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Analysis of CEA Withdrawal Accident Using BEPU Approach

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1. Introduction

Reactivity Initiated Accident (RIA) scenarios, such as Uncontrolled Control Element Assembly (CEA) Withdrawal, require special attention using advanced simulation techniques due to their complexity and importance for nuclear power plant (NPP) safety. While the conservative approach has traditionally been used for safety analysis, it may lead to unrealistic results which calls for the use of best estimate plus uncertainty (BEPU) quantification, especially with the current advances in computational power which makes BEPU analysis feasible [1].

This paper presents the BEPU analysis of a CEA Withdrawal accident at Full Power with concurrent Loss of Offsite Power (LOOP) for APR1400. This is achieved by employing MARS-KS thermal hydraulics system code with point kinetics model coupled with the DAKOTA to propagate key uncertain parameters using an uncertainty quantification framework developed using Python.

2. Literature Review

There are numerous works related to the subject of RIAs, in which uncontrolled CEA withdrawal at full power scenario was investigated. In case of APR1400, the Design Control Document (DCD) presents the conservative analysis of all postulated accidents (PA) and anticipated operational occurrences (AOO) for licensing purposes. For CEA withdrawal at full power scenario, the analysis in DCD was conducted using a model developed with CESEC-III code [2]. Further, Lee et al. developed the Korean Non-LOCA Accident Package (KNAP) where RETRAN-3D code was applied to analyze the CEA withdrawal [3]. Additionally, Jang et al. used iSAM methodology, also based on RETRAN code, to analyze CEA withdrawal for OPR1000 [4]. In a work by Yang [5], the results of applying SPACE code for CEA withdrawal at full power were compared to the results achieved using KNAP methodology for OPR1000 with reasonable agreement. It is, however, important to mention that in all the above cases the results were conservative with some deviations in relation to DCD.

Despite the ease of using conservative analyses, the results can lead to unrealistic safety margins. This necessitates the use of a realistic approach using BEPU analysis which is proposed in this work by coupling MARS-KS thermal hydraulics system code to the statistical tool, DAKOTA. Recent research indicates that BEPU analysis conducted using RELAP5 with one-way coupling using point kinetics is capable of generating satisfying results in regards to transient analyses for pressurized water reactors (PWR) [6]. According to Park [7], as much as the point kinetics model with one-way coupling is convenient in analyses applying conservative approach, it also leads to significant loss of information on the system behavior due to its degree of simplification, and consequently resulting in a decreased safety margin. Therefore, two-way coupling model development is recommended as a future improvement, despite of its complex nature.

3. Model Description

The model nodalization of APR1400 nuclear power plant (NPP) is developed using the MARS-KS thermal hydraulics system code as depicted in Figure 1. The model includes key systems and components of the NPP primary and secondary circuits that are relevant to the analyzed accident. The vital part for the CEA withdrawal at full power accident is the core, which has been split into an average channel and a hot channel in order to estimate the Departure from Nucleate Boiling Ratio (DNBR) as it evolves during the accident. This parameter is crucial for successful safety assessment and constitutes one of the safety criteria to be satisfied for the analyzed scenario, along with the peak linear heat generation rate and RCS pressure according to General Design Criteria (GDC) specified in the US NRC 10 CFR Part 50 regulations document, Appendix A [8]. Minimum DNBR calculation is performed using the W-3 correlation [9].

APR1400 includes two loops, each including one steam generator (SG), one hot leg, two cold legs, and two reactor coolant pumps (RCPs). To control pressure in the reactor coolant system (RCS) the pressurizer is attached to a hot leg of one loop via a surge line. The SGs tube sections are modeled to provide heat exchange between primary and secondary loops.

The secondary circuit consists of the two SGs connected to the Main Feed Water System (MFWS). Two steam lines are connected to the upper part of each SG, directing the steam generated as a result of the heat exchange with the primary loop to the turbine.

Key safety systems relevant to the accident include the Pilot-Operated Steam Relief Valves (POSRVs) attached to PRZ, the Auxiliary Feed Water System (AFWS) which delivers feedwater in the event of LOOP and finally the Main Steam Safety Valves (MSSV) connected to the steam lines to protect the secondary circuit from over-pressurization.

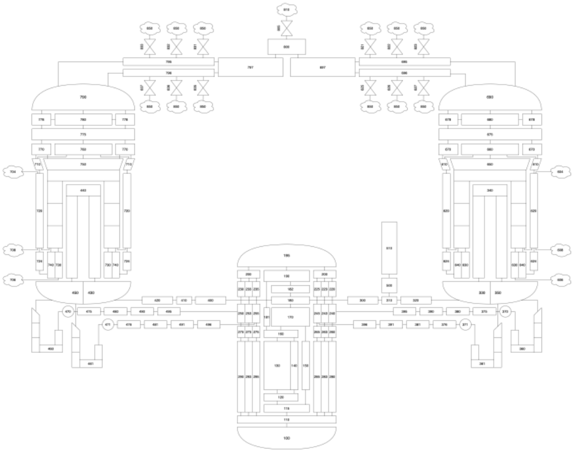

Figure 1. APR1400 Nodalization

4. CEA Withdrawal at Power Accident Description

Uncontrolled Control Element Assembly (CEA) Withdrawal at Full Power condition 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. LOOP concurrent with reactor trip is assumed for conservatism in compliance with the US NRC Standard Review Plan criteria for uncontrolled control rod assembly withdrawal at power [10].

During CEA Withdrawal at Full Power, the fifth group of CEAs is withdrawn from the reactor core, leading to reactor power and core heat flux increase, effectively causing RCS temperature and pressure to increase as well. As such, the Specified Acceptable Fuel Design Limits (SAFDL) may be approached, specifically regarding DNBR and fuel centerline melt temperature, which necessitates action from the Reactor Protection System (RPS). The transient may thus be terminated by either Core Protection Calculator (CPC) on Variable Overpower Trip (VOPT), low DNBR trip, high Local Power Density (LPD) trip, or High Pressurizer Pressure Trip (HPPT). Simultaneous LOOP results in turbine trip, effectively disabling all the equipment dependent on its operation.

5. Methodology and Results

MARS thermal-hydraulics system code developed by the Korean Atomic Energy Research Institute (KAERI) is used for the accident analysis [11]. The model is tuned to fit the accident scenario. One-way coupling using the point kinetics model and relevant reactor kinetics data in the form of reactivity tables was implemented to reflect the core behavior. In order to validate the model, the conservative initial conditions provided in Chapter 15 of APR1400 DCD were set and the reported results compared with model predictions for verification. First, the parameters were selected, so that the system is driven to achieve the lowest possible value of minimum DNBR, as listed in Table 1.

Table 1. Initial Conditions for CEA Withdrawal at Full Power

Parameter	Model	DCD	Deviation
Core power level, MWt	4062.66	4062.66	0,00%
Core inlet coolant temperature, °C	287.74	287.8	0,02%
Core mass flow rate, 10^6 kg/hr	69.64	69.64	0,00%
Pressurizer pressure, kg/cm^2	163.5	163.5	0,00%
Integrated radial peaking factor	1.49	1.49	0,00%
Initial core minimum DNBR	1.839	1.72	6,92%
Steam generator pressure, kg/cm^2	68.262	68.26	0,00%
Moderator temperature coefficient	Most positive	Most positive	N/A
Fuel temperature coefficient	Least negative	Least negative	N/A
CEA worth on trip, %Δρ	-8.0	-8.0	0,00%
Reactivity addition rate, 10^{-4}Δρ/sec	0.315	0.315	0,00%
CEA withdrawal speed, cm/min	76.2	76.2	0,00%

The initial conditions were judged to be accurately translated, given that key system parameters deviate within a 5% bound from DCD, except for the minimum DNBR. This discrepancy may be attributed the different methodology applied for DNBR calculation. In the DCD it is calculated via the CETOP code using KCE-1 correlation, whereas the calculation in the developed model is conducted using W-3 correlation implemented in MARS-KS as a control variable. With the initial conditions verified and set, the model was run and the results for key parameters were generated, as depicted in Figures 2 through 5 and listed in Table 2.

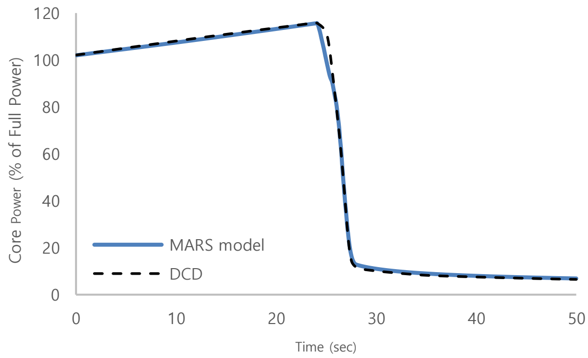

Figure 2. Core Power vs. Time

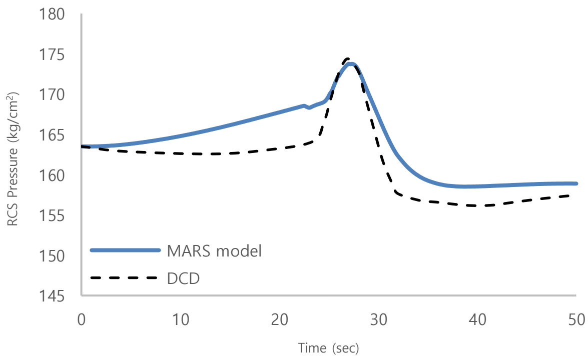

Figure 3. RCS Pressure vs. Time

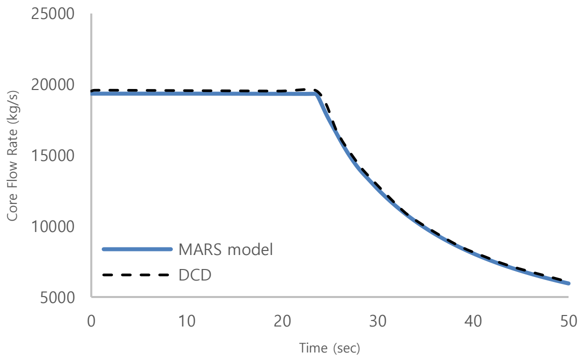

Figure 4. Core Flow Rate vs. Time

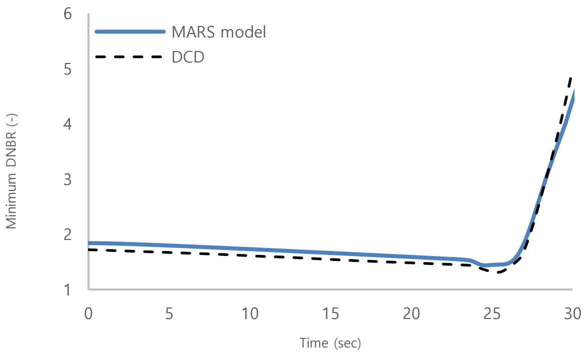

Figure 5. Minimum DNBR vs. Time

Table 2. Extreme values of safety parameters

Parameter	DCD	Model	Deviation
Maximum core power, % of design power	115.56	115.67	0.10%
Minimum DNBR	1.31	1.4364	9.65%
Maximum pressurizer pressure, kg/cm^2	172.97	173.76	0.46%

The model predictions indicate reasonable agreement with the results of the analysis reported in DCD and hence the model was judged as valid. Furthermore, the minimum DNBR and the peak linear heat generation rate remained inside the bounds of regulatory limits.

Next, the initial values of core inlet temperature and steam generator pressure were changed to 295° C and 75.86 kg/cm^2 respectively, while keeping all other parameters unchanged in order to push the system to its peak pressure limit for the considered accident scenario. As a result, it was concluded that the maximum pressure of 180.69 kg/cm^2 at the RCP outlet being the maximum RCS pressure point is below 110 percent of design pressure limit equal to 193.38 kg/cm^2, and therefore the safety criterion is satisfied.

Subsequently, the model initial parameters were adjusted to reflect nominal operation conditions at steady-state based on the data from Chapter 1 of the DCD, and modified to be interfaced with DAKOTA uncertainty quantification framework to perform BEPU analysis [12].

In order to conduct the safety analysis using BEPU approach, the statistical tool, DAKOTA, was coupled with MARS-KS using a Python script to develop the uncertainty quantification framework. This way the random propagation of key uncertain parameters derived from the underlying phenomena which govern the accident progression may be automated. The uncertain parameters were selected based on uncertainty and sensitivity analysis for reactivity initiated accidents performed by Marchand et al. [13]. Given their significance, a total of 19 parameters were chosen and divided into 4 categories according to the Phenomena Identification and Ranking Table (PIRT). Each parameter was assigned a mean value (μ), a standard deviation (σ), a probability distribution function (PDF), and boundary limits (min ~ max).

Table 3. Uncertain Parameters for Uncertainty Analysis

PIRT	Uncertainty parameter (unit)	μ	σ	PDF	min	max
Fuel Manufacturing Tolerances	Cladding outside diameter (mm)	9.40	0.01	Normal	9.38	9.42
	Cladding inside diameter (mm)	8.26	0.01	Normal	8.24	8.28
	Fuel theoretical density (kg/m^3 at 20°C)	10970	50	Normal	10870	11070
	Fuel porosity (%)	4	0.5	Normal	3	5
	Cladding roughness (μm)	0.1	1	Normal	10^{-6}	2
	Fuel roughness (μm)	0.1	1	Normal	10^{-6}	2
	Filling gas pressure (MPa)	2.0	0.05	Normal	1.9	2.1
TH boundary conditions	Coolant pressure (MPa)	15.500	0.075	Normal	15350	15650
	Coolant inlet temperature (°C)	280	1.5	Normal	277	283
	Coolant velocity (m/s)	4.00	0.04	Normal	3.92	4.08
Core Power	Injected energy in the rod (J)	30000	1500	Normal	27000	33000
	Full width at half maximum (ms)	30	0%	Normal	20	40
Thermo-physical properties and key heat transfer models	Fuel thermal conductivity model	1.00	5%	Normal	0.90	1.10
	Clad thermal conductivity model	1.00	5%	Normal	0.90	1.10
	Fuel thermal expansion model	1.00	5%	Normal	0.90	1.10
	Clad thermal expansion model	1.00	5%	Normal	0.90	1.10
	Clad yield stress	1.00	5%	Normal	0.90	1.10
	Fuel enthalpy/heat capacity	1.00	1.5%	Normal	0.97	1.03
	Clad-to-coolant heat transfer	1.00	12.5%	Normal	0.75	1.25

The first category includes parameters related to fuel rod manufacturing tolerances, such as fuel, gap, and cladding dimensions which may differ from the nominal design values. The following category reflects the variability and fluctuations of the thermal hydraulic conditions during operation. Similarly, the core power may slightly deviate from the nominal power level and the control rod worth or CEA reactivity values may be perturbed as identified by the third category. The last category refers to uncertainty in thermo-physical properties as well as heat transfer models and correlations.

The simulation is run multiple times using the uncertainty quantification framework, the values of uncertain parameters are randomly selected by DAKOTA and passed to MARS-KS until a statistically representative sample size is reached. By randomly selecting the combinations of uncertain parameters, the modeled system undergoes CEA withdrawal accident for different initial states without bias. A schematic depiction of the uncertainty quantification framework is shown in Figure 6.

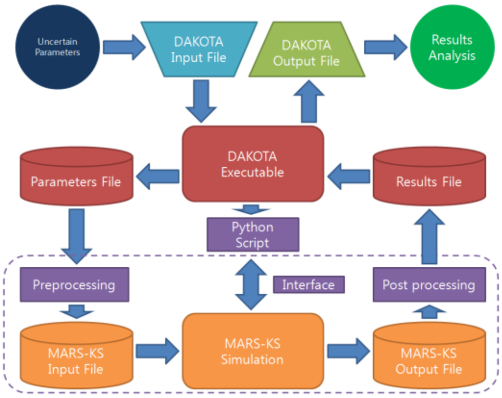

Figure 6. Uncertainty quantification process

The number of simulations necessary to generate credible results has been determined based on the fifth order Wilks formula for one sided distribution. Fifth order was selected for the current work since it is generally used in the safety evaluation, following the work of Han and Kim [14]. For a 95% probability and 95% confidence level, a fifth order Wilks a statistically acceptable sample size is 181. The results for DNBR and RCS pressure from all simulations are shown in Figures 7 and 8.

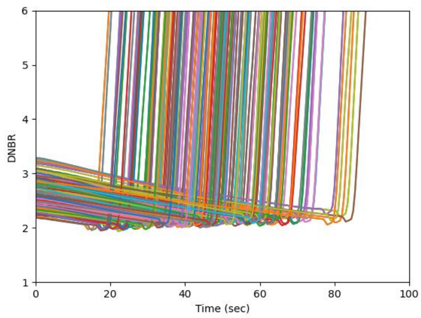

Figure 7. DNBR vs. Time (181 simulations)

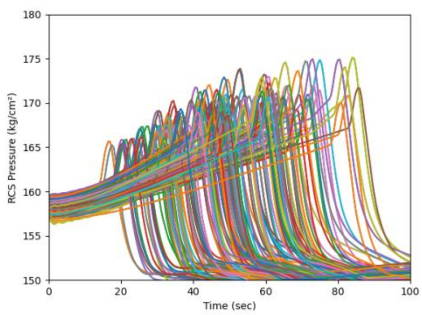

Figure 8. RCS pressure vs. Time (181 simulations)

As explained by Chung et al., the fifth order Wilks formula determines the limiting values that satisfy the 95/95 US NRC rule as being the fifth lowest and the fifth highest values for DNBR and peak RCS pressure, respectively, among the 181 samples. [15]

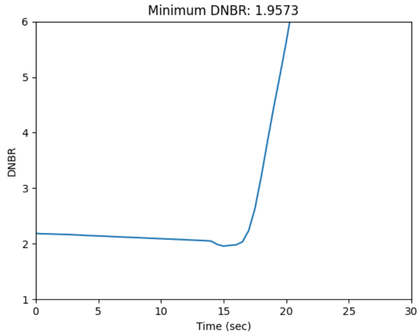

Figure 9. DNBR vs. Time (selected curve)

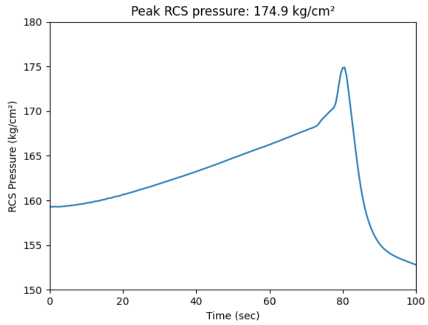

Figure 10. RCS pressure vs. Time (selected curve)

Using BEPU methodology, the safety margin is more realistically represented for the various critical safety parameters compared to those calculated using the conservative approach. This is expected given that several limiting assumptions adopted in the conservative approach are now replaced by realistic inputs along with the consideration of various input uncertainties.

For this class of accidents that involve uneven power distributions with strong feedback mechanisms, high fidelity simulations e.g. using multi-physics simulations where two-way coupling between the thermal hydraulics and neutronics is considered can provide more realistic response and further increase the safety margin. The model is therefore subject to further improvement, and outcomes of the analysis are expected to evolve towards increased accuracy.

6. Conclusions

This paper presents the realistic analysis of a reactivity initiated accident; specifically CEA withdrawal at power using the BEPU approach. A thermal hydraulic model of APR1400 is developed using MARS-KS code and cross-validated against results reported in the APR1400 DCD. Subsequently, an uncertainty quantification framework has been developed by loosely coupling MARS-KS to the statistical software, DAKOTA. Results from the BEPU indicate that a realistic treatment of the accident scenario yields a more reasonable safety margin and is therefore encouraged for accident analysis. Additionally, high fidelity simulations e.g. using multi-physics approach is increasingly recognized especially given the computational potential of modern day machines. This is particularly relevant to the class of accidents that involve uneven power anomalies or in transients that involve strong feedback mechanisms, such as CEA withdrawal at low power or single CEA drop, as well as in other events related to CEA misoperation.

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REFERENCES

[1] D'Auria F., Best Estimate Plus Uncertainty (BEPU): Status and perspectives, Nuclear Engineering and Design, Vol. 352, 2019.

[2] Korea Hydro and Nuclear Power Co., Ltd, APR1400 Design Control Document Tier 2: Chapter 15 Transient and Accident Analyses, Revision 3, 2018.

[3] Lee Dong-Hyuk, Yang Chang-Keun, Kim Yo-Han, Sung Chang-Kyung, APR1400 CEA Withdrawal at Power Accident Analysis using KNAP, Transactions of the Korean Nuclear Society Spring Meeting, May 25-26, 2006, Chuncheon, Korea.

[4] Jang, Chansu, Um Kilsup, Applications Of Integrated Safety Analysis Methodology to Reload Safety Evaluation, Nuclear Engineering and Technology, Vol. 43, pp. 187-194, 2011.

[5] Yang Chang-Keun, Lee Dong-Hyuk, Ha Sang-Jun, OPR1000 CEA Withdrawal at Power Accident Analysis Using the SPACE Code, Transactions of the Korean Nuclear Society Autumn Meeting, October 27-28, 2016, Gyeongju, Korea.

[6] Casamor M., Avramova M., Reventós F., Freixa J., Off-line vs. semi-implicit TH-TH Coupling Schemes: A BEPU Comparison, Annals of Nuclear Energy, Vol 178, 2022.

[7] Park Min-Ho, Park Jin-Woo, Park Guen-Tae, Um Kil-Sup, Ryu Seok-Hee, Lee Jae-Il, Choi Tong-Soo, 3-D Rod Ejection Analysis Using a Conservative Methodology, Transactions of the Korean Nuclear Society Autumn Meeting, October 27-28, 2016, Gyeongju, Korea.

[8] U. S. Nuclear Regulatory Commission, Regulations Title 10, Code of Federal Regulations, Part 50, Appendix A: "General Design Criteria for Nuclear Power Plants", 2007.

[9] Tong L. S., Heat Transfer In Water-Cooled Nuclear Reactors, Nuclear Engineering and Design, Vol. 6, pp. 301-324, 1967.

[10] U. S. Nuclear Regulatory Commission, Standard Review Plan: 15.4.2 "Uncontrolled Control Rod Assembly Withdrawal at Power", 2007.

[11] Korea Atomic Energy Research Institute, MARS-KS Code Manual Volume II: Input Requirements, 2010.

[12] Korea Hydro and Nuclear Power Co., Ltd, APR1400 Design Control Document Tier 2: Chapter 1 Introduction and General Description of the Plant, Revision 3, 2018.

[13] Marchand Oliver, Zhang Jinzhao, Cherubini Marco, Uncertainty and Sensitivity Analysis in Reactivity-Initiated Accident Fuel Modeling: Synthesis Of Organization For Economic Co-Operation and Development (OECD)/Nuclear Energy Agency (NEA) Benchmark on Reactivity-Initiated Accident Codes Phase-II, Nuclear Engineering and Technology, Vol. 50, pp. 280-291, 2018.

[14] Han Seola, Kim Taewan, Numerical Experiments on Order Statistics Method Based on Wilks' Formula for Best-Estimate Plus Uncertainty Methodology, Journal of Environmental Management, Vol 235, pp. 28-33, 2019

[15] Chung Bub Dong, Hwang Moon-kyu, Bae Sung Won, Analysis of APR1400 LBLOCA and Uncertainty Quantification by Monte-Carlo Method, Comparing with Wilks' Formula Approach, Transactions of the Korean Nuclear Society Autumn Meeting, October 27-28, 2011, Gyeongju, Korea.