# SENSITIVITY AND UNCERTAINTY INFORMATION INCORPORATED LOSS OF FLOW ACCIDENT ANALYSES FOR RESEARCH REACTORS

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### ABSTRACT

This paper outlines a system level safety analysis procedure for research reactors incorporating sensitivity and uncertainty components. The protected loss of flow (LOF) accident was selected as an exemplified design basis accident to demonstrate the analysis procedure. The conceptual NIST (National Institute of Standards and Technology) horizontally split-core based research reactor was adopted as a research reactor model in the study. Two system level dynamics codes, RELAP5-3D and PARET, were employed in this work in a comparison study manner. The primary objective of the present work is to demonstrate the analysis capability of integrating sensitivity and uncertainty information in addition to traditional predictions of the system code models for the study of the thermal-hydraulics (T/H) safety characteristics of research reactors under accidental transient scenarios. The canonical transient predictions on the LOF accident yielded from the two system codes mentioned above have demonstrated some noticeable yet acceptable discrepancies. To better understand the discrepancies observed in the simulations, sensitivity and uncertainty analyses were performed by coupling the RELAP5-3D model and the data analytic engines provided by the RAVEN framework developed by INL. The sensitivity information reveals the significances of key figure of merits such as the peak cladding temperature varies with different boundary and initial parameters in both normal operation and design basis transients. The uncertainty analysis informs the deviations of the responses contributed by the errors of various input components. Both the sensitivity and uncertainty information will be incorporated into a safety analysis framework as part of the safety characteristic predictions delivered by the framework.

Keywords: Research Reactor, Sensitivity and Uncertainty Analysis, RELAP5-3D, RAVEN. Accident Analyses

### 1. INTRODUCTION

Nowadays research reactors are widely used in the world as important research or production facilities. In fact, there are more than 250 research reactors with various purposes are currently running in the world. Based on their utilizations, research reactors can generally be classified into three types: material test reactors, isotope production reactors, and beam tube reactors [1]. Therefore, rather than for supplying power, the main function of a research reactor is to deliver neutrons for testing materials, producing radioisotopes, or assisting neutron scattering experiments.

The safety analysis for the research reactor is paramount important as that for the power reactor. The designed basis protected loss of flow (LOF) accident, for example, examine the thermal-hydraulic safety characteristics of the research reactor under the situation of pumps unplanned coastdown due to the loss of onsite power, which could possibly take place at any reactor site. In this study, the LOF accident for the conceptual NIST (National Institute of Standards and Technology) horizontally split-core based research reactor [2] was comparatively studied with the multi-channel T/H safety analysis code PARET [3] and the system level code RELAP5-3D [4]. The power profiles and kinetics parameters used in transient analysis models were provided by neutronics calculations [2]. The steady state and the transient behavior under the reactivity insertion accident for the conceptual reactor have been performed before [5]. The results from these studies verifies the RELAP5-3D outcomes have a good agreement with the ones from the PARET code, which verifies the feasibility of the current model in a certain degree.

The primary design goal of this conceptual NIST research reactor is to produce high-quality neutron sources for scientific experiments [2]. The reactor concept considers 20MW thermal power and a 30-day operating cycle. A plate-type fuel element with low enriched uranium (LEU) - U3Si2-Al - was used in the new design. The reactor core is cooled by a forced downward circulation of light water and surrounded by heavy water in a cylindrical tank. The reflector tank is about 2.5 m in diameter and 2.5 m in height and placed in the center of a larger light water pool that serves as thermal and biological shields. A more detailed description of the core and reactor configurations of the new NIST reactor design can be found in Ref. 2.

The canonical transient predictions on the LOF accident yielded from the two system codes mentioned above have demonstrated some noticeable yet acceptable discrepancies. To better assess the transient modeling capability of the developed model and comprehensively understand the discrepancies observed in the simulations, sensitivity analysis and uncertainty quantification were performed in this paper by coupling the RELAP5-3D model and RAVEN [6]. RAVEN is a flexible and multi-purpose data analysis framework recently developed by INL. RAVEN can be used for uncertainty quantification, regression analysis, probabilistic risk assessment, model optimization, and so on. Depending on the tasks to be accomplished and on the probabilistic characterization of the problem, RAVEN is capable of capturing the uncertainties of the response of the system under consideration by stochastically sampling its own parameters. The system modeling software such as RELAP5-3D is accessible to RAVEN either directly (software coupling) or indirectly (message coupling). The data generated by the sampling process is analyzed using classical statistical and more advanced data mining approaches. Because of these salient features, RAVEN is an ideal tool to be used in this work to quantify the uncertainties during the LOF transients.

The reminder of the paper is organized as following. The computational methods of this work are described in Section 2, Section 3 discusses the simulation results of protected LOF accidents without uncertainty considered. Section 4 presents the sensitivity and uncertainty analysis of LOF accident, followed by the results in Section 5. The conclusions and future works are provided at the end of the paper.

### 2. COMPUTATIONAL METHODS

Reactor system level safety analysis codes (e.g., RELAP5-3D and PARET) were employed as standard mechanistic tools in this work. The PARET code is a T/H analyses tool developed by Argonne National Laboratory (ANL) for plate-type research reactor safety analyses. It consists of a one-dimensional (1-D) T/H model and a point-kinetics model to couple the neutronics and thermal hydrodynamics effects on reactor behavior during normal and off-normal conditions. However, PARET is merely a channel analysis code and unable to model complete cooling loops in the reactor, so the research efforts have been extended to more sophisticated code RELAP5-3D.

The computational modeling and simulation results obtained from the system codes will serve for two basic objectives in this research. First, the RELAP5-3D simulations with nominal input parameters provide reference solutions to the hypothetical accident transients under investigated. The RELAP5-3D model also functions as a working forward model for the sensitivity and uncertainty analysis. Second, the anticipated discrepancies existed between the predictions of RELAP5-3D and PARET stand as one typical epistemic uncertainty source due to the model differences. The uncertainty assessment procedure performed in the second stage of the paper will attempt to quantify these uncertainties and provide the best estimates.

Fig. 1 illustrates the nodalization of the reactor core model developed in RELAP5-3D. A similar multi-channel model was established in PARET as well. As shown in Fig. 1, the thermalhydraulic system of the reactor core is represented by one hot channel (No.100), one average channel (No.110) and one bypass channel (No.120) using the built-in PIPE component in the RELAP5-3D code. The hot channel describes the flow channel with the hottest power peaking factor in the fuel assembly, the remaining flow channels are lumped to one average channel. The bypass channel is developed to consider the side flow that is stuck in the area between fuel assemblies. All three channels are divided into 17 control volumes along the flow direction. The upper plenum (No.130) and bottom plenum (No.160) are modeled to connect and mix the flow at the entrance and exit point of the flow channels. The inlet condition (flow source) was provided using a time-dependent control volume (No.140) and its corresponding time-dependent junction. Similarly, the outlet condition (flow sink) is defined by a single control volume (No.180) and the corresponding single junction. No.170 represents the reactor pool. The primary coolant loop has not yet been fully completed at this moment. Therefore, the core channel model is bounded with inlet and outlet components, which were established with time-dependent control volumes and junctions to provide needed boundary conditions during the transients. Proper boundary conditions were provided with the ones consistent with the PARET model. The heat structure components were developed to accommodate the proper heat power profiles of the core, which were calculated and transferred from the neutronics models [2].



**FIGURE 1:** NODALIZATION OF THE REACTOR CORE IN RELAP5-3D MODEL

The conceptual NIST research reactor is designed to operate with forced flow heat convection mode in normal operating conditions, while natural heat convection mode is enabled during the design basis accident situations. After the reactor is shut down because of such an accident such as the LOF scenario, the decay heat will be eventually removed out of the core by a natural heat convection of a reversed coolant flow (the initial coolant is flowing in a downward direction) controlled by buoyancy and gravity forces due to flow density differences. Both the PARET and RELAP5-3D codes are able to model the decay heat removal by the reverse flows, but the mechanism to establish the natural circulation in PARET and RELAP5-3D has shown some differences.

In the PARET model, there is no primary loop simulation. To enable the natural circulation, the time of flow reversal is detected and saved for each channel. After the flow reversal, the channel exit becomes the inlet. The exit coolant mixes with the pool water (possibly in a plenum) and is cooled. After some time, the reversed flow will draw relatively unheated coolant from the pool. The model assumed that the enthalpy of that coolant well after flow reversal was the same as that of the coolant inlet enthalpy at the start of the calculation.

In the RELAP5-3D model, the reactor was modeled as a pool-type system. The reactor core is immersed in a light water pool, and a closed flow path was established through the pool acting as a primary loop during the natural circulation. A natural circulation valve (NCV) is modeled as a trip valve component and will be activated when the flow is reduced to 10% of its nominal flow, which leads to flow reversal and eventually establish the natural circulation of the coolant in the core to remove the decay heat after reactor shutdown. After the natural circulation established, the core decay heat is removed by the reverse flow loop between a water pool and the reactor core.

Sensitivity analysis and uncertainty quantification were performed with the intention to address the model differences and to better understand the discrepancies observed in the simulations of these two models. Be more specific, the purpose of sensitivity analysis is to determine the contribution of the uncertainty of input variables to the model result. While the uncertainty analysis aims at quantifying the deviations of the responses contributed by the errors of various input variables.

To perform sensitivity and uncertainty analyses (SUA), figure of merit (FOM) that are known of critical importance in analyzing the transient behaviors in the design basis accident need to be appropriately selected. Key input variables that are expected to mostly influence the FOM are also need to be identified in the SUA. The sensitivity coefficient of the response parameters with respect to the input variables were calculated using the central difference method.

The uncertainty quantification process in this work were carried out with the thermal-hydraulic system code RELAP5-3D coupled to the data analysis code RAVEN. These two codes were connected to each other through the interchange of physical parameters. The role of RAVEN is act as a simulation controller of the RELAP5-3D model, by using monitored variables and controlled parameters. In general, the RELAP5 reference model is developed with a list of input variables and their uncertainties defined, then RAVEN randomly samples each variable and creates multiple RELAP5-3D input files and executes each input separately. The interplay mechanism between these two codes can be pictorially illustrated by Fig. 2.



FIGURE 2: INTERACTION BETWEEN RAVEN AND RELAP5-3D

### 3. PROTECTED LOSS OF FLOW ACCIDENT

The protected LOF accident is used as a representative accidental transient to demonstrate the method and theoretical framework described above. Here protected means the reactor SCRAM is tripped during the accident. To mimic the LOF accident, the flow rate reduction caused by the pump coastdown is assumed to follow an exponential function  $\exp(-t/\tau)$ , where  $\tau$ is considered as the time constant of the flow rate decay. In this study, the time constant  $\tau$  is set to be 1 s to mimic the fast loss of flow (FLOF) accident. During the LOF transients, the reactor SCRAM is tripped by a low coolant flow signal when the coolant flow reaches 85% of its nominal operation value. The safety control rods react to the trip signal with a time delay of 0.2 s. This short delay is considered to account for the reaction time needed by mechanical and electronic circuit operations. All reactivity feedback effects and period trips are neglected in the analyses by this moment. The core status is considered at the state of the end of the cycle (EOC).

Fig. 3 shows the transient behaviors of the power and mass flow rate of the hot channel during the LOF accident. The results generated by PARET and RELAP5-3D models are presented in a comparison manner. The LOF accident takes place at the initial time (t = 0). Reactor trip occurs at ~0.5 s in both codes when the flow rate decrease to 85% of its nominal value. As shown in Fig. 2(A), the power rate reduces to the decay heat level nearly immediately after the reactor shutdown. In the RELAP5-3D model, the NCV opens at 2.5 s when flow reached 10% of its nominal flow, then the buoyancy force starts to drive the natural circulation of the flow between the core and the pool. As shown in Fig. 2(b), the reversal flow was established after ~5 s and reaches a stable level of ~0.047kg/s in the hot channel. Compared to the PARET results, it shows the flow reversal was predicted slightly earlier in the PARET code.



FIGURE 3: COMPARISON OF POWER (A) AND MASS FLOW RATE (B) IN THE LOF ACCIDENT.

Fig. 4 shows the transient behavior of peak cladding and peak coolant temperatures during the LOF accident in the hot channel from both codes. As shown in the figure, the temperatures of cladding and coolant initially increase steeply because of the rapid LOF at the early stage of the accident. The temperatures reaches their first peak values in a fractional section. After the reactor trip at  $\sim 0.5$  s, the temperatures start to decrease sharply and arrive at minimum values shortly after 1 to 2 s into the accident. Then the temperatures of both cladding and coolant start to increase again due to the reduction of heat removal and the accumulation of decay heat in the core. The second temperature peaks are observed for both cladding and coolant after the flow reversal occurs and natural circulation is established.



FIGURE 4: COMPARISON OF CLADDING (A) AND COOLANT (B) TEMPERATURE IN THE LOF ACCIDENT

TABLE	1.	PEAK	TEMPE	RATURES	AND	CORRESPO	ONDING
OCCURR	INC	5 TIMES	5 IN THE	LOF ACC	IDENT		

Properties	R5-3D	PARET	Deviation
1st PCT1 [°C]	100.25	104.57	4.13%
1st PCT time [s]	0.50	0.40	25.00%
1st PCoT2 [°C]	59.47	59.72	0.42%
1st PCoT time [s]	0.50	0.40	25.00%
2nd PCT [°C]	123.81	128.67	3.78%
2nd PCT time [s]	7.5	8.00	6.25%
2nd PCoT [°C]	108.77	106.76	1.88%
2nd PCoT time [s]	8.00	8.00	0.00%

PCT = Peak cladding temperature

2PCoT = Peak coolant temperature

Table 1 provides a quantitative summary of the first and second peak cladding temperature (PCT), peak coolant temperature (PCoT) and the corresponding time of occurrence during the LOF Accident. The maximum cladding temperature and coolant temperature during the transient predicted by RELAP5-3D are 123.81 °C and 108.77 °C, respectively. The PARET predictions are 128.67 °C and 106.76 °C, respectively. After the stable establishment of natural circulation, the temperature is shown to be decreasing mildly.

As also can be seen in Fig. 4, the transient temperature behaviors predicted by both codes have shown some differences, particularly the cladding temperature in the period before and after the stabilized natural circulation. The temperature in PARET decreases sharply after flow reversal, while RELAP5-3D predicts the temperature decreases smoothly. This is possibly attributed to the difference existed in the natural circulation modeling mechanisms of both codes as described earlier. RELAP5-3D take the momentum change effect owing to the open of NCV into account, which is hard to be thoughtfully considered in PARET because of lacking of the loop modeling.

### 4. SENSITIVITY AND UNCERTAINTY ANALYSES

In this work, the PCT and PCoT are the two response parameters chosen as the FOM for SUA for demonstration purposes. Both of the two parameters are known of certain importance in evaluating the reactor core safety in the LOF accident.

Key input variables that are expected to mostly influence the FOM are also need to be identified in the SUA. Model input variables with uncertainties are one of the major sources of model uncertainties that were investigated and prioritized in this work. These variables include the initial inlet coolant temperature, the inlet coolant mass flow rate and the reactor core power. The selected key input variables and the possible range of uncertainties associated with these variables are summarized in Table 2.

Uncertain parameter	Nominal value	Uncertainty range	Distribution
Inlet coolant Temp. [°C]	37	±10%	Normal
Inlet coolant mass flow rate [kg/s]	516.83	±10%	Normal
Reactor core power [MW]	20	±10%	Normal

#### **TABLE 2.** UNCERTAIN PARAMETERS

The input variables shown in Table 2 were randomly sampled using a Monte-Carlo method approach in RAVEN. The probability distribution functions (PDF) of the variables were all assumed as normal distributions and their uncertainties correspond to two standard deviations (i.e.,  $2\sigma$ ) based on engineering judgment.

Using the sets of input variables generated by the sampling procedure, RAVEN creates multiple RELAP5-3D input files and executes each input separately. The FOM response parameters contained in the output data were extracted and post-processed for the SUA. In this study, we performed 1000 calculations for both the steady state and LOF transient calculations in a random set to obtain a 95%/95% confidence level.

### 5. RESULTS AND DISCUSSION

## 5.1 Sensitivity Analysis of Input Variables

In order to assess the sensitivities of the selected input variables to the selected FOM response parameters, the sensitivity coefficient of the response parameters with respect to the input variables were calculated using the central difference method as follows

$$\alpha = \frac{x_0}{R_0} \frac{\partial R}{\partial x} \Box \frac{x_0}{R_0} \frac{R(x+h) - R(x-h)}{2h}, \qquad (1)$$

where R stands for a general response parameter, x represents one input variable, and h is the perturbation size to the input variable. The symbols attached with the 0 subscript are the nominal values of the quantity.

Since Eq.(1) is a numerical approximation to the sensitivity coefficient, truncation errors are inevitably introduced by this method. To understand and ensure the accuracy of the calculations based on Eq.(1), the sensitivities were evaluated with different perturbation sizes in this study. Fig. 5 depicts the sensitivity coefficient of the PCT with respect to the three input variables listed in Table 2 with different perturbations. The various perturbations were made in a range of -10% to +10% of the nominal values. All the sensitivity calculations in this paper are performed only at the steady state condition. The sensitivities for the transient situation are deferred for future investigation at this point.



**FIGURE 5:** SENSITIVITY COEFFICIENTS OF PCT AT DIFFERENT ERTURBATIONS OF INPUT VARIABLES

As shown in Fig. 5, the sensitivity coefficients of PCT to the mass flow rate and reactor power have shown a nearly linear variation characteristic, while the sensitivity to the inlet temperature has exhibits some non-linear feature, which indicates the higher-order sensitivity information may be required to fully capture its characteristics.

Fig. 6 shows the rank of the sensitivity of the selected three input variables to the PCT under steady state. The sensitivity coefficient for inlet coolant temperature, reactor power, and inlet coolant mass flow rate are 0.572, 0.133 and -0.112, respectively. It could be found that the inlet coolant temperature is the dominant factor that affects the PCT, while the impact of the other two parameters are relatively small.



**FIGURE 6:** SENSITIVITY RANKS OF PCT TO THE SELECTED THREE INPUT VARIABLES

Similarly, the sensitivity coefficients of PCoT with respect to the three input variables with various range of perturbations were shown in Figure 7.



**FIGURE 7:** SENSITIVITY COEFFICIENTS OF PCOT AT DIFFERENT ERTURBATIONS OF INPUT VARIABLES

Fig. 8 shows the rank of the sensitivity of the selected three input variables to the PCoT under steady state. The sensitivity coefficient for inlet coolant temperature, reactor power, and inlet coolant mass flow rate are 0.857, 0.03 and -0.03, respectively. It could be found that the inlet coolant temperature is again the dominant factor that affects PCoT, while the impact of the other two parameters are relatively small.



Input parameters

**FIGURE 8:** SENSITIVITY RANKS OF PCOT TO THE SELECTED THREE INPUT VARIABLES

### 5.2 Uncertainty Analysis for Steady State

The PCoT and PCT generated from the 1000 outputs are shown in a form of histogram view in Fig. 9. The fitted Gaussian distribution functions are also highlighted with a red line in the figure.



**FIGURE 9:** HISTOGRAM VIEW OF PCOT (A) AND PCT (B) IN THE STEADY STATE

Table 3 summarizes the peak temperature distribution statistics. It can be seen that the small discrepancy between

RELAP5-3D and PARET are within 95%/95% confidence levels (C.L.). For reactor designs, it is required the PCT must not reach the fuel blister temperature, which is taken as 515 °C to 575 °C for silicide LEU fuel [7]. Apparently, the maximum value of peak coolant and cladding temperature are within the safety limit, which indicates the design has sufficient safety margin in steady state.

<b>TABLE 3.</b> PEAK TEMPERATURE DISTRIBUTION STATIST	ICS
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	PCoT [ °C]	PCT [°C]
Mean	54.25	90.17
Standard Dev.	14.88	10.67
95% Lower C.L.	53.32	89.51
95% Upper C.L.	55.18	90.84
Maximum	97.74	122.83

#### 5.3 Uncertainty Analysis for Loss of Flow Accident

For the LOF transient case, 1000 sets of selected three input variables were sampled and fed into RELAP5-3D by RAVEN. The baseline and perturbed peak cladding and coolant temperature evolution are shown in Fig. 10. The perturbed results are presented with a shadowing area surrounding the line plots, which are the baseline results.

It can be seen from Fig. 10 that the PCT in PARET is mostly within the range of the uncertainties generated by the perturbed calculations before the period of the stabilized natural circulation. It is also noticed the discrepancy of the predictions in both codes after the period of the flow reversal cannot be well captured by the uncertainties in this study. The temperature in PARET decreases sharply after flow reversal, while RELAP5-3D predicts the temperature decreases smoothly, which may be caused by other uncertainty resources such as the model difference of natural circulation models in both codes.

### 6. CONCLUSION

This paper presents a sensitivity and uncertainty information incorporated safety analysis for research reactors under the framework of RELAP5-3D and RAVEN. Two system level transient analysis codes, RELAP5-3D and PARET, were employed in the study. The results of RELAP5-3D provide the reference and perturbed solutions for sensitivity and uncertainty analysis. The results of PARET provide prediction discrepancies to the reference solutions and create an uncertainty source due to model incompleteness to enable the investigation. A design basis protected LOF accident is modeled and used as a representative transient accident for this work. FOM response parameters and key input variables are identified for the sensitivity and uncertain analyses. The sensitivity coefficients obtained from the sensitivity analysis procedure provides insights of different level influence impacts of different input variables on the responses. The uncertainty analysis informs the deviations of the responses contributed by the errors of various input components. The research approach and findings presented in this paper provide

an efficient means for the reactor safety analysis under the larger framework of best estimation plus uncertainty (BEPU) reactor design philosophy.



**FIGURE 10:** BASELINE AND PERTURBED CLADDING TEMPERATURE (A) AND COOLANT TEMPERATURE (B) IN THE LOF ACCIDENT

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