Chapter 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, COBRA and CATHARE, ATHLET 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 current work constitutes a pioneering study that brought into light the use of BE methods for Research Reactors and is opinion of the author that it will be the future.

The demonstration of applicability of qualified BE system codes to Research Reactor accident analysis constitutes the key message from this thesis: 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. Through this study the following topics were emphasized for better and future application of computational tools for safety analysis of Research Reactor:

- To consider experimental data and to perform code-to-experiment comparison before any code application to prediction relevant to the Research Reactor design or safety analysis.

- To consider that any best-estimate code, even though supported by the use of the optimised procedures, produces results that are affected by an unknown error, i.e. uncertainty.
- 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 Research Reactor accident analysis.
- To consider innovative techniques including of CFD (Computational Fluid Dynamics) and coupled three-dimensional neutron kinetics codes and thermal-hydraulic system code

On the other hand 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.

References

Adorni M., Bousbia Salah A., D'Auria F., Nabbi R., **2005**, “Analysis of a Reactivity Transient for the DIDO Type Research Reactors Using RELAP5”, International Conference Nuclear Energy for New Europe, Bled (SLO).

Adorni M., Bousbia Salah A., Hamidouche T., Di Maro B., Pierro F., D'Auria F.,**2005**, “Analysis of Partial and Total Flow Blockage of a Single Fuel Assembly of an MTR Research Reactor Core”, Annals of Nuclear Energy , vol. 32, pp. 1679-1692.

Adorni M., Bousbia-Salah A., D'Auria F., 30 April-3 May **2006** “Application of Best Estimate Thermalhydraulic Codes for the Safety Analysis of Research Reactors”, Transactions of the 10th International Topical Meeting on Research Reactor Fuel Management (RRFM), Sofia, Bulgaria, pp.315-319.

Adorni M., Di Maro B., Pierro F., Bousbia Salah A., D'Auria F., **2004**, “Analysis of a total flow blockage of a fuel assembly in a typical MTR research reactor by RELAP5/mod3.3”, International Conference Nuclear Energy for New Europe, vol. 1, pp. 120-130, Portoroz (SLO).

Aksan N., D’Auria F., Glaeser H., Pochard R., Richards C., Sjoberg A., **1993** “A Separate Effects Test Matrix for Thermal- Hydraulic Code Validation: Phenomena Characterization and Selection of Facilities and Tests”, OECD/GD (94) 82, Vols. I and II.

Aksan S. N., D'Auria F., Staedtke H., **1993**, “User Effects on the Thermal-hydraulic Transient System Codes Calculations”, Nucl. Eng. Des., 145, 1&2.

Ambrosini W., Bovalini R., D’Auria F., **1990**, “Evaluation of Accuracy of Thermal-hydraulic Codes Calculations”, Energia Nucleare, 7, 2.

ANL Reactor Homepage website, Argonne National Laboratory, http://www.anl.gov/.

Bousbia Salah A., Jirapongmed A., Hamidouche T., D'Auria F., Adorni M, **2006**, “Assessment of RELAP5 model for the University of Massachusetts Lowell Research Reactor”, Nuclear Technology & Radiation Protection, 1/2006, pp. 3-12.

Bousbia Salah. A., D’Auria. F., Galassi. G.M., Giannotti. W., Lo Nigro. A., **2003**, “Overview of coupled 3D System Thermalhydraulics Neutron Kinetics Code Applications”, IAEA technical meeting on progress in development and use of coupled codes for accident analysis, Vienna.

Bousbia-Salah A. Hamidouche T., **2005**, “Analysis of the IAEA research reactor benchmark problem by the RETRAC-PC code”, Nuclear Engineering and Design, Vol.235, 661–674.

Bousbia-salah A., and Berkani R., **2001**, “Analysis of Reactivity Insertion Transient Experiments in a Nuclear Reactor Core and Comparison with RETRAC-PC Computational Results”, Algerian Review of Nuclear Science, Vol. 3, pp.79-84.

Bousbia-Salah A., D'Auria F, **2007**, "Use of coupled code technique for Best Estimate safety analysis of nuclear power plants", Progress in Nuclear Energy, Progress in Nuclear Energy, Vol. 49, p.1-13.

Bousbia-salah A., Hamidouche T. and D'Auria F., Vienna, 7-12 November **2004**, "Analysis of typical transients in the generic 10MW IAEA MTR research reactor by RELAP5/Mod 3.2", RERTR 2004, International Meeting on Reduced Enrichment for Research and Test Reactors, Austria.

D'Auria F., **2000**. "Nodalisation Development and Qualification". IAEA TCM on Inter-comparison and Validation of Computer Codes for Thermal-hydraulic Safety Analyses of Heavy Water Reactors, Vienna.

D'Auria F., Bousbia-Salah A., 15-20 October **2006**, "Accident analysis in research reactors", 15th Pacific Basin Nuclear Conference (PBNC) at the Sydney, Australia.

D'Auria F., Bousbia Salah A., **2004**, "User effect on code application and qualification needs", at OECD/CSNI & IRS[N] Workshop THICKET-2004 'Seminar on Transfer of Competence, Knowledge and Experience gained through CSNI activities in the field of Thermalhydraulics', vol. III, pp. 140-148, Saclay (F).

D'Auria F., Bousbia-Salah A., Petruzzi A., Del Nevo A., **2006**, "State of the Art in using Best Estimate Calculation Tools in Nuclear Technology", Nuclear Engineering and Technology, Vol.38 N.1, pages 11-32.

D'Auria F., Galassi G.M., **1998**, "Code Validation and Uncertainties in System Thermal-hydraulics". Prog. Nucl. En., 33, p.175.

D'Auria F., Chojnacki E., Glaeser H., Lage C., Wickett T., **1999**, "Overview of Uncertainty Issues and Methodologies". OECD /CSNI Seminar on Best Estimate Methods in Thermal Hydraulic Safety Analysis". NEA/CSNI/R(99) 10.

Damm G., Nabbi R., **2002**, "Status of HEU-LEU Conversion of FRJ-2", International Meeting on Reduced Enrichment for Research and Test Reactors RERTR-2002, Bariloche, Argentina, Nov. 3-8.

Di Maro B., Pierro F., Adorni M., Bousbia Salah A., D'Auria F., **2003**, "Safety analysis of loss of flow transients in a typical research reactor by Relap5/mod3.3", International Conference Nuclear Energy for New Europe, vol. 1, pp. 150-157, Portoroz (SLO).

Dietrich, J. R., **1954**, "Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled, Water-Moderated Reactor. Borax I Experiments, 1954", ANL-5323, (also listed as AECD-3668), Argonne National Laboratory, USA.

Dietrich, J. R., Layman, D. C. , February **1954**, "Transient and Steady State Characteristics of a Boiling Reactor. The Borax Experiments, 1953", ANL-5211 (also listed as AECD-3840), Argonne National Laboratory, USA.

Forbes S.G., Nyer W.E., **1961**, "Dynamic Properties of Heterogeneous Water Reactors", IDO- 16701.

Hamidouche T., Bousbia Salah A., Adorni M., and D'Auria F., May 16-20 **2005**, "Comparatve Study between RELAP5/3.2 system code and channel codes for the safety analysis OF research reactors", International Conference On Nuclear Engineering Beijing (China).

Hamidouche T., Bousbia-salah A, Adorni M., D'Auria F., **2004**. "Dynamic calculations of the IAEA safety MTR research reactor Benchmark problem using Relap5/3.2 code", Annals of Nuclear Energy vol. 31 pp. 1385-1402.

Hamidouche T., Bousbia-salah A., **2006**, "RELAP5/3.2 assessment against low pressure onset of flow instability in parallel heated channels", Annals of Nuclear Energy, Volume 33, Issue 6, Pages 510-520.

Hari S., Hassan Y., Tu J., **2000**, "Analysis of Transient Events without Scram in a Research Reactor Using the RELAP5/3.2 Computer Code", Nuclear Technology, Vol.130, pp.296-309.

IAEA, Safety Reports Series (SRS) no. 23, **2002**, "Accident Analysis for Nuclear Power Plants".

IAEA, TECDOC-233, **1980**, "Research Reactor Core Conversion from the use of high enriched uranium to the use of low enriched uranium fuels. Guidebook".

IAEA, TECDOC-643, **1992**, "Research Reactor Core Conversion Guidebook".

IAEA, Technical Report Series (TSR) no 446, **2006**, "Decommissioning of Research Reactors: evolution, state of the art, open issues".

Ivanov, K., Olson, A., Sartori, E., **2004**, "OECD/NRC BWR Turbine Trip Transient Benchmark as a Basis for Comprehensive Qualification and Studying Best-Estimate Coupled Codes, Nuclear Science and Engineering, 148 (2), 195-207.

Khedr A., Adorni M., D'Auria F., **2005**, "The effect of Code User and Boundary Conditions on RELAP calculations of MTR Re Search Reactor Transient Scenarios", Nuclear Technology & Radiation Protection, 1/2005, pp. 16-22.

Koncar B., Mavko, B., **2003**, "Modeling of low-pressure subcooled flow boiling using the RELAP5 code", Nuclear Engineering and Design, 220,(255-273).

Meftah B., Zidi T., Bousbia-Salah A., **2006**, "Neutron flux optimization in irradiation channels at NUR research reactor", Annals of Nuclear Energy, Volume 33, Issues 14-15, September- October 2006, pages 1164-1175.

Miller R. W., Sola Alain, McCardell R. K., June **1964**, "Report of the Spert I Destructive Test Program on an Aluminum, Plate-Type, Water-Moderated Reactor", US AEC Technical Report IDO-16883, Phillips Petroleum Co.

Nabbi R., Wolters J., **1992**, "Loss of Main Pumps at the Research Reactor FRJ-2 with delayed or without SCRAM", HDT, Thermal Hydraulics of Severe Accidents, Volume 192, pp 27-32, Book No H00750.

Nabbi R., Wolters J., **1995**, "Thermohydraulic Consequences of a Power Excursion resulting from a Hypothetical Reactivity Accident", AIChE Symposium series, Vol. 91, pp.179-188.

NEA, **1992**, "Probabilistic safety assessment: an analytical tool for assessing nuclear safety " (http://www.nea.fr/html/brief/brief-08.html).

ORAUT-TKBS-0007.2, **2003**, "Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL)".

Pierro F., Di Maro B., Adorni M., Bousbia Salah A., D'Auria F., **2004**, "Analysis of a total flow blockage of a fuel assembly in a typical MTR research reactor by Relap5/mod3.3", ASME Int. Conf. on Nuclear Engineering (ICONE-12), vol. 1, pp. 349-357, Arlington (VA, USA).

RELAP5/MOD3.3Beta, **2001**,. "RELAP5/MOD3.3Beta Code Manual volume II: Appendix A Input Requirements" ISL, Idaho, NUREG/CR-5535.

Stacy, S. M., **2000**, "Proving the Principle", http://www.inl.gov/proving-the-principle/, DOE/ID-10799.

Woodruff W.L., **1984**, "A kinetic and thermal-hydraulic capability for the analysis of research reactors". Nuclear Technology, Vol.64.

Woodruff W.L., Hanan N.A., Smith R.S., Matos J.E., **1996**, "A comparison of the PARET/ANL and RELAP5/Mod3.3 codes for the analysis of IAEA Benchmark transients". International Meeting on Reduced Enrichment for Research and Test Reactors, October 7–10, Seoul, South Korea.