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Transient thermal–hydraulic analysis of complete single channel blockage accident of generic 10 MW research reactor

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ABSTRACT

Thermal–hydraulic behavior for a complete blockage of a single fuel channel in a generic 10 MW research reactor is studied by using the system analysis code RELAP5/MOD3.3 which is widely used in the nuclear industry. Fuel assembly geometry is lumped into a 4 channel model to model high and average power cases which are spatially discretized. Various axial power shapes coming from different control rods positions are considered in the analysis, where the minimum wall subcooled margin is found to exist for case with highest peaking for an average powered channel blockage transient. Vapor generation is observed from first and second highest peaking cases where cyclic variation of vapor inventory inside a blocked channel resulted in oscillatory behavior of the fuel temperature. Effect of a presence of an oxide layer is also tested which showed a slight increase in structure temperatures and vapor generation. Point kinetics model is utilized in the analysis code to observe the effect of reactivity feedback and consequences from different application ranges are compared. Analysis shows a consideration of assembly wise feedback results in increased feedback effect and decreased boiling which deviate from single channel wise feedback case. This calls for a detailed multi-dimensional simulation with neutronics and thermal–hydraulics simultaneously considered. Analyses results show that the consideration of feedback improves the outcome in terms of fuel temperature, and its integrity is conserved for all test cases.

1. Introduction

With growing nuclear proliferation concerns, a Reduced Enrichment Research and Test Reactors (RERTR) program was launched in the late 1970s, which triggered a worldwide research effort to replace high enriched uranium (HEU) research reactor fuels with low enriched uranium (LEU) fuels (Stahl, 1978). The modified core and affected system coming from the core conversion call for a new set of safety analyses for a licensing renewal. As part of these activities, International Atomic Energy Agency (IAEA) has published a series of technical documents which touches upon safety and licensing considerations during the core conversion process along with benchmark specifications of the generic 10 MW light water cooled pool type research reactor (IAEA, 1980, 1992a). In the benchmark problem, neutronic parameters were evaluated and thermal–hydraulic behavior during the transients was studied. In the transient calculations, the participants analyzed loss of flow and reactivity insertion accidents (LOFA and RIA). In addition, based upon this problem description, researchers have analyzed the fuel channel blockage accident, which is also one of the accidents with a decrease in the heat removal that need to be re-analyzed as part of the safety analysis for licensing renewal (IAEA, 1992a). When the coolant flow undergoes a rapid decrease due to partial and total channel blockage triggered by inadvertent insertion of foreign materials into the core, the heat transfer between the coolant and the fuel becomes impaired. It may also result in a cladding failure and release of radioactive inventories, which makes it a noteworthy accident to investigate the progression and underlying phenomena.

Adorni et al. (2005) analyzed the partial and total flow blockage of the standard fuel assembly utilizing RELAP5/Mod3.3 code. They adopted a typical research reactor layout and modeled each 25 fuel assembly (FA) as individual flow paths so that the simulated blockage could represent the assembly-wise transient. The simulation results showed that the negative void reactivity feedback from the obstructed channel was strong enough to keep the integrity of the fuel for the partial flow blockage (up to 95%) case, and was not sufficient for the total blockage case. Lu et al. (2009a) investigated the accident utilizing the same code, but a different control volume is used where 9 parallel channels representing flow channels between the fuel plates were modeled. In the study, the fuels

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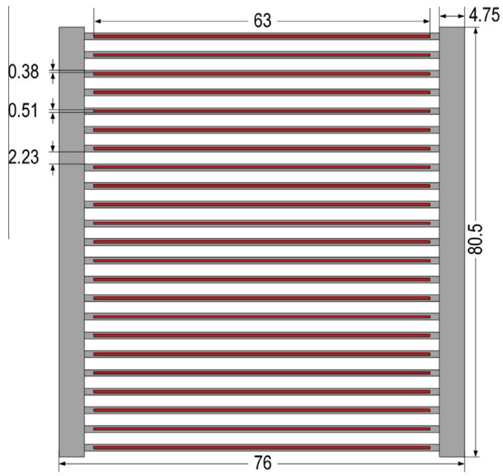

Fig. 1. Cross section view of standard fuel element (dimensions in mm).

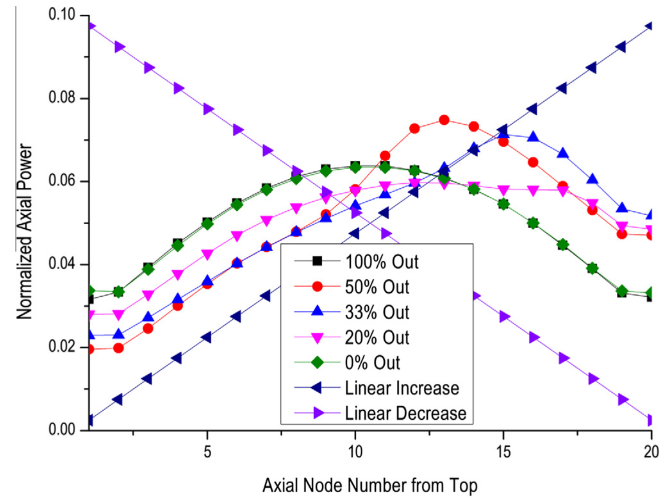

Fig. 2. Normalized axial power profile with various control rods positions.

Table 1
Reactor Design Specification.

Parameter	LEU fuel	HEU fuel
Core thermal power [MW]	10.0	
Fuel type	Plate fuel	
Coolant	Light water	
Fuel material (LEU)	U_3Si_2-Al	UAl_x-Al
U-235 enrichment [%]	20	93
Cladding material	Al 6061	
Fuel height [m]	0.6	
Number of fuel plates/SFE	23	
Coolant inlet temperature [K]	311.15	
Coolant inlet pressure [MPa]	0.17	
Nominal flow rate [m^3/hr]	1000.0	
Normal core flow direction	Downward	
Average heat flux [kW/m^2]	205.4	
Combined radial and local power peaking factor	1.4	
Axial power peaking factor	1.5	
Engineering factor	1.2	
Cycle length [days]	16.7	

are nodalized in order to simulate the heat transfer between neighboring channels, and a neutronic feedback was not used for conservative reasons. Their simulation showed that the integrity of the fuel was conserved even for a totally blocked channel case owing to cooling from the adjacent channels. Lu et al. (2009b) also studied a single channel blockage transient with their in-house homogeneous two phase flow analysis code THAC-PRR where the reactivity feedback was taken into account. The calculations showed that a blockage lead to a redistribution of the coolant flows into the adjacent channels but the increase was minor. In addition, no incipient boiling was seen even for a total blockage case with the power peaking and the engineering factors considered. Salama (2012) utilized a computational fluid dynamics code ANSYS FLUENT 12 to perform a two-dimensional (2D) transient simulation of the LOFA combined with the average channel flow blockage created by an axially deformed fuel plate. The simulations showed that the cladding integrity is conserved up to 80% of the blockage. Salama and El-Morshedy (2012a) adopted the same FLUENT code on the LOFA with a hot channel flow blockage transient in a 2D manner to observe that 20% of the blockage was the upper limit without boiling. Salama and El-Morshedy (2012b) have carried out a steady-state FLUENT simulation on the three parallel flow channels where the boiling was not occurred up to 90% and 100% blockage in the hot and the average channels, respectively. Based upon the above literatures, this study deals with the transient simulation

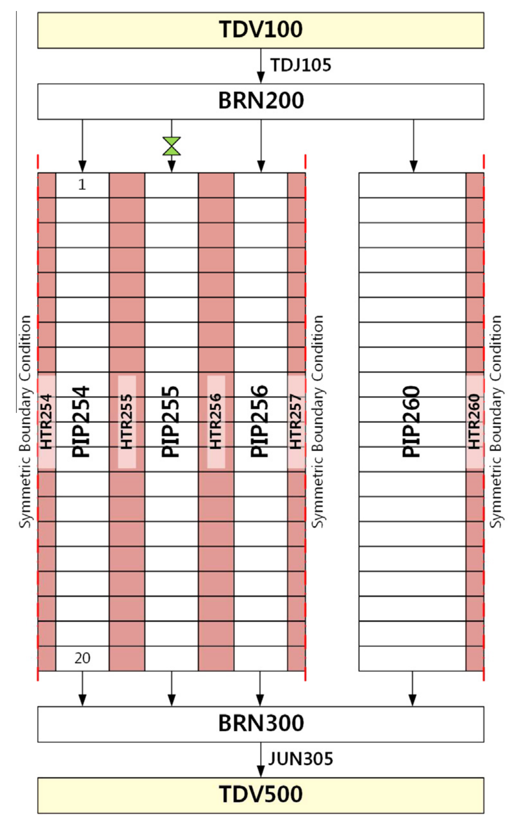

Fig. 3. RELAP5 nodalization map of SFE for single channel blockage transient simulation.

of a complete blockage of the single channel where the diverse results are reported in terms of the boiling phenomena. The sensitivity study is also carried out to observe the effect of various factors such as different power profiles, growth of corrosion layer coming from aging, reactivity feedback, and different fuel enrichment (LEU/HEU) in terms of thermal margin to boiling and peak fuel temperatures. The widely used system analysis code RELAP5/Mod3.3 code is utilized to evaluate the thermal–hydraulic behavior during the transient (USNRC, 2001a).