Nuclear Heat Transfer and Passive Cooling

Volume 1: Introduction to the Technical Volumes and Case Studies

FRAZER-NASH CONSULTANCY

Volume 1 Introduction to the Technical Volumes and Case Studies

Volume 2 Convection, Radiation and Conjugate Heat Transfer

Volume 3 Natural Convection and Passive Cooling

Volume 4 Confidence and Uncertainty

Volume 5 Liquid Metal Thermal Hydraulics

Volume 6 Molten Salt Thermal Hydraulics

Study A Liquid Metal CFD Modelling of the TALL-3D Test Facility

Study B Fuel Assembly CFD and UQ for a Molten Salt Reactor

Study C Reactor Scale CFD for Decay Heat Removal in a Lead-cooled Fast Reactor

Study D System Code and CFD Analysis for a Light Water Small Modular Reactor

Authors:	Richard Underhill	Frazer-Nash Consultancy
	Tim Houghton	Frazer-Nash Consultancy
Contributors:	Adam Owens	Moltex Energy
	Aidan Wimshurst	Frazer-Nash Consultancy
	Akash Dhandhukia	Terrestrial Energy
	David Cuming	DBD International
	Hector Iacovides	The University of Manchester
	Hugo Winter	EDF Energy
	Janne Wallenius	LeadCold
	Milorad Dzodzo	Westinghouse Electric Company
	Paolo Ferroni	Westinghouse Electric Company
	Shuisheng He	The University of Sheffield
	Steven Moor	Rolls-Royce
	Tim Abram	U-Battery
Technical Volumes Lead:	Tim Houghton	Frazer-Nash Consultancy
Approver:	Brian Gribben	Frazer-Nash Consultancy

Document Number: FNC 60148/49308R

Issue and Date: Issue 1, December 2021

Legal Statement

Preface

Nuclear thermal hydraulics is the application of thermofluid mechanics within the nuclear industry. Thermal hydraulic analysis is an important tool in addressing the global challenge to reduce the cost of advanced nuclear technologies. An improved predictive capability and understanding supports the development, optimisation and safety substantiation of nuclear power plants.

This document is part of *Nuclear Heat Transfer and Passive Cooling: Technical Volumes and Case Studies*, a set of six technical volumes and four case studies providing information and guidance on aspects of nuclear thermal hydraulic analysis. This document set has been delivered by Frazer-Nash Consultancy, with support from a number of academic and industrial partners, as part of the UK Government *Nuclear Innovation Programme: Advanced Reactor Design*, funded by the Department for Business, Energy and Industrial Strategy (BEIS).

Each technical volume outlines the technical challenges, latest analysis methods and future direction for a specific area of nuclear thermal hydraulics. The case studies illustrate the use of a subset of these methods in representative nuclear industry examples. The document set is designed for technical users with some prior knowledge of thermofluid mechanics, who wish to know more about nuclear thermal hydraulics.

The work promotes a consistent methodology for thermal hydraulic analysis of single-phase heat transfer and passive cooling, to inform the link between academic research and end-user needs, and to provide a high-quality, peer-reviewed document set suitable for use across the nuclear industry.

The document set is not intended to be exhaustive or provide a set of standard engineering 'guidelines' and it is strongly recommended that nuclear thermal hydraulic analyses are undertaken by Suitably Qualified and Experienced Personnel.

The first edition of this document set has been authored by Frazer-Nash Consultancy, with the support of the individuals and organisations noted in each. Please acknowledge these documents in any work where they are used:

> Frazer-Nash Consultancy (2021) Nuclear Heat Transfer and Passive Cooling,
> Volume 1: Introduction to the Technical Volumes and Case Studies.

Contents

1 Background and Purpose 1
- 1.1 Nuclear Thermal Hydraulics 1
- 1.2 Purpose of Technical Volumes 1
- 1.3 Safety and Design Benefits 2
- 1.4 Passive Cooling Applications 3
- 1.5 Advanced Nuclear Technologies 4

2 The Nuclear Industry and Thermal Hydraulics 5
- 2.1 Motivation 5
  - 2.1.1 The Need for Nuclear Power 5
  - 2.1.2 The Value of Nuclear Thermal Hydraulics Analysis 7
- 2.2 Safety and Licensing Context 9
  - 2.2.1 IAEA Safety Standards 9
  - 2.2.2 Safety Culture 11
  - 2.2.3 Defence in Depth 12
  - 2.2.4 Safety Categorisation and Classification 12
  - 2.2.5 Graded Approach 13
  - 2.2.6 Licensing Context 18

3 The Technical Volumes and Case Studies 20
- 3.1 Technical Volumes 20
- 3.2 Case Studies 22
- 3.3 Assumed Knowledge 23

4 Nuclear Thermal Hydraulics Analysis 26
- 4.1 Overall Analytical Approach 26
  - 4.1.1 Analysis Tools 26
  - 4.1.2 Computational and Experimental Analysis 28
  - 4.1.3 Analysis through the Whole Project Life Cycle 29
- 4.2 Phenomena Identification and Ranking Table (PIRT) 30
- 4.3 Verification, Validation and Uncertainty Quantification (VVUQ) 33
  - 4.3.1 Verification 33
  - 4.3.2 Validation 34
  - 4.3.3 Uncertainty Quantification 35
- 4.4 System and Subchannel Analysis 36
  - 4.4.1 System Analysis 36
  - 4.4.2 Subchannel Analysis 39
- 4.5 Computational Fluid Dynamics (CFD) Analysis 42
  - 4.5.1 Planning a CFD Analysis 43
  - 4.5.2 Mesh Generation 44
  - 4.5.3 CFD Approaches 48

4.5.4 Codes . . . 54
4.6 Experimental Methods . . . 56
4.6.1 Planning and Working Together . . . 56
4.6.2 Basic Tests . . . 58
4.6.3 Separate Effect Tests (SETs) . . . 58
4.6.4 Integral Effect Tests (IETs) . . . 59
4.6.5 Scaling Considerations . . . 59
4.6.6 Facilities . . . 60

5 References 62

6 Nomenclature 67

7 Abbreviations 69

Appendix A Project Context 72
A.1 Thermal Hydraulics Model Development . . . 72
A.2 Project Consortium . . . 74

1 Background and Purpose

1.1 Nuclear Thermal Hydraulics

Nuclear Thermal Hydraulics (NTH) is the application of thermofluid mechanics to nuclear reactor systems. NTH is used to study engineering systems where energy from nuclear fuel is transferred by a coolant to a power generation turbine or to the environment by heat transfer, flow processes or phase change.

NTH shares modelling challenges and tools with many industries where thermofluid mechanics is important (e.g. aerospace, turbomachinery, chemical engineering and meteorology). However, NTH often addresses an unusually wide range of technical challenges simultaneously on a huge range of geometrical scales. D'Auria (2017) provides a more detailed discussion of this point in the context of water-cooled nuclear reactors.

Thermal hydraulic analysis is already a substantial part of justifying the safety of a Nuclear Power Plant (NPP) and used in the design of a wide range of nuclear systems and components to ensure that the plant is efficient, reliable, robust and can operate economically with minimal operational costs and shut-down time.

As with other industries, the appropriate development and acceptance of state-of-the-art modelling tools has the potential to enable a level of design and operational optimisation for future NPP beyond anything seen in the past, potentially delivering improved safety and economic benefits.

1.2 Purpose of Technical Volumes

The first edition of the *Nuclear Heat Transfer and Passive Cooling: Technical Volumes and Case Studies* was funded by the UK Department for Business, Energy and Industrial Strategy (BEIS) as part of a larger thermal hydraulics model development project. The overall project is described in Appendix A, together with details of the consortium that delivered the project.

The aim of this project has been to develop a consistent approach for thermal hydraulic analysis of single-phase heat transfer and passive cooling for use across the nuclear industry. This is intended to improve the design and performance of advanced nuclear technologies by understanding the route to 'good quality' thermal hydraulic analysis.

This has been achieved through the production of a high-quality peer-reviewed set of documents in a useful and accessible form that can be used by different end-users to understand:

- The benefit, role and limitations of thermal hydraulic analysis within the reactor design process and how analysis can be used to reduce uncertainty.

- Current and 'state-of-the-art' computational methods for thermal hydraulic analysis of passive cooling applications with a particular focus on Computational Fluid Dynamics (CFD).
- The availability and capability of different types of analysis and the selection of appropriate approaches (fidelity vs cost) for thermal hydraulic analysis within a civil nuclear context.

Due to the limited scope of this project, the volumes only consider single-phase heat transfer as this is applicable to all advanced nuclear technologies under normal operation and a range of fault scenarios. This means that not all thermal hydraulic phenomena relevant to passive cooling applications are addressed by this set of technical volumes, and the main exclusions are multiphase (gas/liquid) flow, boiling, flashing, evaporation, condensation, melting and solidification.

Current methodologies used for NTH analysis (Section 4.1) tend to vary based on the status of the design, purpose of the analysis and the expertise and familiarity of end-users with specific analysis tools. Within industry, thermal hydraulic design of reactors is usually undertaken by well-established system and subchannel codes. These codes use empirical correlations for flow and heat transfer that have been created and validated using experimental data, which are applicable within a defined range of operating conditions and, in some cases, system geometries.

In academia, thermofluid mechanics is a well-researched area within the nuclear industry and other sectors. This research is often focused on CFD analysis due to its finer (temporal and spatial) resolution compared to system codes. However, due to the high computational requirements of CFD analysis, CFD tools are generally used for component level design and require appropriate validation and justification for use within a nuclear safety case due to the uncertainties associated with aspects such as mesh resolution, turbulence models, initial and boundary conditions and applied numerical schemes.

The documents generated by the current project are intended to inform this link between academic research and end-user needs. They are predominantly focused on CFD methods, but also cover the use and application of system and subchannel codes. A more rounded understanding can be achieved by looking at the information provided in the technical volumes and case studies together.

1.3 Safety and Design Benefits

The safety of NPPs is clearly of paramount importance, and so safety and design analyses are often coupled (i.e. safety informed design). NTH analyses for safety are focused on ensuring and justifying that nuclear systems and components operate within the required margins under normal operation and fault scenarios. Nuclear safety is typically based on multiple levels of defence to prevent or limit the development of adverse conditions during fault scenarios. NTH analyses are often an essential part of every defence in depth protection level, and are used to assess the margin associated with key figures of merit (the difference between a calculated value and an acceptance criterion) under normal operation and fault scenarios.

For a thermal hydraulic analysis tool to be used to support a reactor design assessment or nuclear safety case, it must be assessed using a rigorous protocol that has been reviewed and accepted by the licensing authorities. This is likely to involve demonstrating adequacy and confidence in the models for calculating the scenarios of interest through detailed verification and validation, within the context of a graded approach that ensures proportionality for underpinning NTH analysis

(Downing *et al.*, 2018). The safety and licensing context around NTH analysis is discussed further in Section 2.2.

Thermal hydraulic analyses demonstrate that the plant (primary circuit, secondary circuit and ancillary systems) has been designed to operate within acceptable design or operational limits (such as component temperatures). Such analysis is likely to include simulating normal operation, start-up and shut-down, as well as postulated fault scenarios, which cover a broad range of phenomena occurring over a wide range of lengths and timescales. The value of NTH is considered further in Section 2.1.2, and example benefits of using NTH analysis include:

Improved performance: Thermal hydraulic analysis is a key factor within the reactor design process to optimise the commercial viability, improve the operational envelope, lead to better understanding and therefore more appropriate safety margins. For example, NTH analysis is used to size and design the primary and secondary circuits, maximise heat transfer from the core to the coolant, minimise pressure drop around the circuit and maximise heat exchange.

Improved reliability: Due to the long timescales and cost associated with start-up and shut-down and accessibility of components within the primary circuit, long-term, reliable operation is a key requirement for a well-designed and operated NPP. NTH analysis is used to improve reliability and identify potential showstoppers or design defects early. For example, minimising the potential for fouling/deposition, fretting (wear) due to fluid-structure interaction and weld failure due to thermal fatigue.

Reduced cost: The cost of a NPP is dependent on the size and complexity of the plant and its components, as well as the number of safety systems required. Thermal hydraulic analysis can be used to simplify the design (reducing capital cost), as well as develop and justify the use of fewer, simpler passive safety systems.

1.4 Passive Cooling Applications

Passive safety is a design approach intended to reduce or remove the need for active systems, or any intervention on the part of the operator or control systems, to bring and maintain a reactor to a safe shut-down state in the event of a particular scenario occurring. These design features typically take advantage of natural forces or phenomena such as gravity, buoyancy, pressure differences, conduction, thermal radiation or natural heat convection to accomplish safety functions without requiring an active power source (IAEA, 2009).

All current commercial NPPs make use of both active and passive systems. However, new reactor designs are making more extensive use of passive safety features because they are intended to achieve the same or higher reliability using fewer systems/components, thus reducing capital, Operation and Maintenance (O&M) costs associated with the installation and maintenance of mechanical, control and instrumentation support systems. For example, the NuScale reactor design[1] uses natural circulation of the reactor coolant flow around the primary circuit without the need for main coolant pumps under normal operation and loss of power conditions.

However, the nature of the flows, complex coupling between the flow and temperature field, and weak or unstable driving forces associated with passive safety systems (e.g. natural circulation)

[1] www.nuscalepower.com

can make justification of the plant's operation and safety performance more challenging. Therefore, demonstrating the reliability and performance of passive safety systems and understanding the requirements and validation of thermal hydraulic analysis tools used to simulate these phenomena is essential for new reactor designs with passive safety features. Examples of passive cooling systems include core cooling, containment cooling, safety injection and decay heat removal.

1.5 Advanced Nuclear Technologies

Advanced Nuclear Technologies (ANTs) are designed to maximise the use of off-site factory fabrication of modules to significantly reduce the cost of electricity generation, and is the term used by the United Kingdom (UK) Government to describe two groups of reactors (BEIS, 2019):

Generation III water-cooled Small Modular Reactors (SMRs): These are similar to existing light water reactors, but on a smaller scale and aim to have enhanced passive safety systems. Typically, concept designs are mature and close to demonstration or commercialisation. The current SMRs under development are summarised in IAEA (2020a) and listed in the International Atomic Energy Agency (IAEA) Advanced Reactor Information System (ARIS) database[2].

Next generation Advanced Modular Reactors (AMRs): These reactors use novel cooling systems or fuels to generate electricity and offer new functionality (such as industrial process heat) and aim for further improved passive safety and a step change reduction in costs. Typically, these reactors are at early design stages. AMRs include fusion technologies and the six nuclear energy systems selected for further development by the Generation IV International Forum (GIF)[3] and described in GIF (2018):

- Gas-cooled Fast Reactor (GFR)
- Very High Temperature Reactor (VHTR)
- Lead-cooled Fast Reactor (LFR)
- Sodium-cooled Fast Reactor (SFR)
- Molten Salt Reactor (MSR)
- SuperCritical Water Reactor (SCWR)

The coolants associated with advanced nuclear technologies that are addressed within these technical volumes and case studies, include water, helium, lead, sodium and molten salt.

[2] aris.iaea.org
[3] www.gen-4.org/gif

2 The Nuclear Industry and Thermal Hydraulics

2.1 Motivation

This section outlines the need for nuclear power to support a decarbonised power system and the benefit and value of thermal hydraulics analysis within it. This highlights the need for innovation within the nuclear industry and motivation behind this project.

2.1.1 The Need for Nuclear Power

In June 2019, the UK became the first major economy in the world to pass laws to reduce emissions to net zero by 2050, as recommended by the UK Committee for Climate Change (CCC, 2019). Nuclear power has the potential to decarbonise the whole energy system and directly supports the United Nations Sustainable Development Goal (SDG) of affordable and clean energy (SDG7)[1] and supports all goals to ultimately achieve high living standards, good health, a clean environment and sustainable economy for all (IAEA, 2017).

The UK generated 325 TWh of electricity in 2019 (56 TWh from nuclear) with over 54 % from nuclear and renewables (BEIS, 2020); see Figure 2.1. Fully decarbonising electricity supply can be achieved through increasing the share of low-carbon power generation from around 50 % today to around 95 % in 2050. This will need to be done in the context of meeting significant additional demands from: the electrification of transport, industry and agriculture; the heating of buildings; and potentially production of hydrogen and carbon capture and storage. Therefore, low-carbon electricity generation could need to be as much as four times today's levels by 2050 (NIRAB, 2020).

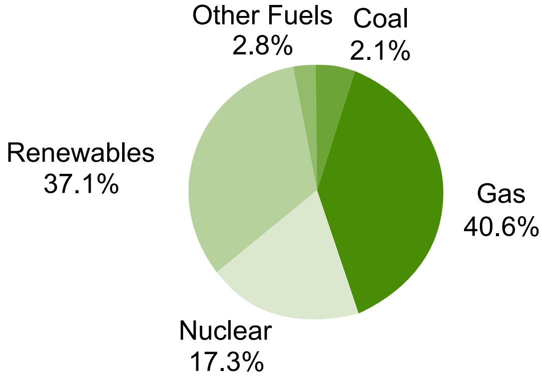


Figure 2.1: Shares of UK electricity generation in 2019, by fuel (BEIS, 2020).

[1] sdgs.un.org/goals

A significant increase in firm low-carbon power is required as part of a reliable, low-cost, decarbonised energy system (CCC, 2019). Firm low-carbon resources include nuclear, natural gas with carbon capture and hydro-electric power, which provide a steady, reliable supply compared to variable renewables sources, like wind and solar. The UK is not geographically well suited to support a significantly increased hydro-electric contribution, but based on the uranium resources around the world and high power density associated with it, nuclear power could provide the UK with a long-term source of firm, low-carbon power (MacKay, 2010).

Nuclear power supplied 10 % of global electricity generation in 2018 (IEA, 2019). However, nuclear fleets around the world are ageing and the addition of new capacity has slowed. Therefore, it is important to extend the life of existing plants and accelerate the development and build of new NPP in order to transition towards a low-carbon energy system. A number of advanced reactor technologies have the additional benefit of operating at high temperatures (above 700 °C) for:

Industrial process heat: Applications include desalination, synthetic and unconventional oil production, oil refining and biomass-based ethanol production, as well as heat networks for industry and domestic homes.

Hydrogen production: Hydrogen can be produced via electrolysis of water using off-peak capacity, high-temperature electrolysis of steam using heat and electricity, and potentially high-temperature thermochemical production using nuclear heat (IAEA, 2013). Hydrogen is already significantly used in industrial processes and large increases in demand are expected by 2050 associated with the need to decarbonise transport and heat generation[2].

However, for nuclear power to make a credible and substantial contribution to the UK decarbonisation strategy, the cost of designing, building and decommissioning nuclear plants needs to be significantly reduced.

Cost can be measured by the Levelised Cost of Energy (LCOE), which is the total cost to build, operate and decommission a NPP divided by the total energy output. This allows a comparison of different methods of electricity generation on a consistent basis, and is effectively the average minimum price (£/MWh) at which electricity must be sold in order to break-even over the lifetime of the plant. Although this calculation does not take into account the potential additional benefits of industrial process heat and hydrogen production.

$$\text{LCOE} = \frac{\text{Design} + \text{Finance} + \text{Licensing} + \text{Construct} + \text{O\&M} + \text{Fuel} + \text{Decommission}}{\text{Thermal power} \times \text{Operating life} \times \text{Thermal efficiency} \times \text{Capacity factor}}$$

Reducing the LCOE for nuclear power is a key commitment in the BEIS (2018) Nuclear Sector Deal, specifically reducing the cost of new build projects by 30 % and achieving savings of 20 % in the cost of decommissioning by 2030. Short term benefits for existing technologies are being realised through fleet deployment benefits, alternative financing mechanisms and reducing the cost of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA) process (BEIS, 2019). Beyond 2030, ANTs provide opportunities for further cost reductions through disruptive advances and innovation in reactor design, construction processes and alternative fuels.

[2] For example, BEIS are currently funding projects to investigate the use of hydrogen in the gas network for domestic and industrial heating.

2.1.2 The Value of Nuclear Thermal Hydraulics Analysis

NTH analysis is a fundamental part of the design of all NPPs, as it is concerned with the design of the systems required to transport thermal energy from the reactor core (fuel) and to transform this thermal energy into process heat or electricity. As such, it plays an important role in determining not only the plants' safety case and operational characteristics, but also its economics through capital cost and efficiency in energy conversion.

This energy conversion process involves multiple connected systems, including the reactor core, primary circuit, heat exchangers, turbines and associated safety systems, as well as pipes, valves, pumps and vessels. It also requires significant interaction with other disciplines, such as reactor physics, chemistry, metallurgy, control and stress analysis throughout the design process.

The most important role of thermal hydraulics from a safety perspective is to ensure that the thermal energy can be removed from the core to the ultimate heat sink under all potential scenarios. This is particularly important in nuclear reactors, compared to other power plants, because ongoing nuclear decay reactions mean that significant thermal energy is released in the core even after the fission chain reaction has been stopped.

Thermal hydraulic analysis needs to be undertaken at a system and component level to optimise the design and maximise plant performance, reliability and safe operation. However, it is important to ensure that the analysis tools are appropriately verified and validated (Section 4.3) for the application to provide sufficient confidence and safety margin in the key predicted performance parameters or Figures of Merit (FOMs), such as peak fuel cladding temperature, fuel centreline temperature, minimum reactor vessel coolant level and maximum containment atmosphere pressure.

As discussed in Section 2.1.1, cost reduction is a key driver for innovation in the development of ANTs in order to ensure that they are competitive and affordable as part of the UK's low-carbon energy system. Improvements in NTH analysis methods can support this need to reduce cost through digital design, improved performance, passive safety and modular construction. The benefits of NTH analysis in these areas are highlighted in Figure 2.2 and described below.

Digital design: The design of a nuclear power plant accounts for about 10 % of the total capital costs (OECD NEA, 2000). Developing digital tools and fundamental scientific understanding will enable advanced nuclear technologies to be designed and built in an accelerated and cost effective way. This will reduce the cost of design and increase the cost effectiveness of testing by reducing the number of tests needed and/or by streamlining the design of testing campaigns. In addition, digital twins are being increasingly used in other industries to improve plant monitoring and substantially reduce plant maintenance costs.

Thermal hydraulic analysis is an essential part of an integrated digital design and assessment process, which enables the reactor design to be optimised at a system and component level in an efficient and cost effective way. This will require improvements in the adequacy, automation and usability of system and CFD codes to accelerate the design of advanced nuclear technologies and couple with plant geometry, neutronics, Control & Instrumentation (C&I) and structural integrity tools.

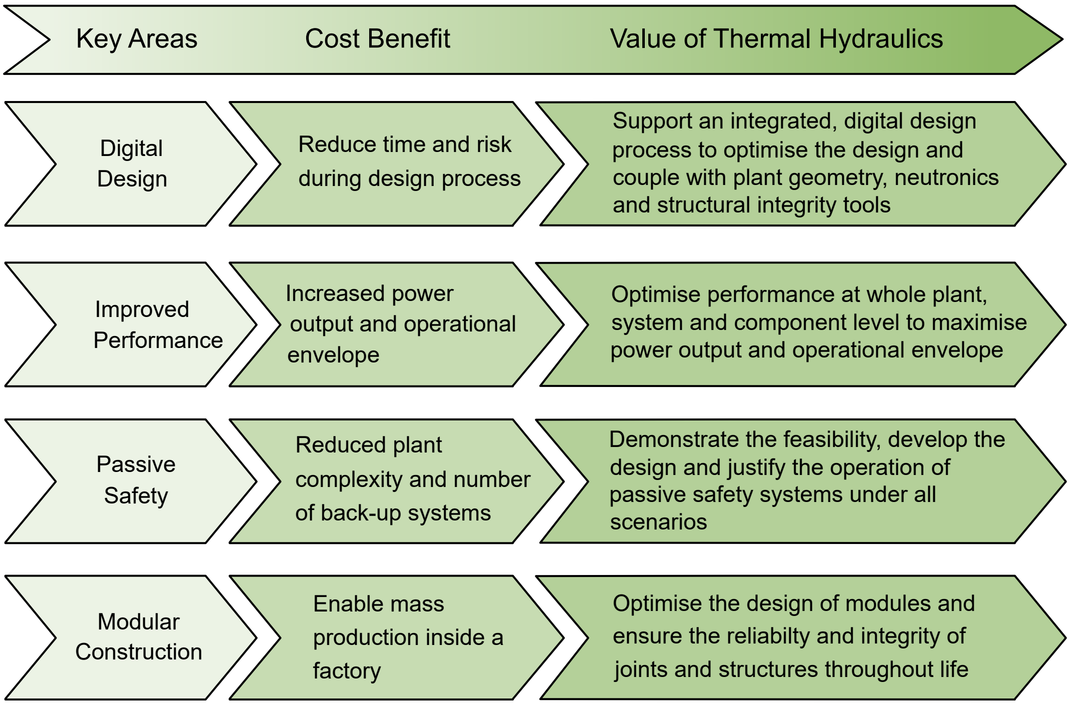


Figure 2.2: Cost benefit and value of thermal hydraulics.

Improved performance: The efficiency of a NPP depends on a number of factors that interplay to optimise the commercial viability, including fuel burnup (utilisation), thermal efficiency and thermal losses. Thermal hydraulic analysis is a key factor within the reactor design process to maximise power output, improve the operational envelope, lead to better understanding and therefore more appropriate safety margins.

Thermal hydraulic analysis is used to improve the performance of a NPP by maximising primary circuit temperature and heat transfer effectiveness in the core, minimising primary circuit pressure drop and maximising heat exchange. By reducing and quantifying the uncertainty in system and component level predictions for advanced nuclear technologies, an increase in the maximum primary coolant temperature can be justified in order to increase the power output.

Passive safety: ANTs have more passive safety features and systems than conventional plants and so are mechanically simpler. Therefore, fewer pumps, valves, auxiliary components and active back-up systems are required in order to bring the reactor to, and maintain it in, a safe shutdown state (e.g. no back-up power supply is required to respond to a loss of off-site power). This reduces the overall size and complexity of the nuclear site (and hence capital cost), O&M costs and number of on-site personnel. They also have the potential to be deployed in more diverse locations which expands the potential siting options.

Since passive safety systems use natural forces or phenomena such as gravity, buoyancy, pressure differences, conduction or natural heat convection, thermal hydraulic analysis is fundamental in order to demonstrate the feasibility of the system and justify that the safety function is achieved under all scenarios. The increased reliance on passive safety systems and the weak driving forces associated with them highlight the need for increased understanding and improved accuracy for simulating passive cooling within a NPP.

Modular construction: Advanced manufacturing techniques and modular construction enable standardised mass production inside a factory, which ensures consistency in quality away from the wind, rain and dirt associated with a construction site. This will drive down unit costs compared to bespoke, one-off designs. A significant advantage of modular design is therefore a large reduction in construction risk, as connecting together modules on a pre-made foundation eliminates a lot of uncertainty in the construction process. This means that small modular reactors could be built in a shorter time than larger reactors, greatly reducing cost.

Thermal hydraulic analysis is essential for sizing components, and so supports the modularisation decision process and the breakdown of the plant/systems into individual modules. In addition, NTH analysis ensures the reliability and integrity of joints between modules through life by understanding and minimising the potential for thermal fatigue, flow induced vibration and flow accelerated corrosion. By improving the adequacy and confidence in thermal hydraulic analysis of nuclear components under all scenarios, the design of a module can be optimised to increase reliability and minimise the likelihood of failures.

2.2 Safety and Licensing Context

Safety is of paramount importance to the future of the nuclear industry. This section provides an introduction to the safety and licensing context relevant to NTH, which may assist engineers, particularly those from non-nuclear industries.

Further information can be found in the references listed, the IAEA website or through the Safety Case Toolkit that has been developed by Frazer-Nash Consultancy as part of the BEIS funded Safety & Security Nuclear Innovation Programme (NIP)[3]. This provides interactive web-based guidance to those seeking to gain familiarity with modern nuclear safety cases in a UK context.

NTH analysis is often used to provide safety justification for a component or system design and support a nuclear safety case. This section provides a high level summary of the fundamental aspects of nuclear safety and how it impacts/influences a NTH analysis.

2.2.1 IAEA Safety Standards

The IAEA is the international forum for nuclear co-operation, working for the safe, secure and peaceful use of nuclear science and technology. As such, the IAEA publishes a range of documents, including international safety standards, technical guides, conference proceedings and scientific reports. The IAEA safety standards[4] are arranged in three tiers (Figure 2.3). A useful glossary of safety-related terms is also available in IAEA (2019b).

Safety fundamentals: This document (IAEA, 2006) establishes the fundamental safety objective for all nuclear facilities (to protect people and the environment from the harmful effects of ionizing radiation) and ten fundamental safety principles, which are summarised below:

1. Responsibility for safety: The prime responsibility should rest with the person or organisation responsible for nuclear facilities and activities over their whole life.

[3] innovationfornuclear.co.uk/toolkit/safety-case-toolkit.html
[4] www.iaea.org/resources/safety-standards

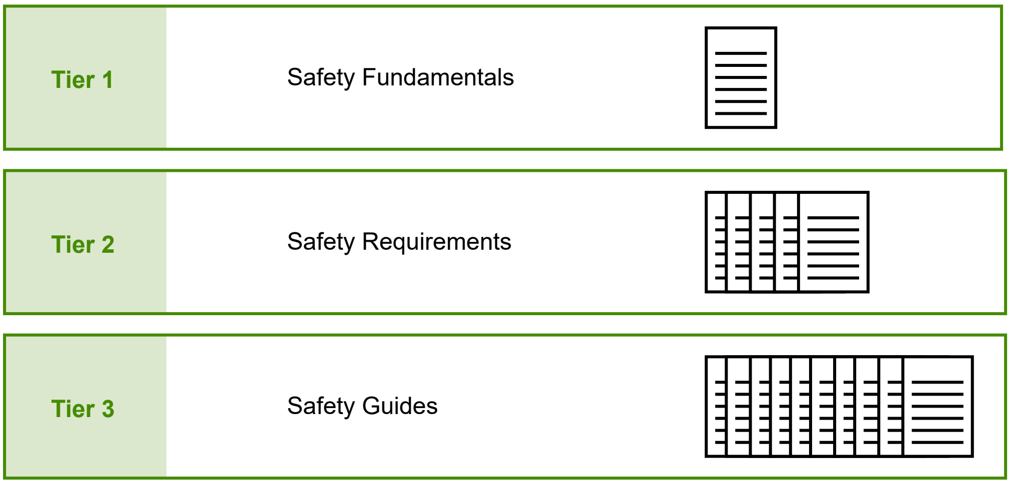


Figure 2.3: The three tiers of IAEA safety standards.

2. Role of government: Regulating safety and licensing nuclear facilities should be a national responsibility discharged through an independent regulatory body.
3. Effective leadership and management for safety: Should be established and sustained in organisations as part of a strong safety culture.
4. Justifying facilities and activities: The benefits of operation should outweigh the risks.
5. Optimising protection: The highest level of safety that can reasonably be achieved should be provided (i.e. As Low As Reasonably Achievable (ALARA) principle).
6. Limiting risks to individuals: No individual should bear an unacceptable risk of harm.
7. Protecting present and future generations: Applicable to people and the environment.
8. Preventing accidents: All practical efforts should be made to prevent and mitigate accidents (primarily through defence in depth).
9. Emergency preparedness and response: Arrangements should be made for responding to nuclear or radiation incidents.
10. Protective actions to reduce existing or unregulated radiation risks: These risks should be justified and optimised.

Safety requirements: These establish general and specific safety requirements that shall be met to ensure the protection of people and the environment. There are a number of these documents, two of which are particularly relevant to the design (IAEA, 2016c) and commissioning and operation (IAEA, 2016b) of NPP. These documents also explain the structure of IAEA safety standards in more detail.

Safety guides: This large suite of documents provides recommendations that should be met and guidance on how to comply with safety requirements. For example, IAEA (2020b) provides guidance on the design of the reactor coolant system.

Many of the ten safety principles are important for NTH analysis tasks, especially aspects relating to safety culture, defence in depth, safety classification and graded approach. These aspects are discussed further below, together with a high level introduction to the regulatory regime for a few individual nations.

2.2.2 Safety Culture

A strong safety culture starts with the framework/processes that an organisation puts in place to manage safety and is essential for everyone involved in nuclear related activities (detailed discussion is presented in IAEA, 1991 and WANO, 2013). Considering NTH analysis, this individual and collective commitment to safety encourages:

- Individuals to take personal and professional responsibility for the quality and accuracy of their work.
- A questioning and learning attitude, whereby computational and experimental work is considered, reviewed and challenged to build confidence in the results, and good practice is recognised and promoted. Where comments are made about work, these are considered on their merit and opportunities to improve future work are taken when reasonable to do so.
- Learning from experience, whereby it is recognised that one of the most valuable sources of knowledge is people with prior experience, and this is learned from to ensure computational or experimental work is performed most effectively and does not have to be repeated.
- Applying a graded approach, understanding the risk context around computational and experimental work and proceeding in an appropriately rigorous and prudent manner (Section 2.2.5).
- Appropriate communication, whereby the key stakeholders relevant to the work are identified and communicated with clearly, to avoid misunderstandings, lack of alignment or unwelcome surprises.

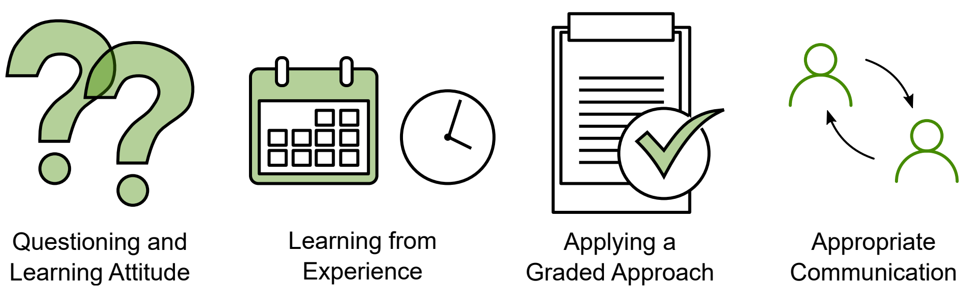


Figure 2.4: Aspects of nuclear safety culture.

Since NTH analysis methods and tools are constantly evolving and developing, it is important that users are aware of the importance and impact of current industry best practice, new software and standard developments. This can be achieved through continuous professional development, competency frameworks, Suitably Qualified and Experienced Personnel (SQEP) registers and knowledge capture within organisations.

2.2.3 Defence in Depth

The primary means of preventing and mitigating the consequences of accidents is to provide successive independent levels of protection (or barriers) that would have to fail before harm could be caused to people or the environment. Since these levels of protection should be independent, no single failure or latent condition should lead to harmful effects. This creates 'defence in depth' (IAEA, 1996, 1999, 2005) within the design and operation of a NPP that ensure the three main safety functions (controlling criticality, providing cooling and maintaining containment).

The value of using multiple independent and redundant levels of protection can be illustrated using a 'Swiss cheese model' (Figure 2.5), a widely used concept in nuclear, healthcare, aerospace and other high-assurance industries (Reason, 1997). An example of the application of defence in depth to the nuclear industry is:

1. Preventing abnormal operation and failures of any Structure, System and Component (SSC) important for safety, to maintain normal plant operation (e.g. through conservative design and high quality of construction and operation).
2. Detecting and controlling abnormal operation, to prevent anticipated operational occurrences escalating to accident conditions (e.g. using control and protection systems).
3. Controlling design basis accidents, to return the plant to a safe state, prevent core damage and avoid releases of radioactive material that could cause harm (e.g. using plant safety features and procedures).
4. Controlling beyond design basis accidents, to mitigate the consequences of accidents that could result from a failure of the third level of defence in depth (e.g. using plant safety features and procedures to prevent accident progression and avoid or minimise off-site release of radioactive material).
5. Mitigating the radiological consequences of significant releases of radioactive material (e.g. using on-site and off-site emergency response).

NTH analyses can be an essential part of every defence in depth protection level, and are often used to assess margins (the difference between a calculated value and an acceptance criterion) under normal operation and fault scenarios.

2.2.4 Safety Categorisation and Classification

It is important that the safety significance of SSCs within a NPP are clearly understood. This is achieved through categorisation and classification (IAEA, 2016c), a systematic, top down process that considers the plant design, its safety analysis and the main safety functions. This analysis generally uses deterministic methodologies (IAEA, 2019a), complemented by Probabilistic Safety Analysis (PSA) and engineering judgement as appropriate.

The safety functions required in various plant states are identified and categorised based on their safety significance, taking into account the consequences of failure, frequency of initiating events and significance of the function in achieving a controlled/non-hazardous or safe/shutdown reactor steady-state (IAEA, 2014, IAEA, 2016a, and BSI, 2010). Safety functions are typically categorised as Category 1(A), 2(B), 3(C) or not categorised (Table 1 in IAEA, 2014). The safety features in

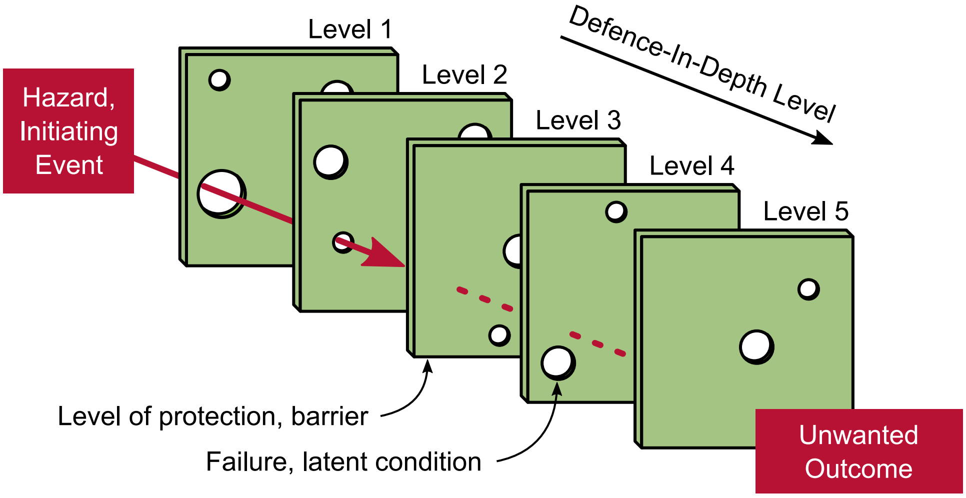


Figure 2.5: Providing defence in depth using multiple independent levels of protection (note that the levels of protection are drawn so they are not the same as each other).

the plant (and their associated SSC) that deliver these categorised functions are then classified accordingly, typically into Class 1, 2, 3 or not classified.

Categorisation and classification (or 'cat and class') enables resources to be focused on the SSCs most important to nuclear safety, and is therefore a key part of ensuring that SSCs are designed, manufactured, installed, commissioned and operated to provide appropriate reliability and integrity. As such, the safety category and classification strongly impacts the level of justification required for a thermal hydraulic analysis to support a safety case, in line with a graded approach (Section 2.2.5). For the lowest classified SSCs, a normal industrial level of justification may be appropriate, but for higher safety classifications the justification work required may be substantially more onerous.

2.2.5 Graded Approach

In assessing risks, it is normal to use a graded approach. This broadly means that the stringency of the control measures applied should be commensurate (as far as practicable) with the level of risk associated with a loss of control (IAEA, 2019b). From the perspective of NTH work, the level of justification that a given control measure is appropriate should therefore also be commensurate (as far as practicable) with this level of risk. Deciding what is appropriate will rely on the judgement of skilled people; this section discusses some of the aspects that should be considered.

2.2.5.1 Risk Context

The risk context or background for NTH work should be understood by the engineers involved, to enable an appropriate level of justification to be developed (or the work to be 'risk informed'). The proportionate assessment of risks is a discipline in its own right, particularly for nuclear safety, where it is a key aspect of safety case work. Therefore, understanding the risk context for computational or experimental assessments (and agreeing an appropriate way forward) is likely to involve communication with other stakeholders who may have no background in NTH. Aspects considered are likely to include:

Nature of the risk: The control of risks to health, safety, security, the environment and the economic case for the plant might all be supported by NTH analysis. Typical approaches and stakeholder and regulatory expectations may well differ between different kinds of risk. For example, the regulatory expectations for justifying a Category 1 safety function are likely to be higher than for justifying an aspect of plant economics, but this difference may well be smaller for internal stakeholders.

Level of risk: This is likely to be determined by considering both the probability (or frequency of initiating events) and the consequences associated with a loss of control. The detailed arrangements for assessing risk are likely to vary between organisations. For risks to nuclear safety, it is likely that the safety categorisation of any relevant safety functions (Section 2.2.4) will be important, but this may not be the only consideration. Additional relevant considerations may include: the safety or other classifications of specific equipment (higher safety class would indicate higher safety risks), the complexity and novelty associated with the SSC used to provide relevant safety features (more complex or novel plant designs would indicate higher safety risks), the characteristics and stage in the lifetime of the plant, risks to operators or the environment etc.

Expected margins: In many cases engineers will have some understanding of how large the margins for a given situation are likely to be (i.e. for safety, the margin between some safety limit and the performance the plant is expected to provide). Where these are considered small, this would indicate a higher risk that the plant performance might be inadequate to meet the relevant safety requirements.

Cliff edge effects: In some cases, a small change in a parameter or in the behaviour of an SSC might lead to severely abnormal operation or a severe increase in the consequences associated with a loss of control. Similarly, a severe scenario is sometimes identified which is just above some threshold (or 'cut-off frequency'). In situations like these, the level of justification might be increased to recognise these potentially severe consequences.

Organisations often have systems for categorising aspects of safety cases based on the associated risks. The process used to develop these categorisations may involve scoring systems, but these are normally used to provide additional context; the judgement of skilled engineers on the level of risk remains important.

2.2.5.2 Assessment Risks

The risks associated with the computational or experimental assessments should be considered in the light of the risk context (Section 2.2.5.1). Whatever process is followed, it is usually appropriate to **consider what are the worst consequences and likelihood of the assessment being inadequate**. As for risk context, the judgement of skilled engineers is important in considering the level of risk associated with an assessment, and the following aspects are likely to be considered:

Complexity: Where the scenario being assessed (analysis approach to problem) or the assessment work itself (execution of analysis) is very complex, there is likely to be an increased risk that the results of the assessment work are inadequate. A low risk example might be performing a simple hand calculation to calculate the time taken for a spent fuel pool to heat up to a certain temperature following a loss of a cooling, considering only thermal mass and neglecting heat losses from the pool. A high risk example might be understanding the impact of a Loss-Of-Coolant Accident (LOCA) caused by a large break in a Pressurised Water Reactor (PWR) primary circuit, considering the cooling of the core and pressurisation of the containment building, for which a large range of phenomena would need to be considered, probably using a number of different assessment techniques and tools.

Novelty: Where the assessment approach is novel or the application of the approach to the scenario of interest is novel, there is likely to be an increased risk that the results of the assessment work are inadequate. A low risk example might be using a system code to study the core cooling resulting from safety injection into a PWR primary circuit, using an approach that has been thoroughly tested through repeated use in similar situations previously. A high risk example might be using a new analysis method from a research environment which has never been used for the situation of interest before.

Uncertainty and/or conservatism: Where the uncertainty associated with the assessment approach is expected to be high or difficult to assess and/or the degree of conservatism is expected to be low or difficult to assess, there is likely to be an increased risk that the results of the assessment work are inadequate. A low risk example might be the pool heating example above, for which conservative assumptions on key parameters such as fuel heat output would be likely to be used. A high risk example might be a complex heat transfer assessment, for which using conservative assumptions for all the inputs might lead to unrealistic or misleading results, and hence a more detailed approach to managing uncertainty and conservatism might be needed (discussed further in Section 2.2.5.3).

Expected margins: As noted in Section 2.2.5.1, where margins are small compared to the uncertainty or conservatism in the analysis, there is likely to be an increased risk that the results of the assessment work are inadequate, and a higher level of justification may be appropriate.

In understanding the risks associated with an assessment, it is important to consider whether the assessment is the only route available to justify the design of the plant. This is likely to be the case for analysis of fault scenarios, since the performance of a reactor under such a scenario is often not possible to test. For example, the performance of a cooling system may be tested during commissioning, but the performance of a water deluge fire suppression system may not be because it might flood the plant, and this would likely feed into considerations of the level of risk associated with modelling system performance.

However, if the performance of a given system or component (e.g. a pump) can be satisfactorily tested during commissioning, the consequences of any analysis being inadequate (e.g. the pump being too small) are likely to be that the component must be replaced at a late stage in the programme (or system modified). This could have serious economic implications, but not necessarily nuclear safety implications (since there may be high confidence that the commissioning test will detect the performance shortfall). Therefore, in this case the measure of risk of underperformance is likely to be economic, with mitigations planned accordingly (e.g. by considering the supplier track record, the similarity to previously supplied pumps and suitable factory acceptance tests).

2.2.5.3 Conservatism, Uncertainty and Margins

These aspects introduced in Section 2.2.5.2 are often the more complex aspects to consider. When considering conservatism, it is helpful to understand the terms below, which are often used to define the kind of assessment being considered:

Best estimate: The assumptions used in the assessment (e.g. inputs, correlations, etc) are chosen to predict real plant behaviour as accurately as is reasonably practicable. This approach is generally used for analysis of normal operation and analysis of scenarios in support of a probabilistic safety case (e.g. PSA). This approach may also be used for analysis of low probability scenarios (such as analysis of severe accidents), for which it may not be appropriate or reasonably practicable to use a conservative approach.

Conservative: The assumptions (inputs, boundary conditions, correlations, etc) used in the assessment are chosen to be *appropriately conservative* (i.e. to bound the variation in inputs to some high confidence level). This bounding approach is generally used for analysis of scenarios within the design basis in support of a deterministic safety case, to provide a high level of confidence in safety margins.

Best Estimate Plus Uncertainty (BEPU): A best estimate assessment is performed and the uncertainty in the conclusions is assessed and accounted for, to show that the performance of the plant is appropriate when these uncertainties are taken into account. This approach may be used to improve plant performance or better justify the analysis where the margin is small or a conservative assessment is unacceptable.

For complex transients, conservative models and individually conservative input data can result in a failure to understand the plant performance and potentially produce non-conservative plant results. So, BEPU is necessary to prevent being misled and not just justify increased margin. BEPU type approaches should therefore be an acceptable/necessary approach to a deterministic demonstration. However, the amount of effort required to quantify uncertainty appropriately may be significant, in line with a graded approach (considered further in Section 4.2 and Volume 4, Confidence and Uncertainty).

Overly conservative: An overly conservative approach may enable high levels of confidence to be developed in analysis of complex plant in a cost effective manner. However, the combination of multiple conservative assumptions in assessments can cause unrealistic or misleading conclusions (which could potentially cause latent conditions or actions to be taken that are detrimental to nuclear safety).

These situations are illustrated in Figure 2.6. The safety limit (or safety criterion) in this illustration is a maximum value of some parameter or variable (e.g. a temperature that must be respected, or a time by which an action must be taken) and is shown in red[5]. For the purposes of this illustration, the safety limit is displayed as the same value in all cases. The variation in assessed plant performance is shown schematically as a normal distribution, but the real distribution may not be normal and the distribution itself may not be known when a best estimate or conservative approach is used.

For a conservative (or 'bounding') approach, using appropriately conservative assumptions is likely to lead to a conservative assessment, giving confidence in the margins presented. For a BEPU ap-

[5] This safety criteria could also be determined conservatively, overly conservatively etc.

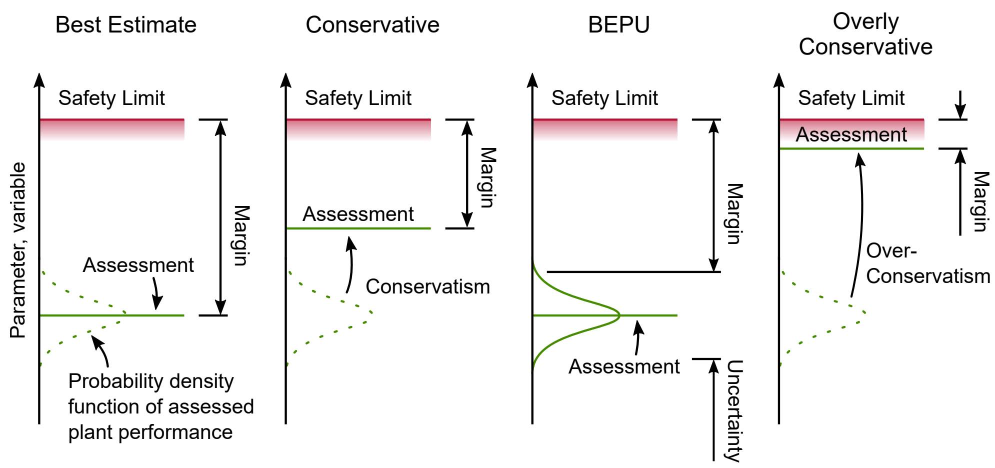


Figure 2.6: Illustration of margins and uncertainty for different types of assessment.

proach, it may be possible to better justify the analysis where the margin is small or improve plant performance while maintaining an acceptable margin, but the variation in assessed plant performance must be understood using uncertainty quantification. In an overly conservative approach the assessment is misleading, suggesting margins are far smaller than they really are.

In general, the starting point for NTH assessments is likely to be a conservative assessment for work supporting a deterministic safety case, although care should be taken to avoid over-conservatism. However, improvements in NTH analysis methods will increase the use and benefit of BEPU to improve plant performance and better quantify the margin and uncertainty as part of a nuclear safety assessment. It is worth noting that in the case of multiple parameters (phenomena) where all of them have the same importance the evaluation of uncertainty can be more complex, as well as the decision of what combination of them is the most conservative assumption.

2.2.5.4 Level of Justification

The level of justification required for an assessment within a graded approach is likely to depend on the risk context and the risk that the assessment is inadequate, considering aspects such as novelty, complexity, conservatism, uncertainty and margins (US NRC, 2005). The level of justification needed extends to all aspects of the analysis, such as:

- The amount of detail used to conduct the Phenomena Identification and Ranking Table (PIRT) (Section 4.2), scaling analysis (Section 4.6.5) and Verification, Validation and Uncertainty Quantification (VVUQ) work (Section 4.3), including aspects such as the justification of the steps taken, amount of validation work and number of sensitivity studies. The appropriate level of uncertainty quantification and tools available are discussed further in Volume 4.
- The resolution or amount of detail considered in the assessments or sensitivity studies themselves (Section 4.5.3).
- The level and rigour of quality assurance carried out on the work. Much more detailed checking would need to be documented where a high level of justification is needed. This is sometimes referred to as ‘a graded approach to quality assurance’.
- The nature and experience of the team used to perform the work (more skilled people are

likely to be needed where a high level of justification is needed). Further, using less skilled users could increase the uncertainty associated with the analysis through ‘user effects’.

- The detail, depth and volume of documentation produced to record the work.

In practice, it is likely to be challenging to justify a high category safety function, and require detailed validation evidence. Indeed, this justification may not be possible if the margins between predicted performance and safety limits are small (or cannot be shown to be sufficiently large) in relation to the uncertainty associated with the analysis. The skill and judgement of engineers in understanding what level of justification is appropriate for the given risk context and what this means in terms of the various assessment options available for a specific task is therefore extremely valuable.

2.2.6 Licensing Context

Individual nations are responsible for sustaining an independent regulatory body and regulatory regime for licensing nuclear facilities and activities. In IAEA member states, the regulatory regime is influenced and guided by the IAEA as well as other bodies. The IAEA assesses the effectiveness of individual regulatory bodies through its Integrated Regulatory Review Service (IRRS) programme, which enables an objective comparison of national nuclear regulation against IAEA guidelines.

This means that the regulatory regimes vary between countries, so while there may be many similarities, the design, safety and security justifications used to address the regulatory requirements of one country are unlikely to be identical to another. For example, the high-level regulatory principles of some countries (the UK, United States and Canada) are discussed below. Other countries with less history of having a nuclear industry may use regulatory arrangements similar to the countries with more established regimes or based on the IAEA approach.

United Kingdom: The ONR[6] approach to enforcement is governed by the principles of proportionality, consistency, targeting, transparency, and accountability (ONR, 2019a). The technical principles which ONR uses to judge a Licensee’s safety case are outlined in the ONR’s Safety Assessment Principles (SAPs) for nuclear facilities (ONR, 2014). These principles are developed further in a series of Technical Assessment Guides (TAGs) for assessors. The UK regulatory regime for safety is largely goal-setting (rather than prescriptive), which means that the ONR does not direct a Licensee to use particular analysis codes; it is up to the Licensee to present accompanying information to underwrite the appropriateness of the analysis techniques used. ONR permissions activities rather than approving safety cases.

This is based on the principle that risks should be reduced As Low As Reasonably Practicable (ALARP). The term ALARP arises from general UK health and safety legislation, and demonstrating that risks are ALARP involves evaluating the risks and considering whether it would be reasonably practicable to implement further safety measures[7]. In many areas, this will not be done through an explicit comparison of costs and benefits, but rather by applying established Relevant Good Practice (RGP) and standards. The ONR has recently updated its GDA guidance to enhance the efficiency and flexibility of the process, particularly for advanced nuclear technologies (ONR, 2019b).

[6] www.onr.org.uk
[7] Further background on wider UK context is available at www.hse.gov.uk/risk/theory and HSE (2001).

United States: The United States Nuclear Regulatory Commission (US NRC)[8] licenses and regulates the operation of commercial NPP under a two-step process (construction permit and operating license) described in Title 10 of the Code of Federal Regulations (10 CFR) under Part 50. In 1989, the US NRC established a combined licensing process in 10 CFR Part 52 to improve regulatory efficiency and add greater predictability. These require a Safety Analysis Report to be submitted that includes design information, hypothetical accident scenarios, plant safety features and details on the proposed site. The US NRC uses risk-informed, performance-based regulation to appropriately consider defence in depth, risk insights and margins of safety using a Probabilistic Risk Assessment (PRA) approach. The US NRC regulations are supported by Regulatory Guides (RGs) to aid licensees in implementing regulations and Nuclear Regulatory Report (NUREG) publications that cover a variety of regulatory, technical and administrative subjects.

Canada: The Canadian Nuclear Safety Commission (CNSC)[9] regulatory documents are organised into three categories: regulated facilities and activities, safety and control areas and other regulatory areas (CNSC, 2018). The CNSC assesses how licensees manage risk by applying concepts such as the ALARA principle and defence in depth. Canada is similar to the UK in that its safety regime is also goal-setting with the onus on the Licensee to demonstrate safety through whichever means it deems suitable. It is committed to continuous improvement and requires licensees to strive to further reduce the risks associated with their licensed activities on an ongoing basis. Therefore, the CNSC also considers how licensees continuously evaluate and further reduce uncertainties through additional safety and mitigation options as techniques and technologies evolve.

[8] www.nrc.gov
[9] www.nuclearsafety.gc.ca/eng

3 The Technical Volumes and Case Studies

This set of documents (six technical volumes and four case studies) are aimed at the technical end-user with some prior knowledge of thermofluid mechanics, who wishes to know more about thermal hydraulic analysis for single-phase heat transfer and passive cooling. General advice on regulation and potential future developments are provided as context, but it is advised that end-users review the latest developments and best practice.

- The technical volumes focus on how to model a variety of different thermal hydraulic phenomena, describe good practice and intend to summarise the state-of-the-art with respect to single-phase heat transfer and passive cooling in a form accessible to industry, including regulators.
- The case studies demonstrate the use of the methods described in the volumes applied to representative industry applications.

An expert industry steering group (consisting of the contributors listed on page i) was formed to develop and review the technical volumes by providing industrial context, reviewing the structure and content and providing specific technical insights to guide the development of this content. The depth and breadth of thermal hydraulic experience within the industry steering group has ensured that the content of the technical volumes and case studies is beneficial and relevant to industry. In addition, these documents have undergone an independent peer review process within Frazer-Nash Consultancy.

3.1 Technical Volumes

Each technical volume provides an overview of the latest methods available to analyse the thermal hydraulic phenomena or topics under consideration within the specific volume. This includes a description of the phenomenon or topic, the methodologies that can be used to analyse it and anticipated future developments. The titles and contents of the technical volumes are:

1. **Introduction to the Technical Volumes and Case Studies:** Summary of the motivation for the project, introduction to thermal hydraulic analysis and description of the NTH analysis methods currently used in industry.
2. **Convection, Radiation and Conjugate Heat Transfer:** This volume covers the temperature variation within solids and fluids and the thermal interaction between them. This process is fundamental to the transfer of heat out of a reactor via conduction, convection and thermal radiation particularly under passive cooling scenarios.
3. **Natural Convection and Passive Cooling:** Natural circulation is used during normal operation and within passive safety systems for a number of advanced reactor designs to cool the

core or containment volume. This volume focuses on passive cooling due to buoyancy-driven flows within loops, channels, pools and plena.

4. **Confidence and Uncertainty:** Two topics that are particularly important for thermal hydraulic analysis within the nuclear industry are: understanding and quantifying the sources and magnitudes of uncertainty, and establishing the level of confidence in the results that is consistent with the significance of the decision that they are being used to support.
5. **Liquid Metal Thermal Hydraulics:** This volume focuses on the thermal hydraulic characteristics that arise in liquid metals (i.e. sodium, lead and Lead-Bismuth Eutectic (LBE)), such as the impact of low Prandtl number on heat transfer.
6. **Molten Salt Thermal Hydraulics:** This volume considers heat transfer in molten salts, which are used as a primary coolant, liquid fuel or secondary/tertiary heat transport. It provides an overview of the physics and chemistry of the salt compositions typically used, and guidance on modelling methods, such as transparency to thermal radiation.

References are provided throughout to facilitate further learning in specific areas of interest, and each volume has a common structure.

Introduction: The thermal hydraulic phenomenon or topic under consideration and how it relates to single-phase heat transfer and passive cooling is introduced and described. This includes a discussion on how it impacts or influences the reactor design in terms of performance or safety with examples provided from existing or proposed reactor designs.

Technical Context: The technical background and theory relevant to the phenomenon is discussed, with reference to passive cooling applications and reactor design, together with a summary of the modelling challenges and issues relevant to all methodologies.

Methodologies: The currently available and robust methodologies for the analysis of the thermal hydraulic phenomenon under consideration are described, including reference to any best practice and relevant new developments from recent research. This begins with a general description of the current methods relevant to the phenomenon under consideration and how they are applied to a reactor design process.

For each phenomenon or topic under consideration the different approaches available for the analysis of the thermal hydraulic phenomenon are identified and reviewed. All appropriate methodologies are described briefly and a balanced discussion of the relative merits and drawbacks of each approach is provided. Recommendations on the application of different methodologies (including their theoretical and practical limitations) is also given to ensure that any methodology can be used correctly.

Future Developments: A brief assessment is provided on the research and anticipated future developments that are relevant to the analysis of the thermal hydraulic phenomenon under consideration (e.g. computational power, automation, increased coupling and uncertainty quantification). This provides a picture of what methods and tools may be available for the design and analysis of ANTs in the near future.

The structure of Volume 4 is a variant of this, because it is not focused on a reactor technology or individual thermal hydraulic modelling method *per se*.

3.2 Case Studies

The technical volumes are supported by a set of four individual case studies. These case studies provide illustrative reactor-specific examples of thermal hydraulic analyses, with an emphasis on passive cooling applications and practical examples showing how the methods outlined in the respective technical volumes can be applied. The case study applications have been chosen to span different thermal hydraulic applications and advanced nuclear technologies:

A. **Liquid Metal CFD Modelling of the TALL-3D Test Facility (TALL-3D):** This study demonstrates the use of CFD analysis for the modelling of both forced and natural circulation in liquid metals and includes consideration of conjugate heat transfer. It also demonstrates the use of test data from the TALL-3D[1] test facility in the Royal Institute of Technology (KTH), Stockholm and high-fidelity methods to inform decision making about modelling methods and improving confidence in analysis results.

B. **Fuel Assembly CFD and UQ for a Molten Salt Reactor (MSR):** Flow through a fuel assembly under forced and natural circulation in the core of a MSR is analysed to demonstrate CFD analysis of molten salt, the derivation of reduced order model parameters, and consideration of input and model uncertainty.

C. **Reactor Scale CFD for Decay Heat Removal in a Lead-cooled Fast Reactor (LFR):** Passive decay heat removal from the whole primary system of a LFR to analyse flow circulation paths and associated heat transfer effectiveness, including heat transfer to/from solid reactor components (conjugate heat transfer).

D. **System Code and CFD Analysis for a Light Water Small Modular Reactor (SMR):** This study demonstrates the role of different fidelities of modelling method through the analysis of a water test facility, with relevance to SMRs. These different methods will be compared to experimental data to demonstrate the insight that can be gained of the more complex aspects of a flow with CFD.

Each case study is intended to illustrate how to analyse a range of passive cooling applications for advanced nuclear technologies. This involves the combination of several thermal hydraulic phenomena or topics; see Table 3.1 for the technical volumes that are applicable to each case study.

		TALL-3D	MSR	LFR	SMR
1	Introduction	-	-	-	-
2	Convection, Radiation and Conjugate Heat Transfer	✓	✓	✓	✓
3	Natural Convection and Passive Cooling	✓	✓	✓	✓
4	Confidence and Uncertainty	✓	✓		✓
5	Liquid Metal Thermal Hydraulics	✓		✓	
6	Molten Salt Thermal Hydraulics		✓		

Table 3.1: Application of the technical volumes to each case study.

Each case study is intended to be illustrative of the application of the methods outlined in the technical volumes and the analysis results are not expected to be directly used. Detailed thermal

[1] The name comes from Thermal-hydraulic ADS Lead-bismuth Loop with 3D flow test section (TALL-3D); operated with relevance to Accelerator-Driven Systems (ADSs).

hydraulic analysis should be undertaken by SQEP professionals when analysing the safety and operational performance of new or existing reactor technologies. The case studies seek to provide engineers and end-users with a better understanding of this type of analysis, and follow a common structure:

Defining the Problem and Planning the Analysis: This is an important step in performing a thermal hydraulic analysis and has a determining impact on its success and efficiency. Each case study will include a description of a realistic scenario under which the analysis is being performed, its relevance to reactor design and safety assessment. The process of planning the analysis will then be described, including the rationale behind the decisions that are made.

Performing the Analysis: A full description of how to set up the models needed and run each of the analyses. The emphasis of this section will be a 'hands-on' guide to reproducing the example analysis.

Application of the Results: Reporting, including explaining what the results mean to project engineers, managers and safety case engineers, as well as to other NTH engineers.

3.3 Assumed Knowledge

It is assumed that readers are comfortable with basic concepts relevant to thermofluid mechanics. These include the mass, momentum and energy conservation principles, and the corresponding constitutive equations that describe the fluid motion and the transfer of thermal energy by fluid motion (these are explained more fully in reference sources such as Rogers and Mayhew, 1992 and Incropera *et al.*, 2011):

Mass conservation principle: Also referred to as mass continuity, this states that the rate at which mass enters a control volume is equal to the rate at which mass leaves the control volume plus the accumulation of mass within the control volume. In differential form this becomes the continuity equation.

Momentum conservation principle: This is the adaptation of Newton's second law of motion to a moving fluid. It relates the sum of the forces acting on an element of fluid to its acceleration, or rate of change of momentum. The forces mainly arise from the pressure gradient, the fluid viscosity which opposes the fluid motion, and the gravitational field. In uniform-density flows the gravitational forces only change the hydrostatic pressure, while in variable density flows they modify the flow through buoyancy. In Newtonian fluids the relationship between the strain rate and the viscous stress results in the widely used Navier-Stokes equations.

Energy conservation principle: This arises from the first law of thermodynamics. In its more general form this takes into account the transport of both mechanical and thermal energy and the conversion between the two forms. In thermal hydraulic analysis which is either concerned with incompressible fluids (liquids), or low Mach number gas flows, and in which the contribution of the viscous force to the generation of thermal energy and thermal radiation can be neglected, the differential form of the energy conservation principle reduces to the widely used heat convection equation.

Bernoulli's principle: This is what the mass and momentum conservation principles can be reduced to for incompressible, inviscid and steady flows: along the flow direction, the total pressure, ($P_T = P_s + \frac{1}{2}\rho U^2 + \rho g h$) is conserved. It is often viewed as an energy conservation equation (for the internal, kinetic and potential energies) and is a very useful analytical tool, but can only be used under the conditions for which the assumptions listed above apply.

Steady Flow Energy Equation (SFEE): Similar to Bernoulli's equation, this is the reduced form of the energy conservation principle, for steady one-dimensional flow. It can take into account heat transfer and work done and changes to kinetic energy, potential energy, internal energy and enthalpy.

Pressure drop: In flows through internal systems (e.g. pipe or duct networks) there is a drop in pressure along the flow direction due to the viscous wall shear stress which opposes the fluid motion. Further losses can be caused by the presence of bends, orifices, expansions or contractions, which further complicate the flow by generating phenomena such as secondary motion or flow separation. At sufficiently high Reynolds number the total pressure loss in each component is proportional to the 'dynamic pressure' ($\frac{1}{2}\rho U^2$, Miller, 2009).

Turbulence: Turbulent flow is fluid motion characterised by apparently chaotic variations in pressure, flow velocity and fluid temperature, as opposed to laminar flow, which occurs when a fluid flows in parallel layers, without small scale unsteadiness. Because the fluctuating turbulent motion has to satisfy the physical laws of fluid motion (mass, momentum and energy conservation) turbulent fluctuations are not random, but instead consist of eddies over a range of sizes, bounded at one end by the mean flow scale and at the other by viscous dissipation. These eddies enhance the transfer of momentum and thermal energy from regions of high to low velocity and high to low temperature respectively, which is why turbulence has such a strong effect on hydrodynamic and thermal performance. Turbulence and transition is discussed in more detail in Volume 3 (Natural Convection and Passive Cooling) because transition is particularly relevant to passive cooling systems using natural circulation. Most current NPP primary circuits operate with forced circulation in the turbulent flow regime.

Either through the process of dimensional analysis, or through the non-dimensionalisation of governing equations such as the Navier-Stokes and energy transport equations, a number of key dimensionless groups can be identified. These can be used to generalise the performance and controlling parameters in flow and heat convection, as well as scaling. The more widely used dimensionless groups are presented below (definitions are provided in Section 6):

Reynolds number (Re): Ratio of momentum forces to viscous forces. Key group for forced convection.

Grashof number (Gr): Ratio of buoyancy forces to viscous forces. Key group for natural convection.

Prandtl number (Pr): Fluid property, ratio of momentum diffusivity to thermal diffusivity. Key group for forced and natural convection.

Rayleigh number ($Ra = Gr Pr$): Characterises flow regime for buoyancy-driven flow. Key group for natural convection.

Richardson number ($Ri = Gr/Re^2$): Represents the relative importance (ratio) of natural convection and forced convection (discussed in Volume 3, Section 2.2.1).

Nusselt number (Nu): Ratio of convective to conductive heat transfer in a fluid at a boundary. Key group for quantifying heat transfer.

Péclet number ($Pe = RePr$): Ratio of heat transfer by motion of a fluid to heat transfer by thermal conduction.

Stanton number (St): Alternative to the Nusselt number. Dimensionless form of the wall heat flux coefficient, only suitable for forced convection flows.

Additional dimensionless groups can be derived by using various scaling analysis methods (Section 4.6.5).

4 Nuclear Thermal Hydraulics Analysis

Nuclear thermal hydraulics analysis is a very large field, so this section provides a high-level introduction with an emphasis on CFD methods. The section starts with an overview of the approaches used to perform analysis, before discussing the key techniques that are used to understand what analysis should be performed and develop confidence in this analysis. An overview of system and subchannel analysis is provided, and key aspects of CFD analysis are introduced. Finally, experimental methods are considered.

4.1 Overall Analytical Approach

This section provides an overview of how different tools are used for NTH analysis, explains how complementary computational and experimental analysis is often used, and highlights how different analysis is important across the lifecycle of a NPP.

4.1.1 Analysis Tools

A large number of computer codes have been developed to design, analyse and operate NPPs. Nuclear reactor systems operate at a level of sophistication whereby human reasoning and theoretical models alone are not capable of providing a full understanding of a system's behaviour. There is, however, an inherent need to acquire such understanding, particularly for safety analyses. Since the 1960s there has been a concerted effort on the part of the power utilities, regulatory bodies, research organisations, industry and academia to develop advanced computational tools for simulating reactor system thermal hydraulic behaviour during real and hypothetical transient scenarios. This section introduces and summarises some of the key techniques and tools that are currently used for NTH analysis and how they are used together. More details on these tools are provided in the following sections.

Computational analysis of systems and components is generally performed using one or more of the following types of analysis, see Figure 4.1:

System codes: These are used to analyse the performance of a whole system, mainly as 1D flow paths to look at overall performance, particularly during transients. These codes vary greatly in complexity and incorporate single- and two-phase models, and can be coupled to more detailed models for specific components (such as subchannel or CFD codes) (Section 4.4.1).

Subchannel codes: These are used to perform reduced-order spatial analysis of complex components within a system (particularly the reactor core) for which the use of more detailed CFD would be too expensive or cumbersome. They generally use a computational mesh too coarse to predict the detailed flow behaviour, but contain sub-grid-scale models to incorporate the impact of single- and two-phase flow behaviour on the bulk flow field. They are generally designed for specific applications, and so trade general applicability for faster run-times by including specific modelling approaches tailored to the application (Section 4.4.2).

CFD methods: These are general-purpose methods that enable detailed spatial and temporal analysis of flow and heat transfer in parts of a system. They use a finer mesh than subchannel codes, and as a result model more of the flow behaviour directly. A range of modelling approaches are available, with varying fidelity and computational requirements (Section 4.5).

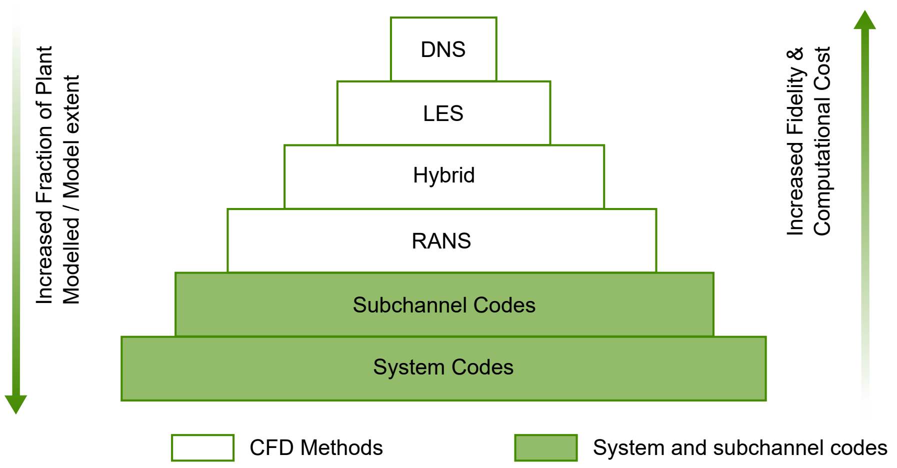


Figure 4.1: General hierarchy of modelling approaches.

Computational analysis using system, subchannel or CFD codes can be performed with boundary conditions that are constant (a steady-state scenario such as operation at full power) or time-varying (a transient scenario such as shut-down). In a transient calculation, the flow variables are solved repeatedly as time passes in the simulation, so these calculations tend to be much more computationally intensive than steady-state calculations.

In general, there is a trade-off between using appropriate modelling fidelity and computational/mesh generation time/cost, and so advanced modelling tools are generally used where lower fidelity tools do not provide enough confidence on their own. It is currently not possible to investigate the performance of a whole system using detailed CFD alone, and so a trusted system code is more appropriate for design iterations. This is particularly true in the nuclear industry, where many of the scenarios of interest are transients (e.g. shutting down the reactor and removing heat from it). System and subchannel codes do not need detailed geometry definition and so are useful during concept design, provided that they adequately represent the key phenomena; while CFD methods need to model a realistic 3D geometry especially in component design iterations for optimisation and trade-space exploration.

For all types of analysis, it is important to select the appropriate modelling approach to resolve the problem by identifying the key phenomena that need to be simulated (e.g. through a PIRT, Section 4.2), and build appropriate confidence in the results through VVUQ (Section 4.3).

Engineers therefore need a range of tools at their disposal to enable them to provide appropriate support to the design and maintenance of a NPP. For example, the overall performance of a cooling system could be investigated using a system code with a subchannel code to model detailed components like the hot channel in a reactor core. However, specific aspects of flow performance (such as the distribution of flow in a plenum) could be investigated using CFD and parameterised in the system code for the particular transient, or more generally the CFD code could be coupled to the system code.

4.1.2 Computational and Experimental Analysis

Synergy between experimental, theoretical, and computational work has been long recognised as essential for the progression of science and engineering, and this is also true for NTH. In this way, while more computational work is being used across industry, neither approach is intrinsically superior to the other, and both approaches remain important. Computational and experimental work are generally viewed as two complementary approaches for building understanding of physical phenomena and increasing confidence in modelling work:

- Computational analysis can often provide more detailed information about a given scenario than experimental work. Computational analysis may also be faster, enabling more sensitivity studies to be run in less time, and can be performed at full scale. However, particular care must be taken over PIRT, scaling analysis and VVUQ.
- Experimental analysis provides real measurements of physical phenomena and data for model validation. However, particular care must be taken over scaling analyses and understanding the uncertainty and accuracy of the measurements.

Many numerical simulations focus on problems that have already been investigated experimentally. Whilst this can often function as part of the Verification and Validation (V&V) process, the nature of numerical work means the results will usually achieve a level of detail well above that obtained in the experiment. This data may be used to inform future experimental work (e.g. indicate further tests that could add value or areas where probes could be placed to learn more about the flow). This two-way interaction offers great value and is key to obtaining full value from both investigatory methods (Section 4.6.1). While Direct Numerical Simulation (DNS) now provides an alternative and often fruitful route for generating validation data, due to the heavy computational demands, DNS can currently only be used to provide validation data for geometrically simple cases at relatively low Reynolds numbers (Section 4.5.3). So for many important NTH scenarios, adequately scaled experiments remain the only viable route for acquiring validation data (Section 4.6).

Such is the breadth of knowledge and expertise required in numerical and experimental research, many scientists and engineers specialise in either area, identifying as ‘experimentalists’ or ‘modellers’. This situation may reduce interaction between these two groups, causing communication gaps that can reduce the value of both aspects. Though the recent advances in computational power and code accessibility have sharpened the need for closer integration, engineers should be proactive in addressing this historical divide (Section 4.6.1). Scaling analysis experts can also provide interaction between these two groups (Section 4.6.5).

4.1.3 Analysis through the Whole Project Life Cycle

NTH analysis is performed throughout the whole life cycle of a NPP, from early studies to decommissioning. The amount of computational and experimental work that is performed is likely to depend on the particular point of a project life cycle and the safety significance of the analysis work for the NPP (in line with a graded approach, Section 2.2.5):

Initial research and feasibility studies: Basic calculations and system code analyses are used to firm-up the reactor concept. University research may also be undertaken to develop understanding of specific aspects of the technology.

Concept design: Confirm the viability of a concept design, reduce the risk of fundamental design changes and secure the economic model. Most thermal hydraulics work is performed using system codes to refine the concept design and underpin the basis of the economic model and safety case. Key aspects of the design (such as the reactor core) are analysed using subchannel or CFD codes. This work will involve a lot of design iterations and dynamic plant simulations. Experiments may also be needed at this stage to provide confidence in the performance of novel aspects of the design, alongside an ongoing Research and Development (R&D) programme.

Preliminary design: Front-end or basic design is focused on reducing the risk of substantial changes during detailed design, securing key aspects of the evidence for the safety case and engaging with regulators. System codes are used to finalise the overall design of the systems and significant effort is spent on VVUQ (Section 4.3) to minimise the risk of expensive re-work later in the programme. CFD and/or experimental methods are used to study flow phenomena, predict aspects of the design performance and perform de-risking substantiation work on the preliminary design. Subchannel or containment codes are also used to develop the design of the reactor and the safety systems. If a performance-critical aspect is identified then higher fidelity analysis methodologies are likely to be used to reduce uncertainty and improve confidence. At this stage, work is carried out to improve and validate the computational tools (system, subchannel and CFD).

Detailed design: Analysis is used to finalise the design and secure the design substantiation for the plant ahead of construction and maintain regulatory confidence. Analysis may also be needed to understand the impact of design changes during the detailed design of the SSCs.

Construction and commissioning: Analysis may be used to understand whether Non-Conformance Reports (NCRs) or adverse commissioning results can be managed within the existing safety case, or if changes to the plant are required.

Operational life: Analysis is used to investigate aspects of plant performance, assess whether unexpected plant monitoring results can be managed within the existing safety case and to support plant modifications (in a similar manner to the design aspects above).

Decommissioning: Analysis is used to support the design of storage facilities (particularly for spent fuel) and the removal of systems in a safe manner.

As the design progress, the questions that need to be addressed and level of detail required will develop. For example, at a concept stage an order of magnitude indication may be needed to inform design decisions, while detailed analysis with uncertainty quantification may be required to support the detailed design. Therefore, a staged analysis approach is appropriate as the design progresses that builds on the level of detail, validation and confidence associated with the analysis.

For a low-power demonstration plant, the amount of analysis work and VVUQ is likely to be smaller than for a commercial NPP, in line with the overall risk and graded approach (Section 2.2.5).

4.2 Phenomena Identification and Ranking Table (PIRT)

The first step in any thermal hydraulic analysis is to understand whether the software being used can adequately model the flow phenomena involved. The PIRT process is a systematic way of identifying the processes and phenomena that have the most dominant influence on the selected Figures of Merit (FOMs). The State of Knowledge (SOK) and importance of each phenomenon is assessed by eliciting the views of a panel of experts in order to identify and prioritise areas where more research, development or validation is required. More systematic, and less based on the expert opinion, quantitative ranking of identified processes and phenomena can be achieved based on scaling analysis results.

This was originally developed for thermal hydraulic analysis of fault scenarios using system codes, but is becoming more widely used within the CFD community and outside the nuclear industry as an important part of any validation process (CSNI, 2015). Although the full PIRT process is quite onerous, engineers may well consider similar steps in an informal manner within the context of a graded approach (e.g. for systems that are simple and have lots of margin).

The PIRT process was developed in 1989 by the US NRC as part of the Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology (US NRC, 1989) to demonstrate the feasibility of the BEPU approach compared to common conservative approaches (introduced in Section 2.2.5.3). The BEPU methodology provides more realistic estimates of the plant safety margins compared to the conservative approach, which may improve plant economics through a higher operational flexibility while still meeting the regulatory requirements.

Wilson (2013) gives a historical perspective of the original CSAU and PIRT processes and subsequent improvements in BEPU methodologies. The PIRT process is also the fourth step in the first element of the US NRC Evaluation Methodology Development and Application Process (EMDAP) guide for assessing models that may be used to analyse transient and accident behaviour (US NRC, 2005). The PIRT process has been continuously refined since 1990 and these advances are embedded in the following nine step process (Wilson, 2013); see Figure 4.2.

1. **Define Issue:** The first step in the PIRT process is to clearly determine and define in detail the problem that needs to be resolved and to what level. This depends on the budget available and level of PIRT that is required to resolve the problem.
2. **PIRT Objectives:** The next step is to define the objectives of the PIRT, which is dependent on its intended use. The primary function is to identify the relative importance of systems, components, processes and phenomena in driving the plant response. However, it can also be used to establish requirements for new experimental programs, code development and improvement or code validation and uncertainty quantification.
3. **Database:** The PIRT team should identify, obtain and review all available experimental and analytical data relevant to the problem, and create a test assessment matrix that lists the

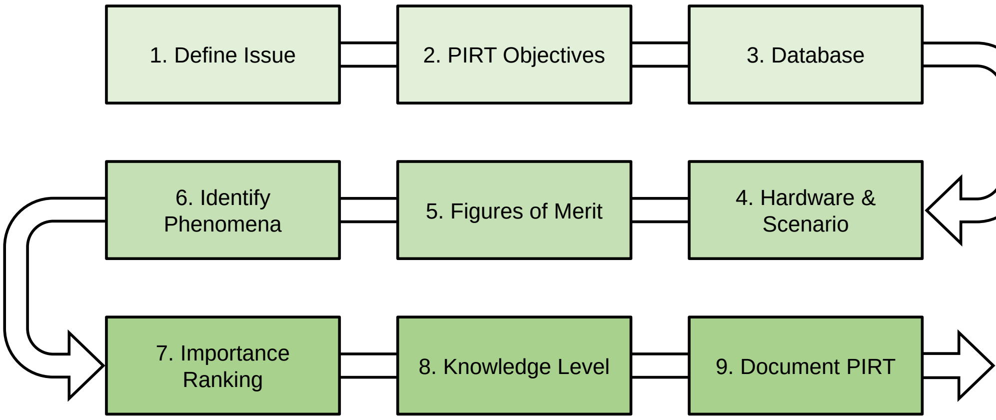


Figure 4.2: Nine step PIRT process.

available test data and their range of application. This will enable the team to develop a knowledge base that represents the state-of-the-art understanding of all relevant factors.

4. Hardware and Scenario: The relative importance of phenomena/processes depends on the scenario under consideration. The combination of plant specific designs and scenario specific features will determine the envelope that must be considered in the PIRT development. The value of the PIRT tends to increase with the specificity of the scenario to a particular application, and so it may be more efficient to do several specific PIRT exercises rather than one large general one.

5. Figures of Merit: FOMs are those criteria against which the relative importance of each phenomenon is judged (e.g. peak fuel cladding temperature). Appropriate figures of merit have distinct characteristics. They should be directly related to the issue being addressed by the PIRT and the phenomena being assessed, easily understood, explicit and measurable.

6. Identify Phenomena: The relative importance of phenomena to the scenario under consideration often changes as the scenario progresses, therefore the scenario should be partitioned into time sequences in which the phenomenological behaviours are consistent. The PIRT panel should first identify plausible phenomena/processes that have some significance to the plant behaviour without any ranking through a brain storming session. Starting with the identification of high level system processes has been shown to be an efficient approach to identifying the components and phenomena that should be considered (Wilson and Boyack, 1998).

7. Importance Ranking: The ranking process is at the heart of the PIRT development, and is intended to rank the importance of each identified phenomenon for each time sequence relative to the impact on the FOM. The phenomenon qualitative ranking is usually done on a scale of low, medium and high. Later, scaling analysis (if performed) can provide quantitative ranking.

8. Knowledge Level: The next step in the process is to assess the current State of Knowledge (SOK) for each phenomenon. This will vary depending on the focus and objectives of the PIRT in terms of whether it is being used for designing an experimental program, code development or quantifying uncertainty in a code (CSNI, 2015). The SOK level is usually done on a scale of low, medium and high.

The importance ranking and SOK for each phenomenon are combined to identify the high priority gaps in knowledge that need to be filled through experimental data or code development and validation. This is highlighted by the darker cells in Figure 4.3 - a phenomenon that has high importance and low SOK requires additional research. It is worth noting that the PIRT development is an iterative process with feedback between the steps as more information becomes available before the first draft of the PIRT is finalised, as well as being revisited at a future point in a development programme.

State of Knowledge	Importance of Phenomenon: High	Medium	Low
High			
Medium			
Low			

Figure 4.3: Table highlighting areas where research is suggested by PIRT process.

9. Document PIRT: Although the ranking table is useful, the real value of the PIRT to the developers and users is in the documentation of the PIRT process results. Experience by Wilson and Boyack (1998) has shown that success in developing useful PIRTs is achieved through well documented output. In addition to the main results of the PIRT (i.e. table(s) grouping the phenomena according to their importance and SOK), the PIRT documentation should also include a description of the system, scenario under consideration and figures of merit, ranking scale used, phenomena and processes definitions, evaluation criteria and the rationale behind each importance ranking and state of knowledge. This practically documents the fourth step in the first element of EMDAP (US NRC, 2005).

The successful development of a PIRT is highly dependent on the composition of the panel. The collective expertise of the panel should be broad with extensive and current knowledge in their field. At least one member should have appropriate, relevant expertise in each of the following fields: experimental programmes and measurements, code development, code application to nuclear safety analysis, operation of the system under consideration and PIRT development. A coordinator should be appointed and experience by Wilson and Boyack (1998) has shown that a small team of about six people is most effective.

The PIRT process is a useful, systematic way of identifying and prioritising the most important thermal hydraulic phenomena, which is being applied to advanced reactor designs (ORNL, 2008, Liao *et al.*, 2019). This enables the areas that require additional research or code development to be identified, together with the verification and validation requirements for the NTH analysis (Section 4.3).

4.3 Verification, Validation and Uncertainty Quantification (VVUQ)

Computational modelling and simulation can reduce the number of physical tests necessary for engineering system development, as long as the confidence in the results is appropriate for the application of interest. VVUQ is the process used to establish the confidence in and adequacy of predictive computations for a particular application, and estimate the error and uncertainty bounds associated with it.

This section provides a high level summary of the key principles associated with VVUQ; more detailed information is provided in Volume 4. The level and extent of validation required for a particular nuclear thermal hydraulic analysis will depend on the safety classification and should be undertaken within the context of a graded approach.

The level of validation required should be agreed before undertaking any analysis and is often identified as part of a PIRT (Section 4.2). In addition to VVUQ, it is important that all models, methods and results are independently checked as part of a formal quality assurance process.

Best practice guidelines have been developed by the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) for CFD applications (CSNI, 2015) and the extension of uncertainty quantification methods to become a routine part of CFD is being pursued (CSNI, 2016). These reference the American Society of Mechanical Engineers (ASME) V&V 20 standard[1] for verification and validation in CFD and heat transfer applications (ASME, 2009), which conforms to US NRC regulatory practices for the licensing of nuclear power plants. In addition, the ASME V&V 30 standard on computational simulation of nuclear systems thermal fluids behaviour is expected to be published in the near future.

Verification ('solving the equations right') and validation ('solving the right equations') are defined in ASME (2006) as:

Verification: The process of determining that a computational model accurately represents the underlying mathematical model and its solution.

Validation: The process of determining the degree to which a model is an accurate representation of the real world from the perspective of the intended uses of the model.

Verification, validation and uncertainty quantification are each discussed separately below.

4.3.1 Verification

Verification, from the perspective of the results of the solution obtained from a particular tool, is formed of two separate parts; code verification and solution verification.

Code Verification: *'Establishes that the code accurately solves the mathematical model incorporated in the code, i.e. that the code is free of mistakes for the simulations of interest'* (ASME, 2009). Therefore, this precedes validation and is intended to find and remove mistakes in the source code and numerical algorithms, and so code verification is predominantly the responsibility of the code developer. Code verification is used to assess the observed order

[1] www.asme.org/codes-standards/publications-information/verification-validation-uncertainty

of accuracy of a code through three types of reference solutions: traditional exact analytical solutions, method of manufactured solutions and benchmark computational solutions.

Solution Verification: *'Estimates the numerical accuracy of a particular calculation'* (ASME, 2009). The solution (calculation) verification is performed, after successful code verification, as part of validation to assure the numerical accuracy of the code output for the problem of interest. Solution verification identifies and minimises the numerical errors due to the choices that a user has made, which includes mesh and time step discretisation, convergence, conservation, round-off errors and other potential computational artefacts (e.g. domain decomposition and parallel processing). Therefore, the solution verification output is a quality assurance step and an error estimate that is applicable to a particular test case, and so is normally considered the responsibility of the code user.

In addition to this, however, verification of an analysis that is to be used as part of a nuclear safety justification, or to make a decision with significant financial implications is required to be broader. It should be subject to independent checking (of appropriate, graded rigour) with particular attention paid to the traceability and provenance of the sources of inputs, such as geometry, boundary conditions and material properties, as well as to post-processed outputs, such as derived or plotted quantities.

4.3.2 Validation

The purpose of validation is to assess how adequately the verified computational solutions compare with the experimental data, where the errors and uncertainties are quantified for both. Validation is typically an ongoing activity, as additional validation exercises provide an increasing level of confidence or expand the validated domain (ASME, 2009).

The level of validation required to support a nuclear safety case should be determined using a graded approach (Section 2.2.5). For example, validating solutions to support a complex Class 1 safety system is likely to be a significant effort, especially if new experiments and extensive validation calculations are required.

Validation should ideally follow a hierarchical approach that separates and simplifies the physical phenomena involved in the system of interest. This should be undertaken as part of the PIRT process (Section 4.2) to identify and rank the most important thermal hydraulic phenomena, and as a part of hierarchical scaling analysis methods (Section 4.6.5). The types of validation tests at each level of the hierarchy are shown in Figure 4.4, and consist of basic tests, Separate Effect Tests (SETs), component tests, Integral Effect Tests (IETs) and finally data from real plant (these aspects are considered further in Section 4.6).

High quality experimental measurements or benchmark data for validation of thermal hydraulic phenomena are hard to find. A validation matrix for system codes was developed in the 1990s by CSNI based on existing SET (CSNI, 1994) and IET (CSNI, 1996) experimental data. Although there is no existing CFD validation matrix, some sources of data include the CSNI NEA CFD benchmarks, ERCOFTAC Knowledge Base Wiki[2] and published DNS data (e.g. Shams *et al.* (2019)).

[2] www.kbwiki.ercoftac.org/w/index.php/Main_Page

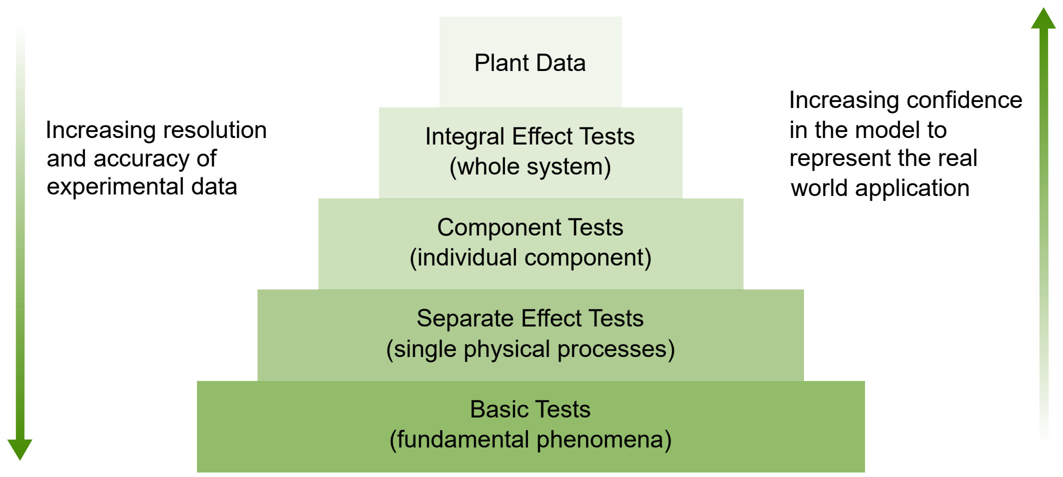


Figure 4.4: Hierarchical validation approach.

4.3.3 Uncertainty Quantification

Uncertainty quantification determines how errors and uncertainties in numerical and physical parameters of inputs and models affect simulation outcomes. Despite being commonly used terms, the definition of error and uncertainty is neither unique nor able to be succinctly stated in a helpful form, but is described in the relevant standard (JCGM, 2008).

Uncertainty quantification includes uncertainties in the simulation result due to the numerical solution of the equations, uncertainties in the simulation result due to code inputs and uncertainty in experimental results, as well as the scaling or extrapolation of the simulation result to reactor conditions (ASME, 2009).

Uncertainty quantification methodologies are based on either propagation of input parameters uncertainty (requiring many simulations or sensitivity studies) or extrapolation of accuracy (requires large experimental database). The propagation of input parameter uncertainties often involves identifying the input uncertainties, defining a range or probability distribution for each parameter and then running simulations according to the sampling approach adopted. Input parameter uncertainties include initial and boundary conditions, material properties, model parameters, modelling options (like mesh size and applied turbulence models or correlations) and geometry simplification. A graded approach may be useful to understand the amount of work required (Section 2.2.5).

A key aspect of uncertainty quantification for nuclear thermal hydraulics is the extrapolation of results from scaled experiments to full scale. Since reduced-scale tests cannot fully respect all non-dimensional parameters, particular attention is required to rigorously quantify the uncertainty at reactor scale (CSNI, 2017).

More detailed information on uncertainty quantification techniques is provided in Volume 4.

4.4 System and Subchannel Analysis

The role of system and subchannel analysis in nuclear thermal hydraulics is introduced in Section 4.1. This section considers system and subchannel analysis in more detail, and introduces some of the codes that are available. More detailed information on the capability and limitations of system and subchannel codes for modelling heat transfer and convection is provided in Volume 2 (Convection, Radiation and Conjugate Heat Transfer) and Volume 3 (Natural Convection and Passive Cooling) respectively.

4.4.1 System Analysis

System analysis tools for NTH, or 'system codes', have been developed since the 1970s to simulate reactor thermal hydraulics in normal operation and during accident scenarios (D'Auria, 2017), and are extensively used in the design and safety analysis of new and existing NPPs. System codes are most widely developed and validated for water-cooled reactors, but are now being applied to ANTs.

4.4.1.1 Approach

System codes model the whole reactor system by using a nodalised representation of all flow and thermal interactions between every sub-system and component, and then solve models for the thermal hydraulics (mass, momentum and energy), conduction and thermal radiation, neutron physics, fuel thermomechanics and control systems. Therefore, system codes must be able to simulate all relevant phenomena and processes, which includes steady-state and transient behaviour. The main benefits and limitations of system codes are summarised in Table 4.1.

Benefits	Limitations
Capable of simulating transient behaviour of whole primary circuit	Unable to resolve detailed mixing and complex multi-dimensional flow features
Extensively validated against experimental data for water-cooled reactors	Only considered acceptable within validated range of scenarios and operating conditions (pools and plena are a particular weakness)
Capable of simulating coolants other than water with some validated for advanced reactor designs	Less experimental data exists for advanced reactor designs to validate the codes

Table 4.1: Summary of benefits and limitations of thermal hydraulic system codes.

System codes are normally used to simulate the physical reactor system as accurately as possible using a best estimate approach (introduced in Section 2.2.5.3). This requires a large number of correlations or closure laws for mass, momentum and energy transfer to be developed from and validated against extensive experimental databases. To support the validation of thermal hydraulic system codes, the CSNI developed a SET matrix (CSNI, 1994) and IET matrix (CSNI, 1996). In order for system codes to be used as part of a safety analysis, uncertainty quantification is required to support a BEPU methodology.

System codes contain many more empirical correlations than CFD codes (Section 4.5), but they have been extensively validated on a much wider range of NPP-specific test cases than CFD and will remain a major NTH analysis tool. One of the key limitations of system codes is the ability to adequately predict multi-dimensional phenomena, such as complex flow features and mixing in plena.

Although most system codes were originally developed and validated for PWRs, they are currently being extended to support advanced reactor designs and include coupling to CFD codes. This enables system codes to simulate the whole reactor system with embedded CFD models of complex mixing regions (Volume 2, Section 3.1).

This increased integration between system and CFD codes is demonstrated through the development of coupling frameworks, such as Idaho National Laboratory (INL)'s open source Multiphysics Object-Oriented Simulation Environment (MOOSE) framework (Volume 2, Section 4.1), and the US NRC's proposed Comprehensive Reactor Analysis Bundle (CRAB) code suite for non-Light Water Reactor (LWR) systems safety analysis (US NRC, 2020).

4.4.1.2 Codes

Some of the principal system codes currently used in the nuclear industry worldwide are ATHLET, CATHARE, RELAP5-3D, RELAP-7, SAM, SAS4A/SASSYS-1 and TRACE. These codes have been selected as there is a large community of users, and licenses are available from the developers depending on the nature of use and export control regulations. The background and history of each code is summarised below, while the theory and implementation of these codes is discussed and compared by D'Auria (2017), Roth and Aydogan (2014a) and Roth and Aydogan (2014b).

ATHLET: ATHLET (Analysis of Thermal Hydraulics of LEaks and Transients)[3] development began in the 1970s at GRS for the analysis of leaks and transients in PWRs and Boiling Water Reactors (BWRs). The latest version, ATHLET3.2, is validated for design basis and beyond design basis accidents (without core degradation) in LWRs, and can simulate advanced reactor coolants (helium, liquid metal and molten salt), although these extensions are subject to further development and validation. ATHLET has a modular structure with four core modules: a fully implicit thermofluid dynamics module, heat conduction and heat transfer module, neutron kinetics module and a control and balance of plant module. Other independent modules can be coupled to ATHLET and interfaces exist to 3D neutron kinetic codes or 3D CFD codes for coupled simulations.

CATHARE: CATHARE (Code for Analysis of THermalhydraulics during an Accident of Reactor and safety Evaluation)[4] code development began in 1979 as part of an agreement between CEA, EDF, AREVA and IRSN. CATHARE2 is a two-phase thermal hydraulic simulator that is used, in particular, for PWR safety analyses, verification of post-accidental operating procedures and R&D.

CATHARE3 has 0D, 1D or 3D modules and can model any water-cooled reactor using a two-phase model with six equations (conservation of mass, energy and quantity of movement for each phase) and up to four non-condensable gases. It is capable of simulating small and large-break loss-of-coolant accidents, steam generator tube ruptures, feed water line breaks, residual heat removal failures and steam line breaks.

[3] www.grs.de/en/computer-code-athlet
[4] cathare.cea.fr

CATHARE3 is the latest version that is currently under development, which includes the following improvements over CATHARE2:

- Improved two-phase flow modelling using multi-field and turbulence models.
- Thinner and non-conforming structured meshes to improve 3D modelling.
- Improved coupling with other codes (e.g. thermal hydraulics and reactor physics).
- Extension to advanced reactor technologies (e.g. SFRs, High Temperature Gas-cooled Reactors (HTGRs) and SCWRs).

RELAP5-3D: RELAP (Reactor Excursion and Leak Analysis Program) code development began in 1975 at INL under US NRC sponsorship for the analysis of transients and accidents in water-cooled NPPs. In the early 1980s, the United States Department of Energy (US DOE) began sponsoring additional RELAP5 development, which continued until 1995 up to the release of RELAP5/MOD3.2. At this point, the code was split into a US NRC version (TRACE) and a US DOE version (RELAP5-3D).

The RELAP5-3D code[5] maintains the proven performance and validation history of RELAP5/MOD3.2 with additional enhancements, including a fully integrated, multi-dimensional thermal hydraulic and kinetic modelling capability, a new matrix solver for 3D problems, new water properties and improved time advancement.

RELAP5-3D's modelling capability has expanded to include large-break loss-of-coolant accidents and operational transients for PWRs, BWRs, MSRs, Liquid Metal-cooled Fast Reactors (LMFRs), HTGRs, supercritical fluid and advanced reactor designs.

RELAP-7: RELAP-7[6] is the most recent evolution of the US DOE RELAP5-3D code, which is being developed for the Risk Informed Safety Margin Characterisation (RISMC) Pathway as part of the light water reactor sustainability programme using INL's open source MOOSE framework. This includes mechanistic calculations to represent plant physics, but has also been designed to be integrated into probabilistic evaluations using the RISMC methodology. The RISMC methodology is intended to optimise plant safety and performance by incorporating plant impacts, physical ageing and degradation processes into the safety analysis.

RELAP-7 retains and improves upon RELAP5-3D's knowledge, extends the analysis capability and improves the accuracy for reactor systems simulation scenarios. The physics in RELAP-7 can be solved simultaneously (i.e. fully coupled) to resolve dependencies and reduce spatial and temporal errors. The main improvements in RELAP-7 are:

- Seven-equation two-phase flow model (liquid, gas and interface pressures).
- Second-order accuracy in both space and time.
- Implicit tightly coupled time integration for long duration transients.
- Tight coupling to higher fidelity physics, such as the BISON nuclear fuels performance application.
- Easy coupling to multi-dimensional core simulators being developed in other US programs, such as Nuclear Energy Advanced Modelling and Simulation Program (NEAMS) and Consortium for Advanced Simulation of Light Water Reactors (CASL).

[5] relap53d.inl.gov/SitePages/Home.aspx
[6] relap7.inl.gov/SitePages/Overview.aspx

SAM: SAM (System Analysis Module)[7] is a new system analysis tool that is being developed at Argonne National Laboratory (ANL) for advanced reactor technologies, such as SFRs, LFRs and MSRs. It aims to provide a fast-running, whole-plant transient analysis capability with improved fidelity using INL's open source MOOSE framework. It includes a multi-dimensional flow model for thermal mixing and stratification phenomena in large enclosures for safety analysis, and can be coupled to other advanced simulation tools.

SAS4A/SASSYS-1: SAS4A/SASSYS-1 (Reactor Safety Analysis System)[8] was developed by ANL primarily to perform safety analysis of SFRs, and was subsequently supplemented with the properties of lead coolant to extend its applicability to LFR systems. The code is capable of simulating the thermal, hydraulic and neutronic behaviour of power and flow transients in LMFRs for anticipated operational occurrences, Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs).

It was originally developed in the late 1960s for the design and analysis of the Fast Flux Test Facility (FFTF). The SAS4A/SASSYS-1 code V&V covers a fundamental test suite for SFRs, and includes validation against the experimental data from the EBR-II shutdown heat removal tests, and FFTF loss of flow without scram tests. Validation efforts are currently ongoing to include experimental data from lead-based facilities, and it supports coupling to a variety of analysis and optimisation tools.

TRACE: TRACE (TRAC/RELAP Advanced Computational Engine)[9] is the US NRC best estimate reactor system code for analysing transient and steady-state thermal hydraulic behaviour in PWRs and BWRs. TRACE development began in 1997 with the aim of combining and consolidating three US NRC codes into a single computational platform: TRAC-P (system code originally designed to model large-break loss-of-coolant accidents and system transients for PWR reactors), TRAC-B (system code originally designed to model small and large-break loss-of-coolant accidents and system transients for BWR reactors) and the US NRC version of RELAP5, which is currently RELAP5/MOD3.3.

TRACE can model thermal hydraulic phenomena in both 1D and 3D space, and is able to analyse small and large-break loss-of-coolant accidents and system transients in PWRs and BWRs. This code is still widely used due to its validation history despite its lack of multi-dimensional modelling capability and limited choices for working fluids, although some work has been performed to adapt and validate TRACE for modelling lead-based systems.

4.4.2 Subchannel Analysis

The whole plant behaviour, as predicted by system codes, is only part of the process of determining the thermal hydraulic operating limits and safety margins of a reactor because the representation of a reactor core is generally coarse grained and the horizontal pressure gradients are normally not correctly modelled (and assumed to be small). The thermal hydraulic safety margins and operating power limits of most reactors are currently assessed using subchannel analysis codes.

In some cases, an initial simulation may be run using a system code for an entire NPP and its output may be used as initial/boundary conditions for a subchannel code to analyse the thermal

[7] www.anl.gov/nse/system-analysis-module
[8] www.ne.anl.gov/codes/sas4a-sassys-1
[9] www.nrc.gov/about-nrc/regulatory/research/safetycodes.html

hydraulic behaviour of the reactor core in more detail.

4.4.2.1 Approach

In subchannel codes, the governing equations of mass, momentum and energy are solved in control volumes that are resolved at a level of the gaps between individual fuel rods. The flow distributions in the rod bundle geometries are predicted by empirical correlations for heat and momentum transfer at surfaces and interfaces, as well as inter-channel mixing models to account for the transport between adjacent sub-channels.

Subchannel codes require empirical correlations for friction loss, heat transfer, mixing and boiling, and are a representation of intermediate level of detail: Coarser than a CFD model, but resolved sufficiently to provide estimations of the local conditions of the sub-channels in the limiting fuel assembly of the core. The main benefits and limitations of subchannel codes are summarised in Table 4.2.

Benefits	Limitations
Predict fuel and fuel cladding temperatures and usually capable of simulating core transient behaviour	Lower fidelity than CFD and so may not be able to predict local phenomena in channels
Extensively validated against experimental data for water-cooled reactors	Empirical correlations are only considered valid within specific range of conditions
Some are capable of simulating coolants other than water for advanced reactor designs	Less experimental data exists for validating the codes for advanced reactor designs

Table 4.2: Summary of benefits and limitations of thermal hydraulic subchannel codes.

Subchannel codes involve more complex representations of the physics of thermal hydraulic phenomena than system codes, and are required to predict local fuel and fuel cladding temperature, void fraction and the margin to Critical Heat Flux (CHF). Increased computing power has allowed subchannel codes to represent entire reactor cores, rather than just a single channel, and have progressively added 3D solution features, resulting in a gradual overlap with lower-resolution forms of CFD. Another technique that has approached this functionality, but from the opposite direction, is Coarse-Grid-CFD (Volume 2, Section 4.2) where flow between fuel pins and medium / large scale flow features are simulated, but at a lower fidelity than ‘true’ CFD.

4.4.2.2 Codes

Examples of subchannel codes in current use and under further development are CTF, FLICA, SUBCHANFLOW and VIPRE. Licenses are available from the developers depending on the nature of use and export control regulations. The background and history of each code is summarised below, while the theory and implementation of these codes is discussed and compared by Moorthi *et al.* (2018), Cheng and Rao (2015) and NSC (2016).

CTF: The COBRA (COolant Boiling in Rod Arrays) software was developed to solve the flow and enthalpy distribution in nuclear fuel rod bundles and cores by Pacific Northwest National Laboratory (PNNL) in the 1960s. This was expanded by PNNL in 1980 under sponsorship of the US NRC to form the original COBRA-TF (Two-Fluid) code. Since then, various organisations have adapted and further developed the code, resulting in many other COBRA variants, such as F-COBRA-TF, COBRA-FLX, COBRA-IE, COBRA-LM, MATRA and ASSERT-PV.

CTF[10] is the name given to the version of COBRA-TF developed and maintained by Oak Ridge National Laboratory (ORNL) and the Reactor Dynamics and Fuel Modeling Group (RDFMG), initially at Pennsylvania State University, and currently at North Carolina State University (NCSU). CTF is based on a separated flow representation of two-phase flow into three fields: liquid film, liquid droplets and vapour. It is capable of modelling solid structure and fluid regions within the core under normal operation and accident scenarios. Improvements in CTF include turbulent mixing, void drift and grid-spacer heat transfer enhancement.

CTF has recently been included in two large projects: US DOE CASL and European Commission (EC) NUclear REactor SAFEty simulation platform (NURESAFE). This has led to a number of developments within the code and resulted in a significant amount of verification and validation testing. CTF has also been integrated into the Virtual Environment for Reactor Applications (VERA) Core Simulator environment, and work is underway to modify and validate CTF for MSR and SFR reactor technologies.

FLICA: FLICA-4 is a 3D two-phase flow code for reactor core analysis that began development in the 1970s at CEA. It uses a four-equation mixture model (mass, momentum and energy of mixture and mass of vapour) with a drift-flux model to simulate the relative velocity between phases.

CEA began developing FLICA-OVAP in 2005, which builds on FLICA-4 and includes both a two-fluid two-phase flow model and a multi-field model that accounts for both liquid and vapour phases. The FLICA-OVAP platform incorporates both subchannel and CFD scale applications and can be coupled to multiscale and multiphysics applications.

FLICA was used as part of the EC NURESAFE project, and can be applied to two-phase flow in PWRs and BWRs.

SUBCHANFLOW: SUBCHANFLOW[11] is based on COBRA and was developed by the Institute for Neutron Physics and Reactor Technology (INR) at the Karlsruhe Institute of Technology (KIT).

SUBCHANFLOW is a single- and two-phase (three-equation mixture model) subchannel code that is used to simulate core geometries built from rectangular and hexagonal fuel bundles. Empirical correlations are then used for pressure drop, heat transfer coefficient and void generation with mixture equations for wall friction, wall heat flux and slip velocity. The solution progresses axially up the core, which limits the code to cases with positive flow up the core.

SUBCHANFLOW was used as part of the EC NURESAFE project, and can be coupled to a number of multiphysics codes. It includes properties for liquid metal, water, helium, and air, and so can be applied to PWRs, SFRs, LFRs and HTGRs.

VIPRE: VIPRE-01 (Versatile Internals and component Program for REactors) was originally developed from COBRA by Battelle Pacific Northwest Laboratories under sponsorship of the Electric Power Research Institute (EPRI). The VIPRE User Group[12] is run by Zachry Nuclear Engineering Inc., who manage the maintenance and development of the code for member organisations and EPRI.

[10] www.ne.ncsu.edu/rdfmg/cobra-tf
[11] www.inr.kit.edu/english/1008.php
[12] www.numerical.com/products/vipre/vipre_01.php

VIPRE-01 predicts the three-dimensional flow field and fuel rod temperatures for single- and two-phase flow in PWR and BWR cores for an interconnected array of channels. It solves the finite-difference equations assuming homogeneous equilibrium (empirical models are included for vapour/liquid slip and subcooled boiling) incompressible flow with no time step or channel size restrictions for stability.

4.5 Computational Fluid Dynamics (CFD) Analysis

CFD simulations provide predictions of 3D and time dependent velocity, pressure and temperature fields by numerical solution of fundamental partial differential equations, normally using meshes which divide the domain of interest into cells. Significant advances in numerical methods over the past four decades and the rapid increase in computing power has led to a number of robust, optimised and vendor supported CFD packages being available. These packages are very powerful tools, and provide access to a broad selection of general models (e.g. for turbulence, multiphase flow and heat transfer) and application specific modules.

CFD is a mature technology, developed and applied in all areas of engineering. However, due to the complexity of its practice, the reliability of CFD results is reliant on user expertise and validation for a given range of applications. Currently the use of CFD in the design of NPP is often limited by both the timescales/cost associated with performing the modelling and uncertainty in the final predictions. Overviews of the use of CFD in NTH analysis are available (for example D'Auria (2017), Chapter 12 and CSNI (2015), Section 4), and the main benefits and limitations of CFD methods are summarised in Table 4.3.

Benefits	Limitations
Capable of high resolution simulations of steady-state and transient flow, mixing and heat transfer	Models are computationally intensive, which limits temporal and spatial extent of models and number of design iterations
Well validated and used for single-phase heat transfer and mixing with no empirical correlations	Needs to be validated for each class of flow and the results can depend on user implementation (mesh, turbulence model and numerical scheme)
Capable and used to simulate coolants other than water for advanced reactor designs	Turbulence models were developed for water and gas, so high (molten salt) and low (liquid metal) Prandtl number fluids need to be considered carefully

Table 4.3: Summary of benefits and limitations of CFD methods.

Although a range of different CFD techniques have been developed, such as Finite-Element methods, Lattice-Boltzmann Method (LBM) and Smoothed-Particle Hydrodynamics (SPH), only the Finite Volume (FV) method is considered in these technical volumes because it is the predominant approach used currently in NPP analysis and industrial CFD codes. The FV method stems from the basic physics principle that rate of change of a quantity inside a control volume or mesh cell is equal to the sum of the fluxes across the bounding (flat) surfaces. Heat, mass or momentum flowing out of one cell is exactly equal to what goes into the neighbouring cell, such that these quantities are globally conserved even on coarse meshes.

There are a number of useful documents providing guidelines for the application of CFD, including CSNI (2015) and ERCOFTAC (2000). In addition, validation benchmarks and best practice development have been undertaken through numerous international efforts, including the NEA/CSNI and CFD for Nuclear Reactor Safety (CFD4NRS) series of workshops[13], ERCOFTAC[14] and NAFEMS[15].

4.5.1 Planning a CFD Analysis

Before undertaking a CFD analysis, it is important to understand and define the purpose of the analysis. It is essential to plan the analysis properly to ensure that it will meet the analysis objectives. This should ideally be undertaken as part of a PIRT process (Section 4.2), which will provide valuable insight into the V&V requirements.

Planning a CFD analysis is usually based on experience and should include the following key steps with practical guidance provided in ERCOFTAC (2000) and NAFEMS (2003):

1. Identify and define the analysis requirements (i.e. what are the objectives of the simulation and the key phenomena anticipated, validation basis for this application and the level of validation required, budget and timescales available and the output needed from the simulation).
2. The extent of the computational domain (fluid and solid regions) should be carefully selected to ensure that the inlet, outlet and any symmetry boundaries are suitably located and do not influence the solution (Section 4.5.2.1).
3. The mesh generation strategy should be planned in advance to ensure that the most important flow features and boundary layers are appropriately captured and resolved (Section 4.5.2).
4. The CFD software and physical models should be selected based on the fundamental flow physics and phenomena involved, such as steady/unsteady, incompressible/compressible, laminar/turbulent, natural/forced convection and conduction/thermal radiation.
5. The numerical discretisation schemes and other numerical settings should be selected based on the CFD software used (generally at least second order spatial and temporal discretisation with double precision applied).
6. The turbulence model and wall treatment (Section 4.5.3) should be selected based on the accuracy required and the time, computational resources and budget available.
7. The quality and accuracy of the geometry, material properties (including their dependence on temperature) and initial/transient boundary conditions need to be properly understood and documented in order to quantify the uncertainty in the predictions (Section 4.3.3).
8. The post-processing requirements and solution monitors should be identified and selected in order to demonstrate and ensure solution convergence in the results of interest.

It is also valuable to plan a staged approach to quality assurance at the outset of a CFD project (so that appropriate validation occurs as work progresses). This increases the likelihood of problems being identified at the earliest stage, enables them to be addressed with the minimum of rework, and improves interactions between originators and verifiers.

[13] www.oecd-nea.org/nsd/csni/cfd
[14] www.ercoftac.org
[15] www.nafems.org/community/working-groups/computational-fluid-dynamics

4.5.2 Mesh Generation

The mesh discretises the geometry into cells on which the flow is solved. The resolution (size) and type of mesh has a significant impact on the solution stability, accuracy, rate of convergence and computational resources required. Producing a good mesh in a time-efficient manner generally benefits from significant judgement and an intuitive insight into the flow field based on experience. Mesh generation remains one of the most important and labour intensive aspects of CFD analysis, although significant development is ongoing to automate the process. Poor mesh quality is one of the most common sources of errors in CFD solutions and can be difficult to resolve.

The mesh itself must be fine enough to adequately resolve the key flow and geometric features, though mesh requirements differ depending on the CFD approach adopted. The size of CFD models has increased significantly over the past 20 years due to the exponential growth in High Performance Computing (HPC) power, which in an R&D context has enabled a thermal hydraulic simulation of over 10 trillion cells to be solved (Gordon Bell 2013 HPC prize). However, the time taken and computational power required to solve these models is prohibitive, so most CFD models for industrial applications are less than 100 million cells at the time of writing.

The most suitable mesh generation approach will depend on the geometry and physics being modelled, the CFD solver being used and purpose of the analysis. The main mesh generation stages, approaches and their benefits are summarised below. Further background and guidance on mesh generation and quality is provided in ERCOFTAC (2000), NAFEMS (2019), Volume 2 (Section 3.4.2) and Volume 3 (Section 3.2.3).

4.5.2.1 Computational Domain

Many problems relating to nuclear thermal hydraulics involve large-scale systems, and modelling these in their entirety using CFD is unlikely to be feasible. It is therefore necessary to define the domain over which the CFD simulation will provide the best value (and which will therefore need to be meshed).

This is likely to encompass the area of interest, with extensions to avoid complex or recirculating flow (e.g. extending the domain up to a pipe containing more stable flow), ease the application of boundary conditions (e.g. a location for which information is available from an experiment or system code) and ensure boundary conditions do not have an undue influence on the results (e.g. by unphysically over-constraining the calculated flow field). In complex flows it may be appropriate to use sensitivity studies to investigate the impact of the boundary conditions on the results. Additional simplifications may be made where these do not influence any known phenomena of interest or prevent unforeseen phenomena from forming, such as:

- Using symmetry planes to enforce symmetry on the flow field and reduce the size of the computational domain. Whilst establishing the existence of symmetry in the geometry is usually straightforward, it might be difficult or impossible to establish in the flow *a priori*. Many flows are asymmetric despite geometric symmetry (e.g. vortex shedding behind a symmetric bluff body). Large homogeneous pools with uniform thermal boundary conditions may be suitable, but care is needed for unsteady, complex or buoyancy-driven flows. Literature may provide useful guidance, or an exploratory computation without the use of symmetry planes could be considered to check the approach.

- Using either translational or rotational periodicity constraints. Whilst this is commonly used to model large or repeating domains (such as very long pipe runs) this will restrict the size of the flow structures which can develop (and may prevent such features from developing at all if the domain is too small). This is especially pertinent in natural convection flow fields, and may prevent plumes and recirculating flows developing correctly. Another example is the flow across tube bundles, such as occur in heat exchangers or boilers.

4.5.2.2 Geometry

Once the computational domain is defined, the relevant geometry can be generated. This may be created top-down (importing geometry and then splitting it into meshable sections) or bottom-up (building geometry from points, lines and faces). Meshing packages often have built in geometry creation tools or can link to Computer Aided Design (CAD) packages to allow meshing directly from the CAD geometry.

In contrast, during a concept design process, the detailed geometry of the components may not be fully understood or known, and the CFD model can be used to understand and optimise the geometry to support the design process. This can be done using either mesh movement in the CFD solver or automatic remeshing within the CAD/mesh generator package.

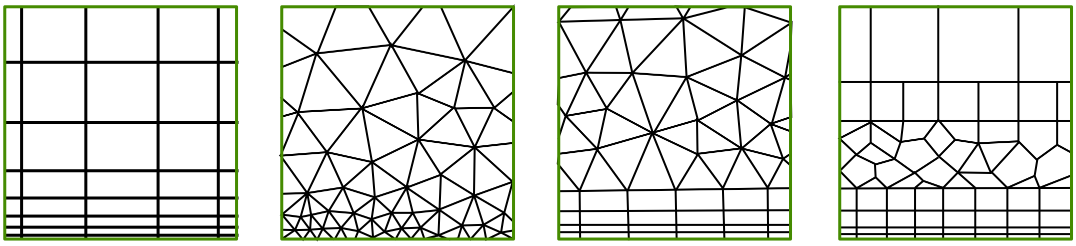
Modelling a geometry that is different from what is actually built can significantly affect the accuracy of predictions for complex flows that are common in NTH. Specific issues to consider include:

- Incomplete or inaccurate geometrical information (e.g. insulation missing from drawings or CAD, unclear old drawings, modifications or as-built details not being included in drawings) which may need to be corrected, or considered for uncertainty analysis (Volume 4).
- Problems with the CAD geometry itself (e.g. poorly connected geometry, slivers, gaps) generally need to be cleaned before meshing, and this is normally best performed by a CFD user who can operate CAD or CFD clean-up tools, not vice versa.
- Differences between the drawn and operational geometry may need to be considered. Most drawings show what should be manufactured, and therefore show the geometry in cold conditions with nominal dimensions and tolerances. However, modelling work is normally analysing the component in service, when it is an ‘as built’ geometry (i.e. dimensions are unlikely to exactly match nominal dimensions due to manufacture inaccuracy) and may also be hot (and may therefore have changed shape due to thermal expansion). These differences may be important in situations where these changes from drawn to as-built geometry have a significant impact on the geometry of the flow.
- Considering the level of geometric detail needed. Due to the large range of different geometric sizes within a NPP (e.g. comparing the size of individual joints or welds to the lower plenum of a reactor) it is often necessary to simplify the geometry of a CFD model compared to the real geometry or represent the impact of it using a porous media approach (e.g. screens and pebble beds). Different levels of simplification may be appropriate in different parts of a domain. This simplification process needs to be carefully understood to ensure that any significant geometrical features are not removed, and the impact of the simplification process and geometrical tolerance should be assessed as part of the uncertainty quantification (Section 4.3.3).

Problems with transferring CAD data into meshing tools (e.g. ensuring the units/scale is correct and the import process does not result in a reduction of geometric accuracy) should also be considered.

4.5.2.3 Mesh Types

There are a number of different approaches for meshing a geometry, the main types of mesh used in industry are illustrated in Figure 4.5 and summarised below. Designing meshes is discussed in more detail in Volume 2 (Section 3.4.2) and Volume 3 (Section 3.2.3).



(a) Structured (b) Unstructured (c) Near-Wall Prismatic (d) Hybrid

Figure 4.5: Four different mesh types used in CFD.

Structured Mesh: In a multi-block structured mesh, the domain is split into a number of structured mesh regions or 'blocks', within which cells can be indexed by i, j, k coordinates. Orthogonal, structured grids aligned with the flow can provide higher quality solutions (reduced numerical diffusion) with fewer cells than unstructured grids, and solution times can be shorter relative to unstructured grids if a suitable solver is employed. However, the time taken to generate a structured mesh for industrial geometries is often prohibitive, and can lead to increased geometrical simplification and undesirable mesh features, such as flow across high aspect ratio cells, sharp changes in grid direction and non-orthogonal or skewed cells that can impact solution accuracy. Additionally, in many industrial cases it will not be possible to develop a structured mesh that is aligned with the flow field, reducing the benefit of using a structured grid.

Unstructured Mesh: These meshes use tetrahedral, hexahedral and polyhedral cells arranged in an arbitrary fashion with no constraints on cell layout. This significantly eases meshing of complex geometries, but means that the flow is generally not aligned with the mesh:

- Tetrahedral meshes can be automatically generated to fit complex geometry, but require significantly more cells compared to structured grids to achieve the same level of solution accuracy (at least a factor of 5). In addition, since tetrahedral cells are not aligned with the flow, it can lead to increased numerical diffusion especially if the cells are stretched or lie in one plane.
- Unstructured hexahedral meshes are harder to generate, although require fewer cells as the grid can be stretched and aligned to the flow direction with well defined normals. Hexcore or cut-cell meshes automatically generate high quality hexahedral elements with the cells subdivided hierarchically until they reach the required level of refinement. This creates polyhedral cells at the interfaces between refinement levels and requires some sort of interface region or trimming to conform with the surfaces of the geometry.

- Polyhedral meshes can be automatically generated by merging together cells in a tetrahedral mesh. This overcomes many of the disadvantages of tetrahedra, as polyhedra have many neighbours, which improves information propagation and reduces the overall number of faces. This means that fewer cells are required compared to tetrahedra for the same solution accuracy.

Near Wall Prismatic Mesh: In order to accurately predict the flows near surfaces ('at walls') it is important to properly resolve the velocity and temperature gradients near walls (discussed further in Volume 2). This may be achieved using a 'prismatic' (or 'inflation layer', or 'boundary layer') mesh that is grown from the surface mesh in layers. In regions where the flow is aligned with the surface (i.e. away from separating or recirculating regions) high aspect ratio cells can be used to significantly reduce the overall cell count, as the gradients in flow variables normal to the wall are likely to be much larger than the gradients along the wall. It is important to use an appropriate first cell height and expansion into the flow field. Prismatic and near-wall meshing is considered in Volume 2 (Section 3.4.2).

Hybrid Mesh: A hybrid mesh uses any combination of tetrahedra, hexahedra and polyhedra. This could include multi-block structured grids in simple geometry regions and unstructured mesh in more complex areas. For example, some mesh packages generate an unstructured mesh using tetrahedra to join a prismatic boundary layer to an unstructured mesh in the free-stream.

At the interfaces between each mesh region it is preferable to use a conformal interface (i.e. the mesh faces are identical and fully connected). However, it is also possible to use non-conformal interfaces, where the nodes/faces lie on the same surface, but are not connected. This can cause errors in the solution across the interface (especially if there are large gradients in flow variables combined with a large difference in the face size from one side to the other) and so should be avoided if possible.

4.5.2.4 Mesh Quality

Since the solution quality depends on the quality of the mesh, it is important to ensure that the mesh is suitably refined to resolve the key flow features, such as regions of high shear or large thermal gradient. The quality of the mesh should be checked:

- Using overall measures of mesh quality within meshing packages (often parameters such as skewness, aspect ratio, orthogonality, expansion/smoothness and perhaps even one overall 'quality parameter'). Acceptable ranges of these parameters are often specific to the meshing and CFD software being used, so software specific documentation should be consulted. Further general guidance is provided in CSNI (2015).
- Using mesh sensitivity studies for Reynolds-Averaged Navier-Stokes (RANS), or resolution checks for Large Eddy Simulation (LES) (Section 4.5.3 and Volume 2, Section 3.4.2).
- Regardless of quality metrics and mesh sensitivity studies, **when a flow solution is available, the predicted flow should be visualised on top of the mesh and reviewed by an experienced CFD user** who can consider whether or not the mesh is really capable of appropriately resolving gradients that may (or perhaps should) exist within the predicted flow field.

In some situations (particularly predictions of transient or unsteady flow phenomena) it may be necessary to use the CFD code to change the mesh based on the flow solution, to capture details of the flow in adequate detail. Some CFD packages can automatically refine (or adapt) the mesh using criteria based on flow variables. Whilst these appear attractive and may (in principle) provide efficient mesh refinement, the algorithms are highly software and case dependent and may not always result in a better mesh. For example, if the initial mesh is too coarse to predict flow features like plumes, the adapted mesh may not either. Further, success may be dependent on the right adaption criteria being used, and the process may be difficult to repeat precisely if the initial mesh is changed. In any case, the adapted mesh should be checked in the same way as any other mesh.

Adjustments are almost always required to correct problems. Meshes are likely to be a compromise between detail, efficiency and the speed/ease at which the mesh can be generated. As discussed in Section 4.3, it is important to undertake a mesh sensitivity study for any new application in order to understand the impact of the mesh resolution on the solution parameters of interest, and ensure that the solution results are appropriately independent of the mesh.

4.5.3 CFD Approaches

As CFD models the three-dimensional flow field, the significant effects of turbulence must be considered. This section introduces the main CFD approaches, which primarily differ as a result of their different approaches to predicting turbulence. The subject of turbulence and CFD is vast (even if purely restricted to NTH). Further details on modelling turbulence and laminar-to-turbulent transition are provided in Volume 3, because these aspects can be particularly important for passive cooling systems. Figure 4.6 illustrates the different approaches to CFD using the kinetic energy spectrum of turbulence (introduced in Volume 3, Section 2.2.3).

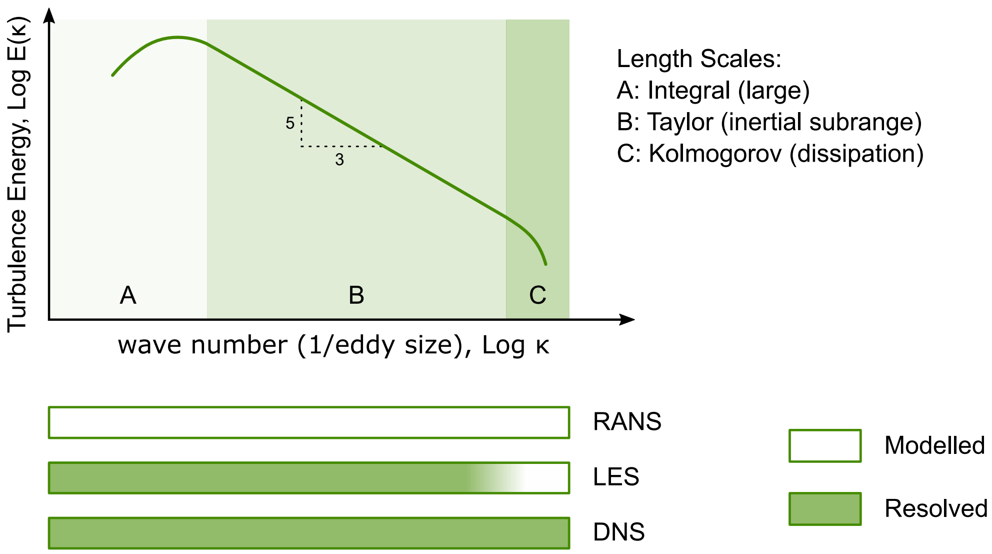


Figure 4.6: Turbulence energy spectrum and main CFD flow modelling approaches.

In essence, DNS seeks to resolve all turbulence (A, B and C), RANS models all turbulence, and LES seeks to resolve the integral and Taylor scales (A and B) and model the Kolmogorov scales. A more detailed overview of each approach is provided below (further detail is provided in D'Auria, 2017). However, there is a trade-off between increased fidelity and computational cost (Figure 4.1).

- DNS is predominantly used by academia to generate benchmark solutions for CFD model validation on simple geometries at low Reynolds numbers.
- RANS techniques currently account for the vast majority of CFD work performed in industry, as it provides useful results with practical computational expense.
- LES use in industry is increasing for detailed investigation of the flow in specific areas of a system, but it is generally limited to small regions and short transients.
- Hybrid methods combine RANS and LES within the same model, so offer increased fidelity over RANS, but are less computationally intensive than LES.

It is recommended that software-specific documentation (e.g. user guides) is consulted for detailed information on the availability, configuration and use of the different modelling approaches and their limitations, as mesh requirements, terms and implementations used can vary between tools.

4.5.3.1 DNS

Turbulent flows are completely described by the Navier-Stokes equations and can be directly solved using DNS. If done correctly, this will resolve all of the length and timescales within the flow and the results can be regarded as a true representation of the real flow system. Whilst regarded as highly accurate, this analysis is extremely computationally expensive, especially at higher Re[16] (or Gr for buoyant flows). DNS is therefore predominately used to study low Re or Gr flows in simple configurations, in order to develop fundamental understanding of these basic flows and obtain extremely detailed data that can be used to improve the models used by other CFD methods (a review of the use of DNS is provided in Moin and Mahesh, 1998). DNS is unsteady and 3D (i.e. predicted flow parameters always vary with time).

The use of DNS is likely to increase gradually as computational power increases, particularly in academia. However, its application to industrial problems for the foreseeable future is likely to be restricted to improving understanding of specific local phenomena, and providing high-fidelity data which can aid development of more industrially useful CFD methods.

4.5.3.2 LES

LES attempts to resolve only the larger (more energetic) scales in a flow, substantially reducing computational expense compared to DNS. To achieve this, a ‘filter’ is applied to the Navier-Stokes equations, removing fluctuations smaller than a given scale. The larger (more energetic) turbulent length scales in a flow are then resolved using the computational mesh, while the effect of the smaller (unresolved) motions is modelled using a Sub-Grid-Scale (SGS) model (which may be similar to the Eddy Viscosity Models (EVMs) used in RANS methods). The basis for this is that the larger, more energetic turbulent motions are likely to have the largest impact on flow behaviour and are more likely to be anisotropic (Wagner *et al.*, 2007). Like DNS, LES is transient and 3D, even if the time-averaged flow is steady and 2D[17]. Common SGS models include Wall-Adapting Local Eddy-Viscosity (WALE) and Dynamic Smagorinsky, although the choice of SGS will depend on the particular situation and the specific software tools being used.

[16] The number of mesh points required to resolve the complete spectrum of turbulence increases as $Re^{9/4}$.
[17] Some attempts have been made to adapt the approach to 2D in the literature, for example Bouris and Bergeles (1999).

Directly resolving the large turbulence scales is a key benefit of LES over RANS methods; the main features of the flow are usually well represented and the SGS models may have less impact on the accuracy of the simulation than the turbulence models used in RANS methods. Used correctly, LES can provide significantly more information, has the potential to be more accurate and offers the prospect of tackling turbulence problems beyond the scope of RANS models. However, the accuracy of LES is more strongly dependent on mesh quality and numerical schemes than RANS. Poor LES work may well produce worse results than good RANS work (Holgate *et al.*, 2019).

In particular, it is important to ensure that the energy containing eddies are resolved, as otherwise the assumptions underpinning LES may not be valid. This generally requires at least 80% of the turbulent kinetic energy in the flow to be resolved (Pope, 2000). A suitable mesh resolution is required to achieve this, so the mesh spacing is likely to be based on precursor flow solutions (discussed in Volume 2, Section 3.4.2). Acquiring suitably accurate inflow turbulence information for LES can be difficult in industrial contexts, which may be particularly problematic for transitional flows.

Resolving only part of the turbulence spectrum significantly reduces the computational cost of LES compared to DNS but generally results in significantly increased cost compared to RANS. The computational cost of LES (which scales with Re and Gr, although less strongly than DNS) remains too high for most industrial problems, particularly so for high Re boundary layer flows. This is because a small near-wall mesh sizing (y^+) is required to model the inner-most portion of the momentum boundary layer, leading to similarly small mesh sizes in both transverse directions (x^+ and z^+). As a result of this, hybrid methods such as Wall Modeled Large Eddy Simulation (WMLES) are often used to reduce the expense of modelling near-wall flows (or higher Re flows may be unresolved near walls). Similar constraints exist for heat transfer (see Volume 3, Section 3.2).

The use of LES is slowly increasing in industry. It is most likely to be used to perform detailed investigation of the flow in specific areas of a system, particularly free shear flows (such as jets, separated and swirling flows) or where detailed predictions of turbulence behaviour are important (such as unsteady mixing and generation of cavitation or acoustics from turbulence). LES is also used to provide data to compare with RANS methods (Ammour *et al.*, 2013). Within the context of a graded approach for NTH, LES is most likely to be used in areas of high safety significance or commercial impact.

4.5.3.3 **RANS**

RANS decomposes the instantaneous flow variables in the Navier-Stokes equations into mean and fluctuating parts (e.g. $U = \overline{U} + U'$) and then applies a statistical averaging procedure. The result is a set of equations governing the *mean* flow which are very similar in form to the Navier-Stokes equations but have an additional term, called the Reynolds stress tensor[18]. The equations are averaged, so no turbulent flow structures are resolved using RANS. All the effects of turbulence on the mean flow are incorporated using a model for the Reynolds stresses. There are two main approaches to doing this:

1. Assume there is a relationship between the Reynolds stresses and the mean flow strains through a turbulent (or eddy) viscosity. This is the Boussinesq eddy-viscosity hypothesis,

[18] Also called the ‘Reynolds stresses’, ‘turbulence stresses’ or ‘second moments’.

and results in Eddy Viscosity Models (EVMs). The most commonly used EVMs assume a linear relationship, but other non-linear approaches have been developed. Simple algebraic models exist which are a function only of the instantaneous mean flow; these are the simplest and quickest to solve, but are limited by narrow ranges of applicability. Much more commonly, transport equations which allow the history of flow development to also influence the eddy viscosity are used. These may use one equation (e.g. Spalart-Allmaras model), two equations (e.g. k - ε or k - ω models) or more to calculate the turbulent viscosity.

2. Solve modelled transport equations for the Reynolds stresses themselves, which are often called Reynolds Stress Models (RSMs). While typically more difficult than using EVM, this approach enables modelling of anisotropic turbulence effects, which may be significant in, for example, buoyancy and boundary layer flows.

Turbulence, transition and near-wall modelling using RANS are considered further in Volume 3 (Section 3.2.6).

A key aspect of RANS methods is the approach to modelling the flow near walls to cater for the no-slip condition at the wall itself and the impact of the wall on the adjacent flow (e.g. damping turbulence). In NTH, this region is often important, since wall heat transfer is strongly affected by the near-wall flow and turbulence behaviour.

The turbulent boundary layer is often subdivided into three near-wall regions, which are categorised by the non-dimensional wall distance, y^+, as illustrated in Figure 4.7, where $\kappa = 0.41$ (von Kármán constant) and $C^+ = 5.0$ for a smooth wall (Schlichting and Gersten, 2017).

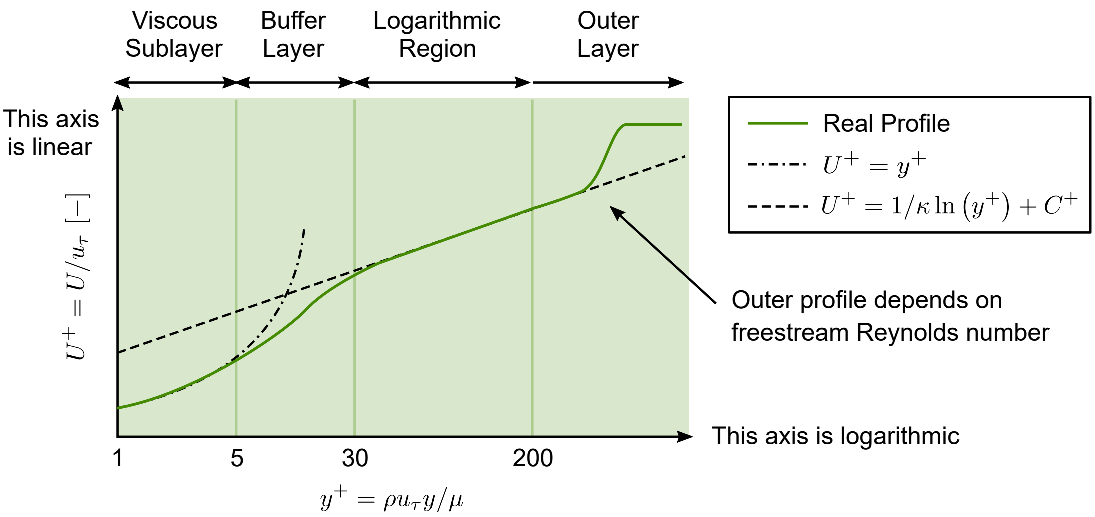


Figure 4.7: Near-wall regions of a turbulent boundary layer.

While the detailed terms and implementation may be specific to the modelling software and turbulence model, the following approaches are commonly available:

Wall resolving: The mesh is refined to the wall so that the viscous sublayer ($y^+ < 5$) is resolved, ideally the height of the first cell at the wall should achieve $y^+ \approx 1$, and so is often called a 'low-*Re*' method as the turbulent Reynolds number (Re_t) is low in the viscous sublayer. This method can be used for any bulk (free-stream or pipe) *Re*, although at high *Re* the near-wall mesh spacing will need to be very small.

Standard wall functions: This 'high-*Re*' method uses a calibrated 'wall function' to model the logarithmic region of the boundary layer, acting as a bridge between the wall and the bulk flow (as described in Launder and Spalding, 1974). The first cell should be placed in the logarithmic region ($y^+ > 30$), and so an overly fine near-wall mesh should be avoided. This can significantly reduce the mesh size, but should not generally be used in complex flows or natural convection if accurate predictions of heat transfer are required.

Enhanced wall functions: In real engineering geometries it is often necessary to use a mixture of both of the above approaches to focus the mesh on areas of interest. As a result, CFD software often includes an approach that combines a wall resolving model with 'enhanced' or 'scalable' wall functions that use a single law of the wall for the entire wall region.

Advanced wall functions: The wall function approaches above may also be extended to manage non-equilibrium conditions (such as exist in complex flows like impingement or separations). While the overall approach is the same, locally 1D analytical or numerical solutions of the boundary-layer forms of the transport equations remove the need to impose the log-law and other assumptions involved in the original wall function approach.

Whatever approach to near-wall modelling is taken, it is often necessary to adjust the mesh based on initial flow solutions. Near-wall meshing for RANS is considered in more detail in Volume 2 (Section 3.4.2).

Classic RANS methods generally assume that the flow is steady and therefore do not model transient behaviour. By contrast, Unsteady Reynolds-Averaged Navier-Stokes (URANS) methods retain the time derivative term in the governing equations, and can therefore capture unsteady behaviour in the mean flow variables. It is noted that:

- An appropriate physical time step must be defined. This can require engineering judgement about the nature of the flow fields being investigated, and a poor choice of physical time step may not be obvious from numerical behaviour or predicted flow fields. Selecting a time step is considered in Volume 3 (Section 3.2.6).
- URANS assumes there is scale separation between the 'mean flow unsteadiness' (at large scale) and the largest turbulent eddies (at small scale). The timescales of flow unsteadiness (which feed into decisions on the physical time step to use) are therefore assumed to be larger than the timescale associated with turbulence (which is modelled by a turbulence model). For many flow fields this assumption may not be correct everywhere in the flow field.
- A time-averaged URANS flow field may or may not be the same as a steady-state RANS solution, depending on the physics of the problem being solved.
- As for LES and DNS, unsteady flow field predictions can be made by URANS. This unsteadiness may not emerge automatically; it may be spontaneously initiated (e.g. by numerical discretisation errors), or may need to be initiated by a user-supplied perturbation to a physical property of the flow or unsteady boundary conditions.
- The frequency content of separated flows is likely to be very restricted, for example downstream of a bluff body, coherent vortex shedding may be incorrectly predicted at high *Re* because of the absence of smaller-scale chaotic features that have been removed by the RANS method.

RANS and URANS are extremely valuable tools and account for the vast majority of CFD work performed in industry, as they can provide useful results with practical computational expense. RANS is likely to feature in most CFD studies, not least because a RANS study is often used to provide flow field information ahead of using more detailed approaches like LES. RANS modelling work can also vary particularly dramatically in complexity (and cost), from simple steady-state calculations to detailed URANS simulations of plant transients with complex physics.

URANS will, for many years to come, be the only practical CFD method for predicting long duration problems such as long reactor shut-down transients (which may extend over several hours), especially those with conjugate heat transfer to structures with significant stored thermal energy (Volume 2, Section 3.4.5). Within the context of a graded approach for NTH, RANS is therefore likely to be used in most situations where CFD is needed, with the complexity of the modelling work reflecting the complexity of the case and the safety significance and commercial impact associated with the work.

Two-equation turbulence models (particularly k-ε or k-ω Shear Stress Transport (SST), introduced in Launder and Spalding, 1974 and Menter, 1994 respectively) are the most commonly used turbulence models, as they are robust and well understood. Enhanced wall functions are commonly used, with the mesh refined to wall-resolving levels in areas of interest where there are complex flow fields or important heat transfer. URANS is used where appropriate, given the additional computational expense and technical work (which is case-dependent).

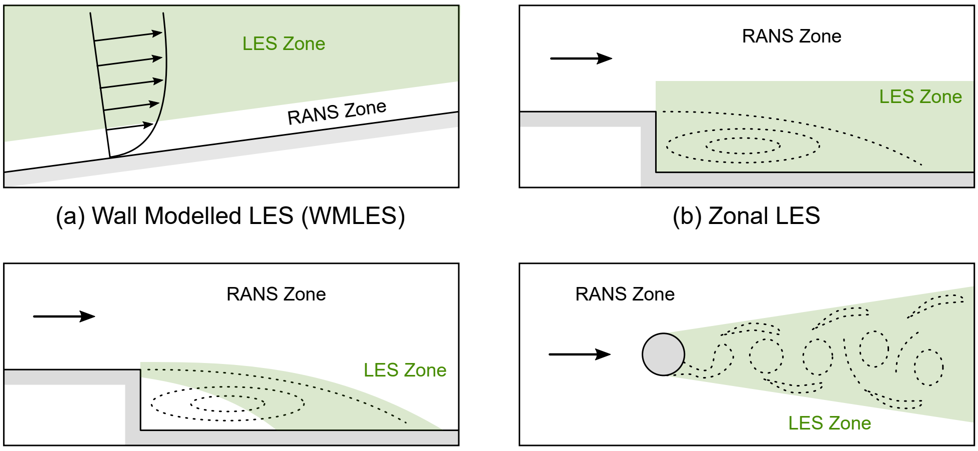
The simplifications fundamental to RANS methods are significant and can pose challenges. As a result, higher fidelity models may well be used to provide comparative data where the performance of RANS methods is uncertain or not considered appropriate.

4.5.3.4 Hybrid Methods

As noted above, modelling wall-bounded or free-stream flows is much less expensive for RANS approaches compared with LES. By contrast, free-shear flows can be solved well with LES with practical expense, but are a weakness of RANS. As a result, a number of hybrid methods have been developed that combine RANS and LES methods within the same model, to take advantages of both. Only the range of scales being modelled by the turbulence or SGS model is different, and models can therefore be developed to switch between the two approaches.

A wide variety of approaches have been proposed, ranging from ones where the user prescribes the switching location, to ones in which the two treatments are blended to give a smooth transition. Some common models are illustrated in Figure 4.8 and include:

- WMLES, where the flow very near the wall is modelled using RANS, but the outer parts of the boundary layer and the rest of the domain uses LES.
- Zonal or Embedded LES, where the domain is split manually by the user into RANS and LES or WMLES zones (generally during geometry and meshing activities).
- Detached Eddy Simulation (DES) and its various derivatives, where the method switches between RANS and LES based on the local mesh resolution and wall distance.
- Scale-Adaptive Simulation (SAS), where the flow is generally solved using RANS, but switches to scale resolving mode in large unstable areas of flow (such as separated areas).



(a) Wall Modelled LES (WMLES) (b) Zonal LES (c) Detached Eddy Simulation (d) Scale Adaptive Simulation

Figure 4.8: The main hybrid methods.

No single hybrid method is likely to be appropriate for all applications, as each approach has different strengths and weaknesses, may require different meshes, have different settings to use, and have different subtleties in results interpretation. The best approach to use is likely to be application and tool specific. It is therefore necessary to obtain a detailed understanding of the approaches available within a given tool before using them (D'Auria, 2017 and Menter, 2015).

A Partially-Averaged Navier-Stokes (PANS) model, like hybrid methods, has also been developed to resolve large scale structures at reasonable computational expense (Girimaji and Abdol-Hamid, 2005). This uses a suite of turbulence closure models for various modelled-to-resolved scale ratios ranging from RANS to DNS. The flow resolution in PANS is controlled by suitably specifying the unresolved-to-total ratio of kinetic energy, and so can be considered as an LES method with an implicit filter.

Whichever approach is chosen, it is important to understand which modelling method is being used at different locations in the computational domain. For example, whether LES resolved turbulence must be generated based on flow from upstream RANS areas and how this is achieved. Within a graded approach for NTH, hybrid methods sit between the points made above for RANS and LES.

4.5.4 Codes

The principal CFD codes currently used in the nuclear industry worldwide are CFX, Fluent, Code_Saturne, OpenFOAM, Nek5000 and Star-CCM+. Commercial licenses are available from ANSYS and Siemens, while Code_Saturne, Nek5000 and OpenFOAM are freely available. All of these codes are highly scalable and have a wide range of capabilities that allow them to model flow, turbulence, heat transfer and reactions for industrial applications. The background and history of each code is summarised below.

CFX: The United Kingdom Atomic Energy Authority (UKAEA) in Harwell developed an internal structured CFD code for use on nuclear applications. This was released in the late 1980s as a general-purpose CFD package called FLOW3D and then renamed CFX-4. AEA Technology acquired TASCflow in 1997 from Canadian company Advanced Scientific Computing

(ASC) and used it as the basis for CFX-5. In 2003 CFX was acquired by ANSYS Inc[19]. CFX uses a vertex-centred approach, which means that it can only handle hexahedral, wedge, pyramid and tetrahedral mesh topologies, and can be customised using the CFX Expression Language (CEL). The ANSYS quality system meets the US NRC rules and regulations for quality assurance in 10 CFR 50 Appendix B, 10 CFR 21 and the ASME NQA-1 quality standard.

Fluent: Development began at the University of Sheffield in the UK in the 1970s. The researchers partnered with Creare Inc. in the US and the first version of Fluent was released in 1983 as a general-purpose CFD package. Development and use of Fluent continued to grow internationally and the code was acquired in 2006 by ANSYS Inc[20]. Fluent uses a cell-centred approach and is able to handle any type of cell or grid structure including polyhedral and hierarchical hexahedral meshes, and can be customised using User Defined Functions (UDF). The ANSYS quality system meets the US NRC rules and regulations for quality assurance in 10 CFR 50 Appendix B, 10 CFR 21 and the ASME NQA-1 quality standard.

Code_Saturne: This is a 3D unstructured finite volume code developed by EDF R&D since 1997. It was made open source in 2007 and is freely available under a General Public License[21]. Code_Saturne uses a cell-centred approach and is able to handle any type of cell or grid structure and the source code is openly available to allow maximum flexibility for user customisation. It is used as part of the EC Partnership for Advanced Computing in Europe (PRACE) project.

OpenFOAM: OpenFOAM (open source Field Operation And Manipulation) was created in 1989 at Imperial College London under the name 'FOAM', and was released open source in 2004 by OpenCFD Ltd. under a General Public License. In 2011, OpenCFD was acquired by SGI Corp. and was then sold to ESI Group in 2012, although the copyright belongs to the OpenFOAM Foundation[22] to ensure that it remains exclusively open source software. OpenFOAM is a C++ toolbox for the development of customised numerical solvers. It uses a cell-centred approach and is able to handle any type of cell or grid structure and the source code is openly available to allow maximum flexibility for user customisation.

Nek5000: Nek5000 is a high-order open source CFD solver that has been developed by ANL[23]. The spatial discretisation is based on the Spectral Element Method (SEM), which is a high-order weighted residual technique similar to the finite element method. It is designed to simulate laminar, transitional, and turbulent incompressible or low Mach number flows with heat transfer and species transport, predominantly using a LES approach. Nek5000 is a 'MOOSE-wrapped' application, and so can be easily coupled to the other NEAMS codes within the open source MOOSE framework.

STAR-CCM+: Star-CD was released in 1987 by CD-Adapco, a partnership between Adapco and Computational Dynamics Ltd., which was formed by researchers at Imperial College London in the UK. In 2004, CD-Adapco rewrote their CFD code and released Star-CCM+ to allow the use of polyhedral meshes and the simultaneous solution of fluid flow and heat transfer prob-

[19] www.ansys.com/products/fluids/ansys-cfx
[20] www.ansys.com/products/fluids/ansys-fluent
[21] www.code-saturne.org
[22] www.openfoam.org
[23] nek5000.mcs.anl.gov

lems. In 2016 CD-Adapco was acquired by Siemens PLM[24]. Star-CCM+ uses a cell-centred approach and is able to handle any type of cell or grid structure and can be customised using user field functions. It was used as part of the US DOE CASL project and complies with the ASME NQA-1 quality standard.

4.6 Experimental Methods

Thermal hydraulics has always played a central role in the design, safety assessment, operation, and maintenance of NPPs, and experiments have long been used to discover, understand and predict phenomena and to help create models. Experimental work therefore formed the basis of safety evaluation for nuclear thermal hydraulics and underpinned the development of computational tools, and the present strong connection between experimental work and the development and qualification/validation of computer codes (Section 4.1) is likely to continue.

In this section, the benefits of close working between experimental and modelling teams and planning a validation experiment is discussed, before the main types of tests (basic, SETs and IETs, as introduced in Section 4.3.2) are summarised. Although these categories of experiments were developed for PWRs, they are applicable to all reactor types. This is followed by an overview of scaling considerations and the facilities used worldwide. The main benefits and limitations of experimental methods are summarised in Table 4.4. Further discussion on experimental methods is provided in Volume 2 (Section 3.5) and Volume 3 (Section 3.3).

Benefits	Limitations
Provide real-world demonstration of NTH phenomena at reactor conditions for licensing purposes	Large scale experimental tests are expensive (space, infrastructure and associated permits) and need to be scaled appropriately to match reactor conditions
Measurement data to validate system, subchannel and CFD methods	Data is limited by instrumentation to point measurements or small measurement planes
Capable and used to test other coolants and support advanced reactor designs	Experimental tests with molten salt and liquid metal are more complicated as the coolants are harder to handle and sometimes require new instrumentation techniques

Table 4.4: Summary of benefits and limitations of experimental methods.

4.6.1 Planning and Working Together

The need for good two-way communication between experimental and modelling teams is stronger than ever. In particular, experiments for validating models should be carefully designed to ensure that they capture the physics of interest and generate the required measurements to the appropriate level of accuracy (CSNI, 2015), otherwise they can suffer from:

- Lack of appropriate measurements in the right locations.
- Insufficient number of measurements.
- Lack of well-defined initial and boundary conditions.
- Lack of understanding and quantification of experimental uncertainty.

[24] www.plm.automation.siemens.com/global/en/products/simcenter/STAR-CCM.html

Both the CSNI Working Group on Analysis and Management of Accidents (WGAMA) CFD Task Group and the SIgnificant Light and Heavy Water Reactor Thermal Hydraulic Experiments Network for the Consistent Exploitation of the Data (SILENCE) network have identified the need to establish more detailed guidelines for future CFD-grade experiments. Some initial guidance is provided in Bestion *et al.* (2019) with more detailed information expected to be published in a NEA/CSNI report soon.

The overall objective of a CFD-grade experiment is to provide the lowest values of δ_{input} (error in input data, e.g. boundary and initial conditions and fluid/solid properties) and δ_D (error in experimental data, i.e. difference between experimental and true value). This often requires highly accurate measurements on well-defined, well-instrumented test facilities. Bestion *et al.* (2019) highlights the importance of the following factors to achieving a CFD-grade experiment:

- Collaboration and close working between experimentalists and code users from the beginning of the concept design to the end of the analysis of its results.
- Clear specification of the fluid/solid volume of interest, inlet/outlet fluid surfaces, solid-fluid boundaries and the external solid boundaries with sufficient accuracy, so that they can be used as input data.
- The number, location and type of measurement instrumentation is important for CFD model validation, but compromises are often necessary.
- Reliable evaluation of measurement uncertainty is difficult, but essential to support CFD code validation and enable the model uncertainty to be determined.

A multi-step process is proposed in Bestion *et al.* (2019) to support the design of a CFD-grade experiment, as illustrated in Figure 4.9 and summarised below.

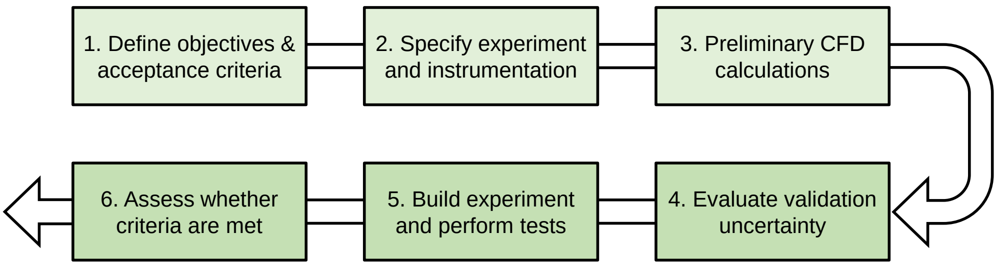


Figure 4.9: Proposed steps to undertake a CFD-grade experiment (Bestion *et al.*, 2019).

1. Define Objectives: It is important to define clear objectives for the experiment in terms of the main physical processes of interest (PIRT), types of model to be validated, safety demonstration methodology (e.g. BEPU and the FOMs / acceptance criteria).

2. Specify Experiment: Experimentalists and code users need to discuss and agree the basic design of the facility, test conditions and measurement techniques to confirm that the modelling needs are experimentally achievable. This includes defining the test conditions and extent of the model/test section domain.

3. Preliminary CFD: Preliminary CFD calculations are useful at this stage to understand the sensitivity to boundary and initial conditions, inlet and outlet boundary conditions, finalise the test section geometry and identify the best measurement locations, which can feed back into the experiment specification.

4. Evaluate Uncertainty: The global validation uncertainty in the defined FOM should be calculated based on the expected accuracy of the measured values, the uncertainty in all of the boundary and initial conditions and numerical uncertainty from preliminary CFD calculations. If the acceptance criteria are not met, then the experiment design should be reviewed.

5. Perform Experiment: The test facility is then built according to detailed design specification and the tests are performed. It is essential that repeatability tests are carried out to check and evaluate the uncertainties in the FOM.

6. Check Experiment Suitability: Often, the actual conditions observed in the experiment do not exactly match the defined test conditions. Therefore, it is important to check the actual experimental uncertainties to prove the quality of the experimental work and ensure that it meets the original acceptance criteria.

Where work must be performed sequentially, the study should be reported as clearly, accurately and comprehensively as possible (e.g. providing detailed geometrical descriptions, making CAD geometry available, reporting boundary conditions, test conditions and material properties).

4.6.2 Basic Tests

These experiments address fundamental phenomena, such as pressure drop, single- and two-phase flow, fluid mixing, heat transfer, boiling, condensation and freezing, critical flow, pressure-wave propagation, and complex phenomena due to combinations of fundamental processes like flooding and counter-current flow limitation.

Basic tests aim to understand the phenomena under simple, steady-state or prescribed transient boundary conditions, sometimes with less reference to actual reactor conditions, including those expected in accidents. Rather, basic tests may reveal information essential for developing models and provide data that can be used to improve confidence in modelling tools.

4.6.3 Separate Effect Tests (SETs)

Validation of codes and models can be more easily addressed when local phenomena are separated from the whole system response where various phenomena interact. SETs vary in scale, potentially addressing multiple phenomena occurring in reactor components and sub-assemblies. They are typically performed on individual, representative test pieces under more complicated boundary conditions than basic tests. Local phenomena suitable for SETs are expected to occur in primary thermal hydraulic regions or zones of reactors and individual components, such as fuel assemblies and heat exchanger channels. Frazer-Nash (2019) summarises a wide range of the thermal hydraulic SET facilities available worldwide for both water and other coolants.

An internationally agreed SET validation matrix for thermal hydraulic system codes was developed by the CSNI (1994) for LWRs. This matrix collected together the best sets of openly available test data for code validation, assessment and improvement, including quantitative assessment of uncertainties in the modelling of individual phenomena. In addition, it was intended to record information that was generated around the world, so that it is more accessible.

This includes the identification and characterisation of the thermal hydraulic phenomena relevant to two-phase flow related to LOCAs and thermal hydraulic transients in LWRs. The SET matrix was developed by selecting individual tests from the test facilities relevant to each thermal hydraulic phenomenon so that it could be used to assess transient thermal hydraulic system codes. The test facilities in the SET validation matrix are listed in the The International Experimental Thermal HYdraulics Systems (TIETHYS) database and the test data is available from the NEA databank[25].

4.6.4 Integral Effect Tests (IETs)

IET facilities are designed and operated to reproduce aspects of a reference reactor's performance and behaviour, including the main safety systems. These are large test facilities that address the performance of an entire reactor system in support of a particular reactor development programme that are performed on a scaled representation of the primary and/or secondary circuit(s) under reactor transient and accident conditions. IETs also provide data to improve confidence in predictive models and substantiate the design. Frazer-Nash (2019) lists most of the IET facilities available worldwide for both water and other coolants.

An internationally agreed IET matrix was developed for the validation of best estimate thermal hydraulic codes for PWRs and BWRs (CSNI, 1996). This identified the main physical phenomena that occur during the considered accidents and the tests suitable for replicating them were selected.

The relevant experimental facilities were identified and a list of experiments was selected for each facility. This matrix collected together and classified the best sets of openly available test data for code validation, assessment and improvement, including quantitative assessment of uncertainties. The test facilities in the IET validation matrix are listed in the TIETHYS database and the test data is available from the NEA databank[26]. There are also similar resources for containment code validation (CSNI, 2014).

4.6.5 Scaling Considerations

The size and complexity of nuclear reactors means that it is impractical to undertake experiments at reactor scale (size, pressure and power), and so experiments need to be designed and performed at a reduced scale/complexity compared to the original system. Therefore, scaling is needed to ensure that the phenomena and performance in the experiment is representative of (and can be used to learn about) the behaviour of the full scale plant. Detailed information on the theory and application of scaling nuclear thermal hydraulic tests is provided in CSNI (2017).

Scaling is the process of converting any parameters of the plant at reactor conditions to those in experiments (or numerical simulations) to reproduce the dominant phenomena in the plant. The objective of scaling analysis is to minimise scaling distortion, which may result from assumptions and simplifications in scaling methods, technological limitations in constructing and operating test facilities and limitations of computer code scalability. Therefore, scaling analysis is an important part of the safety review process, and part of the US NRC EMDAP (US NRC, 2005).

Since it is generally not possible to match all conditions simultaneously in a scaled-down test facility, the most important thermal hydraulic phenomena should be prioritised as part of a PIRT process

[25] www.oecd-nea.org/dbcps/ccvm/indexset.html
[26] www.oecd-nea.org/dbcps/ccvm

and the experiment should be carefully planned and designed (Section 4.6.1). Scaling methods are essential tools that are used at both a local and system level to match the target phenomena. The most suitable approach will depend on the type of test facility and number of phenomena involved.

Basic Tests: Basic tests aim to understand a single phenomenon with little reference to the reactor geometry or operating parameters, and so they have a weak connection with scaling.

Separate Effect Tests: The main role of SETs is to investigate fundamental phenomena and processes at a local level to provide experimental data to develop and validate models. Therefore, scaling of SETs is mostly related to local methods to find the scaling parameters, such as dimensional analysis using Buckingham Pi theorem. However, sometimes, or even more often (nowadays), hierarchical scaling methods are used for scaling SET facilities, than Buckingham Pi theorem.

For example, the dimensionless scaling parameters for single-phase, fully developed forced-convection heat transfer are Nu, Re and Pr. Therefore, SETs often minimise scaling distortions by using full-scale and/or prototypical fluid conditions. However, it is important to ensure that system level scaling is not required and key parameters are not overlooked, such as preserving friction in reduced geometries or structural heat load.

Integral Effect Tests: An IET is intended to provide a similar thermal hydraulic response to a postulated accident/transient in a reference reactor to understand accident phenomena and validate system codes. Therefore, scaling at a local and system level is required to preserve kinematic and dynamic similarities. A number of scaling methods have been developed for IETs (CSNI, 2017): Linear scaling, Power-to-volume scaling, Three-level scaling, Hierarchical 2-Tiered Scaling (H2TS), Power to mass scaling, Modified linear scaling, Fractional Scaling Analysis (FSA) and Dynamical System Scaling (DSS). Each method has its merits and limitations, and so the scaling method should be selected based on the application area.

As part of the scaling analysis it is important to understand and calculate the scaling distortion associated with the experiment. This is an essential part of the reactor safety demonstration process, as it is necessary to quantify the uncertainty associated with the transposition of the model to reactor conditions and ensure that it is still valid.

4.6.6 Facilities

Frazer-Nash (2019) lists most of the thermal hydraulic test rigs and facilities available worldwide for water and other coolants (this report was written in Phase 1 of the project, Appendix A). Other databases of experimental facilities include:

- TIETHYS database[27].
- The IAEA experimental facilities in support of liquid metal cooled fast neutron systems[28].

These facilities are managed and operated by research institutions, national laboratories, commercial organisations and reactor developers, to perform the following functions:

- Understand and investigate thermal hydraulic phenomena.

[27] www.oecd-nea.org/tiethysweb
[28] nucleus.iaea.org/sites/lmfns/Pages/default.aspx

- Measurement of parameters, such as velocity, pressure, temperature and void fraction for the development of correlations for system and subchannel codes.
- Generation of high resolution test data for validation of CFD models
- Testing and qualification of systems and components under reactor conditions to confirm the performance of key reactor components, for example pumps and valves.

The quality, accuracy and repeatability of experimental data is essential to satisfy nuclear industry regulation requirements. This is particularly important for testing and qualification of nuclear components, but is also necessary for experimental data that are used to support a nuclear safety case or validate a model that is used for safety justification or reactor design.

In addition, the quality and type of instrumentation and measurement (including data acquisition and reduction) that is available within a test facility and installed on particular test rigs has a large impact on the benefit and value of a test rig and the data that it generates. Therefore, thermal hydraulic test rigs are often designed for a specific purpose and type of tests that will be undertaken.

5 References

Ammour D, Craft T, Iacovides H (2013) Highly Resolved LES and URANS of Turbulent Buoyancy-Driven Flow Within Inclined Differentially-Heated Enclosures. *Flow, Turbulence and Combustion*, 91(3), 669–696, dx.doi.org/10.1007/s10494-013-9497-1.

ASME (2006) Guide for Verification and Validation in Computational Solid Mechanics. ASME V&V 10-2006, American Society of Mechanical Engineers.

ASME (2009) Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer. ASME V&V 20-2009, American Society of Mechanical Engineers.

BEIS (2018) Industrial Strategy: Nuclear Sector Deal. UK Government Department for Business, Energy and Industrial Strategy.

BEIS (2019) Advanced Nuclear Technologies – a UK Framework. Clean Energy Ministerial 2019. UK Government Department for Business, Energy and Industrial Strategy.

BEIS (2020) Digest of United Kingdom Energy Statistics (DUKES). UK Government Department for Business, Energy and Industrial Strategy.

Bestion D, Camy R, Gupta S, Heib C, Marfaing O, Moretti F, Skorek T (2019) Requirements for CFD-Grade Experiments for Nuclear Reactor Thermalhydraulics. In *CFD4NRS7: Computational Fluid Dynamics for Nuclear Reactor Safety*.

Bouris D, Bergeles G (1999) Two Dimensional Time Dependent Simulation of the Subcritical Flow in a Staggered Tube Bundle Using a Subgrid Scale Model. *International Journal of Heat and Fluid Flow*, 20(2), 105–114, dx.doi.org/10.1016/S0142-727X(98)10053-X.

BSI (2010) Nuclear Power Plants. Instrumentation and Control Important to Safety. Classification of Instrumentation and Control Functions. BS EN 61226:2010, British Standards Institution.

CCC (2019) Net Zero The UK's Contribution to Stopping Global Warming. Committee on Climate Change.

Cheng Z, Rao Y (2015) Strategies for Developing Subchannel Capability in an Advanced System Thermalhydraulic Code: A Literature Review. *AECL Nuclear Review*, 4(1), 23–41, dx.doi.org/10.12943/ANR.2015.00039.

CNSC (2018) Regulatory Fundamentals. REGDOC-3.5.3, Canadian Nuclear Safety Commission.

CSNI (1994) Separate Effects Test Matrix for Thermal-Hydraulic Code Validation, Vol.1 and 2. NEA/CSNI/R(1993)14, OECD NEA Committee on the Safety of Nuclear Installations.

CSNI (1996) CSNI Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients. NEA/CSNI/R(1996)17, OECD NEA Committee on the Safety of Nuclear Installations.

CSNI (2014) Containment Code Validation Matrix. NEA/CSNI/R(2014)3, OECD NEA Committee on the Safety of Nuclear Installations.

References

CSNI (2015) Best Practice Guidelines for the Use of CFD in Nuclear Reactor Safety Applications - Revision. NEA/CSNI/R(2014)11, OECD NEA Committee on the Safety of Nuclear Installations.

CSNI (2016) Review of Uncertainty Methods for Computational Fluid Dynamics Application to Nuclear Reactor Thermal Hydraulics. NEA/CSNI/R(2016)4, OECD NEA Committee on the Safety of Nuclear Installations.

CSNI (2017) A State-of-the-Art Report on Scaling in System Thermal-Hydraulics Applications to Nuclear Reactor Safety and Design. NEA/CSNI/R(2016)14, OECD NEA Committee on the Safety of Nuclear Installations.

D'Auria F (2017) Thermal-Hydraulics of Water Cooled Nuclear Reactors. Woodhead.

Downing J, Jones J, Tehrani A, Moscrop R (2018) A UK Regulatory Perspective on Computational Fluid Dynamics for Nuclear Safety Analysis. In *Proceedings of TOPFUEL 2018*.

ERCOFTAC (2000) Industrial Computational Fluid Dynamics of Single-Phase Flows ERCOFTAC Best Practice Guidelines. Version 1, Turbulence and Combustion European Research Community on Flow.

ETI (2018) Enabling Resilient UK Energy Infrastructure: Natural Hazard Characterisation Technical Volumes and Case Studies. IMechE, IChemE.

Frazer-Nash (2019) Project FORTE - Nuclear Thermal Hydraulics Research & Development Review of Thermal Hydraulic Test Facilities. FNC 53798/46969R Issue 1, Frazer-Nash Consultancy.

GIF (2018) GIF R&D Outlook for Generation IV Nuclear Energy Systems: 2018 Update. Generation IV International Forum.

Girimaji S, Abdol-Hamid K (2005) Partially-Averaged Navier Stokes Model for Turbulence: Implementation and Validation. In *43rd AIAA Aerospace Sciences Meeting and Exhibit*, American Institute of Aeronautics and Astronautics, dx.doi.org/10.2514/6.2005-502.

Holgate J, Skillen A, Craft T, Revell A (2019) A Review of Embedded Large Eddy Simulation for Internal Flows. *Archives of Computational Methods in Engineering*, 26(4), 865–882, dx.doi.org/10.1007/s11831-018-9272-5.

HSE (2001) Reducing Risks, Protecting People. Health and Safety Executive.

IAEA (1991) Safety Culture. INSAG-4, International Atomic Energy Agency.

IAEA (1996) Defence in Depth in Nuclear Safety. INSAG-10, International Atomic Energy Agency.

IAEA (1999) Basic Safety Principles for Nuclear Power Plants 75-INSAG-3 Rev. 1. INSAG-12, International Atomic Energy Agency.

IAEA (2005) Assessment of Defence in Depth for Nuclear Power Plants. Safety Reports Series No. 46, International Atomic Energy Agency.

IAEA (2006) Fundamental Safety Principles. SF-1, International Atomic Energy Agency.

IAEA (2009) Passive Safety Systems and Natural Circulation in Water Cooled Nuclear Power Plants. IAEA-TECDOC-1624, International Atomic Energy Agency.

References

IAEA (2013) Hydrogen Production Using Nuclear Energy. Nuclear Energy Series NP-T-4.2, International Atomic Energy Agency.

IAEA (2014) Safety Classification of Structures, Systems and Components in Nuclear Power Plants. SSG-30, International Atomic Energy Agency.

IAEA (2016a) Application of the Safety Classification of Structures, Systems and Components in Nuclear Power Plants. IAEA-TECDOC-1787, International Atomic Energy Agency.

IAEA (2016b) Safety of Nuclear Power Plants: Commissioning and Operation. Safety Standards Series SSR-2/2 (Rev. 1), International Atomic Energy Agency.

IAEA (2016c) Safety of Nuclear Power Plants: Design. SSR-2/1 (Rev. 1), International Atomic Energy Agency.

IAEA (2017) Nuclear Power for Sustainable Development. International Atomic Energy Agency.

IAEA (2019a) Deterministic Safety Analysis for Nuclear Power Plants. SSG-2 (Rev.1), International Atomic Energy Agency.

IAEA (2019b) IAEA Safety Glossary: 2018 Edition. International Atomic Energy Agency.

IAEA (2020a) Advances in Small Modular Reactor Technology Developments . A Supplement to the IAEA Advanced Reactor Information System (ARIS). International Atomic Energy Agency.

IAEA (2020b) Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants. SSG-56, International Atomic Energy Agency.

IEA (2019) Nuclear Power in a Clean Energy System. International Energy Agency.

Incropera F P, DeWitt D P, Bergman T L, Lavine A S (2011) Fundamentals of Heat and Mass Transfer. Seventh edition, John Wiley & Sons.

JCGM (2008) Evaluation of Measurement Data - Guide to the Expression of Uncertainty in Measurement. JCGM 100:2008 (GUM 1995), Joint Committee for Guides in Metrology.

Launder B, Spalding D (1974) The Numerical Computation of Turbulent Flows. *Computer Methods in Applied Mechanics and Engineering*, 3(2), 269–289, dx.doi.org/10.1016/0045-7825(74) 90029-2.

Liao J, Wright R F, Ferroni P, Sofu T, Jin Lee S (2019) The Importance of Phenomena Identification and Ranking Table in Lead Fast Reactor Development. In *Proceedings of 18th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-18)*.

MacKay D (2010) Sustainable Energy - without the Hot Air. UIT Cambridge.

Menter F R (1994) Two-Equation Eddy-Viscosity Turbulence Models for Engineering Applications. *AIAA Journal*, 32(8), 1598–1605, dx.doi.org/10.2514/3.12149.

Menter F R (2015) Best Practice: Scale-Resolving Simulations in ANSYS CFD. Version 2.0, ANSYS Germany.

Miller D S (2009) Internal Flow Systems. Second edition, Mentor Graphics.

Moin P, Mahesh K (1998) Direct Numerical Simulation: A Tool in Turbulence Research. *Annual Review of Fluid Mechanics*, 30(1), 539–578, dx.doi.org/10.1146/annurev.fluid.30.1.539.

References

Moorthi A, Kumar Sharma A, Velusamy K (2018) A Review of Sub-Channel Thermal Hydraulic Codes for Nuclear Reactor Core and Future Directions. *Nuclear Engineering and Design*, 332, 329–344, dx.doi.org/10.1016/j.nucengdes.2018.03.012.

NAFEMS (2003) How to Plan a CFD Analysis. HT24, NAFEMS.

NAFEMS (2019) General Guidelines for Good Convergence in Computational Fluid Dynamics. https://www.nafems.org/publications/guidelines-for-good-convergence-in-cfd/.

NIRAB (2016) Prioritisation of UK Nuclear Innovation and Research Programme Recommendations. NIRAB–124-1, Nuclear Innovation and Research Advisory Board.

NIRAB (2019) Clean Growth Through Innovation - the Need for Urgent Action. NIRAB-213-3, Nuclear Innovation and Research Advisory Board.

NIRAB (2020) Achieving Net Zero: The Role of Nuclear Energy in Decarbonisation. NIRAB-244-1, Nuclear Innovation and Research Advisory Board.

NSC (2016) International Benchmark on Pressurised Water Reactor Sub-Channel and Bundle Tests Volume III: Departure from Nucleate Boiling. NEA/NSC/R(2015)7, OECD NEA Nuclear Science Committee.

OECD NEA (2000) Reduction of Capital Costs of Nuclear Power Plants. Nuclear Energy Agency, Organisation for Economic Co-operation and Development.

ONR (2014) Safety Assessment Principles for Nuclear Facilities. Revision 0, Office for Nuclear Regulation.

ONR (2019a) Licensing Nuclear Installations. Office for Nuclear Regulation.

ONR (2019b) New Nuclear Power Plants: Generic Design Assessment Guidance to Requesting Parties. ONR-GDA-GD-006, Office for Nuclear Regulation.

ORNL (2008) Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs) Volume 1: Main Report. ORNL/TM-2007/147 Vol 1, Oak Ridge National Laboratory, dx.doi.org/10.2172/1001275.

Pope S B (2000) Turbulent Flows. Cambridge University Press.

Reason J T (1997) Managing the Risks of Organizational Accidents. Ashgate.

Rogers G F C, Mayhew Y (1992) Engineering Thermodynamics: Work and Heat Transfer. Fourth edition, Prentice Hall.

Roth G A, Aydogan F (2014a) Theory and Implementation of Nuclear Safety System Codes – Part I: Conservation Equations, Flow Regimes, Numerics and Significant Assumptions. *Progress in Nuclear Energy*, 76, 160–182, dx.doi.org/10.1016/j.pnucene.2014.05.011.

Roth G A, Aydogan F (2014b) Theory and Implementation of Nuclear Safety System Codes – Part II: System Code Closure Relations, Validation, and Limitations. *Progress in Nuclear Energy*, 76, 55–72, dx.doi.org/10.1016/j.pnucene.2014.05.003.

Schlichting H, Gersten K (2017) Boundary-Layer Theory. Ninth edition, Springer, dx.doi.org/10.1007/978-3-662-52919-5.

References

Shams A, Roelofs F, Niceno B, Guo W, Angeli D, Stalio E, Fregni A, Duponcheel M, et al. (2019) Reference Numerical Database for Turbulent Flow and Heat Transfer in Liquid Metals. *Nuclear Engineering and Design*, 353, 110274, dx.doi.org/10.1016/j.nucengdes.2019.110274.

US NRC (1989) Quantifying Reactor Safety Margins Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident. NUREG/CR-5249 Rev. No. 4, U.S. Nuclear Regulatory Commission.

US NRC (2005) Transient and Accident Analysis Methods. Regulatory Guide 1.203, U.S. Nuclear Regulatory Commission.

US NRC (2020) NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 1 – Computer Code Suite for Non-LWR Design Basis Event Analysis. Revision 1, U.S. Nuclear Regulatory Commission.

Wagner C, Hüttl T, Sagaut P (editors) (2007) Large-Eddy Simulation for Acoustics. Cambridge University Press, dx.doi.org/10.1017/CBO9780511546143.

WANO (2013) Traits of a Healthy Nuclear Safety Culture. Principles PL 2013-1, World Association of Nuclear Operators.

Wilson G E (2013) Historical Insights in the Development of Best Estimate Plus Uncertainty Safety Analysis. *Annals of Nuclear Energy*, 52, 2–9, dx.doi.org/10.1016/j.anucene.2012.03.002.

Wilson G E, Boyack B E (1998) The Role of the PIRT Process in Experiments, Code Development and Code Applications Associated with Reactor Safety Analysis. *Nuclear Engineering and Design*, 186(1), 23–37, dx.doi.org/10.1016/S0029-5493(98)00216-7.

6 Nomenclature

Latin Symbols

A	Area, m^2
At	Atwood number ($At = (\rho_1 - \rho_2)/(\rho_1 + \rho_2)$)
Bi	Biot number ($Bi = hL/k_s$)
c_p, c_v	Specific heat at constant pressure or volume, $\mathrm{J\,kg^{-1}\,K^{-1}}$
d or D	Diameter ($D_h = 4A_{cs}/p_{cs}$ for hydraulic diameter), m
f	Darcy-Weisbach friction factor
Fo	Fourier number ($Fo = \alpha_s t/L^2$)
Gr	Grashof number ($Gr = gL^3\Delta\rho/\nu^2\rho = gL^3\beta\Delta T/\nu^2$, using the Boussinesq approximation $\Delta\rho/\rho \approx -\beta\Delta T$, where ΔT is often taken as $T_w - T_{s,\infty}$)
g	Acceleration due to gravity, $\mathrm{m\,s^{-2}}$
h	Specific enthalpy, $\mathrm{J\,kg^{-1}}$, Heat Transfer Coefficient (HTC), $\mathrm{W\,m^{-2}\,K^{-1}}$ or height, m
I	Radiative intensity, $\mathrm{W\,m^{-2}\,sr^{-1}}$ or $\mathrm{W\,m^{-2}\,sr^{-1}\,\mu m^{-1}}$ for a spectral density, where sr (steradian) is solid angle
J	Radiosity, $\mathrm{W\,m^{-2}}$
k	Thermal conductivity, $\mathrm{W\,m^{-1}\,K^{-1}}$
L	Length or wall thickness, m
M	Molar mass of a species, $\mathrm{kg\,kmol^{-1}}$
Ma	Mach number ($Ma = U/a$, where a is the speed of sound)
n	Refractive index
Nu	Nusselt Number ($Nu = hL/k_f$)
p	Perimeter, m
P	Pressure (P_s = static pressure, P_T = total pressure), $\mathrm{N\,m^{-2}}$ or Pa
Pe	Péclet number ($Pe = RePr$)
Pr	Prandtl number ($Pr = c_p\mu/k_f$)
q	Heat flux (rate of heat transfer per unit area, $q = Q/A$), $\mathrm{W\,m^{-2}}$
Q	Rate of heat transfer, W
r	Radius, m
R	Gas constant (for a particular gas, $R = \tilde{R}/M$), $\mathrm{J\,kg^{-1}\,K^{-1}}$
$\tilde{R}$	Universal gas constant, $8314.5\,\mathrm{J\,kmol^{-1}\,K^{-1}}$
R_{th}	Thermal resistance, $\mathrm{K\,W^{-1}}$
Ra	Rayleigh number ($Ra = GrPr$)
Re	Reynolds number ($Re = \rho UL/\mu$, or for an internal flow $Re = WD_h/A_{cs}\mu$)
Ri	Richardson number ($Ri = Gr/Re^2$)
Sr	Strouhal number ($Sr = fL/U$, where f is frequency)

St	Stanton number ($St = Nu/RePr$)
t	Time, s
T	Temperature (T_s = static temperature, T_T = total temperature), K
u_τ	Wall friction velocity ($u_\tau = \sqrt{\tau_w/\rho}$), m s^{-1}
U	Velocity, m s^{-1} or thermal transmittance, W m^{-2} K^{-1}
v	Specific volume, m^3 kg^{-1}
V	Volume, m^3
W	Mass flow rate, kg s^{-1}
y	Wall distance, m
y^+	Non-dimensional wall distance ($y^+ = yu_\tau/\nu$)

Greek Symbols

α	Thermal diffusivity ($\alpha = k/\rho c_p$), m^2 s^{-1}
β	Volumetric thermal expansion coefficient ($\beta = -(1/\rho)(\partial\rho/\partial T)$), K^{-1}
γ	Ratio of specific heats ($\gamma = c_p/c_v$)
ϵ	Emissivity or surface roughness height, m
κ	Absorption coefficient, m^{-1}
λ	Wavelength, m
μ	Viscosity, kg m^{-1} s^{-1}
ν	Kinematic viscosity and momentum diffusivity ($\nu = \mu/\rho$), m^2 s^{-1}
ρ	Density, kg m^{-3}
σ	Stefan Boltzmann constant, 5.67×10^{-8} W m^{-2} K^{-4}
τ	Shear stress, N m^{-2}
ϕ	Porosity or void fraction

Subscripts and Modifications

b	Bulk (mass-averaged) quantity
cs	Cross-sectional quantity
f	Quantity relating to a fluid
i	Quantity relating to a particular species
T	Total (stagnation) quantity
t	Turbulent quantity
s	Static quantity or quantity relating to a solid
w	Quantity relating to a wall or surface
∞	Quantity far from a wall or in free-stream
$\overline{\square}$	Average quantity
$\tilde{\square}$	Molar quantity
$\square'$	Varying quantity

7 Abbreviations

ADS	Accelerator-Driven System
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
AMR	Advanced Modular Reactor
ANL	Argonne National Laboratory
ANT	Advanced Nuclear Technology
ARIS	Advanced Reactor Information System
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Accident
BEIS	Department for Business, Energy and Industrial Strategy
BEPU	Best Estimate Plus Uncertainty
BWR	Boiling Water Reactor
C&I	Control & Instrumentation
CAD	Computer Aided Design
CASL	Consortium for Advanced Simulation of Light Water Reactors
CEA	Commissariat à l'énergie atomique et aux énergies alternatives
CFD	Computational Fluid Dynamics
CFD4NRS	CFD for Nuclear Reactor Safety
CHF	Critical Heat Flux
CNSC	Canadian Nuclear Safety Commission
CRAB	Comprehensive Reactor Analysis Bundle
CSAU	Code Scaling, Applicability and Uncertainty
CSNI	Committee on the Safety of Nuclear Installations
DBA	Design Basis Accident
DES	Detached Eddy Simulation
DNS	Direct Numerical Simulation
DSS	Dynamical System Scaling
EC	European Commission
EMDAP	Evaluation Methodology Development and Application Process
EPRI	Electric Power Research Institute
EVM	Eddy Viscosity Model
FFTF	Fast Flux Test Facility
FOM	Figure of Merit
FSA	Fractional Scaling Analysis
FV	Finite Volume
GDA	Generic Design Assessment
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
H2TS	Hierarchical 2-Tiered Scaling

Abbreviations

HPC	High Performance Computing
HTGR	High Temperature Gas-cooled Reactor
IAEA	International Atomic Energy Agency
IET	Integral Effect Test
IMSR	Integral Molten Salt Reactor
INL	Idaho National Laboratory
INR	Institute for Neutron Physics and Reactor Technology
IRRS	Integrated Regulatory Review Service
IRSN	Institut de radioprotection et de sûreté nucléaire
KIT	Karlsruhe Institute of Technology
KTH	Royal Institute of Technology
LBE	Lead-Bismuth Eutectic
LCOE	Levelised Cost of Energy
LES	Large Eddy Simulation
LFR	Lead-cooled Fast Reactor
LMFR	Liquid Metal-cooled Fast Reactor
LOCA	Loss-Of-Coolant Accident
LWR	Light Water Reactor
MMR	Micro Modular Reactor
MOOSE	Multiphysics Object-Oriented Simulation Environment
MSR	Molten Salt Reactor
NCR	Non-Conformance Report
NCSU	North Carolina State University
NEA	Nuclear Energy Agency
NEAMS	Nuclear Energy Advanced Modelling and Simulation Program
NIP	Nuclear Innovation Programme
NIRAB	Nuclear Innovation and Research Advisory Board
NIRO	Nuclear Innovation and Research Office
NNL	National Nuclear Laboratory
NPP	Nuclear Power Plant
NTH	Nuclear Thermal Hydraulics
NUREG	Nuclear Regulatory Report
NURESAFE	NUclear REactor SAFEty simulation platform
NVEC	Nuclear Virtual Engineering Capability
O&M	Operation and Maintenance
OECD	Organisation for Economic Co-operation and Development
ONR	Office for Nuclear Regulation
ORNL	Oak Ridge National Laboratory
PANS	Partially-Averaged Navier-Stokes
PIRT	Phenomena Identification and Ranking Table
PNNL	Pacific Northwest National Laboratory
PRA	Probabilistic Risk Assessment
PRACE	Partnership for Advanced Computing in Europe
PSA	Probabilistic Safety Analysis

Abbreviations

PWR	Pressurised Water Reactor
R&D	Research and Development
RANS	Reynolds-Averaged Navier-Stokes
RDFMG	Reactor Dynamics and Fuel Modeling Group
RG	Regulatory Guide
RGP	Relevant Good Practice
RISMC	Risk Informed Safety Margin Characterisation
RSM	Reynolds Stress Model
SAP	Safety Assessment Principle
SAS	Scale-Adaptive Simulation
SCWR	SuperCritical Water Reactor
SDG	Sustainable Development Goal
SEM	Spectral Element Method
SET	Separate Effect Test
SFEE	Steady Flow Energy Equation
SFR	Sodium-cooled Fast Reactor
SGS	Sub-Grid-Scale
SILENCE	SIgnificant Light and Heavy Water Reactor Thermal Hydraulic Experiments Network for the Consistent Exploitation of the Data
SMR	Small Modular Reactor
SOK	State of Knowledge
SQEP	Suitably Qualified and Experienced Personnel
SSC	Structure, System and Component
SSR	Stable Salt Reactor
SST	Shear Stress Transport
STFC	Science and Technology Facilities Council
TAG	Technical Assessment Guide
TALL-3D	Thermal-hydraulic ADS Lead-bismuth Loop with 3D flow test section
TIETHYS	The International Experimental Thermal HYdraulics Systems
TRISO	TRIstructural-ISOtropic
UK	United Kingdom
UKAEA	United Kingdom Atomic Energy Authority
UKRI	UK Research and Innovation
URANS	Unsteady Reynolds-Averaged Navier-Stokes
US DOE	United States Department of Energy
US NRC	United States Nuclear Regulatory Commission
V&V	Verification and Validation
VERA	Virtual Environment for Reactor Applications
VHTR	Very High Temperature Reactor
VVUQ	Verification, Validation and Uncertainty Quantification
WALE	Wall-Adapting Local Eddy-Viscosity
WGAMA	CSNI Working Group on Analysis and Management of Accidents
WMLES	Wall Modeled Large Eddy Simulation

A Project Context

The UK Government's 2013 Nuclear Industrial Strategy described significant ambitions for the UK to grow its nuclear capability, with the key aim of becoming a preferred nation state partner of the global nuclear technology industry. To help fulfil the initial objectives, the Nuclear Innovation and Research Advisory Board (NIRAB) was established in January 2014 and ran until December 2016. NIRAB comprised experts from industry and academia with the objective of advising Government on the approach to, and coordination of, nuclear innovation and R&D in the UK.

The UK Government's commitment to nuclear research was reiterated in the BEIS (2018) Nuclear Sector Deal between the UK Government and the nuclear industry. As a result, NIRAB was reconvened in 2018 to work in partnership with the Nuclear Innovation and Research Office (NIRO) to provide independent advice to government (NIRAB, 2019).

Based on NIRAB's prioritised recommendations for innovation and research programmes (NIRAB, 2016), BEIS committed to invest £180 million in an ambitious Nuclear Innovation Programme (NIP) from 2016 to 2021. This NIP covers research and innovation in advanced reactor design, advanced fuels, recycling and reprocessing, advanced manufacturing and materials, strategic tool kit and facilities and AMR.

The specific benefits of the five-year programme are expected to be:

- Enhanced designs, increased productivity and a step change in the way that nuclear design, development and construction programmes are delivered.
- Increased and widespread uptake of modern digital engineering practices within the UK nuclear industry.
- Improved understanding and safety of through life performance of reactor components.
- A greater predictive modelling capability and understanding of passive safety arguments.
- A highly-skilled workforce able to drive design improvements and underpin operations and regulation of future reactors.
- Leverage to facilitate extended UK participation in associated international activities.

A.1 Thermal Hydraulics Model Development

This set of technical volumes and case studies has been delivered through the thermal hydraulic model development project within the Advanced Reactor Design NIP, and consists of two distinct phases; see Figure A.1.

The first phase of the project, which ran from 2017 to 2019, began with a state-of-the-art review of the current thermal hydraulic modelling capability and experimental facilities around the world.

Phase 1	Phase 2
Literature Review	Focused R&D
User Requirements Capture	Technical Volumes
Modelling Capability Specification	Case Studies
Initial Innovative Research	Integration and Dissemination

Figure A.1: Schematic of the structure of this project.

This was followed by a stakeholder engagement programme to capture the requirements for thermal hydraulic model development and testing, which enabled a specification to be developed for both a UK national nuclear thermal hydraulics test facility and future UK modelling capability. In addition, initial innovative thermal hydraulic model development was undertaken at the University of Manchester and University of Sheffield to kick start research in the UK in advanced nuclear technologies. Further information on Phase 1, together with the published deliverables, are available at www.innovationfornuclear.co.uk/nuclearthermalhydraulics.html.

The second phase of the project, which ran from 2019 to 2021, is intended to build on the work in Phase 1 to create an enduring, coordinated and industry focused nuclear thermal hydraulic capability in the UK. This included focused R&D to address the gaps identified in Phase 1 and support the development of the technical volumes and case studies.

A key part of the NIP is to ensure that the work is integrated together to develop a coherent UK capability. Within the thermal hydraulics programme, this involved integration with:

- Nuclear Virtual Engineering Capability (NVEC): This programme is intended to develop the UK's reactor system and component design, analysis and verification capability by incorporating virtual engineering and associated technologies from high-tech industries to enhance and reduce the cost of nuclear design and development programmes.
- National Nuclear Thermal Hydraulic Facility: The design and delivery of the proposed £40 million thermal hydraulics facility in the north of Wales, as outlined in the BEIS (2018) Nuclear Sector Deal, is being led by the UKAEA, and builds on the test facility specification developed in Phase 1.

Based on the outputs from Phase 1, the technical volumes and case studies are focused on single-phase heat transfer and natural convection to support passive cooling applications, as it has not been possible to address all thermal hydraulic phenomena within the scope of this project. This is consistent with the overall focus of the NIP on advanced nuclear technologies and passive safety arguments, as single-phase flow is applicable to light water SMR designs during normal operation and Generation IV reactor designs during normal operation and a range of fault scenarios.

A.2 Project Consortium

The consortium assembled for the delivery of this project includes experts on nuclear thermal hydraulics for advanced nuclear technologies from different organisations. These partners could call upon a range of experience from many years of work within the field. The consortium was led by Frazer-Nash Consultancy and supported by Checkendon Hill, DBD, EDF Energy, LeadCold, Moltex Energy, National Nuclear Laboratory (NNL), Rolls-Royce, Science and Technology Facilities Council (STFC), Terrestrial Energy, the University of Manchester, the University of Sheffield, U-Battery and Westinghouse.

Frazer-Nash is a leading UK-based systems and engineering technology company with over 30 years of nuclear industry experience. Frazer-Nash supports every operator of a nuclear licensed site in the UK and is one of the largest technical consultancies in the UK. Frazer-Nash successfully helps clients meet the strict nuclear industry regulatory requirements throughout the life of their nuclear projects by addressing thermal hydraulic issues using empirical, system level and CFD approaches.

DBD is an independent global enterprise providing innovative solutions to complex technical, engineering and strategic challenges in highly regulated industries. DBD's portfolio includes the delivery of the comprehensive study of the Chinese HTR-PM under Phase 1 of the BEIS AMR Feasibility & Development project.

EDF Energy, part of the EDF Group, is one of the UK's largest energy companies and its largest producer of low-carbon electricity. EDF is currently developing a 300-400 MWe SMR, called the Nuward, in partnership with Commissariat à l'énergie atomique et aux énergies alternatives (CEA), Naval Group and TechnicAtome. The EDF Energy R&D UK Centre was formed in March 2012 to strengthen its ability to deliver R&D activity in the UK. This work builds on EDF Energy R&D UK's experience delivering a similar set of Technical Volumes and Case Studies on Natural Hazard Characterisation (ETI, 2018).

LeadCold is a spin-off from the KTH Royal Institute of Technology in Stockholm that was founded in 2013 to develop a lead-cooled fast reactor. The SEALER-UK design is a 55 MWe lead-cooled reactor using uranium nitride fuel, which has been submitted for review under Phase 1 of the BEIS AMR Feasibility & Development project.

Moltex Energy, based in the UK and Canada, is developing a Stable Salt Reactor (SSR), which uses molten salt fuel in essentially standard fuel assemblies. Their design was reviewed under Phase 1 of the BEIS AMR Feasibility & Development project, and uses rectangular cores that can be extended module by module to create reactors from 150 MW to 1200 MW power.

NNL is an independent, UK government-owned and operated nuclear services technology provider covering the whole of the nuclear fuel cycle. NNL was established in 2008 and brought together the UK's nuclear R&D capability into one organisation.

Rolls-Royce SMR has been established as an independent company, drawing on decades of Rolls-Royce experience in nuclear design and engineering. The 470 MWe Rolls-Royce SMR technology is designed to reduce cost through factory-built modular construction using tried and tested nuclear technology and advanced manufacturing techniques.

STFC is part of UK Research and Innovation (UKRI). Their researchers and software developers utilise the UK's premier supercomputing environment to provide an outcome based R&D service to help UK industry and academia develop better products, services, processes and software.

Terrestrial Energy was established in Canada in 2013 and is developing an Integral Molten Salt Reactor (IMSR)® power plant that generates 195 MWe with a thermal spectrum, graphite-moderated, molten-flouride-salt reactor system. This uses molten salt as a liquid fuel and builds on research at the ORNL.

The University of Manchester, School of Mechanical, Aerospace and Civil Engineering (MACE) has a 30-year track record in experimental and computational thermal hydraulics, particularly in the development of innovative, world leading strategies for modelling the effects of turbulent fluctuations, especially in the near-wall region. These include CFD methods for forced, natural and mixed convection, conjugate heat transfer, molten salt heat transfer and 3D unsteady cooling flows. Manchester's extensive nuclear R&D is coordinated by the Dalton Nuclear Institute.

Project Context

The University of Sheffield Thermofluids Group has over 20 staff members and conducts research in energy, aerodynamics and fundamental fluid mechanics. Their current nuclear research focuses on core thermal hydraulics of Generation IV and Advanced Gas-cooled Reactors (AGRs) with strong expertise in turbulence modelling and advanced computational approaches. The campus-wide Energy 2050 Centre is the focal point of the University's energy research which includes nuclear thermal hydraulics, waste disposal and advanced materials.

U-Battery is a Micro Modular Reactor (MMR) that is being developed by Urenco in the UK in partnership with Jacobs, Cammell Laird and Laing O'Rourke. The U-Battery design is a 10 MWt high temperature gas-cooled reactor powered by TRIstructural-ISOtropic (TRISO) fuel, which was reviewed under Phase 1 of the BEIS AMR Feasibility & Development project.

Westinghouse is a world leader in the development, commercialisation and servicing of nuclear power plants. Having successfully delivered the new AP1000® reactor design, it is currently developing a 450 MWe class, passively safe, pool-type, liquid lead-cooled, fast neutron spectrum reactor and has recently been involved in further SMR development based on the AP1000 passive cooling technology. The Westinghouse Lead-cooled Fast Reactor was reviewed under Phase 1 of the BEIS AMR Feasibility & Development project.

Volume 1