Transactions of the Korean Nuclear Society Autumn Meeting
Changwon, Korea, October 24-25, 2024

Multiphysics Simulation of an Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

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***Keywords:** IOSGADV, APR1400, RELAP5/3DKIN

1. Introduction

In this paper, a comprehensive safety analysis of the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (IOSGADV) accident for the APR1400 reactor is presented. Analysis of this accident scenario is conducted by two different approaches. Firstly, using a thermal hydraulic model with simple point kinetics by RELAP5, which is evaluated for total 181 cases with different initial and boundary conditions to simulate the BEPU approach using DAKOTA tool. Secondly, using thermal hydraulic model coupled with nodal kinetics using the RELAP5/SCDAPSIM/MOD3.4/3DKIN code package for multi-physics simulation reflecting the real three-dimensional core behavior.

The Advanced Power Reactor 1400 (APR1400) is an advanced Pressurized Water Reactor (PWR) developed with enhanced safety, efficiency, and reliability. This reactor is a significant evolution in Korean nuclear technology, emphasizing improved safety features and operational performance with increased power. Despite the safety advancements, the IOSGADV still remains a critical safety concern [1]. This type of a Design Basis Accident (DBA) is initiated by an unintended release of steam from the Main Steam Line (MSL) from one of the Steam Generators (SGs) directly into atmosphere, posing numerous operational and safety challenges. The Atmospheric Dump Valve (ADV) plays a crucial role in the safe accident management and mitigation strategies within the operating procedures of nuclear reactors. However, the inadvertent opening of the ADV can lead to a range of adverse outcomes, primarily due to the uncontrolled release of steam. This can result in a rapid decrease in SG pressure and, consequent loss of water inventory, which also leads in change of the primary side cooling and is therefore critical in maintaining the reactor thermal and reactivity balance.

In the final stage, both models were utilized with the uncertainty quantification technique using DAKOTA. This involves systematically assessing and quantifying the uncertainties in the model inputs and parameters to better understand the range of possible outcomes and the confidence level in the predicted reactor responses. This step was crucial for providing a robust and reliable analysis of the IOSGADV scenario, accounting for potential variability in the reactor behavior and overall plant response.

2. Methodology

The approach undertaken in this module involves two main steps. First, a thermal-hydraulic model coupled with point kinetics is developed using RELAP5 code. The results are verified against the results published in the APR1400 Design Control Document (DCD) under conservative conditions. Next, a more detailed multi-physics analysis is undertaken by activating the nodal kinetics module coupled within the code package of RELAP5/SCDAPSIM/MOD3.4/3DKIN to reflect the asymmetric cooling of the core that ensues from the IOSGADV scenario. This can be achieved via a three-dimensional representation of the reactor core with real-time feedback between the thermal hydraulics and neutronics. [1]

Finally, an uncertainty quantification framework is developed to enable a realistic analysis of the accident, by applying the Best Estimate Plus Uncertainty (BEPU) methodology using the Design Analysis Kit for Optimization and Terascale Applications (DAKOTA).

2.1 Point Kinetics Model Using RELAP5

A simple point kinetics model is integrated in the RELAP5 to account for the reactor core behavior, including power changes and reactivity feedback. The thermal-hydraulic behavior affects the neutron kinetics through temperature and density changes in the moderator and fuel, while the kinetics influence the thermal-hydraulic behavior by altering the core heat generation rate. The nodalization of APR1400 model used for the IOSGADV analysis is shown in Figure 1 and initial conditions are listed in Table 1. [1, 2]

Table 1. Initial Conditions [1]

Initial Parameter	DCD	Conservative
Core power level, MWt	4062.66 (102 %)	4051.90 (101.73%)
Core inlet coolant temperature, °C	296.1	297.39
Core mass flow rate, kg/hr	85.03	85.65
Pressurizer pressure, kg/cm2A (psia)	163.46	166.52
Pressurizer water volume, m3	13.56	12.27

SG inventory, kg per SG	127131	121184
CEA worth on trip, %Δρ	−8.0	−8.0
MTC, Δρ/°C	−5.4 × 10-4	−5.4 × 10-4
Core burnup	End of cycle	End of cycle
ASI	0.3	0.3
Maximum radial peaking factor	2.0552	2.0552
Doppler reactivity	Least negative	Least negative

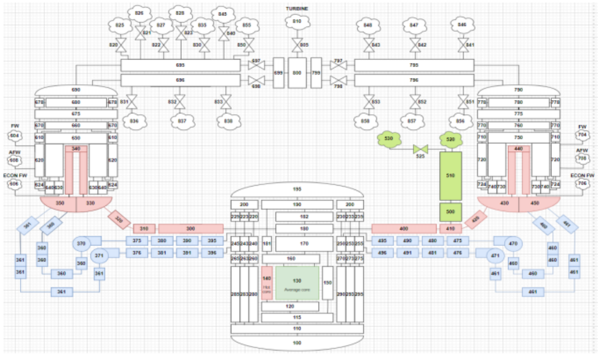

Figure 1. APR1400 Nodalization

2.2 Thermal-hydraulic model using RELAP5/3DKIN

In the RELAP5/3DKIN coupled model, the thermal-hydraulic calculations are performed concurrently with the three-dimensional reactor kinetics calculations. The nodal kinetics approach provides detailed feedback on the power distribution to the thermal-hydraulic model, while the thermal-hydraulic conditions (such as coolant density and temperature) influence the reactivity and neutron flux distribution in the reactor core. This real-time coupling ensures that the complex interactions between reactor kinetics and thermal-hydraulics are accurately represented, particularly during asymmetric or localized transients. [2]

2.3 Uncertainty Quantification

DAKOTA is a powerful toolkit for performing uncertainty quantification, optimization, and sensitivity analysis. When applied to thermal-hydraulic models like RELAP5 and RELAP5/3DKIN, DAKOTA helps in understanding of the impact of input uncertainties on simulation outputs and in improving the robustness of safety assessments. The uncertainty quantification process involves identifying uncertainties, generating samples, running the simulations, analyzing outputs, and sensitivity analysis. For uncertainty quantification, the Wilks-based method is used. This method specify the desired confidence level (e.g. 95 %) and coverage probability (e.g. 95 %) for the output variable. The Wilks-based method is widely used in the regulatory framework and safety analysis in the nuclear industry due to its simplicity. For this paper, 5th order was chosen with total 181 samples. [3, 4]

3. Results

This study presents a comprehensive analysis of the outcomes from two simulation models following the inadvertent opening of a steam generator atmospheric dump valve. The conservative model utilizes initial conditions sourced directly from the APR1400 DCD, representing a scenario that is intentionally biased towards safety. This model is designed to ensure the plant operation remains within safety limits even the under worst-case conditions.

In contrast, the UQ model incorporates variability through 181 samples, selected according to the Wilks' based method at the 5th order. This method allows for a probabilistic evaluation of the system behavior by considering uncertainties in various parameters. By comparing these two models, the analysis provides insights into the differences between a conservative, deterministic approach and a probabilistic approach that accounts for uncertainties.

The key finding from this comparison is the significant margin observed between the two models, particularly in terms of DNBR. The conservative model reports a minimum DNBR of 1.2976, which is significantly lower than the 1.5438 minimum DNBR observed in the UQ model. This difference underscores the conservative nature of the DCD-based model, which is designed to err on the side of caution to maintain safety under extreme conditions.

The higher DNBR value in the UQ model suggests that even when accounting for uncertainties, the system maintains an adequate safety margin. This indicates that the plant could operate safely even under conditions that deviate from the idealized parameters used in the conservative model.

Table 2. Sequence of events for IOSGADV [1]

Event	Setpoint	Time
ADV opens fully	-	0.0
Rector trip	-	1800.0
MFIVs close completely	-	1800.1
Minimum transient DNBR	1.336	1802.1
Pressurizer pressure reaches safety injection actuation signal analysis, kg/cm2A	121.98	1967.0
SG pressure reaches main steam isolation signal, kg/cm2A	57.09	1956.0
Safety injection flow begins	-	2007.0
MSIVs close completely	-	2100.0

SG water level reaches auxiliary feedwater actuation	19.9	2324.6
Operator manually closes ADV	-	3000.0
Operator initiates plant cooldown	-	3600.0

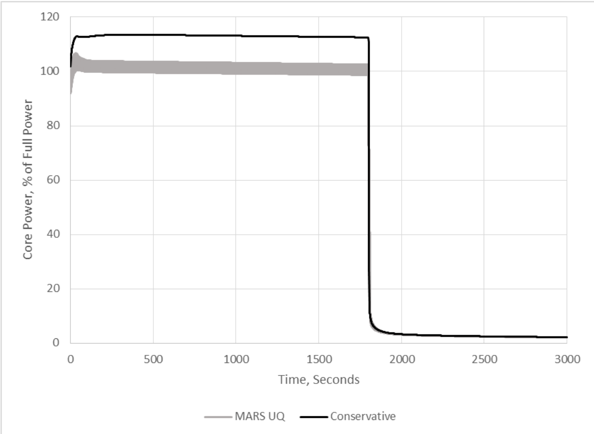

Figure 2. Core Power

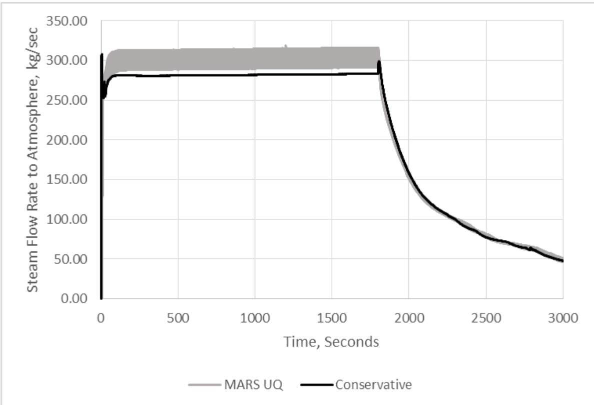

Figure 3. Steam Flow to the Atmosphere

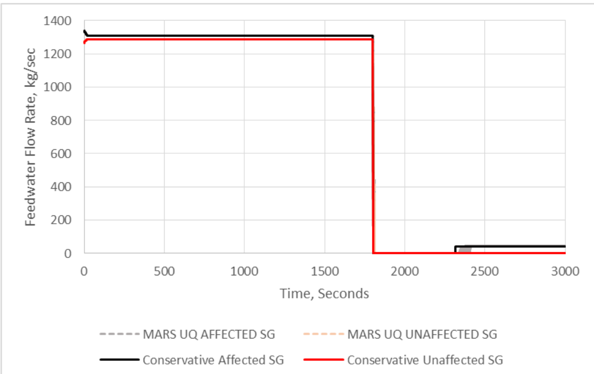

Figure 4. Feedwater Flow Rate

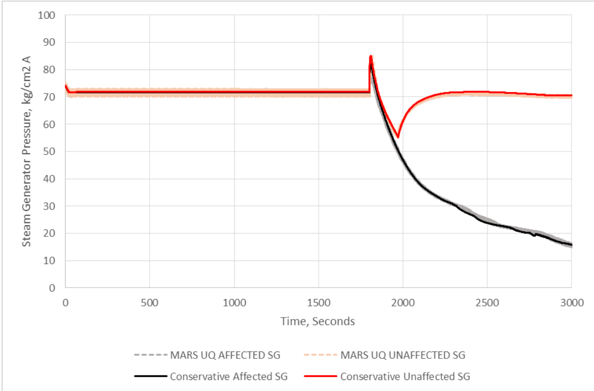

Figure 5. Steam Generator Pressure

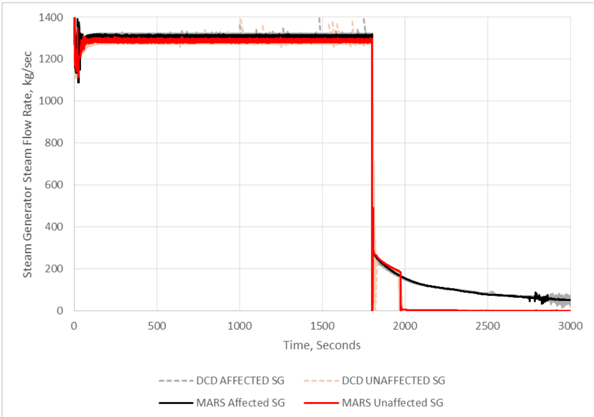

Figure 6. Steam Generator Flow Rate

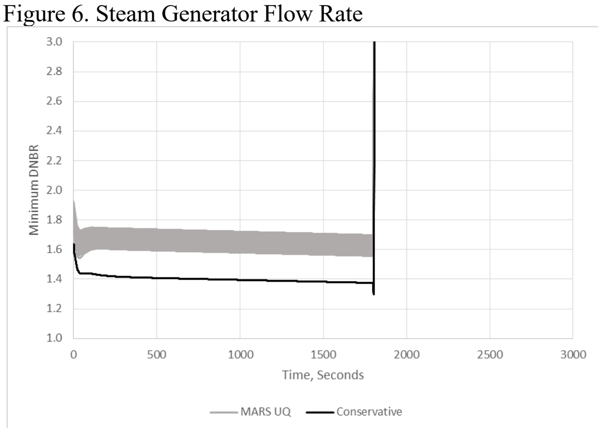

Figure 7. DNBR

The DNBR is calculated based on the W3 critical heat flux (CHF) correlation. For both average and hot channels, the axial power distribution follows the conservative scenario described in the DCD Chapter 15, where the axial offset of +0.3 is used for IOSGADV. The W3 is a widely used CHF correlation for PWR fuel, where DNB is the dominant CHF mechanism [5].

During the accident of IOSGADV, minimum DNBR happen after 1800 seconds, after the reactor trip. SG pressure reaches main steam isolation signal after 1956 seconds. Pressurizer pressure reaches safety injection actuation signal analysis setpoint after 1967 seconds and SIP is activated after another 40 seconds. AFW is activated after 2317 seconds. After 3000 seconds, operator manually closes ADV and in 3600 seconds plant cooldown.

4. Conclusion

In this paper, comprehensive analysis using point- and nodal-kinetics models of IOSGADV for APR1400 was presented.

The application of the models to a dataset of 181 samples gave results that were satisfactorily interpretable. However, it is necessary to recognize that the discrepancies observed between the DCD and the simulated outcomes are largely attributable to variations in input parameters that could not be determined with exact precision. This underscores the need for further investigation into these input parameters to enhance their accuracy and consistency. Refining these parameters is critical for improving the reliability and precision of the model's predictions. Future efforts should be focused on the simulation of RELAP5/3DKIN because the results from Multiphysics simulations will be able to predict the real behave of the APR1400.

The real operational model demonstrates a significantly larger safety margin compared to the conservative model, primarily due to the fact that it does not approach the critical DNBR values. Additionally, the behavior of the affected and unaffected steam generators varies distinctly. In the affected steam generator, the Auxiliary Feedwater (AFW) system is activated. The steam generator pressure on affected side is not maintained.

5. Acknowledgement

This research was supported by the 2024 Research Fund of the KEPCO International Nuclear Graduate School (KINGS), Republic of Korea.

I would like to also express special appreciation to Je Hee Lee Ph.D. for valuable discussions during my internship in FNC Technology. His insightful advice helped me to move forward with research presented in this paper.

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