed to operate at the maximum thermal power of 31kW. Under this mode of operation, the density difference in the fuel channels and the surrounding pool water create a buoyancy force that causes the coolant to flow upward in the fuel channels (Yamoah et al, 2011, Anthony, 2002). Therefore, the channel mass flow rate in NIRR-1 core is directly related to temperature (and density) difference between the channel and the surrounding pool. In order to determine the mass flow rate needed in the input preparation for reactivity accident simulation at a particular initial reactor power level, a number of power specified cases in the PARET code run was use to estimate the mass flow rate at different power for the present NIRR-1 system (figure 1.0).

Note that only the peak value of the axial flow rate is plotted as shown in the figure. It is clearly seen that the mass flow rate in the NIRR-1 core increases as the reactor power increases. The flow rate in the hot channels is greater than the flow rate in the average channels. This is exactly what was expected because the increase in mass flow rate in natural convention is associated with temperature increase, with a corresponding increase in reactor power. At power level of about 31kW for example, an estimate of the mass flow rate associated with each channels in NIRR-1 core is as shown in table 1.0. The mass flow rate in the hot channel is slightly greater than in the average channel at this power level (Table 1 and Fig 1). This particular PARET code run also generated the expected axial temperature profile for the fuel surface, the fuel centerline, the clad surface as well as the coolant temperature at different power level for the hot and the average channels.

Table 1: An estimate of peak mass flow rate (kg/m.s^2) and peak fuel channel temperature (degree centigrade) at 31kW power for the present NIRR-1 system.

Core type	Present HEU system	
Channel type	Hot channel	Average channel
Mass flow rate	41.47	40.40
Fuel centreline temp	50.07	49.48
Fuel surface temp	49.89	49.31
Clad surface temp	49.89	49.23
Coolant temp	38.77	38.60

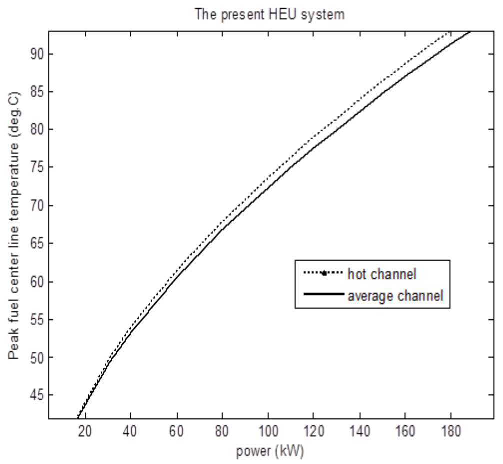

Fig 2: Peak fuel centerline temperature as a function of reactor power level.

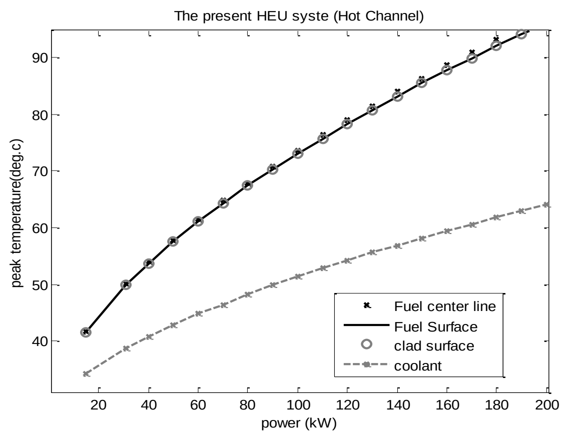

Fig 3: The peak temperature of the core components versus power in the hot channel.

The calculated peak fuel centerline temperatures were far below the melting point temperature (i.e. 650^{o}C) of the UAl4-Al fuel in the core of NIRR-1 system. The value for the fuel centerline temperature in the hot channel is slightly greater than the value for the average channels (Fig 2). The same is true for the fuel surface temperature, the clad surface temperature and the coolant temperature. The fuel, the clad and the coolant temperatures in the two channels increase as the reactor power level increases and no boiling of any kind was observed in the core up to the power level of 250kW. Fig 3 is the plot of the peak temperatures of the NIRR-1 core components as a function of reactor power level. At power levels below 80kW, there is no significant difference between the peak fuel centerline temperatures, the peak fuel surface temperature and the peak clad surface temperature in the hot channel as well as in the average channel (Fig 3 and 4). The difference between the peak fuel centerline temperature and the peak coolant temperature in these channels increases as the power level increases (Fig 3 and 4).

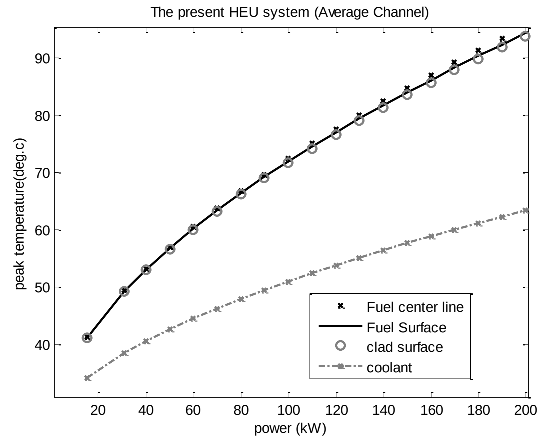

Fig 4: The peak temperature of the core components versus power in the average channel.

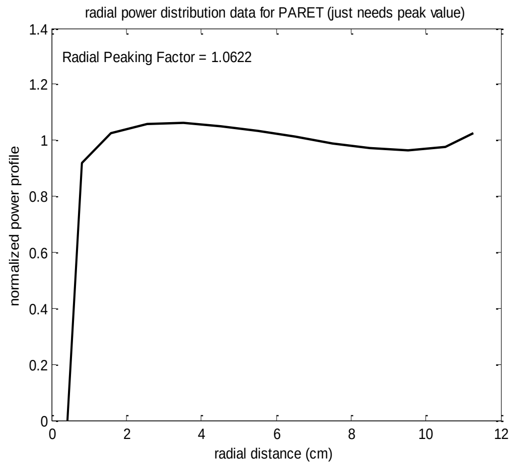

Fig 5: Radial power distribution showing the radial peaking factor for the present NIRR-1.

Before performing thermal hydraulic analysis for any research reactor system, some of the important factor required in the preparation of PARET code input file include the radial peaking factor and the axial power profile that is applicable to PARET code input. The axial (from top to bottom) and the radial power profiles obtained from neutronic calculations using VENTURE Model was used to generate these data with the help of a short Matlab code that was written to perform some preliminary calculation before the actual PARET calculation. The result of the radial peaking factor calculated for the present High Enriched Uranium (HEU) core of NIRR-1 is shown in Fig 5. A spline fit command in this Matlab code was used to fit curve to the axial power distribution obtained from VENTURE model for NIRR-1 system (Fig 6) and then extracted power profile that is appropriate to the dimension of PARET code axial distance. In addition, PARET code requires axial power profile from bottom to top as input and not from top to bottom as obtained from the VENTURE model. Therefore, a fliplr command was finally used to do the necessary axial power profile transformation within the Matlab code (Fig 7).

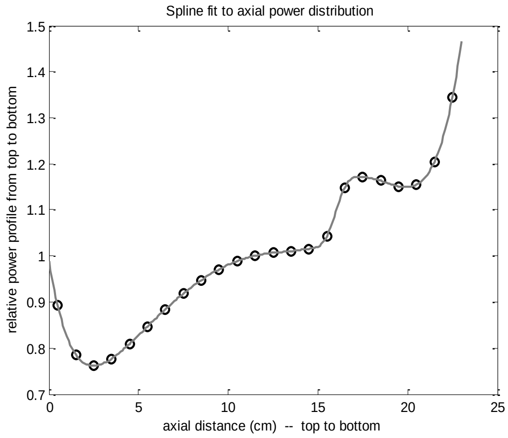

Fig 6: Axial power distribution data from VENTURE Model for NIRR-1

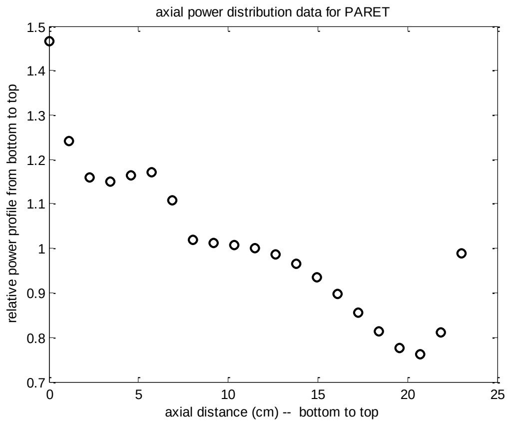

Fig 7: Suitable axial power distribution data for PARET code input preparation

The two types of reactivity accident simulated in this work include step (rapid) and ramp (slow) reactivity insertion. These were used to demonstrate that the present NIRR-1 system can be operated safely under a number of reactivity accident scenarios. The onset of nucleate boiling is generally used to set the operating limit for research reactors irrespective of the maximum power reached during reactivity accident. Therefore the safety condition is said to be met provided the ONB limit is not exceeded during reactivity transient (Anthony, 2002). The reactor behavior was studied in this work under a number of reactivity insertion accident condition with the most import been the insertion of 3.77mk of reactivity at full initial power level of 31kW. Since the excess reactivity of NIRR-1 is about 3.77mk, this simulation represent one of the worst possible reactivity accidents that can happen in the core of NIRR-1 system. This is assumed to be caused by faulty electronic system or an unexpected pressure buildup, forcing the control rod out of the core from a critical position (at full power level) to a fully withdrawn position. Thus, a ramp insertion of 3.77mk of reactivity produces the peak power of about 80kW in approximately 410 seconds (Fig 8). This peak value is in agreement with experimental result (FSAR, 2005) as well as the result from models provided in other literature materials (Dunn et al, 2007, Arne and Jonah, 2007).

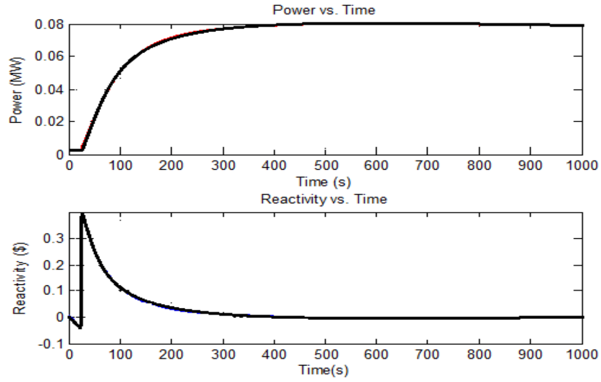

Fig 8: Variation in power/reactivity versus time for a ramp insertion of 3.77mk of reactivity

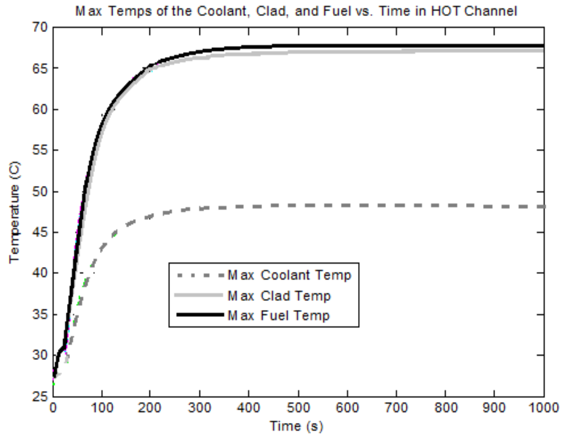

Fig 9: Hot channel analysis for 3.77mk of reactivity insertion for the present NIRR-1.

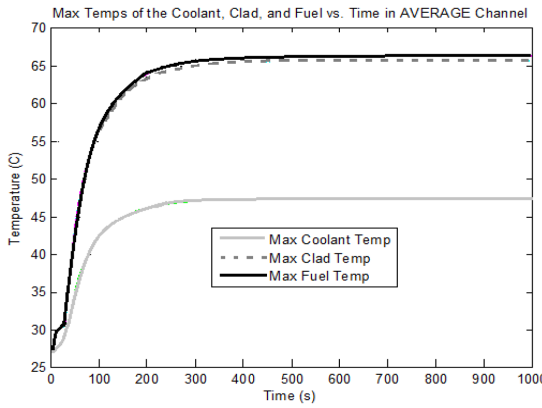

Fig 10: Average channel analysis for 3.77mk of reactivity for the HEU core.

The peak coolant temperatures in the hot and the average channels for a number of important reactor components under this reactivity transient is provided in figures 9.0 and 10.0 respectively. A ramp insertion of 4mk of reactivity produces the peak power of about 87.0 kW and the peak temperatures of the coolant in the two channels were less than 50 degree centigrade.

At full power of 31kW with an initial temperature of 25 degree centigrade, a step change in reactivity was made by inserting 4mk of reactivity in about 0.1 second. This represent the control rod been instantaneously forced out of the core from a critical position at full power level by the buildup of an unknown pressure forces within the reactor core. The reactor power increases to the peak value of about 200kW while the peak fuel temperatures in the hot and average channel were 97 and 94.6 degree centigrade respectively. The corresponding coolant temperatures for the two channels were about 64 and 62.7 degree centigrade, while the peak clad temperature in the two channels was about 2 degree less than the corresponding fuel temperature. Note that boiling of any kind was not observed in NIRR-1 core throughout this simulation exercise.

4 Conclusion

The result from the reactivity accident analysis for the present Nigeria Research Reactor-1 (NIRR-1) has shown that the reactor power can never rise to an uncontrollable level by a ramp or step insertion of up to 4mk of reactivity. Temperature of the important core components (e.g. fuel and clad) in the hot and average channels were far below their melting point temperature during this reactivity accident test. Boiling of any kind was not observed in the core in any of the two channels studied. Therefore the Nigeria Research Reactor -1 (NIRR-1) can be operated safely even if there is an inadvertent addition of up to 4mk of positive reactivity.

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