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NPP Krško Post-UFC Transient Response during MSLB

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ABSTRACT

UpFlow Conversion (UFC) was implemented in NPP Krško during the last outage in order to reduce the pressure differential across baffle plates and the possibility of the fuel damage caused by flow induced vibration. The paper describes the coupled code calculation (RELAP5 and PARCS) of MSLB accident at power for pre and post-UFC configuration of reactor vessel. In the calculation, the split model of the reactor vessel was used to better describe asymmetric conditions in loops. It has been demonstrated that the basic parameters (pressure, temperatures) stayed unchanged and there was little change in the flow rates except in baffle-barrel region of the vessel where both flow direction and amount of flow were changed.

Keywords: *upflow conversion, MSLB, coupled code, reactor vessel model*

1 INTRODUCTION

During NPP Krško refuelling outage in May 2015. a modification in reactor vessel internals was made [1]. Bypass flow in the baffle-barrel region, which was previously a downward flow, was converted to an upward flow. This modification reduced the pressure differential across baffle joints and therefore decreased the possibility of the fuel damage. On the other hand, it increased the bypass flow and thereby slightly decreased the core mass flow. Based on the safety review, the greatest influence should be related to LOCA (Loss Of Coolant Accident) and all other accidents should be unaffected. In this paper, the coupled code calculation of MSLB (Main Steam Line Break) accident for pre and post UFC modification was preformed. Cycle 27 was used for pre and Cycle 28 for post UFC calculation of NPP Krško. The split reactor vessel model was taken into account because it better describes the asymmetric character of MSLB.

2 UPFLOW MODIFICATION

Damaged fuel assemblies have been identified in NPP Krško during 2013. outage refuelling activities. According to Westinghouse, the main reason for those fuel rod failures was flow-induced vibration. This phenomenon, known as “baffle jetting” is common among fuel assemblies in the periphery of core, depending on orientation and condition of baffle plate joints. Baffle jetting is a hydraulically induced vibration of fuel rods caused by a high velocity lateral jet of water. This jet is created by high-pressure water, forced through gaps between baffle plates near upper core plate. In that area, pressure differential across the baffle joints is the largest, and it becomes smaller downward to the lower core plate. Baffle-barrel bypass flow direction is responsible for this significant pressure differential.

Figure 1 shows modification required in reactor vessel discretization to take into account changes in vessel bypass flows. Primary flow passes down through the downcomer region, enters the lower plenum, then upward into the core region, into the upper plenum and out through the outlet nozzle. The portion of the primary coolant does not participate in removing core heat and it is called bypass flow. Following streams belong to bypass flow:

- head cooling spray nozzle – the portion of the flow that flows from the vessel inlet nozzle into the vessel head region,
- outlet nozzle – the portion of the flow that leaks through the gaps between the core barrel outlet nozzles and reactor vessel outlet nozzles and merge with the vessel hot leg outlet nozzle,
- core cavity gap – the flow between peripheral fuel assemblies and baffle plates, which has the same direction as the flow through the core,
- thimble tubes – the flow through thimble tubes, which has the same direction as the flow through the core,
- baffle-barrel region – the flow between baffle and barrel, which had the opposite direction to the flow through the core (for the nodalization before UFC).

Most affected part of bypass flow is baffle-barrel bypass. Before UFC modification, it had the opposite direction to the flow through the core. As the primary coolant entered the downcomer, a portion of the flow diverted and passed through the holes in the core barrel between the top first and second former plate. The pressure in this region is higher than pressure in the core at the same elevation. This pressure differential caused baffle jetting by high pressure water, which passed through gaps between the baffle plates. Through the time, baffle jetting had been causing damage of the fuel rods, which could lead to cladding failure and the dispersal of the fuel pellets into the coolant. To solve this problem, Westinghouse developed an approach where downward flow in the baffle-barrel region was reversed to the upward flow.

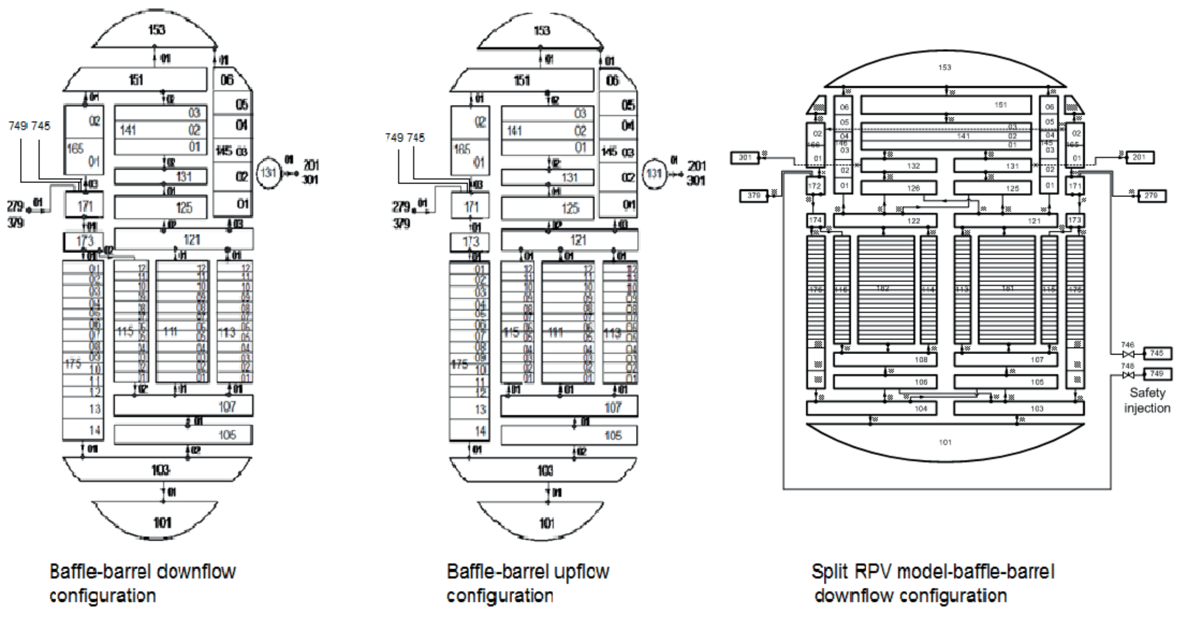

Figure 1: RELAP5/mod 3.3 nodalization of NPP Krško reactor vessel: a) baffle-barrel downflow and b) upflow for one coolant channel, c) split RPV model

The left side of the Figure 1 shows RELAP5 nodalization (single core channel) of NPP Krsko reactor vessel before and after UFC modification. The same type of change apply for split vessel nodalization (two downcomers, two bypass channels, two or more core channels) on the right side of the figure. As part of the modification, 16 core barrel flow holes between top first and second former plate were plugged, and 8 new holes, each having nominal diameter of 2.5 inches, were machined in the top former plate.

The analysis of the UFC modification has proved the decrease in the pressure differential across the baffle joints. Figure 2 shows the difference between pre and post UFC modification. Gray color shows pressure differential before UFC modification, and it is clear, as previously said, that the greatest pressure differential is near the upper core plate and is continuously decreasing towards the lower core plate. After modification, pressure differential is lower, and it alternates from positive values (yellow) to negative values (red).

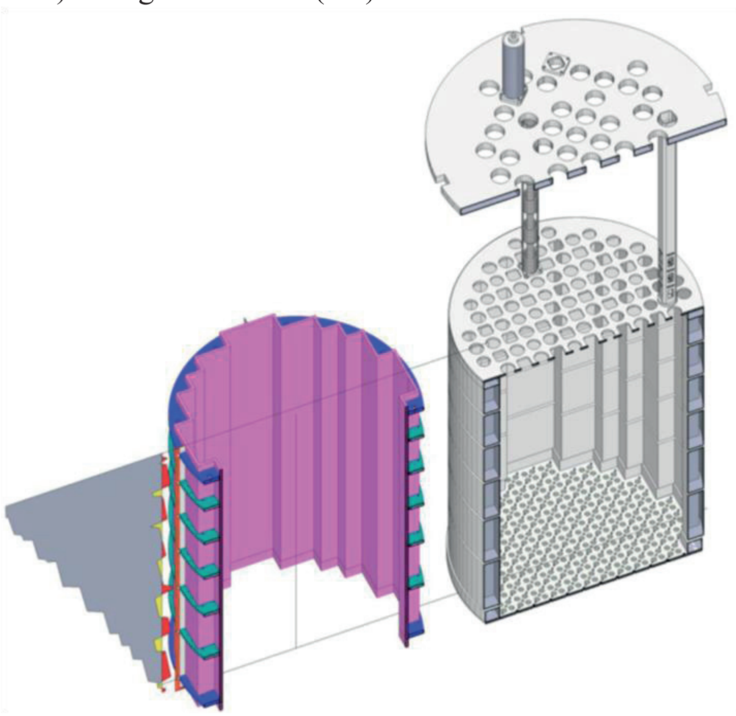

Figure 2: Pressure differential along baffle plates for pre and post UFC configuration

3 COUPLED RELAP5 AND PARCS PROGRAM

Code coupling is a standard methodology used to describe transients having spatial reactivity dependence in the core and thermal-hydraulic influence from rest of the system [7]. The coupled code RELAP5/mod3.3-PARCS v2.5 (R5PA) has been developed at FER. RELAP5/mod3.3 [2] is a code for modeling complex thermal-hydraulic systems and PARCS [3] is a three-dimensional (3D) reactor core simulator. Therefore RELAP5/mod3.3 calculates system thermal-hydraulics, average core channel thermal-hydraulics and heat conduction whereas the code PARCS calculates 3D neutron kinetics. Within R5PA it is possible to use COBRA code to perform core channels thermal-hydraulics calculation within PARCS code.

Taking into account asymmetric character of MSLB accident [6], the split model of the reactor vessel was used (Figure 1). The main difference, compared to the standard nodalization, is that all reactor vessel parts below hot and cold nozzles (downcomer, core inlet plenum, active core, guide tubes, core bypass and core outlet plenum) have been subdivided in two main parts, each corresponding to the one plant loop. The mixing was modeled in inlet and outlet plenum with coefficients 0.4 and 0.5, demonstrating that 70% of the cold leg flow is delivered to the closer region of the core and 75% of the hot leg flow is from the half of the core closer to the loop. There are 18 thermal-hydraulic channels in the core, 9 for each part of the core (loop), and 24 equidistant axial subdivisions for the active core region. The lower plenum is divided into seven CVs: 101, 103 and 104 connected to downcomer parts from the two halves of the vessel (before mixing), 105 and 106 describing the middle parts after mixing and the volumes 107 and 108 representing upper parts of the lower plenum before entering the reactor core. The active core is modeled with 18 channels (181 to 198) that are divided in two halves of the core (volumes 184, 185, 186, 190, 191, 192, 196, 197 and 198 for the loop 1 and volumes 181, 182, 183, 187, 188, 189, 193, 194 and 195 for the loop

2). The core channels are modeled as PIPE components each consisting of 24 volumes. RCCA guide tubes inside core are represented with volume 113 for 1st half and volume 114 for 2nd half of the core. The region between baffle and barrel is also represented with two volumes (CV 115 and CV 116 for each half). The region above the active core is represented with CV 121 and 122 before mixing and with CV 125 and 126 after mixing. Upper downcomer is also subdivided in two halves with corresponding volumes 165 and 166 with which the bypass flow path to the upper head is introduced. Another bypass is modeled in the reactor inlet volume (CV 171) connecting core outlet (CV 125) for the 1st half of the core and volume 172 connecting core outlet 126 for 2nd half.

The detailed description of the NPP Krško RELAP5/mod3.3 nodalization before UFC modification is reported in [4], [5]. Cycle 27 is representative for pre UFC condition of NPP Krško. On the other hand, Cycle 28 is representative of condition after UFC modification. Those two nodalization are very similar except for small UFC related change and usual small variation in the point kinetics and distribution of the power in the core. To take into account change in the direction of the flow path in the baffle-barrel region the junction 02 in branches 106 and 107 is directed upward. The junction 02 in branches 173 and 174 is deleted because no coolant flow is directed downward the baffle-barrel region after core barrel flow hole plugs are installed in the former plates. The new flow path is introduced to model the flow in the barrel baffle region from the upper plate to the outlet plenum of the reactor vessel. The same modifications are done to the 2nd part of the reactor vessel in split model. In addition to reactor split vessel model used till now, alternative model was developed with additional lateral connections between two downcomer halves. The model showed benefits for LOCA modeling and we wanted to see what is its influence for other asymmetric accidents.

Used NPP Krško nodalization, Figure 3, has 1054 control volumes, 1146 junctions, 1157 heat structures (with 10159 mesh points), 733 control variables, with 197 variable and 221 logical trips.

The coupled code steady state calculation was performed for 1000 s at full power nominal conditions.

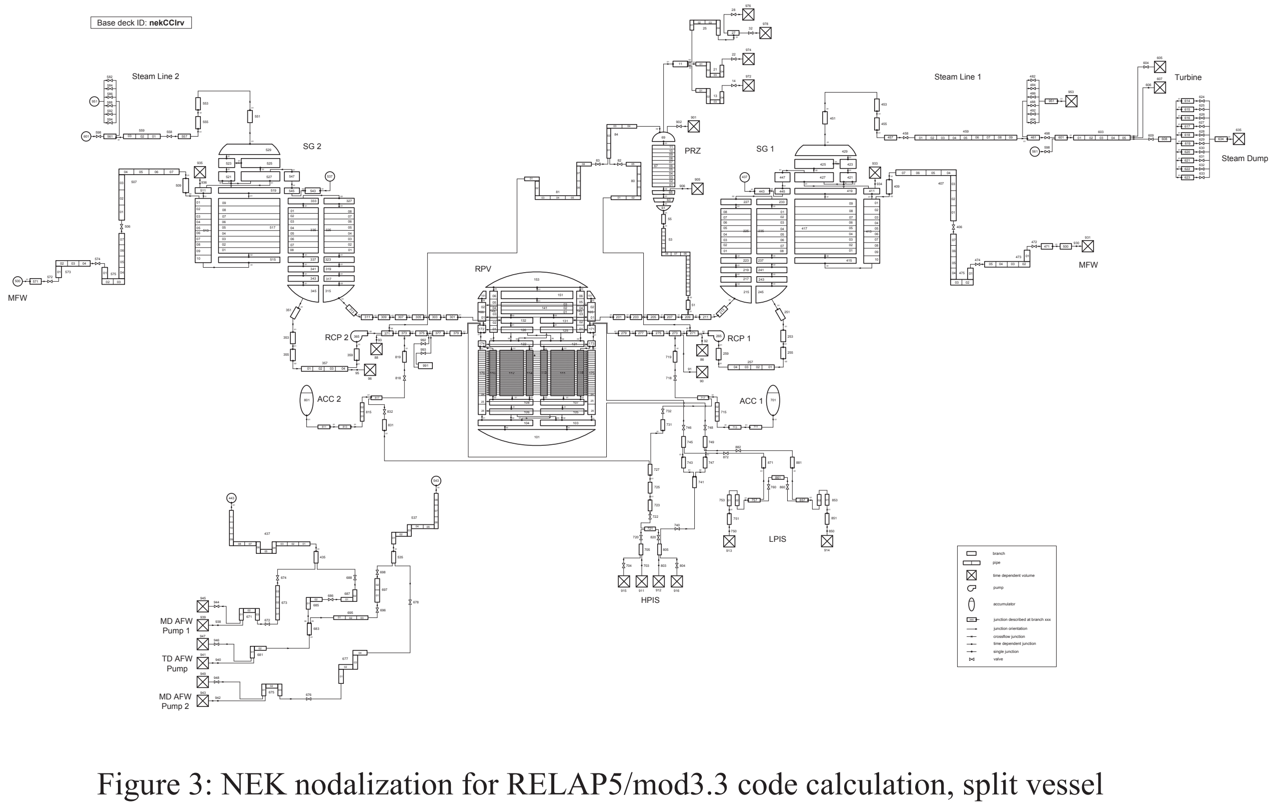

Figure 3: NEK nodalization for RELAP5/mod3.3 code calculation, split vessel

4 RESULTS OF THE CALCULATION

4.1 Accident Description

The analyzed transient was the main steam line break which is classified as ANS Condition IV event. It belongs to accidents which cause increased energy removal from the Reactor Coolant System (RCS) consequently leading to the reduction of coolant temperature and pressure. Main assumptions for the accident are:

- double ended (guillotine) break,
- end of core life,
- hot full power - normally, NPP is at the zero power for MSLB accident because the coolant flow through the break is at the higher rate and the core locally reaches higher power, but it is noticed, that for the UFC modification, the full power has bigger impact on the limiting faults,
- reactor trip occurs on the low pressure signal in the steam line,
- most negative rod cluster control assembly is assumed to be stuck in its fully withdrawn position after reactor trip,
- core boron concentration is 0 ppm
- 0% SGTP
- main feedwater system supplies both steam generators whereas auxiliary feedwater system supplies only broken steam generator,
- reactor coolant pumps trip 60 s after reactor trip.

The transient calculation was performed for 1000 s, yet all the important cooldown related changes happened in first 200 s.

The analysis has shown that there was no return on the power after MSLB accident and that the integrity of the cladding was conserved. Steam generator 1 was assumed to be broken and the steam generator 2 stayed intact during the accident. The steam release arising from the rupture of the main steam line resulted in an initial increase in steam flow. This rupture in the steam line rapidly decreased broken steam generator pressure, Figure 4. As the pressure in the broken steam generator was falling, the steam flow started decreasing during the accident. The increase in the energy removal from the reactor coolant system caused a reduction of the coolant temperature and pressure, Figure 5. The SG 1 pressure was constantly falling until it reached the containment pressure. Pressurizer pressure was also falling, and after approximately 100 s started increasing, shortly after the initiation of the safety injection system and isolation of the broken steam generator. The similar pressure behavior was presented in the unbroken steam generator, firstly the pressure decreased, and then it increased after heat transfer reversal. Due to the cooldown accident and sink in the secondary system, temperatures, both hot and cold legs, decreased. With the split model of the reactor vessel, better asymmetric character of MSLB accident is described, therefore different distribution of cold and hot legs temperature was presented in comparison with standard 1 channel model of the reactor vessel, Figure 6. The reverse heat removal started in the intact steam generator, approximately 5 s after reactor trip. That means that the steam generator became heat sink in oppose to his standard function. This change is noticeable in the intact cold leg temperature which increased and intact steam generator power which became negative after reversal. Mass flow in both loops increased little due to reduction of temperature, Figure 8, and then decreased rapidly after reactor coolant pumps trip.

4.2 Comparison of conditions before and after UFC modification

Figure 4 and Figure 5 show that basic parameters (pressure and temperatures) stayed unchanged after modification. Very small difference in the basic accident behaviour is visible among two nodalizations/core cycles. As expected, due to asymmetric nature of accident, there is difference between coupled code split vessel model and RELAP5 calculation using point kinetics (PK) and one-channel vessel model, Figure 4 and Figure 6. The differences are mostly related to different cold leg temperatures. Differences between old and new model of split vessel (lateral connections) are shown in coolant temperature response in Figure 7. The influence is rather small and it is more significant for pre than for post UFC conditions.

UFC modification has direct influence to core bypass flow, mostly to baffle-barrel region flow. Overall influence is small increase in loop flows (lower hydraulics resistance of the vessel), and decrease in core flow, Figure 8. Total bypass flow is increased with baffle-barrel region flow being larger, Figure 9, and guide-tubes bypass flow being lower than before UFC modification. Figure 9 and Figure 10 show mass flow in the baffle-barrel region in both parts of the reactor vessel, before and after UFC. Mass flow changed direction (sign), as expected, and now has positive value because the direction is the same as in the active core. Furthermore, mass flow rate increased, approximately 20 kg/s in each part of the core, leading to approx 40 kg/s increase for the whole core. Small influence of change in vessel model and change in vessel downcomer model can be seen in Figure 9 and Figure 10, respectively.

As a consequence of small changes in thermal-hydraulics variables the changes in nuclear variables are not expected. Pre and post UFC reactivity components are shown Figure 11, and differences are very small. Nuclear peaking factor is shown in Figure 12. Reference coupled calculation is without stuck rod to be more similar to PK response. When stuck rod is assumed, increase in peaking factor is present during core cooldown. In addition there is small Fq difference due to different mixing in downcomer when lateral connections are used. Taking into account that core power is reduced immediately after break initiation due to reactor trip, the difference in peaking factors has no practical value. Pin power distribution in case of stuck rod is shown in Figure 13. The location of the rod is usually in more affected (cooled) part of the core.

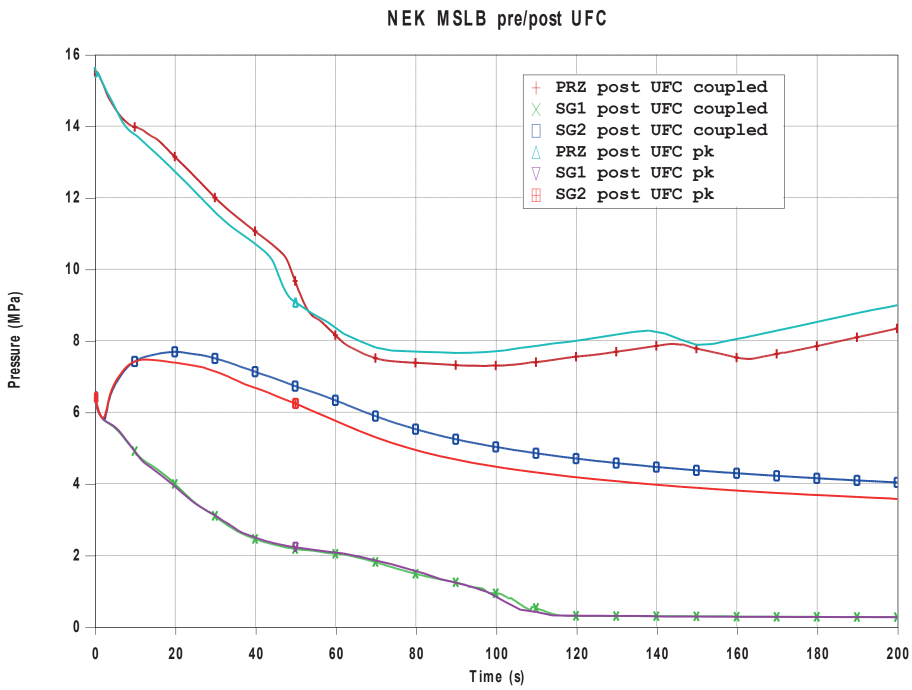

Figure 4: Pressurizer, Steam generator 1 and 2 pressure

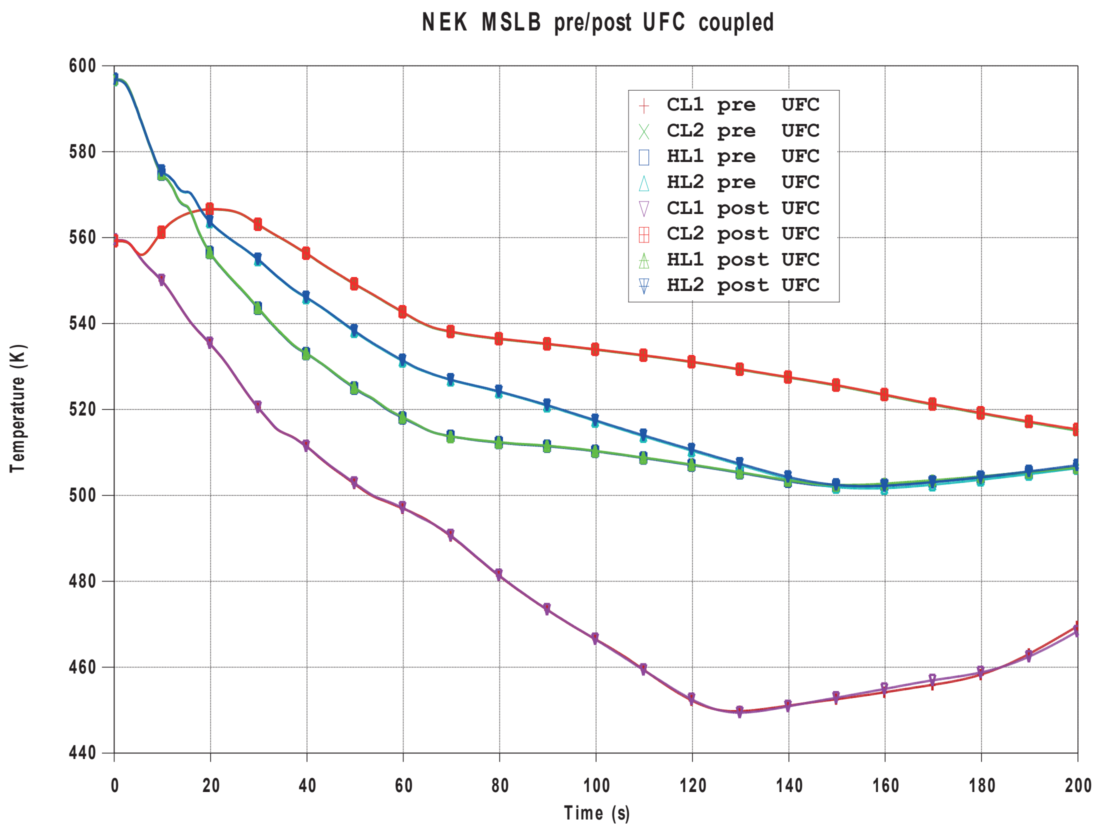

Figure 5: Cold and hot leg temperatures, pre vs. post UFC

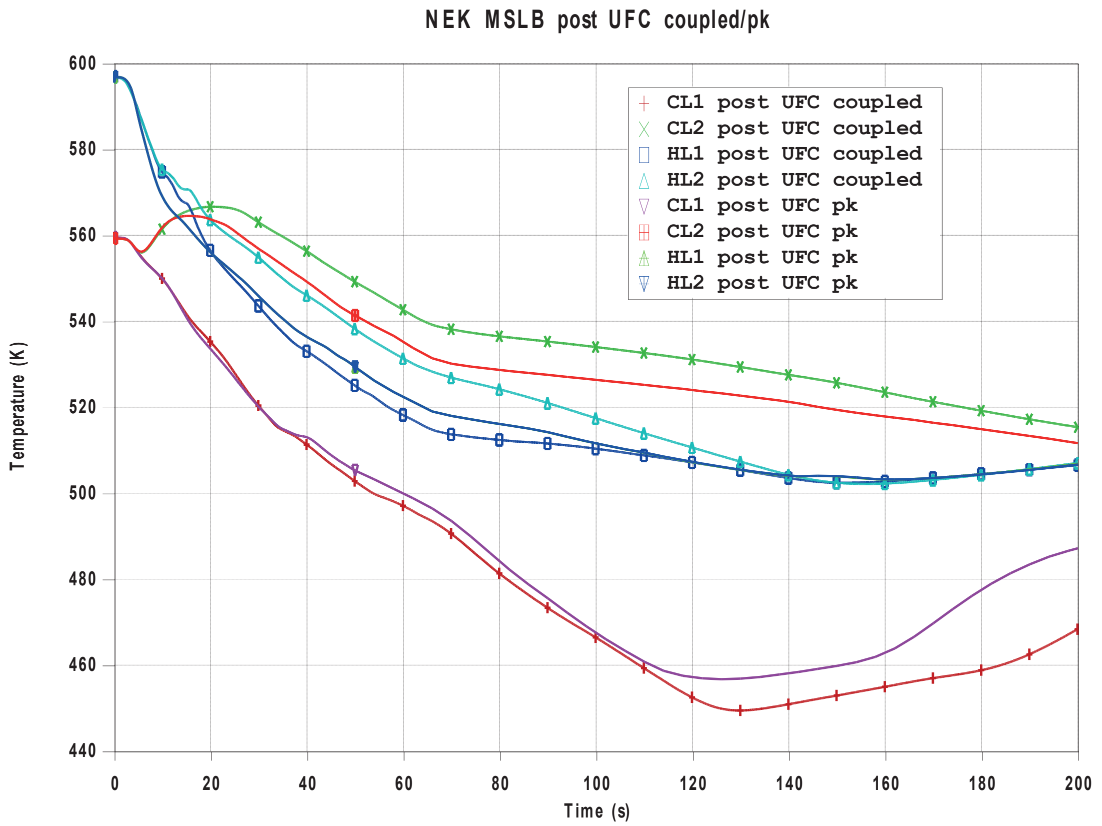

Figure 6: Cold and hot leg temperatures, coupled vs. PK

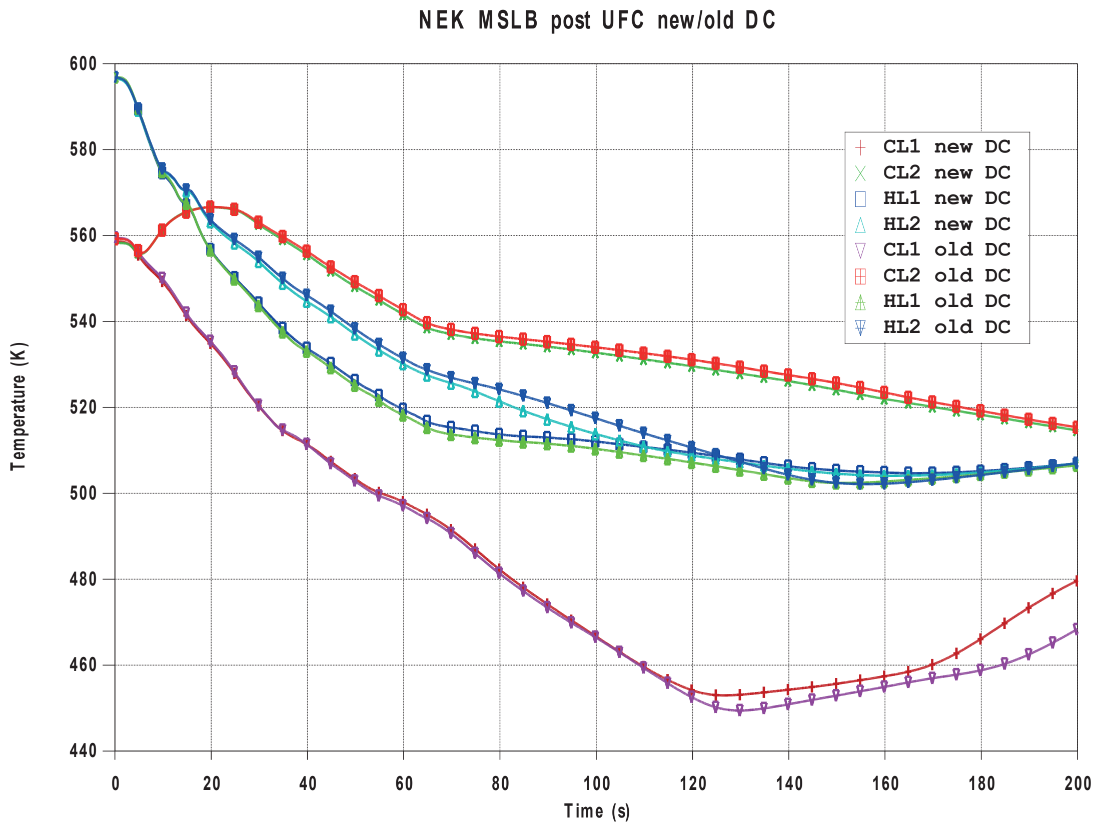

Figure 7: Cold and hot leg temperatures, new vs. old downcomer

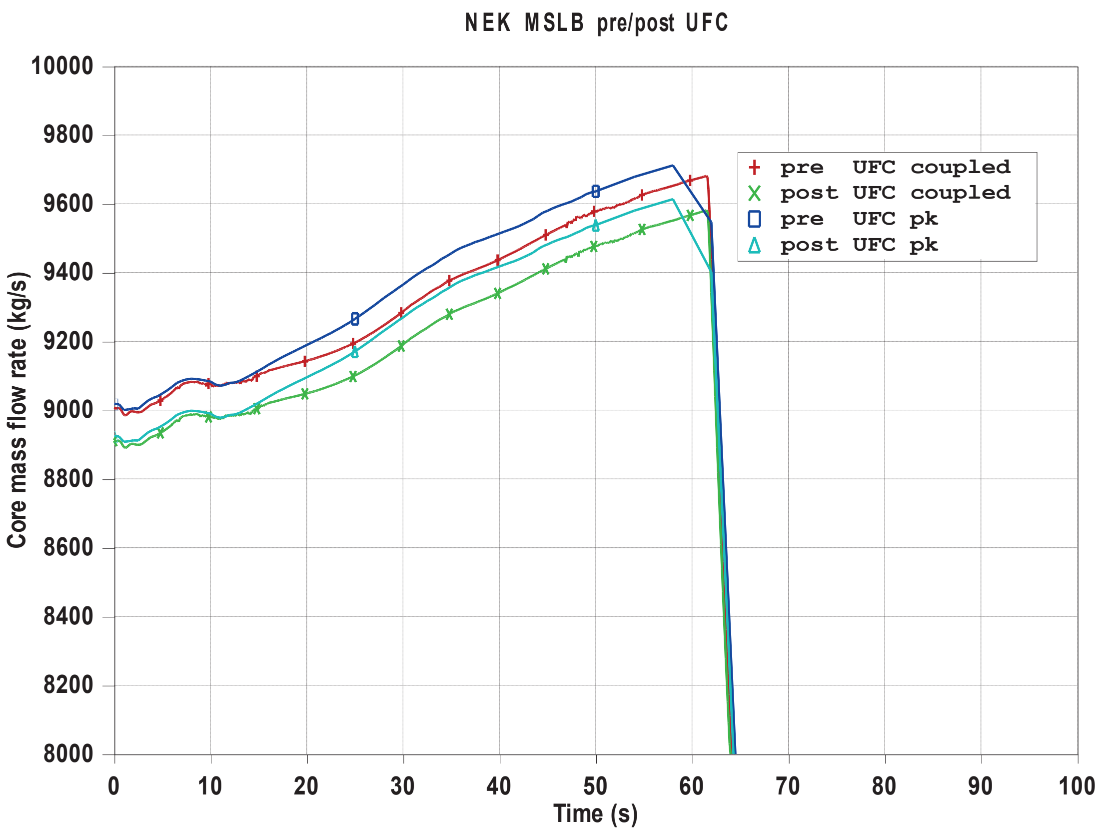

Figure 8: Core mass flow rate, pre vs. post UFC

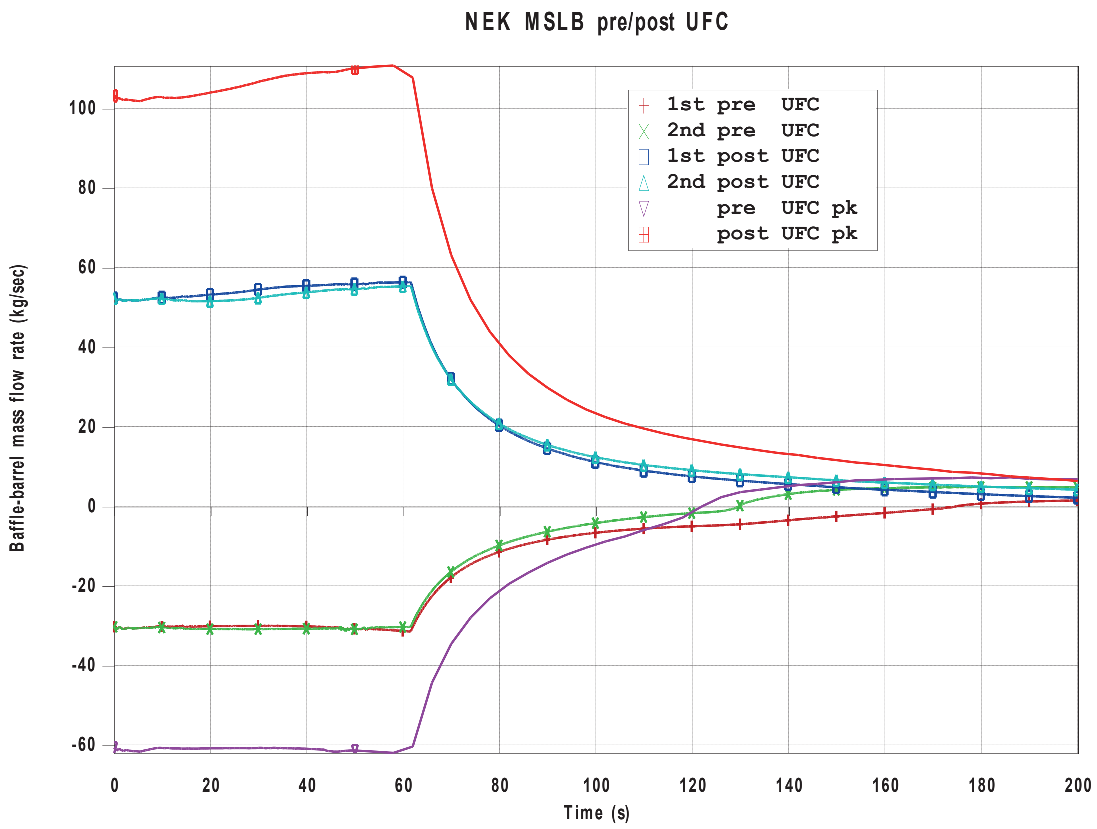

Figure 9: Baffle-barrel mass flow rates, pre vs. post UFC

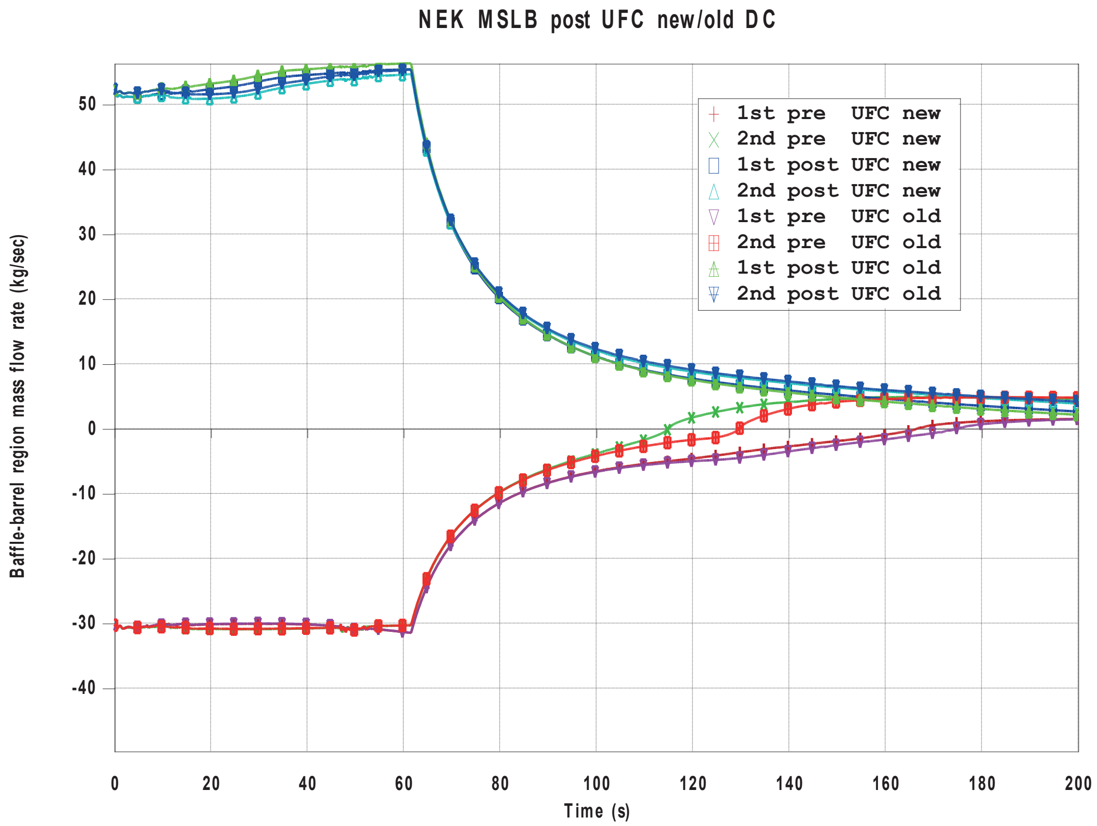

Figure 10: Baffle-barrel mass flow rates, new vs. old downcomer

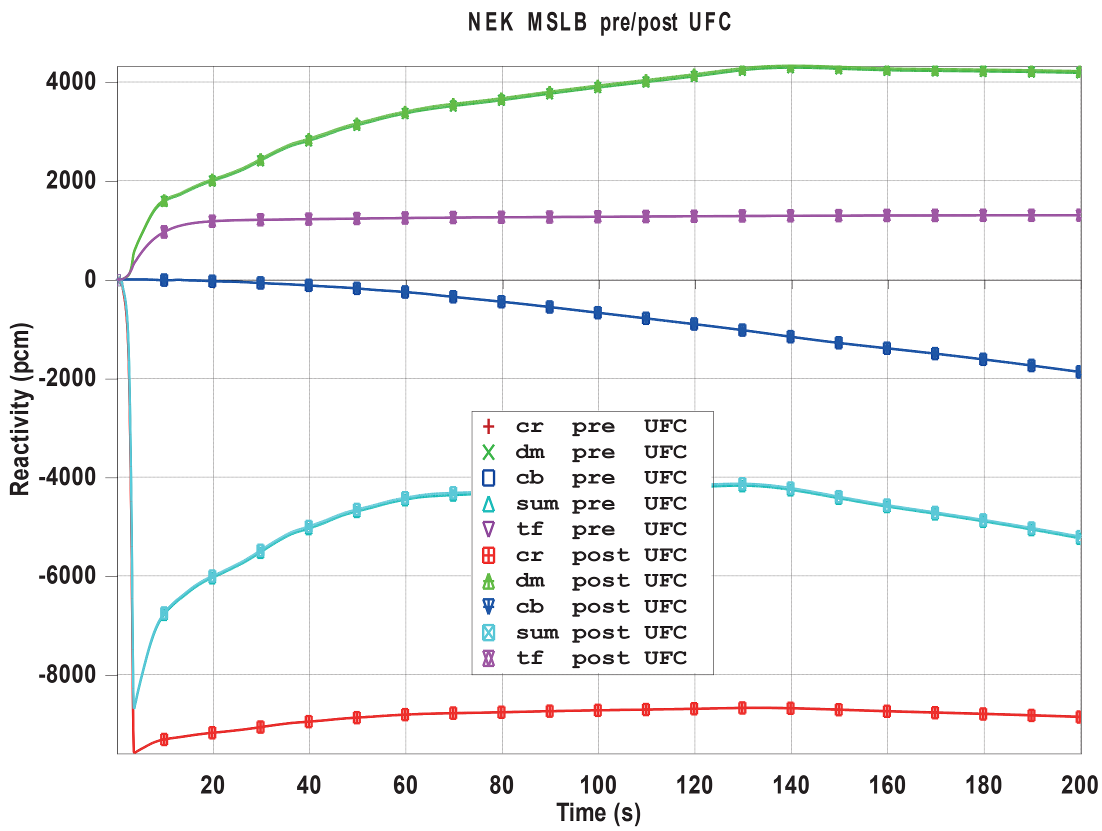

Figure 11: Reactivity components, pre vs. post UFC

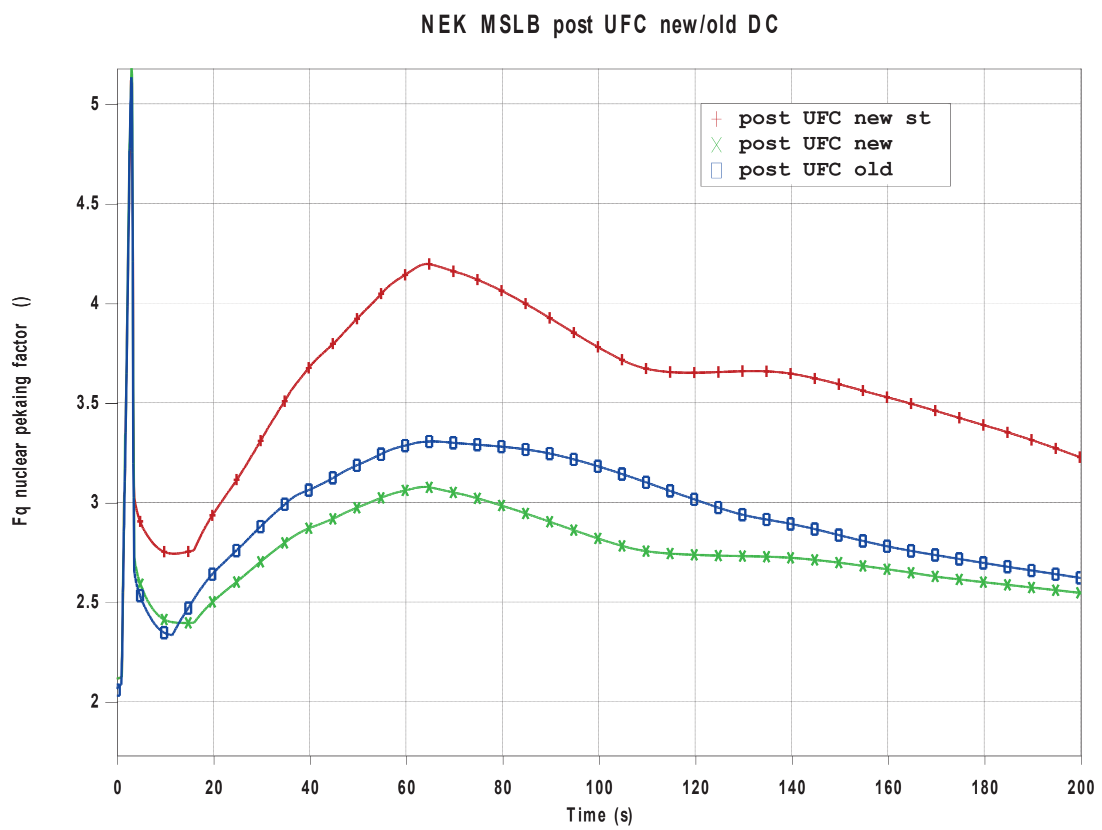

Figure 12: Fq nuclear peaking factor, post UFC, influence of stuck rod

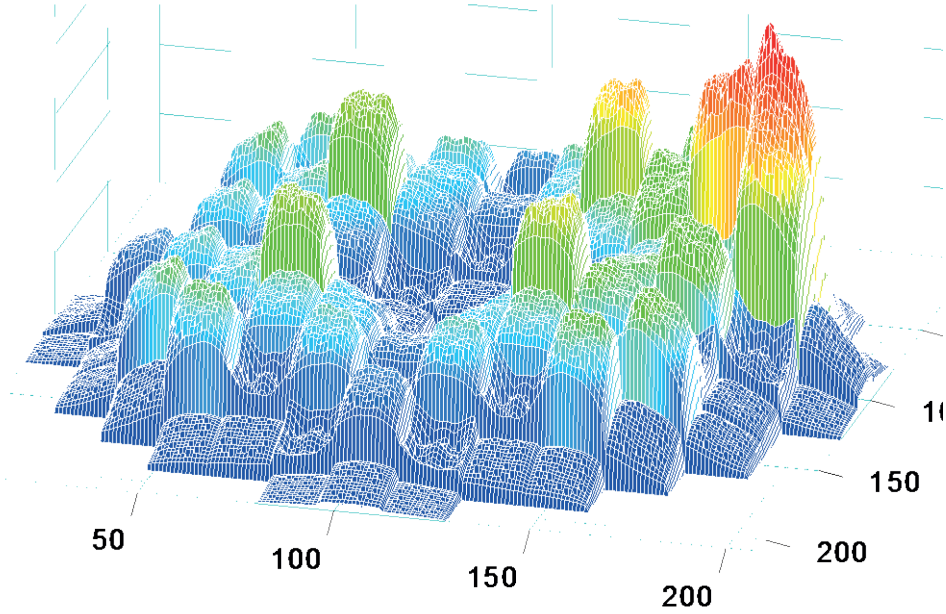

Figure 13: Pin powers at 200 s, post UFC case with stuck rod

5 CONCLUSION

No significant difference was found between the standard 1 channel model of the reactor vessel and split model of the reactor vessel, only a slightly different behaviour of the temperature in loops due to the different mixing. Furthermore, the comparison of the accident behaviour before and after UFC modification did not show difference as it was expected from screening analyses performed before the modification was implemented in the NPP Krško. The results only showed difference in the reactor vessel flows. After UFC modification, the bypass flow increased from 0.5 to 1% of the total coolant flow value, and stayed within design project calculations of the total 6.5% bypass flow. The baffle-barrel flow increased for approximately 40 kg/s and changed sign due to the opposite direction of the flow paths after modification. The guide tubes flow decreased a little bit, less than the baffle-barrel flow increased, therefore the active core flow decreased. The coolant flow toward the upper head stayed practically the same and the mass flow in loops slightly increased. The reactor vessel split model with additional lateral connections between downcomer parts showed similar results as original split vessel model (fully separated downcomer halves).

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