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

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Abstract : Deterministic safety analysis is a crucial part of safety assessment, particularly when it comes to demonstrating the safety of nuclear power plant designs. The traditional approach to deterministic safety analysis models is to model the nuclear core using point kinetics. However, this simplified approach does not fully reflect the real core behavior with proper moderator and fuel reactivity feedbacks during the transient. The use of Multi−Physics approach allows more precise simulation reflecting the inherent three−dimensionality (3D) of the problem by representing the detailed 3D core, with instantaneous updates of feedback mechanisms due to changes of important reactivity parameters like fuel temperature coefficient (FTC) and moderator temperature coefficient (MTC). This paper addresses a CEA ejection accident at hot full power (HFP), in which the underlying strong and un−symmetric feedback between thermal−hydraulics and reactor kinetics exist. For this purpose, a multi−physics analysis tool has been selected with the nodal kinetics code, 3DKIN, implicitly coupled to the thermal−hydraulic code, RELAP5, for real−time communication and data exchange. This coupled approach enables high fidelity three−dimensional simulation and is therefore especially relevant to reactivity initiated accident (RIA) scenarios and power distribution anomalies with strong feedback mechanisms and/or un−symmetrical characteristics as in the CEA ejection accident. The Systems Engineering approach is employed to provide guidance in developing the work in a systematic and efficient fashion.

Key Words : Systems Engineering, APR1400, Multi−Physics Simulation, Deterministic Safety Analysis, Two−Way Coupling, Reactivity Initiated Accidents

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1. Introduction

In order to thoroughly evaluate the safety of APR1400, Design Basis Accidents (DBAs) are divided into 7 different categories, based on the effect that the event has on the system irrespective of the cause. For a conservative analysis, the most limiting conditions are considered regarding the initiating cause, sequence of events, as well as initial and boundary conditions to prove the power plant safety with enough safety margins consistent with the US NRC standard review plan.[1]

The control element assembly (CEA) ejection accident at hot full power (HFP) belongs to the category of reactivity and power distribution anomalies. This event is initiated by the ejection of a CEA, which may occur due to mechanical failure resulting in rapid rupture of the control element drive mechanism housing or associated nozzle, ejecting the CEA drive shaft to the fully withdrawn position. As a result of the ejection, positive reactivity is inserted into the core, leading to the sharp increase of the core power and reactor coolant system (RCS) pressure almost instantaneously. This increase is mitigated by the combined effect of delayed neutron and Doppler feedback mechanisms, followed by reactor trip.

CEA ejection at HFP disturbs the core power distribution causing strong and asymmetric feedback between thermal-hydraulics and reactors kinetics. The interplay between thermal hydraulics and neutronics plays a significant role in the system response. For such an accident scenario, it is indispensable not only to model the three-dimensional core, but to directly transfer the information between the neutronics and the thermal hydraulic parts of the model in real-time. Multi-physics simulation using implicit code coupling enables real-time data exchange between the thermal-hydraulic (TH) code and the nodal kinetics (NK) code to simulate the response more precisely and realistically. For the purpose of this analysis, the multi-physics simulation package RELAP5/3DKIN/MOD3.4 is used where the TH code, RELAP5/MOD 3.4, is coupled with the NK code, 3DKIN.

2. Literature Review

The traditional approach used to simulate CEA malfunction scenarios adopted the point kinetics model with one-way coupling, which simplifies the core phenomena. To mention a few of the relevant studies, Lee et al.[2] and Yang et al.[3] model the CEA withdrawal accident using KNAP methodology and SPACE code, respectively.

However, according to Park[4] 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. More detailed and realistic representation assumes using multi-physics approach by code coupling as recommended by Jang et al..[5]

Park et al. used the 3-D NK code, ASTRA, the sub-channel analysis code, THALES, and the fuel performance code, FROST in their sensitivity studies for 3-D rod ejection. It is worthy to note that the codes are coupled by CHASER system to enable realistic safety analysis methodology.[6] A similar approach is used by Park et al.[7] who conducted a three-dimensional pin-wise analysis for CEA ejection accident. Accident scenarios including CEA malfunction were analyzed by Lee et al.[8] by coupling of CUPID code with MASTER code for multi-dimensional representation of thermal hydraulics parameters for OPR1000 as a base model. Recently, a multi-dimensional, multi-physics, multi-scale simulation was conducted by Park et al. for a main steam line break scenario.[9]

Mahmoud and Diab applied the multi-physics approach to simulate load follow operation.[10] Further, Hruškovič[11] conducted a multi-physics analysis of the CEA withdrawal accident scenario and proved that a higher safety margin and more flexible operation can be achieved using a high fidelity simulation.

As mentioned earlier, this work attempts to undertake a multi-physics analysis of a CEA ejection accident for APR1400. For efficient and timely execution of the current work, it is proposed to use the Systems Engineering (SE) approach. As such, the multi-physics analysisis considered as the 'system'under development for this work.

SE is currently been adopted to guide the design, development and optimization of a conceptual framework for a digital twin electric grid.[12] It has also been applied for innovative and sustainable manufacturing.[13] and for the conceptual design of a simulator.[14]

It is worth noting that the SE approach was successfully used to guide the development of several projects of similar nature as the project at hand.For example by Mahmoud and Diab [15] for the load follow simulation of the APR1400 nuclear power plant and by de Sousa and Diab[16] for uncertainty analysis of a Station Blackout (SBO) scenario. Regarding reactivity initiated accident scenarios, Hruškovič et al.[17] used the Systems Engineering approach to guide the development of a multi-physics analysis for a CEA withdrawal accident scenario.

3. Systems Engineering Approach

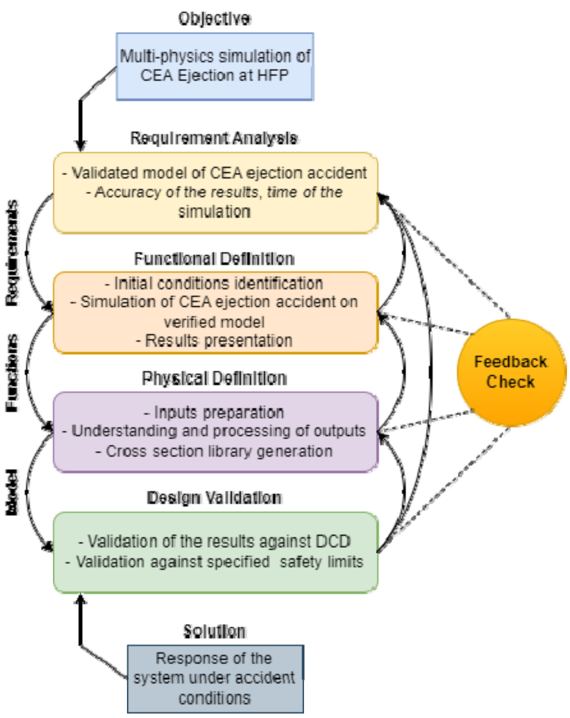
For the purpose of this work, the Kossiakoff method[18] is used for implementation of the SE approach. This method can be divided into the following steps:

- Requirement analysis;
- Functional definition;
- Physical definition;
- Design validation;

For successful implementation of the SE approach, the objective hierarchy, represented in Figure 1, is of utmost importance as it helps guide the work in a systematic and efficient fashion. It starts with identification of the work objective followed by derivation of the requirements based on the stakeholders needs. Next, drawing the system architecture (physical and functional) enables the system development. Finally, the validation step ensures that the execution of the project ultimately satisfies the requirements and hence leads to the achievement of the project objective.

As mentioned earlier, the objective is to conduct a Multiphysics analysis of the CEA ejection scenario. This work objective can be achieved by breaking it down into a set of smaller and manageable tasks. The work breakdown structure involves the following activities:

- Development of TH and NK model of CEA ejection accident analysis.
- Validation of the model against initial conditions and parameters for accident, as specified in APR1400 DCD.
- Performing the transient simulation of the accident, using coupled codes.
- Validation and verification of the obtained results.



[Figure 1] Kossiakoff SE Method Objective Hierarchy

3.1 Stakeholders Identification

Identification of interested and affected parties is paramount in the implementation of nuclear safety analyses and assuring the protection of people and the environment from any radiation risk. Several groups were identified as stakeholders, as listed in Table 1 covering economic, social, and technical needs or interests. All those groups have a keen interest in the safe and reliable operation of the nuclear power plant under normal as well as abnormal or transient conditions, for example under the CEA ejection scenario.

Further, stakeholder identification is particularly important for a key component of the SE approach, i.e. the derivation of the requirements to be met as the relevant tasks can be implemented to achieve the overarching project objective.

<Table 1> Stakeholders Identification

Category	Stakeholder
Economic	Utility Company Government Nuclear Industry Local Community
Social	Public Media
Environmental	Environmental Lobby Groups Neighboring Countries Government Environmental Legislators
Technical	Nuclear Regulator Researchers Contractors Employees

4. Requirements Development

After identifying the stakeholders' needs and interest, the basic requirements to be met at each of the developmental stages can be derived. The requirements related to this work can be divided to four different groups as specified in Table 2: mission, originating, system and simulation requirements.

4.1 Mission Requirements

The mission requirements are related to stakeholders' need to ensure the safe and reliable operation of the NPP system design under normal or even accident conditions. As such, it is deduced that for the current project, making the case for the CEA ejection design basis accident scenario meets the mission requirement.

<Table 2> Requirements for CEA Ejection Analysis

Requirements	Description
Mission requirements	Realistic simulation of accident scenario using multi-physics coupled analysis for APR1400
Originating requirements	Multi-physics simulation will demonstrate the response of APR1400 system during accident while ensuring that safety limits such as RCS pressure, core heat flux are maintained.
System requirements	The fuel design and primary system components must be capable to withstand physical processes during the accident
Simulation requirements	Coupled codes are capable of the analysis of the accident Multi-physics simulation must be capable of modeling the power distribution of the core and in individual assemblies Accurate mapping between volumes to enable information exchange is necessary Definition of convergence is required to determine accuracy of the simulation

4.2 Originating Requirements

The originating requirements are deduced from the stakeholders' needs based on the mission requirements. For a given operational scenario, the objectives are identified and then prioritized based on their relative importance. To achieve the mission objective, it is perceived that a realistic multi-physics reflecting the asymmetry and strong feedback mechanisms inherent in this accident scenario is deemed to be targeting the originating requirements. A precise representation of the plant design, and accurate modeling of key phenomena are precursors to a realistic prediction of the transient response of the power plant under the CEA ejection accident condition.

4.3 System Requirements

During accident conditions, crucial APR1400 systems must be capable to withstand the ongoing physical processes to ensure that, vital parameters are traced, along with safety limits. As the CEA ejection at HFP results in sharp increase of the power, some acceptance criteria need to be checked. Since power increase results in temperature and pressure increase, which may lead to a mechanical failure of fuel rods. The increase in fuel temperature may induce the fuel expansion and rod ballooning, causing strong pellet cladding mechanical interaction (PCMI).

In order to terminate the power excursion, reactor protection system (RPS) is triggered by variable overpower trip (VOPT) to initiate reactor trip and begin cooldown for a safe shutdown purposes.

4.4 Simulation Requirements

The simulation requirements have to be considered to ensure that it can be conducted successfully for analyzed scenario given its initial and boundary conditions. In addition, the capability and applicability of the used codes needs to be verified to ensure that the used tools will be suitable for accurately modelling the complex interactions and attendant phenomena with enough level of detail and high fidelity for this class of accident.

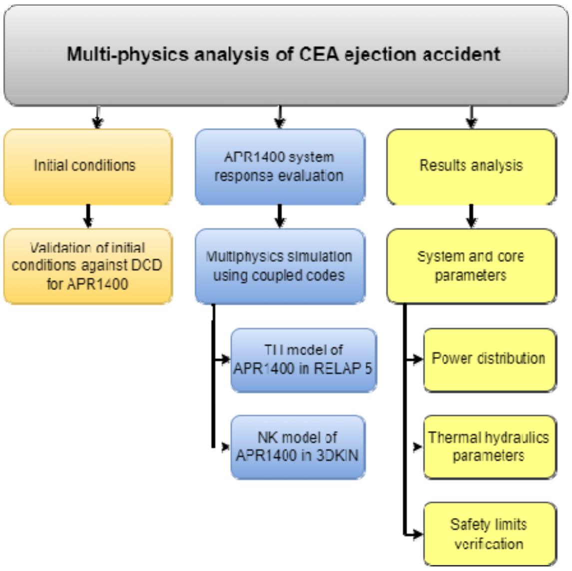
In order to successfully conduct multi-physics simulation, a verification and validation of individual codes must be conducted; specifically, the thermal-hydraulics model and the nodal kinetics must reflect the behavior of APR1400 power plant to ensure the credibility of the simulation. The power distribution representation needs to be precise and must be able to represent in detail the distribution at the assembly level. Moreover, the correct mapping between TH and NK volumes need to be created, considering the limitations of the tools as well as computational resources.

For the purpose of this work, a multi-physics package RELAP5/MOD3.4/3DKIN was selected. RELAP5 is a well-known thermal-hydraulics code, developed by Idaho National Laboratory (INL) and widely used in nuclear industry. 3DKIN is the nodal kinetics module, based on NESTLE code, that was developed by North Carolina State University (NCSU). 3DKIN is coupled to RELAP5 to reflect the core kinetics instead of the traditional point kinetics model. This allow the codes to exchange relevant information in real-time via implicit two-way coupling, hence yielding more accurate prediction of the NPP response.

5. System Architecture

5.1 Functional architecture

This section presents the functional architecture of the simulation which involves three levels as shown in Figure 2. According to the objective hierarchy, the overarching objective is to conduct a multi-physics simulation of the CEA ejection accident. To achieve this objective, a number of tasks (functions) have been identified according to the work breakdown structure. This is used to develop the functional architecture of the system under development.



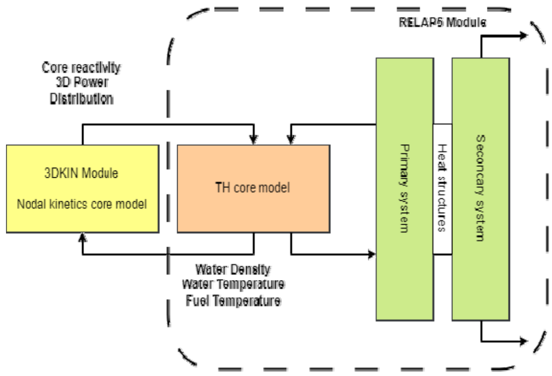
[Figure 2] Functional Architecture

The first level is related to the definition of initial conditions for the CEA ejection accident. The system should be represented using the conservative approach as specified in Chapter 15 DCD of APR1400.[19] For this purpose, conservative initial and boundary conditions are applied to the system and the results of the model are subsequently compared to their counterparts reported in the DCD for validation. Once the model is validated, the second level involves conducting the multi-physics simulation via the implicitly coupled codes, RELAP5 and 3DKIN. This step requires input preparation for each module separately: two input files for RELAP5 and five input files 3DKIN. In addition, proper mapping between RELAP5 thermal-hydraulic volumes and 3DKIN structural volumes to enable the exchange of relevant information between the corresponding nodes.

5.2 Physical Architecture

To carry out the functions listed in the previous section, it is essential to build the thermal-hydraulic and neutronics components of the simulation separately, confirm their credibility as stand-alone units and then integrate them in a fully functional system capable of conducting the multi-physics simulation to predict the NPP response under the CEA ejection accident scenario. This task refers to developing the physical architecture, represented in Figure 3, which illustrates the way the APR1400 system is modeled by using the coupled codes.

Additionally,Figure 3 represents the path of information exchange between the TH and NK codes. At each time step, key parameters calculated by the 3DKIN module, such as reactivity and power distribution are transferred to the TH model and are used as boundary conditions to generate the TH data in the RELAP5 module. Subsequently, the calculated TH parameters are then returned to 3DKIN as boundary conditions. The driver module checks if the boundary conditions exchanged between the codes match, if not, the process is repeated. This is an iterative process conducted at every time step until convergence is achieved.



[Figure 3] Physical Architecture

Inspecting Figure 3, two main steps: the first is related to the development of APR1400 TH model and the second step is related to the development of APR1400 NK model. The development of each of these models will be described next.

5.2.1 TH Model Description

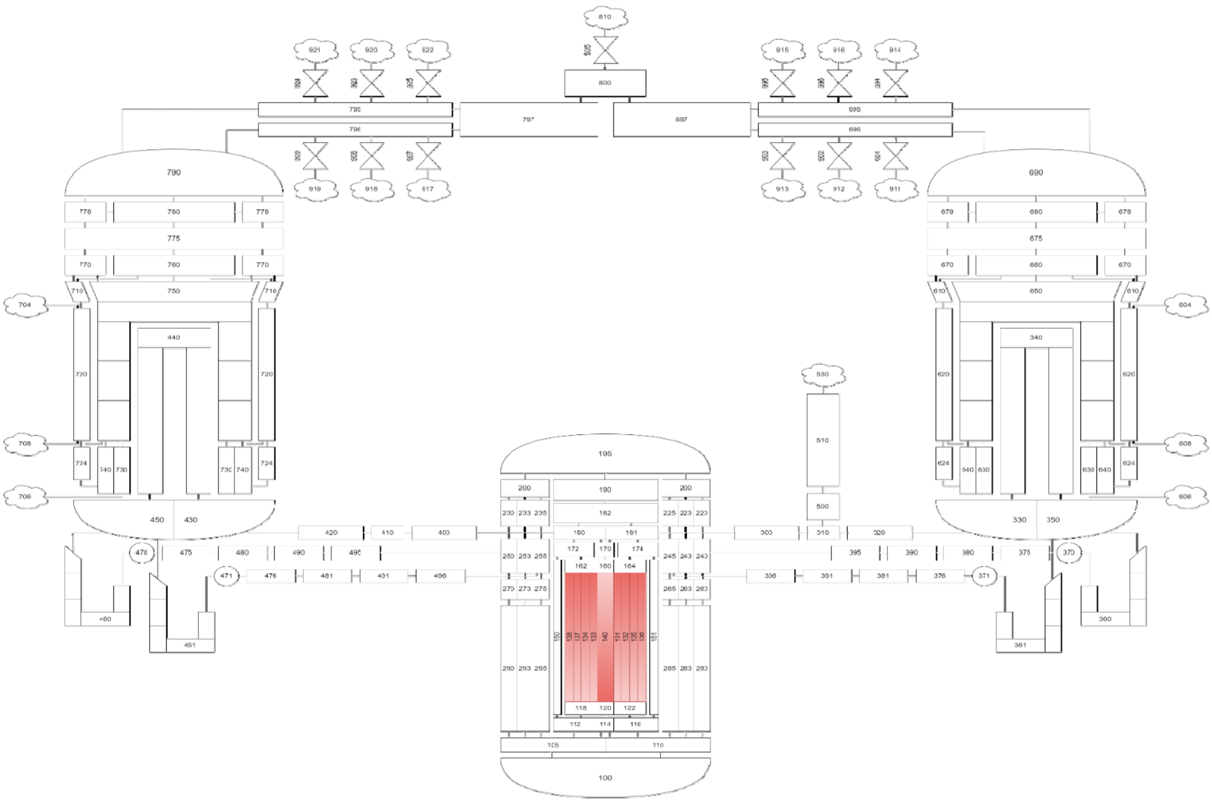
APR1400 is a PWR two-loop pressurized water reactor with a nominal power output of 1400MW. The development of the model by using RELAP5 includes the representation of all key systems and component of power plant, as shown in Figure 4. The nodalization represents reactor pressure vessel (RPV) as one of the key components. The core nodalization is prepared for mapping with 3DKIN. Bypass channels are also modeled, to realistically represent the flow through the core. Two Steam Generators (SG) are included, one for every loop. The pressurizer is connected to the system via a surge line to one of the hot legs. The effect of maintaining the pressure is achieved by a time-dependent volume connected to top of the pressurizer.

In order to maintain the water level within the SGs, the Main Feedwater System (MFWS) is represented via time dependent-volumes connected to downcomer and economizer accordingly. The feedwater is split so that 10 percent of the full-power feedwater flow goes to the downcommer while the remaining flow is directed through the economizer, to realistically reflect the actual feedwater flow.

Safety systems are crucial for the accident analysis and they are therefore implemented into the TH model. Three groups of main steam safety valves (MSSVs) are modeled on each steam line to prevent over pressurization and maintain the secondary system pressure within the design limits. Setpoints and mass flowrates are specified accordingly to the data provided in APR1400 DCD Chapter 4.[20] For conservatism, it is assumed that a loss of offsite power (LOOP) is concurrent with the reactor and turbine trips. The auxiliary feedwater system (AFWS) is modeled, to replace the MFWS when it is not available in the event of LOOP. Lastly, the pilot-operated safety relief valves (POSRVs) are connected to the pressurizer header to protect RCS boundary from over pressurization.

5.2.2 NK Model Description

The NK model is developed by using 3DKIN module. In order to fully reflect the realistic behavior of the core during the simulation, the core was modeled in accordance to the details specified in APR1400 DCD Chapter 4. The core is divided into 241 sections - every section represents single fuel assembly.



[Figure 4] APR1400 Model Nodalization

The arrangement of fuel assemblies is determined in accordance to the core design for the first cycle for APR1400. The quadrant model of the core, showing the arrangement of assem-blies is presented in Figure 5. Depending on the enrichment level, number of rods per assembly and control rod or water tubes, the fuel assemblies are divided into three different groups as specified in Table 3.

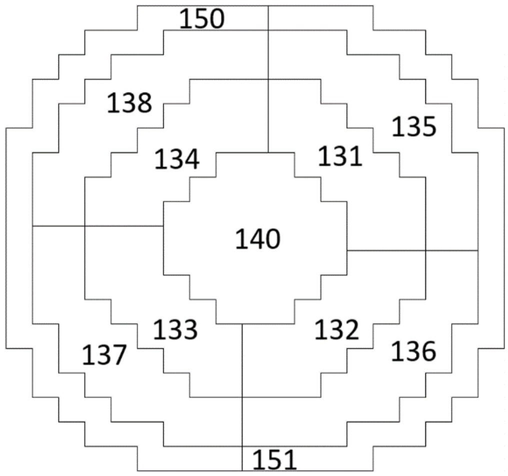
<Table 3> Fuel Assemblies Specification[10]

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

N F	N : BOX Number F : Fuel Type				1 C0	2 B0	3 C0	4 B0
			5 C0	6 C0	7 B2	8 B1	9 B3	10 C2
		11 C0	12 C1	13 B1	14 A0	15 C3	16 A0	17 B3
	18 C0	19 C1	20 B3	21 A0	22 B3	23 A0	24 B1	25 A0
	26 C0	27 B1	28 A0	29 C2	30 A0	31 C3	32 A0	33 B1
34 C0	35 B2	36 A0	37 B3	38 A0	39 B3	40 A0	41 B3	42 A0
43 B0	44 B1	45 C3	46 A0	47 C3	48 A0	49 C2	50 A0	51 C3
52 C0	53 B3	54 A0	55 B1	56 A0	57 B3	58 A0	59 B3	60 A0
61 B0	62 C2	63 B3	64 A0	65 B1	66 A0	67 C3	68 A0	69 A0

[Figure 5] Loading Pattern for a Quadrant Core Model[10]

In 3DKIN module, the active core is divided into 241 radial nodes each with 60 axial nodes, including the reflector. The mapping between TH and NK was achieved by assigning the nodes created in 3DKIN module, to the corresponding TH channels in RELAP5 module as illustrated in Figure 6. This step is necessary for splitting the amount of energy produced within the fuel volumes appropriately amongst the corresponding coolant volumes.



[Figure 6] Core Nodalization

For the NK model, 3DKIN module requires various cross-section (absorption, scattering and nu-fission) data to be provided for every fuel assembly. Also, in order to model precisely the CEA movements, those cross-section data should be obtained for both rodded and un-rodded cases. CASMO code was used to generate the cross section library for 3DKIN module. The 3DKIN code would then interpolate within the cross-section libraries provided to accurately reflect the feedback mechanisms and calculate the core reactivity.

5.2.3 Sequence of Events and Initial Conditions

The postulated mechanical failure of the control element drive mechanism is assumed to cause the ejection of a CEA. As a result, positive reactivity is added instantaneously to the core, which triggers a rapid increase in the reactor core power. This excursion is terminated by delayed neutron and Doppler feedback. Following the CEA ejection, reactor shutdown is initiated by a VOPT signal on high neutron power. Dropping of the shutdown control rods decreases the power rapidly due to the negative reactivity insertion.

The initial conditions and assumptions for the accident were chosen conservatively, to represent the worst possible scenario. For conservative analysis of this accident, LOOP is assumed to occur concurrently with the turbine trip. The core power level is set to 102% and the thermal-hydraulic parameters (maximum RCS coolant temperature and minimal RCS pressure) are set to maximize the energy increase in the fuel. To make the power increase faster and further, it is conservative to assume a minimum delayed neutron fraction for this accident. Further,positive feedback should be maximized which can be achieved by assuming the most positive moderator temperature coefficient (MTC), while assuming the least negative fuel temperature coefficient (FTC) to minimize the Doppler feedback during

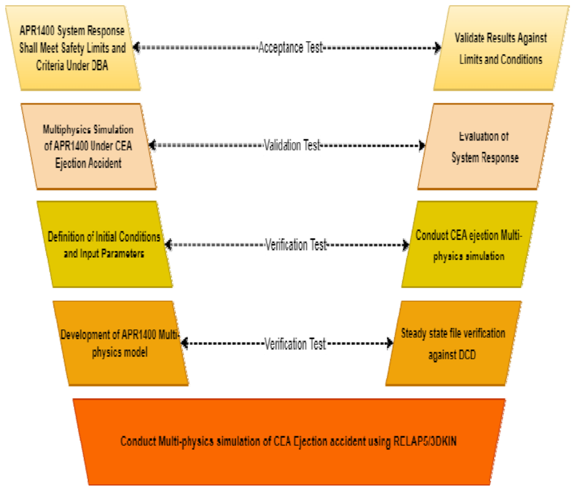
<Table 4> Initial Conditions for CEA Ejection at HFP

Parameter	DCD
Core power level, MWt	4062.66
Delayed neutron fraction	0.00412
MTC, 10−4Δρ/°C	0.00
FTC, Δρ/K1/2	−0.0013 0
Ejected CEA worth, 10 −2Δρ	0.1459
Core inlet coolant temperature, °C	295
Core mass flow rate, 106kg/hr	69.64
Pressurizer pressure, kg/cm2	152.9
Postulated CEA ejection time, sec	0.05

the power excursion. A summary of the assumptions and initial conditions for the CEA ejection analysis at HFP is provided in Table 4.

5.2.4. Model Verification

The model verification was conducted in accordance to the V-model shown in Figure 7, to ensure that V&V activities are implemented precisely at every stage of the development.



[Figure 7] V−Model for Multi−Physics Simulation of CEA Ejection Accident

To verify the credibility of the Multi-Physics model several simulations were conducted by using APR1400 nominal conditions and obtained results were then compared with DCD. If the results were not matching within acceptable limits (under 10% deviations), model was modified to match the APR1400.

Next verification step is related with proper simulation of CEA ejection to satisfy mission requirements that simulation will provide realistic and accurate results. After that, system response evaluation step is conducted to validate system response respecting originating requirements. Finally, acceptance step is done to qualitatively assess the system response that satisfies stakeholder's interest in safe operation of nuclear power plant.

6. Results

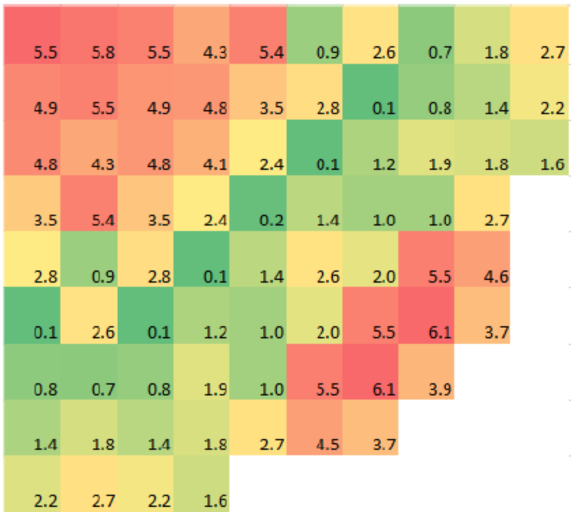
6.1 Steady State Verification

The first step refers to a steady state verification against DCD nominal operation values. This step is crucial in order to successfully run a credible transient simulation. Once the TH and NK are developed, both inputs are coupled by mapping the corresponding volumes to allow information exchange between the codes. The simulation is then conducted and the obtained values are compared to the initial conditions for CEA ejection accident in DCD. As Table 4 shows, the simulation predictions compare well with those published in DCD.

<Table 5> Steady-State Validation

Parameter	DCD	Simulation
Core thermal power, MWt	4026.66	4026.66
Pressurizer pressure, kg/cm^2	152.9	152.89
Core mass flow rate, $10^6 kg/hr$	69.64	69.61
Core inlet coolant temperature, C	295	295.7
MTC, 10-4 Δρ/˚C	0.00	0.00
Postulated CEA ejection time, sec	0.05	0.05

The results of the NK parameters were similarly compared to those of the DCD, but they show bigger deviations. The highest deviations occur in the central part of the core, with a maximum value of 6.1 %, as shown in Figure 8 (quadrant part of core is shown as distribution is symmetrical). The results are compared with nominal power distribution for APR1400. The model is deemed in reasonable agreement with the APR1400 operating conditions with deviations under 10%.



[Figure 8] Deviation in Core Power [quadrant core]

6.2 Future Work

This research is currently work in progress. Future work will cover the transient CEA ejection accident simulation using the multi-physics approach. Once the simulation will be done, the results will be evaluated to verify the system response. The last step of the scope of work is related to the validation of the results against the design limits and safety margin. The simulation results are foreseen to confirm that the multi-physics approach using the coupled package RELAP5/3DKIN can be successfully used for high fidelity analysis of RIA scenarios for APR1400 plant, providing larger safety margins and hence more operational flexibility.

7. Conclusion

In this paper, a Systems Engineering approach is used to guide the development of a multi-physics simulation of CEA ejection accident at HFP. The Systems Engineering approach was demonstrated to be an effective tool to address the project requirements and to show the relationships between each phase of the project development. Four main levels of verification and validation are specified based on the V-model to ensure that all requirements are satisfied.

To achieve the target of the project, RELAP5 TH code and 3DKIN NK code are coupled for multi-physics analysis tool suitable for RIA with unsymmetrical effects and strong feedback mechanisms. The obtained results should confirm that safety limits and criteria will be satisfied during accident conditions and all safety systems will prevent and minimize the consequence of the event.

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