

Multiphysics Analysis of CEA Withdrawal at Power for the Korean APR1400 Reactor

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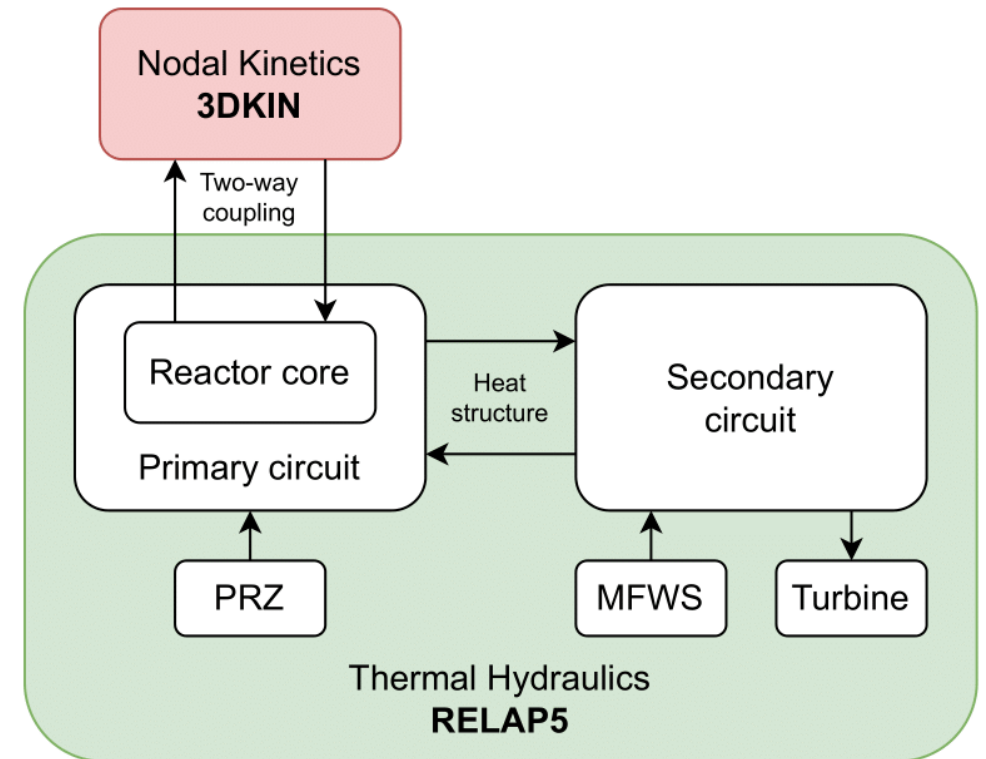
KNS Autumn Meeting 2022

Presentation outline

- Introduction
- Accident description
- Model description
- Methodology
- Results
- Conclusion

Research goal

To conduct a **multi-physics simulation** of CEA withdrawal accident for a more **realistic prediction** of the system performance and compare the results of the conservative one-way coupled analysis using RELAP5 with point kinetics and those via **two-way coupling of RELAP5 and 3DKIN**

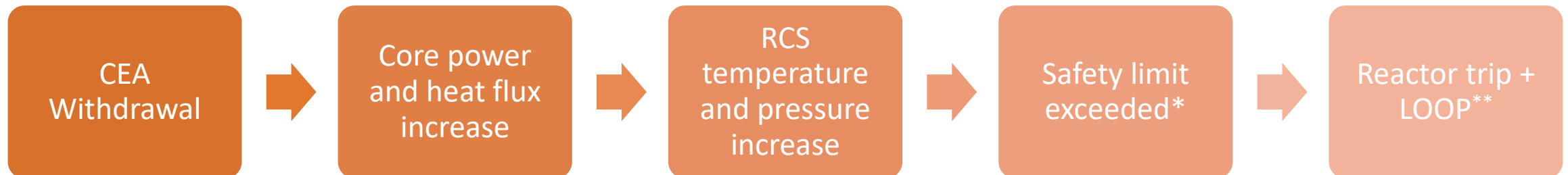


Introduction

- Control Element Assembly (CEA) Withdrawal at Power
 - is a **Reactivity Initiated Accident** (RIA)
 - causing uneven reactivity distribution in the core
 - with **strong feedback** mechanisms and **rapid reactivity** insertion
- Multiphysics simulation using code coupling
 - Thermal Hydraulics code **RELAP5**
 - Nodal Kinetics code **3DKIN**

Accident Scenario

- CEA withdrawal is an **anticipated operational occurrence** (AOO)
- CEA withdrawal may happen due to
 - failure in digital rod control system (DRCS)
 - failure in reactor regulating system (RRS)
 - operator error



CEA: Control Element Assembly

RCS: Reactor Coolant System

LOOP: Loss of Offsite Power

* Either low DNBR, high local power density (LPD), or high pressurizer pressure

** LOOP is assumed for conservatism

Protective Actions

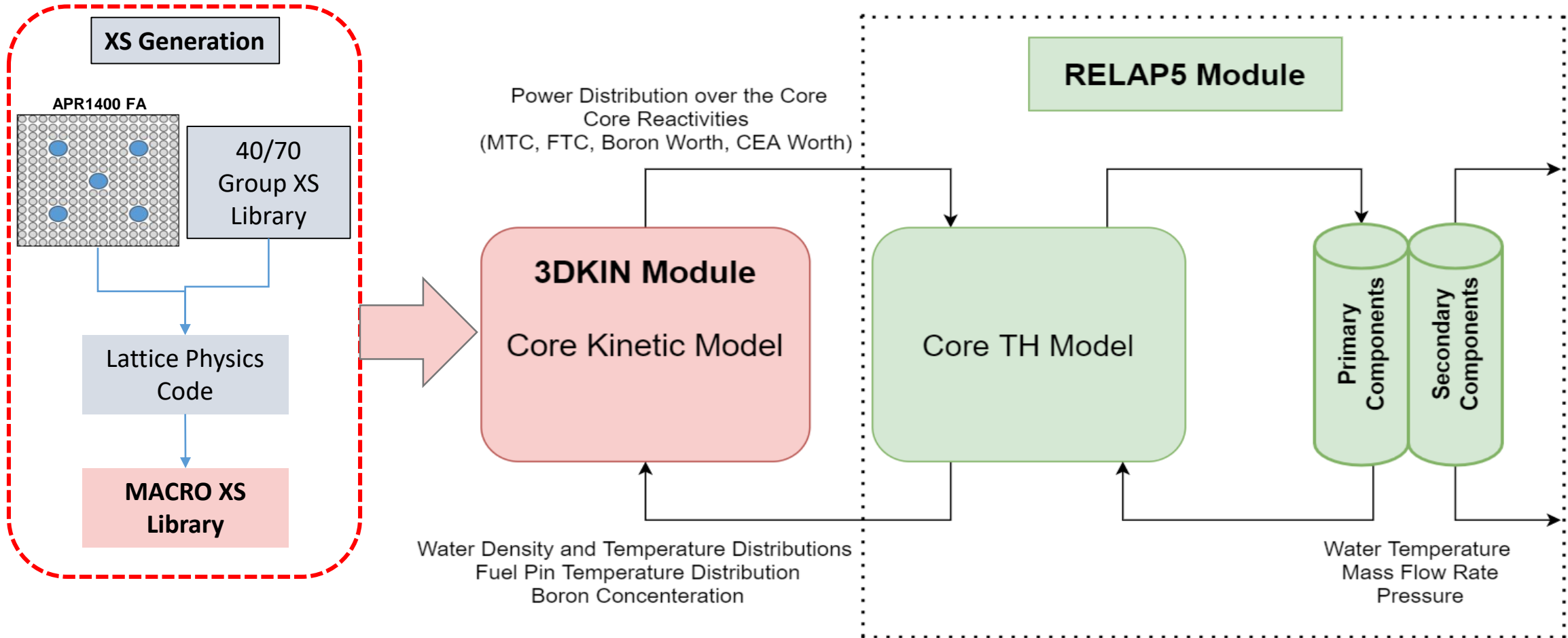
- Core parameters may approach specified acceptable fuel design limits (SAFDLs) on **DNBR** (> 1.29) and **fuel centerline melt temperatures** (2200 °F)
- Action from reactor protection system (RPS) based on
 - core protection calculator (CPC)
 - variable overpower trip (VOPT)
 - low DNBR
 - high local power density (LPD) trip
 - high pressurizer pressure trip (HPPT)

Initial Conditions

Conservative assumptions following APR1400 DCD Chapter 15, with biased parameters for **worst case scenario** and **concurrent LOOP** with turbine trip

Parameter	Value
Core power level, MWt	4062.66
Core inlet coolant temperature, °C	287.8
Core mass flow rate, 10 ⁶ kg/hr	69.64
Pressurizer pressure, kg/cm ²	163.5
Steam generator pressure, kg/cm ²	68.26
Moderator temperature coefficient	Most positive
Fuel temperature coefficient	Least negative

Methodology Overview



Plant Model



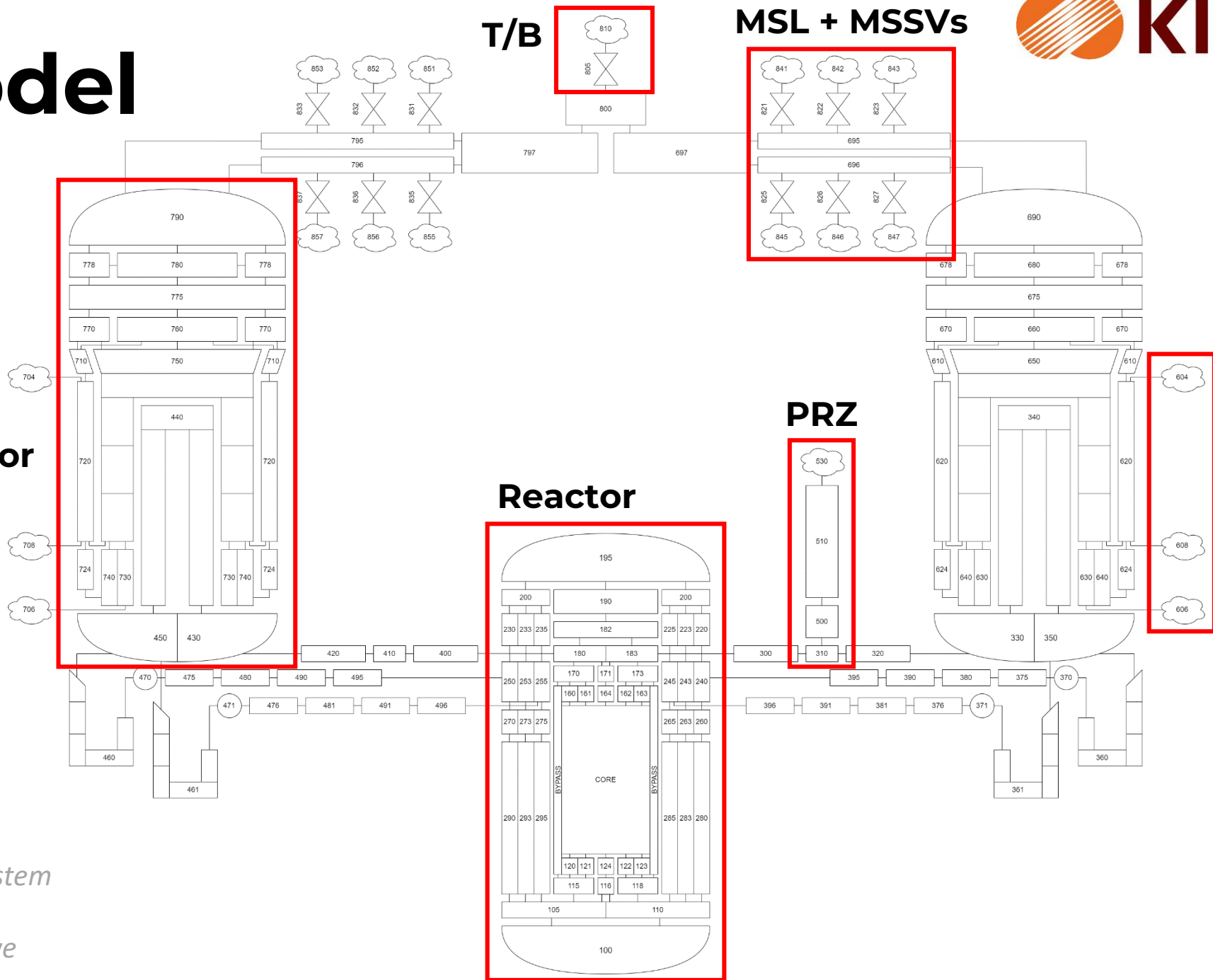
Steam generator

Reactor

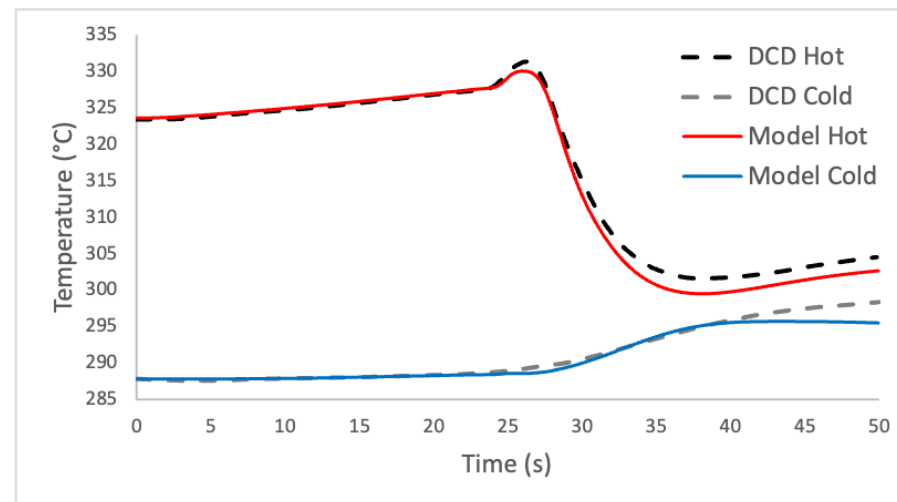
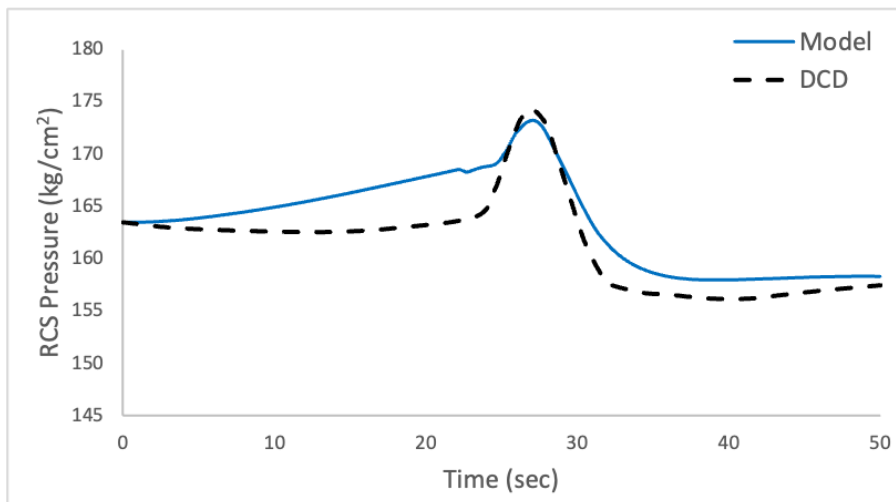
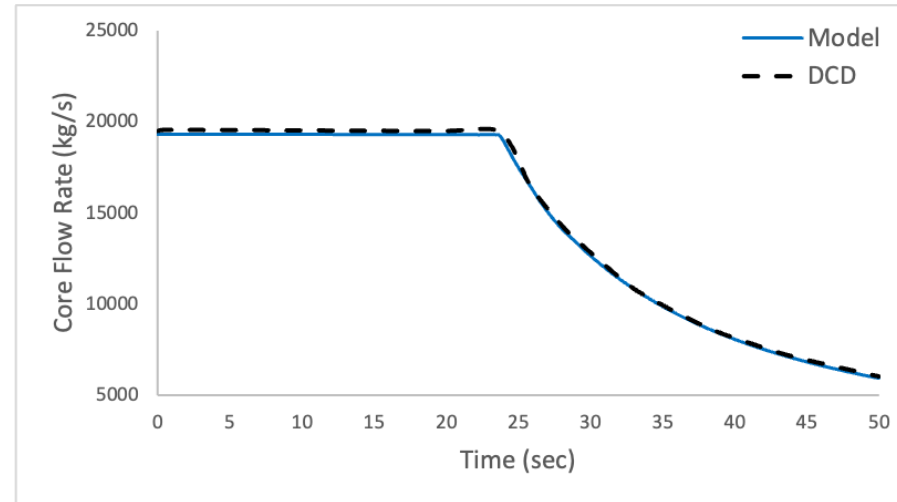
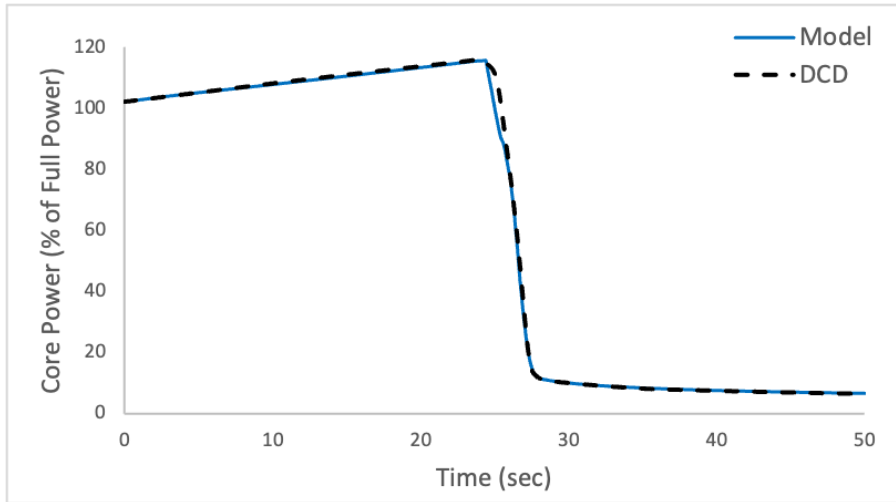
PRZ

**MFWS
AFWS**

T/B: Turbine
PRZ: Pressurizer
FWS: Feedwater system
AFWS: Auxiliary feedwater system
MSL: Main steam line
MSSV: Main steam safety valve

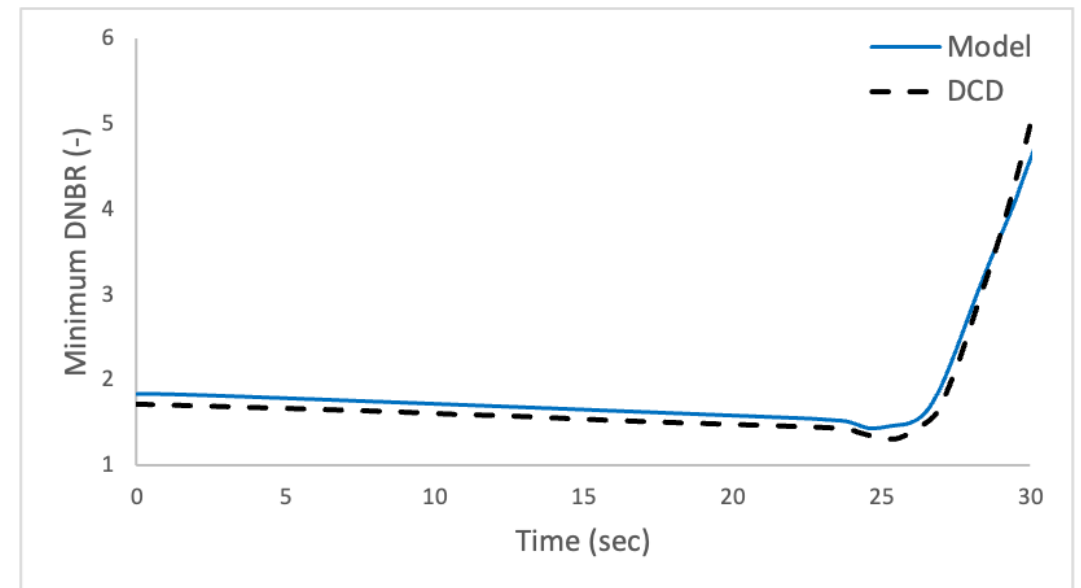


TH Model Validation



TH Model Validation

- DNBR calculated in hot channel using the W3 correlation
- Model minimum DNBR 1.43
- DCD results show non-proprietary KCE-1 CHF correlation
- DCD minimum DNBR 1.31



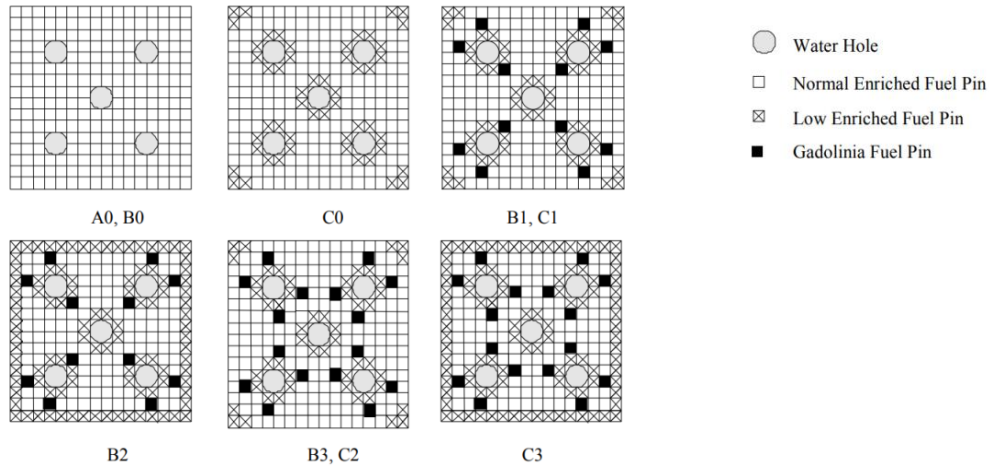
3DKIN Core Model

- 3DKIN requires two-group
 - Transport, absorption, fission and scattering cross-sections
 - Nu (average number of released fission neutrons)
 - Kappa (average energy released by fission)
- For reactivity coefficients, following equation is used:

$$\begin{aligned}\Sigma_{Bu,j} = & \Sigma_{base} + \alpha_1 \Delta\rho_{mod} + \alpha_2 (\Delta\rho_{mod})^2 + \alpha_3 \Delta T_{mod} + \alpha_4 \sqrt{\Delta T_{f,eff}} \\ & + \alpha_5 \Delta N_p + \alpha_6 (\Delta N_p)^2 + \alpha_7 (\Delta N_p)^2\end{aligned}$$

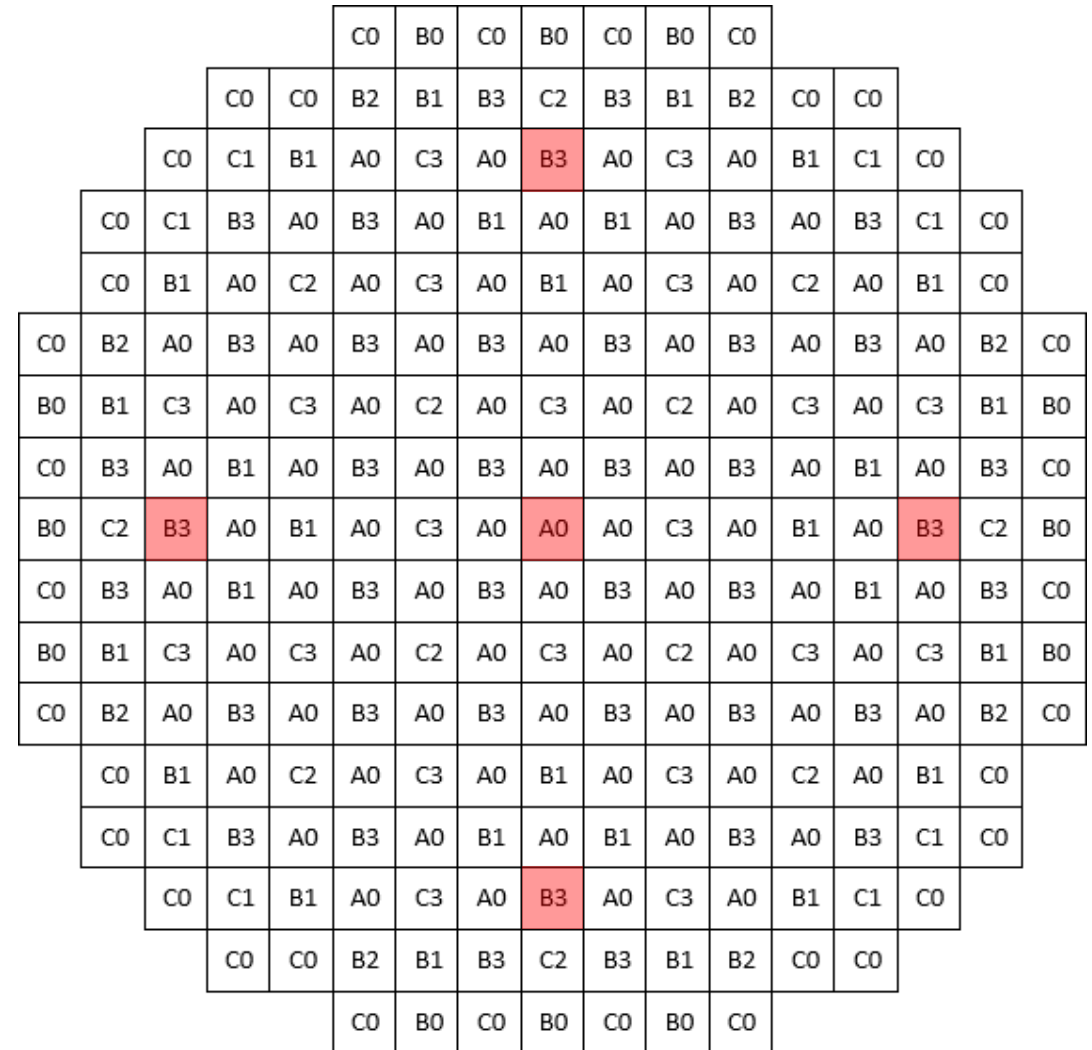
APR1400 Core Model

Composition of fuel assemblies



Fuel assembly parameters

Assembly Type	Number of Fuel Assemblies	Fuel Rod Enrichment (w/o)	No. of Rods Per Assembly	No. of Gd ₂ O ₃ Rods per Assembly	Gd ₂ O 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

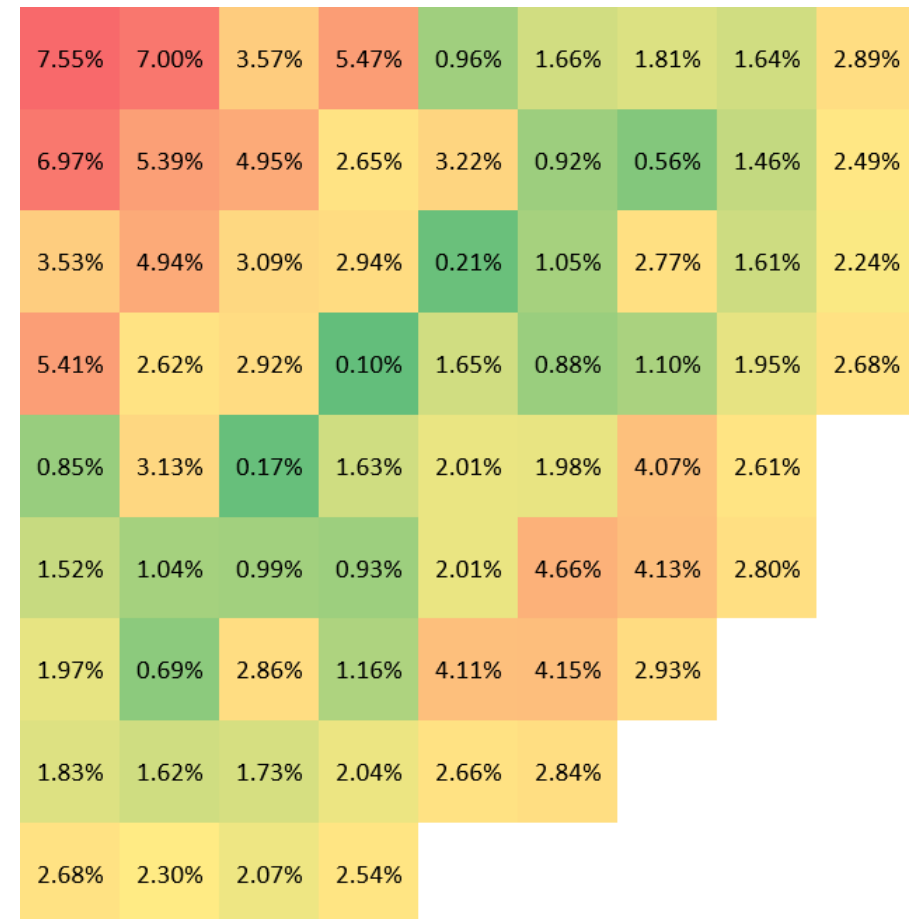


3DKIN Core Model Validation

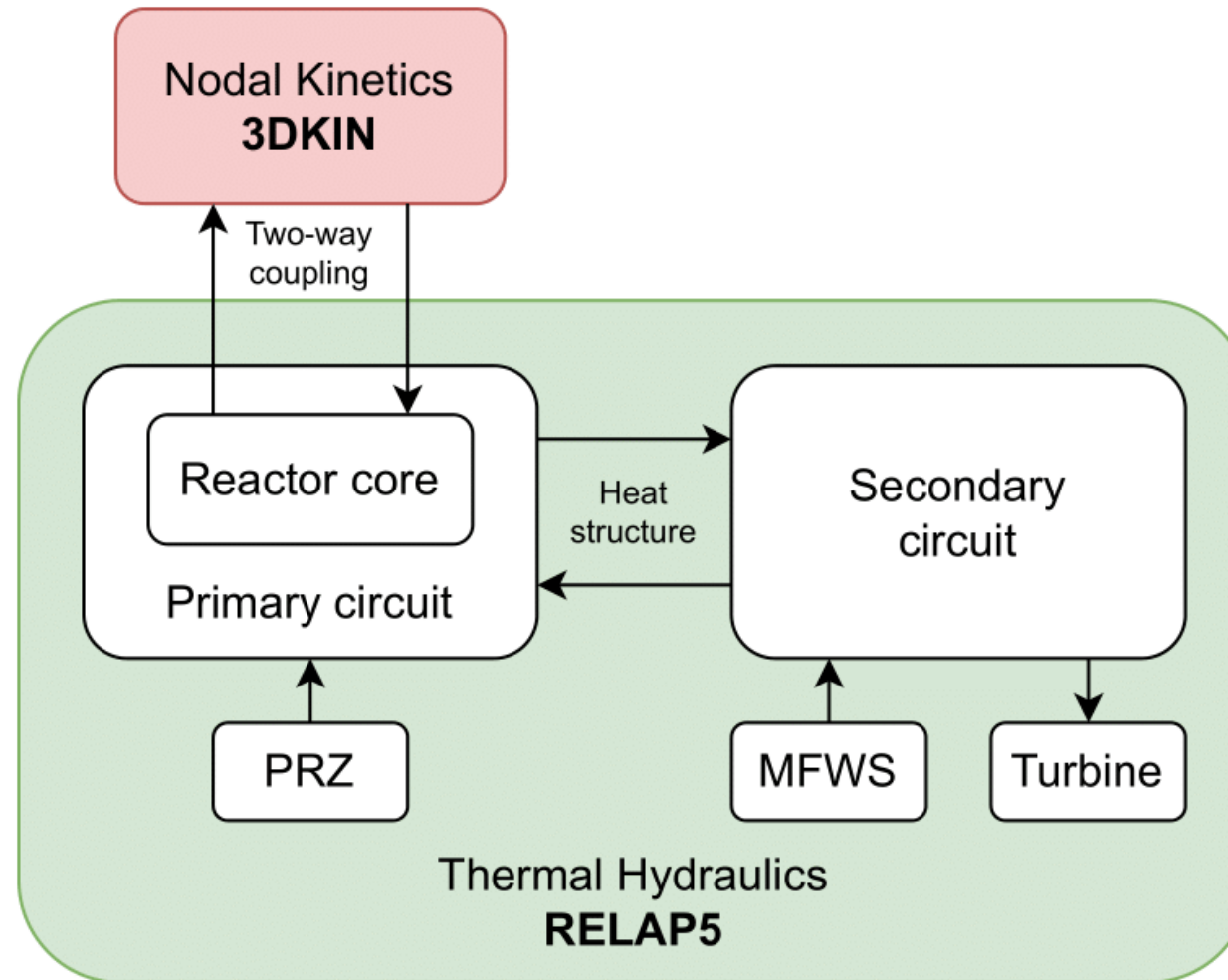
3DKIN core model with fuel assemblies

Parameter	DCD	Simulation	Deviation
Core thermal power, MWt	4062.66	4062.66	0.0 %
Pressurizer pressure, kg/cm ² *	163.5	163.34	0.1 %
Reactor inlet coolant temperature, °C	287.8	288.8	0.3 %
Core mass flow rate, 10 ⁶ kg/h	69.64	71.4	2.5 %
Steam generator pressure, kg/cm ²	68.26	68.27	0.0 %
CEA withdrawal speed, cm/min	76.2	76.2	0.0 %

Deviation in Core Power Distribution

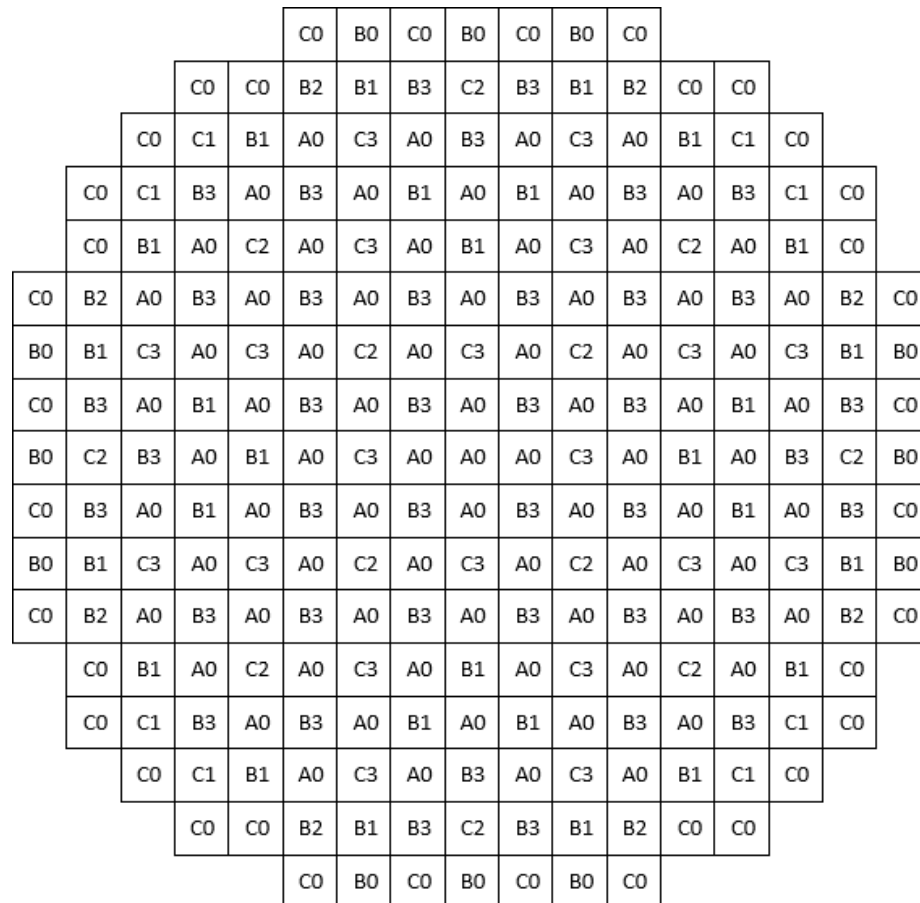


RELAP5/3DKIN Two-Way Coupling

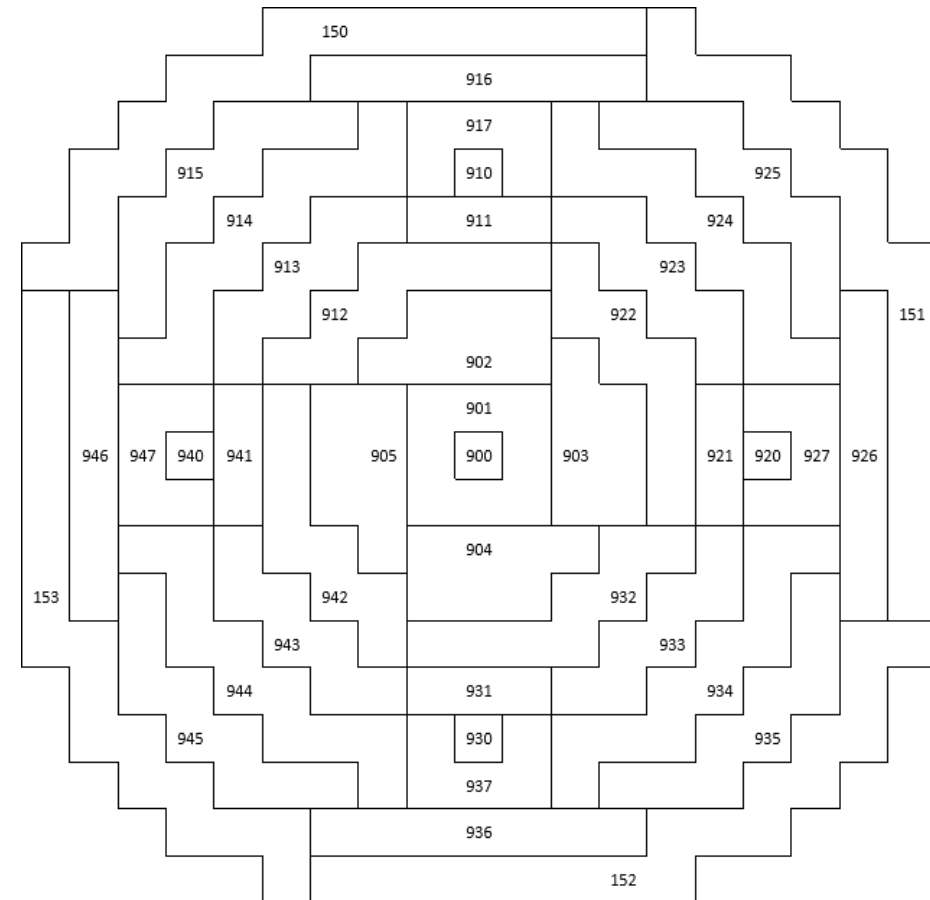


APR1400 Core Model

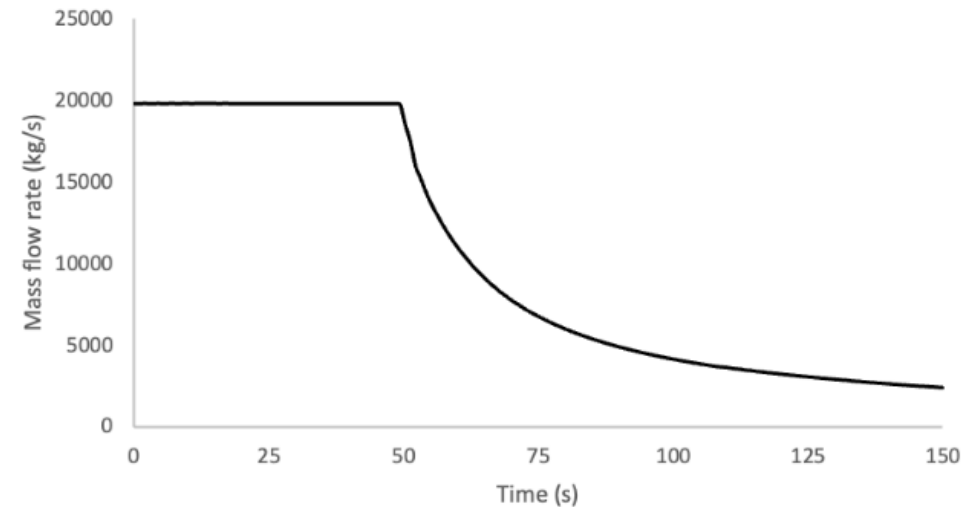
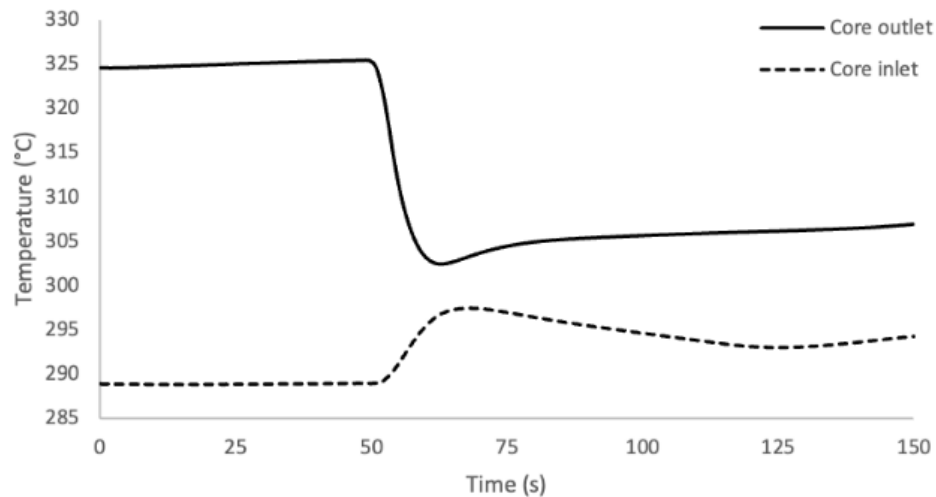
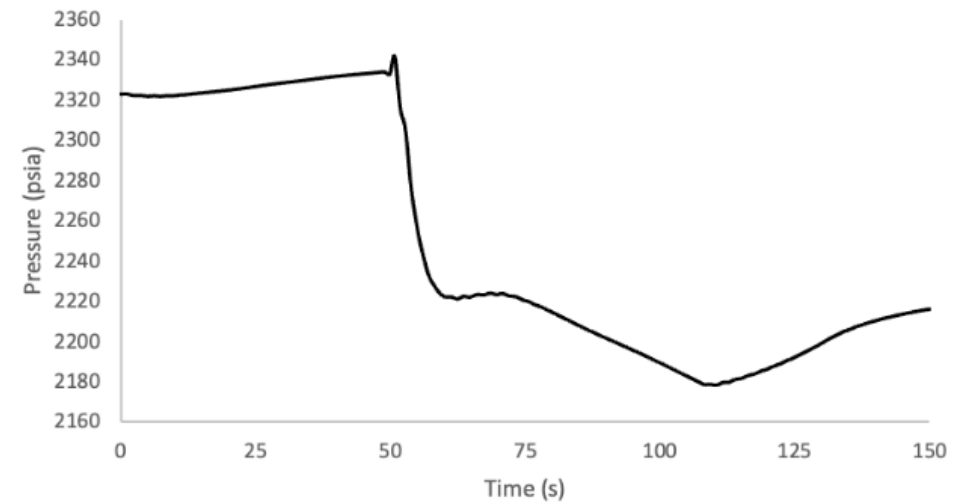
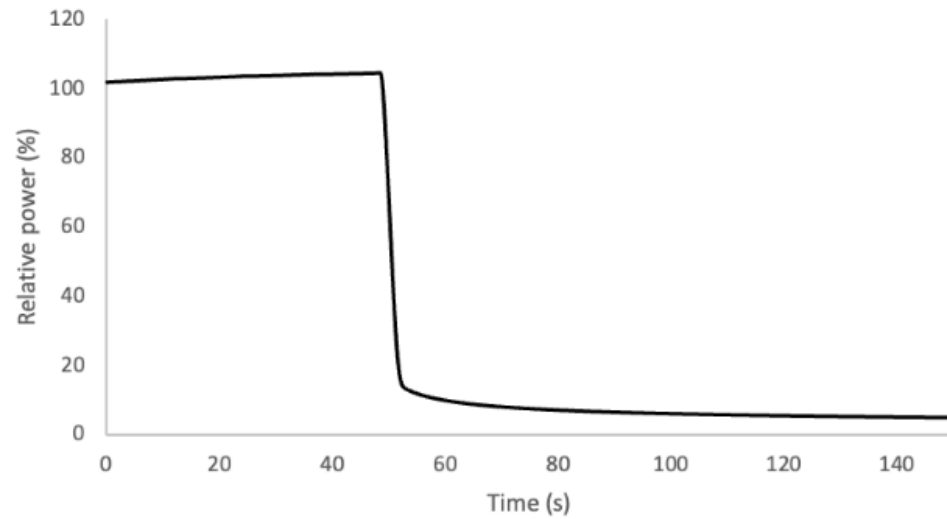
3DKIN core model with fuel assemblies



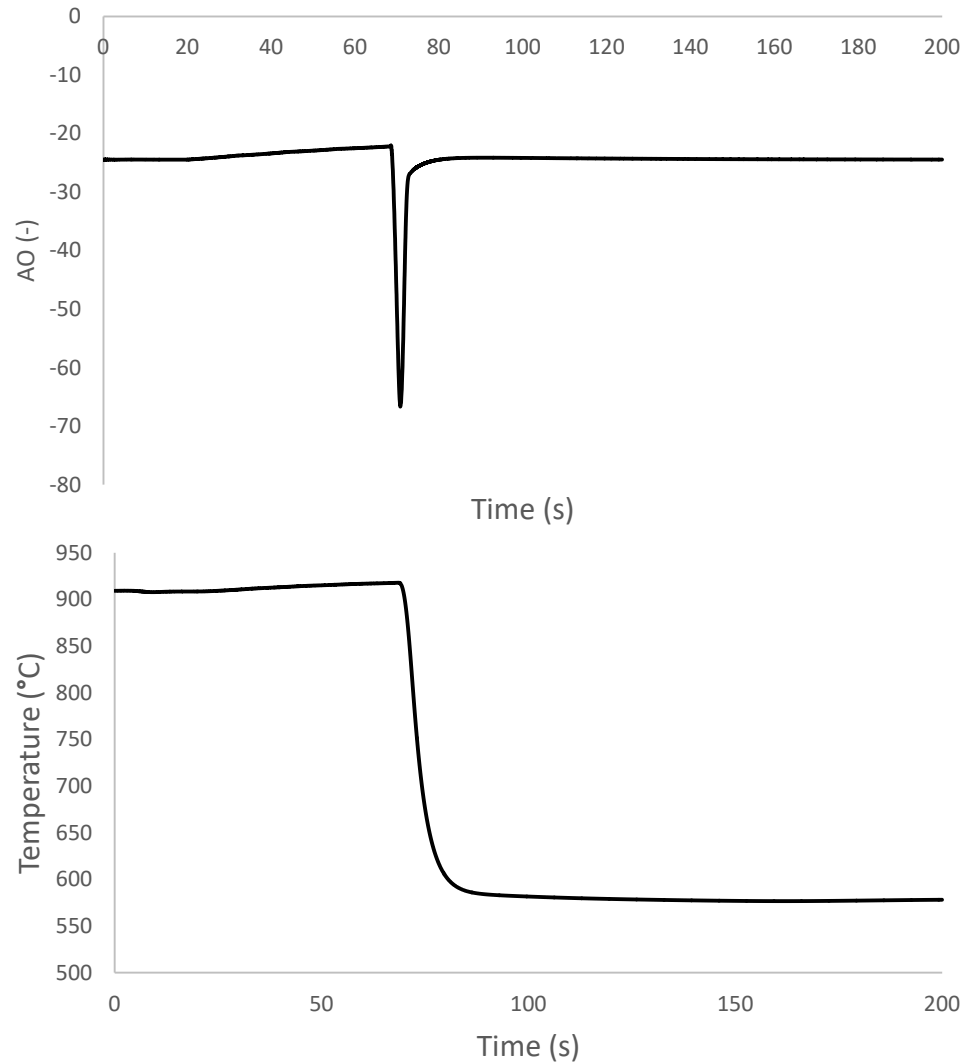
RELAP5 core nodes (channels)



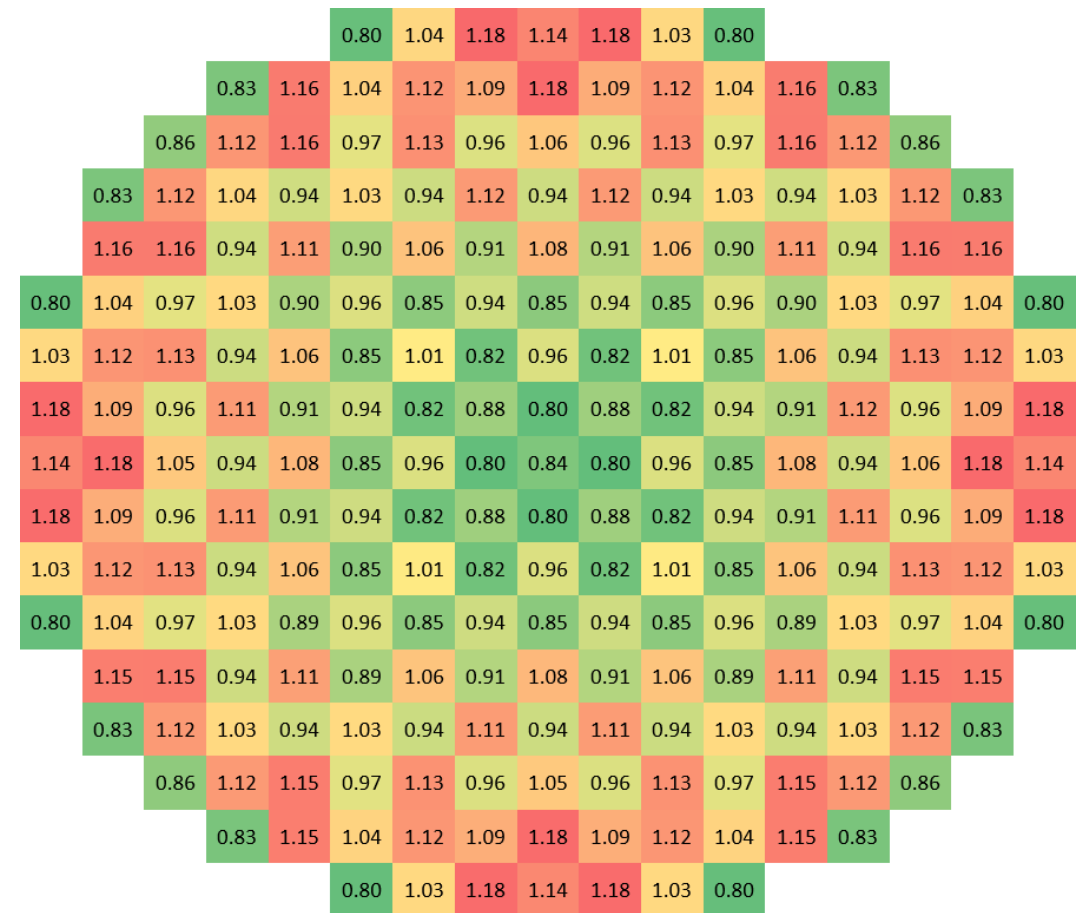
Multi-Physics Results



Multi-Physics Results



Core Power Distribution

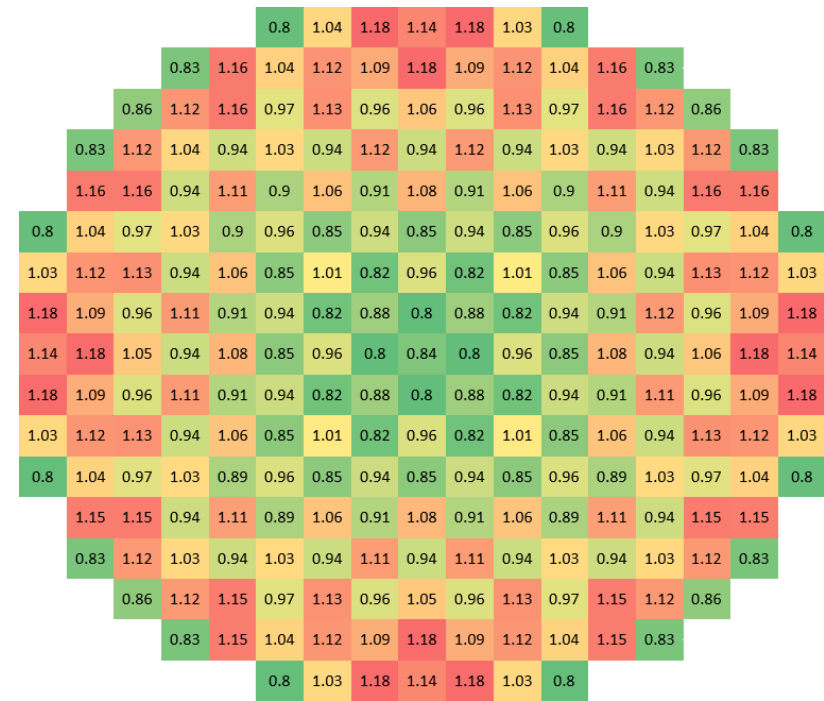
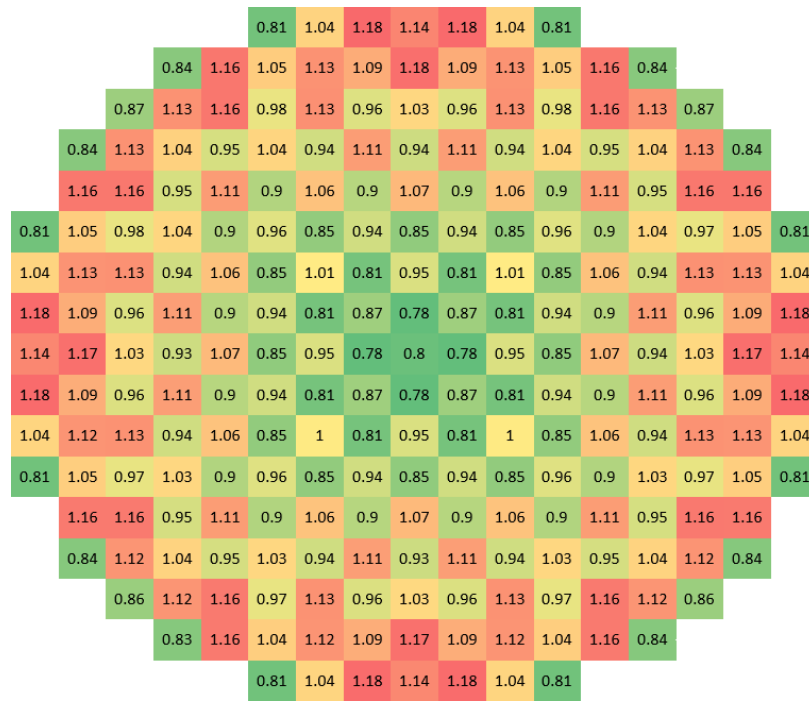
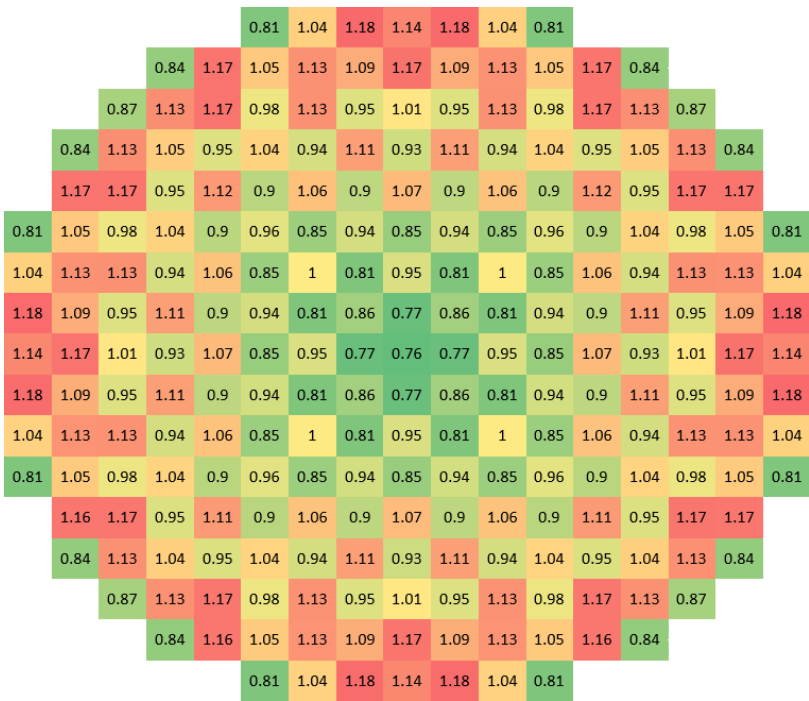


Core Power Distribution

Accident start

Middle of the accident

Start of reactor trip

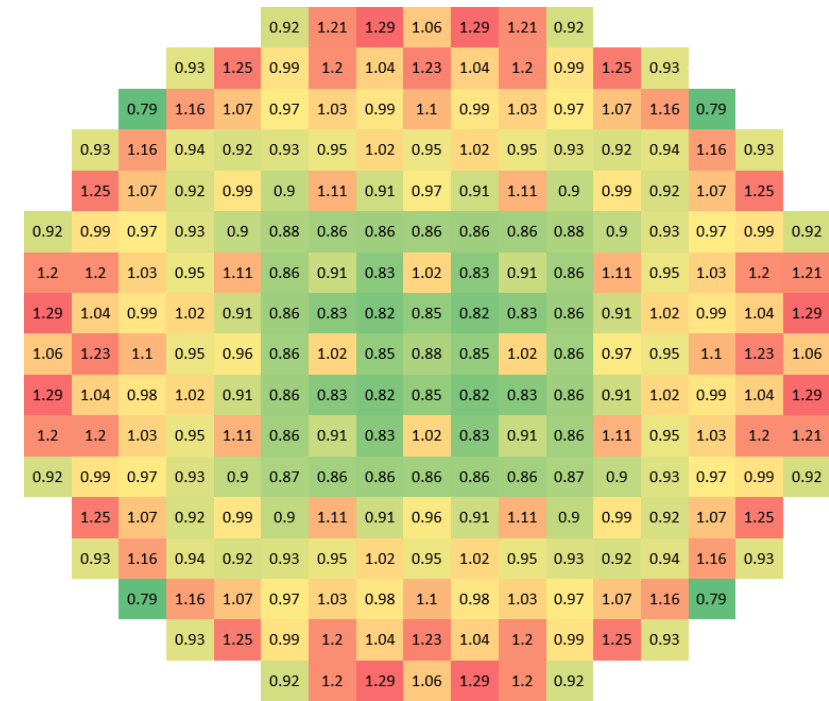
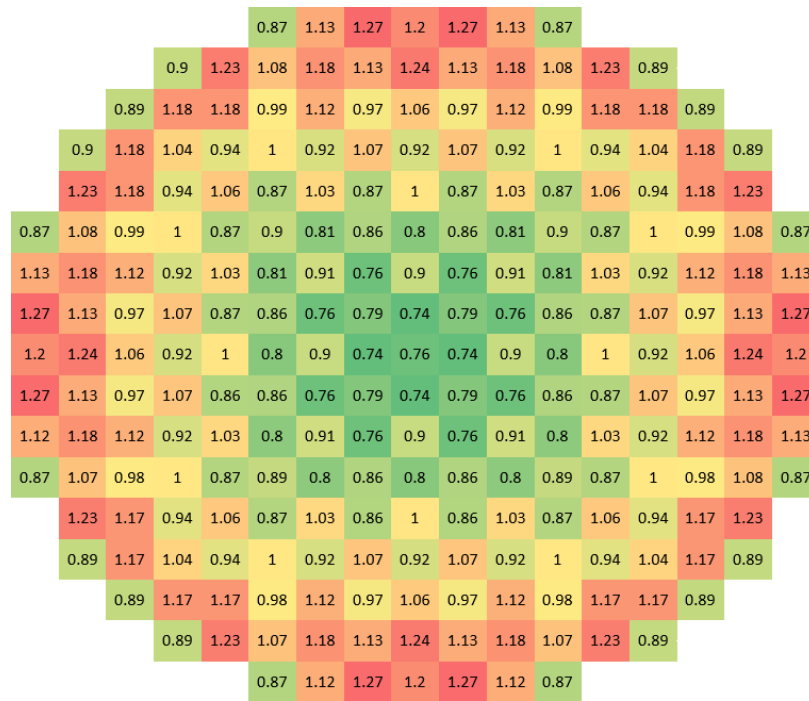
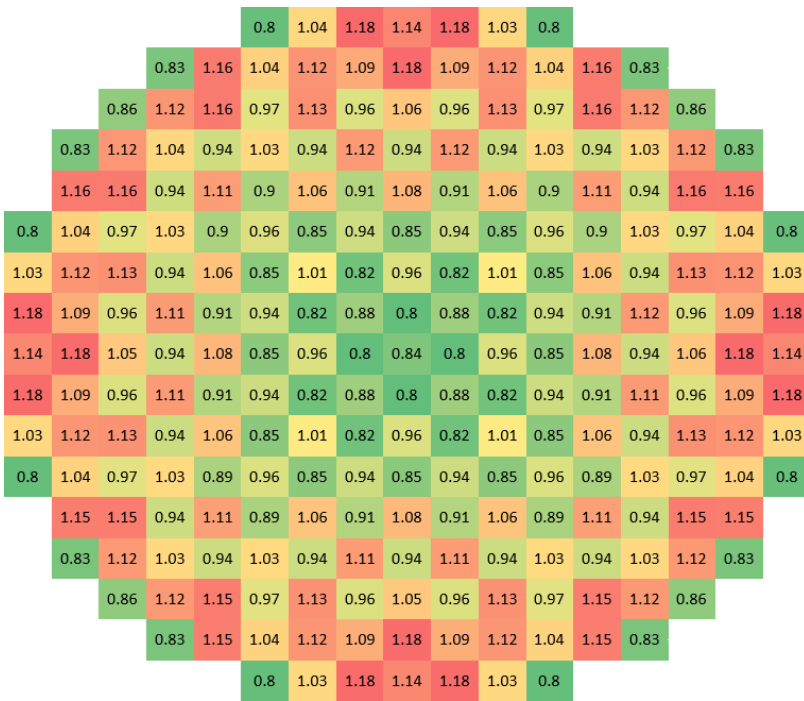


Core Power Distribution

Start of reactor trip

Middle of the trip

End of reactor trip



Conclusion

- Model Development
 - **TH Model validated against DCD** using the point kinetics model and conservative assumptions
 - **NK Model validated against DCD**
- Multiphysics simulation of CEA withdrawal accident
 - **Realistic results** achieved via RELAP5/3DKIN two-way coupling
 - Uneven **reactivity distribution** in the core **detected more precisely**
 - Simulation provides a **larger safety margin**, bringing more operational flexibility
 - Model tuning for more precise results is under development

References

- [1] Korea Hydro & Nuclear Power Co., Ltd, “APR1400 Design Control Document Tier 2: Chapter 15 Transient and Accident Analyses, Revision 3”, August 2018.
- [2] Korea Hydro & Nuclear Power Co., Ltd, “APR1400 Design Control Document Tier 2: Chapter 4 Reactor, Revision 3”, August 2018.
- [3] AKBAS Sabahattin, Victor MARTINEZ-QUIROGA, Fatih AYDOGAN, Chris ALLISTON, Abderrafi M. OUGOUAG, “Thermal-hydraulics and neutronic code coupling for RELAP/SCDAPSIM/MOD4.0”, 2019.
- [4] U. S. NUCLEAR REGULATORY COMMISSION, “STANDARD REVIEW PLAN: 15.4.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER”, 2007.
- [5] TONG L. S., “HEAT TRANSFER IN WATER-COOLED NUCLEAR REACTORS”, 1967.

Acknowledgement

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