Integrated severe accident codes

Advanced safety evaluations and design optimizations that were not possible few years ago can now be performed. Nowadays, it becomes possible to switch to new generation of computational tools in order to get better realistic simulations of complex phenomena and transients. The challenge today is to revisit safety features of the existing research reactors in order to verify that the safety requirements still met and when necessary to introduce some amendments, coming from not only the new requirements but also, in order to introduce new equipments from recent advancement of new technologies. The objective of this work is to give an overview of the state of the art in performing safety analysis of research reactors and to emphasize the need and the provision to achieve such goals.

An attempt to perform standardized safety analyses for RR was proposed by the International Atomic Energy Agency e IAEA. In the framework of core conversion from the use of highly enriched uranium fuel to the use of low enriched uranium fuel. In this regard, a safety related benchmark problem for an idealized generic 10 MW MTR light-water pool-type reactor was specified in order to compare computational methods used in various research centers and institutions. The related benchmark problem covers large steady state kinetic and thermal-hydraulic calculations and wide range of hypothetical dynamic transient conditions. However, almost all of the safety analyses have so far been performed using conservative computational tools.

Nowadays, an established international expertise in relation to computational tools, procedures for their application, including best estimate methods supported by uncertainty evaluation, and comprehensive experimental database exists within the safety technology of Nuclear Power Plants (NPP). The importance of transferring NPP safety technology tools and methods to RR safety technology has been noted in recent IAEA activities. However, the ranges of parameters of interest to RR are different from those for NPP. This is namely true for fuel composition, system pressure, adopted materials and overall system geometric configuration. The large variety of research reactors prevented so far the achievement of systematic and detailed lists of initiating events based upon qualified PSA (Probabilistic Safety Assessment) studies with results endorsed by the international community. However, bounding and generalized lists of events are available from IAEA documents and can be considered for deeper studies in the area.

In the area of acceptance criteria, established standards accepted by the international community are available. Therefore no major effort is needed, but an effort appears worthwhile to check that those standards are adopted and that the related thresholds are fulfilled.

The importance of suitable experimental validation is recognized. A large amount of data exists as the kinetic dynamic core behavior form SPERT reactors tests. However, not all data are accessible to all institutions and the relationship between the range of parameters of experiments and the range of parameters relevant to RR technology is not always established. However, code-assessment through relevant set of experimental data is recorded and properly stored.

An established technology exists for development, qualification and application of system thermal-hydraulics codes suitable to be adopted for accident analysis in research reactors. This derives from NPP technology. The applicability of system codes like RELAP5, COBRA and MARS to the research reactor needs has been confirmed from recent IAEA activities. Definitely, system codes are mature for application to transient analysis in research reactors. However, code limitations have been found in predicting pressure drops as a function of mass flux at low values of mass flux when nucleate boiling occurs. The importance of the Whittle and Forgan experiments shall be mentioned, as well as the dependence of results from the noding (cell subdivision) adopted by the code users.

Several code user choices, including time step may have a significant effect upon prediction, thus confirming the need for detailed code user guidelines. Furthermore, code validation must be demonstrated for the range of parameters of interest to research reactors. The crucial role of uncertainty in research reactor technology has been emphasized, (a) for the design, with main reference to the prediction of the nominal steady state conditions and, (b) for the safety issues, with main reference to the prediction of the time evolution of significant safety parameters.

It has been observed that suitable-mature methods exist, but the spread of these methods and procedures within the community of scientists working in research reactor technology is limited. Therefore, the purpose of the present report is to provide an overview of the accident analysis technology applied to the research reactor, with emphasis given to the capabilities and limits of the used computational tools.

There are many analysis codes for transient and accident analysis and simulating individual phenomena of severe accident. These analysis codes can be categorized into the different groups as shown in Table 1, where various analysis codes are classified into several groups.

The integrated severe accident codes are formed by selecting and combining individual analysis tools. They can be used to model the whole sequence of the severe accident which may occur in the plant system or in the experimental facilities.