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

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1. Introduction

Control Element Assembly (CEA) withdrawal at power is a Reactivity Initiated Accident (RIA), considered as a Design Basis Accident (DBA). In the Design Control Document (DCD) for Korean APR1400 reactor, it belongs to a reactivity and power distribution anomalies category [1]. During this accident, a bank of CAEs is unintentionally withdrawn from the core, causing the RCS power, pressure and temperature to increase. As a result, the core condition approaches the Specified Acceptable Fuel Design Limits (SAFDL) regarding Departure from Nucleate Boiling Ratio (DNBR) and fuel centerline melt temperatures, which prompts the Reactor Protection System (RPS) to mitigate the negative consequences.

This transient perturbs the power distribution inside the core, and the effect of feedback mechanisms is significant and dynamic, which necessitates advanced simulation techniques. This can be achieved using high fidelity multi-physics simulation using real-time code coupling.

For CEA withdrawal at power accident, Thermal Hydraulics (TH) and Nodal Kinetics (NK) codes are coupled to simulate the system response and boundary conditions of each part of the system. Therefore, RELAP5 thermal-hydraulics code coupled with 3DKIN nodal-kinetics code is used for this simulation.

2. Literature Review

Traditionally, conservative analysis has been conducted to simulate CEA withdrawal at power accident using one-way coupling, for example by Lee et al. using KNAP methodology which is based on RETRAN code [2], Yang et al. using SPACE code [3] and Jang et al. using RETRAN code [4]. Given the limitations of one-way coupling techniques to fully represent the complexity of the underlying dynamic and three-dimensional phenomena, several discrepancies are inevitable, specifically regarding the minimum DNBR.

According to Park [5], point kinetics model using one-way coupling is convenient for the conservative analysis approach of RIAs but may lead to poor representation of the safety margin. Therefore, two-way coupling is recommended for accurate system response during those transients.

Two-way coupling of thermal-hydraulics and nodal-kinetics codes, such as MARS-KS and MASTER codes, or RELAP5 and 3DKIN codes, is indispensable to provide high-fidelity simulation results for transients with uneven power distribution and strong dynamic feedback mechanisms. Those include reactivity initiated accidents, such as inadvertent control rod withdrawal at power. [6]

3. APR1400 Model Description

For the accident analysis, a thermal-hydraulics model of APR1400 reactor is developed. A system nodalization containing key systems and components and their parameters is prepared for RELAP5 as shown in Figure 1. This model is composed of the Reactor Coolant System (RCS) with 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. The core inlet and outlet nozzles, downcomer, lower and upper plenum as part of the reactor vessel are modeled as well.

Unlike in one-way coupling using point kinetics model, the core is not simply represented via an average and a hot channel. For the multi-physics simulation, a detailed core model is developed in 3DKIN using 313 radial nodes by 51 axial nodes, as will be explained in detail later.

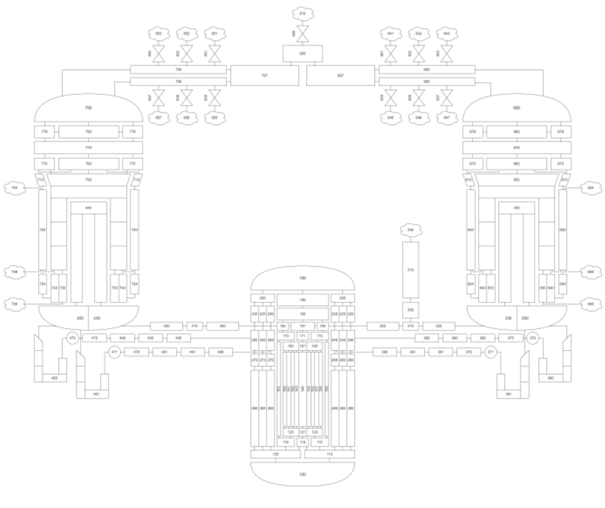

Figure 1 APR1400 RELAP5 model nodalization

3.1 Primary Circuit

The primary circuit includes two hot legs connecting the RPV to the SGs and four cold legs connecting the RPV to four RCPs that forcibly circulate the coolant. The PRZ is connected via a surge line to one of the hot legs to accommodate pressure changes in the system and to maintain the RCS design pressure. Heat is exchanged from primary to secondary circuits via a heat structure, simulating the U-tubes.

3.2 Secondary Circuit

The secondary circuit, as the main part of the Nuclear Steam Supply System (NSSS) contains two SGs. Both of them are connected to the Main Feed Water System (MFWS), where time-dependent volumes reflect the boundary condition of constant feed water flow. At the upper part of each SG, two main steam lines (total four) are connected and leading the steam from SGs to the turbine, which is represented as a time-dependent volume, imposing a pressure boundary condition.

3.3 Safety Systems

Additional safety systems relevant to the accident are implemented into the RELAP5 input deck. Specifically, the Pilot-Operated Safety Relief Valves (POSRVs) attached to the PRZ head to protect the RCS from over-pressurization. Further, the Auxiliary Feed Water System (AFWS) is added to deliver the feed water in case of the Loss Of Offsite Power (LOOP), when MFWS is not available. Finally, the Main Steam Safety Valves (MSSVs) were implemented on each steam line to maintain the secondary pressure. Each MSSV operate according to the pressure set points and mass flow rate capacities stated in the DCD Chapter 4 [7] to reflect the conservative assumptions.

3.4 Fuel Assemblies

As mentioned earlier, a detailed core model is necessary to reflect the realistic behavior of the core during the transient. The core model of APR1400 consists of 241 Fuel Assemblies (FAs) with nine different types. The FAs are divided into three groups – A, B and C, depending on the uranium enrichment level and whether or not gadolinia Burnable Absorber (BA) is used. Each assembly type is described in the Table 1 and Figure 2.

Table 1 APR1400 Fuel Assembly's Parameters [8]

Assembly type	Number of FA	Fuel Rod Enrichment (w/o)	Number of Rods per Assembly	Number 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

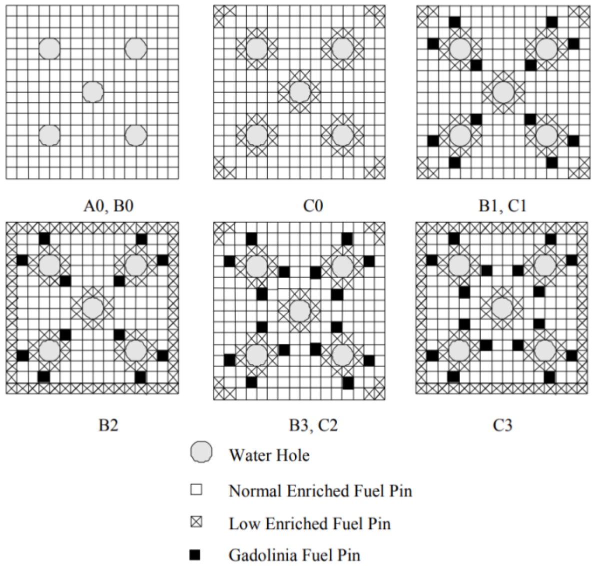

Figure 2 APR1400 Fuel Assembly's Design [8]

3.5 Core Model

The reactor core is composed of the aforementioned FAs for the first cycle, as described in the DCD Chapter 4, maintaining the octant core symmetry. The most important parameters for the simulation are cross-section data of each FA and their position in the core. The configuration for a quarter core is illustrated in Figure 3. In 3DKIN, the active core is represented by 241 fuel assemblies with cross-sections grouped according to the type of assembly, each divided into 51 axial nodes. Additionally, 72 radial nodes with 51 axial divisions represent the radial reflector around the core.

A0	A0	C3	A0	B1	A0	B3	C2	B0
A0	B3	A0	B3	A0	B1	A0	B3	C0
C3	A0	C2	A0	C3	A0	C3	B1	B0
A0	B3	A0	B3	A0	B3	A0	B2	C0
B1	A0	C3	A0	C2	A0	B1	C0	
A0	B1	A0	B3	A0	B3	C1	C0	
B3	A0	C3	A0	B1	C1	C0		
C2	B3	B1	B2	C0	C0			
B0	C0	B0	C0					

Figure 3 APR1400 Core Quadrant Model [8]

4. Accident Description

Uncontrolled Control Element Assembly (CEA) withdrawal at power may occur as a result of a single failure in the Digital Rod Control System (DRCS), Reactor Regulating System (RRS), or due to an operator error. No other single failure listen in Table 15.0-4 of DCD Chapter 15 has any effect on the accident. [1] Therefore, US NRC Standard Review Plan criteria for uncontrolled control rod assembly withdrawal at power are being met [9].

4.1 Sequence of Events

To initiate the CEA withdrawal accident, the fifth group of CEAs is withdrawn from the reactor core at power. This leads to perturbing the neutron flux, and hence creating a reactor power anomaly, which causes the core heat flux to increase. As a result, the RCS temperature and pressure increase. Depending on the CEA withdrawal speed, the initial system conditions and the reactivity feedback mechanisms, a certain amount of reactivity is inserted into the core. Action from RPS is required to mitigate the transient, as there is a possibility to approach the Specified Acceptable Fuel Design Limits (SAFDL) related to minimum DNBR and the fuel centerline melt temperatures. Therefore, based on the Core Protection Calculator (CPC), the reactor trips as a result of a Variable Overpower Trip (VOPT), low DNBR trip, high Local Power Density (LPD) trip, or High Pressurizer Pressure Trip (HPPT). For conservatism, it is assumed that LOOP occurs concurrently with the reactor trip.

4.2 Initial Conditions

Following the DCD, the initial conditions for CEA withdrawal at power accident were chosen conservatively to simulate the worst case scenario. Therefore, high reactor power, RCS pressure, and radial peaking factor; lower RCS inlet temperature, core flow together with maximum CEA withdrawal rate and rod worth is assumed. Regarding the reactivity feedbacks, the accident is assumed to occur at the Beginning Of Cycle (BOC), where the most positive Moderator Temperature Coefficient (MTC) and least negative Fuel Temperature Coefficient (FTC) cause the highest reactivity insertion during the transient. The initial conditions as stated in DCD Chapter 15 [1] are listed in Table 2.

5. Methods and Results

Innovative Systems Software (ISS) provides a multi-physics package with RELAP5 code coupled with 3DKIN code. While RELAP5 is used for the system thermal-hydraulics response, 3DKIN is used for the nodal-kinetics calculation, based on NESTLE nodal-kinetics code. RELAP5 generates and transfers data for water temperature, water flow rate, pressure and fuel temperature along with boron concentration to update the cross-section library necessary to initiate the calculation within 3DKIN, which in turn generates and transfers the 3-D power distribution of the reactor core along with the feedbacks from the fuel, moderator, boron, and CEA reactivities. This process is conducted iteratively at each time step until the results reach a predefined convergence criterion as illustrated in Figure 4.

For the NK model, cross-section data were generated by CASMO nodal-kinetics code for 3DKIN input files. Each individual FA therefore needs to be specified by a transport, absorption and scattering cross-sections as well as nu-fission, kappa-fission, and nu values. Moreover, those data need to be obtained for roded and unroded cases to allow 3DKIN to model the process the CEA withdrawal accordingly. Also, changes of those parameters for different moderator and fuel temperatures are required to reflect MTC and FTC feedbacks. DNBR is then calculated using W-3 correlation [10].

Table 2 Initial Conditions for CEA Withdrawal at Power [1]

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

5.1 Two-way Code Coupling

To enable coupling between RELAP5 and 3DKIN, the nodal-kinetics module in RELAP5 input deck is activated. The reactor core is split into 9 volumes with 20 axial nodes, where each of those are mapped to the appropriate NK structure in 3DKIN, which produce heat as a result of the nuclear fission. This ensures reasonable discretization of the core considering the RELAP5 code limitations. During the multi-physics simulation, the data are exchanged between the two codes, i.e. two-way coupling, which allows the impact of the core TH parameters on the NK core behavior and vice versa to be considers in real time.

5.2 Accident Simulation

To initiate the multi-physics simulation, one-way coupling using point kinetics is firstly used. This also serves to validate the thermal-hydraulics model against result reported in the APR1400 DCD, under conservative accident conditions. The initial conditions of the model were adjusted accordingly to represent the worst-case scenario. Therefore, the core power of 4062.66 MW_t (102 % of nominal power), core inlet temperature of 287.8 °C, PRZ pressure of 163.5 kg/cm^2 and core mass flow rate of $69.64{\cdot}10^6$ kg/hr were used.

The fifth group CEA withdrawal is enabled via a control variable and a trip. It is worth noting that the withdrawal speed of the fifth group CEA is set to 152.4 cm/min and the reactor trips upon reaching 115 % of the core nominal power.

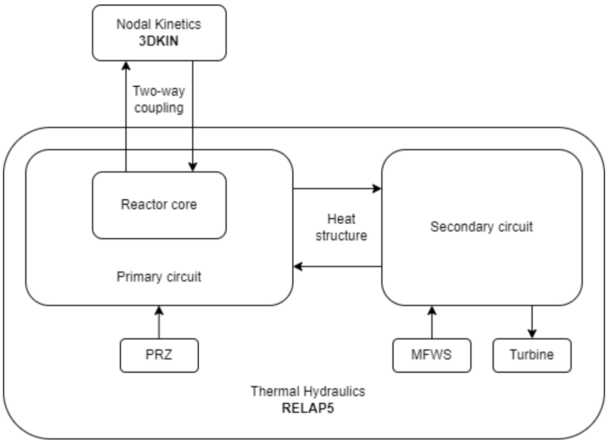

Figure 4 Diagram of two-way TH-NK code coupling

As mentioned earlier, the uncontrolled CEA withdrawal at power accident is initiated by withdrawing the fifth group CEA bank, which results in reactivity insertion and hence a core power increase. Upon reaching 115 % of nominal power, the VOPT signal trips the reactor. For conservatism, concurrent LOOP is assumed to occur and as a result, the MFWS becomes unavailable and instead, the AFWS delivers the feed water to the SGs. Also, as the pressure rises in the secondary circuit, the MSSVs are triggered to release steam and maintain the system pressure.

The results of the simulation are compared to the point kinetics model, i.e. one-way coupling, where the reactor core is split into an average and a hot channel and the reactivity coefficients are set to more realistic values; as well as to the DCD results based on the conservative approach.

When the point kinetics model is tuned to a less conservative but more realistic reactivity coefficients, the results match with reasonable agreement those of the multi-physics simulation with a CEA withdrawal speed of 152.4 cm/min. Clearly, the conservative approach leads to a much faster response. How this impacts the DNBR is yet to be seen since the DNBR calculation is being developed and will be included in the updated version of the model. Further adjustments and model tuning is required for precise simulation and reactivity feedbacks reflection.

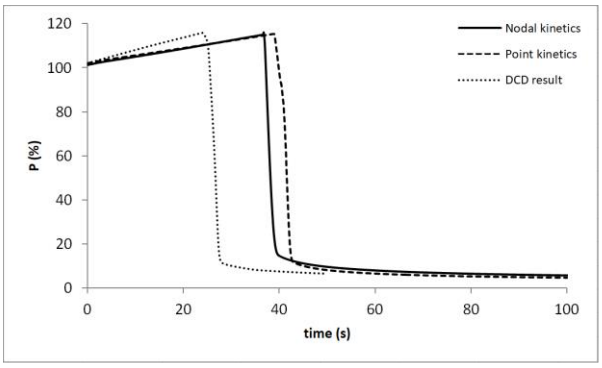

Figure 5 Core power vs time

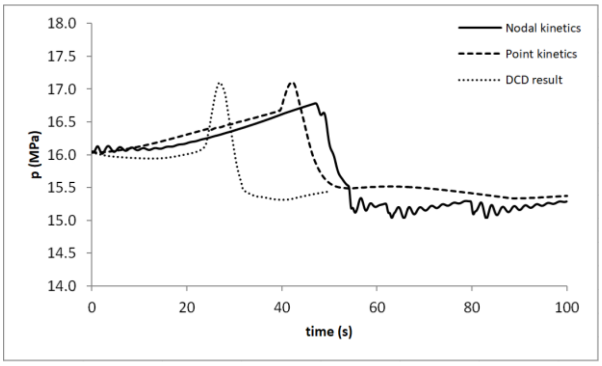

Figure 6 RCS pressure vs time

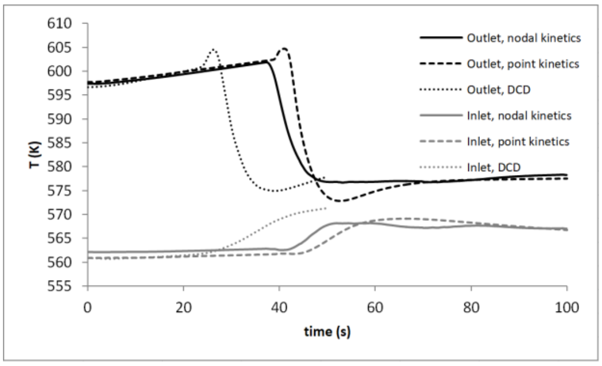

Figure 7 Core temperature vs time

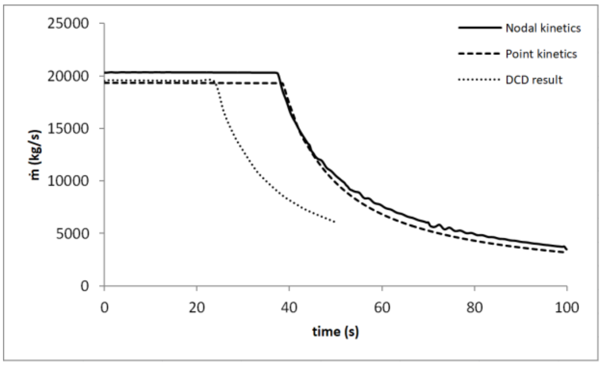

Figure 8 RCS mass flow rate vs time

6. Conclusions

Uncontrolled CEA withdrawal at power for Korean APR1400 reactor using two-way coupling is investigated in this work. The multi-physics simulation is conducted using RELAP5 thermal-hydraulics code coupled with 3DKIN nodal-kinetics code. The coupling allows more precise reactor core behavior to be reflected during the transient. The steady state input was adjusted to match the initial conditions stated in DCD Chapter 15. Finally, the results of the analysis were compared to the point kinetics model, and despite the different modeling approach, the results are in reasonable agreement, considering the used initial conditions. Also, the results of the analysis were compared to those reported in the DCD with more conservative assumptions (i.e. using most limiting MTC), where system response is faster, therefore reaching the core power set point of 115 % in a much shorter time. The fully coupled approach yields more realistic results with larger margin. Some discrepancies were observed and may be attributed to differences in modeling the reactor core structure and CEA properties and further adjustments are therefore needed. Future development consists of model tuning in order to increase the simulation accuracy.

Acknowledgement

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