KAIlpakkam MINI (KAMINI): Beryllium-Oxide-Reflected $^{235}$U-Fuelled Reactor

## EXECUTIVE SUMMARY

In order to make experimental data like neutron multiplication factor ($k_{\text{eff}}$) useful in the validation of the theoretical model and data used in the calculational model, the configuration of the reactor has to be described very accurately. The uncertainties in the geometry, composition and fabrication tolerances contribute to the accuracy of the experimental measurement of $k_{\text{eff}}$. If the error in $k_{\text{eff}}$ prediction due to these uncertainties is less than the targeted accuracy of one percent, then the measurement can be used as a benchmark for the purpose of validating the theoretical prediction of $k_{\text{eff}}$. Benchmarking is the process of describing the reactor very accurately and establishing that the uncertainties in the description will contribute less to less than one percent error in $k_{\text{eff}}$ prediction.

## OUTLINE

In the calculation of $k_{\text{eff}}$ of a reactor lot of nuclear data and complicated computational techniques are used. The present international standard target accuracy in the prediction of $k_{\text{eff}}$ is one percent (1000 pcm). In order to achieve this target one has to validate the nuclear data and calculational techniques against the measured $k_{\text{eff}}$ values. Since the data used is very large the data and calculational methods have to be validated for a number of measured $k_{\text{eff}}$ values. In order to validate the data the reactor where $k_{\text{eff}}$ is measured has to be described very carefully as built. Normally $k_{\text{eff}}$ is calculated using the nominal composition and nominal geometrical arrangement. Practically there will be some engineering tolerances in the fabrication, uncertainties in the composition, density of the materials, impurities in the materials and errors in the measurement of these parameters. These uncertainties will reduce the accuracy of the calculated $k_{\text{eff}}$. In order to know the accuracy in the $k_{\text{eff}}$, the configuration of the reactor has to be described with all the details. The effect of these uncertainties on $k_{\text{eff}}$ has to be estimated and requires lot of sensitivity studies involving the uncertainties in each of the fabricational and compositional error or other uncertainties. Benchmarking is essentially describing the reactor accurately such that error due to the uncertainty in description, lack of data and errors in the measurement will effect $k_{\text{eff}}$ prediction by less than one percent. The factors affecting the accuracy of calculated $k_{\text{eff}}$ can be classified into three broad categories (viz.) uncertainties in: the geometry, the composition and material densities. The effect of various uncertainties is given in Tables 1, 2 and 3. The overall uncertainty due to these uncertainties is found to be 886 pcm which is less than the target limit of 1000 pcm.

Earlier deterministic calculational methods were used where there were lots of limitations in geometrical modeling of the reactor. Finer details could not be modeled. With the advent of fast computers, Monte Carlo codes have come into the forefront of reactor physics calculations. In these calculations, core could be modeled very accurately. In the present benchmarking, the core was modeled as accurately as possible. In the modeling of the reactor, the fuel content of each subassembly is taken accurately as available from the Quality Assurance(QA) reports. In principle the capability exits to model each plate with its own composition. The fuel subassemblies are having 180 geometrical symmetry and plates are not having this symmetry as each plate has got slightly different composition. During the fuel loading, the orientation of the subassembly at the time of loading was not recorded. Due to non-availability of this information each plate could not be modeled and modeling had to be restricted to average compositions of the plates in each subassembly. Few more minor approximations had to be made due to the lack of data. The uncertainty due to the calculation of $k_{\text{eff}}$ due to these approximations is shown to be negligible. The above sensitive studies were used in qualifying this error. For this study, we also used cross-sections generated in continuous energy 'A Compact ENDF/B Routine' (ACER) format by us. Creation of this benchmark established our capability of generation cross-sections accurate modeling of the reactor and describing the reactor accurately to meet the international standards.
**ADDITIONAL BENCHMARKING**

The International Handbook of Evaluated Criticality Safety Benchmark Experiments (ICSBEP) handbook, by design, is a compilation of critical, sub-critical, and radiation-transport benchmark experimental data that have been verified, to the extent possible, by reviewing original and subsequently revised documentation, logbooks, internal memos and letters, and by talking with experimenters or individuals who were associated with the experiments or the fabrication of components of the experimental facility. The experimental data have been reduced to a form, called the benchmark specification that facilitates the development of an analytical model of the criticality experiment. Detailed uncertainty analyses are preformed and an overall uncertainty is assigned to the benchmark model $k_{eff}$ value. When used properly, these data allow criticality safety analysts to validate their analytical techniques and nuclear data evaluators to test data performance without repeating the research and data-reduction steps. Currently, the handbook spans over 38,000 pages and contains 442 evaluations representing 3955 critical, near-critical, or sub-critical configurations, 21 criticality alarm placement/shielding configurations with multiple dose points for each, and 20 configurations that have been categorized as fundamental physics measurements that are relevant to criticality safety applications. The handbook is intended for use by criticality safety analysts to perform necessary validations of their calculational techniques and is expected to be a valuable tool for decades to come. The Handbook is currently in use in 60 countries. With respect to $^{233}$U critical systems, there are presently 196 systems of experimental benchmarks involving $^{233}$U in the Handbook (10 are Metal (Fast spectra); 181 solution (144 Thermal, 29 Intermediate Spectra and 8 Mixed Spectra) and 5 Compound (Thermal spectra) and KAMINI is the 197th configuration.

**KAMINI REACTOR AND CROSS SECTIONS**

KAMINI is a $^{233}$U fuelled tank type, beryllium oxide reflected, demineralised light water cooled low power research reactor. This reactor is designed to operate at a nominal power of 30 kWt and uses a low fuel inventory of 590 g of $^{233}$U in the form of uranium-aluminum alloy plates. This is the only operating reactor in the world that uses $^{233}$U as fuel. Cooling of the reactor is by natural convection and an external heat exchanger cools the tank water to control the inlet temperature. Two absorber type plates (safety control plates) are used for control as well as emergency shutdown of the reactor. This reactor is mainly used for neutron radiography, neutron activation analysis and radiation physics experiments. The horizontal cross sectional view of the core-reflector assembly modeled in the calculation is shown in Fig. 1.

The present trend in Reactor Physics calculations elsewhere is to use Monte Carlo codes which are amenable to model the complicated geometries. Also continuous energy cross-sections have replaced the multi-group cross-sections. We have used the Monte Carlo codes for this study. Also for the analysis of this benchmark continuous energy cross-sections in the ACER format were also generated using in-house capability. The validity of these cross-section generations is verified with the cross-section generated by Korean Atomic Energy Research Institute (KAERI).

**ACHIEVEMENT**

India has successfully completed the task of benchmarking of KAMINI and is formally listed as a participant and member of the USDOE-NEA lead team in the preparation of the ICSBEP. So far the compilation of these benchmarks was not made available to India. After our contribution to the collection of benchmarks India officially provided access to this vast experimental data which will be useful in validating the nuclear codes and nuclear data.

**PUBLICATIONS ARISING OUT OF THIS STUDY AND RELATED WORK**


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