Nuclear Science ISBN 92-64-02083-7

Neutronics/Thermal-hydraulics Coupling in LWR Technology, Vol. 1

CRISSUE-S – WP1: Data Requirements and Databases Needed for Transient Simulations and Qualification

5th EURATOM Framework Programme (1998-2002)

UNIVERSITÀ DI PISA

NEA No. 4452

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14th December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full member. NEA membership today consists of 28 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

FOREWORD

Controlled fission power has been utilised for electricity production worldwide in nuclear power plants (NPPs) based on light water reactor (LWR) technology for several decades. It has proven its efficiency and safety during these years and has manifested itself as a reliable and durable energy source. The foundation pillar in the peaceful utilisation of fission nuclear power has always been the strong emphasis on safety. Safety has been accomplished by continuously pursuing in-depth reviews and re-evaluation of safety-related issues incorporating findings from ongoing nuclear safety research activities worldwide. Specific requirements have been deployed at the design and in the permissible operation conditions of the NPPs in order to always ensure adequate margins against critical system conditions, thus preventing the occurrence of accidents. It is realised that as new findings and analysis capabilities become available safety will be increased, and it is further possible that the safety margins presently employed will eventually be relieved (decreased) without compromising the actual safety. Prevention and mitigation measures, however, must be properly balanced with cost-reduction needs. A thorough knowledge of fundamental issues – in the present case the interaction between neutronics and thermal-hydraulics – allows pursuing the goal of ensuring safety at reasonable costs.

Consistent with this goal, the CRISSUE-S project was created with the aim of re-evaluating fundamental technical issues for LWR technology. Specifically, the project seeks to address the interactions between neutron kinetics and thermal-hydraulics that affect neutron moderation and influence the accident performance of the NPPs. This is undertaken in the light of the advanced computational tools that are readily available to the scientific community today.

The CRISSUE-S activity deals with the control of fission power and the use of high burn-up fuel; these topics are part of the EC Work Programme as well as that of other international organisations such as the OECD/NEA and the IAEA. The problems of evaluating reactivity-induced accident (RIA) consequences and eventually deciding the possibility of NPP prolongation must be addressed and resolved. RIAs constitute one of the most important of the "less-resolved" safety issues, and treating this problem may have significantly positive financial, social and environmental impacts. Public acceptance of nuclear technology implies that problems such as these be satisfactorily resolved.

Cross-disciplinary interaction (regulators, industry, utilities and research bodies) and co-operation within CRISSUE-S provides results which can directly and immediately be beneficial to EU industry. Concerning co-operation at an international level, the participation of the EU, former Eastern European countries, the USA, and observers from Japan testify to the wide interest these problems engender. Competencies in broad areas such as thermal-hydraulics, neutronics and fuel, overall system design and reactor surveillance are needed to address the problems that are posed here. Excellent expertise is available in specific areas, while limited knowledge exists in the interface zones of those areas, *e.g.* in the coupling between thermal-hydraulics and neutronics. In general terms, the activities carried out and described here aim at exploiting available expertise and findings and gathering together expert scientists from various areas relevant to the issues addressed.

Added value for the CRISSUE-S activity consists of proposing and making available a list of transients to be analysed by coupled neutron kinetics/thermal-hydraulic techniques and of defining "acceptability" (or required precision) thresholds for the results of the analyses. The list of transients is specific to the different NPP types such as PWR, BWR and VVER. The acceptability thresholds for calculation precision are general in nature and are applicable to all LWRs. The creation of a database including the main results from coupled 3-D neutron kinetics/thermal-hydraulic calculations and their analysis should also be noted.

The CRISSUE-S project is organised into three work packages (WPs). The first WP includes activities related to obtaining and documenting relevant data. The second WP is responsible for the state-of-the-art report (SOAR), while the third WP concerns the evaluation of the findings from the SOAR and includes outcomes of the entire project formulated as recommendations, mainly to the nuclear power industry and to regulatory authorities. The present report is the result of the first WP and discusses the type of transients that are of interest in relation to reactivity-initiated accidents in LWR NPPs and elaborates on the data required for coupled 3-D neutron kinetics/thermal-hydraulic analysis as well as the data needed to perform associated validations.

A comprehensive report such as the present one, composed of contributions from the different CRISSUE-S participating organisations, unavoidably implies non-homogeneous treatment of the various topics, although an effort was made to provide consistency between the various sections. However, it is realised that the adopted level of detail is not commensurate with the safety relevance or the technological importance of the issues discussed.

The report has been written to accomplish the objectives established in the contract between the EU and its partners. Expected beneficiaries include institutions and organisations involved with nuclear technology (*e.g.* utilities, regulators, research, fuel industry). In addition, specific expected beneficiaries are junior- or senior-level researchers and technologists working in the considered field of research and development and application of coupled neutron kinetics/thermal-hydraulics.

Six plenary CRISSUE-S meetings took place over the course of the project implementation period. The meetings were held at:

- University of Pisa, Pisa, Italy, 25-26 February 2002 (kick-off meeting).
- OECD/NEA, Issy-les-Moulineaux, Paris, France, 5-6 September 2002.
- Technical University of Catalonia (UPC), Barcelona, Spain, 23-24 January 2003.
- SKI, Stockholm, Sweden, 26-27 June 2003.
- European Commission, Luxembourg, 12 November 2003 (status information meeting).
- University of Pisa, Pisa, Italy, 11-12 December 2003 (final meeting).

An Internet site has been established at the University of Pisa and has been kept alive during the project lifetime (2001-2003). The address is www.ing.unipi.it/crissue_s. The site also contains the discussion records of the six meetings.

The importance of the CRISSUE-S project has been expressed by the OECD/NEA Nuclear Science Committee. This interest has also been emphasised by the OECD/NEA Committee on the Safety of Nuclear Installations, as the project discusses many of their activities. It was agreed that the CRISSUE-S reports be published by the OECD/NEA as its contribution to the project.

This report was produced by the members of the CRISSUE-S project for use within their organisations. The present version is being made widely available for the greater benefit of organisations and experts working in the nuclear power area.

Acknowledgements

Many thanks to Nicola D'Amico (ITER Consult, Rome) for his review of the report.

This report is dedicated to the memory of Gianni Frescura, who as Head of the Nuclear Safety Division of the Nuclear Energy Agency provided strong support for this activity and arranged for the co-operation between the OECD/NEA and this EC Project to be an effective one.

TABLE OF CONTENTS

LIST OF CONTRIBUTORS

University of Pisa, Italy – Project Co-ordinator
F. D’Auria, A. Bousbia Salah, G.M. Galassi, J. Vedovi
ANAV-UPC, Spain
F. Reventós, A. Cuadra, J.L. Gago
Studsvik Eco & Safety AB (later Studsvik Nuclear AB), Sweden
A. Sjöberg, M. Yitbarek
SKI, Sweden
O. Sandervåg, N. Garis
University of Madrid, Spain
C. Anhert, J.M. Aragonés
University of Valencia, Spain
G. Verdù, R. Mirò
NRI, Czech Republic
J. Hadek, J. Macek
Pennsylvania State University, USA
K. Ivanov
University of Illinois at Urbana-Champaign, USA
R. Uddin
OECD/NEA, Paris, France
E. Sartori
VALCO Project Co-ordinator at FZR, Germany
U. Rindelhardt, U. Rohde
Royal Institute of Technology in Stockholm, Sweden
W. Frid
Westinghouse Electric Sweden AB, Sweden
D. Panayotov

EXECUTIVE SUMMARY

The CRISSUE-S project

The CRISSUE-S project aims to re-evaluate fundamental technical issues in the technology of LWRs related to the interaction between thermal-hydraulics and neutron kinetics, including aspects related to neutron moderation and associated implications as concerns NPP accident performance. This is undertaken in light of the advanced computational tools that are available to the nuclear energy community today.

The objectives of the CRISSUE-S project can be summarised as follows.

- To establish a state-of-the-art report (SOAR) on the subject.
- To provide results of best-estimate analyses of complex transients in existing reactors.
- To provide recommendations to interested organisations.
- To identify areas of the NPP design for which the design/safety requirements can be relaxed.

The SOAR summarises among other things the findings from recent international projects or activities (*e.g.* OECD/CSNI SOAR on BWR stability, IAEA workshops and TECDOCs, recent OECD/NEA/NSC benchmarks including the PWR-MSLB and the BWR-TT, and boiling stability activities in the EU IV Framework). Within this context, the results of a co-operation established with the EU VALCO project are discussed and consideration is given to the activities in progress within the EU NACUSP project. Selection of relevant transients from proper PSA studies and consideration of the needed ensemble of tools and databases including computer codes, NPP nodalisations (input decks), and realistic boundary and initial conditions, are addressed in the SOAR. Additional topics deal with the determination of criteria for using the mentioned tools and databases including an associated qualification process and uncertainty evaluation.

The establishment of the SOAR implies the availability of information concerning the system performance should relevant selected accidents occur. Therefore, NPP transients have been analysed and relevant results considered in the SOAR along with qualified results available from different sources. The accident scenarios are typically identified as MSLB, ATWS, LOFW, CR ejection, DW stability, MSIV closure and TT. The relevance of three-dimensional (3-D) neutronics/thermal-hydraulics coupling upon the prediction of a LBLOCA scenario is also evaluated. In relation to the mentioned transients, a CRISSUE-S database has been created.

The upper limits of “core survival” are obtained from the analyses, and elaboration on the thermal-hydraulic and neutron kinetics conditions of the core (including fuel types and burn-up) during the course of specific transients will provide information on actual safety margins. Parameters related to rod surface temperature and fission power are considered for the evaluation of the margins. The maximum tolerable fuel burn-up is evaluated in reference cases. The validity of the emergency operating procedures (EOPs) implemented can be challenged and recommendations are given for accident management in situations possibly outside the scope of actual design. Thus potential benefits of the activity also include consideration of NPP core “extreme” conditions. Two examples provide a general impression of the results achieved:

- In the case of PWR LBLOCA-DBA, the “physical” scram occurrence (due to core voiding) has been demonstrated to occur well before (in terms of time after the “break opening”) control rod insertion. In practical terms, this constitutes one of the “green lights” on the road to raising the rated power of the NPP.

- The conservatism of results from the current approach (including calculations with 0-D or point neutron kinetics models) has been demonstrated in the latter case. A more comprehensive analysis, including full 3-D core thermal-hydraulic and neutron kinetics models, reveals reduced power peaking factors which in turn result in associated lower PCT. Again, this constitutes one of the “green lights” to possibly relax some of the current requirements, *e.g.* the boron concentration in various tanks installed in the NPP, or to allow a more aggressive and optimised core fuel loading thus engendering greater fuel efficiency.

The present report (WP1)

The present report constitutes the first report in a series of three which will be issued within the CRISSUE-S framework. This report deals with both the identification of NPP transients that can be expected to benefit from being re-evaluated using the coupled thermal-hydraulic/neutronic techniques, and with the required “input” data needed for the analyses. In relation to the input data, consideration is also given to NPP transients useful for the qualification of the coupled techniques. The second report constitutes the SOAR itself, while the third report deals with recommendations that can be beneficial mainly to the industry and to regulators.

In relation to the current report one main topic is the discussion of probabilistic safety assessment (PSA) and of the associated outcome in relation to NPP transients revealing substantial core reactivity changes, including reactivity-initiated accidents (RIA). Another topic concerns modelling and related capabilities in the areas of thermal-hydraulics and 3-D neutron kinetics, highlighting needed input and specific modelling considerations. Some emphasis has also been placed upon transients measured in nuclear power plants, as a means of complementing the database of results that was created and is described in Annex I of the WP2 report. More specifically, the WP1 report includes and discusses the areas described below.

Probabilistic safety assessment (PSA)

An overview of PSA Level 1 concepts is provided followed by a short review of current PSA studies with a focus on transients, including substantial core reactivity changes. It is noted that the PSA has not yet focused attention explicitly towards such transients, although time periods of reactivity change could be part of the transients discussed. However, it is also noted that current PSA applications indicate that the methodologies involved are used in an increasingly extensive perspective. Thus newer applications also include, apart from traditional PSA, core damage frequency estimations, evaluations of plant modifications, etc., and are envisaged to provide a basis for the process of possible relaxation of current technical specifications. Other important areas are the optimisation of component inspection and maintenance, safety management activities, and risk-informed decision-making and regulation. It is realised that, in the long term, the latter can possibly result in a redefinition of the limiting pipe break size to be considered in LOCA-DBA analyses.

In conclusion, in general terms it seems difficult to obtain an evaluation and ranking of various reactivity transients from a frequency of occurrence and consequence point of view from current PSA studies. Thus only a limited amount of information is available that can be used to guide the direction and scope of deterministic coupled T-H/neutronic revaluation analysis of such transients based on some merits of benefit and importance. However, as more experiences are gained from, for instance, recent PSA activities related to risk-informed decision-making and regulation, this situation is expected to evolve.

A number of transients related to PWRs, BWRS and WWERs are listed and discussed with reference to their core reactivity behaviour. The transients mentioned are known through experience gained from events which have occurred in NPPs and/or deterministic thermal-hydraulic analyses including more simplistic kinetics modelling, to reveal substantial core reactivity changes. In relation to these transients, the transients included in the CRISSUE-S database (being part of the WP2) are also mentioned.

Thermal-hydraulic system simulation

System thermal-hydraulic (T-H) codes based upon six partial differential balance equations, *i.e.* mass, momentum and energy for each of the two phases, solved in 1-D geometry, constitute the current state of the art. Specific techniques are available to construct equivalent ("fictitious") 3-D nodalisations for the core or the vessel regions as needed. Simplifications are introduced to make the equations more tractable and help with the process of implementing the equations in a numerical solution framework. Constitutive relationships and models determine the evolution of the two-phase mixture, *e.g.* the prediction of flow regimes. These models, being mostly of empirical nature, include simplifications and the normal practices in applying the models often result in extrapolation outside the original validity ranges, thus providing uncertainties in calculational results. The numerical solution methods used also influence the results to an extent not fully quantified. This uncounted uncertainty is expected to be further evaluated and more thoroughly quantified in the future when results and conclusions from more validation analyses for specified types of transients are at hand.

Some specific areas are discussed concerning the preparation of the T-H system input models that can be of particular importance with regard to the T-H/neutronic analysis. Issues related to nodalisation are also considered. The discussion is concentrated on T-H phenomena and process models that could have a specific influence on the kinetics responses, including control and trip systems, and thus have to be considered at the NPP model set-up. References are made to PWR, BWR and WWER modelling aspects.

Three-dimensional neutronic and fuel simulation

Similarly, some areas are discussed related to the preparation of the kinetics models and their input decks that can have importance with regard to the coupled T-H/neutronic analysis. This includes both nuclear cross-sections and associated issues (history and instantaneous effects, and assembly discontinuity factors) and data related to the neutron diffusion code calculations.

It is noted that even though approximate diffusion equations are typically adopted in the available neutron kinetics computational tools, reference solutions exist that are obtained with more sophisticated techniques, including Monte Carlo as well as detailed and higher-order deterministic neutron transport. These solutions are used to benchmark the results from codes utilising the diffusion equations. Thus the problem of uncertainty evaluation seems more manageable and the numerical solution methods play a greater role upon the calculation accuracy than in the case of thermal-hydraulics. It is realised that the details of the fuel pins, including extended transient characteristics, are not currently modelled and consequently the effect from changes in the fuel pin geometrical properties are not accounted for.

Coupling of thermal-hydraulic system and 3-D neutronic codes

The coupling issue is discussed with reference to T-H/3-D neutronics code packages, *e.g.* the RELAP5/PARCS and ATHLET/DYN3D packages. Aspects related to PWR (WWER) and BWR coupled analyses are mentioned. Specific consideration is necessary as concerns the core nodalisation overlay or mapping between the core models in the T-H system and neutronic codes respectively, a subject which is discussed in some detail.

NPP modelling data and transient data for validation purposes

An overview is provided of the types of data that are needed for preparing a plant model to be used for coupled T-H/3-D neutronics calculations, with reference to several international code benchmark projects. The data needed are listed and also include items not directly specific to the applications of coupled codes but are general to any separate T-H system NPP modelling and core kinetics modelling.

Discussions are given on what requirements should be met in relation to plant measured data and the utilisation of such data for code validation purposes. A minimum set-up of suitable measured data is proposed. The characteristics of measured data in form of signals from the plant on-line data monitoring system are also discussed. It is noted that the measured data need to be properly recorded during an ample pre-transient steady-state time period to allow for examination of data consistency and adequacy. During the transient time period the dynamic characteristics of the signals must be fully described and understood in terms of possible time delays and applied filtering. Another important aspect concerns whether there are any specific process influences on the gauges' readings due, for instance, to the locations in the plant components.

Some account on measurement sampling frequency is given. It is realised that the frequency must be sufficiently high so as to reveal important dynamic aspects during the course of the transient event, but on the other hand not prohibitively high, leading to the production of an exceptionally large amount of data. In this respect the format used when saving and storing data is also mentioned.

Data added to the CRISSUE-S database

Additional NPP data to be included in the CRISSUE-S database has been provided within the framework of the WP1. This complements the result database that is part of the contents of the second CRISSUE-S report (WP2, the SOAR itself). This database constitutes the Annex 1 of the WP2 report and includes results of coupled 3-D neutron kinetics thermal-hydraulic calculations performed to predict transient scenarios in BWR, PWR, WWER-440 and WWER-1000 NPPs. The considered transients are part of the ensemble of transients recommended for safety analyses.

The additional data, provided in form of data files, concern transients which have occurred in the Ascó (PWR), Oskarshamn 2 (BWR), Peach Bottom 2 (BWR) and Three Mile Island 1 (PWR) NPPs. The information provided for each transient includes plant and transient description, and associated T-H and neutronic data.

Chapter 1

INTRODUCTION

The acronym for the CRISSUE-S project, pursued within the EU 5th Framework, originated from Critical Issues in Nuclear Reactor Technology: A State-of-the-art Report. The idea of the project is linked with the existence of areas of LWR technology for which a lack of comprehensive understanding mandated the putting into place of wide safety margins; this is the time period during which the current generation of NPPs was designed. Those safety margins set limits on the operation of the nuclear power plants, thus causing increases in the cost of electricity production, and have also left islands in the sea of knowledge that must be flooded. The recent availability of powerful computers and their improved calculational abilities together with the continuing accumulation of operational experience motivates the initiation of revisiting activities within the associated areas. Benefits are foreseen in terms of reducing operational costs, improving the common understanding of nuclear safety and of design/operating conditions and, definitely, establishing a basis for advancing the technology.

The subject of the CRISSUE-S activity is the interaction between three-dimensional neutron kinetics and system thermal-hydraulics. This is relevant for both the safety and the design/operation of existing reactors. Therefore, regulatory authorities and industry may benefit from the main findings. An impact is also expected upon the design of innovative water-cooled reactors.

1.1 Bases of the subject and motivation for CRISSUE-S

The nuclear safety issue in relation to the operation of a NPP can be formulated in terms of ability to contain any radioactivity within the plant under any conceivable situation and, further, to exclude the possibility of any harmful impact on the plant personnel. The risk of releasing radioactivity to the environment has been minimised by using a "defence in depth" design philosophy with several barriers to maintain the radioactive material within, firstly, the fuel rods themselves, then fuel cladding, reactor pressure vessel and, finally, within the reactor containment. An important resulting aspect is to maintain the integrity of the core fuel rods, which is closely associated with the avoidance of fuel rod overheating, of excess rod power, and power rate of change (power time derivative) transients. The evaluation of prevailing margins against those effects are obtained by means of detailed thermal-hydraulic (T-H) transient analyses of specified bounding sequences (identified worst cases), of which cases include boundary conditions based on conservative assumptions to minimise fuel coolability and/or maximise fuel power excursions. These types of T-H analyses are performed with advanced computer codes including phenomena-based models (mostly empirical) and have traditionally been made for plant licensing purposes with conservative assumptions applied for critical models. The trend nowadays tends to be more geared towards performing the analyses with a best-estimate (BE) approach, meaning that the T-H phenomena are simulated as accurately as possible (according to present knowledge). If BE analyses are used for plant licensing purposes the nuclear safety regulatory requirements set forth that the analyses have to be accompanied by uncertainty evaluation to reveal the uncertainty bands related to the calculated parameters.

In T-H analyses the prevailing power in the fuel rods is usually explicitly specified, for instance by providing time-dependent functions. It can also be obtained from simplified kinetics models simulating in most cases only the transient overall core reactivity feedback responses (considering the core as a “point” or point kinetics) or at best the core transient axial responses [one-dimensional (1-D) kinetics]. Consequently, by employing these methodologies the simulation of a detailed core power distribution is not possible and an approach using static power factors has been established. However, this practice requires the application of additional conservatisms when specifying those power factors, as they also have to include uncertainties due to the loss in resolution of the transient true spatial power distribution. Thus conservative and unfavourable core power distributions are applied in the T-H analysis to ensure the fuel rods will experience more severe conditions than might be expected to prevail in the real counterpart case.

As another branch, separate from the basic T-H system analysis area, advanced computer codes have been developed for the transient 3-D simulation of nuclear kinetics responses of a nuclear reactor core exposed to various reactivity perturbations. These perturbations can have their origin in control rod movements or changes in the T-H conditions of the core coolant, including changes in the concentration of soluble boron in the liquid phase. The associated power variations can include both global core-wide changes but also local changes restricted to only a few fuel assemblies. The original core power level and operational history (burn-up) have profound influences on the power responses. The codes use nuclear cross-section tables evaluated on a detailed level, providing neutron energy dependent probabilities for specific nuclear reactions to occur. Those tables are used as a basis for determining the core state which includes allowances for burnable absorbers and for flux heterogeneities at fuel assembly borders and core outer regions (reflector sections). The needed core T-H conditions have mostly been obtained from quite simple internal T-H models simulating only the conditions in the core itself, with the need for adequately specified boundary conditions at core inlet and outlet sides, or at best from simple models simulating the RPV internal flow paths.

Traditionally the T-H codes and the nuclear kinetics codes were developed to pursue different objectives and with little or no common connections. However, with recent computer developments resulting in the availability of powerful computation capabilities at reasonable costs, the interconnection between the two disciplines has become feasible. It is now possible to perform detailed dynamic T-H reactor system analysis together with coupled detailed dynamic three-dimensional (3-D) core kinetics simulation even on a readily-available PC system. This type of detailed T-H/neutronics overall simulation capability of transients in LWR NPPs will provide a basis to undertake a more in-depth evaluation of the safety margins found in previous (licensing) T-H simulations for which a point kinetics model or a 1-D model was used. Thus provisions are now at hand to accurately reveal the actual safety margins, which could provide incentives to more efficiently utilise the fuel and obtain cost benefits in the operation of the nuclear power plants while still preserving – and possibly even improving – safety. In addition, this type of coupled T-H/neutronics analyses could provide more detailed insight in the basis used at the specification of operational procedures and could provide guidelines for optimisations of EOPs. On the other hand, coupled T-H/neutronics analyses require detailed and elaborated plant and core specific inputs to reflect important plant and core design features and to represent prevailing T-H and core kinetics operational states. The latter also includes appropriate consideration of the core state changes during past operation periods (power history and burn-up effects).

As mentioned above, the main objective of the CRISSUE-S project is to produce a state-of-the-art report related to ongoing activities in the re-evaluation of basic T-H-kinetics interaction issues with the new type of analysis capabilities based on coupled T-H/neutronics models. The project has been separated into three Work Packages (WPs) and the complete project is reported through three separate documents, one for each WP. The present report constitutes the first of the three; it deals with the required “input” database for the analyses and includes the identification of NPP transients useful for the qualification of the coupled techniques. The second report constitutes the SOAR itself [1] while the third report deals with recommendations, mainly to the nuclear industry and to the regulators, and summarises the outcome of the entire project [2].

The three WPs of the project have briefly the following contents [3]:

- *WP1:*

  – Identification and characterisation of “bounding transients” for each type of NPP concerned. This information is obtained from probabilistic safety assessment (PSA) studies, and includes some evaluation of associated qualification level. Distinction is made between the common LWR NPP types currently in operation worldwide, *i.e.* PWR, BWR and WWER.

  – In relation to “bounding transients” in PWRs, BWRs and WWERs the results of coupled T-H/neutronic analyses are considered. Some discussions are provided on the nodalisation and model qualification level as well as on the evaluations of associated uncertainty studies.

  – A database is generated suitable for future use by organisations other than those involved in the CRISSUE-S project.

- *WP2:*

  – Responsible for issuing the CRISSUE-S SOAR. Attention is focused on the “bounding transients”. Emphasis is placed on findings in WP1 and to the related activities recently completed or in progress at IAEA, OECD/CSNI/NEA and EU.

- *WP3:*

  – Evaluation of the results and achievements of common understanding. This includes conclusion of the CRISSUE-S activities and recommendations. Main subjects comprise discussions of safety margins in existing reactors, realism and risks as concerns NPP life prolongation, and optimised management of RIA events.

In addition to the general aspects of the contents of WP1 indicated above, a more specific objective is to concentrate on data used in relation to coupled T-H/neutronics analyses. To better address the contents of the WP1 report additional directions have been set forth during the course of the project [4]:

- The WP1 report is complementary to the WP2 report (*i.e.* to the SOAR) without reporting or addressing the same information.

- The WP1 report should benefit from PSA-related information gathered within the project and also discusses data needed to perform T-H/neutronics analyses and to validate the results.

- The needed data are related to T-H data and nodalisation, and neutronic data including cross-sections. Data to be used for validation and additional needs in the area are stressed.

The PSA information can provide, in a general sense, indications of the type of transients that can be regarded as “critical” in the safety context. This is a selection basically directed towards conditions where various degrees of fuel damage can be expected. For the purpose of the CRISSUE-S project emphasis is placed upon those transients for which coupled BE T-H/neutronic revaluation analyses can be expected to provide beneficial outcomes in terms of revealing utilisation of (overly) conservative safety margins. Transients of this kind include time sequences where substantial reactivity changes occur, of which RIAs constitute an essential part.

Based on this overall outline of the content of the WP1 report, the following structure of the report has been established:

- Chapter 1 includes the introduction (this section) with some general background information.

- Chapter 2 of the report discusses PSA results in relation to transients with substantial reactivity changes and elaborates on the different transient scenarios.

- Chapter 3 discusses the T-H system modelling and important data required when preparing the T-H models for coupled T-H/neutronics analyses and also provides some account of nodalisation aspects.

- Chapter 4 discusses data needed, including nuclear cross-section data, when preparing the kinetics models for the coupled T-H/neutronics analyses and will also give some account of associated nodalisation aspects.

- Chapter 5 discusses the coupling of the T-H system and the neutronic codes, including aspects concerning consistent core nodalisations and node mapping between the codes.

- Chapter 6 elaborates on experimental or measured data needed to be able to qualify the results from coupled T-H/neutronics analyses. Considerations are given to types of data needed and to requirements concerning data quality. Some specific NPP data are discussed and constitute additions to the CRISSUE-S database.

- Finally, Chapter 7 summarises the main conclusions.

Chapter 2

PROBABILISTIC SAFETY ASSESSMENT

Safety analysis of nuclear reactor systems is usually performed based on two different approaches characterised as deterministic safety analysis (symptom oriented) and probabilistic safety analysis. Although having quite different objectives they complement each other to ensure that, overall, a well-based and consistent safety philosophy has been thoroughly applied. Moreover, recent developments indicate a trend to perform safety analysis wherein a tighter interaction between the two approaches is pursued [5].

As part of a plant's commissioning deterministic T-H licensing analyses must be performed to determine the safety of the plant at various operating and accident conditions. As it is not feasible to perform analyses of all conceivable types of transients under the influence of a variety of boundary conditions, identified "bounding transients" have to be analysed to envelope the vast amount of possible transient combinations. The selection of those transients is based on the assumption that despite measures taken to avoid accidents during the operation of the NPPs, such accidents may still occur. Therefore, the auxiliary systems must be designed to properly mitigate any possible consequences. The transients are initiated by a specific event or combination of events and are often combined with certain restrictions and prerequisites concerning the availability of important mitigative auxiliary systems with the intention of obtaining conservative results for worst case scenarios which most challenge core integrity and safety. The common understanding is that if these types of transients can be handled adequately by the plant system without compromising specified safety criteria, it is probable that any transient embraced by those being analysed can also be adequately handled. The transients to be analysed include several types of LOCAs and other events leading to imbalances between the core power produced and heat removal. Some of these transients comprise the group Design Basis Accident (DBA) which is a category of very improbable events postulated as a basis for the design of various safety systems. Transients which can pose a direct challenge to the mechanical integrity of the core fuel rods are also analysed, *e.g.* transients including pronounced reactivity (power) changes that can result in damage caused by fuel pellet-cladding interaction (PCI) phenomena.

By using a more global perspective possible transients as well as associated initial events can be categorised, including the probability of the sequences to occur. The basis for quantifying the probabilities is obtained from in-depth analysis of the availability of various systems and subsystems including statistical analysis of basic availability data. The interrelationships between systems and subsystems can be evaluated to provide information on common-cause failure modes which can substantially contribute to the overall probability to reach a specific condition. All together these steps comprise the probabilistic safety assessment (PSA) process. PSA studies provide the frequency of an event to occur per year of reactor operation. PSA, also known as probabilistic risk analysis (PRA) in the USA, thus focuses on the identification of event sequences which can lead to a specific end condition, this being normally and historically the damage or melting of the core, but can in principle be any prescribed end condition. This type of analysis, including evaluation of the reliability of various safety systems, can provide results helping indicate weak points in the overall safety design and thus can be used as a basis for improving the safety of the NPP.

Current PSA applications indicate that the methodologies involved are used in an increasingly extensive perspective. Thus applications also include, apart from “traditional” core damage frequency estimations, evaluations of plant modifications (provide ranking based on risk impact) and can help in the process of possible relaxation of the technical specifications used. Other important areas are optimisation of component inspection and maintenance, safety management activities, and risk-informed decision making and regulation. Especially the latter can potentially (in the long term) result in redefinition of the limiting pipe break size to be considered in LOCA-DBA analyses [6].

As mentioned earlier, PSA traditionally concentrates on events that can lead to degraded core conditions. The complete PSA for degraded core conditions comprises three levels, with each successive level incorporating a more comprehensive analysis than the previous level [7]. The first level, PSA Level 1, focuses on the estimation of core damage frequency expressed as a probability of core damage per year of reactor operation. The second level, PSA Level 2, includes Level 1 analysis and in addition a study of the physical processes during the core melt to quantify the amount of various radioactive compounds that can be released from the core (source term). The third level, PSA Level 3, includes Level 2 analysis and examines the dispersion of radioactive substances in the environment and of associated possible consequences to life, health and property.

During the course of reaching a prescribed end condition (certain degree of core damage) time periods including substantial core reactivity changes could occur. These reactivity changes could by themselves be major contributors to the core damage in that they can severely influence the transient coolability of the fuel pins as well as their mechanical integrity. Additionally, in cases where no core damage occurs, large short-term variations of reactivity (local core power) can have a very unfavourable influence on the fuel pin material and thus on the fuel pin ability to withstand future transient conditions. When evaluating the severity of a reactivity variation in relation to associated fuel design limits, guidelines developed for the situation of having the fuel exposed to excessive power escalation can be used. The specification of this type of condition could be in accordance with the RIA fuel failure limit based on the definition in Section 4.2 of the US Standard Review Plan [8], which is used in several countries. This provides a limit on maximum radially averaged fuel enthalpy of 170 cal/g for BWRs and a limit defined as a DNB criterion for PWRs. The trend of extending fuel utilisation to higher burn-ups (approximately 50 MWd/kg and higher) can possibly alter these limits [9]. Thus, for instance, recent research indicates that the fuel rod failure enthalpy for fuel pellet-cladding metallic interaction decreases with burn-up (and oxide layer growth) [10]. It is also noted that other phenomena apart from high burn-up contribute to the risk of fuel failure at lower fuel enthalpy. In Ref. [10] mention is made of the oxide build-up on the fuel rod surfaces, the hydrogen content of the Zr cladding and of the local hydriding. Newer cladding materials, however, can provide improved characteristics. An elaborate review and discussion of fuel-related issues can be found in Chapters 3 and 4 of WP2 [1].

In order to classify and find a probability for a certain event sequence including time periods of RIA behaviour it is adequate to make an associated PSA Level 1 analysis. This analysis is based on a systematic reliability evaluation of systems and components whose level of operational functionality will have an essential impact on the transient response during the course of the event. As is outlined in Ref. [11] this analysis includes several prerequisites such as acquiring an in-depth understanding of the NPP system and collecting pertinent information, identifying initiating events and states, setting up fault tree models of the system to reveal performance and system interactions, and develop databases of component reliability data. Thus it is clear that PSAs are concerned only with specific sequences in a given plant and are thus plant specific.

2.1 General event classification

Obviously it is not feasible to analyse all conceivable types of events in a plant. Alternatively, the events may be classified according to the expected probability of their occurrences, expressed as an occurrence frequency per year of reactor operation, and associated consequences on the plant overall safety features. The event classification is usually based on recommendations given in ANS 51.1 and 52.1 [12], and is made according to the following:

Event	Frequency (per year)	Category
Normal operation		1
Small disturbances during normal operation and controlled by the plant's control systems without interruption of operations	$> 10^{-1}$	2
Anticipated, moderately frequent events which may result in safety chain actuation	10^{-2}-10^{-1}	3
Anticipated, infrequent events resulting in safety chain actuation	10^{-4}-10^{-2}	4
Improbable events postulated for safety system design	10^{-6}-10^{-4}	5
Very improbable events not included in the design bases	$< 10^{-6}$	6

Modifications of this classification are adopted. For instance, for the Swedish BWRs a somewhat modified event classification is used with categories H1 through H5, where category H2 is a combination of categories 2 and 3 in the above table, and subsequently H3 corresponds to category 4 and so forth.

Within each category acceptance criteria have been established as to what consequences should be assumed. The basis is an overall risk evaluation where the risk within a category, *i.e.* the product of probability of an event to occur and the associated consequence, should be constant. Thus an event occurring with a higher probability (occurring more often) must reveal smaller consequences, while larger consequences could be accepted for events with a lower probability to occur.

Examples of events within the categories 2 through 6 are provided in the table below.

Category	Event
2-3	Trip of circulation pump(s) (BWR) Malfunction of level control (BWR) Load rejection Turbine trip Uncontrolled boron injection (PWR) Inadvertent reactor isolation (BWR)
4	Small LOCA Loss of reactor coolant pumps (PWR) Reactor isolation with loss of off-site power (BWR)
5	Main recirculation line break (DBA-LOCA) Main steam line break (PWR) Reactor isolation without scram (BWR)
6	Reactor vessel rupture LOCA without emergency core cooling Transients without reactor shutdown

Most of the events in categories 2-3 result in reactor trip. The events in category 5 are design base accidents (DBA) postulated for safety system design.

The operation of any nuclear reactor requires means for core reactivity control, over both the long- and short-term time frames. For instance, will the start-up procedure involve rather slow, gradual changes of various core state parameters from cold shutdown conditions up to hot full power operation? In this case a change in reactivity is very slow and can easily be controlled. On the other hand several transients can be perceived which include substantial core reactivity changes and which can pose a challenge to the core reactivity control systems. The characteristics of those types of reactivity changes encompass pronounced temporal and spatial gradients. These are caused by fast changes in the core T-H conditions, including changes in boron concentration, but can also result from fast ejection/insertion of control rods. In the latter case, due to statistical considerations, the effect from only a single control rod with high reactivity worth is usually taken into account, although the effects from several control rods and even a control rod bank could be perceived (very low probability to occur).

In all the event categories time periods can be found which include substantial temporal reactivity insertion (positive or negative) into the core and associated power variation with mostly limited duration, but nevertheless can result in approaching specified core integrity limits. Examples are uncontrolled boron injection (PWR) and inadvertent reactor isolation (BWR) in category 2-3, and main steam line break (PWR) and reactor isolation without scram (BWR) in category 5.

In order to limit the following discussion of current PSA treatment of specific events which include periods of reactivity changes, some specific scenarios have been selected. Those basically include such transients which from previous traditional deterministic T-H safety analysis have been shown to reveal RIA or RIA similar characteristics. It is envisaged that the calculational results for any of the mentioned transients can prove beneficial from being re-evaluated using the new T-H/neutronics codes. Especially the more detailed 3-D resolution of the neutron kinetics condition along with adequate T-H core modelling can reveal possibilities to relax the influence of core power distributions on the associated conditions for the fuel pins. Consequently fuel loadings with more favourable (in economical terms) and peaked power distributions could be used while still preserving specified safety margins thus being able to utilise the fuel more efficiently.

2.2 Relevant PWR transients

Operational PWRs are currently of various designs, including the apparently very different configuration groups with UTSG and OTSG. Within one group substantial differences also exist, *e.g.* different core nominal power and number of loops, SGs with and without pre-heater sections, different design of the SGs resulting in variation in the sensitiveness of downcomer liquid level changes, etc. As mentioned earlier this calls for plant-specific PSA in order to identify and rank the transients from a severity perspective. However, an attempt is made below to identify in a more general sense various types of transients that include time windows where pronounced reactivity variation occurs. A short phenomenological description of each event is also provided. Consideration has been given, when selecting transient scenarios, to the recently issued IAEA guideline [13].

The information below is generally based on the treatment of RIA-related transients in the PSA for the Spanish PWR plants Ascó and Vandellòs (Westinghouse three-loop PWRs) but is assumed to be generally applicable.

Following what has been established related to the PSA in the NPPs Ascó and Vandellòs, every transient listed as a “scenario which could pose fuel integrity problems originated from substantial core reactivity changes”, has been internally discussed within expert groups of the NPP utilities where two different aspects have been evaluated:

- The mechanism from the initiating event until reactor protection actuation.

- The classification of the transient.

The discussion has been performed starting from the following list of transients, which are known from deterministic T-H analyses to involve substantial reactivity changes:

- Decrease in primary coolant flow temperature.

- Increase in primary coolant flow.

- Control rod ejection.

- Boron dilution.

- Bent fuel assemblies.

- Main steam line break (MSLB).

- Anticipated transient without scram (ATWS).

An overall discussion on each of these transients is provided below. It should be noted that other conceivable transients can be listed, including loss of feed water flow (LOFW), large break LOCA (LBLOCA, DBA for PWR ECCS) and small break LOCA (SBLOCA). Each of these transients can have specific core reactivity consequences that must be evaluated:

The LOFW is caused by blockage of the FW pumps, which can originate from a "partial station blackout". The primary loop temperature increases and moderator and Doppler neutron feedback contribute to core power decrease. An added requirement of unavailability of scram system (*i.e.* an ATWS condition) can be applied in the analysis and can be justified by considering the (relatively high) accident frequency.

The LBLOCA is originated by the double-ended guillotine break of one CL at a location between the reactor coolant pump (RCP) and the reactor pressure vessel (RPV). The accident constitutes a pillar in the safety demonstration and in the licensing of any LWR with main reference to the evaluation of the ECCS design (*i.e.* a DBA for the ECCS) and, for this reason, is part of the "official" NPP FSAR. The need for 3-D T-H neutron kinetics analysis is mostly related to the quantification of the conservatism introduced by the "highly conservative" core linear power form factors that result in high values of the PCT. The possibility to relax the conservatism can be revealed, thus allowing for more demanding core loadings (using the fuel more efficiently) while still maintaining – or possibly increasing – the safety. A specific LBLOCA-RIA occurs in the case of positive moderator temperature coefficient, which in principle is possible at BOC with high boron concentration.

The SBLOCA is a TMI-type accident originated by a "small" loss of integrity of the primary loop. An added requirement of an ATWS condition can be applied in the analysis and can be justified by considering the (relatively high) accident frequency. Positive reactivity insertion can be assumed having its origin from de-borated water injected from the ECCS, *i.e.* HPIS, accumulators and LPIS.

2.2.1 Decrease in primary coolant flow temperature

The PSA includes all kinds of overcooling transients, such as:

- Pre-heating anomalies leading to low feed water temperature.
- Speed-up of feed water pumps leading to high feed water flow.
- Inadvertent opening of secondary side valves (relief, safety or steam dump).

All these transients lead to a decrease in the SG secondary side pressure/temperature with associated enhancement of the heat transfer between the primary and secondary sides. Consequently the primary side coolant temperature will decrease, which subsequently results in an increase in both reactivity and neutron flux. The latter event can result in a high variation (rate of change or time derivative) of neutron flux signal and associated reactor trip.

The events can have their origin in only one or in some of the available SGs. Uneven conditions amongst the loops will then result. Thus, the core inlet temperature distribution will in those cases be non-uniform and associated regional core power variations can occur.

2.2.2 Increase in primary coolant flow

The PSA includes transients which result in a speed-up of RCPs leading to the increase of primary coolant flow. An increase of the primary coolant flow will, on the average, decrease the core coolant temperature, and consequently these transients lead to reactivity increase and neutron flux increase. The latter event can result in a high variation of the neutron flux signal and associated reactor trip.

It should be noted that the trip on high variation of neutron flux (high rate of change) can also influence the ability of the plant to handle transients which according to the design are acceptable without any trip initiation. So for instance the variation of the neutron flux can be of a greater extent if the initial increase in coolant flow is combined with a (temporary) loss of heat sink. This combined event can evolve during a loss of external grid transient where a transition occurs from full power operation to a level corresponding to the plant “in-house” power consumption (typically about 5% of the normal power). The sudden unavailability of the external grid results in a speed-up of the turbines with an associated increase of the RCP speed and thus an increase in the primary coolant flow. The turbine acceleration protection system initiates fast closure of the turbine stop valves, which is accompanied with opening of the steam dump valves connecting the steam lines directly to the turbine condensers. Normally this valve operation sequence results in a short time period of complete loss of available heat sink. However, if inappropriate timing of the valve operation sequence prevails (*i.e.* delay of dump valves’ opening) the loss of heat sink can exist during a short but still somewhat extended time period. This time can be long enough to allow the primary coolant to be heated up a certain amount. When this hotter primary coolant enters into the core it will result in a fast core power decrease from the previous elevated level and thus in a more pronounced rate of change in the neutron flux signal, which under unfavourable conditions can result in reactor trip.

2.2.3 Control rod ejection

The PSA includes the transient leading to a local reactivity increase with a corresponding neutron flux increase. The associated local power increase results in a high variation of neutron flux signal and a consequent reactor trip.

During the control rod ejection accident it is usually assumed that the minimum DNBR is attained. However, this parameter only provides supplementary information. Subchannel analysis should be performed in this type of transient, as should fuel pin power reconstruction from the fuel assembly homogenised fluxes as calculated in kinetic codes. The results of the reconstruction (for example the ratio of maximum reconstructed thermal power density to the average thermal power density in the node) can be used for a more precise estimation of the hot channel factor, and can especially reveal a possibility to reduce a prevailing high conservative value.

2.2.4 Boron dilution

The PSA separates the analysis of this transient in relation to two different assumptions:

- With control rods operating in automatic mode.

- With control rods in manual mode and not operating.

Following the first assumption, the automatic control of the control rods provides the reactivity compensation. A steady-state nominal condition is retained in this case without any major problem.

In the second case, the subsequent reactivity increase results in a neutron flux increase. The reactor will trip due to either high over-temperature (DNB protection) or high over-power delta-T signal.

Another situation has been conceived that is generally applicable to PWRs [14]. This event could perhaps occur under cold shutdown conditions if an idle RCP was started in a locally boron-diluted loop. The initially stagnant water inventory with low and deviating boron concentration would then be non-uniformly transported into and through the reactor core, causing a reactivity transient that could be severe, possibly resulting in prompt local criticality. This type of event was reported to have an overall probability of about 1.5×10^{-7} per reactor operation year. The mitigative measures are basically of administrative character.

2.2.5 Bent fuel assemblies

This type of event usually originates from a combination of strong spring forces for holding down the assemblies against the lower core plate and high burn-up of the fuel with associated (comparatively large) fuel length expansion. As the spring forces prevent some of the expansion the fuel assemblies will bend, a bending that will also influence adjacent assemblies and will thus propagate radially in the core. The resulting effects on core geometry will be wider gaps between the assemblies with local enhancement of moderation and increased reactivity. In case of more severe bending effects some difficulties in the operation of control elements of single RCCAs can result.

No mention of such scenarios has been found in the PSA.

2.2.6 Main steam line break (MSLB)

This type of event results in improved heat transfer between the SG primary and secondary sides due to the depressurisation of the secondary side as a consequence of the steam line break. The associated steam flow from the SG is usually limited because of the flow restriction nozzle located at the SG dome outlet. The primary coolant in the affected loop will thus experience a temperature decrease (whence an “overcooling” type of transient), and as the coolant reaches the core a reactivity increase is obtained. The resulting uneven temperature distribution of the coolant in the RPV downcomer upper part will propagate through the downcomer and up through the lower plenum and into the core. Although mixing processes along the flow path tend to alleviate the effect, remaining non-uniformity of the core inlet temperature distribution will nevertheless cause a non-uniform reactivity increase in the core with associated non-uniform power responses. This transient is also a DBA for the reactor protection system.

The PSA includes the transient as an event leading to primary HPSI due to secondary side initiator. The safety injection activation will also directly imply reactor trip.

2.2.7 Anticipated transient without scram (ATWS)

All the transients mentioned above (with the possible exclusion of bent fuel assemblies) are from a PSA standpoint classified as transients leading to reactor trip. Thus the shutdown system is assumed to bring the core condition to a safe zero power state with only the core decay heat remaining as a power source.

However, all the transients can be combined with a low probability ATWS condition. This type of transient includes a prerequisite of a certain degree of shutdown system failure and will thus result in what can be characterised as “extreme scenarios”. These transients are classified as category 5 events. In relation to core reactivity insertion this means that the effects from the control rods are restricted according to specified requirements, thus rendering the reactivity effects from Doppler (fuel temperature) and coolant temperature and density (void) more predominate for the transient scenario outcome. Operator actions can be expected to mitigate the severity of the final consequences of the transients but in the short term this cannot be accounted for. Transient events which, when applied with ATWS conditions, have been identified as the most limiting in PWRs include those that cause a rapid decrease in the core flow to power ratio and those resulting in a rapid reduction in the SG heat removal.

Various types of malfunction of the shutdown or scram system can be conceived resulting in the system being more or less unable to terminate the core fission reactions. The malfunctions can be electrical or mechanical in nature, and in order to mitigate the occurrence frequency some measures have already been taken in the reactor design.

The probability of misbehaviour due to electrical reasons can be decreased if the design of the electrical network allows adequate redundancy. This is usually accomplished by dividing the network into several parallel networks (subs) so the scram system can receive the same trip signal from several subs. Physical separation of systems into different rooms is paramount in this respect. If the automatic systems fail, scram can be initiated manually from the control room, although most likely too late to be effective in reducing the immediate reactivity changes.

The mechanical malfunctions can result in one or more of the control rods failing to enter the core despite the reception of trip signals. The core design includes margins such that at normal operation the most reactive control rod group can fail without compromising the safe scram function. This most reactive control rod group varies from cycle to cycle (as does the burn-up exposure) and is dependent on the reactivity worth of each control rod included, meaning that the effect of having this group unavailable has to be evaluated for each specific core design.

It is also possible to enhance the scram effect by using other available systems and thus to obtain scram with a fewer number of control rods. However the core reactivity influences from activating these systems will be rather slow. In PWRs forced boron injection through the CVCS is readily available. The boron concentration of the RCS coolant is higher at the beginning of a core cycle as compared to the end of the cycle to compensate for the higher inherent reactivity of a fresh or newly reloaded core.

2.3 Relevant BWR transients

Similar to the case of PWRs, various types of BWRs exist that basically differentiate in the nominal core power and configuration of the recirculation loops, *i.e.* with external main circulation pumps (MCPs), internal MCPs and jet pumps. The main advantage of internal pumps and jet pumps is that there is no provision for having a large break in any recirculation loop. For many of the still-conceivable break sizes in internal pump BWRs the capacity of the high pressure ECCS is designed to be adequate to keep the core flooded. Nonetheless, in cases where this system is not able to keep the water inventory of the RPV the automatic depressurisation system (ADS) will be initiated to release steam to the containment wetwell pool. Thus the pressure is rapidly decreased to a level where the high-capacity, low-pressure injection system can be activated and more efficiently mitigate any unwanted consequences.

Despite the obvious need for plant-specific PSA, an attempt is made below to identify BWR transients having general interest in relation to RIAs or having RIA-related behaviour. In a similar manner as for the PWR cases above, some evaluation of transients "which could pose fuel integrity problems originating from substantial core reactivity changes", has been pursued. Again the starting point has been a list of transients that from deterministic T-H analyses have been found to involve substantial reactivity changes. Consideration has been given, when selecting transient scenarios, to the recently issued IAEA guideline [15]. The transients are discussed and a short phenomenological description of the expected events is provided. In all BWR transient scenarios the use of coupled 3-D technique is justified by the utilisation of fuel assemblies with external shrouds and by the broad variation of the axial time-dependent linear power distribution between the assemblies.

The following transients have been identified:

- Overpressurisation events, for instance turbine trip without condenser bypass available.
- Large break LOCA, break of recirculation line for external pump BWRs, MSLB for internal and jet pump BWRs.
- Control rod withdrawal.
- Decrease of core inlet temperature, feed water temperature decrease.
- Decrease of core inlet temperature, feed water flow increase.
- Increase of core flow, main circulation pump flow rate increase.
- Overpressurisation event, main steam line isolation valve closure.
- Core instability events.

2.3.1 Turbine trip without condenser bypass available

The turbine trip (TT) event without condenser bypass available constitutes a (relatively) frequent event in BWR operation. Due to the rapid closure of the turbine isolation valve a positive pressure wave propagates from the turbine valve to the RPV and reaches the core from the top (*i.e.* across the dryer and steam separator deck) and from the bottom (*i.e.* across the downcomer and the lower plenum). Core void collapse causes positive reactivity insertion and the associated power excursion is typically mitigated by scram initialisation. The possible opening of steam relief valves to the containment wetwell pool counteracts the effect of the pressure increase in the RPV, but the short-term pressure wave propagation will still influence the immediate core reactivity.

It should be noted that this event is characterised as a limiting sequence for the core in many current BWRs with internal MCPs. In some of those previous designs the MCP trip became a limiting transient because of approaching DO conditions. This was caused by a combination of fast coasting-down of the pump speed (low inertia of the pumps), along with an associated fast decrease in core mass flow rate and core power (through the void reactivity feedback), and equalisation of the fuel pins' radial power distribution. Later redesigns included a means to increase the MCP inertia by electrically connected devices maintaining the rotational energy of the pumps, thus increasing the coasting-down time with less DO sensitivity for the pump trip events.

2.3.2 LBLOCA

The LBLOCA is a DBA for BWRs with external recirculation lines. The accident originates through the rupture of one recirculation line resulting in a 200% guillotine break. The accident is a basis for the ECCS design as well as the design of the containment pressure-suppression system. From a reactivity point of view the fast depressurisation will quickly increase the core void contents with an associated decrease in core power due to the reactivity feedback. The voiding and the immediately initiated shutdown system will quickly terminate the core fission power, thus resulting in the decay heat and possible cladding metal-water reaction heat as being the remaining sources of core power.

The steam line break is similarly a DBA for many BWRs with internal MCPs and jet pumps. The associated steam flow rate is limited by nozzles located in the steam dome outlet connections to the steam lines. A single steam line break will also, depending on the design, result in an increased flow rate in parallel steam lines due to connections with common headers. Despite these design characteristics, quite different flow rates will usually develop in the two ends of the 200% guillotine break due to the very different lengths of the flow paths.

It should also be noted that the DBA for the internal MCP reactors may differ between different countries. The DBA for the ECCS can for those reactors be a break in the low-pressure injection system itself or in the steam line. However, double-ended steam line breaks are normally DBAs for the containment and the pressure-suppression system. In Sweden, BWRs with internal MCPs have been licensed assuming double, double-ended steam line breaks.

2.3.3 Control rod withdrawal

The unintentional CR withdrawal event is generally characterised by a single rod withdrawal from a core position with high reactivity worth, but as the reactivity control usually involves the manoeuvring of control rod banks other configurations can be conceived. The initial condition can include no power (results generally under more severe conditions) and full power operation. Fast rod withdrawal can also be envisioned, during which a control rod is detached from its drive and thus remains fixed in the core while the drive is progressing downwards out of the core. The control rod thereafter loosens and falls out of the core with an associated rapid reactivity increase.

The core design must be such that the shutdown margin is at least 1% with the highest reactivity worth control rod withdrawn from the core. Equally, the shutdown to cold condition should always be possible even if one control rod group is assumed not to function. It should further be noted that compensation for the higher core initial reactivity (in a fresh or newly reloaded core) is not only accomplished by adequate control rod insertion but also by using burnable absorbers in the fuel itself.

2.3.4 Feed water temperature decrease

The FW temperature decrease event can have its origin in malfunction of FW pre-heaters and also in a loss of pre-heater functionality. The latter occurs, for instance, at turbine trip, in which case the extraction of steam from the turbine stages to the pre-heaters are no longer available. Normally the FW temperature is about 180-220°C when entering the RPV and a reduction of the temperature will increase the subcooling at the core inlet, which increases the reactivity. An uneven distribution of the core inlet subcooling can be envisioned if only a single FW pre-heater string is lost.

It is notable that adjustments of the feed water temperature (decrease) can be used as a means to increase the core power at the end of a core cycle when the core condition includes maximum core coolant flow with all the control rods withdrawn (coast-down operation). However, a decrease of the feed water temperature will result in a somewhat lower overall efficiency despite the increase in the generator electrical power.

2.3.5 Feed water flow rate increase

The FW flow increase event results in a similar effect on the core as the FW temperature decrease, and will be obtained if the FW flow is increased due to FW pump control malfunction (for example). A proportionally higher amount of low enthalpy flow will enter into the RPV downcomer where it mixes with the separated flow from the steam separators. The resultant effect will be an increase of the core inlet subcooling.

2.3.6 Main circulation pump flow rate increase

Normally the core power is controlled by the core flow rate and thus by the MCP speed (at a given control rod pattern). An unintentional MCP flow rate increase may be caused by malfunction of possible valves installed in the MCP lines or by a spurious signal controlling the pump speed (power control mode), whatever is realistic for the concerned NPP. The case of start-up of a MCP at an incorrect temperature can also be conceived. Uneven distribution of the core inlet flow rate can be envisioned, resulting in uneven power distribution amongst the affected fuel assemblies.

2.3.7 Main steam line isolation valve closure

The MSIV closure event will include rapid closure of one or several main steam line isolation valves. This closure creates in the short term a pressure pulse (a similar situation to the turbine trip event) that causes void collapse in the core with associated reactivity increase. Opening of steam relief valves to the containment wetwell pool counteracts the effect of the more long-term pressure increase in the RPV but the short-term pressure wave propagation will nevertheless influence the immediate core reactivity. Closure of an isolation valve in one steam line results in increased flow rate in the remaining steam lines, possibly causing several of the associated valves to close, thus resulting in a successive closure of a number of valves with an associated influence on the core reactivity.

2.3.8 Core instability events

These types of transients, whenever they might occur, usually originate at reactor operation with comparatively low power and low recirculation flow rate resulting in operation points being in the "exclusion region" (left upper corner) of the BWR core power-flow map. Conditions corresponding to about 50% core nominal power and 30% core inlet flow rate may be seen as the area in the power-flow map where the highest probability for oscillations occurs. The operation in this area is avoided by means of adequately defined control and trip conditions illustrated as additional "border lines" in the map. Nevertheless, certain transient conditions can still result in time periods where operation in this area will occur, usually accompanied with the observation of oscillating core behaviour. A number of such transients were summarised in [16] and additional occurrences have been reported since then, one example being mentioned in the CRISSUE-S database (Chapter 6). The core two-phase flow itself provides a potential for oscillatory behaviour and the strong feedback between moderator coolant density and core power enhances the effect under certain conditions. In-phase and out-of-phase power oscillations have been observed and both modes can, for large amplitudes, have an unwanted influence on the fuel integrity. Control systems for RPV pressure and DC level (and various BOP components) can influence the oscillatory behaviour in an unfavourable way. Transient analysis should be performed until establishment of steady-state natural circulation has been obtained.

The additional application of an ATWS condition can be considered as an extreme event in screening analyses. Although this condition is beyond the DBA boundaries due to the low probability of occurrence, analysis results can provide valuable insights and be useful to designers and to safety analysts to optimise EOP and to better understand the safety margins of the NPP.

2.3.9 Anticipated transient without scram (ATWS)

From a PSA standpoint, several of the transients mentioned above are classified as transients eventually leading to reactor trip. Thus the shutdown system is assumed to bring the core condition to a safe zero power condition with only the core decay heat remaining as a power source.

However, similar to the PWR case, the transients can be combined with a low-probability ATWS condition with a prerequisite of a certain degree of shutdown system failure. In relation to core reactivity insertion this means that the reactivity effects from Doppler (fuel temperature) and coolant temperature and density (void) will have a predominant influence on the transient scenario outcome. Operator actions can be expected to mitigate the severity of the final consequences of the transients, but this cannot be accounted for in the short term. It is noted that in a BWR there are always other means to control the core power, though such methods are slower in response than the scram system, *e.g.* adjusting the MCP speed and the feed water temperature and flow, and increasing the boron concentration in the coolant. The negative void reactivity feedback will also have a tendency to make the core somewhat less sensitive to reactivity changes with time derivatives of the order of or greater than the fuel rod time constants (about 5 s). Various types of malfunction of the shutdown or scram system can be conceived being of electrical or mechanical nature, and in order to mitigate the frequency of occurrence some measures have already been taken as concerns the reactor design. It is noted that the scram function is achieved by quick insertion (a few seconds) of the control rods by actuation of hydraulic systems that push the control rods into the core. Separate from the hydraulic systems there are also systems that will insert the control rods by means of electrical drives. This functionality is of course slower than when using the hydraulic systems.

The probability of misbehaviour due to electrical reasons can be decreased if the design of the electrical network allows adequate redundancy which, again, is accomplished (for instance) by using separate networks (subs) and by physical separation of the systems into different rooms. The mechanical malfunctions can result in one or more of the control rods failing to enter the core despite the reception of trip signals. The normal core design includes margins such that the most reactive control rod group can fail without compromising the safe shutdown function.

2.4 Relevant WWER transients

WWER is the Russian PWR that differentiates from the Western type basically by the presence of horizontal tube steam generators (HTSG) and by the core fuel pins being arranged in a triangular pitch array (these differences are relevant in the course of the CRISSUE-S activity; other system features also distinguish WWER from PWR but are not discussed here). The WWER-440 and the WWER-1000 NPP classes can be distinguished. The core power (about 400 MWe and 1 000 MWe in the two classes, respectively), the number of SGs (four and six in the two classes), the core configuration (presence of hexagonal fuel boxes and "ejectable" fuel elements in the WWER-440 and open core with cluster CR in the WWER-1000) and the containment design (confinement with various features in the case of WWER-440 and full containment in the case of WWER-1000) are differences between the two NPP classes. Inside each class further differences among individual NPPs can be observed but are less relevant in the present context.

Again, despite the obvious need for plant-specific PSA, an attempt is made below to identify WWER transients having general interest related to reactivity variations. The starting point has been the same list of transients as that provided for PWRs, which from deterministic T-H analyses have been found to involve substantial reactivity changes. Again consideration has been given, when selecting transient scenarios, to the recently issued IAEA guidelines [13]. The transients are discussed along with a short phenomenological description of the expected events. Thus the following transients have been identified:

- Decrease in primary coolant flow temperature.
- Increase in primary coolant flow.
- Control rod ejection.
- Boron dilution.
- Bent fuel assemblies.
- Main steam line break (MSLB).
- Anticipated transient without scram (ATWS).

An overall discussion is provided for each of these transients below. It is noted that PSA results are not available for the mentioned events. As is the case for PWRs, other conceivable transients can also be listed for WWERs including LOFW, LBLOCA, SBLOCA, incorrect connection (start-up) of an inactive (idle) loop, and isolation of one loop combined with an ATWS condition. Each of these transients can have specific core reactivity consequences that have to be evaluated:

The LOFW in WWER is very similar to the corresponding PWR case and is caused by blockage of the FW pumps, which can originate from a "partial station blackout". The primary loop temperature increases and moderator and Doppler neutron feedback contribute to core power decrease. An added requirement of unavailability of scram system (*i.e.* an ATWS condition) can be applied in the analysis and can be justified by considering the (relatively high) accident frequency.

The LBLOCA originates from a double-ended guillotine break of one CL at a location between the MCP and the RPV. Making reference to the WWER-440, current LBLOCA-DBA originates from a “32 mm break” in CL. The coupled 3-D T-H/neutronic analysis may reveal that current ECCS is also suitable for coping with both the “200 mm break” and the “500 mm break”. These two events can thus be added to the WWER DBA list. For WWER 440/213 the double-ended guillotine break (500 mm) constitutes the maximum DBA.

The SBLOCA is very similar to the corresponding PWR case and is caused by “small” loss of integrity of the primary loop. An added requirement of an ATWS condition can be applied in the analysis and can be justified by considering the (relatively high) accident frequency. Positive reactivity insertion can be assumed, having its origin from de-borated water injected from the ECCS, *i.e.* HPIS, accumulators and LPIS.

The transient (including isolation) of one loop may only occur in a WWER-440 NPP equipped with main isolation valves installed in HL and CL. The transient can be considered complementary to the case of incorrect connection of an idle loop. The ATWS condition is added, notwithstanding the low probability of occurrence, in order to reveal the intrinsic safety margin of the system.

2.4.1 Decrease in primary coolant flow temperature

A number of events can be postulated which could result in a decrease in primary coolant temperature.

2.4.1.1 Feed water system malfunctions that result in a reduction in feed water temperature

A feed water system malfunction that results in a decrease in feed water temperature can cause a significant decrease in primary temperature and pressure. The negative moderator and fuel temperature reactivity coefficients and the reactor control system can cause core reactivity to rise as the primary system temperature decreases. This increase in core power – if coupled with a decrease in primary system pressure – could potentially violate the core thermal limits.

2.4.1.2 Feed water system malfunctions causing a flow increase

Feed water system malfunction that results in an increase of feed water flow can cause a significant decrease in primary temperature and pressure. The negative moderator and fuel temperature reactivity coefficients and the action initiated by the reactor control system can cause core reactivity to rise as the primary system temperature decreases. In the absence of a reactor trip or other protective action, this increase in core power – if coupled with a decrease in primary system pressure – could potentially result in a DNBR less than the safety analysis limit.

2.4.1.3 Steam pressure regulator malfunction or failure that results in increasing steam flow

This event is defined as a rapid increase in steam flow that causes mismatch between the reactor core and the steam generator load demand. The steam pressure regulator malfunction or failure that would cause the greatest steam flow increase is assumed to be the opening of the single largest steam valve. The reactor control and limitation system is designed to accommodate a small step load increase, such as an excessive load increase event, without a reactor trip. Large load increases would cause an excessive reduction in reactor coolant temperatures, large coolant contraction and resultant low pressure. Such large step increase may cause a reactor trip actuated by the primary reactor protection system.

2.4.1.4 Inadvertent opening of a steam generator relief or safety valve causing a depressurisation of the main steam system

As a consequence of this accident, the steam release results in an initial increase of steam flow which decreases during the accident as the steam pressure falls. The increased energy removal from the reactor coolant system causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cool-down results in an insertion of positive reactivity.

2.4.1.5 Spectrum of system piping failures

The steam released arising from a rupture of a main steam line would result in an initial increase of steam flow, which decreases during the accident as the steam pressure decreases. The increased energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cool-down results in an insertion of positive reactivity. If the most reactive rod cluster control assembly is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return the power. The core is shut down by the boric acid delivered by the HPIS.

2.4.2 Increase in primary coolant flow

In this event the transient initiator is a speed-up of the MCPs, which leads to an increase of the primary coolant flow and a decrease of core coolant temperature with consequent increase of core reactivity and reactor power. The transient is terminated by the reactor trip. This type of transient has not been investigated in WWER safety analysis.

Another transient, which has been analysed in detail as part of the WWER safety analysis, is the start-up of an inactive reactor coolant loop at an incorrect temperature. This event is caused by starting an idle MCP without first bringing the inactive loop hot leg temperature to correspond to the core inlet temperature. This results in both a significant increase of core coolant flow and injection of cold water into the core. Both of these effects decrease core water temperature, causing a rapid core power increase due to moderator reactivity feedback.

This transient can have a higher probability of occurrence in certain WWERs (WWER-440s) compared to the corresponding case in a PWR. This is attributed to the fact that WWER-440s are equipped with main isolation valves installed in HL and CL.

2.4.3 Control rod ejection

The rod ejection transient is defined as an assumed failure of a control rod mechanism pressure housing such that reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. The consequence of this mechanical failure is a rapid reactivity insertion along with an adverse core power distribution. If this transient did take place, a fuel rod thermal transient that could cause DNB could also occur, together with limited fuel damage. The amount of fuel damage that can result from such an accident will be governed mainly by the worth of the ejected rod and the power distribution attained with the remaining control rods and associated pattern. The transient is terminated by the Doppler effect caused by the increased fuel temperature and by the reactor trip actuated by high neutron flux signals. This occurs before conditions are reached that can result in damage to the reactor coolant pressure boundary, or can sufficiently disturb the core, its support structures or other reactor pressure vessel internals such that the capability to cool the core would be significantly impaired.

The PSA includes the transient leading to a local reactivity increase with its corresponding neutron flux increase. The associated local power increase results in a high variation of neutron flux signal and, as a consequence, reactor trip.

Experience from a control rod ejection accident analysed for WWER-1000 safety purposes has been gained for cases with event classification according to Condition 4 (limiting faults – no occurrence is expected – postulated as the limiting design case) [12]. Acceptance criteria that were evaluated in the analyses include [17]:

- Maximum reactor pressure during any portion of the assumed transient will be less than the value causing the structural stress to exceed the emergency condition stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code (non-acceptable stresses).

- Radially averaged fuel enthalpy in any point of the core should be lower than 840 kJ/kg (acceptable fuel damage limit from the point of view of long-term coolability of the fuel rods).

- Peak cladding average temperature should be lower than 1 649°C.

- The amount of fuel melt in hot spot pellet should be lower than 10%.

- The zirconium-water reaction should be lower than 16% by weight.

- Dose equivalent for the most exposed individuals should be less than 0.25 Sv for the whole body, 1.5 Sv for the thyroid in adults and 0.75 Sv for the thyroid in children.

During the control rod ejection accident reaching of minimum DNBR is usually assumed. Thus, the only purpose of this parameter is to provide supplementary information.

The subchannel analysis methodology should be applied for this type of transient. Associated fuel pin power could be found from pin power reconstruction based on the fuel assembly homogenised fluxes calculated in the kinetic codes. The results of reconstruction (for example by using the ratio of maximum reconstructed thermal power density to average thermal power density in the node) can be used for a more precise estimation of the hot channel factor, especially in view of possible reduction of a prevailing high conservative value.

2.4.4 Boron dilution

The scenario of this transient is boron dilution and coolant temperature disturbance in a hot subcritical state when all control rods are fully inserted. The transient is usually analysed for BOC conditions with a high content of soluble boron in the coolant. The rapid decrease of boron acid concentration and reactor core coolant inlet temperature results in an increase in the initially negative reactivity associated with increasing fuel temperature. The increasing fuel temperature leads to the Doppler feedback effect stopping the rapid growth of reactivity.

2.4.5 Bent fuel assemblies

This type of transient has not yet been analysed.

2.4.6 Main steam line break (MSLB)

2.4.6.1 Main steam line break at HZP initial conditions (WWER-440)

In this scenario the transient is supposed to be initiated at a reactor core condition corresponding to HZP at the end of the fuel cycle. All control rods are inserted into the reactor core with the exception of the most reactive control rod, which is assumed to be fully withdrawn during the transient. The initial decay heat power, initial fission power level and the initial subcriticality of the reactor core must all be carefully defined.

The reactor core power excursion after opening of the MSLB is caused by the combination of two basic phenomena – decrease of reactor coolant inlet temperature and, after activation of HPIS, increase in boron acid concentration in the coolant. The opening of the break causes depressurisation of the steam generators with a resulting decrease in primary pressure and coolant temperature. The decrease of reactor core coolant inlet temperature is associated with increased core reactivity. The increasing fuel temperature results in the Doppler feedback effect, reducing the rapid growth of reactivity. The pressure and volume control systems are in operation and will maintain the system pressure and the pressuriser level throughout the transient. The still-decreasing core coolant inlet temperature will counteract the rapid reduction of the core reactivity as originated from the Doppler effect. The continued increase of the core power is finally stopped by the injection of highly borated water from HPIS. The accumulation of boron acid concentration in the primary circuit results in a gradual decrease of reactivity and nuclear power until the end of transient simulation.

2.4.6.2 Main steam line break at HFP initial conditions (WWER-440)

In this scenario the reactor core is under HFP conditions at the end of its fuel cycle. The control rods belonging to the sixth group of control rods are at position of 175 cm from the bottom of the core. Other groups of control rods are fully withdrawn. The MSLB is usually defined as a double-ended guillotine break in the main steam line of one steam generator. This causes different depressurisation amongst the steam generators with a consequent decrease of primary pressure and coolant temperature and associated reactivity and power increase. The reactor power scram is caused by power level signal "110% of P_{nom}" with given time delay. All control rods fall down with the exception of the most reactive rod, which is assumed stuck in the fully withdrawn position. The turbines are turned off by closing the turbine isolation valves at the scram time point. The pressure and volume control systems are in operation and are immediately activated to correct the system pressure and pressuriser level. The main steam header pressure signal causes the closure of the steam isolation valves and will isolate all SGs from the feed water. The culmination of re-critical reactor power can be observed as a consequence of primary circuit overcooling. The reactor power excursion is terminated by injection of highly borated water from HPIS.

2.4.7 Anticipated transient without scram (ATWS)

Anticipated transients without scram (ATWS) are operational occurrences followed by the assumed failure of one reactor scram function [18]. Available transient analyses of ATWS events indicate that

WWER reactors, like PWRs, have a tendency to shut themselves down if the inherent nuclear feedback is sufficiently negative. Various control and limitation functions of the WWER plants also provide a certain degree of defence against ATWS.

For complete failure of scram, *i.e.* inability to automatically drop control rods into the core, all "anticipated transients" should be assessed. These transients are defined as unscheduled events whose occurrences are expected during the plant lifetime with a frequency of 10^{-2} events per reactor year or higher.

Grouping of initiating events can be based on the expected phenomena following a complete failure of the control rods to drop into the core. Concerning the challenge to the barriers to fission product release, the following groups of events may be defined:

- Transients which cause a rapid decrease in the core flow to power ratio and, as a consequence, the possible occurrence of an associated boiling crisis leading to fuel cladding damage. This includes control rod withdrawal and loss of flow events.

- Transients resulting in a rapid reduction in SG heat removal. These events include turbine trip, turbine load reduction and loss of main feed water. The events cause rapid pressurisation, opening of the pressuriser safety valves and, as a consequence, a decrease of primary coolant inventory. Initial reduction of reactor core power and eventually a possible shutdown occurs by reactivity feedback due to an increase in moderator temperature and an associated decrease in moderator density.

For WWER-1000 NPPs, the loss of feed water (including variations such as loss of condenser vacuum) deserves the utmost attention because it leads to an associated very severe challenge of the fuel rods' integrity [because the main coolant pumps (MCP) are tripped on low SG water level] as well as of the pressure boundary.

As mentioned is Section 2.2, these events have been identified as the most limiting ATWS in PWRs of Western design. Although the WWER-1000 reactors have a significantly different response to loss of feed water than Western PWRs due to MCP trip on low SG level and the characteristics of horizontal SGs, similar consequences can nevertheless be expected. These events are difficult to analyse, basically because of the T-H processes in the SG (heat transfer during simultaneous dry-out and flow decrease) and in the core (large power shape changes will occur as relative power shifts towards the bottom of the core).

Turbine trip should also receive adequate attention due to its relatively high frequency and the fact that it bounds the turbine load reduction and the loss of electrical load.

Transients originating from an increase of heat removal to the secondary side, such as inadvertent opening of the BRU-K valves with a strongly negative moderator temperature coefficient (MTC), must be taken into account. The number of BRU-K valves that might open due to a single instrument failure, considered as an "anticipated transient", should be specified. Most Western plants are designed in such a way that a single instrument failure will not open more than one valve, and thus only a single inadvertent valve opening is considered as an anticipated transient. These transients result in reactivity insertion due to the cool-down of the primary coolant, and cause increased power generation and change in the power shape. In these cases, various degrees of core damage are the primary concern.

If only one steam valve will open for any failure classified as an anticipated transient, then the maximum power increase is limited to about 15%. If the feed water pumps can match the associated steam flow, this event can generally be shown to require no protection. If the feed water pumps cannot match the steam flow, the SG level will then decrease to the MCP trip set point. The resulting transient would be bounded by the loss of feed water transient discussed above.

This class of event also includes turbine control malfunction during low power operation, which causes the turbine steam load to increase to its maximum value (*e.g.* from 40% to 105%). These conditions need to be addressed because the power distribution may be more adverse than transients initiated at full power.

Some initiating events do not normally generate a scram signal unless there is a failure of the limitation system, but may still be ATWS initiators if all or part of the control rods fail to insert due to a mechanical failure. Such events include turbine trip (requires power reduction to 40%), trip of two MCPs (40% or 50%) and trip of one main feed water pump (50%). These events can also be bounded by considering complete loss of reactor coolant flow (all four MCPs), or complete loss of feed water flow.

2.5 Some general observations in relation to RIA and associated PSA treatment

From the above discussions it seems clear that current PSA applications mostly focus on events resulting in end conditions involving various degrees of fuel damage. These events can include time windows during which substantial reactivity changes occur, but the focus of such PSA is still on fuel damage and not directed specifically towards the reactivity transients as such. Apart from events resulting in degraded core conditions various types of "milder" transients are mentioned in the PSA as well, but associated end conditions are in those cases the state of safe reactor shutdown without compromising any fuel safety limits. Generally, it seems difficult from current PSA studies to obtain an evaluation and ranking of various reactivity transients from the frequency of occurrence and consequence points of view, and from such results are guided direction and scope of deterministic coupled T-H/neutronic revaluation analysis of those transients based on the merits of benefit and importance. However, newer trends indicate that the PSA methodologies are increasingly used to evaluate and rank the importance of plant modifications and to provide supportive information in the process of possible relaxation of currently used technical specifications. This situation is expected to change, as more experience is gained from activities like the latter one as well as recent activities related to risk-informed decision-making and regulation.

It should be emphasised, though, that results of recent coupled detailed BE T-H/neutronic deterministic analyses seem to indicate an obvious potential to relax previously used core power factors due to the current usage of highly conservative associated presumptions. This will provide a basic condition for more efficient utilisation of fuel (increasing the core power and applying other optimisation basis at core reloading) while still retaining and possibly even increasing the safety in terms of specified margins against fuel overheating and risk of jeopardising fuel integrity by approaching PCI limits.

2.6 Reference transient scenarios

Comprehensive BE calculations and the resulting data evaluation for the transients specified in Sections 2.2 through 2.4 concerning relevant PWR, BWR and WWER transients constitutes an enormous task, which among other things is plant and plant-cycle specific. Therefore, no information obtained for a specific application may be generalised to different plants or to different situations within the same plant.

Notwithstanding the above, an effort has been made within the CRISSUE-S framework to produce reference data for a number of the transients specified above. The main purpose of the effort is to constitute a database of results that can be used by analysts as well as by decision makers to construct a "quantitative" idea about expected transient scenarios and, definitely, to support decisions about the need to perform coupled calculations related to generic NPPs.

Calculations have been performed utilising computational tools and input decks developed and already utilised for purposes outside the CRISSUE-S area of activity. The adopted computational tools include the CASMO code for neutron cross-section derivation, RELAP5/MOD3.2, RELAP5/MOD3.3 and RELAP5 3D© for thermal-hydraulics, and the PARCS and Nestle codes for 3-D neutronics (see Section 2.2 in Ref. [1] for more details concerning the codes). The Penn State University (PSU) has produced all the sets of cross-sections needed for the performed analyses related to PWR, BWR and WWER-1000 respectively [19-21]. The RELAP5 input decks were developed at the University of Pisa (UPISA) [22-24]. The PARCS and Nestle input decks were developed within the framework of a co-operative effort among PSU, UPISA and Texas A&M University [25-27]. BWR stability analyses were partly performed within a co-operation between UPISA and University of Valencia [28].

The reported transient analyses for typical NPPs, using the coupled T-H/3-D neutron kinetics code systems, have been summarised in Annex 1 of Ref. [1] and include the scenarios provided in Table 2.1 below.

Table 2.1. Overview of analyses performed and transients considered in Annex 1 of WP2 [1]

ID	Transient type	PWR case identity	BWR case identity	WWER case identity
1	MSLB	Case 1-PWR: TMI (X) Case 2-PWR: ASCO (X)	–	Case 1-VVER:VVER-1000 (X) Case 2-VVER: VVER-440 (X)
2	LOFW-ATWS	Case 3-PWR: TMI (X)	–	Case 3-VVER: VVER-1000 (X) Case 4-VVER: VVER-440*
3	CR ejection	Case 4-PWR: TMI (X)	Case 3-BWR: PB2 (X) Case 4-BWR: PB2 CR bank ejection (X)	Case 5a-VVER: VVER-1000 (X) Case 5b-VVER: VVER-1000 CR bank ejection (X) Case 6-VVER: VVER-440*
4	LBLOCA-DBA	Case 5-PWR: TMI(X)	Case 9-BWR: PB2 (Y)	Case 7-VVER: VVER-1000 (Y) Case 8-VVER: VVER-440*
5	Incorrect initiation of an inactive loop	Case 6-PWR: TMI (X)	–	Case 9-VVER: VVER-1000 (Y) Case 10-VVER: VVER-440*
6	MSLB-ATWS	Case 7-PWR: TMI (X)	–	Case 11-VVER: VVER-1000 (X) Case 12 -VVER: VVER-440*
7	SBLOCA-ATWS	Case 8-PWR: TMI (X)	–	Case 13-VVER: VVER-1000 SBLOCA (X) Case 14-VVER: VVER-440*
8	Turbine trip	–	Case 1-BWR: PB2 (X)	–
9	Turbine trip-ATWS	–	Case 2-BWR: PB2 (X)	–
10	FW temperature increase	–	Case 5-BWR: PB2 (X)	–
11	MCP flow rate increase	–	Case 6-BWR: PB2 (X)	–
12	MCP flow rate increase-ATWS	–	Case 7-BWR: PB2 (X)	–
13	BWR stability	–	Case 8-BWR: PB2 (X)	–
14	BWR stability-ATWS	–	Case 9-BWR: PB2 (*)	–

(X) Analysis performed and documented.
(Y) Analysis performed and not documented.
– Analysis not applicable or not considered.
* Analysis not performed.

Each calculation is documented in the Annex 1 by a one-page description and two pages of significant system-related time trends or 3-D snap-shots of core-related quantities.

The following main limitations apply for the calculations performed:

- The cross-section sets have been derived for a specific NPP status that does not coincide with the worst or the most critical status expected for the considered transient. As a consequence, the range of validity of the derived cross-sections may be exceeded when different transients are analysed (by adopting the same set of cross-sections).

- Although an effort was made to achieve qualified results, a comprehensive check of the various qualification requirements was not made. Therefore, the reported results should be considered qualified as-far-as-possible. This is true for the stand-alone RELAP5 or PARCS or Nestle input decks as well as for the coupled input decks.

- Actuation logics of all NPP systems that may have a role during the assigned transients have not necessarily been simulated.

The major findings and conclusions are summarised in Annex 1 of Ref. [1].

It is also noted that the world-wide interest in performing this kind of coupled T-H/3-D neutron kinetics analyses has increased in recent years, and many related activities have been pursued at different organisations. This can be exemplified by the overview presented in Table 2 in Ref. [1]. The transients considered are not necessarily related to the transients mentioned in Sections 2.2 through 2.4, but will merely serve as an indication of the general interest and of the scope of various activities performed in this specific area of NPP transient analysis.

Chapter 3

THERMAL-HYDRAULIC SYSTEM SIMULATION

The evaluation and assessment of the safety of NPPs is closely related to the ability to determine the temporal and spatial distributions of the flow field T-H conditions along with associated effects from heat sources and heat sinks throughout the reactor coolant system, and especially in the core region. On-line measurements at different locations of the NPP primary and secondary systems can provide valuable information in this context, but important T-H details within, for instance, the fuel assemblies will not be revealed through such means. The established method to evaluate those complex conditions is by deployment of advanced numerical T-H simulation tools including comprehensive and validated models to predict the phenomena-based behaviour. It is certainly true that the field of nuclear technology relies more than any other engineering field on simulation efforts to ascertain fulfilment of specified safety criteria. However, it is also realised that this approach will only provide an approximate picture of the true conditions, but that the level of approximation is expected to be relaxed as a result of ever-evolving model development and T-H research activities.

Originally based on different assumptions and equation formulations, the T-H system computer codes generally used, *e.g.* RELAP5, TRAC, ATHLET, CATHARE, are today being developed to have rather similar basis and capabilities. They are now based on governing equations representing the transient, non-homogeneous, non-equilibrium, two-phase flow including heat transfer processes from present solid heat structures like the fuel rods, piping and RPV wall structures, as well as internal support structure components. The associated two-fluid, two-phase flow model is formulated from the spatial and time-averaged conservation equations for mass, energy and momentum of the two basic phases (liquid, vapour) with allowances for soluble components in the liquid phase and non-condensable components in the vapour phase. Due to the averaging process used in the formulation all information concerning the local flow processes is lost, as are the time fluctuating turbulence contributions. Thus these effects have to be re-established by means of proper models based on the available averaged properties of the two-fluid model. However, such re-established models can only approximately simulate true behaviour; this is especially the case when considering the two-phase flow with a large variation in the distribution of interfacial area and its volumetric concentration. The common utilisation of static flow regime maps, based on prevailing void and mass fluxes, can not fully ameliorate the situation. Activities pursued by the USNRC, among others, to develop and implement models for dynamical flow regime maps can probably help to improve this state of affairs [29], but many constitutive models continue to suffer from very simple dynamic characteristics or even exclude them altogether.

The constitutive relationships, usually in the form of empirical algebraic expressions, are used to describe wall friction, heat transfer and inter-phase drag and mass transfer with interfacial area concentration based on the rather static flow regime maps. As these models are based on direct or indirect information emerging from various types of experiments they also have implicitly incorporated allowances for the turbulence and other micro-scale conditions that prevailed in the used experiments. The codes' two-phase models are essentially 1-D, although most codes have various capabilities to (at least in an approximate way) also be able to simulate basic 3-D flow field conditions. In this context it is also realised that, although applying the available 3-D capabilities of the codes, the T-H phenomena are still modelled using basically 1-D assumptions. Thus the multi-dimensionality of the flow field can possibly be simulated only to a certain degree and even then the influence of this multi-dimensionality on predicted phenomena is not taken into account when applying the empirical models, apart from adjustments of pertinent flow velocities.

Due to the numerical approximations and the empirical nature of the models included in the T-H system codes, extensive activities related to validation of the code models have been pursued over the years. At present, a fairly well-based experience has been established as concerns the codes' capability to simulate the T-H conditions prevailing during various time windows of specified transient scenarios. Validation has partly been accomplished using experimental data from specially-designed, scaled-down test facilities simulating major components of a complete reactor system. From these integral effect tests (IETs) [30] information can be found regarding the interaction between different parts of the system and their associated influences on the T-H condition and distribution. Another stage of the validation is constituted by the separate effect tests (SETs) [31], which are conducted at full- or close-to-full-scale with well-defined boundary conditions. These tests are devoted to investigating specific T-H phenomena or to the behaviour of single components (pumps, valves, tees, etc.), and provide additional validation information. Transient data from real NPPs are also important due to the full scale and true geometry, although such data can naturally only reveal conditions for fairly mild transients (operational transients, start-up and commissioning tests). The synthesis of information from IETs, SETs and plant transients form the base foundation of data for T-H safety evaluation. Other important sources of information related to the development of new and improved models and particularly to expanding the experiences and knowledge base of code application characteristics, comprise international projects such as CAMP, as well as activities within the OECD/CSNI (for instance ISPs).

With backgrounds of different intentions and levels of sophistication the T-H analyses are performed with either a conservative or a BE approach. The former has traditionally been used for the so-called licensing analyses, being part of a NPP's commissioning activities and has to be submitted to and approved by the regulatory authority. In those analyses the safety margins obtained are expected to be conservatively large, as certain T-H phenomena as well as certain plant and system features are not credited. It should be noted that the conservative approach does not provide any indications as to the true safety margins nor does it provide a true simulation of a specified scenario. In the BE analyses the T-H phenomena are simulated as accurately as possible (according to present knowledge) and the safety margins obtained will more closely reflect the real margins in the plant. This type of analyses also provides more realistic simulations of the NPP behaviour during the course of the transient scenarios and can consequently reveal detailed system information that can be paramount for the understanding of T-H phenomena interaction. If BE analyses are used for licensing purposes they must be accompanied by uncertainty analyses to quantify the uncertainty of calculated parameters. The uncertainty includes contributions from simplifications introduced both into the governing equations and to the constitutive relationships and models, but also from using such models outside their original ranges of validity. The uncertainty evaluation related to T-H system analysis is discussed in more detail in Ref. [1], Section 2.6. In relation to coupled T-H/neutronics analysis it is realised that in order to be meaningful the T-H simulation must be performed with BE analysis in mind. Using the T-H conservative approach along with a coupled 3-D neutronics analysis would not provide any practical understanding of the actual transient plant scenario as the principal physics have intentionally been too greatly distorted.

3.1 Some general characteristics of the T-H system codes with emphasis on coupled T-H/ neutronic analysis

The T-H system analysis codes (*e.g.* RELAP5, TRAC, ATHLET, CATHARE) have evolved into very useful tools that can be advantageously applied for NPP safety analyses and evaluations of plant responses to specified process disturbances. The T-H system codes that are commonly used within the nuclear safety community are briefly presented in Section 2.2.2 of Ref. [1]. The validation database is enormous, and the accumulated experiences from an ever-growing amount of applications provide comprehensive guidance for the code applications. The experiences are documented in the codes' manual sets, for instance the User's Guide [32] and User's Guidelines [33] of the RELAP5 manuals and the User's Manual [34] of the ATHLET code. However, despite the considerable validation efforts pursued over the years there are still areas of code applications that reveal limited experiences, *e.g.* transients for which pressure wave propagation effects are important and for which profound 3-D and recirculating flow formations occur in certain parts of a system. These kinds of more esoteric flow conditions can have a great influence on the course of certain transients, not the least in relation to core reactivity responses, where for instance it can be essential to have an accurate simulation of the flow field distribution at the core inlet (*i.e.* at the lower core plate).

The codes use a similar set of basic governing differential equations to describe the two-fluid, two-phase model. Differences do exist which can influence the solution strategy, so for instance RELAP5 uses a single pressure for the liquid, vapour, and interface, resulting in elliptic characteristics (*i.e.* the characteristic equations have imaginary roots) while CATHARE uses different phasic and interfacial pressures in such a way that the equation system always reveals hyperbolic characteristics (*i.e.* the characteristic equations always have positive roots). The numerical models used when solving the governing equations are based on first order (linear approximation) finite difference donor cell schemes with staggered mesh, meaning that the momentum computational cells are displaced half a cell length relative to the mass and energy cells. Various degrees of implicitness in the time domain can be introduced to mitigate the need of using a sound speed (pressure pulse propagative) based Courant limit for numerical stability. Thus, RELAP5 makes use of a semi-implicit method which requires a material-based Courant limit for stability, *i.e.* a bulk velocity limit [35]. The implicitness can be carried further as is done with TRAC's Stability Enhanced Two-step (SETS) method, which requires no such restrictions for stability.

An inherent characteristic of the numerical methods used is the diffusive influence they have on the solution, which seems to be somewhat more pronounced for the SETS method. This can be detrimental for certain type of T-H transient simulations. Thus, later versions of TRAC (the USNRC consolidated code TRAC-M or TRACE [36,37]) also have the option of using a semi-implicit method for the solution [38]. Nevertheless, the semi-implicit models also reveal numerical diffusion with a tendency to smooth gradients, which severely can influence the simulation results of, *i.e.* BWR stability transients. Such transients have phenomenological dynamic characteristics being of the same order of magnitude as those originating from pure numerical effects. In order to minimise the numerical diffusion influence (truncation error) in the normally used semi-implicit scheme, being first order accurate in time and space, the so-called Courant number based on the fluid node propagation velocity should be equal to one [35]. In relation to BWR stability analyses with coupled T-H/neutronic codes it is noted that, due to the limited experiences gained so far, it is currently not possible to make accurate predictions, whereas simulations of already-occurred events could be pursued (validation). This is partly because of the solution being influenced by effects from the numerical algorithms used, and this influence is not fully quantified. This situation is expected to change in the future when evaluation of results and conclusions from more validation analyses for these types of transients become available.

For simulation of specific and critical phenomena other numerical schemes have been introduced. RELAP5, for example, can use as an option a second-order accurate Godunov numerical scheme for tracking the boron concentration gradients in the system [39]. This can naturally have a profound influence on the kinetics responses of the fuel and must be carefully evaluated for those types of simulations for which boron tracking is essential.

Through the validation of the codes against well-defined tests, experiences have been gained as concerns how to adequately nodalise various parts of the reactor coolant system, so acceptable simulations of prevailing T-H phenomena can be expected. This type of experience has resulted in recommendations as found in the code documentation, for instance in [33] for the RELAP5 code. It is realised that these recommendations also implicitly incorporate the numerics effects (for instance numerical diffusion) on the results, and although nodalisation studies usually have been performed for verification of converged nodalisations within reasonable limits, extension of the node sizes far outside those recommended can greatly influence the results. In the RELAP5 document node sizes of about one to a few meters are recommended for basic loop nodalisations while more dense nodalisations are recommended for the RPV and for the SGs and pressuriser if present. In the core node sizes of about half a meter are common but shorter nodes can be used if higher resolution is needed. In that context it must be noted that the accuracy will not necessarily increase if a more dense nodalisation is used, due to both numerical effects and stability. If very short nodes are used the influence from numerical diffusion can be substantial and the results must be carefully evaluated. The lower limit on node size is related to the stability of the semi-implicit numerical scheme and has a length roughly that of the average scale used [35], meaning that the minimum cell length to hydraulic diameter ratio should be unity or greater.

In relation to coupled T-H/neutronic analysis it is advantageous to use the same nodalisation in the T-H core simulation and in the kinetics analysis for reasons of consistency. In the latter case the core is usually modelled with one radial node per fuel assembly, which includes up to 25 axial nodes resulting in about cubic calculational nodes. This dense axial nodalisation is rarely used in T-H core simulations and consequently very little, if any, T-H validation information is currently available for this type of nodalisation. It is also presently not feasible (in most codes not even possible) to use too many radial nodes in the T-H core simulations despite the current availability of powerful computers. If possible, it is preferred to use the same nodalisation in the axial (vertical) direction although some kind of lumping of radial nodes (*i.e.* fuel assemblies) is necessary. When undertaking the radial lumping a well-founded strategy must be adopted to reduce influences on the T-H result from fuel assembly differences and from the physical location of the assembly in the core. This is especially important when simulating BWR instability transients and is crucial if regional (*i.e.* out of phase) core oscillations are to be simulated.

The multi-dimensional flow field within a single fuel assembly can not readily be found from the T-H system code core simulation. Instead it is envisioned that this can be adequately evaluated by using a proper subchannel model that is coupled to the T-H system code through an exterior communication interface in much the same way as the kinetics model is coupled to the system code [29]. In this way the 3-D flow field interior to the fuel assembly can be simulated with axial and radial boundary conditions taken from the T-H system code core simulation. Thus the “hottest” fuel pin characteristics can be found; however, to be complete this also requires that an adequate pin power reconstruction model is used in the associated kinetics simulation to obtain the prevailing pin power distribution.

3.2 Specific considerations and data needed for the T-H system models in coupled T-H/ neutronic analysis

The following sections indicate some areas in the preparation of the T-H system input models that can be of particular importance in relation to the T-H/neutronic analysis. The list is by no mean intended to be exhaustive and will not repeat what can be found in the code manuals, but provides items that call for attention in certain situations and that can be a contributing source for unquantified uncertainties. The discussion has been separated in relation to T-H phenomena and process models that could have a specific influence on the kinetics responses, and to be considered when setting up the NPP model. In the latter case references are made to PWR, BWR and WWER modelling aspects.

In a general sense the preparation for an adequate simulation of a specific transient in a NPP system includes some basic requirements (as reported in Section 2.4.1 of WP2 [1]). The first requirement is to select a BE T-H code which should have a documented capability to reproduce the expected transient T-H phenomena. This is closely related to code validation and qualification activities. The version of the code selected should preferably be "frozen", *i.e.* it should have reached a certain degree of maturity with a substantial application reference list and documented associated experiences.

The next requirement includes the preparation of the NPP system model itself, with adequate nodalisation, boundary conditions, etc. It is realised that the amount of data required to fully and accurately describe a NPP T-H system along with associated control and trip systems is substantial, with much user freedom when defining what should be included, what nodalisation should be used, what considerations should be given to certain phenomena, etc. Basic requirements concerning accuracy, level of detail and specific model considerations when preparing and setting up adequate NPP T-H input models have been developed based on experiences gained over the years, not least from various code validation activities. As mentioned earlier, these requirements are provided in the T-H system code manuals in the form of guidelines; they can also be found in other separately published specific works (see Refs. [13,15]). Although nowadays the guidelines have reached a certain level of quality assurance with a fairly well-based foundation, modifications and additions are continuously provided as new experiences are acquired.

In this respect, some principal considerations related to the assurance of the level of proficiency of the code user have evolved [40]. The intention is to increase the awareness of reported "user effects" on the model set-up and the associated influence on the calculational result [41], and how such effects can properly be dealt with.

Once developed, the NPP input model should be adequately qualified, both on a steady-state and transient level. This is discussed in more detail in WP2 [1], Section 2.4.1.

3.2.1 T-H phenomena and process models that could strongly affect the kinetic response

3.2.1.1 Dynamic subcooled boiling

During the course of reactivity insertion and the associated fuel rod local power increase the temperature difference between the rod surface and the adjacent boundary liquid layer will increase and eventually result in positive net void formation on the rod surface. The amount of void is determined by several factors, *e.g.* rod surface characteristics, temperature and flow velocity gradients across the boundary layer at the rod surface. As the two-fluid T-H system codes only have one liquid and one vapour temperature, the temperature gradient effects are not directly taken into account, and thus must be included using specific models and correlations. For example, the wall surface void formation used in RELAP5 is based on the model proposed by Lahey along with the Saha-Zuber method [39]. For rapid transients the dynamics of these models can be questionable, resulting in unrealistic local reactivity responses.

It has further been found that the net vapour generation in RELAP5 is under-predicted at low pressures, and that improvements are needed in this regard [42]. Associated developments are supported by experimental programmes. It is also clear that simulation of the dynamics related to vapour nucleation inception during the depressurisation phases of a transient can be questionable in some cases, *e.g.* the pressure undershoot found in various blowdown experiments [43] is not at all simulated in earlier versions of RELAP5 (MOD 3.2.1.2 from 1997 being one example), while the effect has been over-predicted in later versions (MOD 3.3), the reason for this being the utilisation of different models for interfacial heat transfer.

3.2.1.2 Dynamic CHF

The occurrence of CHF will result in a decrease of the fuel rod surface heat transfer coefficient with an associated increase in rod temperature which, in turn, will influence the reactivity feedback through the Doppler effect. The CHF models have a very empirical nature and have been developed with data from static CHF tests (or as the test procedures usually include a slow increase of rod power and/or decrease of coolant flow characteristics a more correct term might be "quasi-static" CHF tests) in tube or rod bundle geometries. The CHF models' dynamic behaviour during power excursions as a result of core reactivity insertion requires more validation, although transient CHF data from well-defined SETs are very few if any. The indirect validation against IET data provides some insight; however as the T-H conditions for the simulated core in those tests are not systematically varied an in-depth interpretation of the CHF conditions is difficult.

3.2.1.3 Spacers with mixing vanes

This type of spacer is usually located in the upper part of the fuel assemblies to promote turbulence and mixing and also, in consequence, to increase the margins against CHF. Due to design characteristics these spacers also increase the "detachment" of the vapour bubble layer on the rod surface into the bulk of the fluid flow which, in case of subcooled conditions, will enhance the void condensation. Thus there will be a change in the void fraction and also, as a result, in the reactivity and fuel rod power. The spacers are simulated in the T-H system codes through their pressure loss coefficients and area contractions, but no allowances are included to simulate the increased mixing effects on void distribution, although some provisions exist to include the mixing effects on CHF. Thus, there currently exist no means in the codes to simulate influences on reactivity from these types of devices.

3.2.1.4 Heat transfer inside fuel pins

The heat transfer phenomena inside the fuel pins comprise heat conduction in the fuel pellets, heat transfer across the gas gap between fuel pellets and cladding inside surface, and heat conduction across the cladding thickness. The combination of these effects along with the geometrical fuel pin design will influence the fuel pin overall time constant at power variations, though the heat transfer across the gap ("gap conductance") will have the greatest influence. All the effects are greatly dependent on the fuel pin burn-up and, consequently, will also have a variation in the axial direction. It is noted that with normal "heat structure" modelling practice these axial variations can not be taken into account. A possibility could be to axially use a stack of different "heat structure geometry" combinations (RELAP5 nomenclature), however this would impair the usage of the "moving mesh" capability in the event that reflood simulations are needed.

The very details of the calculation of the fuel pin inside heat transfer differ between the various T-H codes. In ATHLET the heat transfer inside the gas gap can be modelled either by using a constant heat transfer coefficient or by means of correlations. By external coupling of the 3-D T-H neutron kinetic code DYN3D [44] with the system code ATHLET, the heat transfer inside fuel pins can also be calculated by means of correlations. In this case information should be provided about reference gap width, reference fuel temperature, reference cladding temperature, cold gas pressure and helium mole fraction for the selected state characterised by a given burn-up. There is the possibility to use either one averaged burn-up state for all fuel assemblies in the core or a separate burn-up for each fuel assembly. There is no provision to include any axial profile effect on fuel assembly burn-up distribution to potentially improve the estimation of the heat transfer coefficient of the gas gap.

In-depth discussions of fuel-related issues are found in WP2 [1], Chapters 3 and 4.

3.2.1.5 Valve characteristics

Valve characteristics, *i.e.* valve area changes as (for instance) a function of time with associated pressure losses, must be provided as part of the valve component input. The exact characteristics of real valves are difficult to attain and thus the effects often have to be evaluated from sensitivity analyses. In relation to reactivity changes important influences from the valve operations can be the BWR core pressure variations at closure of the turbine stop valve which in turn is closely associated with the pressure wave propagation through the steam line and into the RPV. The pressure wave will become more pronounced if the valve area change is incorporating a high valve area time derivative, *i.e.* from a pressure wave influence point of view the often-used linear area change will provide non-conservative results.

The very modelling of the valve is greatly code-dependent. Thus in the ATHLET code the valve simulation model is junction-related and can be specified as a “standard” valve model or as a “discharge” model. The standard valve can be located at every location within a pipe or within a single junction pipe. Its impact on the thermal-hydraulics includes an additional inertia term in the momentum of the valve junction, an additional pressure drop due to form losses in the valve and if the valve cross-section area is less than the junction cross-section area, it is used for the calculation of the fluid velocity and hence may cause critical flow. If the valve closes completely, the remaining mass flow is set to zero. To minimise this discontinuity it is important to decelerate the mass flow adequately during the closing of the valve. The discharge valve model was originally developed for the simulation of any kind of discharge at the thermo-fluid dynamic system edge. Its additional features in comparison with the standard valve are the option of an intrinsic use of different, detailed critical flow models based on the isentropic homogeneous equilibrium model, the Moody model and the 1-D finite difference model. Valve actions have a large influence on the behaviour of the mass flow and consequently may cause severe pressure-mass flow oscillations leading to small time steps. It is therefore necessary to specify the valve behaviour very carefully to avoid discontinuities at very fast changes of the valve cross-section areas. The opening time for the simulation of a discharge valve, then, should not be less than 10 minutes, and for the simulation of a pressure or inventory control valve the opening and closing time should be much larger (an order of magnitude larger) than the expected time size.

3.2.1.6 Pressure wave propagation through the steam line into the dome

The pressure wave propagation as such seems to have a rather limited direct influence on the core reactivity. More important is the T-H influence the pressure variation will have on associated voiding and condensation phenomena which in turn can have more direct influence on core reactivity. Both these phenomena include time constants in the transient responses on pressure variations, and consequently the pressure variation propagation in specified flow paths is only one factor when it comes to reactivity change characteristics, which include time, magnitude and spatial distribution aspects.

The simulation of pressure wave propagation using the T-H system codes is somewhat questionable and should be performed with some caution. Thus a careful consideration of the Courant limit, in these cases based on acoustic wave propagation, and of the calculational node length has to be adopted (see Ref. [33], Section 2.1.2.2). The donor-cell differential scheme has a strong tendency to attenuate pressure waves when using node lengths as usually applied in "bulk" flow transient scenarios and a careful evaluation of nodalisation and usage of time step sizes can possibly ameliorate the situation. It is mentioned in Ref. [33] that if there is any concern whether the code can produce undistorted results in this respect, then the choice of a more adequate code should be made. Though, a recent study of pressure wave propagation in simple pipe networks as simulated by RELAP5 provided encouraging results, careful nodalisation and selection of time step sizes were necessary [45].

In this context the pressure wave propagation into the steam dome of the BWR through the dome outlet nozzle has to be mentioned. This transmission, which is also accompanied by a reflection wave back into the steam line, is rarely simulated, and the resulting dome pressure response must be regarded as very uncertain. As the turbine valve essentially controls the BWR pressure at power operation, it is realised that this type of phenomena can nevertheless be of high importance during BWR instability conditions, as the dome pressure feedback on the turbine valve in principle can create oscillations corresponding to the core coupled T-H/neutronic resonance frequency with associated amplification. Thus the correct simulation of the combined features of the T-H pressure wave phenomenon and the control system characteristics is essential for adequate core void dynamics and associated reactivity responses.

3.2.1.7 Frictional and discrete pressure losses

Accurate pressure loss distribution in the T-H system is an essential prerequisite in order to be able to calculate an accurate flow distribution, which in turn is necessary to obtain an adequate reactivity response. The frictional pressure losses are usually included by assigning appropriate roughness to various surfaces in contact with the fluid (RELAP5) or by specifying constant losses (ATHLET). It is envisioned in relation to the core pressure drop that the roughnesses are influenced by the time period during which the fuel assembly is loaded into the core, *i.e.* by the burn-up, with increased roughness with time due to the growth of the oxide layer. It is also realised that the detachment of possible rod surface vapour layer, as caused by spacers with mixing vanes as indicated above, can influence the resulting downstream frictional pressure loss. This type of phenomenon is rarely simulated by the T-H system codes.

Usually the discrete loss coefficients are assigned a constant value, but for the fuel assembly spacers there are indications that a weak Reynolds number dependence is more adequate, with the loss coefficients decreasing as the Reynolds number increases [46].

3.2.1.8 Phase separation at tees

The phase separation phenomena at tee components can have a dominant influence on the course of the system depressurisation rate [29]. Thus, for a BWR steam line break, a perfect separation results in maximum depressurisation while minimising the loss of inventory. Reaching low pressure while maintaining a high liquid inventory may be non-conservative (by retaining the fuel temperature well below any critical levels, for example). It was also discovered that the separation at tees (branches) in RELAP5 had limited applicability. Activities are underway by the USNRC to review existing models and to evaluate and possibly improve the code capabilities under extended ranges of conditions.

In ATHLET the simulation of branching of a main pipe or a vessel is obtained by applying the T-junction model. It consists of two parts: a model for the simulation of the flow splitting between the main pipe and the branching pipe concerning the momentum equation, and a model for the simulation of liquid entrainment and vapour pull-through phenomena at horizontal, vertical upward and vertical downward branching. The flow-splitting model for branching conduits can be applied for pipes branching orthogonally from the main pipe. For dividing flows the momentum flux term is calculated as full acceleration applying the downstream mixture density; for impacting mass flows a full pressure recovery is assumed. For horizontal or slightly inclined main pipes liquid entrainment and vapour pull-through phenomena can be simulated for horizontal, vertical upward and vertical downward branching. For vertical or vertical inclined main pipes the entrainment model can only be applied if the control volume and the mixture level are defined in the corresponding main pipe.

3.2.2 Consideration of the NPP model set-up in relation to the kinetic response

When making coupled T-H/neutronic analysis a limited model comprising only the NPP core and appropriate boundary conditions could be adequate in specific cases. Normally, though, there is also a need to include the actual influence of the reactor coolant loop system and even "balance of plant" components and control systems such as feed water pre-heater strings, level and pressure control systems and CVCS in PWRs. Thus the complete model could be very complex, requiring a vast amount of plant specific data not only of the geometrical type, but also time-dependent characteristics of various auxiliary systems. It is realised that some data are rarely available, an example being time constants of PWR hot and cold leg temperature gauges located in protective pockets in some NPPs, thus shielding them from the direct coolant flow. In other cases the data can be proprietary, like the detailed performance of the steam separators for an extended range of operating conditions. To help reveal associated influences on system and reactivity responses in such cases, comprehensive sensitivity analyses must often be performed with reasonable variations of identified and thoroughly selected parameter ranges. In the following some areas are indicated which can call for specific attention at the preparation of T-H system code input and nodalisation.

3.2.2.1 Steam separator characteristics

In BWRs the steam separator characteristics have a direct influence on the conditions in the downcomer through the amount of carry-under. Together with the feed water inflow into the downcomer and the mixing processes this will be one factor influencing the core inlet subcooling and thus the core reactivity. The amount of curry-under depends on the liquid level in the RPV (*i.e.* downcomer level), the core coolant mass flow rate and the core power. A low level results in higher curry-under and vice versa, although for modern steam separator designs the variation of curry-under for reasonable changes in liquid level is moderate. The curry-under also increases at lower coolant mass flow rate but decreases as core power decreases. In RELAP5 the separator effects can be simulated by utilising a very basic model with prescribed void fraction limits for determining ideal separation functionality, or by utilising a more mechanistic model simulating GE multi-staged centrifugal separator components. Until more experiences are obtained for these GE separator components default input values are recommended [32]. It is realised that the characteristics for an extended range of operating conditions can be difficult to acquire due to the proprietary nature of the needed data. Usually, inferred characteristics for only a single or very few operation points can be obtained, which naturally hampers a comprehensive modelling of the component. Further, the behaviour at pressure wave propagation passing through the separator model into the core is not very clear, and applying a certain caution must be emphasised when using this component for simulation of those types of phenomena.

It is noted that in WWERs with horizontal steam generators no steam separator components are used and the separation is obtained merely by gravitational effects of the low velocity two-phase flow.

3.2.2.2 FW distribution in BWR upper DC section

In BWRs careful modelling of the feed water inlet sections that distribute the feed water into the downcomer and of the downcomer itself can be essential in order to adequately simulate effects from possible non-uniform feed water distributions. The feed water lines are connected to separate spargers and in some designs groups of the lines are fed by water pre-heated by separate feed water pre-heater strings. Dependent on the operating conditions for each pre-heater string, situations can be envisioned in which different pre-heating can prevail, thus resulting in different feed water temperature in the spargers or groups of spargers. Also, the auxiliary feed water system, used for instance at start-up and shut down sequences, is only connected to some of the spargers, thus resulting in possible non-uniform conditions around the downcomer upper section when that very system is in operation. The associated non-uniform temperature distribution can also (dependent on the downcomer flow conditions) remain to some extent at the core inlet and thus create local or regional reactivity heterogeneities. Thus the utilisation of the T-H system codes' 3-D capability is necessary to be able to simulate the fluid flow temperature field propagation, at least to a certain point.

3.2.2.3 Cold leg flow distribution in PWR RPV upper DC section

In PWRs the modelling of the cold leg inlet sections and the downcomer itself can be essential to the adequate simulation of effects from possible heterogeneities between the coolant flow from the different RCS cold legs. Dependent on the operating conditions in each steam generator and in auxiliary systems (*e.g.* CVCS), situations can be envisioned in which different cold leg temperatures and different cold leg boron concentrations can prevail. The associated non-uniformities in temperature and boron concentration distributions can also, dependent on the downcomer flow conditions, remain at the core inlet and thus create local or regional reactivity heterogeneities. Thus the utilisation of the T-H system codes' 3-D capability is necessary to be able to simulate the fluid flow temperature and boron concentration fields' development.

It is noted that the complete and exhaustive 3-D simulation of the cold leg flow distribution in PWR RPV upper downcomer section is not possible in many of the codes (RELAP5, ATHLET) but can be simulated in an approximate way, for instance by applying the concept of "cross-flow junctions" in RELAP5. Some of the T-H system codes have a more adequate 3-D capability, for instance the RELAP5-3D© code [47], for which a 3-D thermal-hydraulic model describing the flow condition in upper DC section was prepared for the WWER-440/213 reactor pressure vessel [48].

3.2.2.4 Control systems (delays, etc.)

The control system can have a profound influence on the T-H system responses and associated core reactivity behaviour. In some cases the control system can amplify disturbances originated elsewhere, for instance reactivity disturbances in BWRs resulting in dome pressure increase which in turn through the pressure control system influences the turbine valves with associated feedback on the dome pressure, although with some delay. In some plants (at least the Swedish BWRs) there is also a control system that monitors a RPV dome pressure increase event "in advance" by letting the turbine valves be influenced directly by the measured core reactivity changes (APRM) and thus adjusting the valve before the resulting pressure change has occurred. In this case situations can be envisioned for which an unfavourable combination of initial reactivity disturbance and control system characteristics can create – and amplify – a resulting reactivity dynamic response.

It is necessary to model all relevant aspects of the control systems, not only the system themselves. Influences from measurement gauges (for instance the several seconds delay in the measurement of PWR HL/CL temperatures as mentioned above, and similar delays of the FW temperature measurements in BWRs) and sense lines must be taken into account. Other possible effects could originate from, *i.e.* using modern digital control systems, which can introduce delays due to different parts of the system using different operational frequencies. It can also be advantageous to model the dynamics of the actual level measurement systems in more detail, for instance including models of the differential pressure between the DC liquid column and the reference water column.

In some periods during a transient there could also be dynamic pressure contributions to flow velocity measurements due, for instance, to asymmetrical locations of pressure taps. Usually the signals from several gauges measuring the same quantities are calibrated to provide the same signals under a specific operating condition, for instance under normal operation. Under deviating operating conditions the signals will consequently be different, thus resulting in erroneous indications of the flow velocities.

3.2.2.5 Core flow field 3-D effects including effects from BWR core inlet orifices

If heterogeneities of the RPV downcomer flow field are expected to prevail during time windows of a transient scenario, the model used must be able to adequately handle those variations when the flow is entering into the lower plenum and is propagating upwards into and through the core. Thus it is necessary to set up a 3-D nodalisation of the lower plenum and the core so advantage is taken of the T-H system codes' 3-D flow simulation capabilities. In PWRs with the comparably more open space in the lower plenum, the mixing effects will be rather efficient in smoothing out gradients of the downcomer flow. In comparison, the rather confined lower plenum space of BWRs, which includes a large number of control rod guide tubes, inhibits the effective flow communication in the transverse direction. Despite the layout of the lower plenum, a nodalisation which adequately preserves flow field heterogeneities from the downcomer throughout the progression of the lower plenum must be used. The nodalisation in the transverse direction is expected to be more critical and has to be made denser in the BWR case as compared to the PWR due to power and flow distributions across the core fuel assemblies.

Important data when specifying the 3-D nodalisation includes pure geometrical information as well as frictional and pressure loss data. These data can usually be found rather readily for the major flow direction (assumed to be the vertical direction) but require a bit more elaboration when it comes to the transverse directions. An example is the definition of the node flow area in the direction perpendicular to the control rod drives in the lower plenum of a BWR; this is not so easily determined, and usually a combination of an average node flow area and carefully selected junction flow areas has to be used. The friction influence in these directions can mostly be omitted, and instead the discrete pressure drop coefficients related to the selected junction flow areas must be specified (possibly also including Reynolds number dependence) so that the momentum transport is expected to be predicted as accurately as possible. Also the characteristics of the 3-D numerics, with its tendency to strongly diffuse any gradients, have to be taken into account, thus resulting in the calculated gradients (after passing through the lower plenum) being less severe at the core inlet level. This can be non-conservative in relation to core reactivity responses and is of special importance when the intention is to simulate local reactivity events and out-of-phase BWR instability transients.

In PWRs the open core design provides provisions for alleviating the effects of flow field heterogeneities at the core inlet and also of core local reactivity insertions along the core height. Consequently any core model intended to simulate the 3-D effects must be appropriately nodalised and the usually non-conservative influence from numerical diffusion must also be evaluated. The core bypass flow (*i.e.* the amount of core coolant flow not directly involved in the heat transfer from the fuel rods) can in principle be related to the area around the core periphery close to the core baffle. It is noted that in some designs the flow between the core baffle and the core barrel communicates directly with the DC flow while in other designs the communication is with the core flow. The bypass flow is thus included in the T-H system core model by properly defined parallel channels to the DC and to the core main flow paths respectively. These design differences also influence core modelling in the neutron kinetics codes.

Variations of this approach can be found and have been applied in relation to the WWER bypass simulation in the coupled code version DYN3D/ATHLET [34,44], where there are two possibilities for bypass modelling. The first possibility is to model the bypass as a separate non-heated channel. This means that the 3-D core modelling is performed by the kinetic code DYN3D and the bypass is modelled within the ATHLET code. The radial core boundary condition is pre-calculated for the initial stationary state with the use of albedo coefficients at the boundary between the heated reactor core part and the radial reflector. The radial reflector is not taken into account in further transient calculations. The radial albedo coefficients are supposed to be time-independent during the transient. However, it is realised that this assumption cannot entirely be fulfilled during a transient. In the second case the reactor core can be surrounded with a hypothetical radial band of reflector cassettes. The hydraulic parameters for these assemblies are adjusted in order to simulate the bypass conditions. In this case the bypass is fully modelled by the 3-D neutron kinetic thermal-hydraulic code DYN3D. The radial boundary conditions are defined at the boundary reflector-vacuum. This approach has its advantages in transients for which stronger influences of radial boundary conditions are expected, especially in the case with considerable changes of coolant temperature and boron acid concentration in the coolant.

In BWRs each fuel assembly is in principle operated as an isolated channel between common headers and a balance will be attained of the flow rate through each assembly to reach the common overall pressure drop. This overall pressure drop includes elevation, friction, acceleration and local pressure drops (from spacers, inlet orifices and outlet sections) with allowances for the two-phase characteristics in each assembly. In order to compensate for the radial power distribution of the core fuel assemblies, different inlet orifices are used for assemblies located in different core regions. Usually, two or three regions with different inlet orifice loss coefficients are used with higher loss coefficients towards the core periphery [20,49]. Thus the flow rate through the assemblies will differ somewhat, as will the individual two-phase conditions and thus also the stability margins [16]. In order to simulate the effect from these fuel assembly differences in a consistent and adequate manner it is crucial to apply a well-founded strategy when defining the nodalisation and the coupling to the kinetics code. As the BWR core consists of 700-800 and more fuel assemblies, some kind of lumping of the T-H channels is necessary for a tractable and feasible model. This lumping process is even more demanding when preparing a model to be used for stability analysis and that much more so for analysis of out-of-phase core power oscillations.

For BWRs, the core bypass flow includes the flow paths outside the fuel assembly shrouds for which the associated flow is normally subcooled, although the conditions can approach saturation at certain transient scenarios (*e.g.* depressurisation transients). Part of the same space between the fuel assemblies also contains the cruciform control rod blades (one control rod for each “supercell” consisting of four fuel assemblies). A movement of the control rods into the core will consequently reduce the available space to be occupied by the bypass flow thus resulting in somewhat reduced bypass flow. However, this effect is usually small and is in many cases ignored in the T-H system BWR simulations (the neutron kinetics influence is of course taken into account; this will be addressed further on) and instead the practice is to introduce simple core parallel flow paths, possibly being radially interconnected, to simulate the core bypass. Nevertheless it has been found that appropriate modelling of the bypass flow region can be important, although no boiling occurs [50]. It is also important that the direct gamma heating to the bypass coolant (usually about 1.7%) be taken into account as well as the heat transfer from the interior of the fuel assemblies through the shrouds. The associated gradual density change of the bypass flow will have some influence on the reactivity and its spatial distribution.

3.2.2.6 BWR modern assembly with internal water crosses

The fuel assemblies of BWRs have developed into new designs having advantageous features in relation to internal power factors and total power levels. This has been obtained basically by optimising the fuel assembly design, *e.g.* by dividing the fuel assembly into four subassemblies separated by a double-wall structure forming an internal flow channel with a cruciform-shaped flow area. Also in some designs the fuel rods have been made somewhat thinner and the number of fuel rods has been increased from 96 up to 100. The design results in a flatter internal power profile due to more favourable moderation effects from the internal water cross with an associated more uniform moderator distribution. Other characteristics include improved thermal margins (DO margins) and reduced fuel assembly pressure drop accomplished by the introduction of part-length rods and a modified design of the upper tie plate. Due to the thinner fuel rods a shorter time constant at power changes will result and consequently different core responses at reactivity changes compared to the old fuel assembly design. This faster response can have a destabilising effect under certain operating conditions (close to the upper left area in the power-flow rate diagram) and can require some modifications to the reactor control system (control and trip lines in the diagram). Modified reactivity characteristics (reactivity coefficients) will also result from the changed design. At control rod movements it can be envisioned that the two-phase flow distribution amongst the four subassemblies can be changed comparably more than in the earlier assembly designs with an “open” subchannel layout.

Only a limited amount of experience seems to be available regarding how to adequately model this type of fuel assembly in the T-H system codes with allowances for their four subassemblies and internal “flow bypass” through the water cross channel. In this respect, it is obvious that validation against appropriate experimental data is needed, but it is also realised that those data are currently mostly proprietary to the fuel vendors. Thus only tentative models can be prepared with associated high uncertainty in transient behaviour.

Chapter 4

THREE-DIMENSIONAL NEUTRONICS AND FUEL SIMULATION

Normally the neutron kinetics calculation is performed in a two-step procedure wherefore the first step includes preparatory calculations for the second step. In the first step adequate nuclear cross-sections for specified fuel types and core compositions are provided in a form that can be directly used in the second step. The cross-sections are obtained using a solution of the integral Boltzmann transport equation for individual fuel assemblies, including the neutron energy-dependent probabilities for the neutron reactions in various isotopes. These calculations can be performed with, *e.g.* Monte Carlo techniques, but most often deterministic lattice physics codes (CASMO [51], HELIOS [52]) are used for computational efficiency. The results from these calculations include homogenised few-group nodal cross-section and related data, including neutron kinetics data, which can be used in the second step of the calculation procedure. In this second step the transient 3-D diffusion equation is solved to provide the nodal neutron flux distribution of the complete reactor core or some selected symmetry part of the core and associated spatial core power distribution. This is accomplished using a nodal diffusion code of which examples include PARCS [53] and SIMULATE-3 [54].

It is noted that even though "approximate" diffusion equations are typically adopted in the available computational tools, reference solutions exist that are obtained with more sophisticated techniques, including Monte Carlo and detailed neutron transport. These solutions are used to benchmark the results from diffusion equations. Thus the problem of uncertainty evaluation seems more manageable and the numerical solution methods play a greater role regarding the accuracy of the results than is the case for thermal-hydraulics.

The following sections discuss some areas in the preparation of the kinetics models that can be important in relation to coupled T-H/neutronic analysis. The list will not repeat what can be found in the code manuals, but provides items that call for attention in certain situations and can be a contributing source to unquantified uncertainties.

4.1 Neutronic cross-section requirements

The basis for creating relevant core cross-section data is neutron energy-dependent reaction probabilities that are provided for various isotopes through libraries (nuclear data files) linked with the lattice physics codes. These libraries include the ENDF/B-V, ENDF/B-VI and JEF-2.2 data collections. They contain microscopic cross-sections for neutron resonance reactions as a function of neutron energies (up to about 10 MeV) for a vast amount of isotopes. The reactions can be separated into absorption, fission, transport and scattering. From these data libraries cross-sections are condensed to provide cross-sections for up to about 70 neutron energy groups. The groups are concentrated around the neutron thermal energy defined as being below 4 eV for which more than 50% of the groups are found with specific emphasis on the resonance energies of ^{234}U, ^{239}Pu and ^{240}Pu. The Pu contribution is essential, as about 50% of the energy generated in a typical LWR core has its origin in fission of the self-generated Pu isotopes [7]. The libraries also contain yields and decay constants for the various fission products.

From the created energy group structure used in the lattice physics codes neutron flux spectra are generated for each composition through input specified region of the core. The generated neutron flux spectra are used to calculate the fission rate and deplete available burnable isotopes with associated build-up of fission products, and to generate condensed few-group 2-D cross-sections data and other needed data to be used by the various transient 3-D nodal codes.

These few-group cross-sections data are parameterised in terms of burn-up and T-H feedback parameters. Under real reactor conditions, especially in transient situations, the neutron energy spectra change and the 2-D cross-section modelling based on a parameterisation model can only approximately describe the effects of neutron flux distributions, which change in space, time and energy. This "2-D off-line" cross-section generation and modelling constitutes a basic input data uncertainty affecting the results of coupled T-H/3-D neutronics calculations.

There are several shortcomings associated with the cross-section generation, of which can be mentioned:

- The use of 2-D lattice physics codes for cross-section generation. The majority of current lattice physics codes use the collision probability method (CPM), which becomes cumbersome and impractical in 3-D geometries.

- The current methodology to homogenise representative assemblies assuming symmetry (reflective) boundary conditions. This approach introduces significant errors in determination of neutron flow among assemblies in a real reactor core configuration. These errors are somewhat mitigated by the use of *ad hoc* assembly discontinuity factors (ADFs) for conventional reactor core analysis.

The standard cross-section modelling for coupled 3-D steady-state and transient simulations is based on data generated in the so-called base and branch calculations using the lattice physics code. The developed cross-section history and instantaneous dependence models are based on burn-up and local feedback parameters, *i.e.* fuel temperature, pressure, moderator temperature, void and boron concentration provided by the T-H system code. Changing each of the parameters one at a time develops the instantaneous cross-section dependencies. The cross-sections are not just dependent on one parameter but on all parameters, and associated cross-term cross-sections have to be taken into account in the transient analysis. However, the standard methods (currently used in core steady-state, depletion and transient analysis) such as the polynomial fitting procedure (usually based on Taylor expansions) do not take these cross terms into account. Since these methods utilise no cross-term dependencies on local feedback parameters they are especially inaccurate for transients for which large variations from nominal conditions exist. In this way, for instance, parameter perturbations are derived from a reference value to develop cross-sections under different conditions where only one parameter is varied at a time, with all other parameters remaining at average conditions. Once the new cross-section is known, along with the magnitude of the individual parameter variation, a derivative can be constructed which is used directly in a polynomial equation. A cross-section can be calculated at any reactor condition using these derivatives along with the average cross-sections. The most significant problem with this procedure is that it becomes more inaccurate as the parameter variations get farther away from average conditions.

The widely used CASMO/SIMULATE cross-section parameterisation model attempts to model the cross-section cross-term dependence involving an approximate type of cross-section representation [54]. Each cross-section can be evaluated as a summation of base and partial values. The base cross-sections represent the burn-up dependence (exposure, spectral history and control history) while the partial cross-sections represent the instantaneous dependence on local feedback parameters. Performing branch calculations generates the partial cross-sections, where again as with the polynomial fitting procedure one feedback parameter is changed only for a given perturbation. The model tries to account for the cross-term dependence by using separate partial cross-sections for different feedback effects. While the model is an improvement over the polynomial fitting procedure it is, again, limited to small perturbations.

The Adaptive High-order Table Look-up Method (AHTLM) was developed by PSU to derive accurate cross-sections to be used in 3-D coupled transient calculations [55,56]. This method uses the cross-section under average conditions and also the cross-term cross-sections, which are essentially cross-sections calculated by varying two or more parameters at the same time. These cross-sections are then tabulated in multi-dimensional tables which are used for interpolation. The tabulated cross-sections completely encompass the full range of conditions that may be present during the initial steady state and during the transient. In the AHTLM the user develops an operating condition box-envelope which bounds the expected range of change of the feedback parameters during both steady state and transient operation. The cross-sections are then calculated for the bounding box edges and within the box. These reference cross-section values are used to build the multi-dimensional tables, and the cross-sections are calculated using an elaborate table interpolation method. The method takes into account the non-linear T-H feedback parameter phenomena that are critical for accurate prediction of the cross-section behaviour.

To keep the number of lattice physics calculations to a minimum, and also the total amount of data that has to be tabulated while maintaining a particular degree of accuracy, a higher-order interpolation scheme must be used for the tables. The use of a higher-order interpolation over linear interpolation has the advantage of being able to obtain the same degree of accuracy with fewer points.

4.1.1 Depletion and spectral effects

The long-term time-dependent modelling of the core behaviour requires that many parameters be taken into account to properly predict the changing fuel properties. When a depletion model is used, it is very important to include as many of the known properties of the fuel as possible, as well as the operating conditions at which the node of interest depletes during the particular part of time. When performing a depletion calculation to a particular exposure, one has to take into account the changes in power that a reactor undergoes, the insertion or removal of control rods, the shuffling of fuel, and the local T-H changes. All of these factors contribute to the particular properties of fuel in a node at some point in the future, and to obtain accurate cross-sections at a particular exposure point they all must be taken into consideration. This is done through developing accurate input decks for the lattice physics code, which include as many of these effects as possible. In order to include all of these effects to a very accurate degree, a cross-section modelling algorithm is used based on the core analysis performed with a 3-D simulator code. Within an integrated core analysis package a more detailed method for developing accurate cross-sections has to be implemented specifically for the initial steady-state conditions modelled for each calculation node.

Cross-sections are dependent on burn-up, control variables and T-H properties. The burn-up dependence of cross-sections is a 3-D vector of exposure, spectral history and burnable poison history (for PWR) or control rod history (for BWR), and is based on the isotopic depletion. As fuel is burnt the isotopic content it contains is changed and, therefore the cross-section behaviour changes, *e.g.e.g.* with the production of Pu isotopes there is a hardening of the cross-section spectrum due to the increase of Pu in the fuel. Other changes occur due to the decay and production of fission products. This means that even if all other T-H properties are constant (*i.e.* steady-state conditions) there is still a change in the cross-section behaviour due to the long-term change in isotopes in the fuel while it is being depleted. Another parameter that affects the cross-section behaviour is the use of burnable poisons (BPs). The BPs cause the fuel to deplete differently with respect to time and space change of the radial power distribution at each depletion step. When the BPs are removed or depleted the behaviour of the fuel changes. In cases where BPs are present during one period of time in a reactor cycle and then are removed from the reactor during a second time period the effect of the BP in the first time period is still present (history effect). The knowledge that the BP was there must be considered since the fuel in this location has different properties than other fuel without BPs. The effect of control rod (CR) movement during the core cycle is similiar.

4.1.2 Importance of the spectral history modelling

As fuel is depleted in a reactor core, the fission process and the consequent decay of nuclide fragments change many of the fuel properties. The production of nuclides and the decay of others affect the neutron spectrum. These effects depend on the node conditions at which depletion takes place. When the fuel, moderator and structural materials are smeared together in the homogenisation process, any change in the properties greatly affects the cross-sections through shifting the neutron spectrum. It can easily be seen that when there is a non-static condition, the modelling of all of the properties of the core can be very difficult. The lattice physics code calculates the shift in neutron spectrum based on the enrichment of the fissile material, type of fissile material and the other materials present in the fuel. This is accomplished through knowledge of the different decay and production chains. The only control the user has over this process is determining the fuel type and enrichment. The change in BP or in T-H conditions during the depletion process will change the local properties of the fuel in the future.

A strong history effect is seen if the depletion calculation is performed at the core average T-H conditions. This is called a density history effect. The density of the coolant at the core outlet is smaller than the density of the core inlet. This means that if a calculation is performed at core average conditions the neutron behaviour will be modelled inaccurately. At the core inlet, the actual cross-sections contain more moderation than the values calculated at average conditions and less moderation at the top of the core. Further, if the approximate values are used by the nodal code the calculated power is shifted by the density of the water when T-H feedback is considered in the calculation. The density history has a direct effect on the axial power shape, skewing it towards the bottom of the core. The cross-sections at the bottom of the core are generated based on under- and over-moderated conditions at the top of the core. The modelling of the cross-sections for the axial temperature distribution in a reactor core is very important to accurately predict the axial power distribution.

4.1.3 Importance of assembly discontinuity factors (ADFs)

It is well known that the diffusion approximation is not accurate in areas where the flux changes rapidly (*i.e.* there is a steep flux gradient). In a typical core calculation these areas are at the core boundary, near strongly absorbing media such as near a node where a control rod is inserted, and when two very different assemblies are placed next to each other [UO_2 and MOX (UO_2/PuO_2) assemblies]. Assembly discontinuity factors (ADFs) are also calculated by the lattice physics code. When single 2-D assembly calculations are performed there is no knowledge of the neighbouring nodes/fuel assemblies. The homogenisation process preserves the reaction rates in the infinite single assembly geometry. When the node/assembly of interest is put in the real reactor core it experiences a different environment than during the homogenisation process. This means that the homogenised flux solutions will not be continuous at the radial node/assembly boundary [the actual physical (heterogeneous) flux between the assemblies is continuous]. When material properties vary greatly this approximation leads to a large error, as the real core environment is very different from the infinite geometry with reflective boundary conditions. Such places are near nodes with control rods (CRs) inserted, at the reflector region, and are unlike fuel assemblies (*i.e.* uranium fuel next to MOX fuel). ADFs quantify the relative difference between the homogenised flux solution at the boundary and the actual heterogeneous transport solution at the boundary of the node/assembly. Using these, one can construct a correction to the flux at the node boundaries, which accounts for the real core environmental effects. The correction assumes a discontinuous homogeneous flux shape at the boundaries. In the usually employed methodologies the ADFs are treated as the rest of the cross-sections and are based on history and instantaneous dependencies. This consistent modelling is especially important during a transient, where the T-H and control variables are changing. It is most important to note that the greatest improvement is at the interface between the outer nodes and the reflector. There are steep flux gradients at the fuel/reflector boundary and large errors are produced when cross-sections are generated using single assembly calculations. The implementation of ADFs improves the leakage description between nodes/assemblies of different types and the error is reduced significantly.

ADFs are primarily used for the radial boundaries of each node. In the axial direction there are less discontinuities since the fuel is uniform. Axial changes in the properties are observed based on the history of the core, but this change is a continuous one in the fuel and can be accurately represented by the nodal decomposition. In the radial direction the inherent dislocations are caused by different types of fuel placed next to each other. In the axial direction, ADFs are most important at the interface between the top and bottom reflectors. As in the radial model this is the location at which the greatest errors can occur in the axial direction.

4.1.4 Dependencies on instantaneous parameters

The linear interpolation in multi-dimensional tables is widely used to obtain accurate reproducible results. This is dependent on the linearity of the cross-section with respect to the different dimensions (chosen principal feedback parameters) and the distance between the data points. Several studies have demonstrated that the cross-section dependence on fuel temperature, moderator temperature and pressure is not linear. This indicates that the number of points necessary for accurate results will have to be developed so that the behaviour of the cross-section is accurately modelled. The more points used in the development of the transient cross-section library, the greater the increase in the overall size of the library and the total number of spectral calculations.

The linear surface interpolation has been used for the last several years. The model employed is a standard linear interpolation model. The following Eq. (1) shows the formula used to determine the dependent parameter based on the selected independent parameters:

$$y = y_1\left(1-\frac{x-x_1}{x_2-x_1}\right)+y_2\left(\frac{x-x_1}{x_2-x_1}\right) \tag{1}$$

In this equation, x is the independent parameter for which the interpolation should be made. It lies between x_1 and x_2, and y is the new function value. This equation is evaluated for each of the independent parameters up to a maximum of four parameters. Recent modifications extended the capabilities of this subroutine to include extrapolation. If the values lie below the first table entry, the dependent values are derived from Eq. (2):

$$y = y_1\left(\frac{x_2-x}{x_2-x_1}\right)+y_2\left(1-\frac{x_2-x}{x_2-x_1}\right) \tag{2}$$

If the independent value is beyond the last table entry, the linear extrapolation is performed via Eq. (3):

$$y = y_1\left(1-\frac{x-x_1}{x_2-x_1}\right)+y_2\left(\frac{x-x_1}{x_2-x_1}\right) \tag{3}$$

In the case of entries that lie within the bounds of the cross-section table, the same interpolation is performed as described for the original method.

The advantages of using a linear interpolation scheme are that it is fast and extrapolation can be done if necessary. The disadvantage is that there is a loss in accuracy when this method is used due to the non-linearity of the cross-sections. This means that more points have to be used to obtain an acceptable accuracy. In order to achieve an acceptable precision, more spectral branch calculations have to be performed and the size of the library files becomes very large.

It is possible to develop incorrect cross-sections by using only density as a feedback parameter. The density (as a parameter) depends on void fraction, moderator temperature and pressure, and accounts for all the effects on number density due to changes in these parameters. It does not, however, account for the effect of moderator temperature on microscopic cross-sections. One solution to this problem is to fix the thermal-hydraulic state by adding another thermal-hydraulic parameter (*i.e.* moderator temperature or pressure). In so doing, the exact cross-section for this state can be determined. The second problem with this representation is embedded in the method used to determine the range of density values that the cross-sections should be based on. For PWRs the density range is based on the variation of moderator temperature and pressure for both steady-state and transient conditions.

The first iteration for the improved methodology involved adding moderator temperature as an independent parameter to the tables so as to remove the uncertainty in the thermal-hydraulic state. The second iteration involved developing the tables based on simultaneous variation of both density and temperature. This proved to be impossible, as the density is a function of the temperature and this fact introduces non-linear cross-section dependence. It is not possible to create “a space” which will completely cover all ranges of density and temperature during both steady state and transients.

The third iteration solves all of these problems and creates a more flexible methodology. This is accomplished by using pressure and temperature as the tabulated independent parameters for single-phase simulations (*i.e.* mostly for PWR applications). This method is also flexible in that it can easily be further extended. For example, when void is present (as in BWR calculations) the existing methodology proved to be very difficult to develop based on the wide range of densities, which compounded the calculation error of the cross-sections. In the new methodology this can simply be overcome by creating a second library containing void as an independent parameter. This is done by replacing the pressure dependence with void dependence. The pressure is replaced due to the fact that when void is present the prevailing conditions correspond to saturated two-phase, and pressure and temperature are no longer independent parameters. When one changes the other will change in an appropriate corresponding manner. This means that the thermal-hydraulic state cannot be accurately described by pressure and temperature alone. The inclusion of void fraction as an independent parameter will fix the thermal-hydraulic state. In these situations, pressure and temperature are dependent upon one other, so only one of the two is needed in the tables. It was determined that temperature has the greatest effect on cross-section behaviour and therefore this is the independent parameter which is used. Moderator temperature has a large effect on the behaviour of cross-sections during the slowing-down process. It is generally assumed that a neutron in the moderator has the same temperature as the moderator. This means that changes in the moderator temperature will cause changes in the neutron temperature and shift the spectrum accordingly.

4.1.5 Some consideration of needed data for the lattice physics code calculations

In order to provide the macroscopic cross-sections with adequate resolution several core regions have to be defined where each core region can, depending on the desired resolution, include various fuel types, cladding, coolant, fuel assembly shroud, water gap and control rods of diverse design. The input data needed for this type of calculations include specification of parameters related to the sought-for core state points (*i.e.* conditions for the base case and associated variations for the branch calculations), and material compositions used in various defined structures including specifications of burnable poison parameters. A detailed geometry layout has to be provided including assembly cross-sectional layout, fuel pin geometries, control rod geometries and also layout of internal water channels like the water cross in modern BWR assemblies. Other input specifications relate to burn-up quantities and can also include detector response options. More specifically the following can be mentioned:

The physical effects to be considered in the time-dependent 3-D diffusion code calculations should be reflected in the determination of the cross-sections and associated parameterisation against selected feedback variables. The heterogeneity effects that preferably should be considered include inner nodal structures, neighbour node conditions, axial variations of kinetics conditions (adequately selected core planes for the lattice code calculations), careful specification of node boundary geometries and conditions for the determination of ADFs and corner discontinuity factors (CDFs) as well as reflector cell boundaries.

In the calculations performed the macroscopic two-group (or few-group) reference cross-sections are obtained as well as microscopic cross-sections for relevant isotopes (B^{10}, Xe^{135}, Sm^{149}), as are their partial derivatives with respect to the selected feedback variables. The thus-obtained cross-section values (tables) may be generated independently of the burn-up cycle or can be made burn-up dependent as determined by their subsequent application in the nodal diffusion calculations. In the latter case the burn-up steps should be adequately selected if gadolinium poison (Gd_2O_3) fuel pins are used in order to accurately take into account the burn-up characteristics of those pins. In relation to the fuel rods with Gd_2O_3, content should be taken into consideration as regards the heterogeneous burn-up distribution from the fuel pellet surface and inwards, thus requiring the pellet to be divided into several concentric calculational nodes. Additionally, when reload fuel assemblies are included in the core the effect of ^{241}Pu decay should be considered as appropriate.

From the above information it is clear that the total amount of input can be rather comprehensive, although many codes (CASMO being one example) have reasonable default values for various parameters that preferably should be used to reduce the needed input data specifications.

Careful consideration of the input parameters is also required in view of the expected parameter ranges that can be encompassed in the later time-dependent 3-D diffusion code application. By this means the overall accuracy of the lattice physics code calculations can be improved. Some benchmark calculations performed with the CASMO-4 program including comparisons with both Monte Carlo analyses and criticality measurements revealed rather favourable results [57]. Additional validation results have been produced but most are of proprietary nature.

At NRI Rez, libraries with homogenised macroscopic cross-sections and kinetic parameters prepared by the lattice codes KASSETA or HELIOS are used for the WWER T-H/3-D neutronic analyses. The typical data structure includes the tabular form of two- or four-group homogenised macroscopic cross-sections for fuel assemblies, control rods and reflector assemblies. The cross-sections are tabular, and are dependent on enrichment, fuel assembly type (fuel assembly spacer grid type, fuel assembly shroud type and its thickness), fuel temperature, moderator temperature, moderator density, boron acid concentration in the coolant, power density and burn-up. The constants of poisoning as well as kinetic parameters are also provided in a similar tabular form. The reflector assemblies are represented as a mix with different volumetric portions of moderator and steel. The neutronic data are usually transformed from the library to a direct access file or they can be extracted by auxiliary programs and included directly into the 3-D neutronic kinetic code input deck.

The tabular form of homogenised cross-section libraries should cover the greatest possible extent of T-H parameters – especially fuel temperatures, moderator temperatures and densities. In the T-H/3-D neutronic codes the dependency of nodal macroscopic cross-sections on time-dependent T-H parameters is usually calculated from reference cross-sections and coefficients of parameterisation using linear or quadratic parameterisation formulas. The reference cross-sections and coefficients of parameterisation are prepared for selected T-H states. In some cases – for example during a control rod ejection accident – the fuel temperature peak can reach limits which are beyond the tabular values. In those cases the T-H/3-D neutronic codes have to perform some form of extrapolation beyond prepared states, a process which can be a source of uncertainties and even introduce incorrectness.

4.2 Diffusion code requirements

The state of the art of currently used multi-dimensional neutron kinetics models for LWR calculations of core time-dependent spatial neutron flux distribution includes the utilisation of 3-D neutron diffusion equation based on two neutron energy groups and with six groups of delayed neutron precursors. This has been found adequate for steady-state applications and for those transient applications for which direct validation has been possible [58]. The use of ADFs has proven to be sufficient for several applications. However, it is also understood that the utilisation of MOX fuels, with its higher Pu content, requires the inclusion of additional neutron energy groups for an accurate simulation [59]. The Pu isotopes have high absorption and fission resonances at around 1 eV (^{240}Pu absorption) and from 100 eV up to the keV ranges (^{239}Pu, ^{241}Pu absorption and fission).

The major calculation features of the neutron diffusion models, as can be illustrated by the PARCS code [53], include the ability to perform eigenvalue (k_{eff}), transient flux, xenon transient, decay heat and adjoint calculations. Pin power reconstruction capabilities are also available to obtain pin power and associated intranodal neutron flux distributions from the calculated nodal fluxes.

Burn-up (depleting) calculations are features included in several codes, SIMULATE-3 being one example, and can be used for core reloading and optimisation analyses as well as for on-line core following calculations. In relation to the USNRC code PARCS, associated burn-up calculations are performed by means of a separate program (DEPLETOR) which communicates with PARCS through a defined interface [60].

The 3-D capability provides the basis for realistic representation of the complete reactor core, although provisions are included to represent appropriate symmetry sections, *i.e.* half- and quarter-core sections, for computational efficiency. Carefully selected boundary conditions for the symmetry planes are paramount for adequate calculational results in those cases. One-dimensional capabilities are also usually available for simulation of transients with predominant axial neutron flux variations. It should be noted that subcritical conditions are in some cases not directly calculated. Those conditions are, from a mathematical perspective, obtained from partial differential equations representing a source term defined problem. However, the common practice for those cases, proven through an ever-increasing amount of analyses and comparisons to real reactor conditions, is also to deploy the eigenvalue calculations for the subcritical cases, but with the adjustment of some reactivity parameter (usually the boron concentration) to obtain critical conditions. The assumption applied to the thus-obtained result is that the quantified value of the adjusted parameter represents the amount of subcriticality for the analysed conditions.

4.2.1 Some consideration of needed data for the neutron diffusion code calculations

The core nodalisation for the diffusion code calculations is usually performed with one node per fuel assembly in the radial direction and with about the same node height axially, thus providing cubic calculational nodes. This is not a requirement and other configurations can be conceived, *e.g.e.g.* using only a half or quarter of the node size in the radial direction. It is important, however, to take into consideration the mapping strategy between the T-H system code core nodalisation and the associated nodalisation as used for the diffusion code. The coupling between the T-H and diffusion codes is usually made using node average values of T-H parameters, boron concentration and fuel temperature. More discussion of the coupling is provided in Chapter 5.

Important data for the neutron diffusion calculations apart from basic descriptions of the various core parts include tabulated macroscopic cross-sections as a function of T-H and fuel parameters, boron concentration and control rod positions (worths, used banks, etc.) and microscopic cross-sections for Xe and Sm including corresponding dependencies. Other data are decay constants and the ADFs for the fuel assemblies' four sides and CDFs for the corners. All these nuclear-related data are basically obtained from the 2-D lattice code calculations mentioned above for each specified composition. The geometrical details of each composition (*e.g.e.g.* fuel type) are "collapsed" in the cross-sections, etc., and consequently new cross-section data have to be produced if the geometry changes.

The data needed for the core description include the layout of the core in terms of location for each fuel assembly type as well as of reflector assemblies, and of control rod position. The nodal geometrical configuration has to be specified (number of nodes and sizes in the 3-D core domain) as well as the type of core boundary (reflective, zero flux or current, albedo). In case a pin power calculation is performed the pin power reconstruction methodology needs some input concerning the assembly pin layout and pin power form functions to be used.

Important transient input would be each control rod or control rod banks movement in the form of (for instance) tables providing position versus time.

If a true safety margin calculation is to be made, *e.g.e.g.* the DNBR in the PWR case, a subchannel calculation should preferably be performed in the critical zone (hot bundle). Obviously, prevailing transversal cross flows with associated enthalpy between the hot bundle and its neighbour assemblies have to be adequately transferred as boundary conditions to the subchannel analysis model. Also, pin power reconstructions are needed for the fuel pins in this case in order to obtain the prevailing pin power distributions.

It is noted that the current state of the art related to the neutron diffusion codes does not seem to allow a detailed transient simulation of the fuel pin performance. The geometrical and physical properties of the fuel can be expected to change during the course of a RIA transient and should ideally be properly accounted for in the modelling. It is envisioned that in case of a substantial local power increase fuel pin deformation could occur, eventually resulting in fragmentation and release of fission gases. This may have an effect upon neutron kinetic cross-sections as well as upon the thermal properties of the gas gap and of the fuel itself. The feedback in terms of fuel pin power and of power transmitted to the coolant might be non-negligible. No models addressing this type of phenomena seem to be currently implemented in the codes.

Chapter 5

COUPLING OF THE THERMAL-HYDRAULIC SYSTEM AND 3-D NEUTRONICS CODES

Two different approaches are generally utilised to couple T-H system codes with 3-D neutronic kinetics models: serial integration and parallel processing coupling. The serial integration approach includes modification of the codes, usually by implementing a neutronic section (*i.e.* subroutines, functions) into the T-H system code, but this can also be accomplished the other way around. In the parallel processing approach the T-H system and 3-D neutronic kinetics codes are executed separately and exchange needed data during the calculation. In the former case substantial programming effort is needed to properly achieve the integration, while in the latter case only minor modifications are made to already existing codes. In the latter case it is crucial to provide the data exchange between the two codes in carefully and properly selected time sequences; thus, great attention must be paid to the process of data transfer and the associated time control of the execution processes of the two codes.

The 3-D neutronics capability in the T-H system code is usually obtained through the external coupling between the system code and an adequate transient 3-D kinetics code, and is explicit in the time domain. An obvious advantage of this methodology is that the codes are isolated and can independently be updated and maintained. The kinetics model receives T-H data from the T-H system code, such as fuel temperatures, coolant void fraction, phasic densities and temperatures, and boron concentration, and returns the fuel power back to the T-H system code. The USNRC code packages RELAP5/PARCS and TRACE/PARCS are examples of this explicit coupling approach using "message passing" data transfer, in this case on a calculational mesh cell or node level, using data communication routines from the PVM package [61]. Other means of communication, using features directly available as part of the computer operating systems (TCP/IP communication through sockets, or using the concept of shared memory), thus providing gains in execution time, will eventually replace PVM communication routines in TRACE/PARCS [62].

Another example of coupled code packages is ATHLET/DYN3D. The reactor core model DYN3D [44] is coupled with the ATHLET code [34] in two different ways [63,64]. The first method uses only the neutron kinetics part of DYN3D and integrates it into the heat transfer and heat conduction model of ATHLET. The power distribution is transferred from the neutron kinetics of DYN3D to ATHLET and the fuel temperature, moderator temperature, moderator density and boron concentration are provided by ATHLET as feedback parameters for DYN3D. This is a very close coupling; the data have to be exchanged between all core nodes of the single models (internal coupling). Thus a large number of data have to be transferred. In the second coupling method, the whole core is cut out of the ATHLET plant model and the reactor core is completely substituted by the DYN3D code model (external coupling). The thermal-hydraulics of the whole NPP system is split into two parts, the first describing the thermal-hydraulics of the reactor core model and the second the thermal-hydraulics of the rest of the NPP coolant system. The interfaces between the two parts are located at the bottom and the top of the reactor core. The pressures, mass flow rates, enthalpies and boron concentrations are transferred at the interfaces. The exchange of these parameters is performed by the General Control and Simulation Module of the system code ATHLET [34]. All T-H models included in the DYN3D code can be used in this external manner of coupling. The use of larger time steps can cause oscillation of the pressure drop over the core and associated mass flow rates in this type of coupling. These phenomena could be damped by utilisation of low pass filter of first order with a recommended low pass filter time constant of 1 s or larger [64].

5.1 Some considerations related to the coupling between thermal-hydraulic and neutronics nodes

Choosing proper spatial mesh overlays or mappings between nodes of the core models in the T-H system and the 3-D neutronic kinetics codes for a given transient is a challenging task. Careful consideration has to be given to expected possible asymmetric and local core behaviour conditions, as the nodalisations of the core in the two models and their interrelationship have a great influence in determining the local core parameters and hence the power distribution during the simulated transient. Core transients revealing pronounced 3-D asymmetric characteristics constitute a basis for verifying the overall performance of the coupled code systems. The main objective in such calculations is to predict the core 3-D local T-H conditions and associated power distribution as correctly as possible. The verification procedure involves testing the functionality, the data exchange between different modules and the T-H/neutronics coupling, which should have been designed to model the combined effects determined by the interaction of T-H and neutronics.

The available T-H system codes have the option to model the core by means of several parallel channels, with possible additions of cross-flow radial connections (exemplified by the RELAP5 code), or by using a 3-D T-H component, included for instance in the TRAC-PF1 and TRACE codes. In the latter case another option is whether to use cylindrical or Cartesian geometry. The Cartesian option has a provision for a better geometrical correspondence between the T-H core layout and the neutronics core model. The nodes of the latter model (the default core layout being divided assembly-wise) can be directly coupled to a T-H cell and heat structure in both the radial and axial directions. The exact detailed mapping in that case is more easily perceived and provides improved spatial resolution in the coupled calculations.

Sensitivity studies on spatial mesh mappings for analysis of PWR control rod ejection accident (REA) [65] suggest that the local refinement of the Doppler feedback model does not necessarily improve the accuracy of the results. It was shown that such simulations were also sensitive to the spatial coupling schemes, particularly in the radial direction. While the impact of neutronics mesh refinement is well known, it has also been found that the local and global core power predictions are very sensitive to T-H nodalisation refinement. The obtained results indicate that the T-H feedback phenomena are non-linear and cannot be separated, especially in the case of REA analysis, where the local Doppler feedback plays a dominant role.

These conclusions have been emphasised through a comparative analysis of MSLB results with the 3-D core T-H models [66]. The results obtained indicated that the detail and geometry approximation of the core T-H can be an important source of deviations for local parameters, especially for the relative power and fuel temperature distribution in the vicinity of a stuck control rod. During the course of the MSLB transient, a power spike was seen at the position of the stuck rod. However, in the coarse mesh model this assembly was averaged with several of the surrounding assemblies while mapping the neutronics model to the T-H model. This had a significant effect of underestimating the feedback in this part of the core. On the other hand, the detailed model did more accurately predict the feedback (as a result of a better spatial feedback resolution), and therefore also the relative power shape and local safety parameters, near the stuck rod. Refinement in the neutronic or heat structure model did not impact the total power transient evolution during the MSLB simulation.

BWR cores contain a large number of fuel assemblies (up to about 800). The precise and detailed T-H and neutron kinetics modelling of such cores requires significant computational resources. Thus there is a great incentive for introducing nodal optimisation in the coupled T-H/neutronics calculations. Calculation resources could be substantially reduced when collapsing similar fuel assemblies into a single representative T-H channel while maintaining the detailed neutronics modelling. However, such collapsing has a tendency to smooth resultant reactivity feedback effects and thus the associated power distribution as well. Finding an optimised number of T-H channels while still retaining adequate resolution to capture important phenomena helps to improve the calculational accuracy and provides the possibility to extend the transient duration and to perform sensitivity analyses.

When mapping neutronic assemblies to T-H channels different requirements have to be fulfilled. For example one obvious requirement would be to map similar neutronic assemblies in terms of their design to one T-H channel. However, there are also other characteristics of the assemblies that must be considered in the mapping process, such as relative power, coolant flow, void distribution, type of bundle inlet throttling (orificing), type of fuel (enrichment), burn-up, etc. Other requirements relate to retaining core symmetry and characteristics. In addition, any expected transient asymmetric core T-H inlet conditions affecting certain assemblies more than others have to be taken into account when the mapping is being performed.

An often-applied approach is to rank the fuel assemblies according to their initial steady-state power level (which can be obtained from a core diffusion simulator like SIMULATE-3 [54]) and classify them into groups, keeping power ranges as narrow as possible. This approach is known as “power flattering” mapping. The mapping of non-identical (in terms of initial power level) assemblies to a single T-H channel smoothes the power distribution and the resultant reactivity feedback. Decreasing the number of channels leads to overestimation of local parameters demonstrated by the comparison of normalised radial power distribution.

The development of the spatial mappings to be applied for BWR stability analysis requires a more sophisticated approach based on a modal analysis method, especially in the case of regional instability analysis. The mapping of T-H channels is modelled considering not only the power peaks, orificing sizes and general T-H characteristics, but also the neutron flux fundamental and first azimuthal mode during the steady-state point. Specific discussion related to the BWR stability issue can be found in Ref. [1], Section 2.7.

In practical terms the detailed mapping not only relates specific neutronic assembly nodes to given T-H channel nodes or cells but also provides the weights between the two meshes. These weights, ranging between 0 and 1 inclusive, determine both the amount of neutronic power to the T-H and heat structure components as well as the associated T-H feedback to the neutronic nodes. The very details of the mapping process are presented in Ref. [67] using the TRACE/PARCS code package and the OECD MSLB benchmark model [19] as a working example. In relation to TRACE/PARCS, an automatic mapping process to reduce the effort needed to prepare the mapping information has also been developed [67]. This process takes data from both the TRACE and PARCS input files and generates the required mapping information without user intervention.

Chapter 6

NPP MODELLING AND TRANSIENT DATA NEEDED FOR VALIDATION PURPOSES

Analyses of events which have occurred in full-scale NPPs constitute an important source of information for evaluation of the behaviour of various safety systems during transient and possible accident scenarios, and will extend the experience base of NPP operations. Such data are also very important for validation of the advanced T-H system codes and will provide insights for adequate code usage and application. In relation to coupled T-H system and neutronic code analysis, NPP occurrences with associated core reactivity changes are in principle the only type of transients that can provide validation basis for true coupled behaviour. Thus such data are very valuable complements to other more basic T-H data, obtained from, *i.e.* specifically designed integral and separate test facilities, mostly in respect to the full scale and to the influence from the actual plant control systems. The data from the NPPs include various types of transients such as commissioning tests, start-up tests, specifically designed tests and of occurred events. All these transients, perhaps apart from those included in the last group, are naturally of rather mild character but can nevertheless be very useful for code validation to reveal code deficiencies and input model problems. Also, it is noted that some types of transients, for example BRW core instability transients, can only be adequately simulated in scaled-down test facilities with great difficulty, and NPP data in such cases are the only representative source of information for related code validation.

To be able to use data from occurrences in NPPs it is of utmost importance that relevant transient information, including actual core states, be saved by the NPP utilities in an orderly fashion. Regular data-saving procedures have nowadays become more and more part of the normal NPP operating procedures, and thus it can be expected that data from specific transient occurrences would be available. However, it is realised that the data acquisition is basically intended for off-line analyses of occurred events and thus usually relies on data measuring systems available as part of the normal NPP monitoring systems. Consequently, the amount of data saved during an event will not be too exhaustive and will also not necessarily be of a quality useful for code validation. Nevertheless, with a previously determined careful selection of measured signals to be saved and by using adequate sampling frequency the saved data can reveal crucial information concerning transient process conditions and can be suitable for code validations. It is also very informative and, in the case of code validation, even necessary that some estimate of measurement uncertainties be provided, including possible delays in measurement gauges and signal processing.

In recent years several international activities have been pursued to provide experience and validation information in relation to the applications of coupled T-H/neutronic code packages. These include both pure benchmark exercises (comparisons of code results) and what can be characterised as validation-type analyses for which cases measured data are also available, mostly from tests performed in full-scale NPPs, but also from occurred, unplanned plant transients. Such activities include international projects such as:

- PWR Benchmark on Uncontrolled Rods Withdrawal at Zero Power [68].
- Ringhals 1 Stability Benchmark [69].
- Forsmark 1 & 2 Boiling Water Reactor Stability Benchmark [70].
- OECD/NRC Boiling Water Reactor Turbine Trip Benchmark.
- VVER-1000 Coolant Transient Benchmark [71].
- Pressurised Water Reactor MSLB Benchmark [66].
- Shutdown of 2 of 6 Working MCPs at 100% N_{nom} on Dukovany NPP (VALCO activity) [72].
- TRAC-M/PARCS analysis of OECD Peach Bottom Turbine Trip [73].

In many of these activities the basic data required for the analyses were provided by the leading organisations with the objective to limit the amount of freedom of the participating organisations, thus concentrating on system modelling and evaluation of code behaviour. However, in relation to the data provided some general recommendations and guidelines can be formulated which are discussed below.

6.1 Model input data requirements

From the various activities including NPP T-H system and core modelling (see the international activities mentioned above), the overall requirements regarding the needed input data can be inferred. The amount of data needed to fully set up a NPP model from both a T-H and nuclear kinetics point of view is substantial. Some specific considerations have been provided in Chapters 3 and 4 for the T-H and nuclear kinetic input, respectively. Chapter 5 indicates considerations specifically related to the code coupling. As already mentioned, these considerations must be complemented with what is provided in the available code manuals, for instance the User's Guide and Input Requirements [32] and the User's Guidelines [33] of the RELAP5/MOD3.3 manual set. From such (and similar) sources the following list of input items can be compiled. The mentioned items indicate in a general perspective what should be taken into account when preparing a NPP input model to be used in coupled T-H system-neutronic simulations. This list is not exhaustive and naturally includes items not directly specific to the applications of coupled codes, but is general to any separate T-H system NPP modelling and separate core kinetics modelling. The list is divided into sections considering various parts of the NPP and can be used as an overall checklist that has to be appropriately applied to specific transients and NPP modelling.

- Reactor and loop system geometry:
  - Reactor pressure vessel and loop geometry.
  - Reactor internals geometry (LP, UH, UP, DC).
  - Heat transfer areas.
  - Structure material thicknesses and geometries.
  - T-H pressure losses in loops including frictional losses.
  - RPV and loop heat losses to the ambient surroundings.

- Core and fuel description:
  - Number of fuel assemblies.
  - Fuel assemblies' configuration and geometry.
  - Distance between fuel assemblies.
  - Active fuel length.
  - Fuel bundle composition (enrichments, etc.).
  - Fuel bundle loading pattern.
  - Bypass configuration and geometry.
  - Core inlet orifice distribution.
  - Core inlet orifice loss coefficients and flow areas.
  - Core outlet orifice loss coefficients and flow areas.
  - Bypass loss coefficients at bottom and top plate.
  - Loss coefficient at each spacer and flow area.
  - Fuel assembly frictional pressure drop.
  - Loss coefficients in transversal directions with associated flow areas if applicable.

- Core T-H data (in the three dimensions, if applicable):
  - Pin domain hydraulic diameters.
  - Pin domain flow areas.
  - Pin domain friction.

- Fuel rod data:
  - Number of fuel rods per channel (assembly).
  - Fuel rod lattice pitch.

- Fuel rod geometry:
  - Length of uranium section.
  - Length of non-uranium inlet section.
  - Length of non-uranium outlet section.
  - Fuel pellet radius.
  - Gas gap width.
  - Cladding outer radius.
  - Clad thickness.
  - Pellet effective density.

- Core heat transfer data:
  - Pellet thermal conductivity.
  - Pellet heat capacity.
  - Pellet-clad gap heat transfer coefficient.
  - Cladding thermal conductivity.
  - Cladding heat capacity.
  - Assembly shroud thermal conductivity and heat capacity, if applicable.
  - In certain cases (*i.e.* DYN3D/ATHLET) the following additional data for gas-gap heat transfer calculation is needed:
    - Reference gap width.
    - Reference fuel temperature.
    - Reference cladding temperature.
    - Cold gas pressure.
    - Helium mole fraction.

- Control rods:
  - Number of control rods.
  - Control rods' composition.
  - Control rods' type distribution (banks).
  - Control rods' insertion pattern.

- LPRM detectors:
  - Number of detector strings.
  - Number of detectors per string.
  - Core location of detector strings.
  - Axial positions of each detector.
  - Detectors time constant due to the signal delay.

- Nuclear cross-sections:
  - Node-wise homogenised two-group cross-sections for all fuel types.
  - Assembly-wise (node-wise) homogenised two-group cross-sections for axial and radial reflectors.

- Historical data:
  - Burn-up data.
  - Void history.
  - Control rod history.

- Kinetic parameters:
  - Node-wise homogenised values of six-group decay constants (λ).
  - Fractions of delayed neutrons (β), and neutron life times (l).

- Recirculation pump data:
  - Number of pumps.
  - Pump Q-H characteristics at nominal pump speed.
  - Pressure loss coefficients in recirculation loop.
  - Pump recirculation loop flow area.
  - Pump recirculation loop flow length.

- Steam separator data:
  - Number of steam separators.
  - Total separator flow area.
  - Steam separator height.
  - Pressure loss coefficients at inlet, steam outlet and fall-back outlet.
  - Steam separator characteristics if mechanistic model will be used.

- Steam and feed water line data:
  - Number of lines.
  - Total length.
  - Flow areas including inlet orifices.
  - Inlet loss coefficient.
  - Frictional data.

- Important (needed) control and trip system data:
  - Trip set points (possibly process dependent) and logics including hystereses.
  - Control system transfer functions.
  - Measuring gauge delays including dynamic effects from possible sense lines (P, DP).
  - Level measurement system, possible dynamics.

Specifically related to the specification of cross-sections as provided by the 2-D lattice physics code calculations, the following input data requirements can be noted:

- Fuel core:
  - Number of fuel assemblies.
  - Fuel loading pattern.
  - Fuel enrichments in the core.
  - Gadolinium fuel rods loading pattern.
  - Burnable poison pins (BP) loading pattern.
  - Initial burn-up per fuel assembly.
  - Shroud thickness.
  - Control rod banks distribution.
  - Barrel thickness.
  - Barrel outer dimensions.
  - Storage pool time per reloaded assemblies (decay periods).

- Fuel assembly:
  - Uranium weight.
  - Number of fuel rods.
  - Number of guide tubes.
  - Assembly pitch.
  - Active fuel height.
  - Pin rod pitch.
  - Pellet radius.
  - Clad thickness.
  - Guide tube outer radius.
  - Instrumentation tube thickness.
  - Instrumentation tube outer radius.
  - Number of fuel spacers.
  - Fuel spacer weight and material.
  - Sleeves weight and material.

- Control rods:
  - Absorber radius.
  - Absorber clad thickness.
  - Absorber clad outer radius.

- Burnable poison pins:
  - Burnable absorber thickness.
  - Burnable absorber outer radius.
  - Inner clad thickness.
  - Inner clad outer radius.

  - Outer clad thickness.
  - Outer clad outer radius.

- Operational data of the core:
  - Coolant pressure.
  - Coolant input temperature.
  - Coolant output temperature.
  - Nominal thermal power and distribution.
  - Assembly mass flow rate/velocity if present.

- Feedback variable range:
  - Expected initial boron concentration.
  - Expected cycle burn-up.

6.2 General recommendations regarding measured data

The NPP measured data in relation to occurred transients have traditionally been saved and stored on computer-platform-dependent media as part of the normal operational procedure at the plant utilities. An initial common practice seems not have been pursued in terms of what should be saved and for how long. Some transients had in an early stage been found to be "interesting" because they include plant operating aspects that could be important related to operating experiences and optimisations or because of safety considerations indicated by the utilities themselves or by the nuclear safety authorities. Thus different procedures for saving data have been applied and the data are usually not directly available to the international community involved in development and validation of NPP safety computer models and tools. However, in view of the increasing awareness of the advantage of using adequately validated computer programs, both for the assurance of plant safety and for operating optimisations, this situation has very much changed over the years. Today, large amounts of data are saved and can be used for different purposes, code validation being one of these. Still, it seems that when saving NPP data there are somewhat different approaches as to what should be saved and also related to the level of quality assurance of the data. This situation is expected to improve when clear general specifications are provided in this context.

In relation to the use of measured data from occurred transients in NPPs for code validation, the intended use of the data and the associated purpose set forth, measurements must satisfy some general requirements. It is clear that not all transients that have occurred in NPPs are suitable for general validation purposes and even fewer for the specific validation of the coupled T-H/3-D neutronic codes. Some general requirements must be met in this regard, primarily to ensure the quality of the data concurrent with some evaluation of the appropriateness of the data for being applied to validation purposes. The transients suitable to validate a coupled code package must reveal some clear evidence of nuclear kinetics feedback effects, caused for instance by the occurrence of core power asymmetries or other effects. It is mandatory and necessary that selected transients be correctly identified with a clear understanding of the temporal sequences occurring during the transients, expressed in terms of prevailing T-H and nuclear kinetics phenomena. Additional important information that must be available for making a general judgement as concerns data quality includes, *e.g.* general aspects of the reactor and reactor type in which the transient occurred, and date and time of occurrence.

The measured transient data have to include both data related to the core state (including burn-up) and RCS T-H conditions. These include a thorough and consistent description of the pre-transient core state and T-H system conditions. The data must provide information on the core loading, power and power distributions, loop and vessel coolant temperatures and flow rates, including the speed of the main pumps. Additional information required relates to auxiliary system temperature and flow rates, various levels (pressuriser and steam generator for PWRs, RPV downcomer for BWRs), boron concentration, and control rod position (various banks).

The core power and power distribution measurements are obtained from fixed LPRM detectors located in the core in BWRs and most often from detectors outside the RPV in PWRs [74], although in some PWRs fixed in-core detectors are available. The sensitivity of the LPRMs is dependent on the detector burn-up and the LPRMs are calibrated on a regular basis (approximately every 2-4 weeks). This is achieved through a system of movable fission or gamma detectors (TIPs) that can be transversed through the core in tubes adjacent to the LPRMs. Normalisation to core power is obtained by comparison with a core heat balance. In BWRs a LPRM string with four axially located detectors is provided for every "super cell" consisting of four fuel assemblies. The detectors are thus rather evenly distributed over the core. The APRM signals are typically constructed from a selected number of the available LPRMs. Normally, four different sets of LPRMs form four independent APRM signals which are used for reactor protection. Thus the APRMs can be, dependent on the LPRM selection, somewhat unevenly influenced from the different parts of the core. Consequently, the APRMs can reveal some degree of asymmetry in core power distribution.

In PWRs ex-core detectors are located in every quadrant, each including two or in some cases six axial levels. Again TIP calibrations are performed on a regular basis. It is envisioned that the ex-core detectors could perhaps reveal some influence from the conditions in downcomer and thus can have a higher uncertainty in readings than in-core detectors. In some PWRs fixed in-core LPRMs are used in addition to the ex-core detectors.

If any operator manual actions were performed prior to the inception of the transient these need to be adequately recorded. Moreover, the measured data also need to be properly recorded during an ample pre-transient steady-state time window to allow for thorough verification of the data for consistency and adequacy. In relation to the measured data the magnitude and units used must be clearly stated, and in case of units provided in per cent the reference values must be provided as well.

In relation to transient data it must be understood that many measured data, or more appropriate "signals" taken from the on-line data monitoring systems normally used, are severely restricted as concerns their time resolutions. It should be taken into account that the measurement signals are in many cases filtered and/or delayed. So, for instance, many loop temperature gauges (*e.g.* thermocouples in PWRs RCS coolant loops) are located such that the response time will be delayed, and the gauges themselves have large time constants. Flow rate signals are usually obtained by means of pressure differences across some flow path section or device and are often filtered for stability reasons, thus providing an erroneous transient flow rate measurement if taken directly. Knowledge of associated transfer functions or characteristics is essential to mitigate the adverse effect. Also, level signals are obtained from pressure differences and should preferably be free from dynamic influences so that accurate readings can be made. This means that the pressure taps must be located in such a way that essentially only the static pressure is recorded at each tap, the location of which can be a delicate task to determine (at design) in downcomer regions.

The cases discussed here are only examples of the factors that must be taken into account and evaluated in order to characterise the adequacy of the transient measured data. These points are valid both for data given in a time sequence and for "snapshots" of the state of the plant at a given point in time.

Measured NPP transient data to be used for validation purposes must be recorded with a scanning frequency of at least a reasonable minimum value. This minimum value is of course dependent on the transient occurrence, but so as to assign a generally applicable value 25 Hz has been recommended [75]. This value is high enough not to interfere with the expected NPP major process dynamics, thus allowing to reveal important dynamical aspects of the transient, while still not producing an exceptionally large amount of data. Further, the time period during which data are recorded and saved should include, as mentioned above, a few minutes of transient pre-history being of sufficient length to encompass all interesting and relevant events during the transient. Thus it can be expected that the total amount of recorded data points will be huge, though this seems to pose only a minor problem in light of the current availability of data storage equipments at reasonable costs. What could be more cumbersome is the need for thorough documentation and QA status of the data as indicated above and the manner in which the data are stored. Preferably, all recorded data should be stored in a format that is useable on all computer platforms, or at least should be easily converted to such a format. Normally this results in a somewhat less compressed format compared to what is provided by true platform-dependent binary formats, but this trade-off is much counterbalanced by the advantage of having the data in a readily transferable format.

As for the specific format of given data, when time sequences are at issue, it is recommended to use column layout, where the first column (or first two columns) states the date and time and the first few rows the parameter considered and the units used for the parameter. This is the typical format of any worksheet file (for example an Excel file layout) and is readily available on any modern computer platform.

As much measured transient NPP data as possible are of course always advantageous. As most measurements signals or channels have to be specified in advance in many NPP monitoring computers, some general indications of what at a minimum level would be required for coupled T-H/3-D neutronic code validation. Uncertainty figures on measured data would be very valuable but seem difficult to acquire on a regular basis. A minimum set-up of suitable measured data would thus comprise the following signals:

- APRM.
- LPRM.
- Fuel assembly inlet flow rate if available (BWR).
- Recirculation flow rate (BWR).
- RCS flow rates (PWR).
- RCS hot and cold leg temperatures (PWR).
- Reactor vessel pressure.
- Reactor vessel head temperature.
- Pressuriser pressure (PWR).
- Pressuriser level (PWR).
- Pressuriser spray flow and heater power (PWR).

- Pressuriser PORV operation if any (PWR).
- CVCS flow rates and temperatures (letdown, charging) (PWR).
- Steam flow rate.
- Reactor vessel downcomer level (wide and narrow ranges) (BWR).
- Feed water flow rate.
- Feed water temperature.
- Auxiliary feed water flow.
- Auxiliary feed water temperature.
- SG pressure (PWR).
- SG level (wide and narrow ranges).
- Turbine valve behaviour.
- Steam bypass valve behaviour.
- ADS behaviour (BWR).
- Steam line relief and safety valve behaviour.
- Pertinent control system and trip signals

6.3 NPP measured data (additions to the CRISSUE-S database)

As previously mentioned, several activities related to coupled T-H/neutronic code analyses have been and are being pursued. Many of those activities include direct validations against selected measured NPP data, and for those cases the measured data are usually saved within the specific project or in a more general framework, for instance the NEA Data Bank and USNRC Data Bank, and with specific restrictions concerning accessibility. Activities performed under the auspices of the European Commission CERTA-TN project [76] have been devoted to examining the feasibility of using a more general framework for saving and storing plant (and test facility) data using distributed databases, again with restricted and controlled accessibility. Other data are more readily available within specific project frameworks, although still disseminated on a case-by-case basis, as is the case for the ongoing CAMP, the activities and results of which are reported through the NUREG/IA report series.

Several transient scenarios have been simulated as part of the CRISSUE-S project, and a database of results has been gathered within the framework of the project, as already mentioned in Chapter 2 (Section 2.6). The database of results constitutes Annex 1 of the WP2 report [1] and results from coupled T-H/3-D neutronic calculations of transient scenarios in BWR, PWR, WWER-440 and WWER-1000 NPPs are included. The scenarios considered are part of the ensemble of transients recommended for safety analyses as detailed in Chapter 2 (Sections 2.2 through 2.4) of this document.

The main objective of this section is to provide some basic NPP data for validation of coupled T-H/neutronic models using suitable transients of reference plants. Associated measured data in the form of data files are only listed while the files themselves are separated from the general transient information. Specific dissemination levels can easily be applied to data and general information by this means.

In the following paragraphs some different NPP transients are presented with indications of associated data files: Ascó data, Oskarshamn 2 data, Peach Bottom 2 data, and Three Mile Island 1. The information provided for each case is organised into three subsections: Plant description, transient description and measured data. Note that all the data are in a general multi-dimensional format and have been provided in an easily readable way, the specific format depending on the type of data.

6.3.1 Ascó data

6.3.1.1 Plant description

Ascó is a two-unit 1 000 MWe PWR nuclear power plant of Westinghouse design in normal operation since 1984. In 1999, it underwent a power up-rating which boosted each unit from 960 to 1 028 MWe. After this up-rating, start-up tests included a loss of load.

6.3.1.2 Transient description

A power up-rating took place in the reload for cycle 13. After nuclear design, the safety evaluation was completed with several adjustments to the technical specifications. Among these changes, new protection equations were implemented. Simultaneously, and as a result of hot leg streaming studies, new filters were implemented for signals related to hot leg temperatures, in order to avoid spurious turbine run-backs. The design criteria of both modifications were fulfilling. The final global settings had to allow a loss-of-50%-load without reactor trip.

On 15 October 1999, as a start-up test for cycle 13, a loss-of-loss test was scheduled in Ascó unit 2. With the plant functioning under normal operating conditions, the demand had to drop from 100% to 50%, at a rate 200%/min, the maximum allowed. Shortly before, the plant was taken to its nominal operating conditions in a controlled way. Exhaustive measurements were performed. In particular, three "snapshots" of the state of the reactor were recorded with the in-core instrumentation, at powers of 30%, 77% and 100%.

Once the test was performed the reactor scrammed due to over-temperature. The reactor went to a safer situation and for this reason it seemed sensible to think the issue was operational, the practical flexibility of the plant being less than expected. After detailed analysis, it was concluded that the over-temperature set-point would never be reached in such loss-of-load after modifying the steam-dump opening signal by means of an adequate lead-lag filter, since the actual opening would take place some seconds before.

The loss-of-load was repeated successfully on 14 November 1999. The plant had been shut down for almost one month.

6.3.1.3 Measured data

The thermal-hydraulic data presented are the data recorded during the successful loss-of-load on 14 November 1999. As recommended, the data are provided in an Excel file, with the following data in columns:

- Reactor power.
- Average temperature.
- Reference temperature.
- Delta-T.
- Pressuriser pressure.
- Pressuriser level.
- Spray demand.
- Charging pump flow.
- Discharge flow.
- Turbine pressure.
- Secondary pressure.
- Steam flow (each loop).
- Steam generator level – narrow range (each loop).
- Feed water temperature.
- Feed water flow (each loop).
- Feed water header pressure.
- Feed water turbo-pump speed (1 and 2).
- Feed water delta-P set point.
- Feed water aspiration pressure.
- Over-temperature delta-T.
- Over-power delta-T.
- Control rod position (banks C and D).
- Control rod speed demand.
- Steam dump position.

The data acquisition system provides values of each aforementioned parameter every second, starting two minutes before the beginning of the actual transient and for 20 minutes.

The neutronic data available are the in-core instrumentation data, which provides three snapshots of the state of the reactor before the failed loss-of-load. These data are valid even if they were recorded one month before the actual transient, because the plant was shut down during that time.

The data in this case are not in a time sequence, so the format in which they are given is different. The data available are:

- For fission reaction rates:
  - Normalised measured detector fission reaction rates.
  - Total normalised fission reaction rate integrals.

- For the power distribution:
  - Measured and expected assembly power.
  - Radial power distribution.
  - Axial power distribution by fuel assembly.
  - Core average axial power distribution.
  - Average axial linear power (kW/ft).

- For the power tilts and axial offsets:
  - Measured and predicted axial offset.
  - Normalised measured and predicted power tilts (octant/quadrant).
  - Quadrant/octant relative power.
  - Quadrant/octant power tilt.

- For the power peaking factors:
  - Measured fxy(z) radial peaking factors in a x-y plane.
    *Note: fxy = nuclear fxy*fqe*fqu, where fqe is the engineering uncertainty and fqu is the nuclear measurement uncertainty.*
  - fq total local peaking factor at each axial point.
  - Limiting fq at each axial point.
  - f-delta-h factors (fuel assembly and hot rod).
    *Note: fdh = fdhn * fhe * fhu, where fhe is the engineering uncertainty and fhu is the nuclear measurement uncertainty.*
  - Axial peaking factors.

- For temperatures and enthalpies (thermocouples):
  - Measured thermocouple temperatures.
  - Measured delta temperatures.
  - Measured enthalpies.

6.3.2 Oskarshamn 2 data

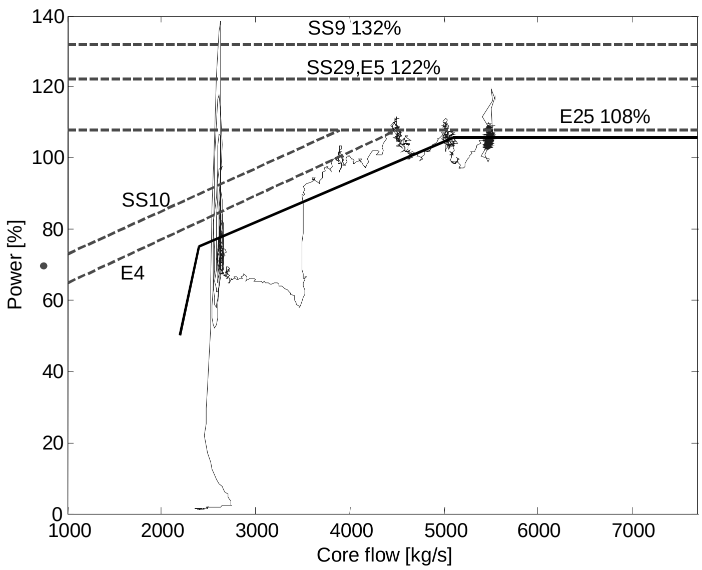
6.3.2.1 Plant description

The Oskarshamn 2 NPP is a BWR with external recirculation pumps designed by ABB Atom AB. Operation of the reactor commenced in 1974. There has been one power up-rating and the nominal thermal power is now 1 800 MW, which corresponds to about 106% of the initial power (1 700 MW). The electrical power is 630 MW. Note that all power references are made relative to the initial power.

The nominal circulation flow at full reactor power (106%) can range from 5 300 kg/s to 7 700 kg/s. For lower flow rates, the allowable reactor power is reduced. The minimum allowed flow is 2 500 kg/s. The maximum allowed recirculation flow is further limited at low reactor power due to the risk of pump cavitation.

Figure 6.1 displays the permitted operation region and the protection lines for Oskarshamn 2. The operating range is protected by various trip lines resulting in actions that are automatically initiated when the allowable range is exceeded. The SS9 signal initiates a reactor scram when the APRM exceeds 132% power. The SS10 signal initiates reactor scram, protecting against overpower in the power-flow dependent region. This signal reacts on filtered APRM, which means that the limit has to be exceeded for a certain time period in order to be activated. All other BWRs in Sweden use the unfiltered APRM signal along this protection line (this was also later modified in Oskarshamn 2). Before reaching scram conditions, the reactor shall encounter other conditions in order to avoid scram. The E25 signal on 108% filtered APRM causes preclusion of control rod withdrawal and coast-down of the recirculation pumps. When the conditions return to the allowed region, the E-signal is reset. The E4 signal has a similar function in the power-flow dependent region.

Figure 6.1. Power-flow map for Oskarshamn 2 with the operational states during the transient



6.3.2.2 Transient description

On 25 February 1999 the reactor operated at full power and a recirculation flow of 5 500 kg/s. Maintenance work was under way; the batteries were in the switchyard. The work was supervised by personnel from one of the reactors at the site (there are three BWRs located at the same site). The other reactors had not been notified that such work was in progress. When the tasks were completed, the battery supply was reinstalled, and during this manoeuvre the voltage was shortly interrupted. All three reactors recorded the voltage drop since they use the same switchyard. Oskarshamn 3 noted this short voltage drop but nothing else happened. In Oskarshamn 1, the voltage drop caused partial scram of the reactor.

A more complex situation occurred in Oskarsham 2. In principle, there were only two condition indicators that were affected by the voltage drop. One was the indicator “station not connected to the grid”, and the other was “station disconnected from the grid”. Although these indicators in common language are synonyms, the short voltage drop caused the first indicator to be “true” and the second to be “false”. In principle, the “true” indicator controlled the turbine and feed water operation and the “false” indicator controlled the reactor.

The general situation was as follows: the turbine control system interpreted the situation so that the external grid was lost, two of the five feed water pre-heaters were bypassed and preparations for supplying “in-house electricity consumption only” were initiated through control of the turbine valves and the dump valves. In fact the station was still connected to the grid.

The reactor control system did not sense the loss of external grid situation and operation continued. Scheduled actions to reduce power, such as stopping one pump and partial scram, were not taken.

The first turbine valve operation caused a peak in the reactor power of short duration. The reactor control terminated the power peak at 117% power and the reactor returned to the allowed operating range. Bypass of the feed water pre-heaters caused the feed water temperature to decrease. The decrease was faster than anticipated in the safety analysis of feed water transients for the reactor. The automatic level control maintained a high feed water flow to maintain the downcomer level, a fact that probably aggravated the temperature decrease at the core inlet.

The reactor responded to the decreasing temperature by increasing the reactor power. The reactor reached the E25 limitation at 108% power several times. Each time the recirculation flow was reduced and the E25 signal reset. Finally, the reactor was outside the allowed operating range.

In the power-flow map (Figure 6.1) the fluctuating line represents the movement of the operational state during the transient. The full lines mark the allowed area. Dotted straight lines indicate operational limits that execute forced reduction in flow and partial scram. Lines marked with “E” indicate forced reduction in flow, while lines marked with “SS” indicate scram or partial scram and forced reduction in flow.

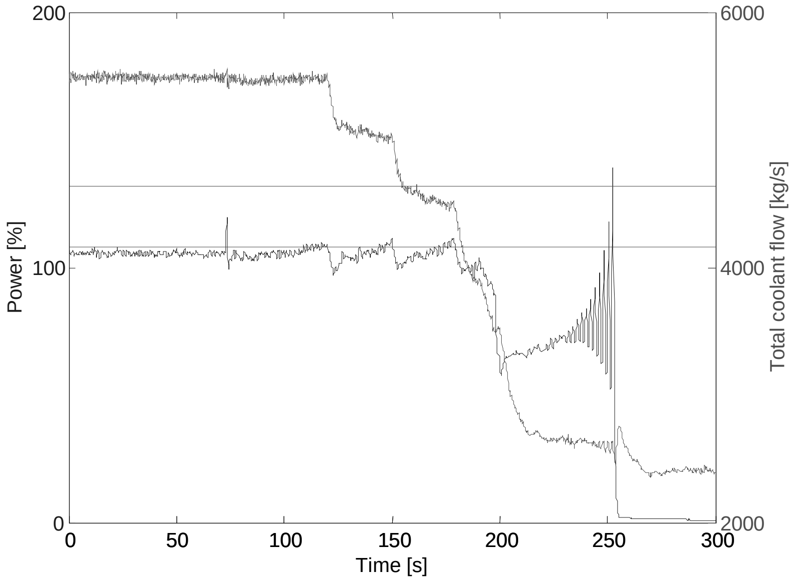
About two minutes after the event commenced, the operator initiated manual partial scram and forced reduction of the coolant flow. The operational state was close to the minimum flow of 2 500 kg/s and reactor power around 60-65%. The introduction of colder feed water with a high flow rate continued, caused by filtering in the controller. Core instability started with growing amplitude. Automatic scram was initiated at APRM = 132% and coolant flow was at 50%. The instability continued over a period of 18 seconds. The scram lines at lower power were adjusted to act with filtering for the APRM signals and therefore they were not reacting by the fast oscillation. The scram line at 132% was not filtered and finally reacted with scram of the reactor.

During scram the disturbance recorder stored measurement data. Important signals like reactor pressure, coolant flow, APRM and so on were stored. The disturbance recorder is constructed to save data starting some minutes before scram. The aim is to be able to evaluate an event after it occurs. Analysis of data from the instability event at Oskarshamn 2 clearly describes the event presented above.

The power increase at the beginning of the transient was caused by cold feed water flow. The overpower limit E25 at 108% was hit three times and power was reduced with flow reduction each time (Figure 6.1). Thereafter the operational state hit the slope of the E4 line with automatic reduction in flow. The reduction in power was, however, not sufficient to move the operational state back within the allowed area. Manual scram was therefore activated. It is clear from the figure that power oscillations started at APRM = 65% and coolant flow at 2 800 kg/s. Many reactor protection lines were passed during the oscillations but they did not react on the fast power oscillations, as they were filtered. The overpower line SS9 at 132% is not filtered and finally caused reactor scram.

APRM and coolant flow are presented as a function of time in Figure 6.2. The disturbance recorder saved 300 s with data and scram occurred after about 254 s. The spike in reactor power at 75 s indicates the time at which turbine was informed about loss of grid. Immediately afterwards, power increases and hits the overpower line at 108% (straight line in the figure). After four reductions in coolant flow, the operational state is outside the allowed area and manual scram was initiated just before 200 s. Instability begins about 30 s after manual scram; the amplitude of oscillation increases rapidly, and scram is initiated when APRM = 132%. This unfiltered scram line is indicated with a straight line in the figure.

Figure 6.2. Reactor power and coolant flow as a function of time during the transient



6.3.2.3 Measured data

The available measured data from the transient are as follows:

- APRM [%].
- Recirculation flow rate [kg/s].
- Reactor vessel pressure [bar].
- Steam flow rate [kg/s].
- Feed water flow rate [kg/s].
- Feed water temperature [°C].
- Reactor downcomer level [m].

The data were recorded with a sample frequency of 10 Hz over a period of 5 minutes.

6.3.3 Peach Bottom 2 data

6.3.3.1 Plant description

Peach Bottom Unit 2 is a single-cycle boiling water reactor (BWR/4) supplied by General Electric and licensed at 1 098 MWe (3 293 MWt). The reactor is operated by Exelon and is located in Peach Bottom, Pennsylvania. Commercial operation of the unit began in July 1974.

6.3.3.2 Transient description

Three planned turbine transients were performed at Peach Bottom Unit 2 prior to shutdown for refuelling at the end of cycle 2 in April 1977. These tests were conducted jointly by Philadelphia Electric Company, General Electric Company and the Electric Power Research Institute to investigate the effects of pressure transients generated in the reactor vessel following turbine trips from three different reactor power levels (47.4%, 61.6% and 69.1% rated power). The measured core neutron flux data and scram data for Turbine Trip Test 2 (TT 2) is shown in Figures 6.3 and 6.4. The reference reactor design data, including thermal-hydraulics system, core data and core neutronics data is provided in Ref. [20].

6.3.3.3 Measured data

In preparation for the sequence of turbine trip transient tests, additional data acquisition equipment was installed in Peach Bottom Unit 2 to supplement the plant control instrumentation. For each of the three tests, a total of 160 measurements were recorded for the resulting database, of which 84 monitored point and average neutron fluxes in the core, 31 were control rod pulse indicators, and 45 recorded other plant parameters such as pressures and temperatures. The data were recorded at 10 000 samples per minute for about one minute.

Figure 6.3. Measured neutron flux and inferred reactivity for TT2

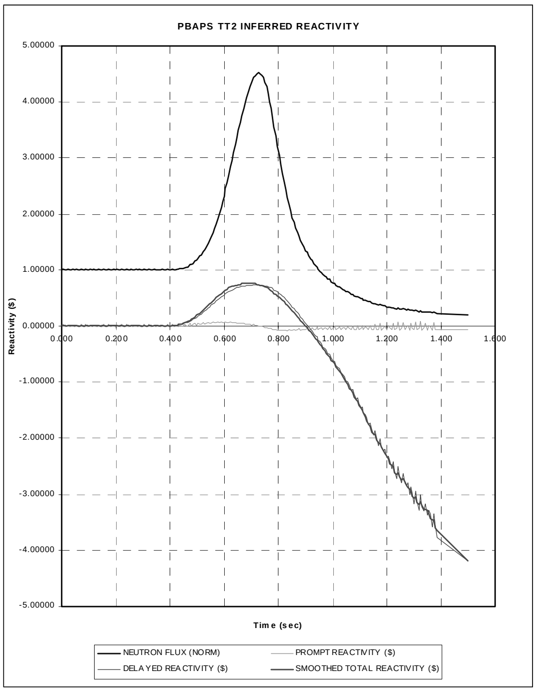
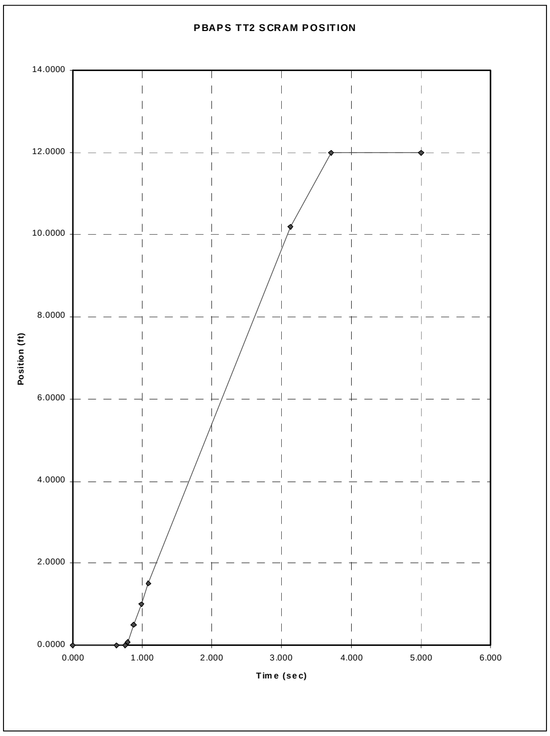


Figure 6.4. Scram data for PB TT2



In order to efficiently transfer the data to disk files, the data were streamlined by reducing the data frequency outside the time frame of greatest interest. All data values were retained within the respective time frame, and the remaining data values were reduced by a factor of ten from the original 166 samples per second to 16.6 samples per second. The associated time frame was the first 2.5 seconds for neutron flux measurements, the first 6.0 seconds for core response measurements, and the first 16.0 seconds for plant response measurements.

6.3.4 Three Mile Island 1 data

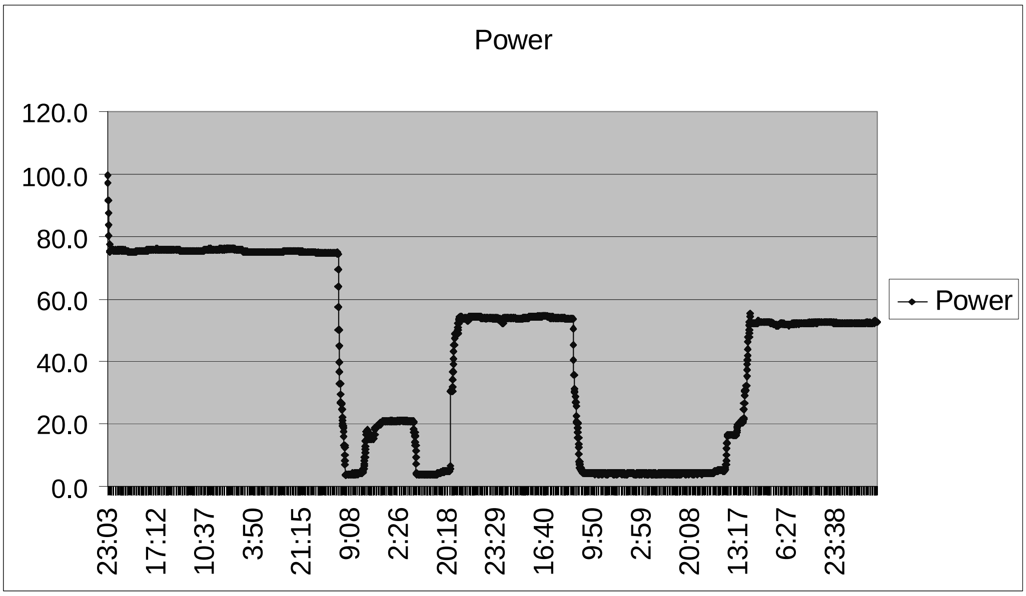
6.3.4.1 Plant description

Three Mile Island Unit 1 is one of two identical PWR units of Babcock & Wilcox design located on an island in the Susquehanna River close to Harrisburg in Pennsylvania, USA. The plant began commercial operation in 1974 with an electrical capacity of 900 MWe. It has two main coolant loops with two recirculation pumps and one steam generator in each loop. The steam generators are of the once-through design with allowances for the steam to be superheated when entering into the steam line. The design results in the steam generators being rather compact with relatively small water inventory compared to the more common u-tube design. Ref. [19] provides the reference reactor design data, including thermal-hydraulic system and core data.

6.3.4.2 Transient description

The specified transient is a power variation manoeuvre that was performed at the TMI-1 as part of the activities related to the main transformer replacement in 2001. The power variation time sequence is provided in Figure 6.5.

Figure 6.5. Power time sequence for TMI-1 power variation



6.3.4.3 Measured data

The available measured data from the transient are as follows:

- Best-estimate core power – 3 539°C.
- Primary core power – 1 710°C.
- In-core imbalance – 3 513°C.
- Average inlet temperature, Loop A – 1 752°C.
- Average inlet temperature, Loop A – 1 753°C.
- Control rod Group 5 position – 3 505°C.
- Control rod Group 6 position – 3 506°C.
- Control rod Group 7 position – 3 507°C.

Chapter 7

CONCLUSIONS

CRISSUE-S WP1 activities have concentrated on identifying NPP transients that could benefit from being re-evaluated using the coupled thermal-hydraulic/neutronic techniques, and on the input data required for the analyses. The identification of NPP transients should preferably be derived from some type of PSA evaluation to guide the direction and scope of deterministic coupled T-H/neutronic analysis, based on the twin merits of benefit and importance. The input data can be related to both T-H data and nodalisation, as well as to neutronic data including, for instance, nuclear cross-sections. Consideration is also given to NPP transients useful for the qualification of the coupled techniques.

In the following some conclusions are provided for each of the major topics covered in the report.

7.1 Probabilistic safety assessment

From the evaluation of current PSA applications it seems clear that they tend to focus on events resulting in end conditions involving various degrees of fuel damage. These events can include time windows during which substantial reactivity changes occur, but the PSA focus is still on fuel damage and is not directed specifically towards the reactivity transients as such. Thus it seems difficult to obtain from current PSA studies an evaluation and ranking of various “bounding” reactivity transients from a frequency of occurrence and consequence point of view. No clear indication can be obtained from current PSA studies to guide the direction and scope of deterministic coupled T-H/neutronic re-evaluation analyses of such transients based on the merits of benefit and importance. However, newer trends indicate that the PSA methodologies are increasingly used to evaluate and rank the importance of plant modifications and as well as to provide supportive information in the process of possible relaxation of technical specifications. As more experience is gained from such activities and also from recent activities related to risk-informed decision making and regulation, this situation is expected to change.

It must be emphasised, though, that results of recent coupled detailed best-estimate T-H/neutronic deterministic analyses make it clear that that there is an obvious potential to relax previously-used conservative core power factors. This will provide a basic condition for utilising the fuel more efficiently while still retaining, and possibly even increasing, the safety in terms of specified margins against fuel overheating and the risk of jeopardising fuel integrity.

7.2 Thermal-hydraulic system simulation

System thermal-hydraulic (T-H) codes based upon six partial differential balance equations, *i.e.* mass, momentum, and energy for each of the two phases, solved in 1-D geometry, constitute the current state of the art. Specific techniques are available to construct equivalent 3-D nodalisations for the core or the vessel regions as needed. Constitutive models are used to determine the evolution of the two-phase mixture. These models, being mostly of empirical nature, include simplifications and are often applied outside original validity ranges, thus providing uncertainties in calculational results. The used numerical solution methods also influence the results to an extent not fully quantified. Such uncounted uncertainties are expected to be further evaluated and quantified in the future, when results and conclusions from more validation analyses for specified types of transients will be at hand.

Several of the models implemented in the T-H codes can affect the reactivity responses but can not generally be influenced by the user (*e.g.* dynamic subcooled void formation), while others require carefully selected input (for instance the steam separator component in BWR modelling). In addition to these models some characteristics of the numerical schemes used (numerical diffusion being an example) are discussed that can similarly influence the reactivity responses and restrict the associated capabilities of current computational tools (*e.g.* pressure wave propagation phenomena inside vessels). For simulation of specific and critical phenomena other numerical schemes have been introduced. An example is to use a higher numerical scheme in simulations for which boron tracking is essential.

In relation to coupled T-H/neutronic analysis the advantage of using a consistent core nodalisation in the T-H simulation and in the kinetics analysis is emphasised. It is noted that in neutronic analysis it is customary to use up to 25 axial nodes, though this dense axial nodalisation has rarely been used in T-H core simulations. As normal T-H system code practice includes much coarser core nodalisation, only a very little T-H validation information is currently available for this dense type of nodalisation. Also it is not presently feasible to use too many radial nodes in the T-H core simulations, despite the current availability of powerful computers. It is thus preferable to use, if possible, the same nodalisation in the axial (vertical) direction in both the T-H and neutronics core modelling although some kind of lumping of radial nodes (*i.e.* fuel assemblies) is necessary. Rendering radial lumping a well-founded strategy must be achieved to reduce influences on the T-H result from fuel assembly differences and from the physical location of the assembly in the core.

It is also noted that modelling of the new types of fuel assemblies in the T-H system codes requires additional validation to fully assure the simulation of dynamical T-H characteristics. One example of this new design is the BWR fuel assembly consisting of four subassemblies separated by a double-wall structure forming an internal flow channel with a cruciform-shaped flow area. Also, the fuel rods have been made somewhat thinner and the number of fuel rods has been increased compared to previous design. Modified reactivity characteristics will similarly result from the changed design. Validation against appropriate experimental data is needed, but those data are currently mostly proprietary to the fuel vendors. Thus only tentative models can be prepared with an associated high uncertainty in transient behaviour.

7.3 Three-dimensional neutronic and fuel simulation

Similarly, some areas are discussed related to the preparation of the kinetics input models that can have importance in relation to coupled T-H/neutronic analysis. This includes both nuclear cross-sections and associated issues such as depletion and spectral effects, and assembly discontinuity factors, but also data related to the neutron diffusion code calculations. It is noted that even though approximate diffusion equations are typically adopted by the available computational tools, reference solutions obtained with more sophisticated techniques (including Monte Carlo and high-order deterministic neutron transport methodologies) do exist. These solutions are used to benchmark the results from diffusion equations. The problem of uncertainty seems be more directly related to the numerical solution methods than in the case of thermal-hydraulics.

Multi-dimensional interpolation algorithms are also discussed in relation to nuclear cross-section evaluation to properly account for variations at changes in the core state. Burn-up, spectral history and instantaneous dependence modelling together with the correct definition of assembly discontinuity factors constitute challenging aspects in neutron kinetics.

It is noted that the current state of the art related to the neutron diffusion codes does not seem to allow a detailed transient simulation of the fuel pin performance. The geometrical and physical properties of the fuel can be expected to change during the course of a RIA transient, at least in more severe cases, and ideally should be properly accounted for in the modelling. This may have an effect upon neutron kinetic cross-sections as well as upon the thermal properties of the gas gap and of the fuel itself. The feedback in terms of fuel pin power and of power transmitted to the coolant might be non-negligible. Currently, no production models seem be implemented in the codes to deal with this type of phenomena.

7.4 Coupling of thermal-hydraulic system and 3-D neutronic codes

The coupling issue is discussed with references to commonly used T-H/3-D neutronics code packages, *e.g.* RELAP5/PARCS and ATHLET/DYN3D. It is noted that specifying proper spatial mesh overlays or mappings between nodes of the core models in the T-H and in the 3-D neutronic kinetics codes for a given transient is a challenging task. Careful considerations must be taken for expected possible asymmetric and local core behaviour conditions, since the nodalisations of the core in the two models and their interrelationship have a large influence on determining the local core parameters and hence the power distribution during the simulated transient.

Requirements for proper mapping include that similar (in terms of design) neutronic assemblies should be mapped to one T-H channel. Other characteristics of the assemblies have to be considered in the mapping process, such as the relative power, coolant flow, void distribution, type of bundle inlet throttling, type of fuel (enrichment), burn-up, etc. Also, any expected transient asymmetric core T-H inlet conditions affecting certain assemblies more than others have to be taken into account when performing the mapping.

Core transients revealing pronounced 3-D asymmetric characteristics constitute a basis for verifying the overall performance of the coupled code systems. Sensitivity studies on spatial mesh mappings for analysis of PWR control rod ejection accident suggested that the local refinement of only the Doppler feedback model did not necessarily improve the accuracy of the results. It was shown that those simulations were also sensitive to the spatial neutronics/thermal-hydraulic coupling schemes, especially in the radial direction. It has also been found that the local and global core power predictions are very sensitive to T-H nodalisation refinement. The results obtained indicate that the T-H feedback phenomena are non-linear and cannot be separated, especially in cases where the local Doppler feedback plays a dominant role.

7.5 NPP modelling data and transient data for validation purposes

An overview is provided of the types of data that are required for preparing a plant model to be used for coupled T-H/3-D neutronics calculation. The data needed are listed, and include items not directly specific to the applications of coupled codes, but are general to any separate T-H system NPP modelling and core kinetics modelling.

A minimum set-up of suitable measured data is proposed for use in code validation. It is noted that the measured data must be properly recorded during an ample pre-transient steady-state time period to allow for examination of data consistency and adequacy. During the transient time period the dynamic characteristics of the measured signal values have to be fully described in terms of possible time delays and applied filtering. Another important aspect concerns whether there are any specific process influences on the gauges' readings due for instance to the locations in the plant components.

Some account of measurement sampling frequency is given. It is realised that the frequency must be sufficiently high to reveal important dynamic aspects during the course of the transient event, but on the other hand not prohibitively high (therefore producing an exceptionally large amount of data). A frequency of 25 Hz has been recommended. The format used when saving and storing measured data is discussed, and the importance of using an easily accessible layout (*e.g.* spreadsheet) is pointed out, as is the need to save the data in a computer platform independent format.

Within the WP1 activities additional data, provided in form of data files, have been provided to the CRISSUE-S database. The data comprise occurred transients in the Ascó (PWR), Oskarshamn 2 (BWR), Peach Bottom 2 (BWR), and Three Mile Island 1 (PWR) NPPs. The information given for each transient includes plant and transient description, and associated T-H and neutronic data.

REFERENCES

[1] CRISSUE-S Partners, *CRISSUE-S WP2-Report, Neutronics/Thermal-hydraulics Coupling in LWR Technology: State-of-the-art Report (REAC-SOAR)*, ISBN 92-64-02084-5, OECD (2004).

[2] CRISSUE-S Partners, *CRISSUE-S WP3-Report, Neutronics/Thermal-hydraulics Coupling in LWR Technology: Achievements and Recommendations Report (FINARES)*, ISBN 92-64-02085-3 OECD (2004).

[3] D'Auria, F., G.M. Galassi, *CRISSUE-S, Outline of the CRISSUE-S Project*, presentation at CRISSUE-S kick-off meeting, FIKS-CT-2001-00185, Pisa, Feb. 2002.

[4] D'Auria, F., *CRISSUE-S, Minutes of the Second CRISSUE-S Meeting*, FIKS-CT-2001-00185, Paris, Sept. 2002.

[5] *Proceedings of the Workshop on Advanced Thermal-hydraulic and Neutronic Codes: Current and Future Applications*, NEA/CSNI/R(2001)2/VOL1, May 2001.

[6] *The Use and Development of Probabilistic Safety Assessment in NEA Member Countries*, NEA/CSNI/R(2002)18, July 2002.

[7] Pershagen, B., *Light Water Reactor Safety*, Studsvik AB, Nyköping, Sweden, ISBN 0-08-035915-9, Pergamon Press (1989).

[8] *US Nuclear Regulatory Commission Standard Review Plan*, NUREG-0800.

[9] *Fuel Safety Criteria Technical Review. Results of OECD/CSNI/PWG2 Task Force on Fuel Safety Criteria*, NEA/CSNI/R(99)25, July 2000.

[10] *Proceedings of the Topical Meeting on RIA*, Aix-en-Provence, France, 13-15 May 2002, Summary and Conclusions from the General Discussion, NEA/CSNI/R(2003)8/Vol. 1 and 2, April 2003.

[11] "Probabilistic Safety Assessment: An Analytical Tool for Assessing Nuclear Safety", *NEA Issue Brief: An Analysis of Principal Nuclear Issues*, No. 8, Jan. 1992.

[12] *Nuclear Safety Criteria for Design of Stationary Pressurized Water Reactors*, ANS-51.1/N18.2 (1973).

[13] "Accident Analysis for Nuclear Power Plants. Guideline for PWR and WWER", *IAEA Safety Report Series*, to be issued, Vienna (2003).

[14] Jacobson, S., *Risk Evaluation of Local Dilution Transients in a Pressurised Water Reactor*, Ph.D. Thesis, Division of Energy Systems, Department of Mechanical Engineering, Linköping University, Sweden (1992).

[15] "Accident Analysis for Nuclear Power Plants. Guidelines for BWR", *IAEA Safety Report Series*, to be issued, Vienna (2003).

[16] *State-of-the-art Report on Boiling Water Reactor Stability* [*SOAR on BWRs*], OECD/NEA/CSNI, OECD/GD(97)13, Jan. 1997.

[17] Tinka, I., "Assumption and Acceptance Criteria for Analyzing Reactivity Related Transients and Accidents in Czech Republic", *5th Symposium of Atomic Energy Research*, Dobogoko, Hungary, 16-19 October 1995.

[18] *Anticipated Transients Without Scram for WWER Reactors*, a publication of the Extra Budgetary Programme on the Safety of WWER and RBMK Nuclear Power Plants, International Atomic Energy Agency, IAEA-EBP-WWER-12, December 1999.

[19] Ivanov, K.N., *et al*., *Pressurised Water Reactor Main Steam Line Break (MSLB) Benchmark. Volume I: Final Specifications*, OECD/NEA Nuclear Science Committee, NEA/NSC/DOC(99)8, April 1999.

[20] Solis, J., *et al*., *Boiling Water Reactor Turbine Trip (TT) Benchmark, (Peach Bottom 2). Volume I: Final Specifications, Revision 1*, OECD/NEA Nuclear Science Committee, NEA/NSC/DOC(2001)1, Oct. 2001.

[21] Ivanov, B., K. Ivanov, *VVER-1000 Coolant Transient Benchmark, Phase I (v1000CT-1). Volume I: Main Coolant Pump (MCP) Switching On – Final Specifications*, OECD/NEA Nuclear Science Committee, NEA/NSC/DOC(2002)6, June 2003.

[22] D'Auria, F., *et al*., "Application of RELAP5/MOD3 to the Evaluation of Loss of Feed Water in Test Facilities and in Nuclear Plants", *Nuclear Engineering & Design*, Vol. 141, No. 3, pp. 409-428 (1993).

[23] Bovalini, R., *et al*., "Analysis of Counterpart Tests Performed in BWR Experimental Simulators", *Nuclear Technology*, Vol. 97, No. 1 (1992).

[24] Tarantino, M., *et al*., "Simulation of a Large Break LOCA in WWER-1000 NPP by RELAP5/MOD3.2 Code", *ANS/HPS Student Conference*, Texas A&M University, College Park, Texas, March 2001.

[25] D'Auria, F., *et al*., "Application of the RELAP5-3D© to Phase I and III of the OECD/CSNI/NSC PWR MSLB Benchmark Related to TMI-1", *ASME/JSME Int. Conf. on Nuclear Engineering (ICONE-9)*, Nice, France, April 2001.

[26] Bousbia, S.A., F. D'Auria, "Analysis of the Peach Bottom 2 BWR Turbine Trip Experiment by RELAP5/MOD3.2 Code", *57th Conf. Italian Thermotechnic Association (ATI)*, Pisa, Italy, Sept. 2002.

[27] Groundev, P., M. Pavlova, "RELAP5/MOD3.2 Investigation of Exercise 1 of a WWER-1000 Benchmark", *OECD/NEA Starter Meeting for the WWER-1000 Coolant Transient Benchmark*, Rossendorf, Germany, May 2002.

[28] Maggini, F., *et al*., "BWR Stability Analysis by RELAP5/PARCS: Validation Against OECD/NEA Ringhals 1 Stability", submitted to the *ICAPP Conference*, Cordoba, Spain, May 2003.

[29] Uhle, J., B. Aktas, "USNRC Code Consolidation and Development Program", *Proceedings of the Workshop on Advanced Thermal-hydraulic and Neutronic Codes: Current and Future Applications*, NEA/CSNI/R(2001)2/VOL 1, Barcelona, Spain, April 2000.

[30] *CSNI Integral Test Facility Matrix for the Assessment of Thermal-hydraulic Codes for LWR LOCA and Transients*, NEA/CSNI/R(96)17, July 1996.

[31] *Separate Effects Test Matrix for Thermal-hydraulic Code Validation*, Volume I, NEA/CSNI/R(93)14/Part.1/Rev., Volume II, NEA/CSNI/R(93)14/Part.2/Rev., Sept. 1993.

[32] *RELAP5/MOD3.3 Code Manual. Volume II: User's Guide and Input Requirements*, Nuclear Safety Analysis Division, Information Systems Laboratories, Inc., Rockville, Maryland, Idaho Falls, Idaho, Dec. 2001.

[33] *RELAP5/MOD3.3 Code Manual. Volume V: User's Guidelines*, Nuclear Safety Analysis Division, Information Systems Laboratories, Inc., Rockville, Maryland, Idaho Falls, Idaho, Dec. 2001.

[34] Lerchl, G., H. Austregesilo, *ATHLET Mod 1.2 Cycle A*, *User's Manual*, Gesellschaft für Anlagen und Reaktorsicherheit (GRS) mbH, GRS-P-1/Vol. 1, Rev.1, March 1998.

[35] *RELAP5/MOD3.3 Code Manual, Volume VI: Validation of Numerical Techniques in RELAP5/MOD3.0*, Nuclear Safety Analysis Division, Information Systems Laboratories, Inc., Rockville, Maryland, Idaho Falls, Idaho, Oct. 1994 (revised version Dec. 2001).

[36] Spore, J.W., *et al*. (LANL, Los Alamos, New Mexico), J.H. Mahaffy, C. Murray (Pennsylvania State University, University Park, Pennsylvania), *TRAC-M/FORTRAN 90 (Version 3.0), Theory Manual*, LA-UR-00-910, July 2000.

[37] *TRACE V4.0, User's Manual*, prepared by F. Odar, C. Murray (U.S. NRC), R. Shumway, *et al*. (Information Systems Laboratories, Inc.), J. Mahaffy (The Pennsylvania State University), U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, May 2003.

[38] *RELAP5/MOD3.3 Code Manual. Volume I: Code Structure, System Models, and Solution Methods*, Nuclear Safety Analysis Division, Information Systems Laboratories, Inc., Rockville, Maryland, Idaho Falls, Idaho, Dec. 2001.

[39] *RELAP5/MOD3.3 Code Manual, Volume IV: Models and Correlations*, Nuclear Safety Analysis Division, Information Systems Laboratories, Inc., Rockville, Maryland, Idaho Falls, Idaho, Dec. 2001.

[40] Ashley, R., *et al*., *Good Practices for User Effect Reduction*, NEA/CSNI/R(98)22, Nov. 1998.

[41] Aksan, S.N., *et al*., "User Effects on the Thermal-hydraulic Transient System Codes Calculations", *Nuclear Engineering & Design*, Vol. 145, Nos. 1 & 2 (1993).

[42] Uhle, J., "US NRC T-H Code Development Activities", presented at the fall *CAMP Meeting 2000*, Annapolis, Maryland, 9-10 Nov. 2000.

[43] *RELAP5/MOD3.3 Transmittal. Developmental Assessment Cases. Marviken Critical Flow Tests 22 and 24*, Nuclear Safety Analysis Division, Information Systems Laboratories, Inc., Rockville, Maryland, Idaho Falls, Idaho, March 2002.

[44] Grundmann, U., S. Mittag, U. Rohde, *DYN3D2000/M1 for the Calculation of Reactivity Initiated Transients in LWR with Hexagonal and Quadratic Fuel Elements, Code Manual and Input Data Description*, 3rd Edition, Research Center Rossendorf, Inc., September 2001.

[45] Tietsch, W., "Applicability of RELAP for Special T&H Problems. Simulation of Pressure Waves in Branched Fluid Systems", presentation at the fall *CAMP Meeting 2002*, Alexandria, VA, 31 Oct.-1 Nov. 2002.

[46] Nylund, O., *et al*., *FRIGG Loop Project. Hydrodynamic and Heat Transfer Measurements on a Full-scale Simulated 36-rod BHWR Fuel Element with Non-uniform Axial and Radial Heat Flux Distribution*, FRIGG-4, R4-502/RL-1253, Dec. 1970.

[47] *RELAP5-3D© Code Manual. Volume I: Code Structure, System Models and Solution Methods*, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho 83415, INEEL-EXT-98-00834, Feb. 1999.

[48] Schmid, J., P. Král, J. Hejna, *Data for the Code RELAP5-3D©. The Discretization for VVER-440/213 Reactor*, Nuclear Research Institute Rez plc, ÚJV 11 436 T, May 2000 (in Czech).

[49] Lefvert, T., *BWR Stability Benchmark, (Ringhals 1), Final Specifications*, OECD/NEA Nuclear Science Committee, NSC/DOC(94)15, March 1994.

[50] *Summary of the First Workshop on OECD/NRC Boiling Water Reactor Turbine Trip Benchmark*, NEA/NSC/DOC(2000)22, Nov. 2000.

[51] Edenius, M., *et al*., *CASMO-4, A Fuel Assembly Burn-up Program*, program manuals (1995).

[52] Stamm'ler, R., *HELIOS – 1.6 Methods*, Studsvik ScandPower, April 2000.

[53] Joo, H.G., *et al*., *PARCS, A Multi-dimensional Two-group Reactor Kinetics Code Based on the Non-linear Analytic Nodal Method*, PU/NE-98-26, Purdue University, Sept. 1998.

[54] Smith, K.S., *et al*., *SIMULATE-3, Advanced Three-dimensional Two-group Reactor Analysis Code Methodology*, STUDSVIK/SOA-92/02 (1992).

[55] Watson, J., *et al*., "Cross-section Generation Methodology for Three-dimensional Transient Reactor Simulations", *TANSAO*, 77, p. 175 (1998).

[56] Watson, J., and K. Ivanov, "Improved Cross-section Modelling Methodology for Coupled Three-dimensional Transient Calculations", *Annals of Nuclear Energy*, 29, pp. 937-966 (2002).

[57] Demazière, C. (CEA), *Boiling Water Reactors with Uranium-plutonium Mixed Oxide Fuel. Report II: A Survey of the Accuracy of the Studsvik of America CMS Codes*, Chalmers University of Technology, Feb. 1998-June 1999.

[58] Diamond, D.J., "Multi-dimensional Reactor Kinetics Modelling", *Proceedings of the OECD/CSNI Workshop on Transient Thermal-hydraulic and Neutronics Codes Requirements*, NEA/CSNI/R(97)4, 5-8 Nov. 1996.

[59] Uhle, J., "Thermal-hydraulic Code Development and Applications", presented at the fall *CAMP Meeting 2001*, Washington DC, 25-27 Oct. 2001.

[60] Staudenmeier, J., "New PARCS Cross-section Model", fall *CAMP Meeting 2002*, Alexandria, VA, 31 Oct.-1 Nov. 2002.

[61] Geist, A., *et al.*, *PVM: Parallel Virtual Machine. A User's Guide and Tutorial for Networked Parallel Computing*, The MIT Press, Cambridge, Massachusetts, London, England (1994).

[62] Mahaffy, J., "Recent Development Activity on the USNRC Consolidated Code", presented at the fall *CAMP Meeting 2002*, Alexandria, VA, 31 Oct.-1 Nov. 2002.

[63] Grundmann, U., D. Lucas, S. Mittag, U. Rohde, *The Further Development and Verification of a Three-dimensional Nuclear Model for VVER Reactors and its Coupling with the Plant Transient Code ATHLET*, Report FZR-84, Rossendorf, April 1995 (in German).

[64] Grundmann, U., S. Kliem, D. Lucas and U. Rohde, "Coupling of the Thermohydraulic Code ATHLET with the 3-D Neutron Kinetic Core Model DYN3D", *Proceedings of the 6th AER Symposium on VVER Reactor Physics and Safety*, pp. 179-191, Kirkkonummi, Finland (1996).

[65] Todorova, N., K. Ivanov, "PWR REA Studies Using Different Coupling Options of TRAC-PF1/NEM", *ANS TANSAO*, 82, pp. 240-242 (2000).

[66] Todorova, N., *et al*., *Pressurised Water Reactor Main Steam Line Break (MSLB) Benchmark. Volume 3: Results of Phase II on 3-D Core Boundary Condition Model*, OECD/NEA Nuclear Science Committee, US Nuclear Regulatory Commission, NEA/NSC/DOC(2002)12 (2002).

[67] *PARCS, A Multi-dimensional Two-group Reactor Kinetics Code Based on the Non-linear Analytic Nodal Method*, user manual, draft Jan. 2003, Provided on the TRACE Version 4.000 beta Transmittal CD, April 2003.

[68] Fraikin, R., *PWR Benchmark on Uncontrolled Rods Withdrawal at Zero Power. Final Report*, OECD/NEA Nuclear Science Committee, NEA/NSC/DOC(96)20, Sept. 1997.

[69] Lefvert, T., *Ringhals 1 Stability Benchmark. Final Report*, OECD/NEA Nuclear Science Committee, NEA/NSC/DOC(96)22, Nov. 1996.

[70] Verdú, G., *et al.*, *Forsmark 1 & 2 Boiling Water Reactor Stability Benchmark. Final Report*, OECD/NEA Nuclear Science Committee, NEA/NSC/DOC(2001)2, June 2001.

[71] *OECD/NRC Boiling Water Reactor Turbine Trip Benchmark – Third Workshop (BWR-TT3) and Starter Meeting for the VVER-1000 Coolant Transient Benchmark (V1000-CT)*, OECD/NEA Nuclear Science Committee, NEA/NSC/DOC(2002)11, Rossendorf, Germany, May 2002.

[72] Hádek, J., J. Macek (NRI, Czech Republic), *Project Meeting VALCO. Shutdown of 2 from 6 Working MCPs at 100% N_{nom} on Dukovany NPP*, VTT Helsinki, Finland, April 2002.

[73] Lee, D., T. Downar, A. Ulses, B. Akdeniz, K. Ivanov, "Analysis of the OECD Peach Bottom Turbine Trip 2 Transient Benchmark with the Coupled Neutronic and Thermal-hydraulic Code TRAC-M/PARCS", *PHYSOR 2002*, Seoul, Korea, Oct. 2002.

[74] *Proceedings of the Workshop on Core Monitoring for Commercial Reactors: Improvements in Systems and Methods. Annex 1: On-line Core Monitoring for BWR and PWR. Overview and Comments on Operational Experience and Further Development*, OECD/NEA Nuclear Science Committee, presented by T. Lefvert, Stockholm, Sweden, Oct. 1999.

[75] Hedelsjö, M., *et al*., *Proposal for Past-history Recordings of Transient Occurrences in Swedish Boiling Water Reactors*, project prepared during the lectures in reactor technology, Royal Institute of Technology, Department of Reactor Technology, Stockholm, Oct. 1995 (in Swedish).

[76] Addabbo, C., *et al*., *CERTA-TN, Consolidation of Integral System Experimental Databases for Reactor Thermal-hydraulic Safety Analysis*, European Thematic Network, Final Report, European Commission, Dec. 2002.

LIST OF ABBREVIATIONS

A

ABB	Asea Brown Boveri
ABWR	Advanced Boiling Water Reactor
ACC	Accumulator
ACMFD	Analytical Coarse Mesh Finite Difference Method
ADF	Assembly discontinuity factor
AEC	*See* US AEC
AFENM	Analytical Function Expansion Nodal Method
AGCR	Advanced Gas-Cooled Reactors
AGR	Advanced Gas Reactor
AHTLM	Adaptive High-order Table Look-up Method
ANAV-UP	Asociación Nuclear Ascó-Vandellòs, Technical University of Catalonia
ANM	Analytic Nodal Method
ANS	American Nuclear Society
AOA	Axial offset asymmetry (guidelines)
APA	Advanced plutonium assembly
APRM	Average power range monitor
AP-1000	Advanced PWR (1 000 MWe)
ASME	American Society of Mechanical Engineering
ASTM	American Society of Testing and Materials
ATHLET	Analysis of Thermal-hydraulics of Leaks (*code*)
ATWS	Anticipated transient without scram
AZ	*Russian acronym for scram*

B

BDBA	Beyond design basis accident
BE	Best-estimate
BFSB	BWR full size bundle test
BiLU	Blockwise incomplete, or biconjugated lower upper (*numerics pre-conditioner*)
BOA	Boron-induced offset anomaly
BOC	Beginning of (fuel) cycle (*into the reactor core*)
BOP	Balance of plant
BP	Burnable poison
BPLU	Border Profiled Lower Upper Matrix Solver (*numerical method*)
BU	Burn-up
BWR	Boiling water reactor
BWRS	BWR stability
BWRTT	Boiling water reactor turbine trip (*see also* TT)
B&W	Babcock & Wilcox

C

CAEC	Czechoslovak Atomic Energy Commission
CANDU	Canadian Deuterium Uranium
CDF	Corner discontinuity factor
CEA	Commissariat à l'Énergie Atomique
CERCER	Ceramic matrix and ceramic fuel
CERMET	Ceramic fuel and metallic matrix
CFD	Computational fluid-dynamics
CFR	Code of Federal Regulation (*US NRC*)
CHF	Critical heat flux
CIAU	Code with capability of internal assessment of uncertainty
CILC	Crud-induced localised corrosion
CL	Cold leg
CMF	Common mode failure
CMFD	Coarse mesh finite difference
CPM	Collision Probability Method
CPU	Central process unit
CR	Control rod
CRGT	Control rod guide tubes
CRHFT	Core region at high (centreline) fuel temperature
CRHST	Core region at high (fuel rod) surface temperature
CRP	Co-ordinated research project
CRISSUE-S	Critical Issues in Nuclear Reactor Technology: A State-of-the-art Report
CSAU	Code Scaling, Applicability and Uncertainty (*US NRC uncertainty method*)
CSC	Cross-section code
CSNI	Committee on the Safety of Nuclear Installations
CT	Coolant transient
CVCS	Chemical and volume control system

D

DBA	Design basis accident
DC	Downcomer (*of RPV*)
DEC	Département d'Études des Combustibles (*CEA Cadarache*)
DF	Discontinuity factor
DNB	Departure from nucleate boiling
DNBR	DNB ratio
DO	Dry-out
DOE	Department of Energy (*US*)
DR	Decay ratio
DUPIC	Direct use of spent PWR fuel in CANDU reactors
DW	Density wave (*originated*)

E

EBA	Enriched boron addition
ECCS	Emergency core cooling system
ENAC	European Nuclear Assistance Consortium
EOC	End of cycle
EOP	Emergency operating procedure
EP	External (recirculation) pump (*BWR*)
EPMA	Electron probe micro analysis

EPRI	Electric Power Research Institute
ES	Eigenvalue separation
ESFAS	Engineered Safety Features Actuation System
EU	European Union
F	
FA	Fuel assembly
FEBE	Forward-Euler, Backward-Euler (*ATHLET module*)
FGR	Fission gas release
FMS	Fuel management system
FSAR	Final safety analysis report
FP	Full power or fission product
FR	Fast (neutron) reactor
FTC	Fuel temperature coefficient
FW	Feed water
FWHM	Full width (of the concerned peak) at half maximum
FZR	Forschungszentrum Rossendorf (*near Dresden, Germany*)
G	
GCSM	Transients General Control Simulation Module (*ATHLET module*)
GE	General Electrics
GI	General interface
GMRES	Generalised minimal residual algorithm
GRS	Gesellschaft fuer Anlagen- und Reaktorsicherheit (*also ID for uncertainty method*)
H	
HCO	Heat Conduction Objects (*ATHLET module*)
HECU	Heat Transfer and Heat Conduction (*ATHLET module*)
HFP	Hot full power
HL	Hot leg
HOSG	Horizontal (tubes) steam generator
HPIS	High pressure injection system
HPLWR	High-performance LWR
HT	Heat transfer
HTA	Heat transfer area
HTC	Heat transfer coefficient
HTGR	High-Temperature Gas Reactor
HTR	High-Temperature Reactor
HWR	Heavy Water Reactor
HZP	Hot zero power
I	
IAEA	International Atomic Energy Agency
IASCC	Irradiation-assisted stress corrosion cracking
ICE	A numerical solution method
ID	Identification
IET	Integral Effect Test (*facility*)
IFPE	International Fuel Performance Experiment
IGSCC	Intergranular stress corrosion cracking
IMF	Inert-matrix fuel
IP	Internal (recirculation) pump (*BWR*)

IRI	Interfaculty Reactor Institute (*Delft University, The Netherlands*)
ITF	Integral test facility
IWGATWR	IAEA Int. Working Group on Advanced Technologies for Water-cooled Reactors
J	
JP	Jet pump (*BWR*)
K	
KAERI	Korean Atomic Energy Research Institute
KKL	Leibstadt NPP
KTH	Kungl. Tekniska Högskolan
KWU	KraftWerk Union
L	
LBLOCA	Large break loss of coolant accident
LMFR	Liquid-metal fast reactor
LOCA	Loss of coolant accident
LOFW	Loss of feedwater
LOOP	Loss of off-site power
LPRM	Local power range monitor
LPIS	Low pressure injection system
LWR	Light water reactor
M	
MCP	Main coolant pump
MFT	Maximum fuel (centreline) temperature
MIT	Massachusetts Institute of Technology
MLIV	Main loop isolation valves
MOX	Mixed U-Pu oxide nuclear fuel
MPI	Multi-processor interaction
MSIV	Main steam isolation valve
MSLB	Main steam line break
MTU	Metric tons of uranium
MWD	Megawatt-day
N	
NACUSP	Natural circulation and stability performance of BWRs
NC	Natural circulation
NCM	Nodal Collocation Method
NCTH	Nuclear coupled thermal-hydraulics
NEA	Nuclear Energy Agency
NEM	Nodal Expansion Method
NEUKIN	Neutron Kinetics (*ATHLET module*)
NK	Neutron kinetics
NKC	Neutron kinetics code
NMCA	Noble metal chemical application, or noble metal clad assembly
NPP	Nuclear power plant
NRC	Nuclear Regulatory Commission (*US*)
NRI	Nuclear Research Institute (*Czech Republic*)
NSC	Nuclear Science Committee
NUGG	Natural Uranium Gas-Graphite
NUPEC	Nuclear Power Engineering Test Center

O

OECD	Organisation for Economic Co-operation and Development
ODE	Ordinary differential equations
O′M	Oxygen-to-metal (*ratio*)
ORNL	Oak Ridge National Laboratory
OTSG	Once-through steam generator

P

PAMS	Post-accident monitoring system
PBMR	Pebble bed modular reactor
PCCI	Pellet-cladding chemical interaction
PCI	Pellet-clad interaction
PCMI	Pellet-clad mechanical interaction
PCT	Peak cladding temperature
PDE	Partial differential equations
PEN	Polynomial Expansion Nodal (*method*)
PF	Peak factors (*for linear power of fuel pins*)
PHWR	Pressurised HWR
PIE	Post-irradiation examination
PORV	Pilot-operated relief valve
PRZ	Pressuriser
PRPS	Primary reactor protection system
PSA	Probabilistic safety assessment
PSU	Pennsylvania State University
PVM	Parallel virtual machine
PWR	Pressurised water reactor
PZ-1	First protection level (*Russian acronym*)
PZ-2	Second protection level (*Russian acronym*)

Q

QA	Quality assessment
QC	Quality control

R

RBMK	Boiling water cooled/graphite moderated (*Russian reactor*)
RCCA	Rod cluster control assembly
RCS	Reactor coolant system
REA	Rod ejection accident
REAC	Reactivity accidents (*ATWS, RIA, BWRS, boron-dilution, low power*)
RFP	Robust Fuel Program (*EPRI*)
RG	Regulatory guide
RHS	Right-hand side
RIA	Reactivity-initiated (or induced) accident
RIT	Royal Institute of Technology (*see also KTH*)
RPS	Reactor protection system
RPV	Reactor pressure vessel
RSK	Licensing guidelines (*Germany*)
R&D	Research and development

S

SAPR	Semi-analytical Perturbation Reconstruction (*numerical module*)
SAR	Safety analysis report
SBLOCA	Small break loss of coolant accident
SCC	Stress corrosion cracking
SEGFSM	Special Experts Group on Fuel Safety Margins
SET	Separate Effect Test (*facility*)
SG	Steam generator
SGTR	Steam generator tube rupture
SI	Safety injection
SIT	Safety injection tanks (*used as synonymous of ACC*)
SKI	Statens Kärnkraftinspektion (*Swedish Nuclear Power Inspectorate*)
SL	Steam line
SOAR	State-of-the-art report
SONS	State Office for Nuclear Safety (*Czech Republic*)
SOR	Successive Over-relaxation (*numerical method*)
SPDS	Safety parameter display system
SRV	Steam relief valve
STP	Standard temperature and pressure
SYS-TH	System thermal-hydraulics

T

TAMU	Texas A&M University
TD	Thoria-based fuels
TFD	Thermo-fluid-dynamics (*ATHLET module*)
TFO	Thermo-fluid-dynamic Object (*ATHLET module*)
T-H	Thermal-hydraulics
THSC	Thermal-hydraulics System Code
TIN	Transverse integrated nodal
TIP	Traversing in-core probe
TMI-1	Three Mile Island Unit 1
TPEN	Triangle-based Polynomial Expansion Method
TT	Turbine trip (*in BWR*)
TTEF	Total thermal energy released to the fluid (*during the calculated transient*)

U

UMAE	Uncertainty method based on accuracy extrapolation
UP	Upper plenum
UPISA	University of Pisa
UPTF	Upper plenum test facility
URB	Accelerated off-loading of the unit (*Russian acronym*)
UT	Ultrasonic (cleaning) technology
UTSG	U-tubes steam generator
US	*See* USA
USA	United States of America
US AEC	US Atomic Energy Commission
UVA	University of Valencia

V

VALCO	Validation of coupled neutronics/thermal-hydraulics codes for WWER reactors
VUJE	Nuclear Power Plant Research Institute (*Trnava*, *Slovak Republic*)
VVER	*See* WWER

W

WANO	World Association of Nuclear Operators
WESE	Westinghouse Energy Systems Europe
WG	Weapons grade
WWER	Water-cooled Water-moderated Energy Reactor

X

XRF	X-ray fluorescence
XS	Xenon samarium

Additional abbreviations

0-D	Zero-dimensional (point model)
1-D	One-dimensional
2-D	Two-dimensional
3-D	Three-dimensional

OECD PUBLICATIONS, 2 rue André-Pascal, 75775 PARIS CEDEX 16
Printed in France.