Uncertainty Quantification for Station Blackout Scenario

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1. Introduction

After Fukushima Dai-ichi NPP accident, mitigating an extended station blackout (SBO) has become a priority. To enhance the coping capability of advanced nuclear reactors, the diverse and flexible coping strategies (FLEX) have been proposed. However, the successful implementation of these strategies demands quantification of the epistemic as well as aleatory uncertainties to ensure a more realistic analysis. In this work, a best estimate with uncertainty analysis (BEPU) is therefore applied to analyze a station blackout for APR1400 nuclear reactor and ensure the successful implementation of emergency operating procedures under SBO conditions. MARS-KS is used as best estimate thermal-hydraulic code loosely-coupled with Dakota software for uncertainty quantification (UQ). Given the high computational cost of Monte Carlo method, the Wilks formula is applied.

BEPU provides a more realistic safety margin and helps improve the emergency operating procedures to prevent progression into a severe accident. The assessment of these uncertainties consist of identification of relevant input parameters and the quantification of their influence on select output, as illustrated in Figure 1. [1] [2]

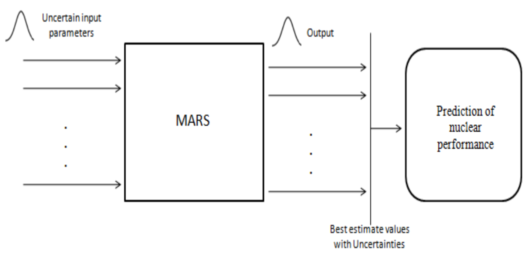

Figure 1 - Propagation of Uncertainties

This paper builds on the work of Kang et al. [3], where the coping strategies of APR1400 were examined considering RCP seal leakage under a station blackout condition. This work attempts to apply uncertainty analysis (UA) in a systematic way to ensure the reliability and robustness of emergency operating procedures under a SBO condition. Therefore, the main goal of this work is to identify the key parameters and quantify the associated uncertainties and using probabilistic methods for APR1400 under SBO accident.

In the next section, the methodology is presented followed by discussion of preliminary results. Subsequently, the last section highlights the key findings and derives the main conclusions.

2. Methods and Results

To meet the objectives of this work, the model of APR1400 with FLEX strategies and optimized operating procedures is developed using MARS-KS based on the work of et al. [3]. Next, the developed model is loosely-coupled with Dakota. Subsequently, a number of input parameters and models have been identified for the UA based on previous work published in the open literature. Finally, the uncertainty quantification and sensitivity analysis is executed.

2.1 Model description

Dakota is used as a platform to run the uncertainty analysis using the thermal hydraulic system code, MARS-KS, for UQ and sensitivity analysis. Dakota is a mathematical toolkit which contains optimization algorithms; uncertainty quantification with sampling, reliability, and stochastic expansion methods; parameter estimation; and sensitivity/variance analysis. It allows the UQ for a station blackout scenario by coupling with a suitable user-supplied simulation code [4], MARS-KS code, for the case at hand.

MARS (Multi-dimensional Analysis of Reactor Safety) is a code for the realistic multi-dimensional thermal-hydraulic system analysis of light water reactor transients already proven and validated. [5]

Figure 2 shows how MARS and Dakota are coupled. MARS-KS is treated as a blackbox and both codes exchange data by reading and writing several text files. A separate process external to Dakota is created to run MARS-KS iteratively.

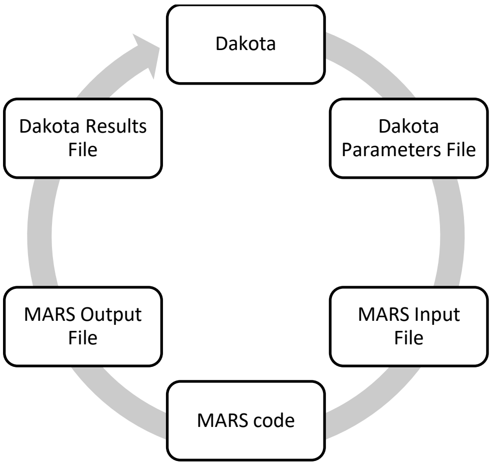

Figure 2 - The loosely-coupled interface between Dakota and MARS-KS code [4]

Fig. 3 shows the APR1400 nodalization used in this study. The model includes the reactor vessel with two hot legs and four cold legs, four reactor coolant pumps, two steam generators (SGs), a pressurizer, the surge line pipe, four main steam lines, one main feed water line from the downcomer of each SG, and one main feed water line from the economizer of each SG. [3]

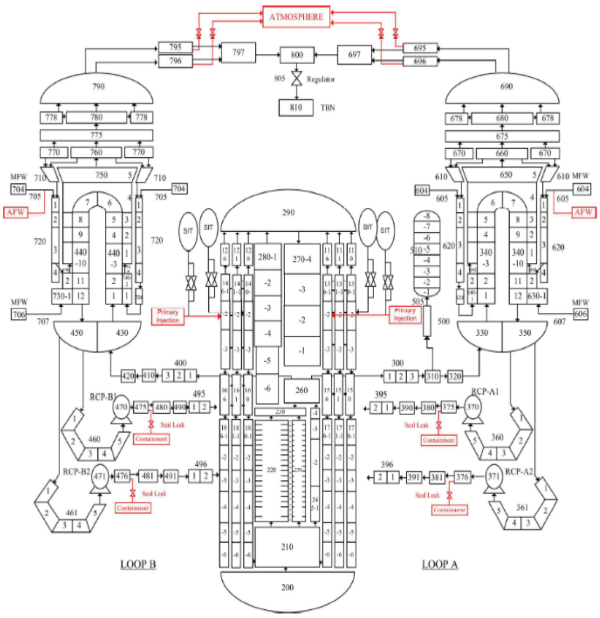

Figure 3 - MARS Nodalization Diagram of APR1400 for extended SBO Scenario [3]

The assumptions during the SBO accident are:

1. FLEX strategies are implemented
2. RCP seal leakage is considered
3. Battery power is guaranteed for 08 hours
4. Feed and bleed is performed on the secondary side
5. Two turbine driven auxiliary feedwater pumps are considered available under certain conditions during the accident
6. Safety injection pump is unavailable
7. Shutdown cooling pump is unavailable
8. Auxiliary charging pump is unavailable
9. Motor driven auxiliary feed water pump is unavailable
10. Operator action are necessary for controlling main steam atmospheric dump valves

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2.2. Uncertainty analysis

Uncertainty analysis consist of 3 processes. First, identifying input uncertainties. Second, propagating these uncertainties through a computational model. Third, performing statistical or interval assessments on the resulting responses. The UQ addresses the question of how safe and reliable is the model. This work uses probabilistic method of UA with input parameters described using probability distribution functions and ranges guided by previous research published in the open literature. [4]

Identification of uncertainty key parameters is an important step in the uncertainty analysis. The initial set of uncertain parameters contains 30 input parameters for example: gap conductance, fuel thermal conductivity; core power; decay heat; Dittus-Boelter liquid HTC; nucleate boiling HTC correlation; accumulator actuation pressure, temperature and inventory. Their respective probability distribution functions and ranges had been published by Kang et al. [3] and Kozmenkov et al. [6]. A sensitivity analysis using linear regression is conducted to isolate the key parameters that significantly affect the overall transient and some specific output variables such as the peak cladding temperature.

The second step is sampling and propagating those parameters into MARS model. Using Dakota both the Monte Carlo sampling method and Wilks's method can be used. However, due to the high computational cost associated with Monte Carlo method, the Wilks's formula with a 5^{th} order is applied to choose the minimum number of samples necessary to guarantee two-sided tolerance limits with 95% probability and 95% confidence as required by USNRC [7]. The advantage of Wilks' method is that the number of calculations is independent of the number of input parameters. Furthermore, the use of higher order Wilks's formula increases the reliability of the estimated safety margin Lee et al. [8]. Accordingly, 181 samples are needed for the desired confidence level.

The third step is assessment of the response. The platform allows the sensitivity analysis and uncertainty quantification. The sensitivity analysis allows the identification of the key input parameters with the most influence on the system response. This paper uses Spearman's rank-order correlation to measure the strength and direction (positive or negative) of association between the uncertain input parameters and output.. The UQ will assess the reliability of the emergency procedures according to the acceptance criteria on condition the model converges as expected. [1] [4]

2.3 Discussion and Results

This section shows the results for the base case of SBO without considering the uncertainties. The steady-state condition is the same adopted by Kang et al. [3]. Table I summarizes the timing of the main events of the developed MARS-KS model. Due to time constraints, the uncertainty analysis could not be completed at the time of preparation of this paper. Hence, identification of key parameters by linear regression, and uncertainty quantification will not be presented in this paper. The complete results of the undertaken uncertainty analysis will be presented at the KNS Conference.

Table I – Key SBO Events Timing

Time (hh:mm)	Scenarios
00:00	Initiating event – SBO
	Reactor trip
	TDAFW pump starts automatically based on SG wide range level with design flow rate (41kg/s)
00:03	RCP seal leakage (120 gpm/pump)
01:00	ADVs are opened 30% for cooldown
01:20	SITs begin injecting
02:00	AFW flow rate is reduced from 41 kg/s to 12kg/s per loop
06:00	AFW flow rate is reduced from 12 kg/s to 5 kg/s per loop
07:00	Core uncovery starts
08:00	Primary-side injection with 6kg/s to the DVI line
13:00	SITs are depleted
32:00	SG level is reached at upper limit (13.68 m)
	Secondary-side injection with 5 kg/s to the AFW line
72:00	Calculation end

7. Conclusion

This is work assesses the necessary condition for successful implementation of the copying strategies under SBO condition for APR1400 nuclear reactor. Thus, UA was performed by probabilistic method using Wilks' method, Spearman's Rank correlation coefficient, and analysis of output response with acceptance criteria to verify the reliability and robustness.

The future goal is to use the same framework in different accident scenarios and explore different uncertainty methods.

Acknowledgment

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