OVERVIEW OF THE METHODS DEVELOPED FOR FISSION PLANTS SAFETY RELEVANT TO THE SAFETY OF FUSION FACILITIES

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Abstract

Safety studies for fusion facilities are commonly conducted using codes originally developed for fission reactor accident analysis and adapted to model the fusion-relevant phenomena. Nevertheless there are many “fission developed” methods still not considered in fusion safety assessment which could offer significant advantages in the fusion power commercialization. Along with solving the safety and licensing critical for the fusion power commercialization will be as well the ability to reduce the cost and increase the efficiency of the power production. Among other means these were achieved in the fission power by limiting or even avoiding the conservatism in safety assessment, by improving the methods and use of the state-of-the art tools. The paper addresses the following topics: Experimental programs, Test Matrixes and Data bases; Phenomena Identification and Ranking Tables (PIRT); and Computer codes development. For each of the above topics a brief presentation of the fission historical development followed by an overview of published adaptations of methods and their applications to fusion safety is reported. The presentation draws in particular on availability of qualified tools for accidents analyses, use of PIRT, the verification and validation of computer codes by means of separate and integral effect tests and establishing benchmark problems as well on code assessment and development of multi-physics, multi-fluids integrated code systems. These efforts should be aimed at developing a systematic safety demonstration defined by an integrated fusion safety assessment methodology.

1. INTRODUCTION

Safety demonstration of fusion facilities are commonly conducted using methodologies, methods and tools developed for fission reactor safety analysis and somewhat adapted to fusion specificities. Thus the computer codes originally developed for fission reactor accident analysis were adapted to model the fusion-relevant phenomena (see for example [1], [2] and [3]). We are not aware of a comprehensive overview of applicable “fission developed” methods to be considered in fusion safety assessment. These methods could offer significant advantages in the fusion power commercialization. Along with solving the safety and licensing issues critical for the fusion power commercialization through ITER, DEMO and Fusion Power Plant (FPP) will be as well the ability to reduce the cost and increase the efficiency of the power production. These had been already observed in the fission power history. Among other means these were achieved in fission power by limiting or even avoiding the conservatism in safety assessment, by improving the methods and use of the state-of-the art tools. Recalling the fission achievements would boost the recognition, usefulness and importance of fission development lessons to the commercialization of fusion power. The fusion power is still in its early 1950s (when the fission power plants were in the design phase) and still on the way to construct their “Obninsk1”, “Colder Hall2” and “Shippingport3” plants. Among the many reasons for looking into the fission safety studies the following seem to be the most important ones:

— Nuclear Regulatory environment will not differ significantly for fission and fusion power as shown in the case of ITER. The very same nuclear safety principles and regulatory limits apply to fission and fusion4;
— Enormous knowledge, experience gained and data collected into fission safety;
— Used of mature and proven methods already accepted by the nuclear regulators;
— List the possible synergies: methods and tools to be used with reasonable effort spend on adaptation, verification, validation and qualification to fusion;

1 http://large.stanford.edu/courses/2015/ph241/morrissey2/ Retrieved 13 September 2018
2 http://www.world-nuclear-news.org/Articles/UK-marks-60th-anniversary-of-Calder-Hall Retrieved 13 September 2018
3 https://www.asme.org/wwwasmeorg/media/ResourceFiles/AboutASME/Who%20We%20Are/Engineering%20History/Landmarks/47-Shippingport-Nuclear-Power-Station.pdf Retrieved 13 September 2018
4 Assuming fusion power will explore the deuterium-tritium fuel cycle
— Overview of on-going fusion safety activities;
— List the existing gaps and development needs of fusion safety;
— Identify the fields in which enhanced international cooperation in fusion safety is needed and maximize the effect of the collaborations;
— Stimulate the systematic development of fusion safety methodologies utilizing the experience and lessons learnt in fission safety.

Finally the conservatism used in ITER safety studies would evolve in somewhat more reasonable conservatism in DEMO design and obviously shall be replaced by best estimate (BE) methods applied to FPP safety. While such evolution is logical and will serve the commercialization of fusion power it can’t be taken for granted. As the history of fission power demonstrated it requires a plenty of effort and wide international cooperation to achieve the economical competiveness of the new power source. Luckily the fusion power has the experience, the lessons learnt and tremendous knowledge acquired in pursuing of fusion power commercialization and its competitiveness development.

The discussed below achievements of the fission plants safety were in my scope as had been working for a long time on the fission thermal-hydraulics, reactor safety, coupled and integrated codes development, verification and validation (V&V) and application. When faced to the challenges in developing a comprehensive fusion breeding blankets accident analyses methodology [4] in condition of limited availability of experimental data for models’ V&V I realized that the experience gained in fission application and situation faced by our fission colleagues about a half a century was very similar to ours. Thus I would try to guide you back in time to see again the evolution of the fission safety tools and methods, experience gained, the approaches followed and achievement reached.

The fission safety studies presented hereafter are solely selection of the author. Although I tried to avoid the bias towards ones that have been (one or other way) engaged in their execution, application or discussion such may still persist. The list of studies doesn’t pretend to be exhaustive although the attention was paid to more contemporary s and judged to be generic and important studies the subjectivity of the selection and deliberate bias towards the areas of interest of the author of this manuscript are recognized. Moreover due to the limited space allocated equally relevant fission safety studies that had been performed in the past or are currently ongoing remained outside this manuscript.

This paper focus on the methodologies and methods related mainly to the accident analyses. Recently brief historical overview of the fusion safety studies is provided in [4] and [5] and there is no need to reproduce it here. The following topics are covered: Experimental programs, Test Matrixes and Data bases (section 2); Phenomena Identification and Ranking Tables (PIRT) (section 3); Computer codes development (section 4).

2. EXPERIMENTAL PROGRAMS, TEST MATRIXES AND DATA BASES

In the mid-seventies experiments and analytical evaluations revealed that multidimensional thermal-hydraulic (TH) phenomena could have significant impact on loss-of-coolant accident (LOCA) transients in PWRs. But even the largest test facilities in operation at that time (e.g., LOFT, LOBI, or PKL) were scaled down geometrically by two or three orders of magnitude. Therefore these facilities could not resolve the issues associated with multidimensional effects on emergency core cooling (ECC).

An illustration of the international efforts in this fields is the 'The International Program on the Thermal-Hydraulic Behaviour of ECC during the Refill and Reflood Phases of a LOCA in a PWR'' (1978-1993) also widely known as "2D/3D Program" because phenomena addressed are strongly influenced by multidimensional (2D and 3D) effects. The 2D/3D Program [6] was carried out by Germany, Japan and the United States to investigate the thermal-hydraulics of a PWR large-break LOCA. Similar to ITER project a contributory approach was utilized in which each country contributed significant effort to the program and all three countries shared the research results. Germany constructed and operated the Upper Plenum Test Facility (UPTF), and Japan constructed and operated the Cylindrical Core Test Facility (CCTF) and the Slab Core Test Facility (SCTF). The US contribution consisted of provision of advanced instrumentation to each of the three test facilities, and assessment of the TRAC computer code against the test results. Evaluations of the test results were carried out in all three countries. A major analysis program involving the development, assessment and use of a BE computer code was carried out in the US. TRAC analyses of PWRs and selected tests were also performed by Japan and Germany. There was no exchange of funds between the participants. This approach fostered technical cooperation among the three countries. The primary objective of research has been to improve
the understanding and modelling of the phenomena so that safety margins can be better quantified and more realistic evaluation approaches can be utilized.

In addition, safety evaluations in the framework of licensing procedures for nuclear power plants employed conservative assumptions and calculation models to envelope the key parameters of principal safety significance. But in the late seventies the need for best-estimate (BE) evaluation of core damage to be expected during a LOCA was recognized. Such analyses were needed for risk assessment studies. The technical results and the experience gained by the 2D/3D Program enabled to close the issues about DBA and concentrate in the future on issues arising from BDBA and accident management. Work on these issues further improved the safety of nuclear energy production [7).

Separate from the 2D/3D Program the USNRC developed a methodology to evaluate TH Code Scaling, Applicability and Uncertainty (CSAU) [8]. The results of the 2D/3D program along with many other experimental studies triggered the establishment of the Organization for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) Separate Effects Test Matrix and Integral test facility validation matrix defining the phenomena of interest, collecting and systematizing the worldwide available experimental data.

2.1. Fission Power Test Matrixes and Data Bases

An internationally agreed Separate Effects Test (SET) Validation Matrix for thermal-hydraulic system codes [9] has been established in early 1990s by a sub-group of the Task Group on for Thermal-Hydraulic System Behavior as requested by the OECD/NEA Committee on Safety of Nuclear Installations (CSNI). The construction of such a matrix was an attempt to record information which has been generated around the world over the last 20 years so that it is more accessible to present and future workers in the field than would otherwise be the case.

The methodology that has been developed during the process of establishing SET validation matrix is summarized as follows

a) Identification of the phenomena relevant to two-phase flow in relation to LOCAs and thermal-hydraulic transients in light water reactors (LWRs).

b) Characterisation of phenomena, in terms of short description of each phenomenon, its relevance to nuclear reactor safety, information on measurement ability, instrumentation and data base. In addition to these points, the present state of knowledge and the predictive capability of the codes is included in the characterisation of each phenomenon.

c) Setting up a catalogue of information sheets on the experimental facilities, as a basis for the selection of the facilities and specific tests.

d) Forming a separate effects test facility cross-reference matrix by the classification of the facilities in terms of the phenomena they address.

e) Identification of the relevant experimental parameter ranges in relation to each facility that addresses a phenomenon and selection of relevant facilities related to each phenomenon.

f) Establishing a matrix of experiments (the SET matrix) suitable for the developmental assessment of thermal-hydraulic transient system computer codes, by selecting individual tests from the selected facilities, relevant to each phenomenon.

Information on facilities and experiment characteristics of 187 test facilities has been consolidated in a systematic way. Sufficiently complete list of relevant phenomena for PWRs and BWRs LOCA and non-LOCA transient applications has been identified. To this end 67 phenomena were identified for inclusion in the SET matrix. In all, about 2094 tests are included in the SET matrix.

CSNI Integral test facility validation matrix for the assessment of thermal-hydraulic codes for LWR LOCA and transients [10] is similar to SET matrix but

i. Integral test facilities (ITF) for the validation of BE thermal-hydraulic computer codes

ii. Revises and combines the works on PWR (SINDOC(86)12) and BWR (SINDOC(86)13) matrices performed on 80s [10].

iii. Includes the tests performed under the 2D/3D program.
iv. Defines phenomenologically well founded set of experiments, for which comparison of the measured and calculated parameters forms a basis for establishing the accuracy of the test predictions. A further step – estimation of the capabilities to simulate real plant behaviour was outside of the scope.

v. Reminds that the comparison of codes against a limited number of SETs may also be of value.

vi. Attempts made to formulate general validation matrices by including phenomena of interest for most cases of plans.

vii. Recognises that under certain conditions some special cases of reactors may display phenomena not adequately addressed by the matrices produced. In such cases, additional validation against experiments addressing these phenomena might be needed.

OECD International Standard Problems (ISPs) programme has contributed to code assessment [11], [12]. Russian VVER plants validation matrix with related tests had been developed by another CSNI group [13], [14].

2.2. Fusion Safety Experiments

The fusion safety experimental activities in 1990 and 2000s were devoted to main physical phenomena governing the accidental sequences related to water/steam discharge into the vacuum vessel or the cryostat. The typical phenomena investigated were the pressurization of a volume at low initial pressure, the critical flow, the flashing, the relief into an expansion volume, the condensation of vapour in a pressure suppression system, the formation of ice on a cryogenic structure, the heat transfer between walls and fluid in various thermodynamic conditions.

The EVITA (European Vacuum Impingement Test Apparatus) facility⁵ [15] investigated the phenomena during ingress of coolant (water/steam) into the cryostat, i.e. into a volume at low initial pressure containing surfaces at cryogenic temperature. Simultaneous ingress of water/steam and non-condensable gas; ice formation kinematics; heat transfer characteristics; total condensed water mass; cryogenic surface temperature; dynamic pressure and temperature in the vessel were the main subjects of interest. Two of the tests series (#5 and #7) are described in [15] and [16].

The LOVA (Loss Of Vacuum Accident) and ICE (Ingress of Coolant Event) facility⁶ tests ([17], [18]) simulated an water/steam discharge into the vacuum vessel and propagation of shock wave into the vessel pressure suppression system. The experiments were reported to investigate fluid flow configurations in the vessel; pressure transients in the vessel without and with available suppression system; the effects of water and wall temperatures, water pressure, heaters compensation and nitrogen control, divertor slit area design, and number of relief pipes.

The dust resuspension data obtained in STARDUST-U (Small Tank for Aerosol Removal and DUST-Upgrade)⁷ facility reproducing experimental conditions comparable to those expected in ITER in case of LOVA or LOCA might be useful for the development of 3D multi-phase numerical models to be implemented in the safety codes [19], [20] and [21].

It has to be noted that with the implementation of EU, China, Korea and Japan DEMO programmes⁸ a significant effort is devoted to experimental studies [22], [23]. However the author is not aware of any justification and planning based on detailed review of the phenomena to be studied and a systematic approach expressed in validation test matrices. There are several steps that might be undertaken to assure the availability of needed experimental data and sufficient knowledge for future FPP license and the following are among them

a) Identify the DEMO⁹ relevant phenomena in a systematic way (by using PIRT for example – see section 3 below). This step should consider the synergies with design and technology demonstration activities.

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⁵ located in Cadarache, France

⁶ operated by JAERI, Japan

⁷ ENEA Fusion Technology, Frascati, Italy

⁸ See also the papers from the recent ISFNT, SOFT and SOFE conferences

⁹ In general there shall be no FPP phenomena not addressed in DEMO – only scaling and application domain might eventually differ.
b) Identify the needs/gaps for experimental studies on both separate effects and integral and plan the facilities and experimental campaigns

c) Systematize the available experimental data into matrices and data bases

d) Perform the sensitivity analyses and evaluate the uncertainties

e) Identify the existing gaps in existing computer codes and development needs for them (see sections 3 and 4 below on Computer codes development, verification and validation, and assessment)

PIRT incorporation into the fusion safety will make possible a systematic approach to the safety studies starting with the experimental programmes and will facilitate the definition of codes development needs as the latter is just a subset of the former.

The experimental programs and available data bases in fission boosted the computer codes development, verification and validation.

3. PHENOMENA IDENTIFICATION AND RANKING TABLES (PIRT)

Phenomena identification and ranking tables (PIRT)\textsuperscript{10} is a systematic way of gathering information on a specific subject, and ranking the importance of the information, in order to meet some decision-making objective, e.g., determining the highest priority for research on that subject \cite{24}. PIRT is a technique used in all five safety topic listed above and as such it will be addressed across the paper. It has been successfully applied to many nuclear technology issues since it was first developed and applied in the late 1980s \cite{8} and then developed into a generalized process \cite{25} (see Fig. 1)

\begin{itemize}
  \item The PIRT process results in lists of phenomena which are germane to a particular subject (a very specific figure-of-merit).
  \item The “phenomena” can actually be the condition of a particular reactor/system/component, a physical or engineering approximation, a reactor parameter, or anything else that might influence the figure-of-merit.
  \item The process proceeds by ranking these phenomena using some scoring criteria in order to help determine what is most important. That ranking, as well as the rationale for the ranking along with the information obtained to explain the ranking, can assist in decision making.
\end{itemize}

\textsuperscript{10} In non-nuclear applications it is often called Phenomena identification and ranking Technique (PIRT).
• An important part of the process is to also identify the uncertainty in the ranking, usually by scoring the knowledge base for the phenomenon.
• Examples of successful PIRT applications exist in thermal-hydraulics, severe accidents, fuels, materials degradation, and nuclear analysis.
• Phenomena and processes are ranked in the PIRT based on their influence on primary safety criteria, and efforts focused on the most important of these. This process has proven valuable in other contexts and its specifications have been broadened over the years.

Interpretation of the PIRT depends on details of the objectives, rankings are with respect to

- design of an experiment => need for accurate measurements and need for care in scaling to properly capture its effect in a full-scale system.
- improve modelling in a simulation code => level of detail required in special models programmed for the phenomenon or process.
- sensitivity study the ranking permits a practical statistical analysis.
- Phenomena with low importance may be dropped from the uncertainty analysis, or their impact estimated with bounding calculations.
- Highly ranked phenomena are treated individually and perturbations of underlying models properly included in statistical methodology.
- Treatment of phenomena with a medium ranking is done on a case by case basis.

The value of the final PIRT is directly proportional to the degree of detail in the initial specification of a transient scenario and system in which the scenario occurs. Creation of a PIRT is an iterative process. After it is first applied results of requested experiments, sensitivity studies, or other results from simulations may require revisions to the original PIRT and associated documentation.

The value of the PIRT process has been recognized outside the nuclear safety community as an important component of any validation process.

The PIRT process demonstrated to be a powerful tool and has shown to be a robust means to establish safety analysis computer code phenomenological requirements. PIRT is a practical and flexible technique that offers a holistic approach to the safety assessment. It results can be used among others: (1) to identify, categorize, and characterize the phenomena and issues relevant to the risk and safety; (2) prioritize research activities to address the safety significant issues; (3) inform decisions regarding the development analytical tools for safety analysis; (4) defining the course of accident sequences and defining safety system success criteria; (5) technical basis and cost effective organization for new experimental programs; and (6) provide insights for the review of safety analysis and supporting data bases.

Overview of the PIRT applications to the following fields related to the fusion accident analyses: i) experimental programs, separate effects and integrated test matrices; ii) codes’ development; iii) code assessment and; and iv) uncertainty evaluation was present in [27]. The application of PIRTs for the definition of the fusion accident analysis specifications, assessment and selection of analysis code(s), development and qualification of the codes models and uncertainty evaluation was illustrated on the example of EU Test Blanket Systems and generic DEMO plant. At present we are not aware of any PIRT application to fusion safety and accident analyses although we note the systematic approach followed by the Korean DEMO project [28], [29] that foresees application of five analytical tools to develop the safety requirements: qualitative safety features review, PIRT, objective provision trees, probabilistic safety assessment (PSA), and deterministic and phenomenological analysis [28].

PIRT have the potential to be employed widely in fusion safety – for example in the following [27].

• Identify the DEMO relevant phenomena in a systematic way
• Identify the needs/gaps for experimental studies on both separate effects and integral and plan the facilities and experimental campaigns
• Systematize the available experimental data into matrices and data bases
• perform the sensitivity analyses and evaluate the uncertainties
• Computer codes assessment - identify the existing gaps and development needs
• PIRT has to be incorporated into the Fusion Safety Assessment
4. COMPUTER CODES DEVELOPMENT

In parallel with the experimental programmes described in section 2 the TRAC and RELAP codes development was supported by CSAU [8] project carried out in the beginning by US NRC and complemented later by other international cooperation (mainly of NEA) projects. These decades long codes’ development went through 3D TH capability development, coupling it with 3D core neutron kinetics modelling and integrating (fuel) thermal-mechanics (TM) behaviour. Recent efforts are focused on the integrated platforms with neutron and reactor physics, TH, TM (including CFD), source-term and atmosphere dispersion and some specialized codes (in the EU) or on advanced simulation and virtual environment reactor applications (in the USA).

The TRAC series of computer codes were extensively assessed against data from the 2D/3D Program [6]. Over the course of the program, results from the assessment calculations performed in the program were continually fed back to the TRAC developers and this contributed significantly to improve the quality of the code. RELAP5 codes have been further developed within the International Code Assessment Program (ICAP) [30]. This section will focus on the codes development while code assessment will be discussed in the following section.

4.1. Coupled Codes

In the late 1980s and early 1990s it became evident that the use of point kinetics approximation in the system codes might be non-conservative. For the first time this has been widely demonstrated in the frame of OECD NEA Pressurised Water Reactor Main Steam Line Break (MSLB) Benchmark [31] based on real plant design and operational data for Three Mile Island – Unit 1 Nuclear Power Plant (NPP)11. This lead to further development of so-called coupled codes in which system and core thermal-hydraulics (TH) capability was coupled with (core) 3D neutron kinetics (NK) capability. The earlier examples of such coupling were RELAP/RAMONA, TRAC-PF(BF)1/NEM, QUABOX/CUBBOX-ATHLET, ATHLET/DYN3D, RELAPS/PANTHER, CATHARE/FLICA4/CRONOS2, RELAPS/PARCS, etc. In this early stage of computer codes coupling different approaches were elaborated. The three reports of reference [32] offer a comprehensive study on the State-Of-the-Art (SOAR) in Neutronics/Thermal-hydraulics coupling in LWR Technology. The EU funded CRUSUE-S project re-evaluated the fundamental issues in the technology of light water nuclear reactors with emphasis on the interactions between neutron kinetics and thermal-hydraulics. Providing results of best estimate analyses of complex transients in existing reactors the issued SOAR focused on fuel performance, fundamental thermal-hydraulics and neutronics modelling and on 3-D neutron kinetics / thermal-hydraulics coupling techniques. Within this framework, results of selected transient scenarios involving the above coupling for PWR, BWR, WWER-440 and WWER-1000 reactors were discussed.

The more obvious (and least effort) ways and thus the first ones were so called “direct”, or “iterative loose” or “external” or “parallel” coupling. Different in their nuances all they preserved the original equations systems12 and solvers of the two codes and used simple exchange of data to be used as boundary conditions at the core inlet and outlet interfaces for the next time step solutions. However it turned out that “Coupling of different codes that model the same phenomena or processes by different equation systems and constitutive equations and make use of different solvers may lead to non-physical numerical oscillations and perturbations in the solutions and the simulation results” ([32] volume 3). Any tools developed to smear out the perturbations normally utilizing numerical filters failed as they were not able to distinguish the physical oscillations (to be preserved) from the numerical ones (to be filtered out). The “internal” coupling required much significant initial development effort but lead to robust and coherent solutions by using the kinetics equations from the core simulator and applying the system code TH also to the core meshes. Full “internal coupling” incorporates all equations in a single (stiff) system solved by a single solver simplifying significantly the time step management procedure, while used in some cases two different equation systems for kinetics and TH each with its solver had to be complemented by time step synchronization management.

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11 Performed comparison demonstrated that the 3-D analysis removes some of the conservatism inherent in point-kinetics analysis. The differences are believed to be caused by the inability of the standard point-kinetics approach to properly account for the moderator density feedback, dynamic scram simulation, local effects and flux redistribution which occur during the transient. As a result the 3-D core transient modelling provides a margin to re-criticality over the point-kinetics approach during an MSLB analysis.

12 Normally the core simulators had their own TH capability although quite simplified in most of the cases.
Other examples of successful coupling are the coupling of System TH code with CFD for example when the greater (generally 3D parameter distribution) detail is required in some location of the reactor loop as in the lower plenum in case of boron dilution incident caused by non-borated water ingress or reactivity insertion by cold water ingress into the reactor coolant loop. Coupling of System TH code with severe accident code (like SCDAP/RELAP5, SCDAP/RELAP-3D, RELAP/SCDAPSIM) were used for the detailed analyses of accidents that evolve from Design Basis Accident (DBA) into severe accidents and are in particular interest in the development of emergency response procedures.

4.2. Integrated Codes

By the mid-1990s, US DOE Office of Nuclear Energy, Idaho National Laboratory (INL), other DOE national laboratories, and private companies developed a real-time version of the RELAP program for use on operator training simulators. Thus RELAP5-3D was born [33]. Over time, many new features and capabilities have been incorporated into RELAP5-3D and major physics improvements included: full three-dimensional hydrodynamics with rectangular, cylindrical, and spherical geometries; variable gravity to model moving systems; multidimensional neutronics with nodal kinetics; massively parallel neutron kinetics; one- and two-dimensional heat transfer including conduction, convection, and radiation; gas diffusion, radiological transport, and alternate heat conduction to fluids; new hydrodynamic components: Emergency Core Cooling mixer, feedwater heater, and compressor; Godunov second-order-in-space boron tracking model; numerous working fluids and noncondensable gases, etc.

The industry went further by integrating the core simulators normally used for design and reload analyses of operating plants and (fuel design and analyses) thermal-mechanics codes into the TH system code otherwise used mainly for transients and accidents analyses. Westinghouse POLCA-T code is an example “internal” coupling not only using a single equation systems and solver of but also utilizing (with proper development and adaptation) the very same set of nuclear data for the core design and accident analyses [34], [35] and [36]. The code and related methodologies were in late 2000s successfully licensed in the US for BWRs design and accident analyses including transients/accidents and also reactivity initiated accidents (RIA) and BWR stability analyses [37]. This type of codes was often called also integral [38].

Coupled TRAC/PARCS codes appeared as part of US NRC codes consolidation project [39] that culminated in the release of TRACE engine [40]. The code performs computational simulations of transients in NPPs or any other system involving two-phase flow and heat transfer. Specific research tasks under the project included: Advanced Numerical Methods (Higher Order and Implicit); Multi-field modeling of droplets (dynamic flow regime); Improved modeling of two-phase choked flow; Improved code architecture; Interface for distributed parallel computing; Code verification and validation. This software has been applied by the US NRC in licensing decisions on new reactor designs (AP1000, ESBWR), and on existing plants.

4.3. Integrated Platforms

In this section we consider three European and three US project developing the integrated platforms for safety analyses.

European Union NURESIM (2005-2008) project [41] started the integration of different simulation codes under a two unique platforms: Core Physics and Thermal-hydraulics. The objective former was to integrate the nuclear data; advanced Monte Carlo codes; lattice codes; advanced deterministic diffusion and transport methods, core simulators; neutron kinetics methods and benchmark the platform while the latter platform considered the sub-channel codes; fuel behaviour TM codes; system TH codes; CFD codes and focused on pressurized thermal shock and critical heat flux modelling. Multi-physics, sensitivity and uncertainty analysis were also in the scope

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13 Called also Beyond Design Basis Accidents (BDBA) and in post-Fukushima definition of accidents as Design Extension Conditions (DEC) (see Safety of nuclear power plants: design. IAEA safety standards series no. SSR-2/1 (Rev. 1) IAEA 16-01014, ISBN 978–92–0–109315–8, Vienna, (2016).).

14 TRACE is NRC’s modernized thermal-hydraulics code designed to consolidate the capabilities of NRC’s 4 legacy safety codes - TRAC-P, TRAC-B, RELAP, and RAMONA. It is able to analyse large/small break LOCAs and system transients in both PWRs and BWRs. The capability also exists to model thermal hydraulic phenomena in both 1-D and 3-D space. It also includes the ability to model 3-D kinetics through integration with the Purdue Advanced Reactor Core Simulator (PARCS) code.
of the project and were further developed in the NURISP (2009-2012) project [42]. The third project NURESafe (2013-2015) [43] expanded to include multi-physics applications involving core physics; multiscale analysis of core thermal-hydraulics from direct numerical simulation (DNS) to sub-channel modelling; multiscale and multi-physics applications of thermal-hydraulics and education and training. The NURESIM platform makes use of open-source software SALOME to provide the dynamic 3D coupling of the codes simulating the different physics of the problem into a common multi-physic simulation scheme [43]. Each code provides an interface, and the coupling is done by calling the methods/services of the interface. Thus a standardized state-of-the-art code system is applied to support the safety analysis of LWRs. NURESAFE project confirmed the compatibility of NURESIM platform simulation schemes with CEA developed open-source URANIE\(^1\)\(^4\) platform for uncertainty and sensitivity analysis. URANIE aims at providing methods and algorithms about Uncertainty and Sensitivity, and Verification and Validation analyses in the same framework. NURESIM platform made use of URANIE for Uncertainty Quantification (UQ).

The US DOE Consortium for Advanced Simulation of Light Water Reactors (CASL) project [44] has developed and tested what amounts to a virtual nuclear reactor. The Virtual Environment for Reactor Applications (VERA) includes computational tools and supporting infrastructure that represent the cutting edge in light water reactor modeling and simulation, and can be used to solve neutronics, thermal-hydraulics, fuel performance, and coupled physics problems with advanced uncertainty quantification tools. VERA integrates physics components based on science-based models, state-of-the-art numerical methods, and modern computational science, and is verified and validated using data from operating PWRs [45], single-effect experiments, and integral tests [44]. CASL’s integrated, coupled solutions with VERA provide a more realistic representation of the reactor’s behaviour.

US DOE Nuclear Energy Advanced Modeling & Simulation (NEAMS) Program [46] is developing a simulation tool kit using leading-edge computational methods that will accelerate the development and deployment of nuclear power technologies that employ enhanced safety and security features, produce power more cost-effectively, and utilize natural resources more efficiently. The project foresees use of two integration platforms: VERA (see CASL project above) for LWRs (and water-based Small Modular Reactors) and Workbench for Advanced Reactors. The platforms utilize the system analysis codes like RELAP-7 for plant model. Core Analysis Neutronics might be pin resolved or Monte Carlo with complimentary kinetics and depletion, and cross-sections codes. While the low resolution core TH is achieved by sub-channel COBRA family code the CFD code is foreseen for high resolution core TH. Core structural mechanics is assessed either by an external code or in-house code Diablo. Fuel Analysis addresses the continuum, microstructure, component aging and chemistry each simulated by a dedicated code.

RELAP-7 (Reactor Excursion and Leak Analysis Program-7) is the nuclear reactor system safety analysis code currently under development at the INL as part of the Light Water Reactor Sustainability Program [47]. It is an evolution in the RELAP-series reactor systems safety analysis applications. The RELAP-7 code development is taking advantage of the progress made in the past three decades to achieve simultaneous advancement of physical models, numerical methods, coupling of software, multi-parallel computation, and software design. RELAP-7 uses the INL’s open source platform MOOSE (Multi-Physics Object-Oriented Simulation Environment) framework for efficiently and effectively solving computational engineering problems. MOOSE provides numerical integration methods and mesh management for parallel computation. Therefore RELAP-7 code developers have been able to focus more upon the physics and user interface capability. There are currently over 20 different MOOSE based applications ranging from 3-D transient neutron transport, detailed 3-D transient fuel performance analysis, to long-term material aging. Multiphysics and multi-dimensional analysis capabilities, such as radiation transport and fuel performance, can be obtained by coupling RELAP-7 and other MOOSE-based applications through MOOSE and by leveraging with capabilities developed by other DOE programs. Unlike the traditional system codes, all the physics in RELAP-7 can be solved simultaneously (i.e., fully coupled), resolving important dependencies, and significantly reducing spatial and temporal errors relative to traditional approaches.

\(^1\) based on the data analysis framework ROOT, (https://root.cern.ch/) an object-oriented and petaflopic computing system developed by CERN
4.4. Fusion codes DEVELOPMENT

Development of a new fusion accident code or system [48] - [51] didn’t prove successful. Thus Safety studies for fusion facilities are commonly conducted using codes originally developed for fission reactor accident analysis. Some of these codes have been modified, and have additional physical models to treat fusion-relevant phenomena. The most are widely applied for fusion accident analyses are fusion-adapted versions of MELCOR code [1] -[3], [16], [52] - [57]. The fusion-modified codes are validated against the limited available fusion experimental data or through benchmarking against validated code(s) or code version(s). Among other adapted codes is worth to mention ASTEC [58], ATHENA (adaptation of RELAP5/mod3) [54], CATHARE [59] codes and recently modified SIMMER code and [60]. Among the fission codes applied to fusion safety one may refer to RELAP-3D applications to fusion breeder blanket accident analyses [61], [62]. Recently INL incorporated tritium transport code TMAP capabilities into fusion adapted MELCOR version 1.8.6 [3].

Code development for fusion safety would have to address the mechanistic (phenomena based) modelling of

- Breeder/multiplier reactions with air, water and steam
- Plasma facing components maximum heat (and pressure) loads on first wall during plasma transients (plasma disruptions, plasma termination, pellets or gas insertion)
- Dust transport and retention
- Activation Corrosion products transport
- Irradiated Pebble beds properties under range of thermal/pressure/mechanical loads
- Magneto-hydrodynamic fluid dynamics
- Tritium transport and retention
- Hydrogen transport and retention, passive systems (hydrogen re-combiners) phenomena

Eventually fusion community will develop one day an integrated computer code with multi-fluid TH; mechanistic models listed above; structural analysis capability (RAFM failure criteria); 2- or 3D modelling capabilities where needed (in-vessel LOCA, breeder blanket pressurization, breeder manifolds, MHD effects, CFD coupling/models/features, etc.

5. CONCLUSIONS

The parallel between the fission and fusion safety approaches and accident analyses has been drawn in the topics of experimental programs, test matrixes and data bases; PIRT; and computer codes development. For each of the above topics a brief presentation of the fission historical development followed by an overview of published adaptations of methods and their applications to fusion safety is reported. Although to complete the overview one need also to address the computer codes verification and validation; computer codes assessments; conservative or best estimate methodology; and uncertainty estimation methods as it was planned initially but dropped because of limited volume of the paper this topics will be covered in a future publications. The needs of further work, needs to enhance the collection of fusion specific experimental data, use of PIRT, the verification and validation of computer codes by means of separate and integral effect tests and establishing benchmark problems as well the development of multi-physics, multi-fluids integrated code systems were discussed. These efforts should be aimed at developing a systematic safety demonstration defined by an integrated fusion safety assessment methodology.

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