Overview of accident analysis in nuclear research reactors

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Abstract

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.

Keywords: Research reactors safety; Accident analysis; Reactivity accidents; LOCA; Loss of heat removal; Loss of flow accidents; Validation; Channel codes; Best estimate computational tools

1. Foreword

Nuclear research reactors have been of great support in the development of nuclear science and technology. According to the IAEA inventory (IAEA-TECDOC-1387, 2004), 651 nuclear Research Reactors (RRs) have been built in 56 countries around the world. To date almost 284 research reactors are currently in operation, 258 are shut down and 109 have been decommissioned. More than half of all operating research reactors reached their first criticality more than thirty years ago. Therefore, some face concerns of obsolescence of equipment, lack of experiment programmes, lack of funding for operation and maintenance, loss of expertise, equipment ageing and retirement of staff. The 258 research reactors that no longer operate are in some form of shutdown. Some of these are expected to restart sometime in the future, some are waiting for decommissioning, with or without nuclear fuel in the facility, and the remaining reactors have no clear definition about their future. The undefined status of these remaining research reactors gives rise to safety concerns, in particular in the areas of loss of corporate memory, personnel qualification, maintenance of components and systems, and preparation and maintenance of documentation.

In general, the purpose of nuclear research reactors is not for energy generation; the maximum power generated within does not exceed 100 MW. They are commonly devoted for generation of neutrons for different scientific and social purposes. However, high power densities are involved in the core and specific features are necessary to ensure safe utilization of these installations. In addition to their particular characteristics, including large variety of designs, wide range of powers, different modes of operation and purposes of utilization, special attention should be focused for their safety aspects. Thus accurate safety evaluations, for instance in case of core reloading, planned power up-rating, or as part of required analyses of occurred events, should be considered. Generally speaking, the content of the safety report has to include...
be modified whenever a new type or design of fuel is to be used in the reactor core. As the existing plants have well established licensing procedures, including well founded analysis methods, the application of new innovative analysis methods has to be thoroughly evaluated, with specific emphasis to the capabilities of producing results that in general terms might be beneficial related to the RR operations.

An attempt to perform standardized safety analyses for RR was proposed by the International Atomic Energy Agency — IAEA (IAEA-TECDOC-233, 1980) 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, the facility operator would be required to submit an amendment to, or a revision of, the Safety Report. For this purpose, 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 (IAEA-TECDOC-643, 1990; Woodruff, 1984; Baggoura et al., 1994).

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 (Forbes and Nyer, 1961). 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

![Fig. 1. UMLRR nodalization scheme.](image-url)
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.

Fig. 2. Pool heat-up during the heat-up experiment.

Fig. 3. Status of the liquid level in the pool and the core after the postulated LOCA.

Fig. 4. Status of a partially submerged fuel element.

Fig. 5. Partially submerged fuel element model.
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 (IAEA-RCP J7-10.10., 2002). Therefore, the purpose of the present paper 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.

2. Examples from code qualification process

Currently, with widespread use of research reactor, there is a real need to get more realistic simulations of the phenomena involved during steady state and transient conditions, and eventually the identification of design/safety requirements that can be relaxed or enhanced (Hamidouche et al., 2004). Several attempts were performed to assess the applicability of Best Estimate codes to RR operating conditions (Woodruff et al., 1996; Hamidouche et al., 2004). Relevant assessments were applied against the following cases.

2.1. UMLRR research reactor

The UMLRR is a 1 MW MTR pool-type nuclear research reactor. The main features connected with this reactor are the fact that operational experimental data are available online and constitutes a real source for detailed measurement data for various code validations and system analyses (Bousbia-Salah et al., 2006a). Fig. 1 shows the REALP5 nodalization of the reactor, while Fig. 2 shows typical results obtained for the simulation of a pool heat-up experiment.

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Fig. 6. Evolution of the maximum fuel plate temperature for various core immersion heights.

Fig. 7. MTR benchmark transversal core cross-section.

Fig. 8. Radial fast flux distribution of 93% — fresh core.
2.2. NUR research reactor

In this case, a general approach used at the NUR MTR pool-type research reactor for neutron flux optimization in irradiation channels is considered. In order to allow the implementation of the new core configuration in the operation scheme of the reactor, detailed neutronic and thermal hydraulic studies are performed. The safety steady state analysis were carried out using combined lattice code WIMS-D4 (Askew et al., 1966) with CITATION diffusion code (Fowler et al., 1971) whereas transients calculations were performed using channel codes such as PARET (Obenchain, 1969) and the in-house LODEHR (Bousbia-Salah et al., 2006b,c) codes. The objective was to ensure that the allowed operational safety limits of the reactor are not exceeded (Meftah et al., 2006).

The schematic representation of core uncovering during a loss of coolant transient is shown in Figs. 3 and 4. Fig. 5 shows the adopted nodalization for the LODEHR code, which take into account the level tracking approach. While Fig. 6 shows the transient evolution of the core temperature during the postulated core uncoveracy accident.

2.3. The IAEA MTR fuel type research reactor

The IAEA Benchmark is based upon one of the SPERT series test reactors (Forbes and Nyer, 1961). The reactor is an open pool 10 MW MTR fuelled core type. The core configuration is shown in Fig. 7. The grid contains 21 Standard MTR Fuel Elements (SFEs) and 4 Control Fuel Elements (CFEs). The SFEs contain 23 standard plates whereas the CFEs contain 17 standard plates with special region to receive the 4 fork type absorber blades. The core is cooled and moderated by downward forced circulation of light water. The benchmark consists in performing static and dynamic calculations. The static computations were performed (Bousbia-Salah et al., in source).

\begin{figure}[h]
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\caption{Radial thermal flux distribution of 93% — fresh core.}
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\begin{figure}[h]
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\includegraphics[width=0.8\textwidth]{fig10.png}
\caption{Typical representation of an MTR research reactor applied for IAEA Benchmark reactors.}
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\begin{figure}[h]
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\includegraphics[width=0.8\textwidth]{fig11.png}
\caption{Outlet fluid temperature during RIA transient.}
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\begin{figure}[h]
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\includegraphics[width=0.8\textwidth]{fig12.png}
\caption{Fuel temperature and relative mass flow rate during slow LOFA.}
\end{figure}
press) using the Monte Carlo MCNP5 code (Jeremy, 2003). They concern criticality estimations and derivation of key kinetic parameters, as the spatial neutron flux distribution and the multiplication factor.

Figs. 8 and 9 show the MCNP5 results for the thermal and fast neutron flux, respectively. On the other hand, the dynamic calculations were performed (Hamidouche et al., 2004) using the Best Estimate thermal–hydraulic system code RELAP5/3.3. A standard representation of an MTR RR as shown in Fig. 10 has been adopted and the nodalization has been developed and qualified according to the qualification procedure used at the University of Pisa (Hamidouche et al., 2004). The benchmark problem consists in analyzing protected transients in MTR Highly Enriched Uranium (HEU) and Low Enriched Uranium (LEU) cores. The analysis concerns mainly rapid kinetic transient initiated by positive Reactivity Induced Accidents (RIA), and relatively slow transient (thermal–hydraulic driven) cases related to complete LOss of core coolant Flow (LOFA). Representative results are given in Figs. 11 and 12, where comparison is made, when applicable, with reference results obtained by the RR devoted codes PARET and RETRAC-PC (Bousbia-Salah and Hamidouche, 2005).

2.4. The Whittle and Forgan and THTL tests facilities

Relevant experimental data related the separate effect Onset of Flow Instabilities (OFI) in the Oak Ridge National Laboratory Thermal Hydraulic Test Loop (THTL) (Siman-Tov et al., 1994; Siman-Tov et al., 1995), as well as some representative Whittle and Forgan (W&F) experiments (Forgan and Whittle, 1966; Whittle and Forgan, 1967), and RELAP5 calculation results are compared in Fig. 13 for the THTL case and in Fig. 14 for the W&F case (Hamidouche and Bousbia-Salah, 2006). The objective of the work was to investigate the RELAP5/3.2 system code capabilities in predicting phenomena that could be encountered under abnormal research reactor’s operating conditions. Similar work was performed for the validation of ATHLET code (Lerchl and Austregesilo Filho, 1995) under research reactor operating conditions (Hainoun and Schaffrath, 2001).

2.5. SPERT tests

Validation process was also performed against experimental tests as (Bousbia-Salah and Berkani, 2001) and those issued from the full scale SPERT (Special Power Excursion Reactor Test) cases (Montgomery et al., 1957). Through these tests a check of the models of codes such PARET, RETRAC-PC, and RELAP5 is carried out. The SPERT III system was designed to investigate the effects of the initial system conditions on the core behaviour under step reactivity insertions’ events. The SPERT III reactor is light-water-cooled and moderated MTR core with operating power of 60 MW. The power self-limiting tests were performed by considering step positive reactivity insertions ranging from 0.2 $ to 1.25 $ (Toole, 1964).

Results of the validation process are shown in Fig. 15, which summarises the calculation obtained for a large spectrum of

![Fig. 13. Channel pressure drop for THTL hot cases.](image)

![Fig. 14. Pressure drop characteristic curve for W&F TS2.](image)

![Fig. 15. Comparisons of SPERT III-C 19/52 data with PARET calculations.](image)
self-limited transients in the SPERT III-C 19/52 core. Acceptable agreement is obtained when some adjustment of the feedback parameters are performed (Bouaouina et al., 2007).

3. Topics of interest for accident analysis in RR and current status

A list of topics relevant to the deterministic accident analysis in research reactors is provided below.

- **Postulated Initiating Events.** The identification of accident scenarios, typically derived by considering probability of occurrences and severity of consequences, constitutes the first step needed for performing deterministic safety analyses.
- **Acceptance criteria.** The availability of ‘thresholds of acceptability’ for consequences of accident, as a function of the probability of occurrence of the event, constitute the second requirement for deterministic safety analyses: namely results of the analyses shall be compared with ‘limiting values’. Acceptance criteria are imposed by national authorities and are not connected with the deterministic safety analysis.
- **Experimental database.** Computational tools are used to perform deterministic analyses and experimental data are needed to demonstrate the quality of those tools. Experimental data can also be used directly to improve the design and the performance of research reactors.
- **Qualification of system codes.** System codes, such as RELAP5 (Fletcher and Schultz, 1995), CATHARE (Barré and Bernard, 1990), ATHLET, CATHENA (Hanna, 1998), widely used within the safety technology of NPP (IAEA TECDOC-1351, 2003; NEA/CSNI/R(99)10, 1998), could be used for the deterministic analysis of accidents in RR. The application must be based upon the evaluation of the complexity of the transient: in a number of cases owing to the ‘simple’ configuration of RR compared with NPP, simpler tools including analytical-hand calculations should be used. However, for the cases when system codes are needed, proper demonstrations of qualification must be provided.
- **Uncertainty in research reactor technology.** The role of uncertainty has been considered at 2 levels: (a) design of research reactors, mostly addressed to the calculation of nominal steady state operating conditions, and (b) evaluating the results from best estimated predictions performed by thermal—hydraulic system codes, mostly addressed to the calculation of transient scenarios. Origins and impacts of uncertainty within both the frameworks are discussed in (D’Auria et al., 2006; Bousbia-Salah et al., 2006c). Methods and procedures to deal with uncertainty are presented in the same reference.

4. Conclusions and recommendations

The demonstration of applicability of qualified best estimate system codes to RR accident analysis constitutes the key message from this paper: a proper accident analysis technology should be developed for RR that could benefit of the experience available from NPP, considering that the risk level and the cost associated with RR are orders of lower magnitude. Recommendations are provided hereafter distinguishing between potential RR system thermal—hydraulic code users and decision makers in the area.

Recommendations to the users of computational tools are as follows:

- To consider experimental data and to perform code-to-experiment comparison before any code application to prediction relevant to the RR design or safety analysis.
- To demonstrate that any code adopted for design and safety is qualified.
- To consider that any best estimate code, even though supported by the use of the optimized procedures, produces results affected by an unknown error, i.e. uncertainty.

Recommendations to decision makers focus on establishing an international understanding in the area:

- To plan “benchmark” exercises in conditions where neutron kinetics and natural circulation are relevant.
- To promote the use of PSA techniques, establishing detailed PIE (Postulated Initiating Events) lists.
- To make an effort to establish ‘validation-matrices’ for computational tools.
- To plan suitable training in the area of RR accident analysis.
- To consider innovative techniques including of CFD (Computational Fluid Dynamics) and coupled three-dimensional neutron kinetics codes and thermal—hydraulic system codes.

References


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