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A Systems Engineering Approach to Multi-Physics Analysis of a CEA Withdrawal Accident

Jan Hruškovič[1)], Kajetan Andrzej Rey[1)], Aya Diab[1),2)*]

1) Department of NPP Engineering, KEPCO International Nuclear Graduate School, Ulsan, South Korea
2) Mechanical Power Engineering Department, Faculty of Engineering, Ain Shams University, Cairo, Egypt

Abstract : Deterministic accident analysis plays a central role in the nuclear power plant (NPP) safety evaluation and licensing process. Traditionally the conservative approach opted for the point kinetics model, expressing the reactor core parameters in the form of reactivity and power tables. However, with the current advances in computational power, high fidelity multi-physics simulations using real-time code coupling, can provide more detailed core behavior and hence more realistic plant's response. This is particularly relevant for transients where the core is undergoing reactivity anomalies and uneven power distributions with strong feedback mechanisms, such as reactivity initiated accidents (RIAs). This work addresses a RIA, specifically a control element assembly (CEA) withdrawal at power, using the multi-physics analysis tool RELAP5/MOD 3.4/3DKIN. The thermal-hydraulics (TH) code, RELAP5, is internally coupled with the nodal kinetics (NK) code, 3DKIN, and both codes exchange relevant data to model the nuclear power plant (NPP) response as the CEA is withdrawn from the core. The coupled model is more representative of the complex interactions between the thermal-hydraulics and neutronics; therefore the results obtained using a multi-physics simulation provide a larger safety margin and hence more operational flexibility compared to those of the point kinetics model reported in the safety analysis report for APR1400. The systems engineering approach is used to guide the development of the work ensuring a systematic and more efficient execution.

Key Words : Systems Engineering, Nuclear Engineering, Multi-Physics Simulation, APR1400, RELAP5, Accident Analysis, Reactivity Initiated Accidents.

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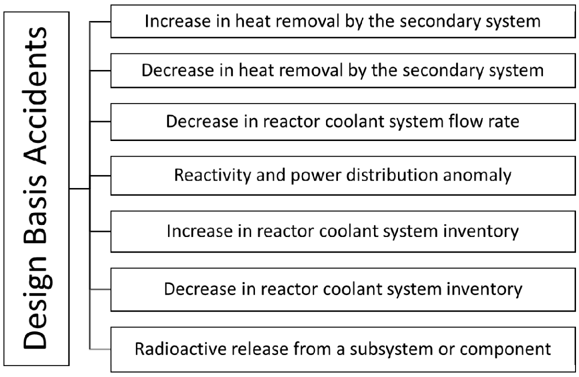
* 교신저자 : Aya Diab / KEPCO International Nuclear Graduate School / aya.diab@kings.ac.kr

1. Introduction

The Design Control Document (DCD) for the Korean APR1400 reactor categorizes design basis accidents (DBAs) based on their principal effect on potential degradation of fundamental safety. For each category, several possible initiating events are considered with the initial and boundary conditions set conservatively and the resilience of the power plant tested to withstand the single failure criteria while satisfying the safety limits.

The fourth category is focused on reactivity initiated accidents (RIAs) where the core experiences reactivity and power distribution anomalies, for example due to control element assembly (CEA) withdrawal at power.[1] In this scenario, the fifth control bank of CEAs is unintentionally being withdrawn from the reactor core, causing an increase in the core neutron flux, and thereby leading the reactor coolant system (RCS) power, temperature, and pressure to increase as well. Under these conditions, critical core parameters may approach the Specified Acceptable Fuel Design Limits (SAFDL), especially regarding Departure from Nucleate Boiling Ratio (DNBR) and fuel centerline melt temperature limits. As such, the reactor protection systems (RPS) is triggered to trip the reactor in order to prevent the negative consequences and together with the safety systems mitigate the accident.

CEA withdrawal at power perturbs the power distribution inside the reactor core, and the effect of feedback mechanisms can be significant and dynamic. This necessitates advanced simulation techniques, which can be achieved using high fidelity multi-physics simulation using real-time code coupling. For this accident, the thermal-hydraulics (TH) code, RELAP5/ MOD3.4, and the nodal kinetics (NK) code, 3DKIN, are internally coupled, using two-way data exchange to simulate the APR1400 system response for a more realistic safety evaluation.



[Figure 1] Design Basis Accident Categorization

2. Literature review

Traditionally, conservative analysis has been adopted to simulate CEA withdrawal at power accident using one-way coupling with point kinetics model; 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] Since one-way coupling using point kinetics does not fully represent the complexity of the underlying dynamic and three-dimensional phenomena in the reactor core, several discrepancies are inevitable, specifically regarding the power distribution, heat generation and therefore minimum DNBR.

According to Park[5], one-way coupling using the point kinetics model is convenient for the conservative analysis of RIAs but may lead to poor representation of the safety margin. Therefore, multi−physics simulation using two−way coupling is recommended for accurate representation of the APR1400 system response during those transients.

Two−way coupling of thermal−hydraulics and nodal kinetics codes, such as MARS−KS and MASTER codes[6], or RELAP5 and 3DKIN codes[7], 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.[8]

Systems Engineering (SE) is quite helpful in guiding the development of complex projects in a systematic and efficient manner. Mahmoud and Diab[9] used the SE approach to develop a multi−physics simulation of load follow operation (LFO) for the Korean APR1400, while Udrescu and Diab[10] used SE to assess the success window for in−vessel retention strategy during severe accident. In this work, the SE approach is used to guide the process of developing a multi−physics simulation of CEA withdrawal at power.

3. Systems Engineering Approach

3.1 Objective and method

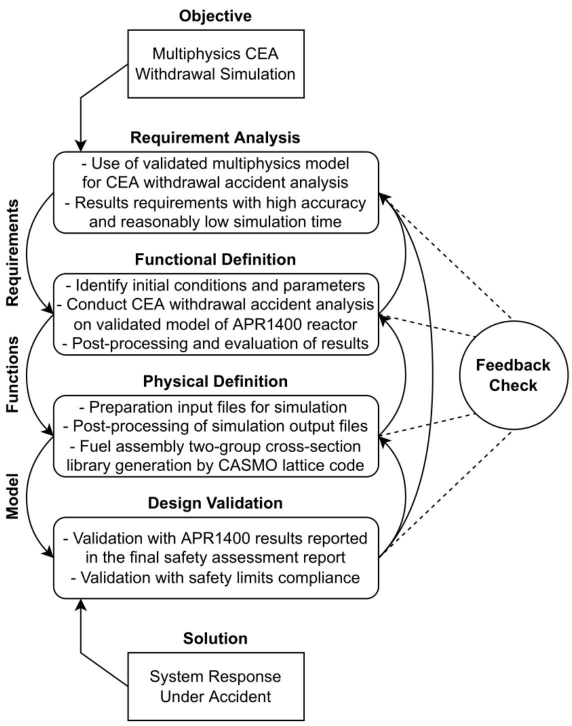
As mentioned earlier, the main objective of this paper is to analyze CEA withdrawal at power accident for the Korean APR1400 reactor using multi−physics simulation and check the system resilience against the safety criteria. The target system in the multiphysics CEA withdrawal at power accident simulation.

For this purpose, a state−of−the−art package, RELAP5/SCDAPSIM/MOD3.4/3DKIN, is used. With two−way coupling between RELAP5 and 3DKIN codes, the multi−physics simulation helps to address the complexity of the underlying phenomena and the dynamic interaction between the physics and thermal−hydraulics. This real−time coupling reflects the strong and dynamic feedback mechanism while simultaneously preserving the three−dimensional characteristics of the system response.

The systems engineering (SE) approach is adopted to facilitate the process of development of the multi−physics simulation of the CEA withdrawal by breaking down the work structure into smaller and easily−manageable tasks. This is achieved by adopting the Kossiakoff SE method[11] which involves the following four steps:

1. Requirement analysis;
2. Functional definition;
3. Physical definition;
4. Design validation.

According to the Kossiakoff method[11], to successfully complete the accident simulation, we start by clearly defining the objective hierarchy with important tasks identified as shown in Figure 2. The precise specification of the main objective as well as the solution and the feedback check in each step is indispensable to obtaining valid results from the multi−physics simulation.

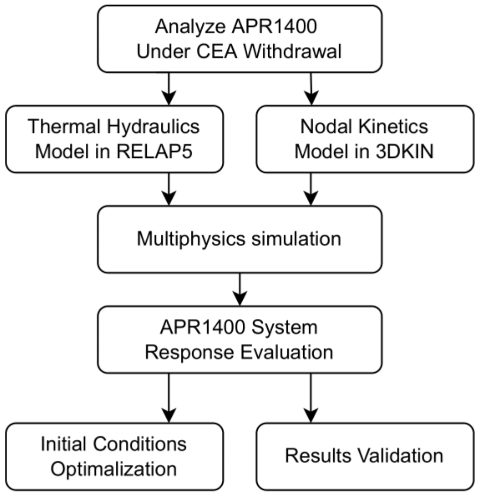


[Figure 2] SE Method Objective Hierarchy

3.2. Work Breakdown Structure

Once the systems engineering methodology is established, the work breakdown structure definition follows. This structure consists of six main stages:

1. Develop, verify, and validate an APR1400 thermal−hydraulics and nodal kinetics model, including the safety systems of APR1400 involved in CEA withdrawal mitigation.
2. Adjust the model parameters to match the initial conditions as stated in the DCD Chapter 15 of APR1400, such as higher core power, or higher RCS pressure and temperature.
3. Run a steady−state simulation and validate APR1400 system parameters with DCD values.
4. Perform the transient simulation of CEA Withdrawal.
5. Quantify the uncertainties in results for more reliable and realistic simulation.
6. Analyze results of the simulation and validate against already published work.



[Figure 3] Main Steps of the Accident Simulation

4. Requirements Development

The requirements considered for CEA withdrawal at power accident analysis are listed in Table 1. These can be divided into four categories:

- the mission requirements,
- the originating requirements,
- the system requirements,
- the simulation requirements.

The mission and systems requirements are connected with the requirements of APR1400 system (power plant) and originating and simulation requirements then allocate the requirements for the multiphysics CEA withdrawal accident simulation.

4.1 Mission Requirements

Enhancing the NPP safety and preventing any negative consequences on the plant employees, public and environment is a priority. One way to enhance the NPP safety may involve implementing additional safety features, but that comes at an economical cost which needs to be considered. High fidelity simulations using multi−physics analysis tools can also provide more realistic assessment of the plant response, which not only increases the safety margins in compliance with the regulatory requirements but also provides for more operational flexibility with economic benefits without impacting the plant's safety and reliability.

4.2 Originating Requirements

The originating requirements are allocated for the multiphysics CEA withdrawal simulation. As stated in the objective's hierarchy, those objectives are identified and evaluated based on the level of importance.

4.3 APR1400 System Requirements

System requirements of APR1400 system are based on the originating requirements and provide further description of the system parameters that must be considered during the simulation.

Under an accident condition, key APR1400 system parameters are monitored and safety limits cannot be exceeded to ensure the plant's safety. Those are mainly the core power distribution, minimum departure from nucleate boiling ratio (mDNBR), core heat flux, pressurizer pressure and temperature. As the CEA is being withdrawn from the reactor core, the core neutron flux, therefore power, RCS pressure and temperature increase, and may cause the specified acceptable fuel design limits (SAFDL) related to minimum DNBR and the fuel centerline melt temperatures to be breached.

Such a violation must trigger the reactor protection system (RPS), which in turn initiates the reactor trip and cooldown to a safe shutdown condition to ensure the plant safety.

4.4 Multiphysics Simulation Requirements

In order to conduct a successful simulation of the given scenario, the simulation requirements need to be considered. Those define the needs of the multi−physics simulation, such as applicability of the used tools and codes, physical parameters, models, and correlations, as well as proper initial and boundary conditions that reflect the CEA withdrawal at power accident.

First, verified, and validated packages of thermal−hydraulics and nodal kinetics codes with the capability of modeling complex phenomena must be selected to analyze the effect on APR1400 system behavior. Further, the simulation efficiency and accuracy must be considered as the multi−physics simulation is usually a computationally intensive undertaking.

Various coupling options then can be applied for those codes. Coupling can be done internally or externally, using implicit, semi−implicit or explicit coupling methodology. Additionally, for precise data exchange between the codes, it is important to apply proper mapping of core structures. Detailed

<Table 1> Requirements of CEA Withdrawal at Power Accident Analysis for APR1400 Reactor.

Requirements	Description
Mission Requirements	• APR1400 system response shall meet the safety criteria under a design basis accident condition.
Originating Requirements	• Multi-physics simulation shall show APR1400 system response under CEA withdrawal at power accident while satisfying safety limits (pin peaking factor, minimum DNBR, core heat flux, RCS pressure and temperature).
System Requirements	• APR1400 system should comply with a design basis accident condition. • The power plant should withstand the CEA withdrawal accident with reasonable conservatism and safety margin. • Accident is considered at beginning of cycle, where core feedbacks have the worst effect and the withdrawing speed should be the maximal of 76.2 cm/min. • Minimum DNBR of 1.29, maximum pressurizer pressure of 2475 psia (110% of nominal pressure) and maximum peak linear heat generation rate of 656 W/cm cannot be exceeded.
Simulation Requirements	• Coupled RELAP5 and 3DKIN codes must be capable of complex analysis of the given accident. • Multi-physics simulation must be capable of modeling the core structure as individual fuel assemblies and calculate its power distribution. • For accurate results of the simulation, proper mapping between fuel assemblies and core volumes and heat structures to exchange information is necessary. • A convergence criterion is defined to determine the simulation accuracy and iteration steps.

mapping can provide high-fidelity results, considering the system's three-dimensionality, however certain code limitations might constrain the simulation or if successful, may become computationally expensive.

Considering all the previously mentioned requirements, a multi-physics package, RELAP5/SCDAPSIM/MOD3.4/3DKIN, had been selected for the analysis. This package is developed by Innovative Systems Software (ISS), based in the US. The package consists of three modules: RELAP5, SCDAPSIM and 3DKIN. RELAP5 is a thermal-hydraulics lumped parameter system code, which was developed by Idaho National Laboratory (INL) and is widely used in the nuclear industry. SCDAPSIM, also developed by INL, is an add-on module to RELAP5 with severe accident modeling capabilities. Finally, 3DKIN, is the nodal kinetics module based on NESTLE code that was developed by North Carolina State University (NCSU). 3DKIN replaces the original point kinetics code of RELAP5 and the codes exchange power and heat flux on the one hand, and TH parameters e.g., temperature, density, and flow rate on the other hand. In this package, the codes are implicitly two-way coupled, using internal coupling method and serial integration.

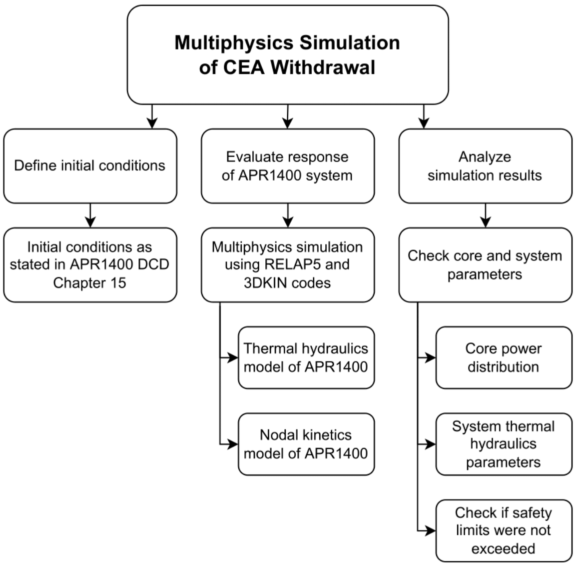
5. System Architecture

5.1 Functional Architecture

In this section, the functional architecture of the multi-physics simulation of APR1400 is described, as shown in Figure 4. The first level of the functional architecture defines the three main functions required to analyze the APR1400 response under CEA withdrawal accident.

Firstly, it is important to define the initial and boundary conditions as well as the key physical parameters, components and structures which have an impact on the APR1400 system behavior during the accident. To maintain the conservative approach, the initial conditions are selected according to the DCD Chapter 15 of APR1400 NPP.

After defining the initial conditions, the second main function is to evaluate the plant response by performing the safety analysis using the multi−physics simulation package. The third step includes analysis of obtained results from the simulation and checking the parameters against safety limits.



[Figure 4] Functional Architecture

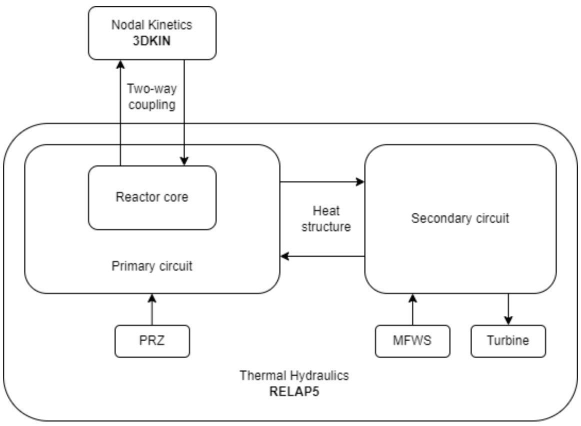
For this simulation, the plant model is developed to reflect the real NPP system description (nodalization) and during the simulation itself, the major primary and secondary system parameters are cross−checked against the DCD.

The nodalization used in this study includes the primary−side represented by the Reactor Coolant System (RCS) and two Steam Generators (SGs) on the secondary−side. The RCS consists of the Reactor Pressure Vessel (RPV), two Hot Legs (HLs), four Cold Legs (CLs) and four Reactor Coolant Pumps (RCPs). A Pressurizer (PRZ) is connected to the HL via the surge line and at its top, one Pilot Operated Safety Relief Valve (POSRV) is modeled to simulate the release of RCS inventory to protect against over−pressure or for depressurization. The water level in the SGs is controlled automatically over the full operating range by the Main Feedwater System (MFWS). On the secondary side, the main steam system transfers the steam from the SGs to the turbine through the Main Steam Line (MSL). Six Secondary Main Steam Safety Valves (MSSVs) with different pressure set points, two Main Steam Line Atmospheric Dump Valves (MSL−ADVs), two Main Steam Line Isolation Valves (MSLIVs) and Turbine Isolation Valve (TIV) are modeled on the MSL which is connected to the upper head of the SGs. The MSSVs prevent over−pressurization of the SG, TBV is used to isolate the Turbine and the ADVs are used by the operator to depressurize the SGs. The turbine is represented as a boundary condition using a time−dependent volume. Similarly, the containment is represented as a boundary condition by a time−dependent volume.

Lastly, the third main function is to analyze the results of the simulation. This includes checking the core parameters and ensuring that the safety limits, such as minimum DNBR, or linear heat generation rate, are not exceeded.

5.2 Physical Architecture

To perform the simulation, the APR1400 system, described above, is modeled, and analyzed using the two modules of the chosen package, namely RELAP5 and 3DKIN which communicate as illustrated using the physical architecture shown in Figure 5.



[Figure 5] Physical Architecture and Code Coupling

In a two-way coupled analysis, the three-dimensional core parameters, such as power distribution, are calculated using 3DKIN and then transferred to the thermal-hydraulics model, RELAP5, as a first guess which in turn uses this output from 3DKIN as a boundary condition. Then, after this iteration is done, relevant thermal-hydraulics data, such as volume flow rate, temperature, pressure, are transferred to the nodal kinetics module to be used in turn as boundary conditions to calculate the reactivity feedback and check the core power calculated earlier. At each time step, the simulations are repeated until convergence is achieved within the tolerance limits specified by the multi-physics simulation. Once the iterations converge within the specified tolerances, the simulation moves forwards in time.

6. Engineering Development

6.1 APR1400 Model Description

For the multi-physics accident simulation, a thermal-hydraulics model of APR1400 reactor was developed. The plant system nodalization reflecting the key systems and components is prepared for RELAP5 code as shown in Figure 6. The model consists of the reactor coolant system (RCS) including the reactor pressure vessel (RPV) as a main component, with two hot and four cold legs, the reactor circulating pumps (RCPs), the pressurizer (PRZ). The core inlet and outlet nozzles, downcomer, lower and upper plenum, detailed reactor core and core bypass channels are also modeled as part of the reactor vessel. The secondary side includes two steam generators (SGs), each connected to two main steam lines with a total of twenty safety valves.

Unlike in one-way coupling using point kinetics model, where the core is simply divided into an average and a hot channel, in this model a full core representation is provided via 3DKIN. The detailed 3DKIN core model is developed using 361 radial nodes, each divided into 20 axial nodes, as will be described later.

6.1.1. Primary and Secondary Circuit

The primary circuit consists of two hot legs, which connect the RPV to the SGs, and four cold legs connecting the RPV to four RCPs that maintain the coolant forced flow in the RCS.

The pressurizer is then connected to one hot leg by a surge line in order to maintain the RCS design pressure and accommodate pressure changes during the plant operation. Heat is exchanged from primary to secondary circuits by a heat structure, representing the SG U-tubes.

As the main part of the nuclear steam supply system (NSSS), the secondary circuit contains of two SGs, which are connected to the main feed water system (MFWS), represented using a time-dependent volume and a time-dependent junction, i.e., reflecting the constant feed water flow boundary condition. Two main steam lines are connected at the upper part of each SG to lead the steam to the turbine, which is represented as a time-dependent volume and a single junction, i.e., imposing a pressure boundary condition.

6.1.2. Safety Systems

Safety systems relevant to the accident under consideration are implemented into the thermal-hydraulics model. In particular, the pilot-operated safety relief valves (POSRVs) attached to the pressurizer head to protect the RCS from over-pressurization. Further, the auxiliary feed water system (AFWS) is added to deliver the feed water, when MFWS is not available, for example in case of the loss of offsite power (LOOP). Lastly, the main steam safety valves (MSSVs) were added to each steam line to maintain the secondary pressure. Each of those relief valves operate according to the pressure set points and mass flow rate capacities stated in the DCD Chapter 4[12] to reflect the conservative assumptions. The control logic of the valve operation as well as that of the AFWS, CEA withdrawal and reactor trip are programmed using the control module in RELAP5.

6.1.3. Core Model

A detailed 3DKIN core model is developed to reflect the realistic behavior of the core during steady-state and into the transient. The core model of APR1400 reactor consists of 241 fuel assemblies (FAs) with nine different types, divided into three groups – A, B and C. This division is mainly determined by the uranium enrichment level and whether gadolinia burnable absorber (BA) is used. Each FA type

<Table 2> Fuel Assembly Parameters[13]

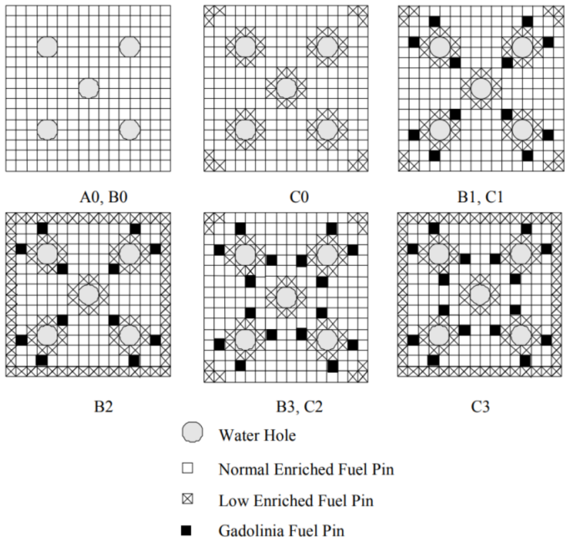
Assembly	Number of Fuel Assemblies	Fuel Rod Enrichment (w/o)	No. of Rods Per Assembly	No. of Gd_2O_3 Rods per Assembly	Gd_2O_3 Contents (w/o)
A0	77	1.71	236	–	–
B0	12	3.14	236	–	–
B1	28	3.14/2.64	172/52	12	8
B2	8	3.14/2.64	124/100	12	8
B3	40	3.14/2.64	168/52	16	8
C0	36	3.64/3.14	184/52	–	–
C1	8	3.64/3.14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

is described in the Table 2 and Figure 7.

The reactor core consists of the aforementioned FAs designed for the first cycle, maintaining the octant core symmetry, as described in the DCD Chapter 4. The quarter core configuration is shown in Figure 8.

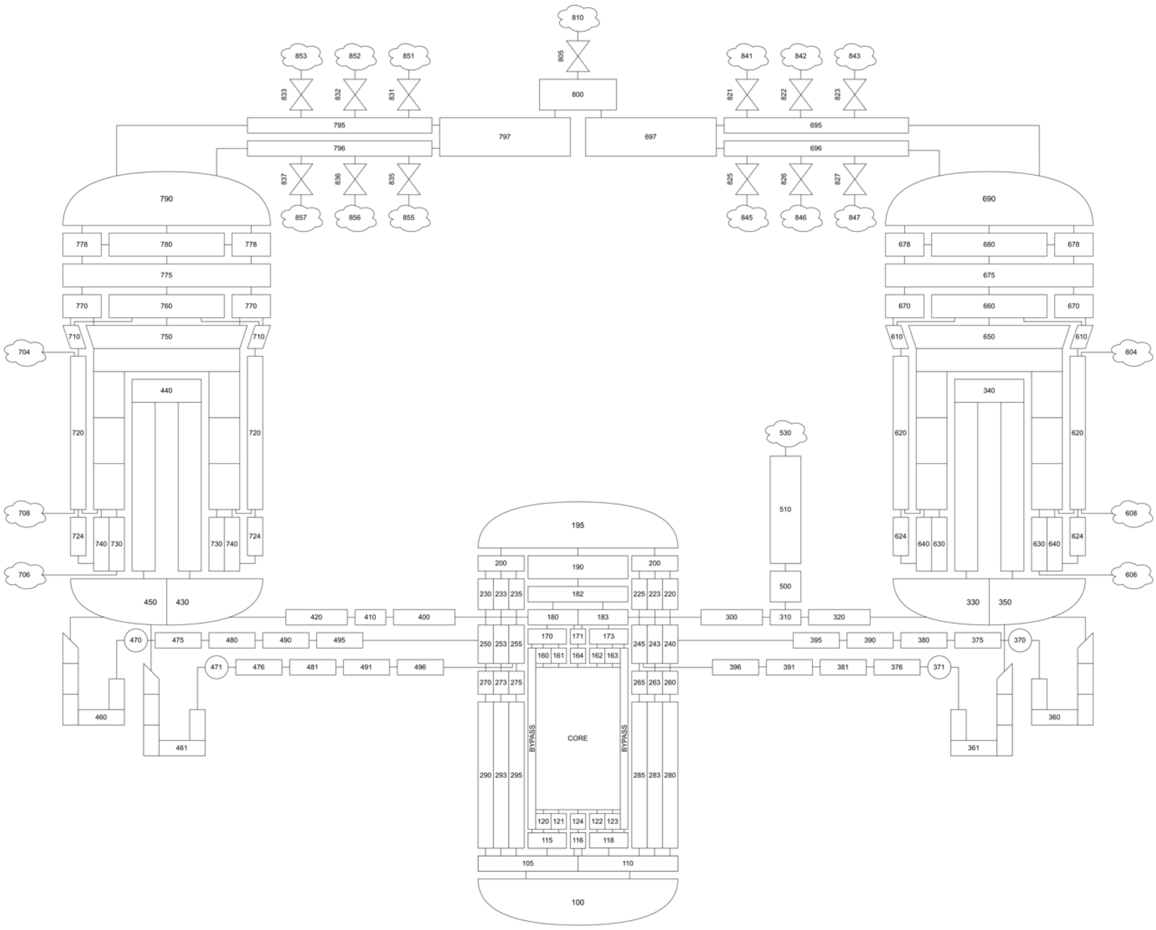
The active core is represented by 241 fuel assemblies in 3DKIN code, with cross-sections grouped by the assembly type, and each having 60 nodal-kinetics axial nodes, including the core reflector.

For the NK model, the 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,



[Figure 7] Fuel Assembly Design[13]

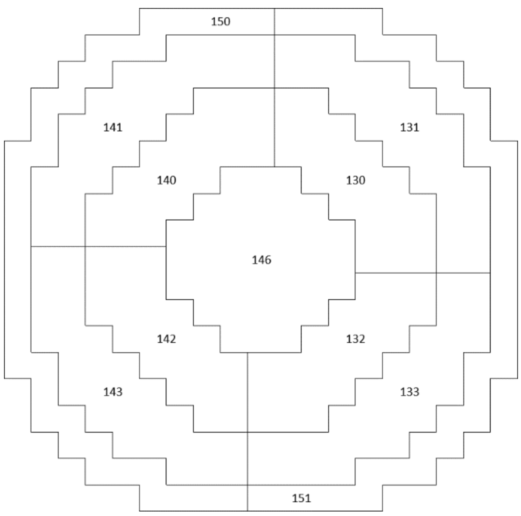
absorption and scattering cross-sections and



[Figure 6] APR1400 TH Model Nodalization nu-fission and kappa-fission values. Moreover, those data need to be obtained for rodded and un-rodded cases to allow 3DKIN to model the process the CEA withdrawal and reactor trip 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.[14]

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 8] Core Quadrant Model[13]



[Figure 9] Thermal-hydraulics Nodes of the Core[13]

6.1.4 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 single failure from Table 15.0-4 listed in DCD Chapter 15 has any effect on the accident.[1] This ensures that the US NRC Standard Review Plan criteria for uncontrolled control rod assembly withdrawal at power are being met.[15]

6.1.5. Sequence of Events

The CEA withdrawal accident is initiated by the fifth group of CEAs withdrawal and for conservatism, it is assumed that the reactor core operates at full power. The CEA withdrawal perturbs the neutron flux, therefore creating a reactor power anomaly, causing the core heat flux to increase. The RCS temperature and pressure increase following this power change. Based on the initial system conditions, the CEA withdrawal speed, and the reactivity feedbacks, a certain amount of reactivity is inserted into the core, hence increasing the power. As such, action from reactor protection system is required to control the transient. Because of the possibility of approaching the specified acceptable fuel design limits (SAFDL), mainly related to minimum DNBR and the fuel centerline melt temperature, based on the core protection calculator (CPC), the reactor may trip as a result of a variable overpower trip (VOPT), low DNBR trip, high local power density (LPD) trip, or high pressurizer pressure trip (HPPT).

When the reactor is tripped, the turbine trips and for conservatism, it is assumed that a LOOP occurs concurrently with the turbine trip.

6.1.6. Initial Conditions

The initial conditions for this accident were chosen conservatively, following the DCD Chapter 15, to simulate the worst-case scenario; i.e., high reactor power, RCS pressure, and radial peaking factor; low RCS inlet temperature, core flow rate together with maximum CEA withdrawal speed and rod worth are chosen. 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) feedbacks cause the highest reactivity insertion. The initial conditions are listed in Table 3.

To represent the worst-case scenario, the initial core power of 4062.66 MWt (102 % of nominal power), core inlet temperature of 287.8 °C, pressurizer pressure of 163.5 kg/cm2 and core mass flow rate of 69.64·106 kg/hr were assumed. Withdrawal of the fifth group CEA is enabled via a control variable and a trip logic in RELAP5 input deck. For the reactor trip, variable over power trip is set to 115 % of the core nominal power and high-pressure trip to 174 kg/cm2 (110 % of nominal pressure).

<Table 3> Initial Conditions for CEA Withdrawal

Parameter	DCD value
Core thermal power, MWt	4062.66
Pressurizer pressure, kg/cm^2A	163.5
Reactor inlet coolant temperature, °C	287.8
Core mass flow rate, 10^6 kg/h	69.64
Steam generator pressure, kg/cm^2A	68.26
CEA withdrawal speed, cm/min	76.2

6.2 Accident Simulation

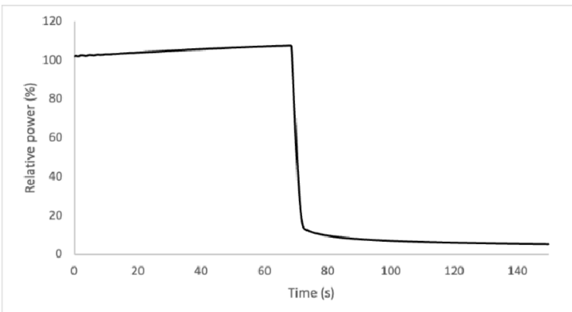
To initiate the multi-physics simulation one-way coupling using the point kinetics model is used to enable the simulation approach the steady-state condition faster and serves to validate the thermal-hydraulics model against result reported in the APR1400 DCD. Next, the nodal kinetics model is activated.

As described earlier, the uncontrolled CEA withdrawal is initiated by withdrawing the fifth control group CEA bank, which results in the reactivity insertion and hence a power increase. Upon reaching any of the safety limits, the reactor is tripped. To maintain the conservative approach, concurrent LOOP is assumed to occur together with the reactor and turbine trip. As a result, the MFWS becomes unavailable and instead, the AFWS delivers the feed water to the steam generators using a control logic that maintains the SG inventory within the wide-range operation limits. Also, as the pressure reaches certain set point in the secondary circuit, the MSSVs are triggered to release steam to the atmosphere and hence maintain the secondary system pressure.

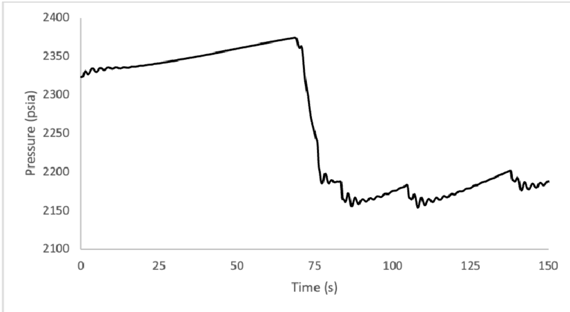
When the results of the current nodal kinetics model are compared to the conservative one from the DCD, the simulation shows a slower course of the accident, as the power, pressure and temperature in the core does not increase as rapidly. Clearly, the conservative approach leads to a much faster response, when the conservative MTC, FTC and CEA worth are used. The Impact on the DNBR is yet to be seen, as its calculation is currently under development and will be included in the updated version of the model. Additional adjustments and model tuning are required for precise core behavior simulation and reactivity feedbacks reflection during the transient.

6.3 Simulation Results

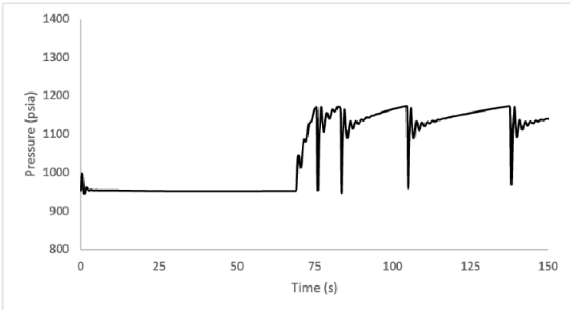
Results of current simulation are shown in Figures 10-14. During the CEA withdrawal, the core power increases and pressure and temperatures of RCS follow this trend. After 68.5 seconds of the accident initiation, the reactor is tripped, the turbine trip follows with a concurrent LOOP. Considering the delay in APR1400 system response, the PRZ pressure peaks in response to the power increase. Due to the LOOP, the core mass flow rate decreases as the RCPs coast-down and MFWS becomes unavailable. As the decay heat builds up in the plant after reactor trip, the pressure in the secondary side increases due to the limited heat removal capacity, which triggers the MSSVs to cycle (open/close) to release the steam and maintain the pressure. Long-term cooling through the plant shutdown must be maintained to offset the decay heat.



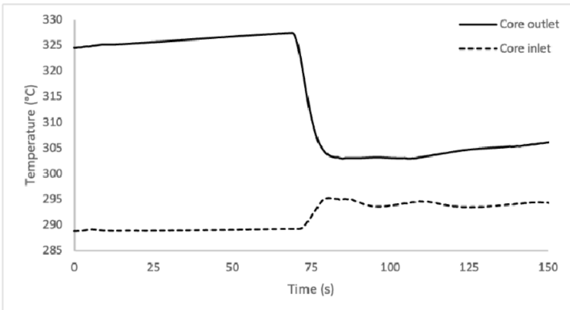
[Figure 10] Core Relative Power



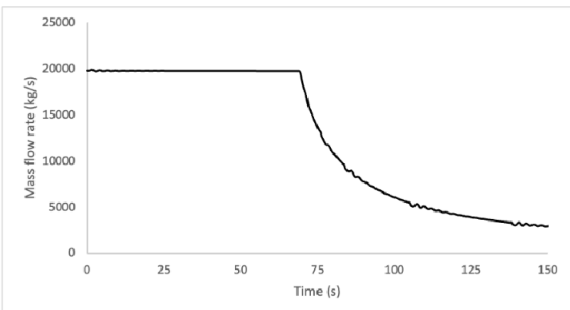
[Figure 11] Pressurizer Pressure



[Figure 12] Steam Generator Pressure



[Figure 13] Core Inlet and Outlet Temperatures



[Figure 14] Core Mass Flow Rate

6.4 Verification and Validation

The verification and validation process were conducted according to the V-Model Diagram shown in Figure 15, consisting of four main stages. Testing at each stage of the multi-physics simulation development is implemented to ensure that all the systems requirements are met.

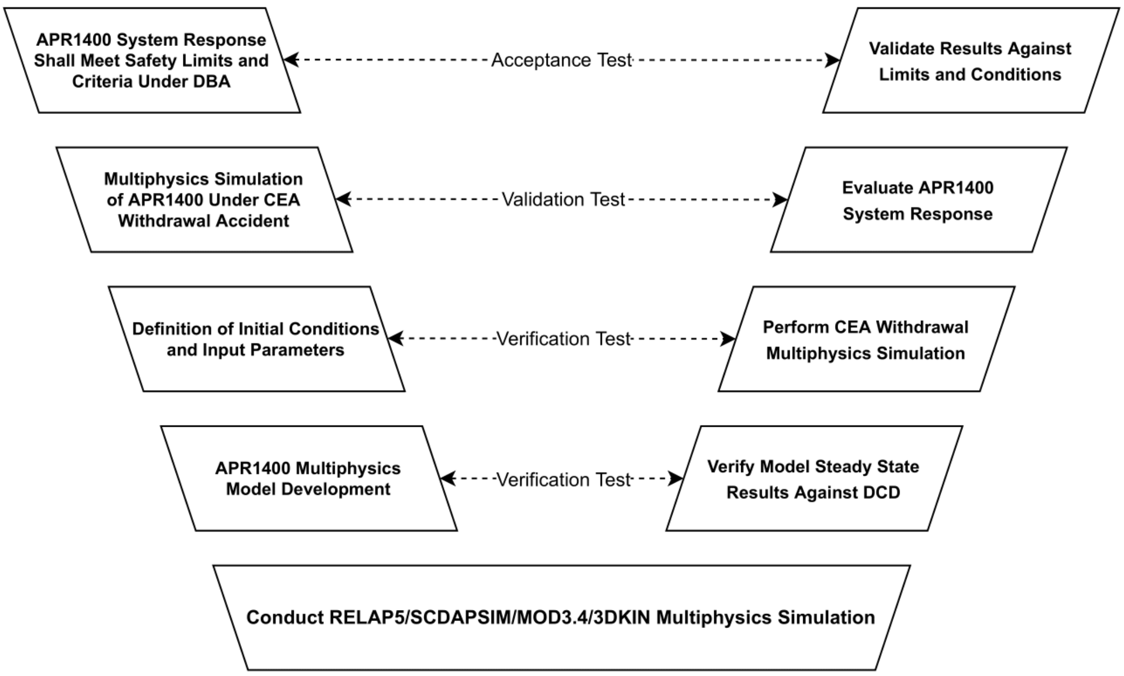
After the thermal-hydraulic and nodal kinetics models are developed, validation is done by comparing the simulation steady-state values to the values stated in the DCD Chapter 15, as shown in Table 3. The steady-state major simulation parameters are in reasonable agreement with the DCD values, with a deviation of less than 3 %, as shown in Table 4. The nodal kinetics 3DKIN model was compared at a nominal power condition, as can be found in the DCD Chapter 4. One of the most important parameters, the core power distribution, is also in a reasonable agreement, when compared to DCD.

<Table 4> Validation of Steady State Simulation

Parameter	DCD	Simulation	Deviation
Core thermal power, MWt	4062.66	4062.66	0.0 %
Pressurizer pressure, kg/cm²*	163.5	163.34	0.1 %
Reactor inlet coolant temperature, °C	287.8	288.8	0.3 %
Core mass flow rate, 10? kg/h	69.64	71.4	2.5 %
Steam generator pressure, kg/cm²*	68.26	68.27	0.0 %
CEA withdrawal speed, cm/min	76.2	76.2	0.0 %

* Absolute pressure

As shown in Figure 16, the highest deviation in the fuel assembly power is 7.03 %, which means the deviations are maintained within



[Figure 15] V-Model for Multi-physics Simulation of CEA Withdrawal Accident reasonable values given the nodal approach adopted in this simulation, reflects more precisely the core parameters, such as temperature, pressure, and mass flow rate.

Next, the CEA withdrawal scenario is simulated using the multi-physics approach. At this stage, the acceptance test shall satisfy the mission requirements, which were previously stated. The simulation results satisfy the research goal and show that the multi-physics simulation using RELAP5 and 3DKIN can be adopted in accident analysis for APR1400 plant.

7.03%	6.73%	4.00%	6.01%	1.05%	3.10%	0.31%	1.18%	2.84%
6.73%	5.19%	5.46%	3.11%	3.17%	0.14%	1.03%	0.96%	2.32%
3.99%	5.46%	3.41%	2.76%	0.09%	1.33%	1.53%	1.36%	1.97%
5.99%	3.10%	2.76%	0.23%	1.55%	0.97%	0.99%	2.35%	2.01%
0.83%	3.15%	0.11%	1.55%	2.32%	2.08%	4.50%	3.73%	
2.87%	0.12%	1.31%	0.98%	2.08%	5.01%	5.16%	4.33%	
0.50%	1.00%	1.54%	1.00%	4.51%	5.16%	4.33%		
1.22%	0.98%	1.38%	2.36%	3.74%	4.33%			
2.82%	2.30%	1.95%	2.00%					

[Figure 16] Deviation in Core Power Distribution

Table 5 is used to verify and validate that all systems requirements have been met.

<Table 5> Requirements and V/V Matching Process.

Requirements	V/V
APR1400 system response shall meet the safety criteria under a design basis accident condition.	APR1400 withstands accident condition while meeting safety criteria.
Multi-physics simulation shall show APR1400 system response under CEA withdrawal at power accident while satisfying safety limits (pin peaking factor, minimum DNBR, core heat flux, RCS pressure and temperature).	Multi-physics simulation shows that APR1400 system under CEA withdrawal at power accident does not exceed safety limits.
APR1400 system should comply with a design basis accident condition.	APR1400 system complies with a DBA conditions.
The power plant should withstand the CEA withdrawal accident with reasonable conservatism and safety margin.	Power plant can withstand the accident conditions, when conservative approach and safety margins are implemented.
Accident is considered at beginning of cycle, where core feedbacks have the worst effect and the withdrawing speed should be the maximal of 76.2 cm/min.	Simulation is conducted with BOC conditions, as well as maximum CEA withdrawal speed.
Minimum DNBR of 1.29, maximum pressurizer pressure of 2475 psia (110% of nominal pressure) and maximum peak linear heat generation rate of 656 W/cm cannot be exceeded.	Neither one of minimum DNBR, maximum pressurizer pressure and maximum peak linear generation rate are not exceeded.
Coupled RELAP5 and 3DKIN codes must be capable of complex analysis of the given accident.	Used codes are capable of multi-physics simulation of APR1400 accident.
Multi-physics simulation must be capable of modeling the core structure as individual fuel assemblies and calculate its power distribution.	Tools used for multi-physics simulation are able to model reactor core with individual fuel assemblies and conduct their analysis.
For accurate results of the simulation, proper mapping between fuel assemblies and core volumes and heat structures to exchange information is necessary.	Proper mapping between fuel assemblies and thermal-hydraulics structures is implemented for accurate simulation results, reflecting reactivity feedbacks.
A convergence criteria is defined to determine the simulation accuracy and iteration steps.	Appropriate simulation parameters are set to satisfy the convergence criteria.

7. Conclusion

In this paper, a systems engineering approach is adopted for multi-physics analysis of the Korean APR1400 reactor under CEA withdrawal at power accident condition. The systems engineering method together with the objective hierarchy are developed for the target system, multiphysics CEA withdrawal at power accident simulation. The functional as well as physical architecture described.

For the multi-physics analysis, the package RELAP5/SCDAPSIM/MOD3.4/3DKIN is used and thermal-hydraulics model of APR1400 together with reactor core kinetics model developed and adjusted for the analysis. Four main stages of verification and validation activities based on the V-Model are followed to ensure that predefined requirements are met with success criteria.

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