Towards Standard Methodology in the Safety Analysis of Research Reactors

Ali Hainoun

Division for Reactor Safety and Energy System Analysis, Nuclear Engineering Department, AECS, P.O. Box 6091, Damascus, Syria

Email of the corresponding author: ahainoun@aec.org.sy

A large spectrum of research reactors (RR) types has been designed and operated during the last decades. Although the general purpose of RR is to serving as source of neutrons for research purposes, their wide application ranges- which spread from neutron physics to material testing and isotopes production- has dictated the observed large variety in design and operation. Furthermore, the short tarry time of fuel element and the limited inventory of radioactive fission products reflected in a significantly lower radiological potential hazard of RR compared to power reactors has dampened the effort to establish detailed standard approach and the development of comprehensive safety analysis code for RR. Nevertheless, the development of compact type high power density RR together with the increased safety requirements of nuclear installations led to adopt safety analysis approach for RR similar to those for power reactor taking into account the specific features and special design of RR.

Meanwhile it is agreed that despite the above mentioned differences between research and power reactors, the safety objective is the same whenever these differences require flexibility in the implementation of the requirements to achieve the safety objective in research reactor facilities [1].

This contribution demonstrates the recent trends in the safety analysis of RR and deals with the following:

- Classification of RR types using power neutron flux dependency;
- Specific fuel elements design and technical features affecting the safety analysis of RR;
- Procedure of extension and validation of advanced thermal hydraulic codes for the application on the safety analysis of RR;
- Example of the application of an improved thermal hydraulic code for the DBA of RR from MTR type.
- Initiative on Adopting/development of an advanced code for the safety analysis of RR (with the possible support of IAEA);
- Outlook to the perspectives of 3D neutronics thermal hydraulic safety codes for RR and the expected role of CFD codes.
- Classification of RR types using power neutron flux dependency;

Design and Safety Analysis Scheme of RR

FIG. 1. presents the general scheme for a comprehensive procedure on design and safety analysis of RR. Starting from criticality calculation to achieve a certain nominal neutron flux, the geometry specification of core and fuel elements (FE) can be specified in respect to an acceptable core power density. The neutronic analyses for different fuel element status (begin and end of life cycle) provide the key dynamic parameters required for the thermal hydraulic analysis (THA). THA provides the operation limits at nominal power and for operational transients represented by the space distribution of coolant flow rates, system pressure and temperature for coolant, clad and fuel. At this stage the operational limits and conditions (OLC) are established. Nevertheless, the design limits are still not defined. In respect to Defence-In-Depth principle, the realization of safety objective by establishing and maintaining effective defences against radiological hazards in RR facility requires the performance of comprehensive safety analysis (using adequate analysis tools) to demonstrate the safe control of RR facility within the design basis limits. These limits are chosen to prevent fuel-cladding damage and can be estimated based on selected design basis accident (DBA) that can differ from RR type to others and are mostly classified in the category of RIA, LOFA and LOCA. The Analysis of

DBA can provide the design limits for RR fuel element represented by OFI (onset of flow instability) and DNB (departure from nucleate boiling). Should the specified design limits be not guaranteed within specific safety margins the geometry, core arrangement and other neutronic and thermal hydraulic features of the facility has to be modified and the design and safety assessment process is repeated iteratively to achieve at the end the design objective of the RR facility.

Neutronic analyses are meanwhile performed using deterministic and probabilistic tools in parallel. For this purpose well qualified (verified and validated) codes have been established, e.g. CITATION, MCNP. For the thermal hydraulic and safety analysis advanced deterministic tools has been qualified for the application on RR. However, development efforts are still required to cover various phenomena, like 3D neutron dynamic and fluid dynamic codes that are very important for an advanced safety analysis.

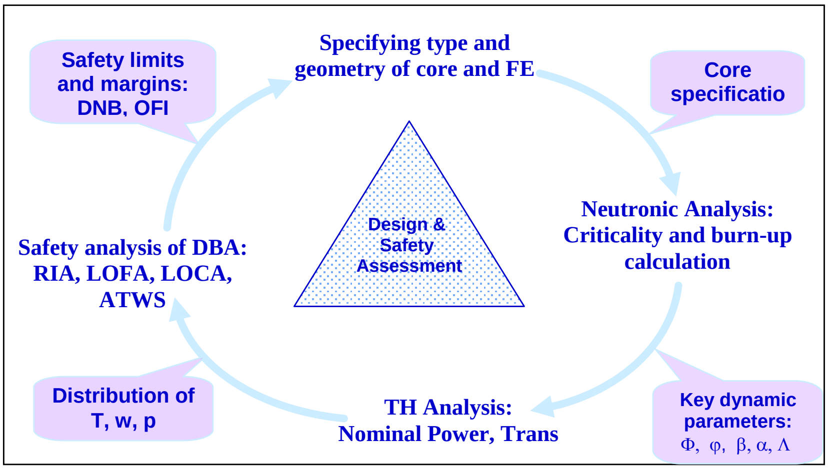

FIG. 1. Scheme for design and safety assessment of research reactors.

References

[1] The Safety of Nuclear Installations, Safety Series No.110, IAEA, Vienna

Plus Additional References

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[2] A. Hainoun, Thermalhydraulic Design and Safety Analysis of Research Reactors, Proceeding of Inter national Conference on: Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management, 10-14 November 2003 Santiago, Chile, , IAEA-CN-100/137.

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