# Uncertainty Quantifications for the MIT Reactor Thermal-Hydraulic Analysis

Yu-Jou Wang, Kaichao Sun, Lin-wen Hu*

Massachusetts Institute of Technology, Cambridge, MA 02139-4307, USA
* lwhu@mit.edu

*The objective of this study is to demonstrate a new methodology to evaluate the impact of measurement, design, and fabrication uncertainties of the MIT Research Reactor (MITR) on thermal-hydraulic licensing analysis. The best-estimate plus uncertainty propagation technique is adopted, with the use of coupled reactor system code RELAP5-Mod3.3 and statistical toolkit DAKOTA. The methodology is verified through comparing the HEU statistical results with the conventional engineering hot channel factors approach. The result shows that this method can eliminate the unnecessary conservatism with sufficient confidence levels. The approach is further applied on the thermal-hydraulic analysis of LEU-based core design, for both steady-state and loss-of-flow transient analysis. It is expected that the quantified engineering uncertainties can serve as a guide to the LEU conversion evaluation.*

## INTRODUCTION

To reduce the risk of nuclear weapon proliferation, converting the high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel has drawn an increased attention around the world. In this context, efforts have been made to design and evaluate the MITR LEU fuel element for safely replacing the current HEU fuel

Thermal-hydraulic limits are established to provide sufficient safety margins for nuclear reactors[1]. Deviations from nominal engineering specifications, such as measurement errors and fabrication tolerance, could notably affect thermal-hydraulic system behavior and hence pose threats in reactor safety. Conventionally, engineering hot channel factors (ECHFs) is adopt to consider uncertainties of different thermal-hydraulic system parameters. This method is typically based on single channel analysis and may lead to overly conservative results, since the culminated effects are derived from a number of so-called sub-factors. These are coefficients representing the uncertainty levels of each key thermal-hydraulic system parameter and are usually based on the semi-statistical vertical approach[2].

In the past decades, thermal-hydraulic analysis with best-estimate models receives increased interest for uncertainty quantification problems[3]. The uncertainties of engineering specifications are generally implemented with the system code calculation input parameters. This approach can evaluate the propagation of parametric uncertainties with specified probability distribution of the input parameters.

The present study aims at quantifying the uncertainties of engineering specifications for the MITR thermal-hydraulic analysis using best-estimate model plus uncertainty propagation techniques. The methodology is first verified through comparing the statistical results with the conventional EHCFs method of the present HEU-based core. The thermal-hydraulic limits of the new MIT LEU core would then be investigated and quantified systematically with this approach.

## MIT REACTOR

The MITR is a research reactor designed primarily for experiments using neutron beams and in-core irradiation facilities. It is moderated and cooled by light-water and has a heavy-water reflector. The thermal power is 6 MW. It uses rhomboid-shaped fuel elements. There are 27 in-core positions for fuel elements and/or irradiation experiments. These positions are divided into three concentric rings with 3, 9 and 15 rhomboid-shaped areas, respectively. The edge-to-edge distances of the three concentric hexagons of the fuel region are 12.4 cm, 25.5 cm, and 38.4 cm. Each fuel element contains 15 fuel plates, which consist of ~ 93% enriched uranium sandwiched between sides of aluminum cladding. The surrounding core tank has a cylindrical shape, 52 cm in diameter and 73 cm in height. There is a heavy water reflector surrounding it from the sides and the bottom. In addition, a three-meter high light-water plenum, which is capable of effective neutron shielding, sits above the core tank.

The MITR LEU-based core design is proposed by the RERTR program[4]. To maintain the neutron performance, the nominal power is increased from 6MW to 7MW considering the expected neutron loss of the $U^{238}$ absorption. Since the upgraded power might pose threats to the power peaking in the outer concentric ring due to the extra moderation, the fuel meat thickness is decreased for the first three plates from each end plate by 45%, 30%, and 30%, respectively. The current fuel adopts un-finned design, which supports the manufacturing feasibility and lowers the reactivity/power density. The coolant flow rate is increased from 2000 gpm to 2400 gpm to increase the heat removal capability.

## METHODOLOGY

### <u>*Best-estimate model of the MITR HEU and LEU core*</u>

The best-estimate analysis is performed with RELAP5-Mod3.3 code. The MITR core model is developed by MIT Nuclear Reactor Laboratory[5, 6]. The HEU-based core consists of one hot channel and one lumped average coolant channel, representing the rest 329 coolant channels. The LEU-based core model is similar to that of the HEU-based core. Since the fuel meat thickness is not uniform in the core, the LEU core is modeled into five types of channels: (1) inner channels (2) mid1 channels (3) mid2 channels (4) mid3 channels (5) end channels. Each type of channels is further divided into one lumped-average channel and one hot channel.

Hot channel is a theoretical channel assuming that the most critical case occurs and thus the thermal-hydraulic analysis focuses on it. The limiting safety system settings (LSSS) aims to prevent onset of nuclear boiling (ONB) during steady-state operation. The cladding softening temperature is adopted as the safety limit under transient conditions.

*Uncertainty Quantifications*

The initial step is to identify the key parameters that have high importance for the thermal-hydraulic limits. The following seven input parameters, suggested in the MITR SAR, are currently considered: (1) reactor power measurement, (2) power density measurement/ calculation, (3) plenum chamber flow, (4) flow measurement, (5) flow area tolerance, (6) heat transfer coefficient, and (7) eccentricity. It is assumed that they follow the normal distribution and their uncertainties correspond to three standard deviations based on the given sub-factors in the SAR.

To perform parametric uncertainty propagation, the RELAP5 model is coupled with Sandia National Laboratory code DAKOTA, a toolkit for system analysis and optimization. The latter code generates multiple combinations of input parameters, i.e. the random sample sets, based on Latin Hypercube sampling (LHS) method. To ensure the reliability of the statistic results, the present study performs 459 calculations in a random sample set to obtain a 99% / 99% confidence level based on Wilk's formula[7]. Figure 1 summarizes the scheme of the uncertainty quantification.

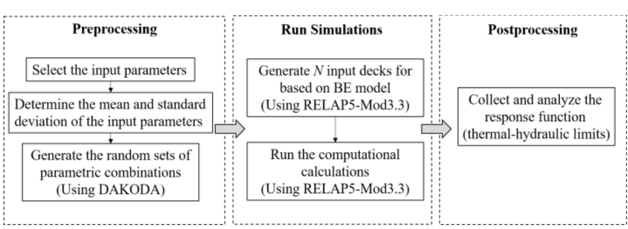


Fig. 1. The uncertainty quantification scheme.

**RESULTS**

*HEU uncertainty analysis for steady-state operation*

The statistical results based on the best-estimate model plus uncertainty propagation technique are shown in Figures 2. It can be concluded that, with 99% / 99% confidence levels, the thermal-hydraulic limits would not exceed the value of $T_{c,max} \leq 81.49$ ℃ , $T_{w,max} \leq 100.16$ ℃ , and $q_{max}" \leq 581.96\ kW/m^2$ , where $T_{c,max}$ , $T_{w,max}$ and $q_{max}"$ represent maximum coolant temperature, maximum cladding temperature and maximum heat flux. Comparing with the thermal-hydraulic limits obtained from the EHCFs method, the uncertainty analysis is shown to capable of eliminating certain over conservatism, while ensuring adequate confidence for the MITR operation safety. In particular, the cladding temperature is considered most critical to the steady-state LSSS, i.e., ONB occurrence. A significant margin (about 2 °C) is observed by replacing the EHCF method with the newly implemented uncertain propagation techniques.

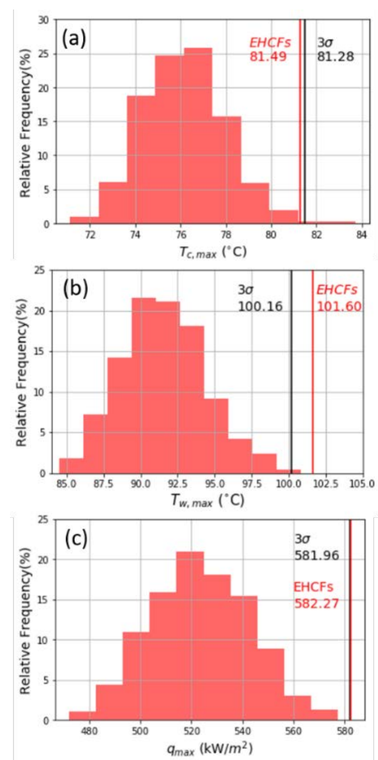


Fig. 2. Histogram of maximum (a) coolant temperature (b) cladding temperature (c) heat flux in the HEU-based core design under steady-state operation.

*HEU uncertainty analysis for loss-of-flow transient*

During loss-of-flow (LOF) transient, the temperature evolutions is shown in Figure 3. Again, a total of 459 calculations are performed to ensure a 99% / 99%

confidence level. The statistical results show that, during LOF, the maximum cladding temperature will not exceed 119.8 ℃, which is well below the aluminum based clad softening temperature of 450 °C. The band width of the cladding temperature evolution after the pump coastdown is found to be 6~7 ℃.

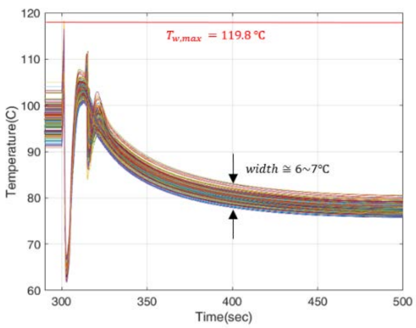


Fig. 3. Cladding temperature evolution during LOF in the HEU-based core.

*LEU uncertainty analysis for steady-state operation*

The uncertainty propagation technique is further applied on the LEU-based core design, as shown in Figure 4. It can be concluded that the, with 99% / 99% confidence levels, the thermal-hydraulic limits would not exceed the value of $T_{c,max} \leq 56.92$ ℃ , $T_{w,max} \leq 99.75$ ℃ , $q_{max}" \leq 701.26\ kW/m^2$ . The statistical results indicate that, with 99%/99% confidence, the design has sufficient safety margin during nominal operation.

*LEU uncertainty analysis for loss-of-flow transient*

The temperature evolution of LEU-based core during LOF transient is shown in Figure 5. It is found that, with 99%/99% confidence level, the maximum cladding temperature would not exceed 115.29 ℃, which is far below the failure limit, the blistering temperature, 365 ℃, of the selected material. Again, the thermal-hydraulic analysis shows that the proposed LEU fuel design would remain integrate during the LOF transient. In addition, the band width of the cladding temperature evolution is quantified within the range of 9~10 ℃, which is slightly higher than that in the HEU-based core design.

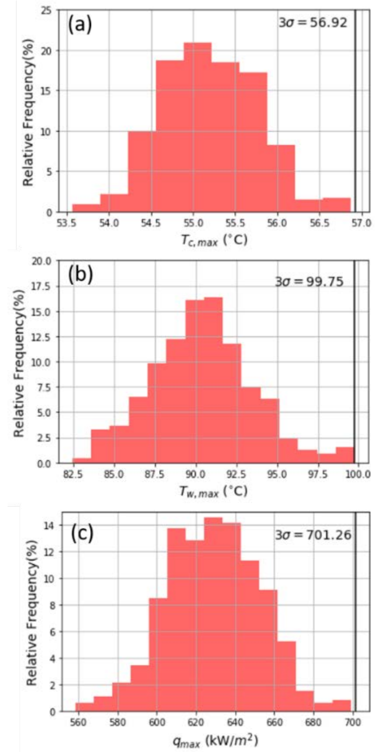


Fig. 4. Histogram of maximum (a) coolant temperature (b) cladding temperature (c) heat flux in the LEU-based core design under steady-state operation.

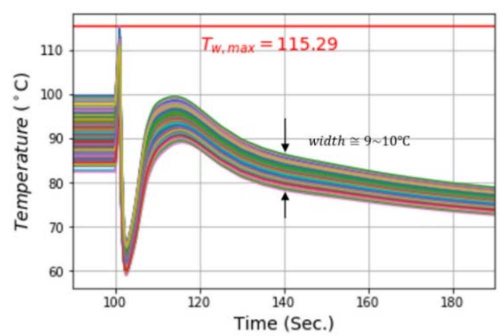


Fig. 5. Cladding temperature evolution during LOF in the LEU-based core.

*Sensitivity analysis of input parameters*

The rank values based on Pearson Product Moment Correlation Coefficient are also investigated in the present study. The rank values could be seen as indications which suggest the sensitivity of input parameters to output functions. If the rank value of an input parameter approaches +1.0, it suggests a strong correlation; whereas -1.0 indicates a strong reverse relationship. A close-to-zero value implies weak bonding between input parameter and output function.

Figure 6(a) and 6(b) shows the sensitivity ranks of the input parameters to the maximum cladding temperature under steady-state and LOF conditions in the HEU based core. It can be seen that reactor power, power density, and heat transfer coefficient are the dominant factors that are relatively impactful for cladding temperature under steady-state operation. The effects are nearly identical in the LOF transient condition, but it can be seen that, during the transient, the rank values indicates generally weak correspondence between input parameters and out function. Such a fact further implies the combining effects on the peak cladding during HEU LOF transient.

The rank values of input parameters to the maximum cladding temperature during steady-state operation and LOF transient in the LEU-based core is shown in Figure 6(c) and 6(d). It can be seen that in the LEU-based core design, the heat transfer presents the most dominant effect; the factors of reactor power and power density also strongly correlated with the maximum cladding temperature. Other factors have nearly no significance to it. To be noted, the effect of heat transfer coefficient still prominent in the LOF transient, which is not the case in the HEU-LOF transient. The result indicates that the present LEU design is more inclined to be affected by certain parameters, rather than combining effects.

## CONCLUSIONS

A best-estimate reactor system model with uncertainty propagation techniques is developed for quantifying the uncertainties of the MITR thermal hydraulic analysis. The corresponding computational tool is the coupled RELAP5-Mod3.3 and DAKOTA.

Analyses were performed for both steady-state operation and loss-of-flow transient conditions with 99%/99% confidence levels. The methodology is first verified through comparing the statistical results with the conventional engineering hot channel approach. Result shows that the best-estimate plus uncertainty propagation approach is capable of reducing over-conservatism while ensuring reactor safety. The method is further applied on the evaluation of the new MIT LEU-based core design, and the thermal-hydraulic limits could be quantified in both steady-state and transient conditions.

The sensitivity tests of the considered input parameters are also investigated. The results show that reactor power, power density and heat transfer coefficient are the dominant factors that influence the maximum cladding temperature. In HEU-based core, it is found that the maximum cladding temperature is more subject to the combining effect of multiple factors, which is even more pronounced in the LOF transient. On the other hand, the sensitivity study shows that the LEU-based core design is prone to be affected by the factor of reactor power, power density and heat transfer coefficient, rather than combining effects.

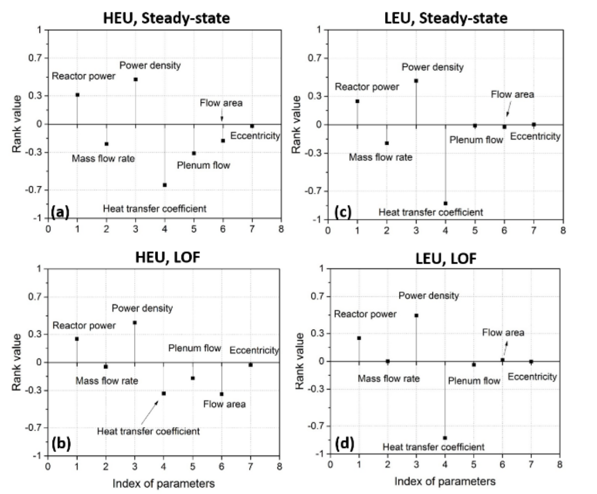


Fig. 6. Rank values of sensitivity study in the case of (a) HEU steady-state (b) HEU LOF (c) LEU steady-state (d) LEU LOF.

## REFERENCES

1. *NUREG-1537-Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*. 1996: US Nuclear Regulatory Commission.
2. N.E. Todreas and M.S. Kazimi, *Nuclear systems: thermal hydraulic fundamentals*. Vol. 1. 2012: CRC Press.
3. Glaeser, H., "GRS method for uncertainty and sensitivity evaluation of code results and applications," *Science and Technology of Nuclear Installations*, 2008.
4. *The Reduced Enrichment for Research and Test Reactor (RERTR) Program*. Available from: http://http://www.rertr.anl.gov/.
5. Ko, Y.-C., *Thermal hydraulics analysis of the MIT research reactor in support of a low enrichment uranium (LEU) core conversion*. 2008, Massachusetts Institute of Technology.
6. Kim, S.J., L.-W. Hu, and F. Dunn, "Thermal-Hydraulic Analysis for HEU and LEU Transitional Core Conversion of the MIT Research Reactor," *Nuclear Technology*, **182** (3) 315-334 (2013).
7. D'Auria, F., et al., *Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation.* IAEA Safety Report Series, 52 (2008).