Progress in Nuclear Energy 66 (2013) 115–123

Contents lists available at SciVerse ScienceDirect

Progress in Nuclear Energy

journal homepage: www.elsevier.com/locate/pnucene

Study of successive ramp reactivity insertions in typical pool-type research reactors

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ARTICLE INFO

Article history:
Received 19 December 2012
Received in revised form
20 February 2013
Accepted 18 March 2013

Keywords:
MTR LEU system
Research reactors
Multiple ramp reactivity insertion

ABSTRACT

A comparative safety assessment of both HEU and the proposed LEU1 & LEU2 cores having UAl_x–Al, U_3O_8–Al and U_3Si_2–Al fuels respectively in a typical MTR system has been carried out using the PARET code. Super prompt-critical transients initiated with the insertion of single as well as double ramps at low power start-up have been studied in this work. The results for the standard HEU core with single reactivity ramps up to $2.2/0.5 s show excellent agreement with the already published data. According to these simulations, for large time gap values ($\Delta t = 1$ s) between double ramps, the first peak power dominates for all three cores and each system remains safe up to $2.2/0.5 s ramp rates. However, for closely spaced double ramps ($\Delta t = 0.1$ s), for both HEU and the LEU2 cores, the simulations indicate clad melting for $2.4/0.5 s ramp rates. The peak power, fuel centerline & clad surface temperature values remain within safe limits for single as well as for double reactivity ramps up to $2.2/0.5 s for all three cores.

1. Introduction

Clad meltdown poses one of the main safety concerns in all research reactors and can result in release of fission product activity to the environment. Large reactivity insertions can produce such an accident. Efforts have been made in past to determine the reactivity insertion limits for various research reactors both experimentally and theoretically. In order to obtain the limits of reactivity insertion and the corresponding transient response of various types of cores, a series of tests were carried out under the Special Power Excursion Reactor Test (SPERT) program, which consisted of a series of tests conducted at the national reactor testing station Idaho, USA. It was primarily focused on safety of nuclear reactors. The program used a series of reactors in an open tank of light water with reactor grid and forced coolant flow loop (Cardell et al., 1967; Taxelius, 1969). Several transients were studied for a wide variety of cores, configurations and conditions. Both metallic and oxide types of fuels were tested and experimental results including the reactor power, system pressure, coolant flow rate and clad surface temperatures at about 20 axial nodes was recorded for each transient. The SPERT cores were changed from MTR plate type fuel to UO_2 fueled systems representative of typical PWR cores (Cardell et al., 1967; Taxelius, 1969).

The PARET code was initially developed as a part of the SPERT program to study reactor transients theoretically (Obenchain, 1969). The code validation was done by comparing its predictions with the actual experimentally measured transient response in various SPERT cores. The computed values of reactor power at the peak burst using the PARET code showed a good agreement with the corresponding experimental data. The transient response of the B-12/64 and D-12/25 SPERT cores was estimated using the PARET code and good agreement was found between the computed and the experimentally measured dependence of the peak power on the inverse period (Woodruff, 1984).

A super-prompt critical excursion in the RA2 critical facility in Argentina occurred in September 1983. This accident took place while the reactor core was being reconfigured through a series of fuel shuffling all assumed to be in the sub-critical state. While changing from the sub-critical D-configuration to super prompt-critical configuration E, the reactor power increased by over ten orders of magnitude in fraction of a second (Waldman and Vertullo, 1987). The analysis by Waldman and Vertullo indicated that a rapid insertion of reactivity in excess of $1 took place when the core of the RA-2 critical facility was being reconfigured and the reactivity insertion crossed the safe limit for the system. This accident clearly shows that the knowledge of the safe limit for the reactivity insertion is needed.

During the past twenty years, the Reduced Enrichment for Research and Test Reactor (RERTR) program was started to covert

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0149-1970/$ – see front matter
http://dx.doi.org/10.1016/j.pnucene.2013.03.021

Highly Enriched Uranium (HEU) cores to new Low Enriched Uranium (LEU) cores. Then several studies were carried out to find effects of reactivity, power and flow rate induced reactor accidents (Metos et al., 1992; Iqbal et al., 1997; Mirza et al., 1998; Mirza, 1997; Nasir et al., 1999; Housiadas, 2000; Salama, 2012). Metos et a1., studied the transient response of the High Enriched Uranium (HEU) fueled core, Low Enriched Uranium (LEU) fueled core and mixed cores for ramp insertions of reactivity. In their studies, they used generic 10 MW MTR cores along with theoretically calculated estimates of the reactivity feedback coefficients.

A series of benchmark transients for IAEA 10 MW research reactor have been analyzed using a simulation model. They studied ramp reactivity insertion transients with initial power set at 1 W and scram level at 12 MW (Gaheen et al., 2007). Their computed values of the peak reactor power, clad peak temperatures etc., obtained by using various computer programs including PARET, RELAP5 and RETRAC-PC shows good agreement between the corresponding data values.

Analysis of the reactivity insertion accident (RIA) for the IAEA 10 MW reactor and the Egypt's second Research Reactor (ETRR-2) have been carried out for super prompt-critical transients after $4/4 s reactivity insertion (Khater et al., 2007). They concluded that ETRR-2 core can withstand the uncontrolled withdrawal of a control rod.

Recently, reactivity insertion accident at startup for the IAEA 10 MW MTR system was studied using the MERSAT computer code (Hainoun et al., 2010). For the range of inserted ramp reactivity values, for only $1.5/0.5 s value the peak power reached the highest value of 133.66 MW in 0.625 s and subsequently sub-cooled boiling and void formation occurred. Comparison of the MERSAT predicted and peak coolant temperature values showed good agreement with the corresponding data obtained using various other codes including RELAP5/Mod 3.2, PARET and RETRAC-PC. In a separate study, the transient response of a typical MTR under partial ad full channel blockage leading to loss-of-flow accident (LOFA) has recently been investigated using the Shear-Stress Transport (SST) F06B-F077 model in the fluent (ANSYS 12, 2010) framework. Under the average channel scenario, they concluded that the system may be safe under fast LOFA up to 80% blockage.

Burnup dependent reactivity feedback coefficients calculated using WIMSD/4 and CITATION codes have been employed for the super prompt-critical transient studied using the RELAP5/Mod3.4 and the PARET codes (Muhammad et al., 2012). They found enhanced inherent safety near end-of-cycle (EOC) mainly due to a larger value of the prompt neutron generation time (003F) for the reference HEU core of PARR-I.

The transient response of a typical MTR system under ramp insertion of reactivity up to $1.35/0.5 s were studied for various clad materials (Muhammad and Majid, 2010). They found the use of stain-less-steel or zircaloy-4 as clad material leads to a higher degree of thermal-hydraulic stability by reducing the maximum attainable power level and increasing the maximum value of the DNB ratio. However, both of these materials yielded higher values of fuel center-line temperature.

In order to assess the point neutron kinetic model of the RELAP5/Mod3 code, reactivity insertion transients in the IAEA MTR system have been studied recently (Hamidouche and Bousbia-Salah, 2010). They have simulated $1.5/0.5 s super prompt positive reactivity insertion transients in an initially critical reactor at 1 W power level with scram set at 12 MW. The predicted power transient modeled by using RELAP5/Mod3 has been compared with the corresponding values predicted by the RETRAC-PC (Hamidouche et al., 2002) and the PARET code (Woodruff, 1982). They concluded that the RELAP5 point kinetics model yields unphysical power evolution for some values of time step.

The Pakistan Research Reactor-I (PARR-I) was also upgraded from 5 MW HEU to 9 MW LEU core under the international RERTR program. Extensive experimental measurements of various parameters of the upgraded Pakistan Research Reactor-1 (PARR-I) core were carried out including the measurement of various coefficients of reactivity (Iqbal et al., 1997). These measurements show that for the PARR-I upgraded core, the fuel temperature coefficient of reactivity is about 21% lower as compared to HEU type. Similarly, the void coefficient of reactivity is about 31% higher and the moderator temperature coefficient of reactivity is about 24% lower as compared with the corresponding previous estimates for HEU core. Also, the rod drop times and both the differential and the integral worth of the PARR-I control rods have also been measured experimentally (Iqbal et al., 1997; Mirza et al., 1998). Also reactivity induced transients were analyzed for both HEU and LEU cores in the range of $0.05 to $1.0 having different reactivity insertion limits at various power levels. Simulations showed that the LEU core is more sensitive to perturbations at low power as compared to transients at full power (Mirza et al., 1998).

In this work we have employed experimentally measured values of various reactor dynamic parameters for the PARR-I upgraded core to determine the safe limits for fast single and multiple ramp reactivity insertions in a typical U_3Si_2–Al fueled LEU2 core. We have carried out similar analysis for the UAl_x–Al fueled HEU (93%) core and for a hypothetical low enriched oxide (LEU) core having U_3O_8–Al fuel. A comparison of the computed values of the safe reactivity insertion limits for HEU, LEU1 and LEU2 cores has also been performed.

2. Reactor description

The Pakistan Research Reactor-I (PARR-I) is a pool type Material Test Reactor composed of 24 standard fuel elements (SFE), five control fuel elements (CFE) and one central flux trap facility (CFT) as shown in Fig. 1. The upgraded core of PARR-I has been designed to operate at full power of 9 MW with primary and secondary coolant loops dumping the generated energy in the air using a cooling tower. During the full power operation, the coolant flow rate through the core is kept constant at 900 m^3/h level (Iqbal et al., 1997).

The low enriched uranium (LEU2) core is composed of 23 U_3Si_2–Al fuel plates in each standard fuel element and 13 in the control fuel element. This core has 290 g loading of 20% enriched ^{235}U loading per standard fuel element. A summary of the design and thermal hydraulic parameters for the LEU2 core is given in Table 1

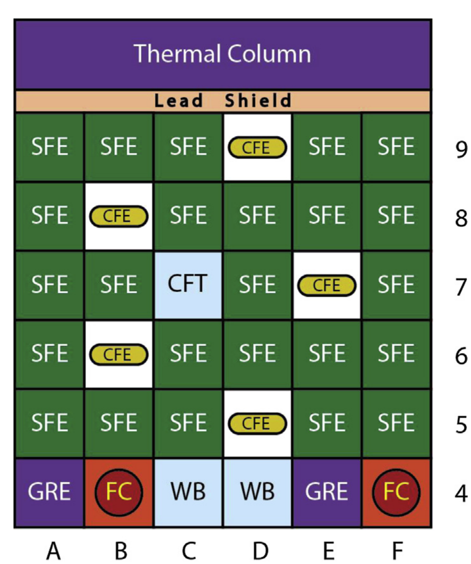

Fig. 1. The core configuration of a typical system for PARR-I.