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Accident Analysis in Research Reactors

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ABSTRACT

With the sustained development in computer technology, the possibilities of code capabilities have been enlarged substantially. Consequently, advanced safety evaluations and design optimizations that were not possible few years ago can now be performed. The challenge today is to revisit the safety features of the existing nuclear plants and particularly research reactors in order to verify that the safety requirements are still met and - when necessary - to introduce some amendments not only to meet the new requirements but also to introduce new equipment from recent development of new technologies. 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 of computational tools.

1 INTRODUCTION

Nuclear research reactors (RRs) have been of great support in the development of nuclear science and technology. The first nuclear research reactor went critical on *2 December 1942*. According to the IAEA inventory about 830 research reactors have been constructed worldwide to date, with some 20 more under construction or planned. Of these, about 290 are in operation and about 520 are shut down and at various stages of decommissioning. Approximately 25% of research reactors currently in operation around the world are over 40 years old. There are also several dozen research reactors that have not yet completed the decommissioning process [1].

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) [2]. 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. 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 to the research reactor needs has been confirmed from recent IAEA activities.

Several code user choices [3], 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.

2 GENERAL OVERVIEW OF RESEARCH REACTORS

In general, the *purpose* of nuclear research reactors is not for energy generation, the maximum power generated within didn't 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.

As the existing plants have well established licensing procedures, including well founded analysis methods, the application of new innovative analysis methods have 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 IAEA in the framework of core conversion from the use of highly enriched uranium fuel to the use of low enriched uranium fuel [4], [5], [6], [7].

2.1 Types of Research Reactors

The categorization of reactor types is difficult because of a lack of any standardization. According to IAEA research reactor database, research reactor types may be categorized according to the criteria outlined in Table 1 [1].

In view of type diversity and wide capacity range, research reactors may also be classified (there is no general consensus on the clasification creteria of RRs) in different groups using the criteria of *core power* density resulting in the relation between neutron flux and core power (see Figure 1).

Type of reactor	Number of reactors	Subtotals
Pool type		
TRIGA	74	
SLOWPOKE/MNSR	19	
Others	160	253
Tank type		
Heavy water	48	
ARGONAUT	29	
Pressurized	22	
Others	90	189
Homogeneous liquid	45	
Homogeneous solid	44	
Fast	37	
Graphite	44	170
Others		
Zero power[a]	135	
Miscellaneous	78	
Unknown	7	220
Total		832

[a] Including critical and subcritical assemblies, prompt burst and pulsing reactors.

Table 1: Classification of research reactors

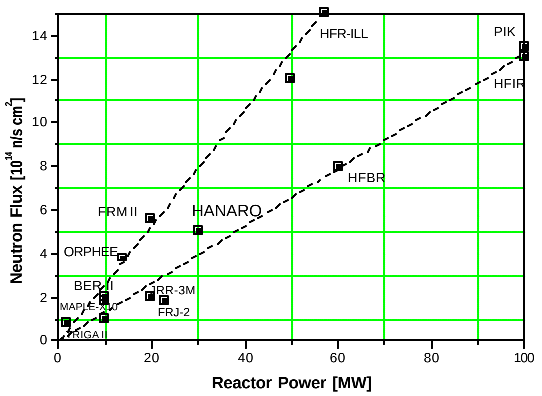

Figure 1: Selected examples on the relation between neutron flux density and reactor power for various reactor types

3 ACCIDENT ANALYSIS METHODS

Most research reactors have a small potential for hazard to the public compared with power reactors but may pose a greater potential hazard to operators. Considering the different types of research reactors and their associated utilization, application of a graded approach to safety analysis shall be commensurate with the potential hazard, ensuring that the design and operation of a research reactor lead to adequate safety (i.e. SL-1 accident) [8].

Two properly balanced complementary *methods of safety analysis*, deterministic and probabilistic, can be used in evaluating the safety of a research reactor. Typically, deterministic methods are used for safety evaluation of RR. Deterministic techniques are characterized by conservatism because they use single numerical values (taken to have a probability of 1), leading to a single value of the result. Deterministic safety analysis is based on a defined set of rules for event selection, analytical methods, parameter specifications (e.g. initial operating state and the initiating event) and acceptance criteria. Through the use of these methods, reasonable assurance is provided that the ultimate objective, to limit release of radioactive materials, can be achieved without performing complex calculations, because these methods tend to overestimate the amount of the release.

Computational models using deterministic approach are normally employed for transient and accident analysis of the DBA, BDBA or any of the anticipated operational occurrences. These models (and codes) have to be applicable over the expected range of operational parameters, yield conservative predictions, represent all physical important phenomena and have been properly validated. Confidence in the results of such modeling, and consequently in the safe design as well as safe operation of the reactor, depend strongly not only on the capability to model the physical phenomena, but also on the input parameters and initial conditions. Validation of that capability therefore through relevant experimental programs and/or real plant operational data (obtained during start-up tests, steady state operations, or during operational events) therefore, is desirable. Deterministic analysis is typically focused on neutron kinetics, thermal-hydraulics, radiological and structural aspects, which are often analyzed with different computational tools.

4 DESIGN LIMITS FOR RESEARCH REACTORS

In general, the escape of fission products from fuel and fuelled experiments and their release to unrestricted environment would be the most hazardous radiological accident conceivable at research reactor. However, research reactors are designed and operated so that a fission product release is not credible for most. Therefore, the release under accident conditions can reasonably be selected as the DBA, which bounds all credible accidents and can be used to illustrate the analysis of events and consequences during the accident release of radiological material.

Most of research reactors have compact core structure with multi parallel channels and are therefore subject to the so-called thermal hydraulic instability (THI) - or flow excursion (FE) - which differs from critical heat flux (CHF) that would occur at a fixed channel flow rate. During postulated design accidents in research reactors, THI would occur before CHF limit is reached [9]. Thus, THI is defined as design limit of research reactors of plate type fuel. In this regard the operation limits of nuclear reactors have been set to prevent fuel-cladding damage. By the design basis accidents, damage would occur by PWR at departure from nucleate boiling (DNB) resulting from vapour film forming at the fuel surface; and by BWR when the liquid film on the fuel surface dries out. The heat flux at DNB or liquid film dry out is called CHF. PWR and BWR are designed in such a way that the fuel clad doesn't exceed temperature referred to CHF.

The compact core structure and the special design of the fuel elements of research reactors make particular demands on the thermal hydraulic construction and on the technical safety measures. Due to the high heat flux and the low system pressures various accidents, such as impairment of forced cooling by failure of the coolant pumps, thermal hydraulic flow instabilities may arise in the narrow cooling channels due to steam formation mostly in sub-cooled boiling regime, which results in the critical heating surface being exceeded within a few seconds affecting the fuel plates which are damaged due to their extremely low melting point (~ 660 °C). Furthermore, many other thermal hydraulic phenomena are resulting from the above mentioned specific design of research reactor cores.

4.1 Design Basis Accidents

The accident analyzed should range from such anticipated events as a loss of normal electrical power to a postulated fission product release with radiological consequences that exceed those of any accident considered to be credible. This limiting accident is termed the Design Basis Accident (DBA) for research reactors.

The initiating events in each group should be evaluated systematically to identify the limiting event selected for detailed quantitative analysis. Limiting events in each category should have potential consequences that exceed all others in that group. The DBA selected should bound all credible potential accidents at that facility, yet should be an event that is not likely to occur during the life of the facility.

4.2 Beyond Design Basis Accidents

Accidents that are more severe than DBAs are termed Beyond Design Basis Accidents (BDBAs). Depending on the potential hazard from the reactor, specific design features for emergency planning may be considered and derived from analyses of BDBAs in the research reactors.

5 BEST ESTIMATE CALCULATIONS AND QUALIFICATION: AN EXAMPLE

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. Several attempts were performed to assess the applicability of Best Estimate codes to RRs operating conditions.

5.1 University of Massachusetts Lowell Research Reactor

5.1.1 Reactor Description

The University of Massachusetts Lowell Research Reactor (UMLRR) is a 1 MW, light-water moderated and cooled, graphite-reflected, open-pool type research reactor (Figure 2) that has been in operation since January 1975. 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. A REALP5 nodalization was performed for the reactor (Figure 3) [10].

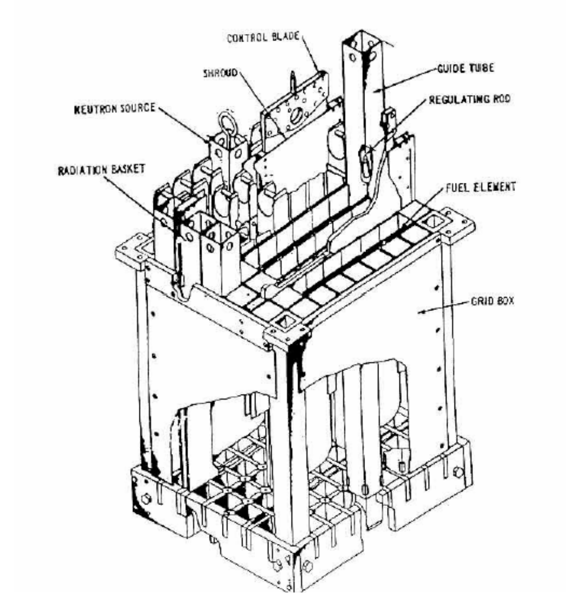

Figure 2: Three-dimensional detail of reactor core

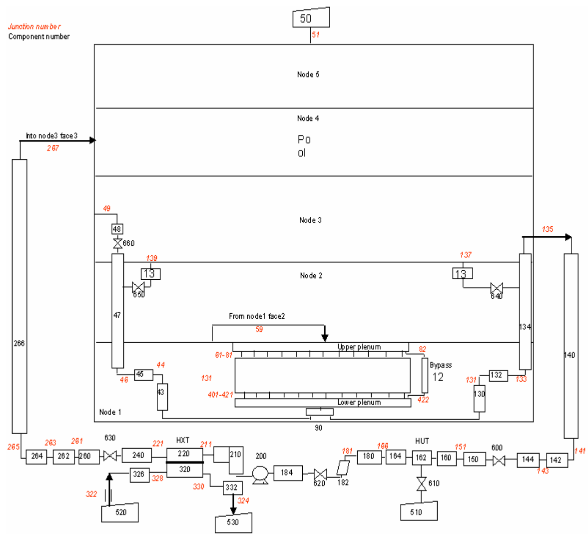

Figure 3: UMLRR Nodalization

5.1.2 Model Validation

The first step concerns the assessment of the RELAP5 response to steady state conditions through the comparison of the calculated parameters with experimental data. In a second step, transient conditions are simulated with the RELAP5 on the adjusted nodalization

5.1.3 Steady state

The validations of the RELAP5 nodalization pass through the demonstration that the RELAP model reproduces the measured steady state conditions of the UMLRR with acceptable margins. For this purpose, a number of parameters are selected for comparison with measured data. Two steady state experiments under forced and natural convective regimes were performed.

5.1.3.1 Forced convection steady state

In this case, the reactor operated at steady state of about 900 KW with primary pump turned on (forced convection mode). The water is forced to flow downward into the core with a pump flow rate of 360-380 m^3/h. A RELAP simulation of the reactor was run for about 100 seconds in order to obtain stable calculation results. Table 2 shows the key parameters from the RELAP5 simulation and the measured data from the experiment. The calculated results show good agreement with the measured data and the deviation in the worst case didn't exceed 6%.

Table 2: Comparison of steady state forced convection mode

	RELAP	**Experiment**	**Error* %**
Power (MW)	0.9	0.88 - .92	2.2
Pool Inlet temperature (C)	23.91	23.66	1.0
Core inlet Temperature (C)	23.01	22.70	1.3
Core Outlet Temperature (C)	25.12	24.70	1.7
Delta Core Temperature (C)	2.11	2.0	5.5
Primary Flow Rate (gpm)	1650	1700	2.9
Primary Pump	ON	ON	-
Steady State	yes	yes	-

* error =(RELAP-Experiment)/Experiment

5.1.3.2 Steady state natural convection

In a second step, the natural convection operation mode is calculated by RELAP5. In this case, the primary pump is turned off and the coolant flow though the reactor core depends only on the core temperature difference. The operating power under natural convection is fixed at 80 KW. Figure 4 shows the mass flow rate through the fuel assembly as calculated by REALP5. Due to the fact that it is not possible to compare the value of the mass flow rate since the UMLRR instrumentation does not have the capability for such purpose, a code-to-code comparison was considered using the NATCON code.

Figure 5 shows the results of the two codes for the axial clad temperature profiles where some minor deviations are observed. The clad temperature as calculated by RELAP5 is underestimating the NATCON data by 3° C whereas it overestimated at the top by 2°C. These deviations are expected to be due to the heat transfer correlations used in each code for natural convection regime. As a consequence, the total temperature increase across the core computed by RELAP5 is 10 °C while it was 15 °C in case of NATCON.

However, since the RELAP5 is underestimating the clad temperature in reference to PLTEMP code, a deeper investigation on the correlation used in RELAP5 should be undertaken in future.

5.1.4 Transient analysis of pool heat-up

In a second assessment step, a series of transient situations were managed experimentally and calculated by RELAP5. The purpose of this section is to assess the thermal-hydraulics model of the RELAP5 code by recalculating the whole system heat up. For this case, the nodalization used includes only the primary piping system and the reactor pool.

The basic idea of that experiment was to operate the reactor at critical constant power when the secondary circuit is disabled. The primary pump was turned on whereas the secondary pump was turned off in order to induce an increase of the pool temperature. In this experiment, the four control blades were positioned to make the reactor critical. The regulating rod was set in auto mode to control and compensate any reactivity changes in the core in order to keep the reactor critical at the fixed power. Key parameters used for this case are summarized in Table 3. The calculations were run with the primary and secondary systems turned on for about 400 seconds to establish the steady state condition. Then the secondary system was shut off by setting the flow rate through the secondary side to zero. The temperature coefficient data for point kinetics model were set to zero since the regulating rod was used to compensate the temperature feedback and to maintain core power level constant.

During the steady state condition, acceptable agreement is observed between the calculated and the measured data (up to 400 seconds) as could be seen on Figure 6 and 7 for the pool and core plenums; the discrepancies are less than 1°C. These discrepancies are completely bounded by the measurement uncertainties.

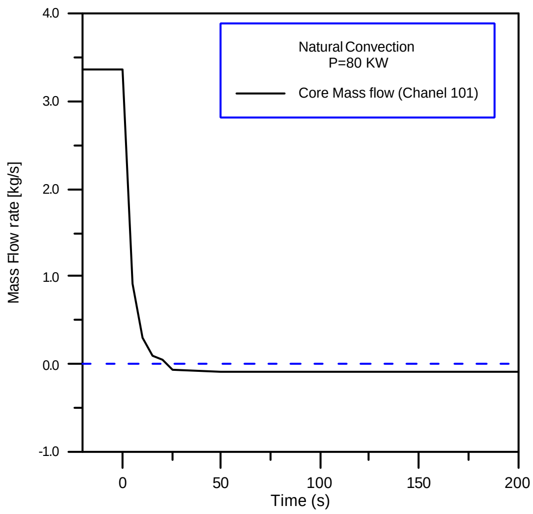

Figure 4: Fuel assembly mass flow rate

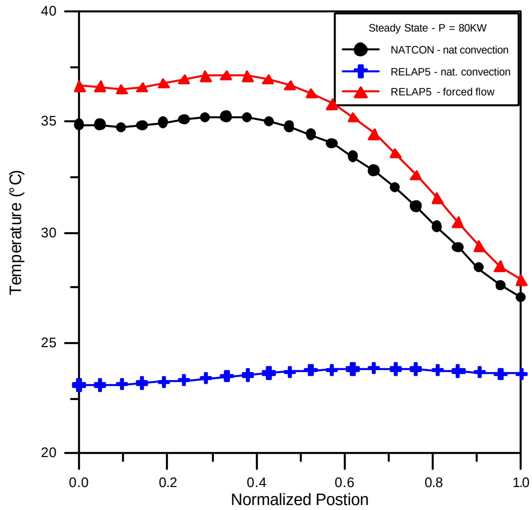

Figure 5: Clad temperatures for natural convection mode

After the beginning of the transient, the temperatures increase as consequence of no energy removal from the primary since the secondary system was off. All the temperatures are shown to increase gradually at a mean rate of 2.7 °C per hour because of continuous power production. Figure 6 and 7 show temperature evolution at different positions in the pool (core plenums and otherwise in pool) as obtained by RELAP5 and by measurements. In general, the same trends are observed and a good agreement is achieved. Furthermore, we observe that the discrepancies are reduced further a long time after the beginning of the transient; such behaviour could indicate that the calculations are well reproduced "macroscopically" by the RELAP5 model when neutronic interactions are not considered.

Table 3: System conditions

Core Power (MW)	0.9
Convection mode	forced
Primary system	on
Secondary system	off
Inlet temperature (c) for the pool (1st guess)	23
Inlet temperature (c) HEX 1st side (1st guess)	26
Inlet temperature (c) HEX 2nd side (1st guess)	25
Feedback coefficients	0

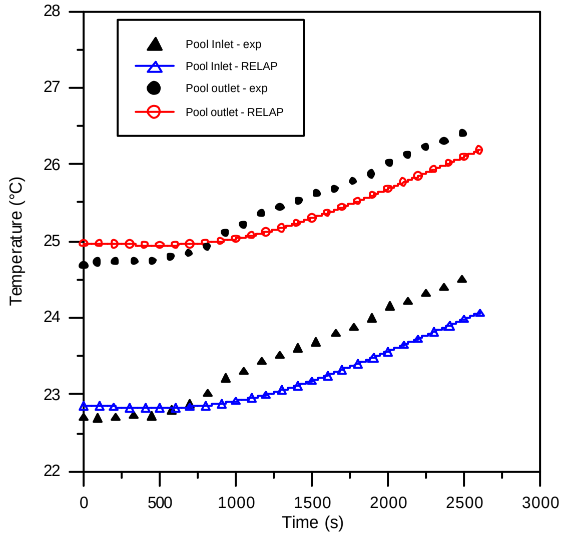

Figure 6: Pool heat up during the heat up experiment

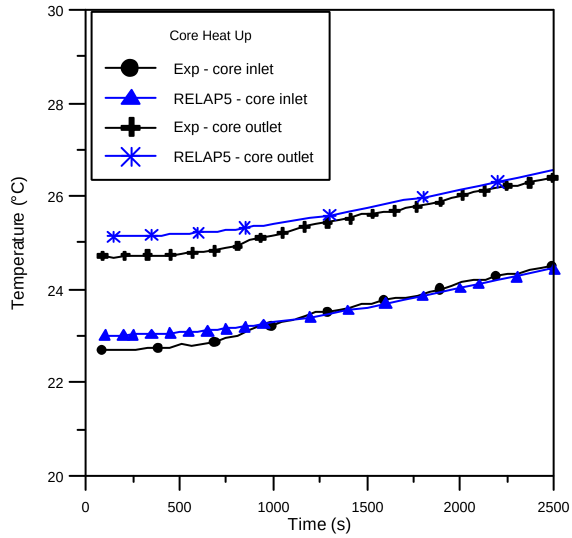

Figure 7: Core (plenums) inlet and outlet temperatures

6 CONCLUSION

Increased consideration to safety issues for research reactors has emerged as a consequence of their enlarged commercial exploitation. So far, conservative computational tools were used to perform safety analyses for the design and development of such reactors. Nowadays it becomes necessary to review such limiting tools by using BE calculation methods.

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 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.

The demonstration of applicability of qualified BE system codes to Research Reactor accident analysis constitutes the key message from this paper: a proper accident analysis technology should be developed for Research Reactor that could benefit of the experience available from NPP, considering that the risk level and the cost associated with Research Reactor are orders of magnitude lower.

The following attainment have been emphasized:

- Confirmation of the necessity to apply BE computational tools in future applications relevant for safety analysis in order to accomplish with international trend in the safety analysis area.
- Confirmation that the Best Estimate codes are mature enough for safety application in Research Reactors.
- Even though the analysis is still not yet supported by other relevant applications the methodology applied and the results obtained appear adequate.
- A possible limit of the analysis is due to the fact that a proper PSA analysis is not yet been applied.

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