Nuclear Science ISBN 92-64-02085-3

Neutronics/Thermal-hydraulics Coupling in LWR Technology, Vol. 3

CRISSUE-S – WP3: Achievements and Recommendations Report

5th EURATOM Framework Programme (1998-2002)

Università di Pisa

NEA No. 5434

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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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 third WP and summarises the results, selects the most important findings and indicates the industry position on related subjects. It emphasises achievements, highlights the most important conclusions reached in the second WP and briefly refers to the first WP.

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. Several of the graphics in the report are in colour; interested readers can request a colour version of the report on CD-ROM from the NEA.

Acknowledgements

Many thanks to Nicola D'Amico (ITER Consult, Rome) for his review of the report, and to Amanda Costa for contributing to the final editing.

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.

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EXECUTIVE SUMMARY

The CRISSUE-S project began on 1 January 2002, with the aim of re-evaluating fundamental issues related to the safety technology of light water reactors (LWR) involving interactions between neutron kinetics (NK) and thermal-hydraulics (T-H).

Along with the emphasis on NK/T-H analysis, the 12 CRISSUE-S partners organised their work into three work packages and in co-operation with the VALCO project. The objectives of the project where established as follows:

- To prepare a state-of-the-art report.
- To provide results of BE analysis in existing reactors.
- To identify areas of the NPP design where design/safety requirements can be relaxed.
- To provide recommendations to target institutions.

Work Package 1 is devoted to the assembly and structure of the existing database related to the subject at hand. The report produced constitutes an important starting point, as it summarises the resources presently available to researchers, utilities and regulatory bodies.

Work Package 2 provides the state-of-the-art report. The complete fulfilment and the detailed explanation of the central goals of the project can be found in the WP2 report.

The aim of Work Package 3 is to summarise results, to select the most important findings and to indicate the industry position on the related subjects.

In order to fulfil the above mentioned aim, this report emphasises achievements. It highlights the most important conclusions reached in the WP2 report, and only briefly refers to WP1. It is structured as follows.

The list of transients recommended to be studied for coupled analysis as well as the thresholds of acceptability of coupled results have a significant importance in the present report.

As the role of fuel is one of the topics addressed by the project, a summary is included dealing with fuel degradation during the transient and other elements such as change of geometry, physical properties, cross-sections, etc.. The summary is intended to demonstrate the connection between fuel engineering and neutron kinetics calculations.

The best-estimate approach (along with uncertainty analysis) is treated not only as a methodology, but with some emphasis on uncertainty sources (sub-cooled heat transfer coefficient, pressure wave propagation, radiolysis, fuel safety margins, etc.).

The specifications of the databases needed and created are relevant results of the project. They include not only the database in a general sense (code, NPP input, general report, initial conditions) but also that of transients suitable for validation and the results of its available analysis. This report focuses on suitable codes, as they become the guidelines of the remaining items from the standpoint of the utility decision makers.

The requirements for the qualification of analysis tools include the T-H field (nodalisations and codes), the NK field (codes and cross-sections) and those related to coupling both kinds of codes.

The licensing status and the use of findings by industry are addressed in the recommendations and final conclusions.

Chapter 1

TRANSIENTS TO BE STUDIED FOR COUPLED ANALYSIS

As a result of the CRISSUE-S project a list of transients is recommended to be studied using a coupled analysis. Due to existing plant configurations, transient analysis becomes plant specific.

1.1 PWR transients

- *MSLB*. Accident originated by the double ended guillotine break of one SL. Positive reactivity is caused by the cooling the primary water following depressurisation of the SG. The plug of cold water typically reaches the core a few seconds after the break occurrence in the steam line. A regional core power increase may occur due to partial mixing in the RPV downcomer of cold water from the affected SG with hot water from the intact SG. The potential regional nature of the transient and the amount of positive reactivity introduced justify the coupled analysis. HZP and FP initial conditions can be investigated at BOC and EOC conditions.

- *LOFW-ATWS*. The blockage of FW pumps originating from partial station blackout cause LOFW. Temperature increase of the primary loop and moderator and Doppler neutron feedbacks contribute to a power decrease. ATWS analysis is deemed necessary considering the (relatively high) accident frequency. The accident scenario originated by MSIV closure, again with the ATWS condition (*i.e.* MSIV closure-ATWS) could be studied with similar assumptions (apart from the initiating event).

- *CR ejection*. A regional reactivity increase is expected following CR ejection. This justifies the application of the 3-D coupled techniques. The highest worth CR should be considered. HZP and FP initial conditions can be investigated at BOC and EOC conditions. Ejection of CR banks or of group of CR can be of interest. The CR ejection is connected with an SBLOCA due to the damage of the holding mechanism of the ejected rod (though this is mostly not considered in the current analyses).

- *LBLOCA-DBA*. The accident is originated by the double-ended guillotine break of one CL located between the RPV and the MCP. 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 and, for this reason, is part of the official NPP FSAR. The proposal for 3-D neutron kinetics/thermal-hydraulic analysis is mostly connected with the need to quantify the conservatism introduced by the highly conservative peak factors (PF) for linear power that cause high values for PCT. The use of advanced coupled neutron kinetics/thermal-hydraulic techniques leads to the removal of that conservatism and emphasises the industrial relevance of the same techniques. A specific LBLOCA-RIA occurs in the case of positive moderator temperature coefficient, which in principle is possible at BOC with a high boron concentration.

- *Incorrect connection (start-up) of an inactive (idle) loop*. The inactive loop is assumed to have de-borated water in the loop seal having a volume consistent with the system geometry. This transient is representative of transients originated by the presence of boron in the primary loop (see also the SBLOCA-ATWS, below).

- *MSLB-ATWS*. The MSLB constitutes a DBA transient (see description above). The ATWS feature is not justified by any PSA study. Rather, the recommendation for the 3-D coupled analysis derives from the bounding nature that this transient might have (*i.e.* in terms of input reactivity for the core) and from the consideration that core integrity can be predicted for such an extreme situation.

- *SBLOCA-ATWS*. This is a TMI-type accident originating from a small loss of integrity of the primary loop. The relatively high frequency of occurrence justifies the ATWS condition. Positive reactivity insertion can be assumed to come from de-borated water emerging from any ECCS, *i.e.* HPIS and ACC (or SIT) and LPIS. The amount of fresh (un-borated) water injection should be in accordance with individual plant features and maintenance programmes.

1.2 BWR transients

- *TT without condenser bypass available*. TT constitutes a (relatively) frequent event for BWR operation. A positive pressure wave propagates from the turbine isolation valve to the RPV and reaches the core from the top (*e.g.* across the steam separator and dryer) and from the bottom (*e.g.* across the downcomer and the lower plenum). Void collapse causes positive reactivity insertion and power excursion typically stopped by scram occurrence. Opening the condenser bypass valves renders the effect of the pressure wave propagation milder, though this is neglected in the proposed scenario.

- *LBLOCA-DBA*. The rupture of one recirculation line (if present) originates the accident. The regulatory framework and the reasons for the analysis are the same as discussed in relation to PWR (see above).

- *CR ejection*. The same considerations made for PWR apply here.

- *FW temperature decrease-ATWS*. Malfunction of FW pre-heaters (*e.g.* sudden depressurisation in one pre-heater on the heating side) and of FW pumps (*e.g.* DC level control system) may cause FW temperature decrease that results in colder water at the core inlet. This creates the potential for reduction of the steam occupied volume into the core and a consequent increase in moderation and in fission power. This partly covers the event “FW flow rate increase” (see below).

- *MCP flow rate increase*. The flow rate increase may be caused by the malfunction of valves installed in the MCP lines or by a spurious signal controlling the pump speed (whatever is realistic for the concerned NPP). This partly covers the event “FW flow rate increase”.

- *MSIV closure-ATWS*. The closure of the main isolation valves creates a pressure pulse (same situation as TT) that causes void collapse into the core and reactivity excursion. The ATWS condition must be considered as a bounding condition in safety analysis. The application of the 3-D neutron kinetics/thermal-hydraulic technique has the potential to show that, even in such extreme and unrealistic (low probability) conditions, the safety margins embedded into the system design allow recovery of the core.

- *Stability analysis*. The stability of BWR performance has been thoroughly investigated during the entire nuclear era. Synthesis reports have been issued, and recently published analyses show the applicability of the 3-D coupled techniques. The use of different types of fuel in the same core renders the proposed investigation potentially even more valuable and necessary. The recommended transient can originate at nominal power and include MCP trip, bringing the plant into the exclusion region of the BWR flow map. Transient analysis should continue until NC steady state is established, at approximately 50% nominal core power and 30% nominal core flow. The FW and the SL flow rates should be varied according to the core power and the RPV pressure should be kept (nearly) constant during the simulation. Control systems for RPV pressure and DC level (and various BOP components) can be simulated or their role in determining the stability performance of the system should be evaluated.

- *Stability analysis-ATWS.* The same analysis as before shall be performed assuming failure of the scram system. Although this condition is beyond the DBA boundaries due to the low probability of occurrence, results can be useful to designers and to safety analysts in terms of optimising EOP and better understanding the safety margins of the NPP.

A similar list has been established for VVER plants; it can be found in Chapter 1, Section 1.4 of the WP2 report.

Notwithstanding the above discussion, the licensing process of a specific plant has particular transients that become crucial depending on the reactor history and cycle design. Knowing what they are for each individual case is essential to help focus on transients that are truly suitable for analysis.

More information on transients to be studied for coupled analysis can be found in Chapter 1, Sections 1.2, 1.3 and 1.5 of the WP2 report and in Chapter 2, Sections 2.2, 2.3 and 2.4 of the WP1 report.

Chapter 2

THRESHOLDS OF ACCEPTABILITY OF COUPLED RESULTS

The precision needed for any code calculation must derive from its specific application. However, the capabilities of computational tools – including the computer power – may have a strong role in assigning a target precision. Due to this reason and to the continuous (over the last 30-50 years) growth of both computer power and of capabilities of numerical techniques, target precision is seldom requested in complex code applications for nuclear technology.

Notwithstanding the above an attempt is made hereafter to formulate criteria and to define thresholds of acceptability for the results of coupled 3-D neutron kinetics thermal-hydraulics calculation. A straightforward process is pursued: a minimum reasonable number of "quantities of interest" is identified at first, requested error is assigned based on engineering judgement that takes into account of the current state of the art. A short description of the suitable level of detail requested for coupled calculation is given before the selection of quantities of interest and the related acceptability thresholds.

2.1 Quantities of interest

A variety of quantities of interest for the design and safety evaluation of LWR are the output of coupled 3-D neutron kinetics/thermal-hydraulics calculation. These range from the k_{eff}, to the boron concentration to reach criticality, to the worth of CR, to the transient pressure, the fuel temperature, the flow rate, the level (*e.g.* in the PRZ) and the amount of radioactive liquid discharged from a relief valve (SRV or PORV). The identification of these quantities is linked to the type of transient (*e.g.* ATWS or LBLOCA) and the application type (notably licensing or design analysis), and is also related to the duration of the transient. Definitely, each application is associated with a set of "quantities of interest". A minimum reasonable set of quantities of interest is defined below making reference to a transient duration of the order of 100 s. The following quantities are selected and related point values are considered:

- **[Quantity 1 and 2]**. Peak pressure in RPV (UP location) and in PRZ (if applicable), related FWHM (if applicable) and time of occurrence.
- **[Quantity 3]**. Peak total core power, related FWHM (if applicable) and time of occurrence.
- **[Quantity 4]**. CHF (or DNB) occurrence time.
- **[Quantity 5]**. PCT and time of occurrence.
- **[Quantity 6]**. Maximum fuel temperature (MFT) and time of occurrence.
- **[Quantity 7]**. Total thermal energy released to the fluid during the transient (TTEF – total thermal energy released to the fluid).

- **[Quantity 8].** Maximum % of core, in terms of heat transfer area (HTA) of the active region where, at any time, rod surface temperature > 1 000 K occurs (CRHST – core region at high surface temperature).

- **[Quantity 9]**. Maximum % of core in terms of volume occupied by fuel pins in the active region where, at any time, fuel temperature > 3 000 K occurs (CRHFT – core region at high fuel temperature).

2.1.1 Acceptability thresholds for the quantities of interest

The definition of acceptability thresholds encounters the same difficulties as those mentioned in above for the quantities of interest. The dimension of the core, including the nominal power, the power generated per unit volume and various other nominal operating conditions (*e.g.* nominal pressure, set-point for valves opening) may also affect the identification of thresholds of acceptance.

However, the errors that are acceptable (acceptability thresholds) for the identified quantities are defined hereafter. In the present approach it is assumed that the overall error for a point value quantity is the combination of two independent contributions: the quantity errors and the time error. Therefore, the predicted point value quantity stays in a rectangle whose geometric centre is the BE prediction and whose edges are given by the quantity and time error values.

- **[Quantity error, situation 'a']**. For pressure pulses characterised by FWHM < 0.1 s in reference BE evaluation, the acceptable threshold error is 10% nominal pressure of the considered system.

- **[Quantity error, situation 'b']**. For pressure pulses characterised by FWHM ≥ 0.1 s in reference BE evaluation, the acceptable threshold error is 2% nominal pressure of the considered system.

- **[Quantity error, situation 'a']**. For core power pulses characterised by FWHM < 0.1 s in reference BE evaluation, the acceptable threshold error is 100% nominal power or 300% initial power of the considered system, which ever is smaller.

- **[Quantity error, situation 'b']**. For core power pulses characterised by FWHM ≥ 0.1 s in reference BE evaluation, acceptable threshold error is 20% nominal power or 100% initial power of the considered system, which ever is smaller.

- **[Quantity error, situation 'b']**. For PCT, the acceptable threshold error is 150 K (larger errors can be tolerated for BE prediction of PCT below 1 000 K).

- **[Quantity error, situation 'b']**. For MFT, the acceptable threshold error is 200 K.

- **[Quantity error]**. For TTEF the acceptable threshold error is 10% of the energy released to the fluid in nominal conditions or the 100% of energy released to the fluid at the actual initial power, during the considered transient duration, which ever is smaller.

- **[Quantity error]**. For CRHST the acceptable threshold error is 10% of core HTA.

- **[Quantity error]**. For CRHFT the acceptable threshold error is 10% of core pin volume.

- **[Time error, situation ‘a’]**. The acceptable time error that can be associated with the prediction of time of occurrence of Quantities 1 to 3 is 100% of the BE value.

- **[Time error, situation ‘b’]**. The acceptable time error that can be associated with the prediction of time of occurrence of Quantities 1 to 6 is 20% of the BE value.

- **[Time error, situations ‘a’ and ‘b’]**. The acceptable time error that can be associated with the prediction of FWHM for Quantities 1 to 3 is 20% of the BE value.

The proposed acceptable threshold errors are consistent with the expected capabilities of current computational tools assuming the best practice of their application.

Further information on transients to be studied for coupled analysis can be found in Chapter 2, Section 2.5 of the WP2 report.

Chapter 3

THE ROLE OF FUEL

A significant part of the CRISSUE-S project findings are related to the role of fuel in the context of safety critical issues. For this reason a chapter in both the WP2 and WP3 reports is devoted to explaining the relevance of fuel properties and fuel failure in coupled analysis.

Reactor engineers within the organisations responsible for nuclear power plants are devoted to the task of determining – and ensuring – correct fuel behaviour under operation conditions, which could become more severe than those currently employed.

Various issues are considered, among them: increasing discharge burn-up of used fuel elements, high power peaks, high evaporation rates, high crud deposition products or chemistry programmes with high pH and lithium. All these issues are safety related and could become critical. That is why safety research must focus on avoiding clad corrosion in order to maintain the integrity of fuel rods and ensuring dimensional stability and absence of failures.

The use of new clad materials, such as Zirlo®, the implementation of intermediate flow-mixing grids (IFM) and other factors have allowed a relevant discharge burn-up increase.

It seems clear that, in the near future, fuel will work under more severe operating conditions than today. Because of this, it will also be necessary to verify the plant behaviour during safety scenarios that take today's trends into account.

Property changes and the fuel degradation, occurring mainly over the course of the fuel lifetime but also during the transient itself, produce variations of fuel geometry as well as the corresponding cross-sections. These changes obviously have an impact on NK calculations and on the subsequent safety analysis.

3.1 Thermo-physical properties of materials for LWR

Recommended data for the thermo-physical properties of nuclear materials have been published in the MATPRO publications. The latest issue of MATPRO data, in the SCDAP/RELAPS/MOD2 code manual from 1990, contains descriptions of material properties subroutines available in that database. The adequacy of the available data as concerns the appropriate physical and chemical properties of some nuclear materials has been reviewed in the report EUR-12404 (1989) published by the Commission of European Communities. A variety of other recommendations have also been made for the thermal conductivity and specific heat capacity of fuels.

3.2 Fuel failure mechanisms

Water reactor fuel continues to perform reliably in power reactors around the world. However, fuel failures from knows mechanisms still occur, occasionally with a significant impact on plant operation, and the current average fuel rod failure rate of around 10^{-5} are still high compared to the "zero failure" goal, which in today's understanding should correspond to a failure rate near 10^{-6}. Improving fuel reliability beyond current levels will be even more challenging for countries involved with programmes to further increase fuel discharge burn-up. A good knowledge of fuel performance is essential to identify the root causes of failures, and to define remedies at the design, manufacturing and operational levels.

The IAEA review has two major aims. The first addresses the above technical background with the aim of updating and supplementing the earlier IAEA publication of 1979. The second aim for this study is to provide a comprehensive presentation of the world-wide experience in this field in order to promote an exchange of information and future co-operation. The reactors covered by this study are "Western"-type boiling and pressurised light water reactors (BWR and PWR), and Soviet-type pressurised light water reactors (VVER). The acronym LWR will refer hereafter to the three types, BWR, PWR and VVER.

Recent surveys on fuel failures in general or those related to regional experience have been provided in several papers. In a comparison of the situations in 1979 and today, two factors are noteworthy. First, the fuel rod failure rates in LWR have been significantly reduced, on average by approximately one order of magnitude from the range above 10^{-4} to levels near 10^{-5}, *i.e.* a few rods per 100 000 rods in operation. Second, it is interesting to note that a majority of the mechanisms of earlier fuel failures are still active, mostly in combination with new root causes and contributing factors.

Chapter 3 (Sections 3.3 and 3.4) of the WP2 report provides an explanation on the mechanisms and root causes of fuel failure along with various details of its relation with coupled analysis. A list of the identified mechanisms follows:

- Manufacturing defects.
- Primary hydriding.
- Pellet-clad interaction.
- Corrosion.
- Dry-out.
- Cladding collapse.
- Grid-rod fretting.
- Debris fretting.
- Baffle jetting.
- Assembly damage.

3.3 Final remarks on the role of fuel

Important operating initiatives are taken to prevent fuel failure during plant operation. Some of them are related to operating procedures such as the surveillance of coolant activity, the guidelines for power changes and load following operation, the strategies for prevention of severe degradation or water chemistry controls. Others belong to the field of fuel handling, *i.e.* fuel loading/unloading practices and storage/disposal of failed fuel.

All these initiatives are having positive results and leading toward an improved fuel performance. Besides this main outcome, the improved knowledge of fuel behaviour must be utilised as a necessary feedback leading to a more accurate treatment of the key parameters of safety analysis. The CRISSUE-S project identifies the need to co-ordinate between the knowledge reached through operating practices concerning fuel and the future techniques of coupled NK/T-H safety analysis.

Chapter 4

THE BEST-ESTIMATE APPROACH

4.1 Methodology: BE versus conservative approach and the need for uncertainty evaluation

The word "uncertainty" and the need for uncertainty evaluation are connected with the use of best-estimate (BE) codes as opposed to conservative codes or assumptions in the code application. The application of coupled 3-D neutron kinetics/thermal-hydraulic codes implies the choice of the BE approach. What follows, then, is a short discussion in relation to the need for BE, the difference between BE and conservative, the origin of uncertainties and the current status of uncertainty evaluation. Additional details can be found in Chapter 2 (Section 2.6) of the WP2 report and the corresponding references.

The selection of a best-estimate analysis in place of a conservative one depends upon a number of conditions that are distinct from the analysis itself. These include the available computational tools, the expertise inside the organisation, the availability of suitable NPP data (*e.g.* the amount of data and the related details can be much different in the cases of best-estimate or conservative analyses), or the requests from the national regulatory body. In addition, conservative analyses are still widely used to avoid the need to develop realistic models based on experimental data, a task that may be revealed unrealistic in the case of BDBA, or simply to avoid the burden of changing an approved code and/or the approaches or procedures needed to obtain the licensing.

The conservative approach does not provide any indication of the actual margins between the actual plant response and the conservatively estimated response. In contrast, the uncertainty estimate provided in the best-estimate approach is a direct measure of such margins. As a result the best-estimate approach may allow for the elimination of unnecessary conservatism in the analysis and may permit the regulatory body and plant operating organisation to establish a more consistent balance for a wide range of acceptance criteria. A conservative approach does not give any indication about actual plant behaviour, including the time-scale for preparation of emergency operating procedures, or for use in accident management and preparation of operation manuals for abnormal operating conditions.

Although the acceptability of the approach to be used for an accident analysis needs to be defined by the regulatory body, the use of totally conservative approaches (conservative models, input data and plant conditions) is unwarranted nowadays, given the broad acceptance of best-estimate methods (*e.g.* mature best-estimate codes are widely available around the world, an extensive database exists for nearly all power reactor designs and best-estimate plant calculations are well documented). The use of totally best-estimate approach implies "the difficulty" of quantifying code uncertainties for every phenomenon and for every accident sequence.

4.2 Uncertainties

Uncertainty analyses include the estimation of uncertainties in individual modelling or of the overall code, uncertainties in representation and uncertainties in plant data for the analysis of an individual event. Scaling studies to quantify the influence of scaling variations between experiments and the actual plant environment are included in this definition. In some references, code scaling and uncertainty analysis are identified separately. These concepts are applicable to any of the CSC, NKC and THSC.

A number of uncertainty evaluation methods proposed by different institutions are mentioned in the WP2 report.

4.2.1 CIAU method and its extension

All of the uncertainty methodologies suffer from two main limitations:

- The resources needed for their application may prove to be prohibitive, ranging up to several man-years.
- The achieved results may be strongly methodology user dependent.

The basis of the CIAU can be summarised by the following three items:

- *Build-up of NPP status*. Each status is characterised by the value of six relevant quantities (or phases) and by the value of the time since the start of transient. Each of the relevant quantities is subdivided into a suitable number of intervals that may be seen as the edges of hypercubes in the phase-space. The transient time or duration of the transient scenario is also subdivided into intervals.
- As*sociation of uncertainty with NPP status*. Accuracy values derived from the analysis of experimental data are associated with each NPP status.
- *Use of the method*. At each time, the CIAU code calculation result is associated to a time interval and to a hypercube, *i.e.* a NPP status, from which the uncertainty values are taken and associated with the current value of the prediction.

The CIAU concept can be applied with any uncertainty method. In other words, any existing uncertainty method (*e.g.* CSAU, GRS method, etc.) can be used to generate uncertainty values, thus filling the hypercubes and the time intervals. The root idea of the current CIAU is connected with the status approach.

First, quantities are selected to characterise in a multi-dimensional space the thermal-hydraulic status of a LWR during any transient. Then, accuracy is calculated from the analysis of experimental data. The combination of accuracy values and time intervals allows the derivation of continuous uncertainty or error bands enveloping any time-dependent variables that are the output of a system code calculation.

The RELAP5/MOD3.2 system code and UMAE uncertainty methodology have been coupled to constitute the CIAU. Therefore, the uncertainty has been obtained from the extrapolation of the accuracy resulting from the comparison between code results and relevant experimental data; these may be obtained from integral test facilities as well as from separate effects test facilities.

Within the CRISSUE-S framework, the methodology has been extended to 3-D neutron kinetics/ thermal-hydraulics calculations.

A number of input parameters connected with neutron kinetics calculations are uncertain:

- Rod worth ±10% or ±15% (depending on reference).
- Fraction of delayed neutrons (β) ±5%.
- Doppler coefficient ±20%.
- Moderator coefficient ±30%.
- Fuel heat capacity ±10% (this is also a relevant thermal parameter).
- Boron reactivity coefficient ±15%.
- Change in the reactivity per unit change in the fuel and moderator temperature when fuel and moderator are at the same temperature $\pm 0.36 \times 10^{-4}$ $\Delta\rho$/°C.
- Critical boron concentration at 100% core power ±50 ppm.
- Power distribution (at intermediate level and at 100% power) ±0.1 * relative power density for each measured assembly power.

In a static calculation, the main output parameter is constituted by the k_∞. More detail on the subject can be found in Chapter 2 (Section 2.12) of WP2 report and the related references. The reported values can be interpreted as minimum acceptable values for the error in calculating k_∞.

Basically, the demonstration of the feasibility of the approach has been achieved, and sample results related to the BWRTT transient in the Peach Bottom NPP are shown in Annex 1 of WP2 and related references. More information is available in Chapter 2 (Sections 2.6 and 2.12) of the same report.

4.3 Sources of uncertainty

4.3.1 Fuel-related

As was said in the introduction, fuel plays a very relevant role in the subject of critical safety issues related to the analysis of nuclear power plant behaviour using coupled NK/T-H codes. That is why a specific chapter (Chapter 3) has been devoted to this particular topic.

Although the main subject of Chapter 3 is clearly related to fuel management, an implicit message can be derived: fuel is probably the main source of uncertainty in the context of coupled predictions.

Traditionally, in conservative calculations, assumptions have been made to take into account the uncertainties related to fuel properties, and the final analysis becomes penalised and allows less flexibility for operation.

Each and every subject listed or summarised in Chapter 3 results in an uncertainty that is added to the key parameter involved. Among fuel uncertain parameters, some are clearly NK related and others are T-H related. The variation of the most of them has an impact on both NK and T-H analysis due to feedback.

Some NK/T-H parameters and phenomena characterised as sources of uncertainty and not addressed in Chapter 3 are listed below:

- Radiolysis in fast reactivity transients.
- Dynamic subcooled boiling.
- Dynamic CHF.
- Volume void weighting on HT surface for two-fluid models.
- Spacers with mixing vanes.

4.3.2 Related to other phenomena or components

Valve characteristics

Valve characteristics, *i.e.* valve area changes such as 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 establish and thus the effects are often evaluated from sensitivity analyses. In relation to reactivity changes important influences from the valve operations can include BWR core pressure variations at closure of the turbine stop valve and (even more so) the associated pressure wave propagation through the steam line and into the RPV. The pressure wave will become more pronounced if the valve area change incorporates 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.

Frictional and discrete pressure losses

Accurate pressure loss distribution in the T/H system is an essential prerequisite for the calculation of 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. It is envisioned when it comes to the core that the roughness is influenced by the time period during which the fuel assembly has been loaded into the core, *i.e.* by the burn-up, with an increased roughness with time due to the growth of the oxide layer. It is also realised that the detachment of a possible rod surface vapour layer, 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.

Phase separation at tees

The phase separation phenomena at tee components can have a dominant influence on the course of the system depressurisation rate. So will (for a BWR steam line break) a perfect separation result in maximum depressurisation while minimising the loss of inventory. Reaching low pressure while maintaining a high liquid inventory may be non-conservative (for instance by retaining the fuel temperature well below any critical levels). Activities are underway by USNRC to review existing models and to evaluate and possibly improve their capabilities under an extended range of conditions.

High transient thermal flux

As consequence of CR ejection, high local thermal flux occurs. Subcooled voids can be formed in a bulk of high-speed liquid. Void formation creates a volume increase in initially nearly incompressible fluid environments like the RPV of a PWR. Local thermal power exchange may reach values one order of magnitude larger than under nominal conditions and pressure may locally increase up to several MPa. DNB is expected. Validation of convection heat transfer correlations is questionable under these conditions.

Positive pressure pulse propagation

In a BWR core this causes void collapse and consequent positive reactivity excursion. Thus the condition for sudden re-evaporation of the condensed liquid is created. The range of validity of convection heat transfer correlations is again questionable under these conditions. Therefore a careful consideration of the Courant limit, in these cases based on acoustic wave propagation, and of the calculation node length has to be adopted. 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. The pressure wave propagation in the steam dome of the BWR through the dome outlet nozzle should be noted. 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. Also, the CR movement creates local fluid displacements that locally may be significant.

4.3.3 Related to models and codes

Heat transfer modelling inside pin and fuel modelling

The heat transfer phenomena inside the fuel pins comprise heat conduction in the fuel pellets, heat transfer across the gap between fuel pellets and the 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 will consequently 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, though this would impair the use of the moving mesh capability in cases where re-flood simulations are needed.

The list of pin heat conduction and fuel-related phenomena or related computational aspects that may require further investigation to validate the results of coupled neutron kinetics/thermal-hydraulic calculations includes the following:

- Minimum number of concentric rings for the fuel rod that bring to the convergence of results for CSC, NKC and THSC (this number must be consistent in the three steps of the analysis). The suitable number of rings for the gap should be identified as well.

- Pellet gap performance during the transient with main reference to variations in physical properties and geometry. The evaluation of such aspects is not within the typical current NKC and THSC capabilities.

- Fuel deformation and change in physical properties during the transient (namely steep power rise). Coupling between THSC and fuel codes might be needed.

Neutron transport and diffusion

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 a 3-D neutron diffusion equation based on two neutron energy groups and with six groups of delayed neutron precursors. This has been found to be adequate for steady-state applications and for those transient applications where direct validation has been possible. The use of ADF has proven to be sufficient for several applications. It is also realised that the utilisation of MOX fuels, with its higher Pu content, requires additional neutron energy groups to be included for accurate simulations. The Pu isotopes have high absorption and fission resonance at around 1 eV (^{240}Pu absorption) and from 100 eV up to keV (^{239}Pu, ^{241}Pu absorption and fission).

The major calculation features of the neutron diffusion models include the ability to perform eigenvalue calculations (k_{eff}), transient flux calculations, xenon transient calculations, decay heat calculations, depleting (burn-up) calculations and adjoint calculations. Further, pin power reconstruction capabilities are available to obtain pin power and associated intranodal neutron flux distributions from the calculated nodal fluxes. The 3-D capability provides the basis for realistic representation of the complete reactor core, although provisions are included to use appropriate symmetry sections, such as half and quarter core parts for computational efficiency. Carefully selected boundary conditions for the symmetry planes are paramount for adequate calculation results in those cases. One-dimensional (1-D) capabilities are also usually available for simulation of transients with predominant axial neutron flux variations.

Important data for the neutron diffusion calculations, apart from pure geometrical descriptions of various core parts, include tabulated macroscopic cross-sections as function of T/H and fuel parameters, boron concentration, control rod positions (worth, used banks, etc) and microscopic cross-sections for Xe and Sm including corresponding dependencies. Other data are decay constants and the ADF for the fuel assemblies' four sides.

The list of neutron kinetics phenomena or related computational aspects that may require further investigation to validate the results of coupled neutron kinetics/thermal-hydraulic calculations includes:

- Identification of a suitable number of neutron energy groups.

- Influence of resonance absorption cross-section in individual layers of pellets (partly linked to the previous item).

- Systematic identification of the influence of material discontinuities (*e.g.* due to the presence of fuel box, CR, at the border between reflector and core active region, etc.).

Chapter 5

DATABASE NEEDED: SUITABLE CODES

Although the database needed for NK/T-H calculations should include items other than the code itself, the focus of this summary is indeed the code. Details on NPP input, initial conditions or general plant description can be found in WP1. The interest of directed toward the codes has to do with the utilities' decision makers. Once the decision to use a particular code has been made, the NPP input is created with the chosen code in mind, and plant details and initial conditions are explicitly recorded following the interaction with the code user.

Although the person responsible for plant operation wants to have results from more than one (normally two) calculation option, this chapter is not the suitable forum to discuss the issue. It is very important to have a common plant description as a base to build input decks for different codes, but the point can be treated in another context. Chapter 9 of this report (*Use by industry*) elaborates upon the relevance of the issue.

Complex codes are required for three steps in the application of the 3-D coupled techniques:

- Codes for deriving suitable neutron kinetics cross-sections (CSC = cross-section code).
- Thermal-hydraulic system codes (THSC).
- Neutron kinetics codes (NKC).

The codes described here are internationally known and suitably qualified. The list of codes is by no mean exhaustive, but serves as an indication of the types of codes available today within the area of coupled THSC/NKC. The features and capabilities of the codes are discussed in the following sections. Details can be found in the reported references

The consistent application of CSC, THSC and NKC is required to perform a full 3-D coupled neutron kinetics/thermal-hydraulic calculation. However, the CSC can be used "out of line" and THSC and NKC must be coupled and interact at each time step.

5.1 Cross-section codes

CASMO

The widely used CASMO cross-section parameterisation model attempts to model the cross-section cross-term dependence involving an approximate type of cross-section representation. 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.

5.2 Thermal-hydraulic system codes

ATHLET

The thermal-hydraulic computer code ATHLET (Analysis of Thermal-hydraulics of Leaks and Transients) is being developed by the Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) for the analysis of anticipated and abnormal plant transients, small and intermediate leaks, and large breaks in light water reactors.

It has a highly modular structure and allows an easy implementation of different physical models. ATHLET is composed of several basic modules for the calculation of the different phenomena involved in the operation of a light water reactor.

RELAP5 USNRC version

This is a thermal-hydraulic system code extensively used for the analysis of any kind of transient in LWRs. The code is based upon the solution of six partial differential equations that are coupled with (or that needed for their solution):

- A wide range "heat transfer surface" for the determination of the HTC.
- The equations for the conduction heat transfer into the solid.
- A variety of constitutive equations (*e.g.* interfacial drag).
- A variety of external models (*e.g.* two-phase critical flow, pump, separator).
- The equations for tracking various non-condensable gases.
- The equation for tracking boron into the system.
- Zero-dimensional (0-D) neutron kinetics equations.

Material properties are embedded into the code, but can also be supplied by the user. The numerical solution method has been specifically adapted for the code and is based upon the use of a semi-implicit finite-difference technique. The code is capable of modelling the primary and secondary circuits (where applicable) of NPPs as well as all the components belonging to the BOP, including the related actuation logic.

The code is classified as a 1-D code, though fictitious 3-D nodalisations (input decks) can be configured to simulate three-dimensional flow configurations in open zones.

RELAP5-3D © DOE version

All attributes discussed in the previous section related to RELAP5 NRC apply here as well. It can be added that RELAP5-3D © has the capability to apply the 3-D thermal-hydraulic general model in selected zones of the NPP. The NESTLE 3-D neutron kinetics is embedded in this RELAP computer code version.

CATHARE-2 CEA code

All attributes discussed in the previous section related to RELAP5 NRC apply here as well. Further, CATHARE was directly developed for PWR plants, though it has been recently applied to the analysis of transients in VVER and BWR NPP. Related to RELAP5, a different numerical solution method is adopted and a different structure of the input deck must be put into place by the user. The code has the design capability to simulate the containment.

CATHARE is modular; several basic modules may be assembled to model the primary and secondary circuits of any pressurised water reactor or of any analytical test rig (separated effects) or integral test rig (system effects). Zero-dimensional (0-D), 1-D and 3-D modules are available. All of the modules can be connected to walls, exchangers and fuel rods (radial conduction, thermo-mechanics). A 2-D conduction calculation is generally made to cope with the quenching of a hot reactor core during a re-flooding process. Other submodules are used to compute point kinetic neutronics, pump speeds, accumulators, sources and sinks. All the modules above use the two-fluid model to transport steam-water flows mixed to four non-condensable gases. Thermal and mechanical non-equilibrium are described through mass, energy and momentum balance equations written for each phase.

TRAC-PF1

Los Alamos National Laboratory (LANL) developed the original TRAC thermal-hydraulic code in the mid 70s for PWR analysis. The TRAC-PF1 code is considered to be one of the most advanced thermal-hydraulic codes because of its full six-equation, two-fluid model of the vessel component, such that accurate modelling of the transient can be achieved. The code can treat a non-condensable gas in a vapour field as well as dissolved solute in the liquid field. TRAC-PF1 is a best-estimate system transient analysis code, which has a 3-D thermal-hydraulic analysis capability. A modified version of TRAC-PF1/MOD2 version 5.4 is currently being used at the Pennsylvania State University (PSU). Version 5.4 incorporates a one-dimensional decay heat model that dynamically computes the decay heat axial shape during the transient. The code solves the general transient two-phase coolant conditions in one, two or three dimensions using a realistic six-equation, two-fluid, finite-difference model combining 3-D vessel hydrodynamics and 1-D balance-of-plant modelling. Two-dimensional treatment of fluid-wall heat transfer is incorporated through the ROD and SLAB components. The ROD component provides the treatment of power generation and its transfer to the coolant. The six-equation representation of the two-phase flow conditions, in conjunction with specialised empirical modes for a variety of PWR primary- and secondary-loop components and control systems, allows TRAC-PF1/MOD2 version 5.4 to accurately model both mild and severe T-H transients.

TRAC-M

The United States Nuclear Regulatory Commission (USNRC) is currently in the process of consolidating its four T-H codes (TRAC-P, TRAC-B, RELAP and RAMONA) into a single code. The goal of this effort is to combine the capabilities of the current suite of codes while reducing the maintenance and development burden. The user community will then be able to focus on one code instead of four, thereby enhancing the knowledge base. A modernised and modularised TRAC-P code, now called TRAC-M, serves as a basis for the consolidation. The architecture has been revamped and the language migrated to FORTRAN90 to produce a more modular, readable, extendable and developer-friendly code. A neutronics package has been coupled with TRAC-M using parallel virtual machine (PVM) to provide 1-D and 3-D kinetics models without having to add this functionality to the TRAC-M code itself. This allows the ability to independently improve the neutronics or T-H models in TRAC-M.

TRAC-BF1

TRAC-BF1 is a best-estimate, full-system, thermal-hydraulics code for boiling water reactors. The two-fluid flow models are solved in one and three dimensions. The code is modular in nature and is based on component models.

POLCA-T

POLCA-T is a coupled 3-D core neutron kinetics and system thermal-hydraulics computer code. The code is able to perform steady-state and transient analysis of BWRs. The code, to varying extents, is based on models and tools for BWR and PWR analysis used in the POLCA7, BISON and RIGEL codes. The code utilises new advanced methods and models in neutron kinetics, thermal-hydraulics and numerics.

5.3 Neutron kinetics codes

DYN3D

The code DYN3D was developed at FZ Rossendorf and is used for investigations of steady-state and reactivity transients in cores of thermal power reactors with hexagonal or quadratic fuel assemblies. The three-dimensional neutron kinetics model HEXDYN3D of the code is based on the nodal expansion method for solving the two-group neutron diffusion equation. The thermo-hydraulic part FLOCAL consists of a fuel rod model and a two-phase flow model describing coolant behaviour and based on four differential balance equations for mass, energy and momentum of the mixture and mass balance of the vapour phase. Several safety parameters such as temperatures, DNBR and fuel enthalpy are evaluated. The input data of the code includes macroscopic cross-sections (depending on the thermo-hydraulic parameters) and boron concentration.

NEM

NEM is a 3-D multi-group nodal code developed and used at Pennsylvania State University (PSU) for modelling both steady-state and transient core conditions. The code has options for modelling 3-D Cartesian, cylindrical and hexagonal geometry. This code is based on the nodal expansion method for solving the nodal equations in three dimensions. It utilises a transverse integration procedure and is based on the partial current formulation of the nodal balance equations. The leakage term in the one-dimensional transverse integrated equations is approximated using a standard parabolic expansion using the transverse leakages in three neighbour nodes. The nodal coupling relationships are expressed in partial current formulation and the time dependence of the neutron flux is approximated by a first order, fully explicit, finite difference scheme. This method has been shown to very efficient although it lacked the precision of the advanced nodal codes. Recently an upgrade of the method has been completed, replacing the fourth-order polynomial expansion with a semi-analytical expression utilising a more accurate approximation of the transverse leakage.

This code has been benchmarked for Cartesian, cylindrical and hexagonal geometry. NEM is coupled with the Penn State versions of TRAC-PF1 and TRAC-BF1.

NESTLE

NESTLE is a multi-dimensional neutron kinetics code developed at North Carolina State University which solves the two- or four-group neutron diffusion equations in either Cartesian or hexagonal geometry using the nodal expansion method and the non-linear iteration technique. Three, two or one-dimensional models may be used. Several different core symmetry options are available including quarter, half and full core options for Cartesian geometry and 1/6, 1/3 and full core options for hexagonal geometry. Zero flux, non-re-entrant current, reflective and cyclic boundary conditions are available. The steady-state eigenvalue and time-dependent neutron flux problems can be solved by the NESTLE code. The new Border Profiled Lower Upper (BPLU) matrix solver is used to efficiently solve sparse linear systems of the form AX = B. BPLU is designed to take advantage of pipelines, vector hardware and shared-memory parallel architecture to run fast. BPLU is most efficient for solving systems that correspond to networks, such as pipes, but is efficient for any system that it can permute into border-banded form. Speed-ups over the default solver are achieved in RELAP5-3D© running with BPLU on multi-dimensional problems, for which it was intended.

PARCS USNRC version

PARCS is a three-dimensional reactor core simulator developed at Purdue University which solves the steady-state and time-dependent neutron diffusion equation to predict the dynamic response of the reactor to reactivity perturbations such as control rod movements or changes in the temperature/fluid conditions in the reactor core. The code is applicable to both PWR and BWR cores loaded with either rectangular or hexagonal fuel assemblies.

The neutron diffusion equation is solved with two energy groups for the rectangular geometry option, whereas any number of energy groups can be used for the hexagonal geometry option. PARCS is coupled directly with the thermal-hydraulics systems codes TRAC-M and RELAP5, which provide the temperature and flow field information to PARCS during the transient.

QUABOX

QUABOX is a neutron kinetics code developed in the seventies at Gesellschaft für Anlagen und Reaktorsicherheit (GRS) in Germany for 3-D core neutron flux and power calculations in steady-state and transient conditions. It solves the two-group neutron energy diffusion equation by the method of local polynomial approximation of the neutron flux.

5.4 Example of coupled 3-D neutron kinetics and thermal-hydraulics codes

Among the three examples presented in Chapter 2 (Section 2.2.4) of the WP 2 report, the case of SIMTRAN is briefly elaborated on here.

SIMTRAN was developed as a single code merge, with data sharing through standard FORTRAN commons, of our 3-D neutronics nodal code SIMULA and the multi-channel (with cross-flows) thermal-hydraulics code COBRA-IIIC/MIT-2. Both codes solve the 3-D neutronics and T-H fields with maximum implicitness, using direct and iterative methods for the inversion of the linearised systems.

SIMULA solves the neutron diffusion equations in one or two groups, on 3-D coarse-mesh nodes (quarter or full fuel assemblies) using our linear-discontinuous, finite-difference scheme, where the interface net currents are given in terms of the actual node average and the corrected interface averaged fluxes, using synthetic interface flux discontinuity factors for each group and node interface. SIMULA uses these synthetic coarse-mesh discontinuity factors in the XY directions, pre-calculated by 2-D pin-by-pin two-group diffusion calculations of whole core planes. In the axial direction it performs embedded iterative 1-D fine-mesh two-group diffusion solutions for each node stack, with the radial leakage terms interpolated from the nodal two-group solution.

COBRA-III-C/MIT-2, is a public code for thermal-hydraulics calculations with implicit cross-flows and homogeneous two-phase fluids. It is used world-wide for DNBR analysis in LWR subchannels, as well as for 3-D whole PWR core simulation with one or more channels per fuel assembly.

More information on suitable codes and examples on coupling them can be found in Chapter 2, Sections 2.2, 2.3 and 2.4 of the WP2 report.

Chapter 6

DATABASE CREATED: TRANSIENT ANALYSIS RESULTS

6.1 Transients suitable for validation

The main objective of this chapter is that of Subtask 3 of WP1: to provide guidelines for validation of neutronic/thermal-hydraulic coupled models using suitable actual transients of the reference plant. The results of this subtask are explained in Chapter 6 of the WP1 report.

Although the transients and the data involved are highly dependent on the characteristics of the reference plant, these guidelines are meant to be general enough to determine if any transient of any plant could be used to validate the coupled codes.

The guidelines are given on the basis of specific cases that are used to exemplify the requirements that have to be met.

Obviously not all of the transients occurring in a plant are suitable to validate the coupled codes. Some requirements must be met, to ensure, first of all, the quality of the data. Other requirements allow determining if using a coupled code is appropriate. Among these, it is necessary that the transients involved be correctly identified, temporally and with a description of the state of the plant where they occurred at the moment they did so. Thus, the reactor in which they occurred, the date and the time when they occurred, the core burn-up, power, average temperature, boron concentration and control rod position (banks D&C) at the moment they occurred must all be clearly stated. Eventually, a manual action performed needs to be signalled. Moreover, the data need to be properly recorded, stating the magnitude measured and the units in which they measure it. This is valid both for data given in a time sequence or for “snapshots” of the state of the plant at a given point in time.

The transients suitable to validate a coupled code must show some evidence of feedback effects, caused by the occurrence of asymmetries or by any other reason.

Chapter 6 of the WP1 report specifies data format and provides information on four different actual transients which occurred in Ascó (PWR with UTSG), Oskarhamn 2 (BWR), Peach Bottom (BWR) and Three Mile Island 1 (PWR with OTSG).

The information provided for each case is divided into four categories: plant description, transient description and measured data (neutronic and thermal-hydraulic). Note that all these data are in general multi-dimensional and must be given in a readable way; the specific format depends on the type of data.

These four transients are a good example of what is, generally speaking, available for validating NPP models.

6.2 Results of available calculations

One of the activities of the CRISSUE-S project has been the preparation of a database of reference transient calculations. Existing results have been assembled and are explained in Annex 1 of WP2.

The main purpose of this effort is to constitute a database of results that can be used by analysts as well as by decision makers to obtain a quantitative idea about expected transient scenarios and, definitely, to support decisions about the need to perform coupled calculations related to a generic NPP.

Calculations have been performed utilising computational tools and input decks developed and already utilised for different purposes. Each calculation is documented by a one-page description and two pages of significant system-related time trends or 3-D snapshots of core-related quantities. The last section of Annex 1 of the WP2 report is devoted to the calculation of the enthalpy release to the fuel for some of the most significant calculations (*i.e.* calculations that produced the highest local power excursion) documented in previous sections of the same Annex.

The following main limitations apply for the performed calculations that are documented in Annex 1 of WP2:

- 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 has been made to achieve qualified results, a comprehensive verification of the various qualification requirements has not been undertaken, as already mentioned. Thus, the reported results should be considered qualified as far as possible. This is true for stand-alone RELAP5 or PARCS or NESTLE input decks as well as for the coupled input decks.

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

The aim of the summarised annex is to provide an overview of how the database performed for a typical NPP using coupled 3-D neutron kinetics/thermal-hydraulic calculations. The adopted format describing each calculation (including figures and tables) is as follows:

- *NPP type and simulated transient*. This section includes a short description and two tables, the titles of which are *Main boundary and initial conditions* (initial core power, core inlet flow rate and temperature, SG pressure, etc.) and *Imposed sequence of main events* (break occurrence or initial failure, actual scram signal, isolation of MSIV or of FW line, MCP trip, etc.).

- *Resulting sequence of main events and description of the transient scenario*. This section includes one table and "main parameters" figures:

  – The table reports the list of resulting events (*e.g.* time and value of peak power and FWHM, time of scram, end of the calculation, etc.).

  – The figures typically show total core power, reactivity, pressure and fuel temperatures.

- *Significant results*. The main results and findings from the analysis are summarised in a few statements.

Calculation results and related notes are added in relation to the energy released to the fuel during the most relevant among the considered transients.

Table 1. Overview of analyses performed

ID	Transient type	Reactor type – analyses ID and status		
		PWR case identity	BWR case identity	VVER case identity
I	MSLB	Case 1-PWR: TMI [X] Case 2-PWR: ASCO [X]	–	Case 1-VVER:VVER-1000 [X] Case 2-VVER: VVER-440 [X]
II	LOFW-ATWS	Case 3-PWR: TMI [X]	–	Case 3-VVER: VVER-1000 [X] Case 4-VVER: VVER-440 (*)
III	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 *
IV	LBLOCA-DBA	Case 5-PWR: TMI [X]	Case 9-BWR: PB2 [Y]	Case 7-VVER: VVER-1000 [Y] Case 8-VVER: VVER-440 *
V	Incorrect insertion of an inactive loop	Case 6-PWR: TMI [X]	–	Case 9-VVER: VVER-1000 [Y] Case 10-VVER: VVER-440 *
VI	MSLB-ATWS	Case 7-PWR: TMI [X]	–	Case 11-VVER: VVER-1000 [X] Case 12 -VVER: VVER-440 *
VII	SBLOCA-ATWS	Case 8-PWR: TMI [X]	–	Case 13-VVER: VVER-1000 SBLOCA [X] Case 14-VVER: VVER-440 *
VIII	Turbine trip	–	Case 1-BWR: PB2 [X]	–
IX	Turbine trip-ATWS	–	Case 2-BWR: PB2 [X]	–
X	FW temperature increase	–	Case 5-BWR: PB2 [X]	–
XI	MCP flow rate increase	–	Case 6-BWR: PB2 [X]	–
XII	MCP flow rate increase-ATWS	–	Case 7-BWR: PB2 [X]	–
XIII	BWR stability	–	Case 8-BWR: PB2 [X]	–
IVX	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.

Chapter 7

QUALIFICATION REQUIREMENTS

This chapter is strongly related to that of *Thresholds of acceptability* (Chapter 2). Computer codes require preparation of the mathematical model that can adequately simulate all or part of a nuclear power plant and/or the processes occurring therein. This mathematical model consists of the computer code itself, and the set of input data grouped in a file (or files) that substantially describe the plant or facility within the boundaries and assumptions of the code models. The preparation of such a model is not only the source of the largest number of errors, but also of uncertainties that affect the use of best-estimate codes. A full knowledge of the computer code models is not sufficient for the error-free preparation of the nodalisation. It is well beyond the purposes of this project to discuss in detail code assessment requirements and related present status. However, some guidelines of acceptability are presented.

Neutron kinetics and thermal-hydraulic nodalisation requirements are summarised in the following paragraphs, starting with the suitable level of detail of the input deck.

In a following subsection thermal-hydraulic nodalisation development is discussed along with the acceptability of nodalisation at both the steady-state level and the on-transient level.

The remaining subsections are less detailed due to the fact that the neutronic and coupling qualification process is not as advanced as the thermal-hydraulic one.

7.1 Suitable level of detail of input deck

Nodalisations that allow the evaluation of neutron kinetics cross-section, 3-D neutron kinetics and thermal-hydraulic nodalisations should be distinguished, as has already been mentioned. Various degrees of freedom are left to the user when setting up nodalisations for performing the simulation of a requested transient in an assigned system. The recommendations listed here consider the capability of current computers (year 2003) and aim at reducing the user effect (or the degrees of freedom):

- Nodalisations for CSC, THSC and NKC must be consistent; the same elementary cell must be considered in each application.

- Individual fuel elements (minimum level of detail) should be considered in each of the codes concerned. These typically differ amongst one other as concerns burn-up, initial enrichment, presence of burnable poison and of control rod, position in the core and (in some cases) for the number of rods.

- Lumping together similar fuel elements, typically in the THSC nodalisations, should be avoided as much as possible.

- Symmetry assumptions must be justified and applied in similar way for the various codes.

- Axial node subdivisions of the order of twenty (or greater) must be adopted, *i.e.* each fuel assembly must be divided into 20 or more parts, typically equal in terms of axial length.

- In THSC nodalisation, the level of detail of different parts of the RPV should be consistent with the level of detail of the core region.

- Energy ranges for neutrons must be the same for CSC and NKC nodalisations.

- Care must be taken in identifying the range of variation (validity) of the cross-section from the CSC calculation. It must be ensured that those limits are not exceeded in the coupled THSC-NKC calculation.

- The core bypass (typical THSC nomenclature) and the reflector (typical NKC nomenclature) nodalisation must be consistent over the various steps of the coupled calculation (*e.g.* volume of the core bypass, input parameter for THSC, must be consistent with the reflector thickness utilised in the NKC input deck).

7.2 Qualification of T-H nodalisations and codes

Thermal-hydraulic nodalisations (input deck) development

Four fundamental pre-conditions should be fulfilled for the correct application of a complex thermal-hydraulics system code to the prediction of transient scenarios expected in NPP:

- The code should be frozen.

- The code should be properly qualified through a wide-ranging (preferably international) assessment programme, as mentioned earlier.

- The developer of the nodalisation should be a qualified code user for the selected code.

- The nodalisation (of the plant) once developed should be properly qualified.

To a large extent, engineering judgement is normally used to develop an input deck. The importance of establishing a procedure for the setting-up and qualification of nodalisation is a consequence. Such a procedure can be split into the following steps:

1) Gathering of a verified set of NPP data.

2) Set-up of the plant nodalisation (input deck for nominal steady-state conditions).

3) Qualification of the nodalisation.

A nodalisation can be considered as qualified when:

1) It has a geometrical fidelity with the plant in question.

2) It is reproducing the nominal measured steady-state condition of that plant.

3) It shows a satisfactory behaviour under time-dependent conditions, in accordance with the time-dependent data of any test performed or of any actual transient in the nuclear power plant (if available).

Thus, qualification of the nodalisation has been divided into two separate processes: steady-state and on-transient.

Acceptability of the nodalisation at the steady-state level

In order to address this question, the University of Pisa proposed a number of acceptability conditions that are summarised in the WP2 report and the related references. These conditions have been adopted for over ten years at the time of issuing of the present report.

Two sets of conditions can be distinguished: the first (rows 1 to 11 in Table 2) deals with the values of the quantities that are part of the input deck, the second (rows 12 to 25) deals with the results predicted by the code at the end of the steady-state calculation.

Table 2. Acceptability criteria for thermal-hydraulic nodalisation qualification at the steady-state level

	Quantity	Acceptable error (°)
1	Primary circuit volume	1%
2	Secondary circuit volume	2%
3	Non-active structures heat transfer area (overall)	10%
4	Active structures heat transfer area (overall)	0.1%
5	Non-active structures heat transfer volume (overall)	14%
6	Active structures heat transfer volume (overall)	0.2%
7	Volume vs. height curve (*i.e.* "local" primary and secondary circuit volume)	10%
8	Component relative elevation	0.01 m
9	Axial and radial power distribution	1%
10	Flow area of components like valves, pumps and orifices	1%
11	Generic flow areas	10%
(*)		
12	Primary circuit power balance	2%
13	Secondary circuit power balance	2%
14	Absolute pressure (PRZ, SG, ACC)	0.1%
15	Fluid temperature	0.5% (**)
16	Rod surface temperature	10 K
17	Pump velocity	1%
18	Heat losses	10%
19	Local pressure drops	10% (^)
20	Mass inventory in primary circuit	2% (^^)
21	Mass inventory in secondary circuit	5% (^^)
22	Flow-rates (primary and secondary circuit)	2%
23	Bypass flow rates	10%
24	Pressuriser level (collapsed)	0.05 m
25	Secondary side or downcomer level	0.1 m

(°) The % error is defined as the ratio: |reference value – calculated value|/|reference value|. The dimensional error is the numerator in the above expression.

(*) With reference to each of the quantities below, following a 100 s transient steady-state calculation, the solution must be stable with an inherent drift < 1%/100 s.

(**) And consistent with power error.

(^) Of the difference between maximum and minimum pressure in the loop.

(^^) And consistent with other errors.

Acceptability of the nodalisation at the on-transient level

The demonstration of the nodalisation quality at the steady-state level does not ensure that the prediction of a transient scenario is phenomenologically correct or even that the nodalisation (input deck) is free of errors. Errors can be part of an input deck that has been qualified at the steady-state level. In order to achieve the full qualification the developed NPP input deck, already qualified at the steady-state level (following Chapter 2, Section 2.4.1.1), must undergo the on-transient nodalisation qualification process.

This is aimed at reproducing (or simulating) one or more transient scenarios in integral test facilities (ITF) that have similar characteristics as the scenarios that constitute the objective for the NPP-related application. In other words, the NPP nodalisation must be used to predict an experimental scenario. This is achieved through the so-called Kv-scaled procedure suitable to derive input boundary conditions for the NPP nodalisation, and to evaluate the comparison between NPP predictions and ITF data.

The quality demonstration of the output of the Kv-scaled calculation is obtained from the qualitative and quantitative accuracy evaluation, adopting suitable analytical tools.

The Kv-scaled calculation may be substituted by a proper benchmark activity, where NPP calculation results are compared with results of other qualified nodalisations and by performing the evaluation of suitable operational transients in relation to which data from the concerned NPP are available.

More information on suitable codes and examples on coupling them can be found in Chapter 2, Section 2.4 of the WP2 report.

7.3 Qualification of neutron kinetics nodalisations and codes

The qualification process of neutron kinetics nodalisations and codes is not as advanced as in the thermal-hydraulic field, and thus the statements and ideas discussed here should not be considered as a final solution of the issue.

Neutron kinetics nodalisation (input deck) development and qualification

In order to achieve a neutron kinetics nodalisation a much smaller number of code-user decisions and choices is needed in comparison to a thermal-hydraulic nodalisation. This can easily be understood by considering that only the core is concerned, rendering the process of developing a 3-D neutron kinetics nodalisation more straightforward.

Nevertheless, the generic recommendations given in Chapter 2, Section 2.4 and the acceptability criteria discussed in relation to thermal-hydraulic nodalisation (Section 2.4.1), can be translated into a number of preliminary requirements.

A first group of requirements (nodalisation development) includes the identification of:

- The core array matrix (square, triangular, plate, etc.).
- The variety of fuel elements or fuel assemblies (FA) that constitute the core.
- The variety of fuel rods that constitute each FA (burnable poisons, etc.).

- The relative position of control rods (CR) and FA and of fuel rods, as needed.

- The role of the fluid-dynamic core bypass and of the neutronic reflector including massive structures in radial and axial positions (barrel, fuel boxes, core support plates, etc.).

Ranges of validity of parameters, namely of cross-sections, should be compared with ranges of variations expected during the transient calculation. Thermal-hydraulic input parameters (required by the stand-alone 3-D neutron kinetics codes) should be properly identified and consistently fixed. User choices, including ADF, xenon consideration, etc., shall be properly justified, together with the selected value of the time step. Reproduction of reference code runs is recommended, before any use of the code.

More information on suitable codes and examples on coupling them can be found in Chapter 2, Section 2.4.2 of the WP2 report.

7.4 Coupling qualification

There are certain requirements concerning the coupling of thermal-hydraulic system codes and neutron kinetics codes that ought to be considered. The objective of these requirements is to provide accurate solutions in a reasonable amount of CPU time in coupled simulations of detailed operational transient and accident scenarios. These requirements are met by the development and implementation of six basic components of the coupling methodologies:

- Coupling approach – integration algorithm or parallel processing.

- Ways of coupling – internal or external coupling.

- Spatial mesh overlays.

- Coupled time-step algorithms.

- Coupling numerics – explicit, semi-implicit and implicit schemes.

- Coupled convergence schemes.

Various international exercises, mainly benchmarks, have been organised with the aim of providing a basic level of qualification of the tools utilised for the analysis. Recent NSC/OECD benchmarks have played an important role in fulfilling the purpose of qualifying the coupling methodologies.

More details related to these basic aspects of qualifying coupling methodologies are given in the WP2 report and related references.

Chapter 8

LICENSING STATUS

Since one of the CRISSUE-S project objectives is preparing a state-of-the-art report, the present chapter, devoted to licensing, is organised according to this aim. In this sense the WP2 report establishes the licensing status in different countries and focuses on the items in which coupled codes are currently used to solve licensing issues. These are: the ATWS issue, BWR stability and the prescribed RIA analysis Low Power.

The three issues are explained in the main report with emphasis to the acceptance criteria, applied list of quantities and threshold values acceptable to different regulatory bodies. The purpose of this chapter is to give an idea of the current status of the interaction between the regulatory requirements and the coupled 3-D neutron kinetics/thermal-hydraulic techniques.

8.1 The ATWS issue

Anticipated transients without scram (ATWS) are anticipated operational occurrences followed by the failure of one reactor scram function. Available transient analyses of ATWS events indicate that VVER 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 VVER plants also provide a degree of defence against ATWS.

The interpretation of the ATWS rules may vary dependening on country and reactor type. For example, there are not any specific ATWS rules in effect in Sweden since the probability for ATWS was considered to be very low because of the possibility to insert control rods by electrical motors in addition to the fast hydraulic insertion. The Swedish BWRs also have a manual boron system. The primary objective of the boron system is to keep the system subcritical in case several adjacent control rods can not be fully inserted.

The electrical control rod insertion system and the boron system do not fulfil all requirements that exist on safety systems, and new regulations are being discussed. The final proposal will probably be that all anticipated transients during plant lifetime should be analysed with respect to failure to scram. The success criteria will be that of the design basis accidents, *i.e.* fuel melting, cladding temperature and oxidation, and integrity of the pressure boundary.

8.1.1 Acceptance criteria (ATWS)

In general, ATWS acceptance criteria are comparable to the criteria for the postulated accidents and address issues such as primary system integrity, fuel cooling capability, shutdown reactivity and off-site radiological consequences. In particular:

- International practice accepts less restrictive reactor coolant system (RCS) pressure limits for ATWS. ATWS limits are generally based on allowable stress limits for ASME Service Level C or 135% of design pressure (for VVER-1000 reactors, this is also the primary circuit hydro test pressure).

- Fuel cooling capability should be demonstrated, *i.e.* core damage and calculated changes in core geometry do not prevent long-term core cooling. However, there is no international consensus as to acceptance criteria.

- The analysis of ATWS events should demonstrate that a safe and stable shutdown of the reactor is achieved. Common international practice is to accept a subcriticality of 1% as an adequate indication of reactor shutdown.

- For radiological assessment, a generally accepted approach is to assume that fuel cladding failure (leakage) occurs for all fuel rods having a potential for a boiling crisis, *e.g.* having a DNBR less than the value that gives 95% probability at 95% confidence level that DNB does not occur. It has not been necessary to apply less conservatism such as to demonstrate that site boundary limits for postulated accidents are not violated.

8.2 The BWR stability issue

The stability of BWR reactor systems has been a concern from the outset, and extensive experimental and theoretical studies have been performed in order to design a stable fuel and core configuration. The requirement of stability is expressed in the general design criteria for nuclear power plants as: "The reactor core and associated coolant, control and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed".

The problem seemed to be solved for quite some time, as no oscillation occurrence was experienced in a BWR for several years. Since then a number of fuel and core design changes have been performed and the power densities of the core have increased. Indications appeared in the late 80s that instability might still be a problem. Instability events had been observed in TVO-1 and Forsmark 1 (1987) and also in LaSalle (1988). Subsequently, authorities in all countries required a review of the stability features of their BWRs. All authorities included analyses in the safety analysis reports. Authorities in some countries determined that changes in the procedures and plant safety systems were required. In other countries, recommendations were made. In many cases the stability question is an integral part of the reload analyses using advanced methodology. Such technology is not generally used, and in many cases only engineering judgement is available. For other reactors possible exclusion regions are identified by measurements as a part of the start-up procedure.

Authorities have also emphasised the importance of training the operators to handle instabilities. One problem with this is that the computer programs' full-scope simulators can not truly simulate an occurrence of instability, a fact which also calls for development of improved simulation methods for operator training.

The major safety concern associated with instability is the cooling of the fuel and cladding integrity. One safety concern is the possibility of large undetected reactivity insertions taking place (for instance) during regional oscillations. In cases where the two halves of the core oscillate are in opposite phase, the oscillation may not always show up in the APRM signal, which is used by the reactor protection system. It has also been observed that severe un-damped oscillations may occur as a result of other kinds of transients such as feed water transients. Cases where external disturbances occur at or near the resonance frequency of the core are also of importance.

Modern technology in the control room may also make detection of instability more difficult. At one Swedish plant digital instrumentation had been introduced that prevented detection of instability although the amplitudes were large. Therefore dedicated stability monitors are in many cases used to give the operators a warning when unstable areas are approached.

In the long term, modifications to the protection systems are under consideration in many countries. These modifications should ensure that the reactor instrumentation and protection system can reliably protect the core against various kinds of instability without being erroneously activated under other circumstances.

8.2.1 Acceptance criteria (BWR stability)

One important parameter which characterises the stability features of a reactor is the decay ratio. The decay ratio is a measure for how quickly a perturbation of the reactor will die out. Limits of maximum allowed decay ratios in the core design are used in many cases. Such limits express the uncertainty associated with the specific method used and may vary between reactors and applications. In the past, such maximum limits have also been specified by the regulators.

As operations have become more aware of the safety concern related to regional oscillations, various criteria are being developed to be able to reliably detect and suppress such occurrences. However, such criteria or algorithms are parts of commercial products and not generally available.

The greatest concern is the possibility of fuel failure. One acceptance criterion is therefore to ensure that dry-out of the fuel does not occur. Another criterion may be to ensure that re-wet does occur after a dry-out of short duration, such that fuel failure does not occur.

8.3 The RIA issue and the events to be considered

A description of the methodology used when analysing control rod ejection in the context of a group of reactivity-initiated accidents is provided. The first part is focused on the methodology used by Westinghouse for rod ejection analysis in the Temelin NPP. The second part is devoted to the methodology used by VUJE Trnava (Slovakia) for control rod ejection analysis at the Bohunice NPP. Within this framework, the essential conclusions valid for VVER control rod ejection analysis, taken from the IAEA guidance, are accounted for and described.

The list of events required to analyse established by Czech Instructions and by US NRC RG 1.70 can be found in WP2 and the related references.

8.3.1 Acceptance criteria (RIA)

The rod ejection event is classified as a postulated accident. Postulated accidents are of such low frequency ($<10^{-2}$/year) that they are not expected to occur, though they are considered in the design. Acceptance criteria are listed hereafter:

1) Average fuel pellet enthalpy at the hot spot shall remain below a certain limit which may be dependent on fuel burn-up. This criterion ensures that core coolability is maintained.

2) Average clad temperature at the hot spot shall not exceed a certain limit and the oxidation of zirconium should be less than a certain fraction of the amount. This criterion ensures that clad melting and excessive clad embrittlement is avoided.

3) Peak reactor coolant pressure shall remain at a value lower than the value which could cause stress to exceed the faulted condition stress limits. This criterion ensures that the structure of the reactor coolant boundary is not threatened.

4) Fuel melting is limited to less than a certain fraction of the fuel volume at the hot spot, even if the average fuel pellet enthalpy is below the limits of criterion no. 1 above. This criterion ensures that fuel dispersal into the coolant will not occur.

More detailed information can be found in Chapter 6 of the WP2 report.

Chapter 9

USE BY INDUSTRY

The nuclear power industry, being the license holder of NPPs, must evaluate safety-related critical issues for their existing plants, for instance in case of core reloading, planned power up-rating, or as part of required analyses of occurred events. As the existing plants have well established licensing procedures, including well-founded analysis methods, the application of innovative analysis methods has to be thoroughly evaluated, with specific emphasis on the capabilities of producing results that in general terms might be beneficial as concerns the NPP operation.

Each licensing process related to a specific plant includes the analyses of specific transients that can become crucial depending on the selected core state, on the core fuel loading pattern employed, on burn-up and in general terms on the reactor operational history. In view of the most beneficial applications, any innovative analysis method should preferably focus on such crucial transients. In the PWR licensing process, for example, four or five licensing transients belong to this group, and the NK/T-H coupled evaluation is needed to fully explore the NK/T-H interactions and the associated influences on safety margins.

Every analysis method used in the licensing processes must be well founded and must have proven its capability to provide acceptable safety margins (best-estimate or conservative) from both the license holder perspective and that of the nuclear safety authority. A generally applicable role in relation to the nuclear safety of existing plants is that any modifications in the plant and/or in the applied licensing analysis methods have to reveal (at least) retained safety margins. When using new analysis methods a comprehensive validation database must confirm the adequacy of the methods and also, in the case of best-estimate methods, provide information of associated uncertainties in calculated values. With the new NK/T-H coupled analysis methods, provisions are at hand to accurately estimate the actual safety margins, which could provide incentives to more efficiently utilise the fuel and obtain cost benefits in the operation of the NPPs, while still preserving – and also possibly improving – the safety. It is also envisioned that the coupled T-H/neutronics analyses could provide more detailed insight on the basis used at the specification of operational procedures and could provide additional guidelines for optimisation of EOPs.

Thus, if the feasibility of a licensing process including new analysis methods is of high priority to the industry, the possibility of gaining margins is closely related. An adjusted margin evaluation normally results in a more flexible cycle scheme and consequently a more efficient use of fuel.

Preserving technical knowledge and competence in the fields needed for engineering support of operating plants is currently of utmost importance for the organisations responsible for NPP operations. In the context of deregulation and an open market for distributing electrical power, utilities are facing this issue as the conservation of an “investment” performed over the years. In this sense any innovation on coupled NK/T-H analysis can be even more acceptable if it has a feature using previous experience to support future studies, thus incorporating already gained knowledge.

In order to efficiently pursue coupled NK/T-H analyses there is an obvious need for more profound organisational capabilities than those likely to emerge with the mere symbiosis of neutron-kinetic and T-H safety experts. It can be envisioned that a well co-ordinated team of both types of specialists will take advantage of a helpful synergy to deal with common subjects. If this vision is fulfilled the results of any coupled analysis will thus be less dependent on any single user effect, providing an increased adequacy. State-of-the-art studies on any related subjects are normally welcomed with the intention of taking advantage of the main findings of such studies. This point applies not only to critical issues and subjects of first priority, but also to non-critical issues related to everyday plant operation.

Utilities are often in the position of having more than one alternative for the general control and quality assurance of calculations and results, especially in cases related to nuclear safety or fuel management. The use of alternative analysis methods or computer codes, developed in an independent framework, can be beneficial when comparing the results of the calculations performed by designers using licensing codes.

In core design, interaction between the core design team and an engineering team having developed an alternative method can be of great importance with regard to cycle-charging scheme optimisation and also for allowing validation of the performed calculations. In addition, this feedback is regarded by the regulatory authority as a proper procedure for the utility to verify the results based on benchmarks against a methodology that uses alternative tools.

Safety is strongly integrated in NPP operation and is regarded as an ever-present, natural element of day-to-day operation activities. NPP operation is strongly related to system and fuel design, control system design and general plant engineering, and has to properly account for nuances in associated properties. This is why flexibility concerning “every day” operation is very important from a utility standpoint. Such subjects are crucial when safety conclusions are applied and, definitely, they can also benefit from coupled NK/T-H analysis results. Coupled codes prove to be powerful tools that help improve the understanding of the phenomena taking place in the power plants, and provide realistic boundary conditions for system studies related to plant operation.

In some PWRs, discharge burn-up has reached more than 50 000 MWd/T, and tests performed at special locations in commercial reactor cores have brought test fuel assemblies to a discharge burn-up of almost 70 000 MWd/T. The interest of the industry in high burn-up is a reality. In the near future, an even higher burn-up will be considered and probably used, most likely along with new types of fuels. New analytical tools must be compatible with the use of such fuels, and must by applicable to the complete problem, including fuel pin performance and the tight interaction between core neutron kinetics and thermal-hydraulics. The validation has to be thoroughly pursued, and in order to be meaningful the analyses should be accompanied by associated uncertainty analyses so as to adequately determine the actual safety margins.

Chapter 10

RECOMMENDATIONS AND CONCLUSIONS

A detailed description of the CRISSUE-S project's fulfilment of its objectives can be found in the final chapter of the WP2 report. This present chapter is intended to emphasise, on one hand, the main findings of the project and, on the other hand, the open items that will explicitly or implicitly produce relevant recommendations for the future.

A discussion of competing techniques and the advancement of the state of the art have been studiously undertaken and reported with a detail characterising the involved phenomena, establishing the availability of tools and databases, and identifying areas of application as well as limitations.

The CRISSUE-S project has provided an opportunity for fruitful co-operation between industry, utilities, regulators, research institutions and between European and North American countries. It has also promoted dialogue and collaboration between users of different technologies. In addition to maintaining a close link with the VALCO project, CRISSUE-S has also had an interesting dialogue with organisations such as TEPCO, NUPEC and PSI, representatives of which attended at least one CRISSUE-S meeting as observers and have been informed of the progress of the project.

A relationship has also been established with the NSC and the GAMMA group of the OECD as well as with the IAEA.

Participant utilities have allowed a limited but significant use of NPP measured data to verify the qualification of the tools involved in the analysis.

The project has also managed to produce results in harmony with ongoing or proposed international programmes, including:

- Benchmark on NUPEC BWR Full-size Bundle Test (proposed by NEA/NSC).
- Benchmark on VVER-1000 Coolant Transient (supported by NEA/NSC and NEA/CSNI).
- Assessment of the Interface Between Neutronic, Thermal-hydraulic, Structural and Radiological Aspects in Accident Analysis (launched by IAEA).
- VALCO project (EC-funded project in the 5th Framework Programme).

The most significant aspects of the main issues (thermal-hydraulics, neutron kinetics, the role of fuel, the regulatory position or general databases of both results NPP data) are covered by CRISSUE-S. Although operator training seems to be a subject not fully integrated in the process of obtaining suitable analytical results, the project has also identified its relevance in cases of RIA.

10.1 Main findings

The connections or the links existing among a number of subjects (*e.g.* derivation of neutron cross-sections, BWR stability, numerical solution methods, fuel modelling) that have been separately treated in the recent past have been highlighted. This may be seen as the first finding of the activity, and emphasises the need to pursue an interdisciplinary approach in the future for the subjects concerned.

- *Capabilities of 3-D thermal-hydraulic computational tools*. Codes are considered here that allow the 3-D porous media modelling approach (*i.e.* those codes that do not have the capability to predict the local turbulence-associated phenomena like the velocity or temperature profile inside a cross-section). The conclusion is that these codes are available and ready to be used but their qualification level is not acceptable.

- *List of transients in relation to which analysis is recommended*. The prediction of NPP scenarios constitutes the final goal of coupled neutron kinetics/thermal-hydraulics techniques. However, the full exploitation of these techniques implies streamlining both their use and their accompanying recommendations. Therefore transients have been selected within the present framework whose analysis is recommended by coupled techniques. A distinction has been made between BWR, PWR and VVER (Chapter 1, Sections 1.2 to 1.4), and only the PWR list of transients is proposed hereafter. This includes:

  – MSLB.

  – LOFW-ATWS.

  – CR ejection.

  – LBLOCA-DBA.

  – Incorrect connection (start-up) of an inactive (idle) loop.

  – MSLB-ATWS.

  – SBLOCA-ATWS.

- *Identification of thresholds of acceptability*. Answering the question “How good is good enough?” constitutes the timeless, daunting task that cannot be “solved” once and forever. The progress of science always further increases the acceptability of any technological product; such is the case for the results of code applications to nuclear reactor safety. However, an attempt has been made (Chapter 2, Section 2.5) to fix acceptable targets for calculation precision, distinguishing between (acceptable) quantity and time errors. The considered values give an idea of the current capabilities of computational tools, provided that qualification methods are adopted. Four examples of acceptability thresholds from more than a dozen considered in Chapter 2, Section 2.5 are reported below:

  – For pressure pulses characterised by FWHM < 0.1 s, the acceptable error is 10% nominal pressure.

  – For core power pulses characterised by FWHM < 0.1 s, the acceptable error is 100% nominal power or 300% initial power, whichever is smaller.

- For core power pulses characterised by FWHM $\geq$ 0.1 s, the acceptable threshold error is 20% nominal power or 100% initial power, whichever is smaller.

- The acceptable time error that can be associated with the prediction of time of occurrence of pressure or core pulse is 100% of the best-estimate value.

The acceptability of neutron kinetics calculation shall be consistent with discrepancies in the application of “state-of-the-art” codes to reference problems (see Chapter 2, Section 2.12 of the WP2 report). Errors in predicting k_{eff} (or k_{∞}) can be expected to be of the order of 3-4% without adaptation to measurements of actual core states. However, the common practice is to apply some kind of adaptive strategy to modify or correct calculated values to match specific core measured data. With such an adaptation the prediction of corresponding “off-state” conditions will provide much higher precision, and the errors in calculated k_{∞} (or k_{eff}) will usually be of the order of a few hundred pcm, which can be viewed as acceptable.

- *Relevance of the role of fuel*. The minor role of the fuel within the coupled neutron kinetics/ thermal-hydraulic environment was noted. Actually, transient modelling of fuel performance must become the “third ring” of an ideally un-broken chain which includes the two other disciplines. Changes in the geometric and physical properties of the fuel are expected during RIA and must be properly accounted for in the modelling. In case of a local power, for example, increased fuel fragmentation and release of fission gases may have an effect upon the neutron kinetic cross-section as well as upon the thermal properties of the gap and of the fuel itself. The feedback in terms of neutron power and of power transmitted to the coolant might be non-negligible. No approach deals with this phenomenon at present.

- *Uncertainty*. Due to a various reasons discussed in the report, the results from coupled 3-D neutron kinetics/thermal-hydraulic calculations are not precise. Therefore, uncertainty must be connected to any prediction. Two main findings emerge from the performed activity. The first one is linked with the determination that the Internal Assessment of Uncertainty method, based upon the interpolation of the detected accuracy, is applicable for coupled techniques even though relevant resources are necessary to develop a suitable error database. The second one is connected with the identification of the main sources of uncertainty. These can be summarised as follows:

  - Subcooled HTC, namely in transient conditions, characterised by a large time derivative for the produced power.

  - Pressure wave propagation inside the complex vessel geometry when a single phase of two-phase fluid is present.

  - Effect of (prompt) radiolysis upon neutron cross-section following fast transients.

  - Effect of direct energy release to the coolant following a fast transient.

  - Detailed specifications about mechanical components such as valves and control rods. The function valve net cross-sectional flow area versus time during opening or closure events is irrelevant for the majority of thermal-hydraulic transients, but may become important when neutron kinetics feedback is computed. Analogously, the velocity as a function of time during a control rod ejections transient is relevant (*i.e.* not only the total ejection time).

– Parameters connected with fuel must be calculated during RIA transient allowing proper feedback (*e.g.* neutron kinetic cross-section and thermal parameter, like gap thickness and conductivity should be calculated as a function of time and provide suitable feedback for the power evaluation).

Uncertainty sources connected with the selected neutron kinetics parameters are mentioned in Chapter 2, Section 2.12. As examples, rod worth, fraction of delayed neutron and Doppler coefficients are typically defined with uncertainties of ±10%, ±5% and ±20%, respectively.

- *Licensing status*. The main finding is that great attention is paid by regulatory bodies toward issues like control rod ejection and ultimate fuel resistance. However, the importance of the use of coupled 3-D neutron kinetics/thermal-hydraulic techniques is not recognised. Making reference to the ATWS issue, the survey of the regulatory position in eight countries brings attention to the following additional main findings:

  – ATWS analyses are mandatory (*i.e.* foreseen in the licensing process) in a small number of countries compared with the number of countries that were taken into consideration. However, generic safety analyses are performed in almost all countries to address the issue.

  – The attention toward coupled 3-D neutron kinetics/thermal-hydraulic techniques from the regulatory bodies is insignificant in almost all cases.

- *Created database*. The database of results presented in Annex I gives an idea of the transient behaviour of the selected NPP in cases when neutron kinetics/thermal-hydraulic coupling is relevant and constitutes a reference for those code users engaged in similar analyses.

10.2 New frontier

The new frontier for research and/or recommendations for future development can be derived from the performed analysis and, specifically, from the items summarised in this chapter. Therefore, a straightforward list of recommendations is:

- Systematic qualification of the various steps of the process of the application of coupled neutron kinetics/thermal-hydraulic techniques is needed, including the adoption of available tools and procedures.

- Full consideration shall be given to the identified sources of uncertainties (see above).

- It must be clear that results obtained by 0-D neutron kinetics coupled with thermal-hydraulics are not necessarily conservative. This observation must provide the impetus for extended applications of coupled 3-D neutron kinetics/thermal-hydraulic techniques in the nuclear safety domain.

- The integration of nuclear fuel related models into neutron kinetics thermal-hydraulics coupling is needed. This also includes the chemistry, with main reference to the processes of crud formation and release.

- The industry and the regulatory bodies should become fully aware of the capabilities (and the limitations) of the concerned techniques. It is expected that regulatory bodies consider the list of transients in LWR whose analysis is recommended by coupled techniques: BWR stability, ATWS and those transients characterised by asymmetric core behaviour are the most important. On the industry side, operators should be made aware of transients that are calculated by 3-D coupled neutron kinetics/thermal-hydraulic techniques and simulators, control rooms and EOP should benefit from the experience gained from the application of those techniques.

The prepared database is a good starting point that can be improved and completed with the information of other databases or in connection with organisations such as WANO.

The search for NPP data during the CRISSUE-S project in order to create the database helped to clarify future needs for data recording in commercial reactors. In this sense it is recommended to improve in-core parameter recording and to qualify such plant data both in PWR and BWR.

Finally, although industry faces safety-related critical issues with mainly realistic improvements of existing licensing procedures in mind, utilities accept to take into account innovative methods when they have the capability to produce a benefit in general terms. Consequently, it seems mandatory to use these new techniques for the design of a new generation plant.

Each licensing process is related to a specific plant, and has specific transients that become crucial depending on different aspects that are analysed in CRISSUE-S. With regard to fruitful application, any innovative method should focus on the analysis of crucial transients.

Industry believes that gaining margins comes along with the improvement of the licensing process, and that improvements usually result in a better use of fuel.

Having more than one alternative for the general verification of calculations, particularly if they are related to safety or fuel management, and also conserving technical knowledge applied to the field of engineering support to operating plants, are currently important concerns in the operation of NPPs that can be addressed if the recommendations of the CRISSUE-S project are applied on a realistic basis and if the technical dialogue between utilities, regulators and research institutions used during the project are maintained.

A natural continuation of CRISSUE-S project was the project developed by some of the partners and presented in at the POST-FISA Workshop 5 on 13 November 2003 as *US-EC Co-operation in State-of-the-Art Nuclear Safety Research – Pilot Project on Accident Analysis Codes and Uncertainty Analysis Methodology*. The scope of this project is intended deal to with:Investigations on the completion of the uncertainty analysis methodology for coupled 3-D neutron kinetics/thermal-hydraulics codes.

- Studies on improving coupled calculations by utilising CFD results in the coarse-mesh thermal-hydraulics models.
- Investigations on extending the modelling capabilities of coupled codes to provide predictions of local safety parameters (on pin level).
- Studies on modelling coupled neutronics/thermal-hydraulic/mechanical phenomena.
- Investigations on extending uncertainty analysis for local (pin level) coupled code predictions.

A consolidation of the knowledge and technical background analysed during the CRISSUE-S project is expected from this development, as is the wider application of the involved techniques to current and future power plants.

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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.