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Analysis of CEA Ejection Scenario for the Korean APR1400 Reactor using BEPU Approach

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1. Introduction

This project aims to analyze CEA ejection accident on the APR -1400. As a first step, the accident is simulated on RELAP5/MOD3.4, using one way coupling of the thermal hydraulics model with a point kinetics model. To reflect the realistic behavior of the plant, it is intended to develop a full core model with real-time reactivity feedback from the neutronics code, 3DKIN. Though a multi-physics approach is deemed more suitable for this analysis due to the asymmetric nature of this accident, simple analysis based on point kinetics and the best estimate plus uncertainty quantification provides valuable insight and helps establish the foundation for a high-fidelity model using multi-physics approach.

2. Literature review

Until recently, safety analysis of NPPs have predominantly relied on the conservative approach especially for Generation II reactors. This was due to lack of knowledge and limited computation resources, with the conviction that biasing the system to create the worst-case scenario is the safest route [1]. While these methods have served the nuclear energy industry well in the 70s, with new knowledge accumulated over the past few decades, newer methods have shown that it is possible to assess the realistic NPP response with minimum conservatism and the plant would still be safe. Best Estimate Plus Uncertainty (BEPU) provides a more realistic alternative to the widely used conservatism. Although relatively new, BEPU is an established approach, and has been used in licensing activities of certain accidents at Angra-2 in Brazil, Kozloduy-3 VVER-440 in Bulgaria, Smolensk-3 RBMK in Russia, Balakovo-3 VVER-1000 in Russia, Atucha-2 in Argentina, and others [2]. By using realistic assumptions and data considering various uncertainties, BEPU allows more flexible and economic NPP operation and effective accident management [3].

The CEA ejection accident belongs to a group of accidents called reactivity-initiated accidents (RIA). Uncontrolled ejection of CEA results in a positive reactivity insertion, which causes rapid increase in power, an increase in fuel temperature and thermal expansion of fuel pellets. Reactivity is first lessened by Doppler feedback and subsequent reactor trip [4]. The CEA ejection accident is classified as a reactivity initiated and power anomaly design basis accident.

The time dependent diffusion theory is used to analyze the dynamic behavior of the reactor and often replaced with the quasi-static, multipoint or one point kinetic in conjunction with thermal-hydraulic calculations [5]. Given the nature of the accident at hand, reactivity feedback from the fuel and coolant are important in investigating RIAs, hence the need to couple the thermal hydraulic model with a three-dimensional neutronics model of the core using relevant cross sections to reflect the real-time feedback mechanisms. However, as a starting point, this work focuses on using the point kinetics model to evaluate the thermal hydraulic response under the effect of various uncertainties. It is intended to develop this work further using a multi-physics approach.

3. Methodology

This section describes the methodology applied in this work which involves two basic steps namely the development of a thermal-hydraulic model, and an uncertainty quantification framework as illustrated in Figure 1.

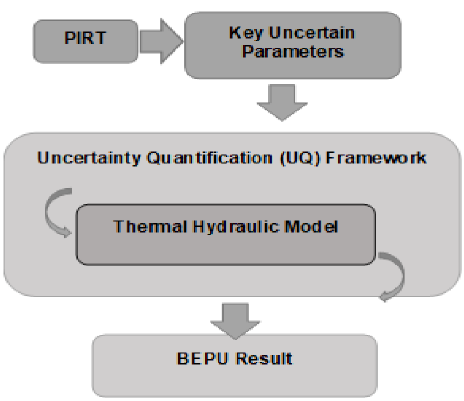

Figure 1: Overview of Methodology

3.1. Thermal-Hydraulics Model

The thermal-hydraulics model is developed in RELAP5/MOD3.4 system code, developed by the U.S NRC [6], to simulate the NPP response under CEA ejection accident scenario. The nodalization shown in Figure 2 contains key systems and components of the Korean Advanced Power Reactor with 1400 MWe nominal power.

APR1400 nuclear power plant is a two-loop pressurized water reactor (PWR). On the primary side, the Reactor Coolant System (RCS) with a reactor pressure vessel (RPV), hot legs, cold legs, reactor circulating pumps (RCPs), pressurizer (PRZ) and two steam generators (SGs) along with main steam lines and safety valves are represented using appropriate thermal-hydraulics components. The core is modelled with inlet and outlet nozzles, downcomer, and lower and upper plenum as part of the reactor vessel. The reactor core is represented using an average channel and a hot channel, each is discretized using 20 vertical nodes. The reactor consists of 241 fuel assemblies, the CEAs are made of boron carbide (B4C) and are located within the fuel assemblies.

CEA ejection is assumed to be caused by mechanical failure resulting in rupture of the control element drive mechanism (CEDM), and therefore subsequent full withdrawal of the CEA and drive shaft caused by pressure of the RCS. Loss of offsite power (LOOP) is assumed to coincide with the turbine trip that follows the accident.

Parameters of interest for this accident are peak fuel rod temperature and fuel rod enthalpy, for evaluation of cladding temperature failure, pellet cladding mechanical interaction (PCMI) failure and core coolability. For thermal hydraulic modelling in RELAP5/MOD3.4 is coupled with nodal kinetics using 3DKIN based on NESTLE [7]. Two-way coupling is required because of the asymmetric nature of RIAs as feedback is required from the thermal hydraulic and kinetic models. However, in this work, a simple model was developed based on point kinetics as a first step.

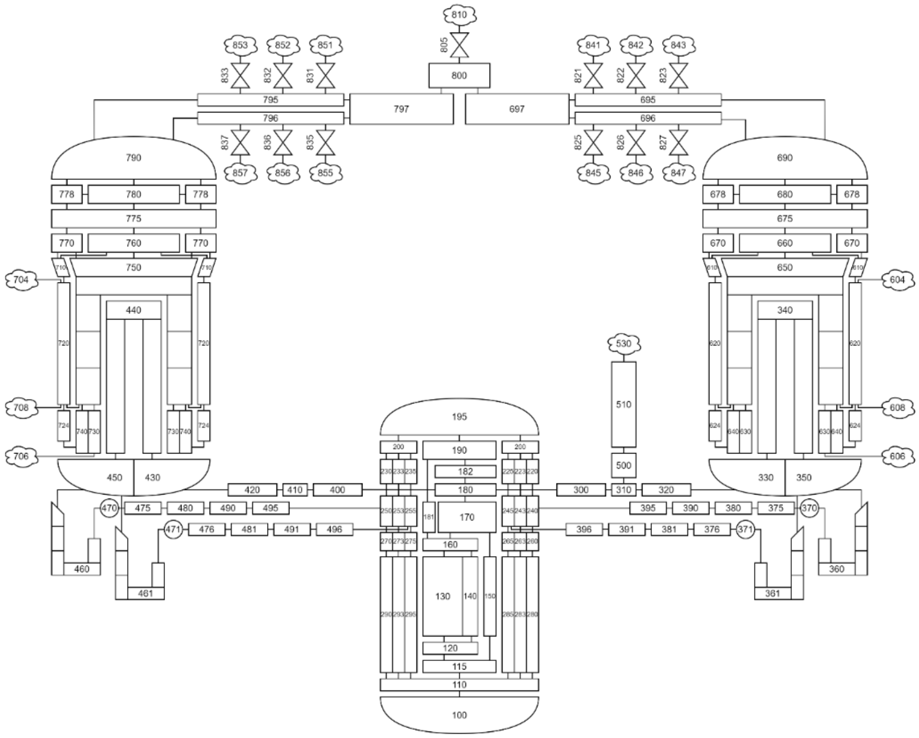

Figure 2: System Nodalization

For conservatism, the negative reactivity insertion due to the control rod worth is not taken into consideration according to the APR1400 Design Control Document (DCD). The system is initialized using the nominal conditions provided for CEA ejection in Chapter 15 of APR1400 DCD [4], which are listed in Table 1.

3.2. Uncertainty Quantification Framework

The BEPU methodology is applied by propagating key uncertain parameters into the thermal-hydraulics model as listed in Table 2. These uncertain parameters are derived from the Phenomena Identification and Ranking Table (PIRT) developed for reactivity-initiated accidents [8]. A total of 19 parameters were chosen and divided into 4 categories according to the PIRT, based on their significance (Table 2). Each parameter was assigned a mean value (μ), a standard deviation (σ), a probability distribution function (PDF), and a range (min ~ max).

Table 1: Conservative Conditions for CEA ejection at Full Power **[4]**

Parameter	**HFP**
Core Power level, MW_{th}	4062,66
Delayed neutron fraction β	0,00412
MTC (10-4) $\Delta\rho/^{o}C$	0
Doppler temperature coefficient $\Delta\rho/\sqrt{K}$	-0,0013

Ejected CEA worth 10-2 $\Delta\rho$	0,1459
post ejected 3d peaking factor	4,32
Ratio of the 3-d post ejected to pre-ejected peaking factor	3,17
Total CEA worth available for insertion on reactor trip, 10^{-2} $\Delta\rho$	-5
Postulated CEA ejection time, sec	0,05
Core Inlet coolant Temperature ^{o}C	295
Core mass flow rate, 10^6 kg/hr	69,64
Pressurizer pressure, kg/cm^{2}A	152,9

Table 2: Uncertain Parameters for Uncertainty Quantification **[8]**

PIRT	Uncertainty parameter (unit)	μ^*	σ^{**}	PDF	min	Max
Manufacturing Tolerances	Cladding outside diameter (mm)	9.40	0.01	Normal	9.38	9.42
	Cladding inside diameter (mm)	8.26	0.01	Normal	8.24	8.28
	Fuel theoretical density (kg/m^3 at 20°C)	10970	50	Normal	10870	11070
	Fuel porosity (%)	4	0.5	Normal	3	5
	Cladding roughness (µm)	0.1	1	Normal	10-6	2
	Fuel roughness (µm)	0.1	1	Normal	10-6	2
	Filling gas pressure (MPa)	2.0	0.05	Normal	1.9	2.1
Boundary Conditions	Coolant pressure (MPa)	15.500	0.075	Normal	15350	15650
	Coolant inlet temperature (°C)	280	1.5	Normal	277	283
	Coolant velocity (m/s)	4.00	0.04	Normal	3.92	4.08
Power & Reacti	Core Power (MW)	3982	39.82	Normal	3909	4062
	Reactivity insertion rate ($/s)	54.3e^{-4}	9.07 e^{-4}	Normal	36.2e^{-4}	72.4e^{-4}
Thermo-Physical Properties & Heat Transfer Models	Fuel thermal conductivity	1.00	5%	Normal	0.90	1.10
	Clad thermal conductivity	1.00	5%	Normal	0.90	1.10
	Fuel thermal expansion	1.00	5%	Normal	0.90	1.10
	Clad thermal expansion	1.00	5%	Normal	0.90	1.10
	Clad yield stress	1.00	5%	Normal	0.90	1.10
	Fuel heat capacity	1.00	1.5%	Normal	0.97	1.03
	Clad-to-coolant heat transfer	1.00	12.5%	Normal	0.75	1.25

$^*\mu$*: mean* $^{**}\sigma$*: standard deviation*

For uncertainty quantification, the thermal hydraulic input decks for RELAP5/MOD3.4 is coupled with DAKOTA software, developed by Sandia National Laboratories [9], using a python interface. The uncertain parameters derived from the PIRT are entered into DAKOTA, to produce a combination of parameters that are chosen at random by the software using Monte-Carlo sampling technique. Those randomly chosen parameters are passed to the thermal-hydraulic code, RELAP5/MOD3.4 to simulate the CEA ejection accident until a statistically representative sample is achieved. The simulation is run multiple times to acquire a large sample. Once processed in DAKOTA a database with NPP response is generated. Figure 3 illustrates the uncertainty propagation framework.

For BEPU analysis, either Monte Carlo or Wilks theorem can be used [10]. The fifth order Wilks method is adopted in this work as it satisfied the 95% probability with 95% with reasonable computational efficiency.

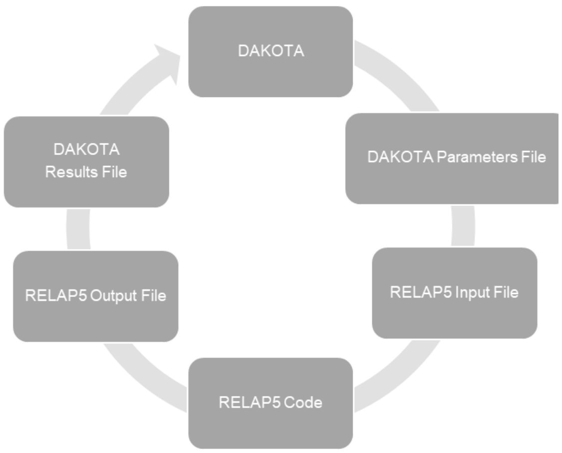

Figure 3: Uncertainty Propagation Framework

4. Preliminary Results

Initial and boundary conditions in the model were verified against DCD values. Table 3 indicates that the model was acceptable due to minimal deviation from DCD values. The system response was evaluated by comparing results from the model against DCD graphs. Figure 4 - 8 is an illustration of the comparison.

Initial conditions were met for the steady state model and the accident response reported in the DCD, Figure 4 shows a slight but acceptable quantitative difference. In Figure 5 the same power behavior was noted for the hot channel. This can be attributed to the manner in which the core was modelled. The core was modelled as an average channel, and hot channel. The power of the hot channel was peaked to reflect the hot part of the core. The behavior of heat flux in the average channel and core Figure 6 and Figure 7 showed qualitative and quantitative difference, however these will be improved during continuation of this project. Further, the results of the BEPU analysis and uncertainty quantification will be compared to those of the conservative analysis and reflected in the final paper.

Table 3: Steady State System Response for CEA Ejection at Full Power

Parameter	DCD	Model	Deviation (%)
Core Power level	4062,66	4062,66	0
Delayed neutron fraction β	0,00412	0,00412	0
Moderator temperature coefficient MTC (10-4) Δρ/ºC	0	0	0
Doppler temperature coefficient Δρ/K	-0,0013	-0,0013	0
Ejected CEA worth 10^{-2} Δρ	0,1459	0,1459	0
post ejected 3d peaking factor	4,32	4,32	0
Ratio of the 3-d post ejected to pre-ejected peaking factor	3,17	3,17	0
Total CEA worth available for insertion on reactor trip, 10^{-2} Δρ	-5,00	-5,00	0
Postulated CEA ejection time, sec	0,05	0,05	0
Core Inlet coolant Temperature K	568,15	568,65	0,0880
Core mass flow rate, 10^{6} kg/sec	19344,44	19349,0	-0,0002
Pressurizer pressure, kg/cm^{2}A	152,9	152,46	0,0029

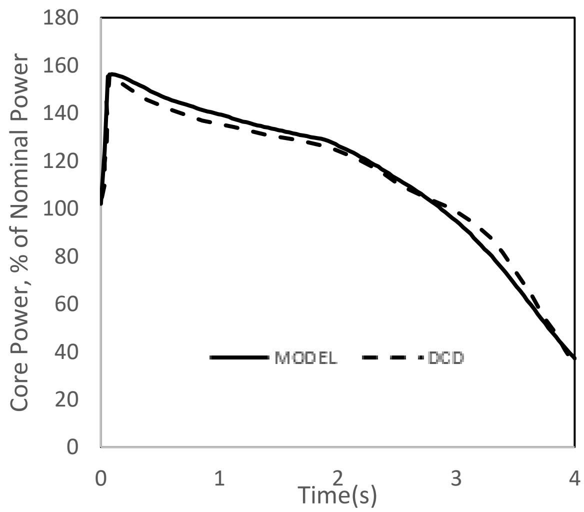

Figure 4: Core power comparison between DCD and simulation

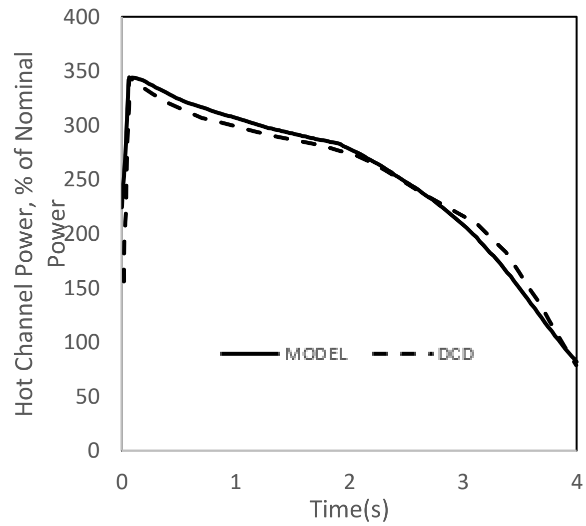

Figure 5: Hot channel power comparison between DCD and simulation

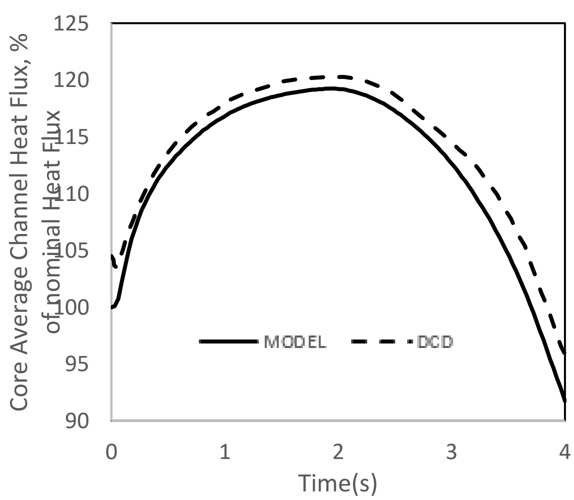

Figure 6: Average channel heat flux

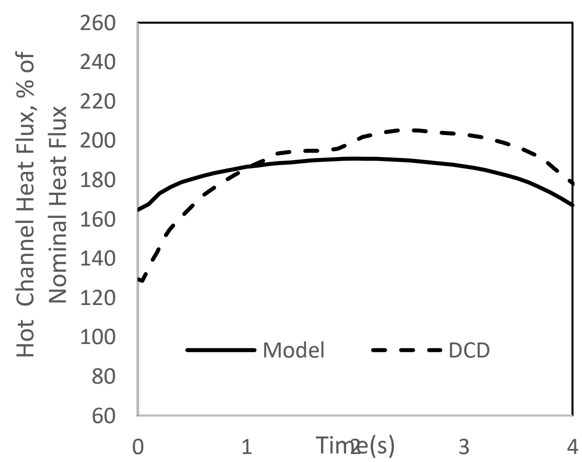

Figure 7: Hot channel heat flux

5. Conclusion

This project aims to model the CEA ejection accident on the APR-1400. As the initial step of this study, one way coupling of thermal hydraulics model with point kinetics provided insight on the behavior of the core during RIAs. The progression of the study provides further understanding and knowledge of simulations involving three-dimensional reactivity feedback in real-time. To achieve this goal, a more sophisticated model is required to realistically simulate the feedback between the thermal hydraulics and neutronics model by coupling the thermal hydraulics code, RELAP5/MOD3.4, with a nodal kinetics code, 3DKIN, which is being implemented. Furthermore, the study will quantify uncertainty using a statistical tool, DAKOTA. The results of this study will be prediction of a more realistic NPP response for the CEA ejection accident.

ACKNOWLEDGEMENTS

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