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INFLUENCE OF REACTOR VESSEL NODALIZATION IN THE COUPLED CODE ANALYSIS OF ASYMMETRIC MAIN FEEDWATER ISOLATION

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ABSTRACT

Asymmetric Main Feedwater Isolation (AMFWI) transient in one Steam Generator (SG) for NPP Krško using RELAP5 standalone code and coupled code RELAP5-QUABOX/CUBBOX (R5QC) was analyzed. In the RELAP5 standalone calculation, a point kinetics model was used, while in the coupled code a three-dimensional (3D) neutronics model of QUABOX with different RELAP5 nodalization schemes of reactor vessel was used. Both code versions use best-estimate thermal-hydraulic system code for all components in the plant and include realistic description of plant protection and control systems. Two different types of calculations were performed: with and without automatic control rod system available. The AMFWI transient causes the great asymmetry of the transferred heat in the SGs and subsequently the asymmetry of the power produced across the core due to different reactivity feedback resulting from the thermal-hydraulic channels assigned to different loops. The work presented in the paper is a part of validation of the 3D coupled code R5QC in the analysis of asymmetric transients.

1 INTRODUCTION

Best-estimate codes are required to perform realistic simulation of operational transients and accidents in the NPP. Realistic models are being developed in two directions. First, development of computer codes with the possibility of realistic simulation of interaction between multidimensional neutron kinetic and NPP dynamic behavior enables realistic analysis of transients characterized by strong coupling between neutronics and the thermal-hydraulics in the primary circuit. This is particularly beneficial if asymmetrical processes take place in the core and asymmetrical spatial power distribution results. Nowadays, coupled codes consisting of a system thermal-hydraulic code and a multidimensional neutronic code are being developed in order to accomplish that task. Secondly, development of models of protection and control systems as well as development of the qualified plant nodalization is important for the realistic system analysis of operational transients and accidents. Finally, validation of developed computer codes and of NPP model is necessary to gain more experience in their use in realistic analyses.

At FER Zagreb, a coupled code R5QC consisting of the thermal-hydraulic code RELAP5 and the 3D neutronic code QUABOX is being developed for several years. In the coupled code, RELAP5 calculates thermal-hydraulics and heat conduction in the whole system as well as thermal hydraulics for the average core channels in the core while QUABOX calculates only 3D kinetics. RELAP5 takes the organization role in the coupled code, i.e. synchronization of both parts of the calculation (thermal-hydraulics and neutron kinetics) and of the time stepping. In the coupled code an interface module serves to interchange data between the two codes (nuclear power per neutronic cell calculated in QUABOX are sent to RELAP5, while fuel temperatures and coolant densities calculated in RELAP5 are sent to QUABOX for the calculation of the thermal-hydraulic feedback). Additionally, the interface module serves to transfer the data describing the current position of modeled control rod banks from RELAP5 to QUABOX where the cross sections affected by the presence of control rods are changed. A comprehensive NPP Krško model after power uprate (reactor power 1994 MW) for RELAP5/mod322 has been developed. For the realistic analysis of operational transients and accidents the model encompasses all major control systems.

The analysis of AMFWI transient presented in the paper was performed using RELAP5 standalone code (point kinetics) and 3D coupled code R5QC with different RELAP5 reactor vessel nodalization schemes. Two kinds of analyses were performed: with and without control rod system available. In the analyzed transient asymmetric conditions in the coolant loops lead to asymmetric reactivity feedback in the related core halves which results in asymmetric radial core power profile. The presented analyses are aimed to assess the influence of different methods (point kinetics, 3D coupled code) and reactor vessel nodalization (non-split and split reactor vessel) in the analysis of asymmetric transients. In addition, the analysis is expected to show benefits of use of multidimensional code and detailed split reactor vessel nodalization against simplified point kinetics method. Actions of control systems (steam dump, control rod system) and particularly of loop related control systems (SG level control system) influence final results and potentially increase the asymmetry within the core.

2 CALCULATIONAL MODEL AND MODEL OF THE PLANT NODALIZATION

The model of plant nodalization in both RELAP5 standalone and coupled code calculation is based on standard NEK RELAP5/mod322 nodalization developed at FER for NPP Krško after modernization and steam generator (SG) replacement, [4]. The basic nodalization (RELAP5 standalone code, one thermal hydraulic channel in the core, non-split reactor vessel) has 582 volumes and 614 junctions. The nodalization includes a detailed description of main and feedwater bypass system, [10]. A coupled code calculation was performed with 18 RELAP5 channels in the core, Figure 1 a. The number of subdivisions in the core is 24 for both point kinetics and coupled code calculation. Two reactor vessel nodalization schemes for 3D calculation were used: non-split reactor vessel for use in RELAP5 standalone, and split reactor vessel, Figure 1 b, in order to enable separate thermal-hydraulic feedback calculation from different loops. Total number of heat structures is 408 with total number of mesh points of 2353. In the split reactor vessel model, mixing at the reactor vessel inlet as well as outlet was assumed (Figure 1 b, volumes 103, 104, 105 and 106 for inlet plenum and 121, 122, 125 and 126 for outlet plenum, respectively). The flow coefficients for respective flow paths were adjusted in order to obtain 75 % of cold leg flow delivered to the closer region of the core and 75 % of hot leg flow delivered from the half of the core closer to the loop, respectively. The comprehensive RELAP5 input data set for NPP Krško encompasses the models of all main plant protection and control systems (steam generator level control system, pressurizer pressure and level control system, steam dump). A realistic RELAP5 model of the control rod system has been developed ([8], [9] and [10]). The model enables the simulation of different operation modes: automatic control rod system –

load follow from turbine, rod withdrawal stop and turbine runback, control rod transients (e.g. rod ejection, rod stuck position).

The nuclear data are based on NPP Krško Cycle 17. The nuclear characteristics are calculated at BOL (Beginning of Life, 150 MWD/MTU) and HFP (Hot Full Power) conditions. Boron concentration was determined for ARO (All Rods Out) and EQXE (Equilibrium Xe) conditions and it was 1390 ppm for 3D calculation (For point kinetics 1401 ppm was used.). Delayed neutron data and prompt neutron lifetime were taken from Core design report for cycle 17, [5]. Thermal-hydraulic feedback in 3D coupled code is calculated using thermal-hydraulic dependence of cross sections. The cross section libraries for material compositions were created using CORD-2 package, [7]. The influence of control rods is taken into account using rodded cross section library. Discrete points for cross section data were prepared for the 3D linear surface interpolation: 6 points for fuel temperature change (600 – 1500 K), 6 points for moderator density change (650 – 800 kg/m^3) and 3 points for boron concentration change (800 – 2200 ppm), respectively. Thermal-hydraulic feedback in point kinetic case is modeled using SEPARABL option. Fuel temperature and moderator density reactivity tables were calculated using LEOPARD code from FUMACS package, [6]. Reactivity tables were calculated for 29 discrete values for fuel rod temperature and 28 values of moderator densities. The reactivity weighting factors for both moderator density and fuel temperature feedback were determined using power*power weighting. Inserted control rod worth in point kinetics is obtained from Core design report, [5]. Insertion rate is based on the curves that describe normalized distance versus time and reactivity versus normalized position, respectively, [2].

a) b)

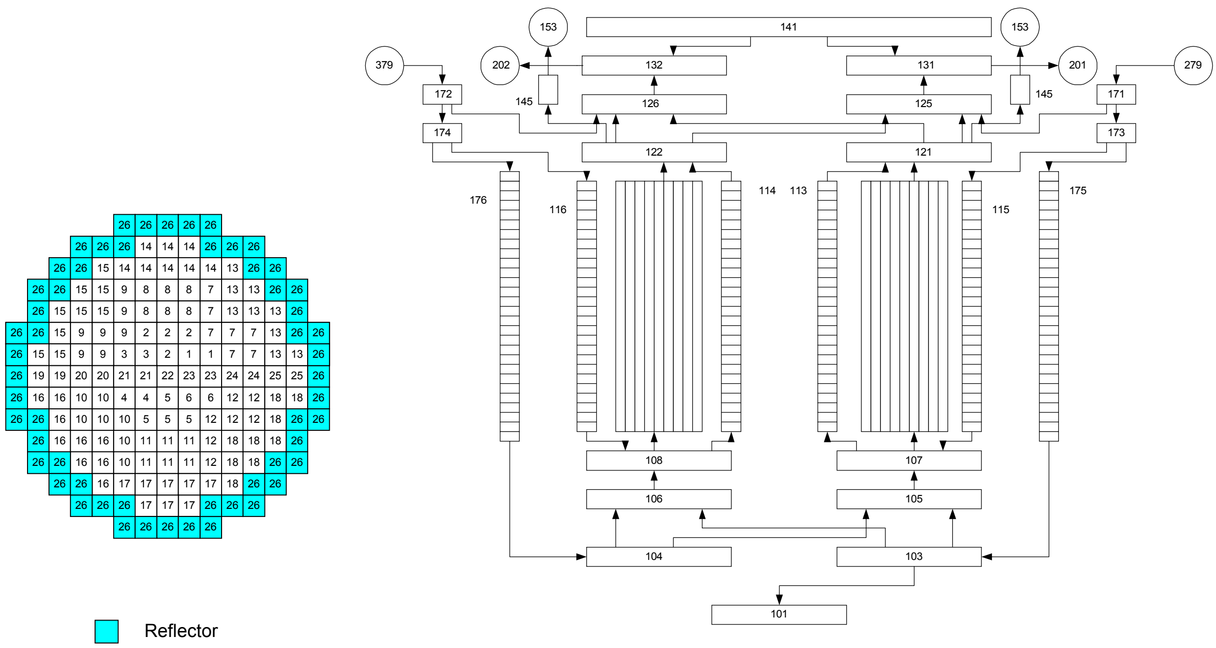

Figure 1: a) Arrangement of thermal-hydraulic channels in the core (coupled code calculation) and **b)** Reactor vessel nodalization (split vessel, coupled code calculation)

3 TRANSIENT ANALYSIS

Asymmetric isolation of main feedwater was initiated in the loop with pressurizer after 200 seconds steady state calculation. An ATWS (Anticipated Transient Without Scram) case was assumed. The protective action in this case is the AMSAC (ATWS Mitigation System Actuation Circuitry) action. It consists of turbine trip and auxiliary feedwater actuation as the level in the affected steam generator falls below 8 %. The aim of this action is twofold. First, primary temperatures increase after turbine trip and nuclear power decreases due to negative temperature reactivity feedback thus alleviating the lack of scram. Secondly, together with turbine trip, auxiliary feedwater is actuated in both loops thus preventing the loss of heat sink on the secondary side. AMSAC action was assumed to be activated with 15 seconds delay. In the unaffected loop it was assumed that main feedwater remains active four minutes after the setpoint for feedwater bypass had been reached (setpoint: turbine power < 15 %). Two different analyses were performed: with (CASE 1) and without control rod system available (CASE 2). The AMFWI transient is characterized by large variations in primary coolant conditions on one side and by asymmetric core inlet and outlet conditions on the other side. The asymmetric reactivity feedback affects the spatial distribution of power produced across the core.

The most important events during simulation are summarized in Table 1. The related labels are for RELAP5 standalone code (point kinetics - POINT) and the coupled code (3D kinetics, non-split reactor vessel – 3D-A, and split reactor vessel – 3D-B). In the point kinetics as well as 3D calculation with non-split reactor vessel, a perfect mixing at the reactor pressure vessel inlet and outlet is assumed. The transient was initiated by an instantaneous closure of main feedwater in the loop with pressurizer (LOOP 1). As a consequence of feedwater closure level in the affected SG drops, Figure 2. During initial phase of the transient, steam flow coming from the affected SG is larger than the flow from the unaffected loop. This is caused by the presence of saturated mixture at the very entrance of the SG heat exchanger section as the main feedwater flow in that SG is terminated. Almost simultaneously for all applied methods at t= 48 s after transient begin, SG level in the affected SG drops below 8 %. Turbine mass flow was assumed to be constant till the begin of AMSAC action. At t= 63 s, a turbine trip together with auxiliary feedwater was actuated in all used methods. Steam dump system was actuated after turbine trip (50 % steam dump valve capacity – turbine trip mode) actuation. The SG level control decreases the feedwater flow in the unaffected loop in order to maintain the SG level at the setpoint value (69.35 %). After loss of inventory in the affected SG, Figure 2, heat transferred to its secondary side decreases rapidly. As a consequence, related cold leg temperature rises, Figure 3. Starting from the time of emptying of affected SG which coincides with the time of AMSAC actuation, differences in results obtained with different methods became greater, Table 1.

Analysis with control rod system available (CASE 1)

Control rod D insertion, Figure 5, started after loss of feedwater because of difference between programmed and measured average temperature. Following a turbine trip, power mismatch channel (turbine - reactor power) in the automatic rod control system generates an impulse negative error signal and control rod insertion (insertion sequence: control rods D, C, B) results. A combination of the negative moderator reactivity feedback after temperature rise, Figure 3, and negative reactivity due to control rod insertion causes reactor power decrease, Figure 4. The resultant coolant temperature and subsequent specific volume decrease causes an outsurge from the pressurizer, primary pressure drop until setpoint for Safety Injection (SI) is reached, Table 1. In the calculation with split reactor vessel, steam dump valves remain open for the longer time period than in other two calculations (point kinetics, non-split reactor vessel) due to higher temperatures in the initial phase of the transient. Contrary to the POINT and 3D-A calculation, in the 3D-B calculation the resultant pressurizer pressure drop led to an earlier SI actuation. In this calculation the Main Steam Line Valve (MSIV) was closed on a signal: high steam line mass flow, low average temperature, SI actuation, Table 1. In the 3D-B calculation higher primary temperatures were reached, Figure 3, and the Over Temperature (OTDT) trip setpoint in the affected loop was overridden, Table 1. Radial core power distribution for the 3D calculation with split reactor vessel at time of greatest asymmetry is presented in Figure 6 a.

Analysis without control rod system (CASE 2)

Nuclear power, Figure 4, decreases after turbine trip only because of negative reactivity feedback. SI signal together with MSIV closure signal is actuated on low steam line pressure signal, Table 1, which is a result of high steam flow from the unaffected SG. This signal protects against steam line break. The difference in times of SI and MSIV actuation for the applied methods is less than in the analysis with rod control system available, Table 1. In the 3D calculation the OTDT protection setpoint was reached in the affected loop due to coolant temperature rise in the initial phase of the transient. In this analysis the absolute change of core conditions and the accompanying asymmetry in the core is less than in the analysis with control rod system available (Figure 6 a and b). Reactor Coolant (RC) pumps were tripped, Table 1, on a sub-cooling signal coincident with SI (60 s delay).

Table 1: Time table of events (seconds), transient begin at t= 0 s: 3D-A non-split reactor vessel, 3D-B split reactor vessel

EVENT	POINT (CASE 1)	POINT (CASE 2)	3D-A (CASE 1)	3D-A (CASE 2)	3D-B (CASE 1)	3D-B (CASE 2)
AMSAC	48.03	48.05	48.02	48.8	47.5	47.8
TURB. TRIP, AUX. FEEDW.	63.07	63.1	63.06	63.12	62.98	63.03
SI	296.97	431.4	293.19	444.69	224.46	447.35
MSIV closure	-	431.46	-	444.74	236.04	447.38
OTDT loop 1	-	-	-	89.9	93.98	87.99
RC pumps trip	-	508.3	-	504.7	-	507.4

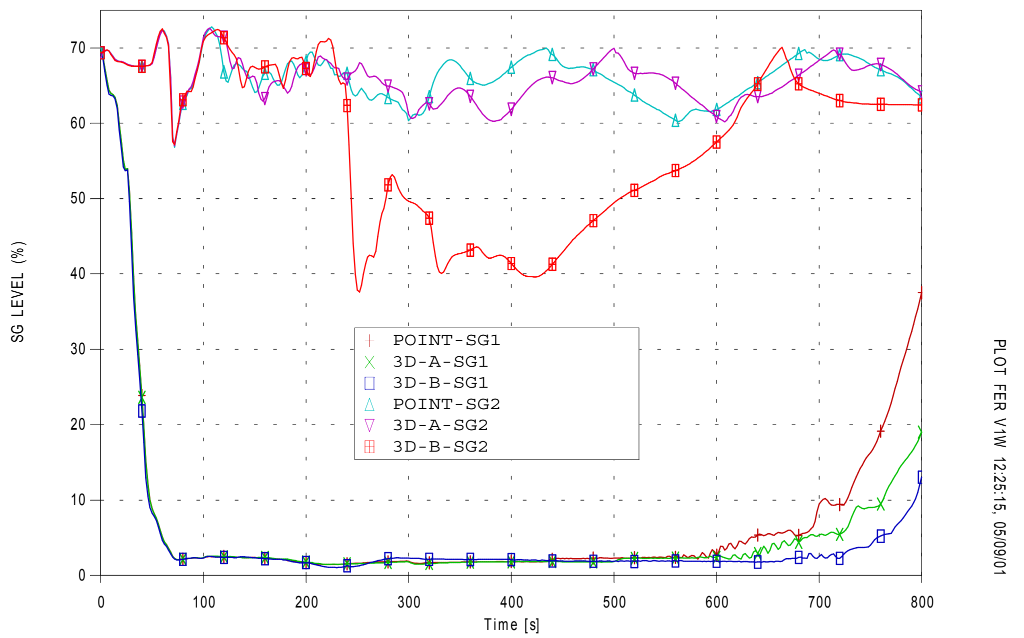

Figure 2: SG level - CASE 1, 3D-A non-split, 3D-B split reactor vessel

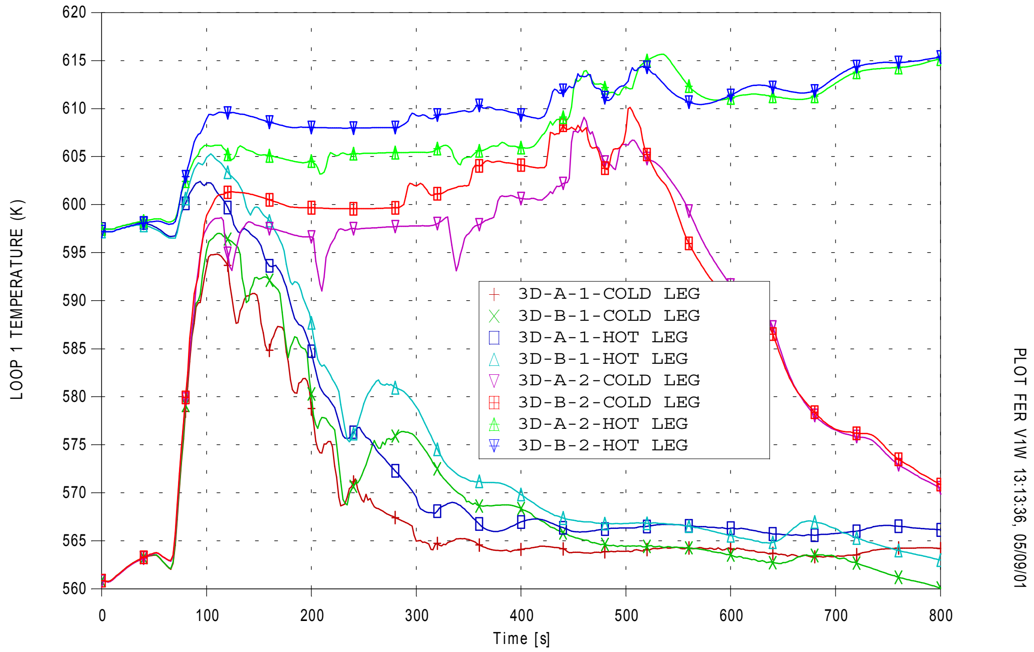

Figure 3: Primary coolant temperature (LOOP 1, 3D calculation, -1 CASE 1, -2 CASE 2, 3D-A not-split, 3D-B split reactor vessel)

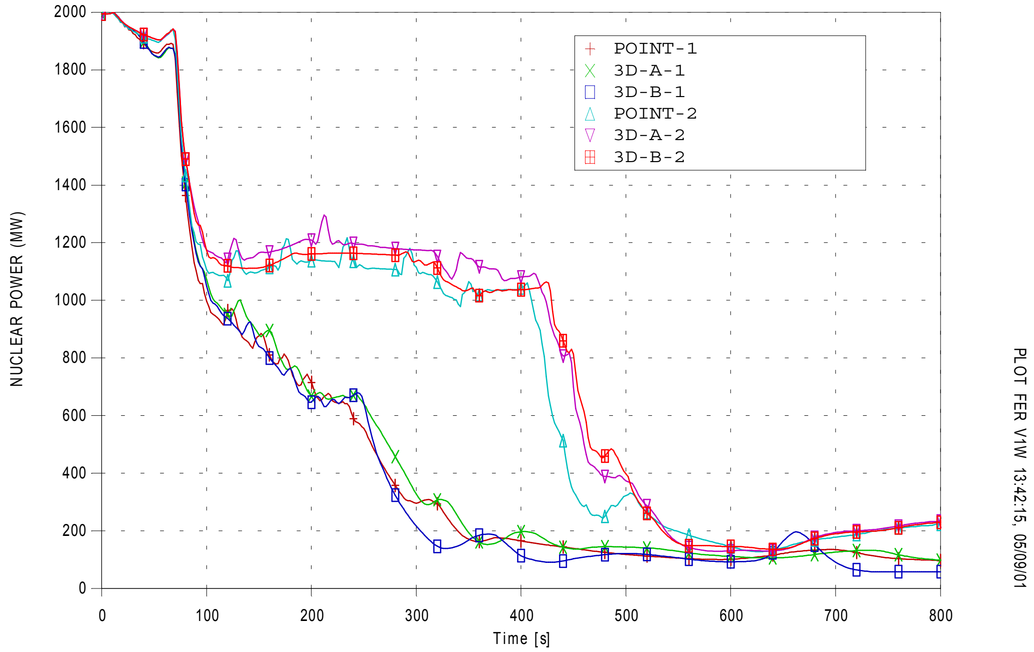

Figure 4: Nuclear power (-1 CASE 1, -2 CASE 2, 3D-A non-split, 3D-B split reactor vessel)

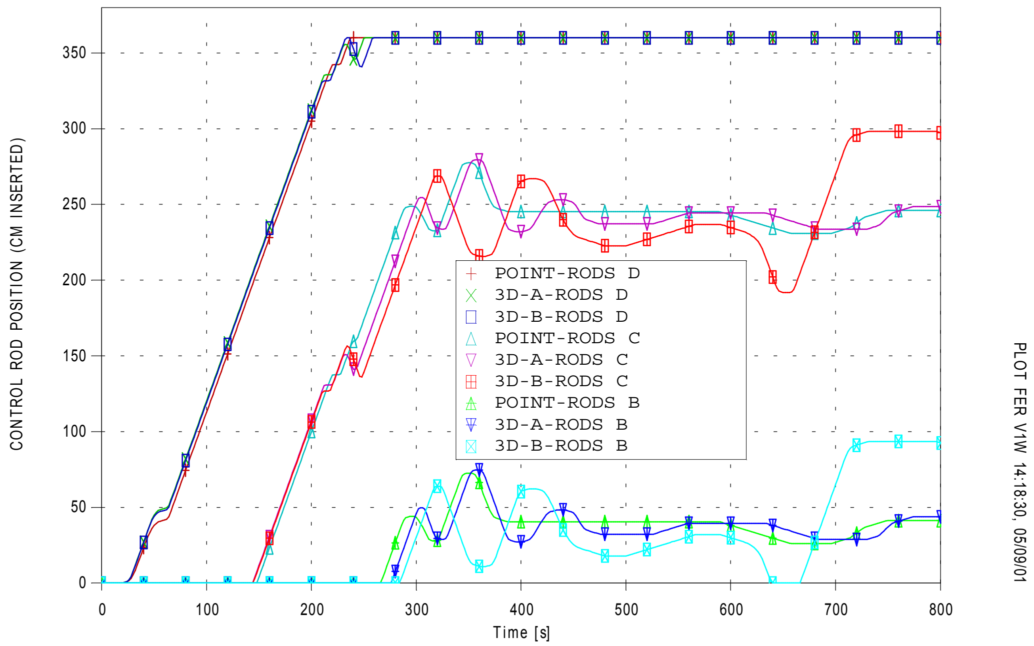

Figure 5: Control rod position – CASE 1, 3D-A non-split, 3D-B split reactor vessel

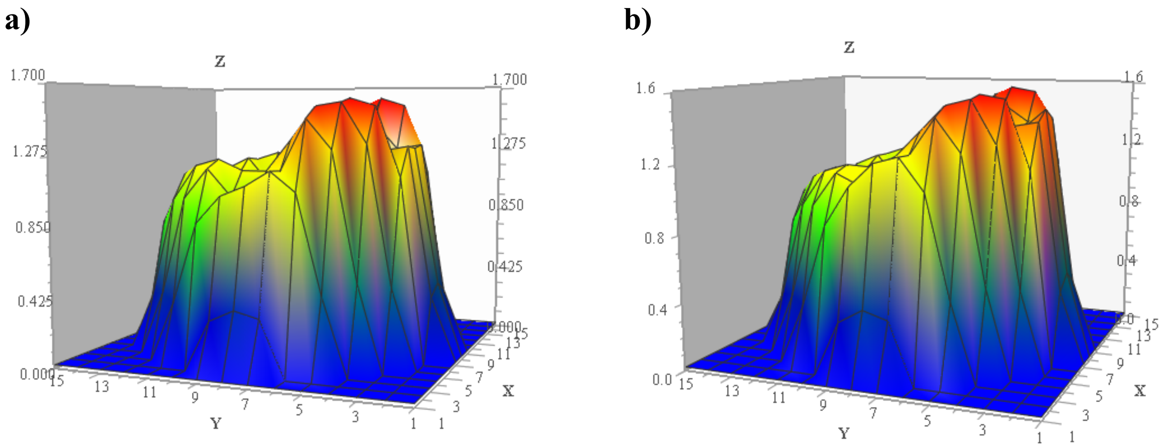

Figure 6: Radial core power distribution for 3D-B calculation at a time of a greatest asymmetry of core conditions (Affected loop is on the left side.) **a)** CASE 1 (t= 165 s) **b)** CASE 2 (t= 173 s)

4 CONCLUSION

In the paper the results of the AMFWI transient analysis are presented. The analysis using RELAP5 standalone (point kinetics) and two 3D coupled code R5QC analyses (non-split and split reactor vessel) with 18 thermal-hydraulic channels in the core were performed. CPU times for 1000 s simulation on Pentium II 350 MHz PC was 1.26 hours for point kinetics and 8.16 hours for 3D calculation with non-split reactor vessel. In the CASE 1 analysis (control rod system available) better agreement between point kinetics and 3D analysis with non-split reactor vessel (3D-A) than between the two 3D analyses (split and non-split reactor vessel) was obtained. The CASE 1 is characterized with larger change in core conditions as well as with larger influence of reactivity feedback than the CASE 2 (control rod system not available). It was shown that the asymmetry across the core was more significant in the CASE

1 than in the CASE 2 calculation. This explains less difference between the two 3D methods in the CASE 2 than in the CASE 1 calculation. The performed analyses have demonstrated the appropriateness of developed NPP Krško nodalization and input data set which includes models of plant control systems in realistic plant analyses.

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