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Multi-physics analysis of CEA drop accident

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1. Introduction

This paper aims to analyze control element drop assembly accident for APR1400 by using a multi-physics approach. Traditionally, this accident is analyzed by using the point kinetics model, where the reactor core is divided into an average and a hot channel with reactivity provided using tables. However, this conservative approach does not reflect the real core behavior. For the purpose of multi-physics analysis, two-way implicit coupling of the thermal hydraulics (TH) code, RELAP5/SCDAP/MOD3.4, and the nodal kinetics (NK) code, 3DKIN 5.2.1, was chosen to achieve a realistic, high fidelity response. This approach is beneficial for accidents with asymmetrical core power distribution, and strong feedback mechanisms such as for reactivity accidents (RIAs).

2. Literature review

Reactivity initiated accidents (RIAs) in nuclear reactor cores are very complex multi-physics transients. The traditional conservative approach used to simulate RIA employs the point kinetics model by using one-way coupling, which simplifies the core and certain important local phenomena cannot be monitored in detail.

According to Park [1] even though it is convenient to use point kinetics model for conservative analyses, the over simplification of this approach leads to significantly poorer representation of the safety margin. In addition to that, comparison of point kinetics approach with a multi-physics simulation using a sub-channel code was investigated by Price et al. Main focus was on the simplifications that are assumed using the conservative approach, and therefore it was found that multi-physics approach can lead to more accurate core parameters tracing (parameters such as core power distribution, pin peaking factors etc.), therefore leading the plant into a safer and more efficient operation [2].

Park et al.[3] used the 3D NK code, ASTRA, the sub-channel analysis code, THALES, and the fuel performance code, FROST in their sensitivity studies for 3D rod ejection. It is worthy to note that the codes are coupled by CHASER system. That complex approach enables realistic safety analysis methodology.

A similar approach is used by Park et al. [4] as they conducted a three-dimensional pin-wise analysis for CEA ejection accident by using the same coupling method. Accident scenarios including malfunction of CEA were analysed by Lee et al. [5] by coupling of CUPID code with MASTER code for multi-dimensional representation of the thermal hydraulics parameters for OPR1000 as a base model. CEA ejection and drop scenarios were analysed and multidimensional multi-physics analysis was successfully performed, clearly showing the non-uniform radial distributions of the core power.

The 12-finger CEA drop accident was investigated by Jeong et al. [6] by using code coupling. MARS TH code was coupled with MASTER to attain more accurate predictions for nuclear system transients that involve strong interaction between neutronics and thermal hydraulics. Detailed analysis related to CEA drop accident using multi-physics approach was conducted by Jonsson et al. [7]. RELAP5 was coupled with a best-estimate nodal reactor kinetics code SIMULATE-3K. Extended analysis of rod drop accidents, assuming that any rod can drop at anytime was represented. The obtained results demonstrated the applicability of coupling codes for RIA analysis as detailed tracking of individual fuel pins allow to trace specific fuel enthalpy and peaking centerline temperatures values more accurately.

3. Model description

3.1 Thermal hydraulics model description

For the purpose of accident analysis, a thermal hydraulics model of APR1400 is developed using RELAP5/SCDAP/MOD3.4. Nodalization representing all key system and components is shown in Figure 1. The model includes a Reactor Pressure Vessel (RPV) as a central component with bypass channels to realistically represent the flow through the core. The Reactor Coolant System (RCS) with four Reactor Coolant Pumps (RCPs) – one per each cold leg – to circulate the coolant. Two Steam Generators (SG) are included – one for every loop – to dissipate the heat generated in the core and generate steam. The pressurizer is connected to one of the hot legs via a surge line to maintain the system pressure. To simplify the pressurizer model, the effect of maintaining the pressure is achieved by connecting a time-dependent volume to the top of the pressurizer.

In order to maintain the water level in the SGs, the Main Feedwater System (MFWS) is represented using time dependent-volumes connected to the downcomer and economizer regions. For a realistic representation, the flow is split so that 10 percent of the full-power feedwater flow is directed to the downcommer while the remaining flow is directed to the economizer.

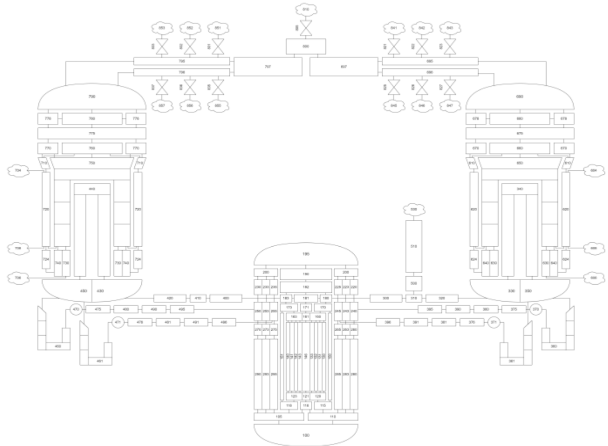

Fig. 1. APR1400 model nodalization

3.2 Nodal kinetics model description

The nodal kinetics model is an extended version of the point kinetics model allowing more detailed core representation. Unlike in conservative analysis, where reactor core is divided into hot and average channel, for the multi-physics analysis purposes, nuclear reactor core is modeled in detailed way by using 3DKIN 5.2.1 nodal kinetics code. The core is divided into 241 axial sections – every section representing a single fuel assembly (FA) and 60 axial nodes, including axial reflector. Using such finer discretization allows prediction of the reactor response in real-time with high fidelity. It is worth noting that the arrangement of fuel assemblies is determined in accordance to the core design for the first cycle for APR1400 [8], as shown in Figure 2.

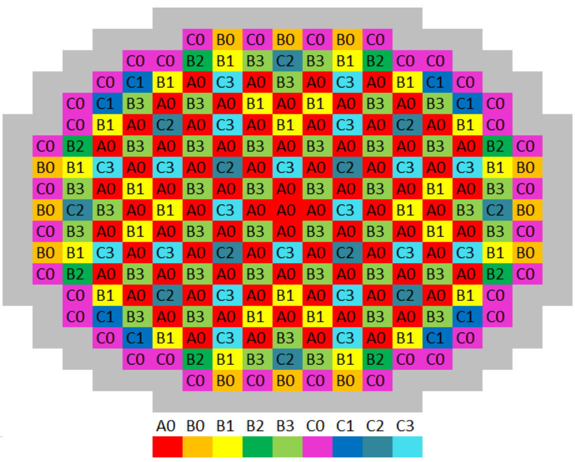

Fig. 2. Loading pattern for NK model

APR1400 consists of 241 FA divided into 9 different groups, based on enrichment, burnable absorber presence etc [8]. Detailed parameters of each group are shown in Table I.

Table I: Fuel Assemblies Parameters

Assembly Type	Number of Fuel Assemblies	Fuel Rod Enrichment (w/o)	No. of Rods Per Assembly	No. of Gd_2O_3 Rods per Assembly	Gd_2O_3 Contents (w/o)
A0	77	1.71	236	-	-
B0	12	3.14	236	-	-
B1	28	3.14/2.64	172/52	12	8
B2	8	3.14/2.64	124/100	12	8
B3	40	3.14/2.64	168/52	16	8
C0	36	3.64/3.14	184/52	-	-
C1	8	3.64/3.14	172/52	12	8
C2	12	3.64/3.14	168/52	16	8
C3	20	3.64/3.14	120/100	16	8

In order to properly represent each group of FAs, physics details such as the two-group macroscopic cross-sections for transport, absorption, fission and scattering needs to be provided to 3DKIN 5.2.1. Those parameters were generated by using CASMO-3 lattice code for each assembly. All those cross-sections are provided with all control rods inserted (all rods in) and withdrawn (all rods out). To reflect the movement of CEAs and reactivity changes during these movements, 3DKIN 5.2.1 interpolates between those two positions. Similarly, the moderator temperature coefficients (MTC) and Doppler reactivity coefficients (FTC) are reflected, based on calculations for different fuel and moderator temperatures using the cross section libraries specified by CASMO-3.

3.3 Two-Way Code Coupling

For the Multiphysics simulation, RELAP5/SCDAP/MOD3.4 code is internally coupled with 3DKIN 5.2.1 allowing two-way data exchange as shown schematically in Figure 3. In order to enable this implicit coupling and information exchange, TH volume nodes are mapped with appropriate NK core structures in the input deck.

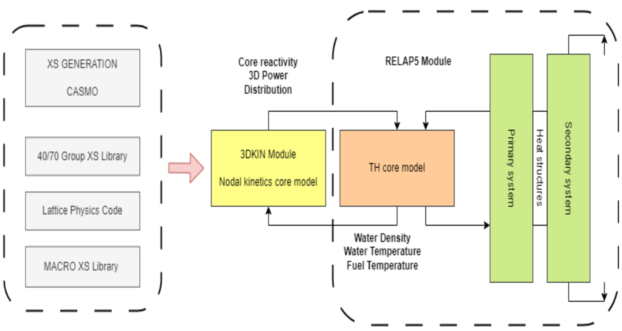

Fig. 3. Two-way coupling diagram

4. Accident description

The transient is initiated by the release and subsequent drop of a single CEA, due to an interruption in the electrical power to the control element drive mechanism (CEDM) holding coil of a single CEA. This interruption can be caused by a holding coil failure or loss of power to the holding coil. The transient initiates a reduction in core power and primary to secondary side power-to-load mismatch. This mismatch results in a cooldown of the Reactor Coolant System (RCS) due to excess heat removal by the secondary side. Negative moderator temperature coefficient (MTC) adds positive reactivity and the core power tends to return to initial power level. The limiting case is the single CEA drop that does not cause a reactor trip but results in an approach to specified acceptable fuel design limit (SAFDL) on the DNBR [9].

5. Results and discussion

In this section, the results from the point kinetics and the nodal kinetics models will be presented. This section is divided into three parts. The first part will be dedicated to the validation of CEA drop accident against those reported in the Design Control Document (DCD) at steady state using the conservative approach. Next, the nodal-kinetics model will be validated against the nominal conditions of APR1400, to assess the model credibility. Finally, the transient results of the multi-physics analysis will be presented.

5.1 Point kinetics model validation

To validate the model against the results reported in the APR1400 DCD, the initial conditions of the model were adjusted accordingly to represent the worst-case scenario. A summary of the steady state conditions is presented in Table II.

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Table II: Initial conditions for the Single CEA Drop [9]

Parameter	DCD	Simulation
Core power, MWt	4062.66	4062.66
Core inlet temp, ^{0}C	295.0	295.77
Core mass flow rate, 10^6 kg/hr	69.64	69.64
Pressurizer pressure, kg/cm^2	152.9	153.1
SG pressure, kg/cm^2	75.86	75.84
Integrated radial peaking factor	1.37	1.37
ASI	-0.3	-0.3
Initial minimum DNBR	1.81	1.8135
MTC and Doppler reactivity	Most negative	Most negative

Next, the transient response is traced as a function of time to validate the model under CEA drop accident conditions. The evolution of key system parameters are reported and compared to those of DCD as shown in Figures 4-8. Namely: core power, heat flux, pressurizer pressure, steam generator pressure and DNBR.

Due to the reactivity insertion upon dropping the CEA, the core power and core heat flux decreases rapidly. However, due to the strong reactivity feedback the reactor power is brought back to the initial conditions. Due to the mismatch, RCS cooldown is observed, along with drop of pressure in primary and secondary system.

Hot channel minimum DNBR is dropping significantly, as a result of pressure drop and dropped CEA that disturbs and restrict the coolant flow rate in that region, creating favourable conditions for reduction of heat transfer capabilities.

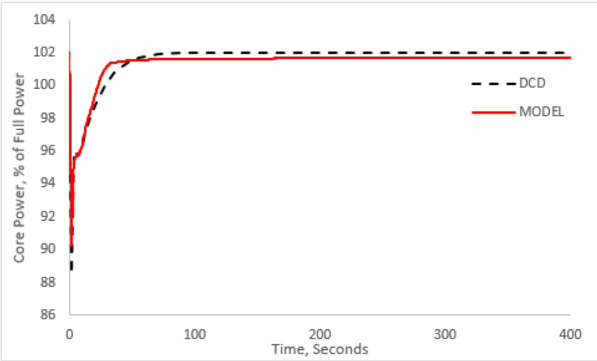

Fig. 4. Core power vs time

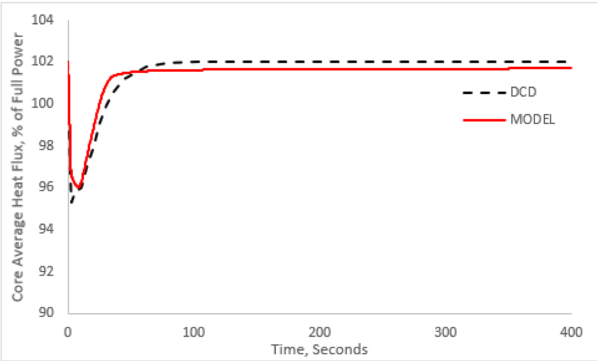

Fig. 5. Core average heat flux vs time

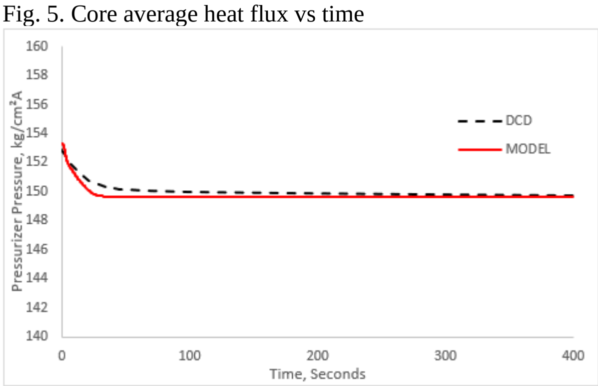

Fig. 6. PZR pressure vs time

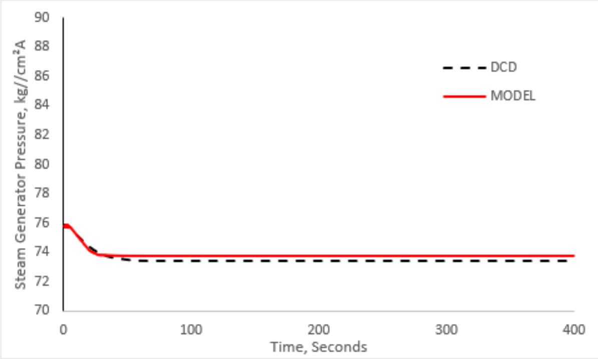

Fig. 7. SG pressure vs time

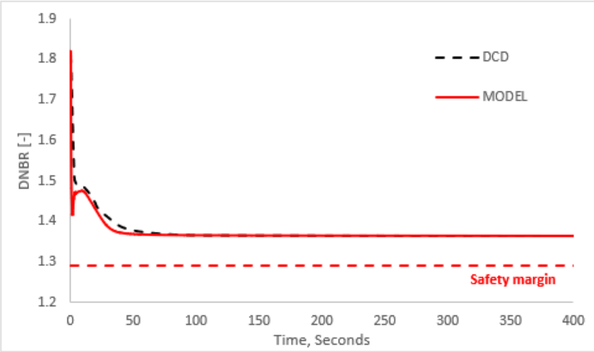

Fig. 8. DNBR vs time

Qualitatively the model follows the NPP transient response quite well; however, quantitatively, some discrepancies can be observed. This may be attributed to the fact that the conservative analysis is conducted by using the conservative code CESEC-III [9], whereas the code used in this analysis is the best estimate code, RELAP5/SCDAP/MOD3.4.

It is important to note key parameters such as the maximum system pressure and the minimum DNBR. The timestamp for key events is shown in Table III.

Table III: Sequence of events for the Single CEA Drop [9]

Time (sec)	Event	Value (DCD)	Model
0.0	A single CEA begins to drop	-	
0.0	Max. PZR pressure, kg/cm^{2}A	152.9	153.1
382.5	Minimum DNBR	1.36	1.3645

5.2 Nodal kinetics model validation

To validate the nodal kinetics model, a simulation was conducted using nominal power conditions for APR1400 reported in DCD [9] and compared with planar average power distribution for unrodded core. In Figure 9, the power distribution for a quadrant core is represented with percentage deviations, compared to DCD.

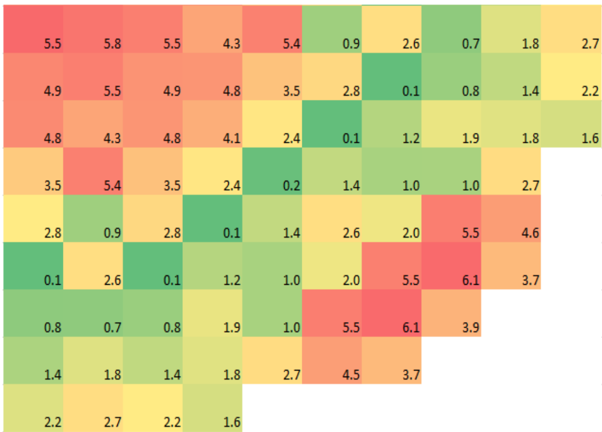

Fig. 9. Deviation (in %) in core power distribution using two-way coupling

The highest deviations occur in the central and outer parts of the core, with a maximum value of 6.1 %. Considering differences in model and approach along with simulation tools and limitations, results are assumed to be in reasonable agreement.

5.3 Multi-physics simulation of CEA Drop accident

Once the credibility of the nodal kinetics model has been established, the multi-physics transient simulation of CEA drop accident was conducted. Results, including comparison with point kinetics model are shown in Figures 10-14.

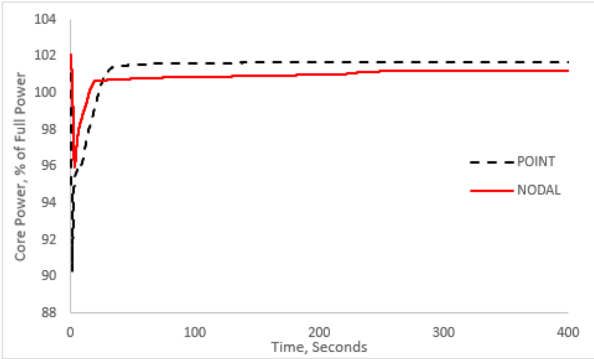

Fig. 10. Core power vs time

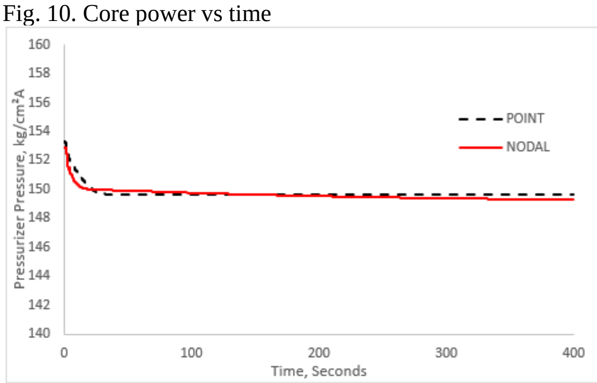

Fig. 11. PZR pressure vs time

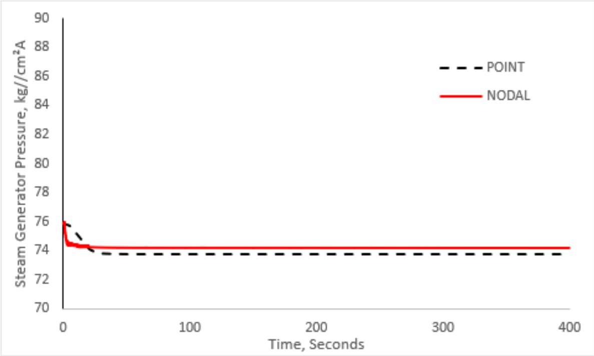

Fig. 12. SG pressure vs time

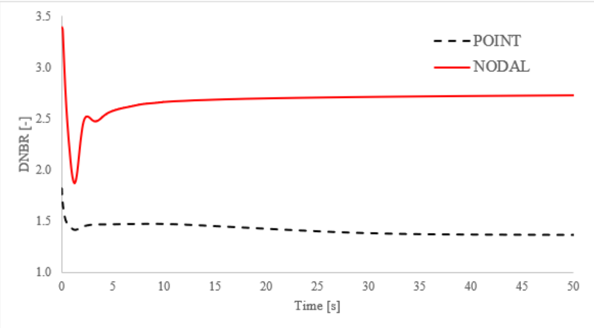

Fig. 13. DNBR vs time (50s)

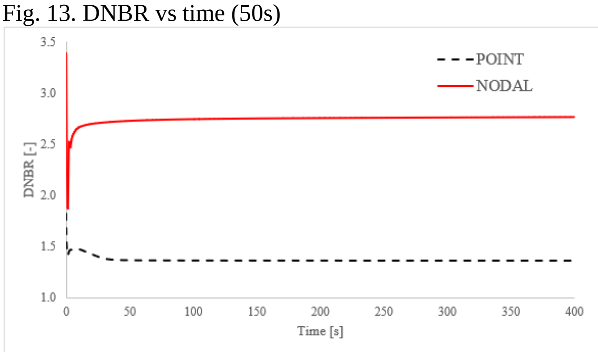

Fig. 14. DNBR vs time

Comparing the results obtained from the multi-physics simulation to those of the point kinetics using the conservative analysis, some differences are visible. The power decrease attributed to the negative reactivity insertion from the dropped CEA is not as strong as in the conservative analysis using the point kinetics model, due to differences in modelling the core. For the core model developed using 3DKIN 5.2.1, the reactivity feedback related to the CEA insertion is based on the cross-sections data prepared in CASMO-3, not on the reactivity insertion values used by the point kinetics model for CEA worth which is based on assuming the worst-case scenario. It should be noted that the power excursion does not reach 102% as in the conservative analysis, but rather stabilizes at a lower value, around 100.8%. Given that the return to power is a safety concern, all parameters are set up in the conservative analysis to maximize the power. However, with a more realistic representation of the core, the multi-physics simulation results in a bigger margin, based on accurate MTC and Doppler reactivity feedbacks that are not chosen conservatively.

Predicted pressurizer pressure for Multiphysics analysis is closer to conservative analysis, as drop is more linear that the one obtained from point kinetics. It

DNBR drop corresponding to rapid insertion of CEA is clearly visible, reaching a minimum value of 1.875, which is higher than the value obtained for conservative analysis, hence multi physics analysis provides higher safety margin, based on real core representation, rather than simplified model with hot and average channel.

Figures 15-17 represent the detailed results obtained using RELAP5/SCDAP/MOD3.4 with 3DKIN 5.2.1, by showing the core power distribution during the accident. The position of dropped CEA is highlighted (using a black box) on all figures. The effect of negative reactivity insertion is clearly visible and affects the core power distribution significantly by lowering the peaking factors in the quadrant in which CEA drop appeared.
Using the detailed core representation confirm the asymmetric power distribution character of this accident which cannot be reflected using the simplified point kinetics model.

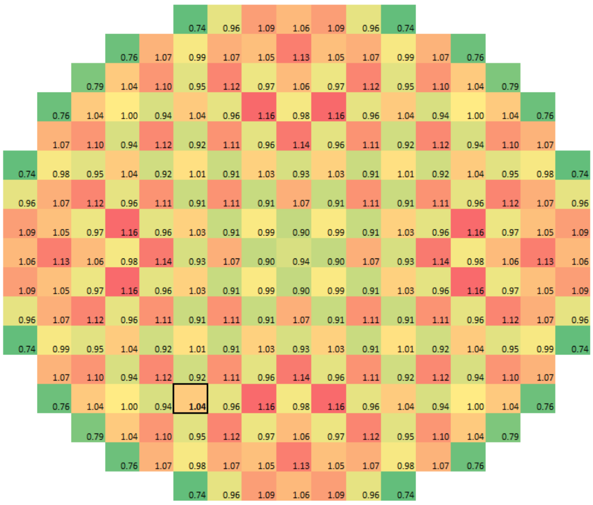

Fig. 15. Core power distribution at the beginning of the accident (time= 0s)

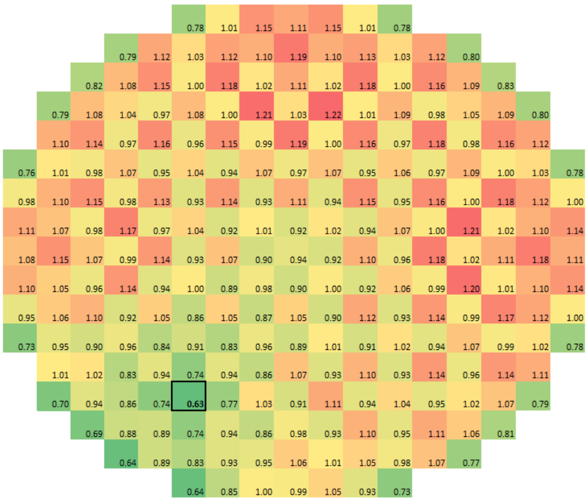

Fig. 16. Core power distribution at the end of CEA drop (time=2s)

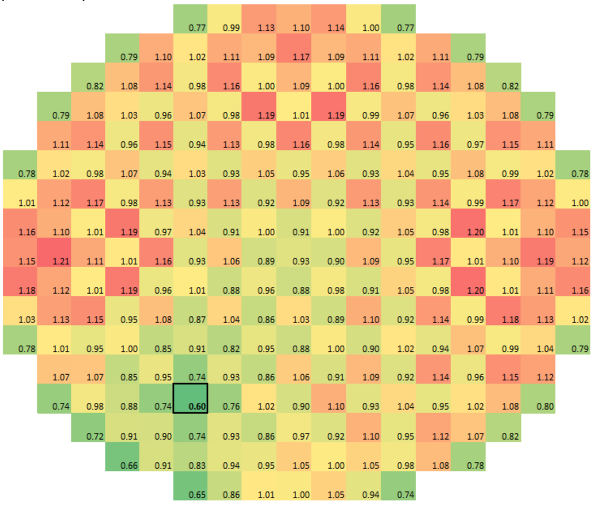

Fig. 17. Core power distribution at the end of transient (time= 400s)

6. Conclusions

In this paper, a control element assembly (CEA) drop accident was investigated, with APR1400 reactor as the modelled power plant. A multi-physics approach is adopted using two-way implicit code coupling of RELAP5/SCDAP/MOD3.4 and 3DKIN 5.2.1.

The development of the project was divided to several stages. First, a thermal hydraulic model with point kinetics was developed to match the conservative analysis for validation purposes. For this model, the core was divided into a hot and average channel. The model captures the NPP response with reasonable agreement.

Next, the simplified core model was replaced by a more detailed core structure, which was developed using 3DKIN 5.2.1 along with CASMO-3 lattice code. In order to allow code-coupling and information exchange, the meshing process was conducted by mapping the thermal hydraulic volumes with nodal kinetics structures.

Finally, a multi-physics analysis of the CEA drop accident was conducted and the results were generated and investigated. Thanks to the multi-physics analysis, the three-dimensional core power distribution and detailed core behavior were predicted with high fidelity and results confirm higher margin for DNBR.

Additionally, this work can be expanded to Best Estimate Plus Uncertainty (BEPU) analysis in conjunction with the multi-physics simulation to estimate the safety margins more realistically. Compared with the traditional conservative approach using high fidelity simulations lead to a larger safety margin and hence more flexible and economical operation of nuclear power plant.

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