Project FORTE - Nuclear Thermal Hydraulics Research & Development

Thermal Hydraulic Capability User Requirements

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FRAZER-NASH CONSULTANCY

SYSTEMS AND ENGINEERING TECHNOLOGY

An introduction to Project FORTE

The Department for Business, Energy and Industrial Strategy (BEIS) has tasked Frazer-Nash Consultancy and its partner organisations to deliver the first phase of a programme of nuclear thermal hydraulics research and development.

Phase 1 of the programme comprises two parts:

- The specification and development of innovative thermal hydraulic modelling methods and tools; and
- The specification of a new United Kingdom thermal hydraulics test facility.

The work is intended to consider all future reactor technologies including Gen III+, small modular reactors and advanced reactor technologies.

Our project partners

The team is led by Frazer-Nash Consultancy and includes:

For more information, visit www.innovationfornuclear.co.uk/nuclearthermalhydraulics.html

Executive Summary

Project FORTE is Frazer-Nash's designation for the BEIS Research and Development project on nuclear thermal hydraulic engineering. BEIS vision for the work is based on the fact that:

Government has indicated that nuclear energy could play a significant role in the UK's future energy mix. To support this aim, R&D is needed to inform, underpin and deliver Government policy on nuclear technologies. The Nuclear Innovation and Research Advisory Board (NIRAB) has provided advice to Government on priority R&D programmes needed to inform and underpin Government policy and to deliver an integrated, overarching Nuclear Innovation Programme for the UK.

As part of this vision, NIRAB identified two priorities for nuclear thermal hydraulics R&D: creation of a major new UK Nuclear Thermal Hydraulic Test Facility and the development of new Nuclear Thermal Hydraulic Modelling techniques and tools. The key thrust of Phase 1 of Project FORTE is the development and expansion of these two titles into well-defined specifications supported by evidence and documentation to allow them to be realised in a future funding release. Project FORTE has therefore developed multiple strands of investigation to understand and characterise nuclear thermal hydraulics and the requirements for research across different reactor types.

In the research contributing to this report, Frazer-Nash Consultancy, supported by our partner organisations, has sought feedback from a large number of organisations and participants with specific expertise in nuclear thermal hydraulics to understand the needs that should drive development of the capability. As a result, input has been received from a wide range of reactor developers, academic institutions, service providers and the UK regulatory body covering a diversity of reactor types and development aspirations. A series of standardised pro-forma were used to capture feedback information from the organisations in a common format for ease of comparison; these were combined with more informal discursive sessions to allow respondents the freedom to highlight items of particular importance to them.

Systems Engineering methods have been used to structure this feedback into a series of User Requirements which define the desired capability in terms of thermal hydraulic testing, the wider test facility infrastructure and simulation and modelling. Taken as a complete set, the User Requirements define work that is needed to deliver progress or improvement in nuclear thermal hydraulics for each reactor technology considered. The primary purpose of this report is to document the process undertaken and the requirements captured. The further development, prioritisation or potential technical solutions to the requirements are not considered at this stage.

Although there are clearly some significant differences in the reactor technologies considered, some common themes have become apparent within the requirements set: These include:

- The quantification and bounding of uncertainty in Computational Fluid Dynamics (CFD) to increase 'trust' in advanced thermal hydraulic models;
- The need for high quality validation data to support model development and reactor design activities;
- The innovative combination of different modelling tools and techniques to enable a more complete picture of the physics and/or gain results in practical timescales;
- Improvements in the understanding and simulation of four thermal hydraulic phenomena: natural convection, two-phase flow, single phase turbulent mixing, and fluid flow driven component fatigue.

The requirements have been subject to further analysis to develop the specification documents for the test facility and the modelling capability. Issue 2 of this document captures additional user requirements identified during these subsequent stages of the work.

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1 Introduction

1.1 Project Context

The UK Government's 2013 Nuclear Industrial Strategy described significant ambitions for the UK to grow our nuclear capability; with the key aim of becoming the preferred nation state partner of the global nuclear technology industry. To help fulfil the strategy's initial objectives, the Nuclear Innovation and Research Advisory Board (NIRAB) was established in 2014. NIRAB comprised experts from both industry and academia with the objective of advising Government on the approach to and coordination of nuclear innovation, research and development in the UK. NIRAB existed from January 2014 until December 2016, publishing annual reports and companion documents, analysing and describing the civil nuclear Research and Development (R&D) landscape. In March 2016, a set of recommendations for innovation and research programmes was published (Reference 1), the recommendations being prioritised in a subsequent report in November 2016 (Reference 2).

In October 2015, the Government set aside £250m for civil nuclear R&D activities "…to re-ignite the nuclear industry in the UK". Approximately half of that budget was to be used in activities relating to Small Modular Reactors (SMR); the remainder was applied to more general civil nuclear R&D. Control of a large part of that budget was inherited by the Department for Business, Energy and Industrial Strategy (BEIS) from its predecessor, The Department of Energy and Climate Change (DECC). This funding is being used specifically to address NIRAB's high priority recommendations.

Control of thermal hydraulic phenomena is at the heart of all current and future nuclear reactor designs, underpinning both reactor performance and safety arguments. Therefore, a thorough understanding of thermal hydraulic effects, the capability to simulate them accurately and validate the key prediction methodology experimentally is essential for efficient design and safe operation through life.

Historically the UK has had a strong capability in thermal hydraulic science and engineering, which was derived from and could be applied across many industries. Several of the most successful commercial Computational Fluid Dynamics (CFD) codes have their origins within UK research institutions. There remains considerable UK-based expertise and impetus in the development of thermal hydraulic modelling tools and techniques to service the needs of other industries. However, the general lull in activity in the UK civil nuclear industry since Sizewell B was commissioned in 1995, is also evident in nuclear thermal hydraulic research. The UK currently has no major civil nuclear thermal hydraulic test facilities and activity in modelling R&D has been limited by a lack of funding or other stimulus, from either the UK industrial base or government.

In order to begin to arrest this trend, and within the overall aim of allowing nuclear energy to play a significant role in the UK's future energy mix, two of the NIRAB recommendations relate to nuclear thermal hydraulics:

- The development of a major new UK Nuclear Thermal Hydraulic Test Facility;
- The development of new Nuclear Thermal Hydraulic Modelling techniques and tools.

The development of specification documents to support the delivery of these two recommendations is the main objective of Phase 1 of Project FORTE; the full programme is described in Reference 3. The Project FORTE team is led by Frazer-Nash Consultancy and includes: the University of Manchester; the University of Sheffield; STFC Daresbury; EDF Energy and Westinghouse Electric Company.

The contribution of this document to meeting the project objectives is described in Section 1.2.

It is worth highlighting that nuclear thermal hydraulics is an extremely active research area internationally. This is evident in the numerous projects and initiatives managed under the Generation IV International Forum, many of which identify thermal hydraulics as an area where research is required. The recent NURETH-17 conference (Reference 4) is just one of a number of major conferences held regularly. Many hundreds of research papers were presented and the conference reports that over 2,500 were submitted. International engagement and, where appropriate collaboration, is likely to be extremely important to the success of whatever is done in the UK. Additionally, it is important to be realistic regarding the scope of what can be done with the comparatively small amount of funding available.

1.2 Report Structure and Objectives

The primary objective of this report is to document the ‘user requirements' (see Section 2 for definition of users) for nuclear thermal hydraulic modelling capability development and testing which will be used to develop the specifications for both the UK test facility and UK nuclear thermal hydraulic model development for both Phase 2 of this R&D programme and, potentially, the next two decades.

A requirements engineering process is a systematic approach to the specification and management of requirements. Requirements are defined as:

- A need or desire perceived by a stakeholder;
- A capability or property that a system should have;
- A documented representation of a need, capability or property.

User requirements describe the capability that a particular system should deliver for its users. They express the outcomes, or effects, that the users of the capability need to be able to achieve in the area of nuclear thermal hydraulics modelling and testing.

The principle of the approach is that user requirements can then be developed into system requirements which capture the functionality and performance of the ‘system’ necessary to deliver the user requirements. These system requirements will be reported in the specification documents for the modelling capability and test facility respectively.

As is always the case, the precise approach taken needs to be tailored to the circumstances. In this case, the subject matter is extremely broad, the potential stakeholders are unbounded and the level of detail to which the requirements are known is often low (unsurprising in a research environment). This has necessitated a high level approach to the output and the requirements are documented in the form of a ‘Statement of Need’ supported by a ‘Justification’ or ‘Benefit’. This report also aims to describe the approach taken to gathering and analysing the requirements.

The remainder of this document is structured as follows:

- Section 2 describes the approach taken to the gathering of user requirements including the selection of stakeholders and potential users;
- Section 3 describes the capability structure and the scope of the study;
- Section 4 explains the methods used to analyse the requirements information;
- Section 5 presents the user requirements in tabular form;

- Section 6 discusses the outcomes of this stage of the work, drawing together common themes in preparation for the prioritisation of the requirements and development of the specifications in the next stage of the work;
- Section 7 contains a summary of the abbreviations used in this document;
- Section 8 contains a list of references.

It is a consequence of the size of the international civil nuclear community and the pace of nuclear thermal hydraulic research that a number of the statements in this document may be out of date by the time it is issued. It is also worth stating that this document is not considered to be comprehensive. Capturing all of the relevant needs would be an impossible task and, given the large number of needs already listed in Section 5, it is not considered to be a benefit at this time in increasing this number with likely lower priority requirements.

A number of the stakeholder engagement activities yielded additional useful and relevant information regarding the possible prioritisation of research, potential technology transfer or opportunities for future collaboration. This information has been captured and will feed into future deliverables, but are beyond the scope of this document.

Issue 2 of this document captures additional requirements that have been identified during the development of the specification documents (References 5 and 6). It should be noted that this update does not duplicate the detailed quantitative information reported in the Test Facility Specification (gathered from both users and consultation with existing facilities), user requirements are captured at a higher level in this document where relevant.

1.3 Acknowledgements

As described in Section 2 of this report, the approach taken has involved extensive engagement with the civil nuclear industry and with researchers in the area of nuclear thermal hydraulics. Many of the individuals contacted have taken the time to answer challenging questions and on many occasions have supported both telephone and face-to-face meetings.

Without this level of engagement, the process would have been severely compromised and we would like to acknowledge the contribution of the organisations listed in Section 2 and offer them our sincere thanks.

2 Approach

There are a variety of appropriate approaches which could have been used for gathering information to input into the specification for a civil nuclear thermal hydraulic modelling and testing capability. The approach that was chosen to carry out engagement activities involved a representative selection of stakeholders to capture the user requirements directly. This activity was then backed up by targeted research to better understand the issues raised and fill in any identified gaps.

This approach, whilst time consuming, has a number of key advantages:

- It directly addresses the UK Government requirement that any R&D that they fund is clearly linked back to benefit to the nuclear energy industry.
- It specifically involves the early dissemination of the objectives and potential outcomes of this work directly to relevant stakeholders thereby promoting future engagement in the programme.
- It removes the risk of further work being biased towards the preconceptions of, or the areas of most interest to, the core project team.
- It promotes an outcome that balances industry needs with research interests and UK strengths.
- It initiates international engagement with the programme, thereby maximising the potential for future collaboration.
- It allows for freedom of contributions from technologies with different levels of maturity, including engagement with smaller organisations and early stage reactor designs.

A consequence of this approach is that a greater level of emphasis and depth is given to those technologies where good engagement was received. However, this aligns with an indication of specific interest in the UK nuclear industry from at least one representative of that technology, thereby making it more likely that the technology will be either developed or deployed in the UK.

The breadth of the task is described in Sections 2.1 and 2.2.

2.1 Technologies Considered

In alignment with the NIRAB recommendations, this document considers the needs of current new build, small modular reactor designs and Generation IV (GenIV) reactor designs[1].

It is recognised that it is extremely unlikely that all GenIV technologies will be developed in the UK. However, as it is currently unclear which (if any) will be developed in the UK, an attempt has been made to capture the needs of as many technologies as possible. It is noted however that, at the current time, more information regarding some technologies have been received than others.

In addition, requirements were also captured from thermal hydraulics practitioners with expertise in the current UK Advanced Gas-cooled Reactor (AGR) fleet. Although not the main focus of this work, many of the requirements have relevance to ageing reactors in general and therefore will be of importance to all reactor designs.

The specific technologies considered are listed in Table 1.

[1] See Reference 7 for a description of GenIV technologies.

Reactor Technology	Reactor Type
Pressurised Water Reactor (PWR)	Gen II PWR Gen III (+) PWR Small/modular PWR
Boiling Water Reactor (BWR)	Gen II BWR Gen III ABWR Advanced RBWR Small DMS BWR
Advanced Gas-cooled Reactor (AGR)	Gen II AGR
Liquid Metal Fast Reactor (LMFR)	Sodium-cooled Fast Reactor (SFR) Lead-cooled Fast Reactor (LFR) Lead-Bismuth Eutectic (LBE) fast reactor
Molten Salt Reactor[2] (MSR)	Solid Fuel designs Stable Salt Reactor Dissolved Fuel designs
Supercritical Water Reactor (SCWR)	Pressure vessel type Pressure tube type
High Temperature Gas-cooled Reactor (HTGR)	Very High Temperature Reactor (VHTR) Gas-cooled Fast Reactor (GFR)

Table 1: Reactor Technologies Considered

2.2 Stakeholders

In order to capture user requirements for a UK test facility or UK thermal hydraulic model development, the relevant stakeholders need to be identified.

These have been taken as:

- UK and Welsh Governments;
- Menai Science Park (M-SParc) owners;
- Potential Users of a UK based nuclear thermal hydraulic test facility;
- Potential Users of nuclear thermal hydraulic modelling tools;
- UK and international nuclear thermal hydraulic researchers/projects;
- Other BEIS funded nuclear R&D projects.

Of these, the potential users are key to the generation of requirements (although all stakeholders have been given the opportunity to express their views). The identification and classification of ‘Users’ is described further in Sections 2.3 and 2.4.

[2] There are a large number of different variants of Molten Salt reactor design and it has not been possible to capture information on all designs. The range chosen is expected to capture representative thermal hydraulic challenges.

2.3 Identification of Users

Potential users for both the test facility and thermal hydraulic modelling tools are considered to be from both industry and research organisations. It was therefore important to gather the views of both groups in order to form a complete picture of the requirements.

2.3.1 Selection of Industry Users

The selection of users presented a number of challenges. UK based organisations, or those with UK industry interests, were not enough in themselves to provide adequate coverage of GenIV technologies or provide a large enough pool of contributors to determine common ground between technologies. However, if you consider reactor design world-wide, the scope becomes enormous. For example, in addition to existing Gen II and III technologies, over a 100 SMR designs alone means that there are thousands of experts worldwide, all of which potentially have value to add.

Given that the remit of this project is not to solve all of the world's problems, but to create capability to support the UK nuclear industry, a process of down-selection needed to be applied. Therefore, the industry stakeholders have been selected as follows:

1. Specific knowledge and expertise in nuclear thermal hydraulics, and;
2. Existing presence in UK nuclear power generation, or;
3. Planned presence in UK nuclear power generation as part of current new build programmes, or;
4. A clear indication of an interest in the UK civil nuclear market by (for example), approaching the UK government with a design, idea or requirement and/or entering the recent UK 'SMR competition'.

This created a shortlist which was further refined using the level of development of the design in question and whether the specific organisation had sufficient, in house, thermal hydraulics expertise (where in doubt stakeholders have been included). Any organisation who proactively approached the project team having heard about this work was also included. The final list of contributors is given in Section 2.5.1.

2.3.2 Selection of Researchers and Subject-Matter-Experts

As UK regulator, the Office for Nuclear Regulation (ONR) was identified as an important contributor to this work and a relevant Subject-Matter-Expert.

In addition to capturing requirements from industry, there was considerable value in drawing on the expertise of researchers in this area as their insight into the challenges would be of value in the context of the industry requirements. In addition, researchers do form part of the set of potential 'users' of both the test facility and modelling capability. The 'use' would be less directly linked to power generation, but research activities form an important aspect of international collaboration and are a key part of a progressive industry.

The academic individuals have been down-selected according to the following criteria:

1. Specific expertise and knowledge in nuclear thermal hydraulics to support design (rather than decommissioning), and;
2. Experience and knowledge of working directly with industry in this area (i.e. experience in higher level TRL research), and;
3. Expression of interest in this area by joining the Nuclear Thermal Hydraulics (NTH) Special Interest Group (SIG), or;
4. Specific recommendation by academic or industrial colleague indicating relevant expertise.

2.3.3 BEIS Programme Cross-Cutting

While the priority for this project is the advancement of thermal hydraulics, there are benefits to exploiting opportunities for collaboration with the other research and development themes. For example, by delivering a test facility specification that exploits opportunities to perform tests focused on other areas, the value of the test facility is maximised over its lifetime.

The approach used for the identification of cross-cutting opportunities relating to the thermal hydraulic test facility is documented in Reference 8.

The 'Digital Reactor Design - Virtual Engineering' project was specifically identified as relevant for engagement with regard to thermal hydraulic model development.

2.3.4 Global Interest

By identifying shortlists of industry and research users as described in Sections 2.3.1 and 2.3.2 not all of the credible GenIV technologies were covered. Recognising the UK's ambitions to: a) become an international partner of choice in nuclear, and; b) develop capability in GenIV technologies, the purpose of our global stakeholder engagement is therefore to:

1. Identify further thermal hydraulic modelling or test requirements;
2. Identify potential international organisations for collaboration in the future;
3. Inform an exploitation strategy for the test facility, by identifying where it may be useful to international programmes.

The challenge is that the global landscape is huge. After discussion with BEIS, it was agreed that the Generation IV International Forum (GIF) was an excellent starting point. This potentially provided access to the thermal hydraulic requirements of technologies not included in the UK SMR competition.

A list of GIF contacts were provided by BEIS, together with a number of US National Laboratories. These, along with a number of other international points of contact that have been made in the course of this project, were included in our engagement activity.

2.4 Classification of Users

To better understand the requirements, it is important to recognise that there are different types of users with different objectives. For example, a nuclear regulator will be entirely focused on aspects that improve or assure nuclear safety. A reactor designer, however, will likely also be interested in areas that improve design performance or reduce development costs and timescales.

The classification of 'Users' that that has been used in the context of the user requirements capture process is displayed in Table 2.

From a thermal hydraulic perspective, it was generally found not necessary to distinguish between small and large reactors with equivalent technology (i.e. there is a lot of commonality between the thermal hydraulic requirements of small and large reactors). Where the requirements generated relate specifically to reactors of a particular size, this is clarified in the individual requirement.

User Classification	Description	Examples
Gen III (+) Reactor Designer	Designer/vendor of a reactor utilising currently well understood Gen III technology.	Hitachi-GE Westinghouse EDF
GenIV Reactor Designer	Designer/vendor of a reactor utilising less mature GenIV technology.	Moltex Hydromine GenIV international reactor development projects
Fuel Vendor	Designer/Vendor of nuclear fuel.	Westinghouse Areva
Researcher	Individual or team engaged in nuclear thermal hydraulic research activities. This could include individuals from industry in addition to University researchers.	The University of Sheffield Imperial College Hitachi-GE
UK Reactor Operator	Operator (licensee) or future operator for a UK based nuclear reactor.	EDF Energy NNB GenCo Horizon
Regulator	UK Nuclear Regulator	ONR
Code Developer	Developer/vendor of a modelling code with capability relevant to nuclear thermal hydraulics.	Siemens ANSYS US DOE
Provider of Technical Services	Supplier of nuclear thermal hydraulic services to the industry.	Frazer-Nash Consultancy NNL
Cross-Cutting	BEIS Nuclear Innovation Programme projects that may benefit from the outputs of this project.	Wood plc (Formally AMEC Foster Wheeler) NNL

Table 2: Classification of Users

2.5 Information Gathering

The initial intention was to gather requirements via a series of workshops. However, whilst a successful workshop was held to discuss PWR thermal hydraulics, it was realised that to continue via this method alone was insufficient.

Specific logistical challenges arose where relevant thermal hydraulic experts were based abroad. Furthermore, the raising of ‘challenges’ in the ability to predict reactor performance was considered sensitive information by some contributors and they preferred to make a more controlled contribution rather than raise issues in a workshop in front of potential competitors.

To address this issue, a series of questionnaires were produced and sent to a variety of stakeholders who had expressed an interest in making a contribution. The questionnaires were tailored to the needs of each group of users. Three examples are included in Annex A of this document for interest.

In addition to the distribution of questionnaires, requirements were also gathered via numerous e-mails, telephone conversations and dedicated meetings. E-mails and telephone conversations were used to:

- Introduce organisations to the project and initiate engagement;
- Identify the most appropriate individuals within an organisation to make a contribution (noting that a detailed knowledge of thermal hydraulics was required);
- Discuss requirements and ideas with those based outside of the UK.

Meetings gave contributors a chance to discuss and expand on their requirements or to involve a larger number of individuals from within their organisation.

2.5.1 Responses

The users identified by the approach described in Section 2.3 comprised 59 organisations. Table 3 and Table 4 list all of the organisations contacted, the type of user they represent and the manner of their contribution to date. In some cases the organisation has not provided a contribution at all. In some cases a number of individuals from each organisation made a contribution and some organisations represent more than one category of ‘User’.

Organisation	Classification	Outcome of Engagement
Frazer-Nash Consultancy	Provider of Technical Services Cross-Cutting	Good contribution in writing and face-to-face.
Westinghouse	Gen III (+) Reactor Designer Fuel Vendor	Good contribution in writing and face-to-face.
EDF Energy UK	UK Reactor Operator Researcher	Good contribution in writing and by telephone.
EDF s.a.	Gen III (+) Reactor Designer Code Developer	Contacted, but no contribution to date. Contribution expected from code developers on modelling R&D.
AREVA	Gen III (+) Reactor Designer Fuel Vendor	Contacted, but no contribution to date.
Hitachi-GE	Gen III (+) Reactor Designer Researcher	Good contribution in writing and face-to-face.
Horizon	UK Reactor Operator	Contribution via telephone and via Hitachi-GE
NNB GenCo	UK Reactor Operator	Initial contribution in writing and via telephone.

Organisation	Classification	Outcome of Engagement
NuScale Power	Gen III (+) Reactor Designer	Good contribution in writing and by telephone.
Rolls-Royce	Gen III (+) Reactor Designer	Good contribution in writing and face-to-face.
Moltex Energy Ltd	GenIV Reactor Designer	Good contribution in writing and by telephone.
GE-Hitachi	GenIV Reactor Designer	Initial engagement no contribution to date.
Advanced Reactor Concepts LLC	GenIV Reactor Designer	Contacted, but no contribution to date.
Holtec	Gen III (+) Reactor Designer	Good contribution in writing and face-to-face.
Sealer-UK (LeadCold)	GenIV Reactor Designer	Contacted, but no contribution to date.
Hydromine	GenIV Reactor Designer	Good contribution in writing and face-to-face.
GF Nuclear	GenIV Reactor Designer	Initial interest shown, but no contribution to date.
U-battery collaboration (Urenco and AMEC)	GenIV Reactor Designer	It is understood that a contribution is in preparation.
Terrestrial Energy	GenIV Reactor Designer	Contribution by telephone.
Wood plc (Formally AMEC Foster Wheeler)	Provider of Technical Services Cross-Cutting	Good contribution face-to-face. Good contribution face-to-face.
Siemens PLM	Code Developer	Initial interest shown, but no contribution to date.
ANSYS	Code Developer	Initial interest shown, but no contribution to date.
University of Sheffield	Researcher Cross-cutting	Good contribution in writing and face-to-face. No cross-cutting contribution to date.
The University of Manchester	Researcher	Good contribution in writing and face-to-face.
Science and Technology Facilities Council (Scientific Computing Department and Hartree Centre)	Researcher Provider of Technical Services	Initial contribution in writing.

Organisation	Classification	Outcome of Engagement
Imperial College London	Researcher	Good contribution face-to-face.
Cambridge University	Researcher	Good contribution in writing
Leeds University	Researcher	Good contribution in writing
Liverpool John Moores University	Researcher	Good contribution in writing
Bangor University	Researcher	Good contribution by telephone and face-to-face.
NNL	Provider of Technical Services Cross-Cutting	Initial interest shown, but no contribution to date. No cross-cutting contribution to date.
UKAEA	Provider of Technical Services	Good contribution in writing
National Nuclear User Facility (NNUF)	Provider of Technical Services	Initial contact made, but no contribution to date.
ONR	Regulator	Good contribution face-to-face.
Brunel University	Cross-Cutting	Initial contact made, but no contribution to date.
Cammell Laird (NAMRC)	Cross-Cutting	Good contribution face-to-face.

Table 3: Users Selected from Process Described in Section 2.3

Country	Organisation/Project	Outcome of Engagement
Australia	Australian Nuclear Science and Technology Organisation (ANSTO)	Contact made, no engagement to date.
Belgium	MYRRHA Project	Contact made, no engagement to date.
Canada	Natural Resources Canada	Contact made, no engagement to date.
Canada	Canadian National Laboratories	Good contribution in writing and by telephone.
Croatia	EURATOM Research & Innovation Programme Committee representative	Contact made, no engagement to date.
Czech Republic	EURATOM Research & Innovation Programme Committee representative	Contact made, no engagement to date.
France	CEA	Initial engagement by e-mail.

Country	Organisation/Project	Outcome of Engagement
Italy	EURATOM Research & Innovation Programme Committee representative	Contact made, no engagement to date.
Japan	Japan Atomic Energy Agency (JAEA) Fast Reactor Research and Development Centre	Contact made, no engagement to date.
Japan	JAEA Advanced fast reactor cycle system R&D centre	Contact made, no engagement to date.
Korea	Seoul National University	Contact made, no engagement to date.
Korea	Korea Atomic Energy Research Institute (KAERI)	Contact made, no engagement to date.
Netherlands	Nuclear Research Group	Contact made, no engagement to date.
Poland	NCBJ	Contact made, no engagement to date.
Slovakia	Nuclear Safety Division, VUJE, Inc	Contact made, no engagement to date.
South Africa	Department of Energy Republic of South Africa	Contact made, no engagement to date.
Switzerland	Paul Scherrer Institute	Contact made, no engagement to date.
USA	Argonne National Laboratory	Contact made, no engagement to date.
USA	Oak Ridge National Laboratory	Contact made, no engagement to date.
USA	Idaho National Laboratory	Good contribution in writing.
USA	Los Alamos National Laboratory	Interest received
USA	Sandia National Laboratory	Contact made, no engagement to date.
USA	University of Pittsburgh	Contact made, no engagement to date.

Table 4: Potential International Users

It should be noted that the emphasis at this stage is very much user requirements. Engagement for other reasons such as technology transfer or international collaboration has clearly been promoted by our approach. However, a discussion of the results of this engagement are outside of the scope of this document.

2.5.2 Other Research

In additional to gathering requirements information from potential users, information has also been gathered regarding outstanding challenges from published literature. This has been used to better understand the requirements and to generate high level requirements in areas where engagement has been low.

There has been no attempt to make this research comprehensive, as the breadth of the project is far too large to reasonably achieve this in the timescales. There is clearly a risk in using this approach that the information gathered is out of date. However, where possible, information has only been sourced from references less than 5 years old.

The primary sources of information of this type are listed in Table 5.

Source	Types of Information Gathered
Generation IV International Forum (GIF)	GIF technology roadmap information Information regarding which partners are actively pursuing which technologies. Technical publications containing requirements or further details on requirements.
IAEA Scientific and Technical Publications and Advanced Reactor Information System (ARIS) database.	Technical publications containing requirements or further details on requirements.
Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), including from the Committee on the Safety of Nuclear Installations (CSNI)	Technical publications containing requirements or further details on requirements.
European Commission, Community Research and Development Information Service (CORDIS)	Links to EU research funded under FP7-EURATOM and Horizon 2020 nuclear thermal hydraulics programmes such as THINS, SESAME, and SAMOFAR. Individual programmes provided technical publications containing requirements or further details on requirements.
Academic journals and conference proceedings, including: *Nuclear Engineering and Design* *International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH)*	Technical publications containing requirements or further details on requirements.
Publications available from international nuclear research bodies, including: *Argonne and Idaho National Laboratories (ANL and INL, USA)* *Commissariat à l'énergie atomique et aux énergies alternatives, la Direction de l'Énergie Nucléaire (CAE-DEN, France)* *Studiecentrum voor Kernenergie - Centre d'Étude de l'énergie Nucléaire (SCK-CEN, Belgium)*	Technical publications containing requirements or further details on requirements.

Table 5: Sources of Additional Information

3 Capability Structure

3.1 Capability Definition

The UK's nuclear thermal hydraulic capability is considered to be all of the nuclear industry accessible UK-based expertise, facilities and organisations, which facilitate understanding of nuclear thermal hydraulic effects and provide the ability to simulate and experimentally validate thermal hydraulic processes. This work has focused on two technical elements of this capability: modelling and simulation and test facilities. In order to provide meaningful full-system analysis of the capability derived from these two elements, Frazer-Nash recognise the need to investigate the supporting enterprises, services and products required to deliver them.

This project is primarily focused on the technical features of the capability - namely producing engineering specifications of the capability's technical requirements. However, Frazer-Nash's Systems Engineering approach recognises that considering technical requirements in isolation during capability development presents risks. Limiting the scope of the task in this way may lead to immature or underdeveloped elements needed to support the technical aspects of the task and cause delivery issues. The systems that deliver this capability are highly interdependent and a failure to recognise this explicitly has the potential to cause significant adverse cost, timing or quality effects later in the programme's development.

Frazer-Nash has therefore taken a more comprehensive view in this phase of the task. In this section, the non-technical characteristics of this capability are considered to allow a development strategy to be created which considers a broader spectrum of risk and is therefore less prone to unforeseen challenge. In some areas, further consideration of these non-technical aspects is either within the scope of subsequent tasks in Phase 1 of this programme or could be planned into Phase 2 of the programme. This is clarified in Sections 3.2 and 3.3.

The approach used widely in Defence engineering captures eight characteristic 'Lines Of Development' (LODs) which should be considered when defining a capability; the eight LODs are: Training, Equipment, Personnel, Information, Concepts and Doctrine, Organisation, Infrastructure, and Logistics. Frazer-Nash has had success applying the LODs approach to non-defence capabilities.

Each LOD has been used as a prompt, to produce an outline statement of need for each of the LOD characteristics across the capability, together with recommendations of the tools and techniques which might be appropriate to deepen understanding of the characteristics.

Two different LODs have been developed: the first considers the scope of a thermal hydraulics modelling capability (Section 3.2), the second studies the capability associated with a thermal hydraulic test facility (Section 3.3).

3.2 Thermal Hydraulics Modelling Capability

Computational modelling techniques are established as integrated processes throughout modern engineering, science and technology. They are used in a diverse range of applications to produce useful datasets, develop real world understanding, inform design processes and facilitate organisational decision-making. Nuclear thermal hydraulic engineering is no exception, with modelling used at all stages of the planning, design and justification of nuclear facilities.

The modelling capability considered by this report fundamentally consists of the execution of mathematical models in order to simulate performance or investigate fluid flow and heat transfer in a civil nuclear context.

Specific thermal hydraulic simulations tend to be developed and run in support of the development and safety analysis of a particular reactor design. They often use bespoke models developed for a particular reactor and focus on the flow regimes and heat transfer mechanisms important for that reactor technology and geometry. In addition, there are a range of commercially available modelling tools which aim to provide more comprehensive coverage of fluid flow and heat transfer scenarios. Computational Fluid Dynamics (CFD) codes are an example of tools that aim to be applicable to a much more complete subset of fluid dynamic and heat transfer scenarios.

The full spectrum of modelling and simulation tools available from the market is broad and this report defines a set of technical user requirements, which will inform the selection, curation and generation of a capability from these tools. An analysis of the other needs of the capability presented around the lines of development, to expand on the elements required for this equipment to properly function, is presented in Table 6.

Line of Development	Outline of need	Approaches/ tools for consideration
Training	Successful delivery of modelling and simulation requires highly trained and qualified individuals. Minimum training requirements would be a Science, Technology, Engineering and Mathematical (STEM) undergraduate degree or equivalent; with postgraduate qualifications and specialist training in tools and techniques highly likely. Specialist post-graduate MSc courses in both reactor technology and nuclear science and technology are offered by a number of UK Universities. University involvement in training course development is likely to be required in order to accurately capture the state-of-the-art. Career-long Continuous Professional Development (CPD) will be required to keep individuals current with the latest modelling and simulation approaches. New tools, techniques and models created by this project will also require promotion to the rest of the industry. The outputs of the work represent an opportunity to develop training material and courses in themselves.	Training Needs Analysis (TNA) Syllabus development for nuclear thermal hydraulics modelling training courses
Equipment	The technical requirements produced by this project directly contribute to the scope of the ‘equipment’ required to develop and deliver the modelling capability. In this case, equipment has been defined as modelling tools (i.e. software). The development of equipment is likely to be a major part of Phase 2 of this project.	This project will adequately cover the equipment need.
Personnel	The UK has a deficit of professionals with the STEM expertise required to deliver detailed and complex modelling work. The UK Engineering Council has identified this shortfall in a number of reports and has produced guidance and strategy to help to combat it.	UK Government Industrial Strategy Measures in UK Engineering Council report

Line of Development	Outline of need	Approaches/ tools for consideration
	This national shortfall presents risks to the development of the capability in several industries; including nuclear thermal hydraulics. The exact strategy that the UK adopts will affect the scope and scale of the modelling capability needed and therefore number and capability of professionals required, the general shortage will drive up salaries and make recruitment difficult. The UK government's 2017 Industrial Strategy also recognises the lack of STEM professionals and has proposed a variety of measures, including boosting STEM education spending and R&D funding.	
Information	Modelling and simulation is an industry built on the generation and transfer of information with datasets and validated simulation results existing as tradeable entities between different research institutions and organisations. Questions of data ownership and which parties have the rights to exploit the resultant information are likely to be some of the most onerous barriers to generating the collaborative working which is required in order to make progress in this sector. The strategic use of the intellectual property produced as a result of the capability development within the next phase of the project will be an essential precursor to the development of a working capability.	Development of a Nuclear Modelling and Simulation Body of Knowledge could form part of Phase 2 of this project.
Concepts and Doctrine	Modelling techniques are well established within nuclear engineering for providing understanding of reactor performance. First principles research, early stage design and outline performance prediction have had their costs reduced and can be highly iterative when enabled by thermal hydraulic modelling. However, the UK regulatory position is that simulation alone is not sufficient to support robust safety claims and must therefore exist in conjunction with physical testing in order to validate their use in the assurance of nuclear safety. Nevertheless, simulations play an increasing 'support' role in safety arguments and the development of processes to minimise the need for testing would have considerable economic benefits. The potential to make major changes to current modelling techniques and practices is recognised worldwide with the establishment of consortia funded by central government. There will be a need for something similar in the UK to fully realise and expand on the benefits of the first 5 years of the programme.	Development of a Modelling and Simulation roadmap Development of industry wide best practice and uncertainty evaluation procedures could form part of Phase 2 of this project.

Line of Development	Outline of need	Approaches/ tools for consideration
Organisation	The ‘ecosystem’ of organisations that deliver modelling and simulation is diverse: from academic developed modelling capability, through small and medium enterprises and specialist consultancies, to multi-national utilities and large technology developers. Each of these organisations have strengths and weaknesses and bring their own approaches and competencies to the models that they create and run. The UK strategy will need to find a way of engaging all of these different types of organisations and integrating their unique competencies into a holistic capability.	Soft systems methodology to understand the scope and scale of the UK-based ‘nuclear modelling and simulation enterprise’.
Infrastructure	Fixed locations for delivering modelling and simulation are less necessary than they were 10 or 15 years ago. The advent of cloud computing and the ability to hire processing by the hour, along with improved bandwidth on data connections offer alternatives to multi-processor computing facilities (however, fixed infrastructure will still be required for very high-fidelity simulations, for example). Modelling for the purpose of nuclear engineering has, at the very least, restrictions around import/export of information and commercial sensitivities for specific designs. Access to secure UK based high performance computing is needed for UK based industry to make full use of advanced modelling capability. Access to a distributed capability may be appropriate for research purposes.	Registry of suitable security cleared High Performance Computing (HPC) facilities and services
Logistics	The worldwide scope and scale of the nuclear industry, the data focussed nature of modelling and the ability to transfer datasets easily via international data connections, mean that the ability to collaborate internationally is a key element of this capability. Secure transfer of data between facilities is likely to be required to enable the capability.	Clear statement of security requirements and compliance for collaborating facilities.

Table 6: LOD Analysis for a UK Nuclear Thermal Hydraulics Modelling Capability

3.3 Test Facility Capability

Despite the significant advances in the sophistication and fidelity of thermal hydraulic modelling, physical testing of nuclear systems, structures and components is still vital at all lifecycle stages of a reactor design to understand, predict and validate thermal hydraulic performance. The test facilities in which these experiments are run are a key part of the physical and organisational infrastructure relied on by the international nuclear industry. Not only do these facilities provide the space and supporting elements required to run thermal hydraulic experiments, but national facilities are important locations for organisational collaboration, information exchange, training and skill development. A physical test facility can act as the focal point for a nation’s nuclear industry where useful collaboration is fostered and concepts are exchanged and improved. National and local governments may also aim to attract research and development testing to facilities via incentivisation programmes, including tax breaks, co-funding of training and development and direct research grants.

A large number of test facilities are built for and often owned by industry and are intended to support the development of a specific reactor design. Even facilities owned by national institutions in other countries are usually built as part of a wider national programme for design and development activities relating to a specific reactor. It should be acknowledged that building a national test facility in the UK without such a programme represents a significant risk.

This report defines a set of technical user requirements, largely related to the equipment needed by both the facility and the proposed test rigs which will operate within it. An analysis of the other needs of the facility presented around the lines of development, to expand on the elements required for this equipment to properly function, is presented in Table 7. It should be noted that the currently planned programme (as identified in Reference 3) does not include any provision for the completion of these tasks.

Line of Development	Outline of need	Approaches/ tools for consideration
Training	A diverse range of skills are required for test rig development and operation. Deep academic analytical expertise is required to specify, design and deliver programmes of tests. This skill set needs to be augmented by practical, technician-level skills to construct, operate and maintain the experimental set ups.	Training Needs Analysis (TNA) Apprenticeship programme Local college involvement Collaboration with international experts
Equipment	The technical requirements produced by this project will be used to scope much of the equipment (i.e. test rigs and experimental equipment) required to develop and deliver the test capability.	This project will adequately cover the equipment need regarding the test rigs
Personnel	Appropriate staffing of the test facility will be vital to its success. Depending on the operational philosophy that is chosen for the facility, several personnel and resourcing approaches will be required due to the diverse skill requirements. These may include: recruitment of experienced technical and professional staff, STEM graduate recruitment, technical apprenticeships, clerical and secretarial staff. The approach taken to this recruitment (permanent/ temporary staffing) will depend on the operational concept chosen for the facility; a variety of stakeholders will need to be consulted.	Development of a Concept of Operations for the test facility
Information	Data and information resulting from the experiments and testing conducted are the primary output of the facility's activities. A strategy for the logging, processing, security and transmission of	Development of data management strategy Development of partnering framework and contracting

Line of Development	Outline of need	Approaches/ tools for consideration
	the data will need to be developed depending on the specific design and operational concept for the facility. Data ownership and the rights to exploit the resultant information are likely to be some of the most onerous barriers in attracting partners to the facility. An appropriate contractual framework to allow a balance between the needs of partner organisations and the facility will be an important element of the facility's planning.	structure to enable collaboration
Concepts and Doctrine	The operational concept and doctrine, as much as the physical capabilities of the facility, are likely to drive the attractiveness of using the facility for academic and industrial researchers. There are some significant outstanding unknowns regarding the way that this facility will be operated including: ownership, economics, corporate structure, how partnerships will be set up and the operational philosophy that will be taken. All of these will have an effect on the facility's ability to attract investment and partners to run and fund experiments within it. A clear concept of operations for the facility, in addition to the technical specification, will be required to answer these questions and without one it is likely that confusion and delay will result.	Development of a Concept of Operations for the test facility
Organisation	Institutional considerations are likely to drive the organisational structure used to plan and run the facility. Depending on the combination of organisations involved in the planning, the aspirations for the organisation's structure (governmental organisation, university, company, charity, or hybrid entity) are likely to be divergent. There are likely to be advantages and disadvantages to all of the different possible organisational structures which are beyond the scope of Frazer-Nash's advice, but clarity on the operational concept should help with the selection.	Development of an organisational structure in line with the Concept of Operations
Infrastructure	The test facility will be a substantial infrastructure investment; some contributors have suggested additions to the infrastructure beyond the test rigs themselves. The wider infrastructure will be important to the overall success of the facility and so there is a need to consider these aspects in parallel with the test rig design and construction.	Some requirements listed in this report make suggestions for the infrastructure, but further analysis will be needed

Line of Development	Outline of need	Approaches/ tools for consideration
		Additional specification work may be required for peripheral equipment
Logistics	The location of the facility will be important to ensure suitable transport links for equipment and personnel, electrical grid connection, suitable water supply, suitable waste disposal facilities and access to suitably trained and experienced personnel. Secure transfer of data between facilities is also likely to be required to enable the capability.	The M-SParc site will be subject to preliminary review as part of this project, but it may be prudent to review other potential sites as well

Table 7: LOD Analysis for a UK Nuclear Thermal Hydraulics Test Facility

4 Requirements Analysis

In order to understand how the user requirements raised can be developed into functional requirements and ultimately into a roadmap for future development it is helpful to analyse them with regard to how they may contribute a specific benefit to the civil nuclear industry.

Use Cases (UCs) look to define generic ways in which users will interact with the nuclear thermal hydraulics capability. They are intended to set the context of the requirements by describing how the capability will be used and by whom. Further analysis methods will be used to develop these user requirements into functional requirements and recommendations (for example consideration of specific thermal hydraulic phenomena of interest), although this is not within the scope of this document.

Table 8 defines seven Use Cases, which are described further in Sections 4.1.1 to 4.1.7.

UC No	UC Statement	Purposes	Users
UC1	Users design and conduct integral system tests encompassing multiple thermal hydraulic phenomena with high fidelity related to a specific reactor design.	Validating one or more aspects of reactor safety in a way acceptable to Regulators. Providing useful performance data against which reactor design can be optimised. Providing data to either validate or provide input data to system level software codes (i.e. characterise sets of components). Extend knowledge and understanding of thermal hydraulic performance of reactor system components. Provide a test bed for manufacturing techniques, materials or components suitable for use under reactor conditions.	Gen III (+) Reactor Designer GenIV Reactor Designer Fuel Vendor UK Reactor Operator
UC2	Users design and conduct separate effects tests to measure a single thermal hydraulic phenomenon, or the interaction of several different thermal hydraulic phenomena, under controlled conditions, but potentially with no specific requirement to reflect a particular reactor design or geometry.	To improve understanding of a specific thermal hydraulic phenomenon. Validate and/or develop the ability of a modelling code to predict a specific thermal hydraulic phenomenon. Developing physical data sets which can form tradeable Intellectual Property. Extend knowledge and understanding of thermal hydraulic phenomena relevant to nuclear reactors. Extend knowledge and understanding of thermal hydraulic performance of reactor system components. Develop/test instrumentation for measuring thermal hydraulic phenomena.	Gen III (+) Reactor Designer GenIV Reactor Designer Fuel Vendor Researcher

UC No	UC Statement	Purposes	Users
UC3	Users design and conduct tests under representative reactor thermal hydraulic conditions to investigate/ measure an aspect of the performance of a test piece not directly relating to thermal hydraulics.	Investigate specific performance of manufacturing techniques, materials or components where the precise thermal hydraulic conditions are key to the investigation. Provide a general test bed for manufacturing techniques, materials or components intended for use under reactor conditions.	Gen III (+) Reactor Designer GenIV Reactor Designer Fuel Vendor Researcher
UC4	Users develop and/or run thermal hydraulic models to predict the overall performance and response of the reactor primary circuit (e.g. system codes).	Provision of evidence for one or more aspects of reactor safety in a way acceptable to a Regulator. Investigating whole system performance envelope for early designs. Understanding/predicting long transient events including the impact of operator actions and control system responses. To validate and/or improve the model or code to better represent the reactor.	Gen III (+) Reactor Designer GenIV Reactor Designer UK Reactor Operator Regulator Code Developer Provider of Technical Services
UC5	Users develop and/or run thermal hydraulic models representing a sub-section of the reactor core to predict the detailed performance of the core/fuel (e.g. sub-channel codes).	Provision of evidence for one or more aspects of reactor safety, usually relating to fuel performance. Provision of higher fidelity data for input into (or coupling with) a system level code. To validate and/or improve the model or code to better represent the reactor core.	Gen III (+) Reactor Designer GenIV Reactor Designer UK Reactor Operator Regulator Code Developer Provider of Technical Services
UC6	Users develop and execute detailed, high fidelity 3D thermal hydraulic (i.e. CFD) simulations of reactor components.	Optimising component design. Characterising components for input into system or sub-channel level code. Enhancing knowledge and understanding of detailed thermal hydraulics in specific area of reactor. Prediction of complex 3D flow and heat transfer to provide evidence in support of nuclear safety. To validate and/or improve the model or code to better represent the important thermal hydraulic phenomena.	Gen III (+) Reactor Designer GenIV Reactor Designer Fuel Vendor UK Reactor Operator Regulator Code Developer Provider of Technical Services Researchers

UC No	UC Statement	Purposes	Users
UC7	A national facility provides focus for nuclear thermal hydraulics	To provide a mechanism for enhanced collaboration both with nuclear thermal hydraulics and other technical areas. To provide a training vehicle for improved understanding of nuclear thermal hydraulics. To provide a facility for enhanced public awareness of nuclear technology.	Potentially relevant to all users and a number of other stakeholders.

Table 8: Use Cases

4.1.1 UC1

Integral testing is extensively used by the nuclear industry to demonstrate performance of systems of reactor components and assemblies. The tests are performed with the intention of determining the overall effect of multiple thermal hydraulic phenomena. The testing is usually carried out because it is essential to provide evidence to support the nuclear safety claims of the design, although on some occasions it is also used to support design activities. This type of testing also provides data for the development of empirical correlations for use in modelling codes.

These types of tests are usually set up to replicate the scaled geometry and conditions of the specific design of interest as closely as possible. For this reason, the test sections, and often the entire test rig, is generally capable of very limited types of tests and the results are only of relevance to a single reactor design.

The cost of testing is a significant burden on a reactor or fuel vendor. With this in mind many integral test facilities are designed to minimise this cost. Under these conditions, instrumentation is limited to measurements of the most important parameters only (i.e. the overall effect). Often financial, time or space constraints result in a test rig which is unable to replicate the precise thermal hydraulic conditions or physical size. In these cases, the test facility, test conditions and results must be ‘scaled’, although this greatly increases the complexity of using the test results to support nuclear safety.

4.1.2 UC2

Separate effects testing is regularly used as part of research activities to investigate a single (or carefully controlled combination of) thermal hydraulic phenomena. The tests are performed with the intention of increasing understanding and developing modelling methods. Additionally, separate effects testing is used in an industrial context to investigate or confirm the performance of specific components or assemblies under controlled conditions.

In a research environment, the emphasis of these tests is on detailed measurement and observation. The design of the test piece is usually kept as simple as possible to make understanding the results easy. There is only a need to replicate reactor thermal hydraulic conditions or specific geometry if it has a direct impact on the phenomena under investigation. The results are therefore scientifically widely applicable, although the simplifications often mean that they are not directly applicable to any specific reactor design.

In an industrial context separate effects testing often refers to the experimental characterisation of the performance of a component. It is often possible to perform the tests at full scale as only a geometrically small portion of the reactor is represented by the test piece.

Separate effects testing is overwhelmingly the most common type of nuclear thermal hydraulic testing.

4.1.3 UC3

It is important that all components of a nuclear reactor are designed to operate in its specific environment. There are many aspects of this environment that need to be taken into consideration and thermal hydraulics is one area of importance.

A test environment that can replicate the thermal hydraulic environment of a reactor is used to expose components or assemblies to this environment and the results are used to investigate or assure their performance.

These types of tests may be carried out at a variety of scales and are often of a longer duration than required for thermal hydraulic investigation. Depending on the objectives of the test, specifically relevant instrumentation may be required or it may be necessary to examine the test piece after the test is complete.

4.1.4 UC4

In nuclear thermal hydraulics, 'system code' is used as a generic name for modelling tools that are intended to predict the behaviour of a fluid 'circuit' within the reactor and, in many cases, the whole plant. Codes like this are used to predict the performance of the reactor primary circuit and plant components as a whole. The purpose of a system code is primarily to predict the transient thermal hydraulic response of the reactor primary circuit (and the whole plant) to various inputs. They are therefore commonly used to assess start-up and shut-down procedures and the evolution of faults.

System codes usually include a 1-dimensional representation of the flow physics and capture the components of the system at a relatively low fidelity. The low fidelity of systems codes have both advantages and disadvantages:

- Advantages: System codes are able to produce results quickly using commonly available computing hardware; have a limited number of user defined inputs, simplifying quality assurance, validation and repeatability; tend to 'characterise' components rather than explicitly represent the geometry, making modifications quick, and provide the ability to model reactor and plant designs at a very early stage.
- Disadvantages: The low fidelity nature of the models makes them highly dependent on experimentally derived empirical data, which is only valid within certain limits; the model is limited to the prediction of only those parameters considered important when it was created; the model is of limited use for the improvement of thermal hydraulic understanding, and are not able to accurately predict the flow in areas of the reactor and plant components where the low fidelity assumptions are not valid (e.g. areas of poor mixing or counter-current flow).

4.1.5 UC5

In nuclear thermal hydraulics, the term 'sub-channel code' is specifically used to describe modelling tools which represent a subsection of a reactor core, often a single fuel assembly. The model captures the geometry of the fuel at a high level of spatial resolution allowing pin-by-pin temperature predictions. The representation of the physics is still simplified compared with a full CFD model and empirical models are implemented to predict complex flow phenomena (such as turbulent mixing) rather than making high level assumptions about the results of these effects. This results in a method that is dependent on a large number of experimentally derived empirical correlations (such as friction factors and heat transfer coefficients), but produces spatially detailed results.

Sub-channel codes, like system codes, are primarily used in analysis to support nuclear safety claims and predict key performance parameters (such as fuel clad temperature) during normal operation and fault scenarios. They are usually set up and validated for specific reactor/fuel design, making the results easy to validate and repeat.

4.1.6 UC6

The highest fidelity (and most advanced) thermal hydraulic modelling methods employ continuum mechanics equations to accurately capture 3D flows and geometry and solves them for the majority of the relevant physics. Such methods are broadly referred to as CFD in this document. These methods are able to represent (effectively) any component or flow conditions from first principles, however, effort is needed to create a specific model of the component or region of investigation.

These methods often contain aspects of semi-empirical modelling when dealing with highly complex flow phenomena (for example turbulence), but the number of assumptions and dependence on empirical data is much lower than for models that are generally used in UC4 and UC5. This means that, in principle, they can be used to predict thermal hydraulic performance outside of the bounds of existing experimental data, or where experiments are expensive to perform, as well as providing a high level of spatial resolution.

The highly detailed approach requires considerable computing power and time to produce solutions. In addition, the level of freedom in the codes to represent any object means that they require skilled and experienced practitioners to produce accurate predictions. The quality assurance of the results is also more challenging as the general-purpose nature and complexity of the code makes quantifying and understanding the confidence that can be placed in the results difficult. In the civil nuclear industry, these methods are used to study specific areas of the reactor or fuel to improve understanding of thermal hydraulics and develop designs. Their use in analysis to support nuclear safety claims is more to support the systems code results, or provide more detailed information, and Regulators often require specific validation evidence to accompany the models.

4.1.7 UC7

The existence of a national nuclear thermal hydraulics facility would create a focal point for the growth and development of nuclear thermal hydraulics expertise, and a physical location around which to grow and develop the nuclear industrial base. The facility is likely to become an important location for organisational co-operation, information exchange, training and skill development where useful collaboration is fostered and concepts are exchanged and improved.

Although the first priority of the facility is to deliver the technical requirements for testing, opportunities exist for other uses. These could include: using the facility for industry events, for training of undergraduate or postgraduate students or professionals and as a visitor location for increasing public awareness.

The facility and wider supporting infrastructure also has the ability to help sustain co-located services or commercial partners. One of the UK Government’s Catapult organisations, for example, could be placed at the facility with a suitable mission to enhance nuclear thermal hydraulic innovation and drive future economic growth.

5 User Requirements

The outputs of the process of engagement, research and evidence gathering is a set of User Requirements (UR). These define what work is needed to deliver progress or improvement in nuclear thermal hydraulics for each reactor technology considered.

The following three sections present tables of requirements:

- **Section 5.1:** Test Requirements, where the need for thermal hydraulic physical experiments has been articulated, listed per-phenomena or by reference to a specific reactor location.
- **Section 5.2:** Facility Requirements, where requirements for the wider facility infrastructure have emerged as part of the process of gathering Test Requirements[3]. These are not specific to any one test, and a complete set of Facility Requirements can only be defined in detail once a specific test rig has been chosen (i.e. these are user defined Facility Requirements only, not ones resulting from the Test Requirements).
- **Section 5.3:** Modelling and Simulation Requirements, where there is usually a strong association with one or more of the Test Requirements. This is because there is an emphasis on deriving data from experiments to validate modelling and simulation tools.

Each table is divided into segments by reactor technology, and has four columns:

- **UR No:** Each requirement is given a unique identifier to allow traceability of how they are addressed in future. The structure of the identifier is *RRR_X_NN*, where *RRR* is the abbreviation for the reactor technology defined in Table 1, *X* can be either *T, F*, or *M* for 'Test', 'Facility' or 'Modelling', and *NN* is a two digit integer.
- **Statement of Need:** The feature or aspect of thermal hydraulics that is required, in the context of the reactor technology.
- **Justification/Benefit:** The reasons why this requirement exists, what type of benefit this could confer, and how this improves on or provides capabilities that cannot be currently achieved.
- **Linked Use Case:** The 'Use Case' that this requirement is classified as representing (Section 4).

The level of detail and specificity in the requirements varies between the technologies. For technologies that are more mature, where there has been direct feedback from participants in these industries, the requirements are more focussed on resolving particular challenges. Where the technologies are less developed and/or the requirements have been largely derived from the academic literature, they are more high-level and generic.

The requirements listed are the result of the process described in Section 2, and are not intended to represent a comprehensive or exhaustive capture of all possible activities that could be undertaken. They are not prioritised, ranked, or presented in any specific order.

A number of requirements raised in the context of a particular reactor technology are likely to be more generally applicable. Areas of common ground are explored in Section 6.2. In this section, these requirements are left in the context in which they were raised. This does result in some repetition between technologies.

[3] To produce a specification to meet the majority of the 'Facility Requirements', as defined in this section, is beyond the scope of this project. However, they are included as opportunities for future development.

5.1 Test Requirements

UR No	Statement of Need	Justification/Benefit	Linked Use Case
Reactor Technology: Pressurised Water Reactor			
PWR_T_01	The ability to undertake detailed, small scale tests of two-phase boiling flow phenomena to support the development of mechanistic (capturing the physics in detail, rather than just the overall effect) models. This includes the measurement of bubble and film formation and development on very small length scales using the most advanced instrumentation available.	1. Significant effort is currently underway across the community to predict boiling and the critical heat flux. Improvements in this area are to be expected in the coming years, and such advances are required to design safer light water reactors. 2. Correlations are only applicable within the range of conditions for which validation data is available. A more mechanistic approach would enable more accurate predictions of CHF to support nuclear safety under conditions where physical testing is difficult/impossible.	UC2
PWR_T_02	Ability to measure 3D temperature and velocity distributions in single-phase flow in large pool volumes with various distributions of internal heat sources, sinks, adiabatic internals, and various distributions of inlet and outlet locations and flow conditions. Measurement of turbulent fluctuations should be included if possible.	1. This would provide improved benchmark data for the thermal hydraulic predictive tools which could further optimise the plant design and result in greater plant efficiency, reduced operation and maintenance costs, increased profitability and improved cost of electricity. 2. This is of particular importance to passive cooling and decay heat removal and therefore has relevance to improving safety and reducing margins.	UC2
PWR_T_03	The ability to perform detailed measurements of flow within a fuel bundle including measurements of velocity and temperature distribution, pressure drop, heat transfer and turbulence structure. The ability to perform these tests under forward and reversed flow conditions and to provide measurements in local regions of flow reversal.	1. This would provide improved benchmark data for the thermal hydraulic predictive tools, especially CFD and sub-channel modelling which could further optimise the plant design and result in greater plant efficiency.	UC1, UC2

UR No	Statement of Need	Justification/Benefit	Linked Use Case
PWR_T_04	The ability to perform prototypical sub-cooled and saturated nucleate boiling tests (including DNB) and CHF testing.	1. The safety margins associated with fault scenarios such as CHF and DNB are significant. By performing representative tests, confidence in predictions and design will enable the safety margin to be reduced and the economy of the design to be improved. 2. There is currently a lack of CHF experimental data for low flow/low pressure regimes. As such, there is a lack of reference data in these ranges, and reliable correlations have not been developed. The ability to undertake CHF/DNB testing on fuel bundles under conditions more representative to a small PWR (lower pressure and flow rate) would be an enabler for the development of small PWR reactor technologies within the UK. 3. Accident tolerant fuel represents one of the next major steps forward in LWR operation. The ability to test new fuel types would promote UK involvement in this area.	UC2
PWR_T_05	A test facility that can perform high pressure condensation heat transfer testing inside tubes of varying diameters (3/4 in to 2 in). Configurations to include: vertical and horizontal U-tubes; straight and spiral tubes; single tubes and tube bundles.	1. Accurate modelling of high pressure condensation heat transfer is required for reliable predictions of the performance of steam generators and heat exchangers. 2. The acquisition of test data is key to the development and validation of improved modelling methods.	UC2
PWR_T_06	The ability to perform testing to support the development of models to predict crud deposition and its effect on nucleate boiling.	1. The formation of crud can cause detrimental effects in a reactor (e.g. uneven transfer of heat across fuel rods and other components). Ultimately this can cause failure of fuel and other components. It is therefore important that the drivers for the formation of crud, and its effects can be accurately modelled.	UC2
PWR_T_07	The ability to undertake testing to support component qualification and the development of improved flow induced vibration and fretting	1. Flow Induced Vibration (FIV) is often observed in upper core components and fuel bundles within a PWR.	UC1, UC2

UR No	Statement of Need	Justification/Benefit	Linked Use Case
	modelling. This would require the testing of components under representative reactor pressures temperature and flow rates.	2. This capability would improve predictions of component vibration and consequent through-life wear (e.g. grid-to-rod fretting). This will help inform design and operating limits for reactor components and potentially reduce maintenance costs.	
PWR_T_08	Measurements of heat transfer across solid and fluid material layers to support improvements in the methods used to predict conjugate heat transfer across multi-layered components.	1. Aspects of plant, especially the fuel have multiple layers of different materials, often separated with narrow gas gaps. The accurate measurement of heat transfer through these layers is important for the validation of prediction methods.	UC2
PWR_T_09	The ability to perform tests to improve understanding of natural convection within the primary circuit of a reactor. This would include measurement of the circulation rate, identification of flow regime boundaries (such as stall limits) and measurement of surface heat transfer.	1. Natural convection of the primary circuit is used to perform post trip cooling in a number of fault conditions for PWRs. Furthermore, in some SMR designs the primary circuit circulation is driven by natural convection under normal operating conditions. 2. Reliable and accurate measurements are necessary to provide clarity in test results and to validate modelling. Improved confidence in prediction methods would enable reductions in core limit margins and design improvements.	UC1, UC2
PWR_T_10	The ability to perform integral tests to provide validation of systems that provide fault recovery cooling by means of two-phase natural convection i.e. natural convection including boiling and condensation of water.	1. Passive cooling systems have the potential to enhance nuclear safety and reduce costs. However, confidence in their successful operation is lower than for active safety systems as they have been less extensively used/demonstrated and the thermal hydraulics can be significantly more complicated. Tests are required to demonstrate the successful operation of such systems.	UC1
PWR_T_11	The ability to perform prototypical integral testing of CHF and tube dry out for the development of steam generators.	1. Data from tests can be used to support the creation of new or extended correlations to predict steam generator performance. 2. With improved confidence in analysis, the requirement for extensive physical testing can be reduced, thus reducing the significant costs and	UC2

UR No	Statement of Need	Justification/Benefit	Linked Use Case
		durations associated with validation and verification programmes and increasing regulator acceptance.	
PWR_T_12	The ability to perform separate effects tests to improve understanding of the heat transfer rates due to boiling (including CHF and DNB) in steam generators, other heat exchangers, economisers and downcomers.	1. CFD is used to influence decisions on a broad range of typical operation and accident scenarios for the plant. It is therefore important that the model is as accurate as possible. Performing separate effects tests will inform the development of more accurate CFD models which in turn will result in improved confidence in numerical methods and a reduction in safety factors. 2. By improving confidence in analysis, the requirement for extensive physical testing may be reduced, thus reducing the significant costs and durations associated with validation and verification programmes and increasing regulator acceptance.	UC2
PWR_T_13	The ability to perform separate effects tests to understand the impact of surface defects (e.g. increased roughness, cracks, weld beads, etc.) on both near wall and far field flows (including the effect on surface heat transfer).	1. Often both modelling methods and many tests assume a 'perfect' surface instead of modelling the effects of real surface defects. Performing separate effects tests to explore the effect on the flow field will inform the development of more accurate models which in turn will result in improved confidence in numerical methods and a reduction in safety factors. 2. By reducing design margins, the plant could be better optimised to meet its specification, and by improving confidence in analysis the requirement for extensive physical testing may be reduced, thus reducing the significant costs and durations associated with validation and verification programmes.	UC2
PWR_T_14	Access a qualified public data bank for validation of CFD tools and methods for nuclear applications.	1. The applicability of modelling methods is largely limited by what tests have been undertaken to validate them for specific applications. However, experimental data obtained for specific designs is generally propriety information. Obtaining this information is technically challenging, costly and time consuming. A source of more generally	UC2

UR No	Statement of Need	Justification/Benefit	Linked Use Case
		available information would decrease costs and timescales of reactor design.	
PWR_T_15	The ability to perform separate effects tests to better understand the thermal hydraulic drivers behind flow distributions and pressure drop at typical reactor vessel high pressures and temperatures.	1. These tests are needed for the advancement of modelling techniques and tools by providing validation data for specific phenomenological models. 2. Separate effects tests promote improved understanding of specific phenomena thereby enhancing UK and worldwide knowledge.	UC2
PWR_T_16	The ability to perform separate effects tests to enable validation of the prediction of relevant phenomena with CFD codes in a broader range of extreme circumstances specifically relevant to PWR reactor faults.	1. CFD has the potential to greatly improve the detail of analyses; however, confidence in the validity of results is low, especially under less well understood conditions, such as two-phase flow or counter-current flow and mixing. 2. A high pressure high temperature test rig is necessary to appropriately reproduce PWR conditions.	UC2
PWR_T_17	Testing to investigate the effect on DNB and CHF of an atypical fuel assembly geometry or location i.e. fuel bundle at the edge of the core or distorted fuel (fuel bowing).	1. DNB and CHF tests are commonly carried out on a small subsection of a fuel assembly representative of a mid-core condition. Far less test data exists for the validation of conditions close to the edge of the core, or even the edge of a fuel bundle. Equally, less data exists for the analysis of distorted fuel. 2. Improvement in the understanding of heat transfer in a PWR core improves nuclear safety and reduces margins.	UC2
PWR_T_18	The ability to perform separate effects tests to improve understanding and performance of the two-phase separation primary moisture separators and dryers.	1. This would provide test data for system codes and CFD validation thereby improving plant efficiency and turbine performance.	UC2
PWR_T_19	The ability to perform separate effects tests to study thermal mixing and streaming to evaluate	1. This would provide test data for CFD validation and development of the process to evaluate risks of thermal fatigue.	UC2

UR No	Statement of Need	Justification/Benefit	Linked Use Case
	pipe network materials thermal stresses and fatigue.		
PWR_T_20	The ability to perform scaled separate effects tests to study flow and acoustic induced vibrations in pipe networks.	1. This would provide data for design and operational limits of the new and existing pipe networks and possibilities for power uprates. 2. The test data generated could be used for the validation of CFD and acoustic analysis software.	UC2
PWR_T_21	The ability to test reactor coolant pumps in prototypical conditions under various accident scenarios. Data of interest to include pump head, flow rate and endurance. Test variables include distribution of velocity and temperature at inlet and fluctuations in these distributions.	1. Reactor coolant pumps are extremely important PWR components and an accurate picture of the performance and response under accident conditions is vital for nuclear safety. Replicating the relevant conditions in a test is difficult, but valuable for the development and improvement of new designs.	UC2
PWR_T_22	The ability to simulate and measure mixing, circulation and stratification in the containment volume.	1. The containment is a vital part of the plant structure with regard to ensuring safety during accident scenarios. This is needed to evaluate containment conditions, in particular to contribute to the evaluation of containment pressure response.	UC2
PWR_T_23	Separate effects tests to measure condensation within the containment volume.	1. This is needed to evaluate containment pressure response and heat exchanger performance during various accident scenarios.	UC2
PWR_T_24	Scaled separate effects test to evaluate the interaction of the plant with the environment. For example, the influence of wind direction on containment passive cooling.	1. The environment of a plant potentially has an impact on both its safety margins and its performance. This is further complicated if there is more than one plant on a single site. 2. Test data could be used to support the development and validation of environmental modelling.	UC2
PWR_T_25	Ability to test methods to retain a melted core inside the reactor vessel during accident	1. Proving the integrity of the reactor vessel with a melted core requires accurate heat transfer data and the ability to test methods for cooling the vessel.	UC2

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	scenarios. Measurements relating to heat transfer would be required.	2. These tests could support improved designs for emergency systems under these challenging conditions.	
Reactor Technology: Advanced Boiling Water Reactor			
BWR_T_01	The ability to conduct critical power tests on a full-scale ABWR fuel bundle for normal operation and accident scenarios.	1. The safety margins associated with fault scenarios that achieve the critical power ratio for BWRs will be significantly greater than one. By performing representative tests at full-scale, confidence in the design and predictions, particularly those associated with natural circulation and pressure drop in the channel, will enable the safety margin to be reduced and the economy of the design to be improved. 2. By creating a UK test facility with this capability, the UK will increase its practical expertise and understanding of thermal hydraulics. 3. By creating a UK test facility with this capability, the UK will increase its expertise and understanding of the ABWR design in the UK.	UC1
BWR_T_02	The ability to conduct critical power tests on a one-quarter-scale BWR fuel bundle for normal operation and accident scenarios.	1. The safety margins associated with fault scenarios that achieve the critical power ratio for BWRs will be significantly greater than one. By performing representative tests at one-quarter-scale, confidence in the design and predictions, particularly those associated with critical power, will enable the safety margin to be reduced and the economy of the design to be improved. 2. By creating a UK test facility with this capability, the UK will increase its practical expertise and understanding of thermal hydraulics. 3. By creating a UK test facility with this capability, the UK will increase its expertise and understanding of the ABWR design in the UK.	UC1
BWR_T_03	The ability to measure void fraction and distribution using 3D computed tomography during tests.	1. Such measurements are necessary in order to provide a level of clarity in the test results that will help develop the understanding of and confidence in the performance characteristics of the test piece. This could enable design improvements or reductions in safety margins.	UC1, UC2

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BWR_T_04	The ability to measure void fraction using wire mesh sensor technology during tests.	1. Such measurements are necessary in order to provide a level of clarity in the test results that will help develop the understanding of and confidence in the performance characteristics of the test piece. This could enable design improvements or reductions in safety margins.	UC1, UC2
BWR_T_05	The ability to measure film thickness using ultrasonic sensor technology during tests.	1. Such measurements are necessary in order to provide a level of clarity in the test results that will help develop the understanding of and confidence in the performance characteristics of the test piece. This could enable design improvements or reductions in safety margins.	UC1, UC2
BWR_T_06	The ability to measure flow velocity using advanced high resolution laser techniques during tests. This is not an innovative technique in itself, but the capability to do this under high pressure, high temperature conditions is challenging.	1. Such measurements are necessary in order to provide a level of clarity in the test results that will help develop the understanding of and confidence in the performance characteristics of the test piece. This could enable design improvements or reductions in safety margins. 2. By applying this technology in full-scale critical power tests, the UK will be applying cutting edge instrumentation to the rig. This will in turn attract interest from other global research programmes.	UC1, UC2
BWR_T_07	The ability to replicate and measure BWR boiling transition and dryout phenomena over a large scale (i.e. close to full-scale).	1. To improve physical understanding of boiling and film development (in the UK and contribution to world knowledge). 2. To provide test data to validate analytical BWR models. Both of these will lead to improved analytical models resulting in reduced safety margins.	UC2
BWR_T_08	The ability to measure boiling, boiling transition and film development phenomena at microscopic levels using state-of-the-art instrumentation.	1. Such measurements are necessary to better understand the performance limitations of BWRs. By taking microscopic measurements of these phenomena, confidence in reactor designs and predictions will enable the safety margins to be reduced and the economy of the designs to be improved. 2. These measurements can be used to support/validate the development of mechanistic boiling models.	UC2

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		3. A UK-based test rig with state-of-the-art instrumentation would appeal greatly to a number of national and international research programmes.	
BWR_T_09	The ability to conduct critical power testing of a large-scale or full-scale novel fuel bundle using different clad materials or non-uniform axial power distribution.	1. The RBWR uses a novel hexagonal fuel bundle design. By performing representative tests at full-scale, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved. 2. Accident tolerant fuel represents one of the next major steps forward in LWR operation. The ability to test new fuel types would promote UK involvement in this area.	UC2
BWR_T_10	The ability to conduct flow induced vibration testing of a 4.5m RBWR fuel bundle at high water pressures.	1. The RBWR uses a hexagonal fuel bundle design. Flow induced vibration tests have been conducted on less representative geometries in atmospheric pressure. High pressure tests on more representative geometries are now required to provide sufficient confidence in the design and predictions, which should in turn enable the safety margins to be reduced and the economy of the design to be improved.	UC2
BWR_T_11	The ability to conduct large-scale testing of free-surface separation characteristics in the natural circulation driven Double MS (Modular Simplified & Medium Small) Reactor (SMR).	1. Free-surface separation is a key design feature of the Double MS Reactor. To-date only small-scale free-surface separation tests have been performed. Large-scale tests are now required in order to provide sufficient confidence in the design and predictions, which should in turn enable the safety margins to be reduced and the economy of the design to be improved.	UC1
Reactor Technology: Advanced Gas-cooled Reactor			
AGR_T_01	A test facility with the ability to validate 3D flow modelling and heat transfer through boilers. Specifically this would involve measurements of	1. The ability to accurately model gas flow in ageing boilers is required to predict the temperature of boiler components, reduce operating margins, increase reactor power operation and underpin the safety case. Validation data is key to the development of these models.	UC1

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	flow structure and detailed heat transfer as well as the overall component performance.	2. Ageing boilers are one of the main issues associated with AGR lifetime extension. Accurate models and predictions of the behaviour of ageing boilers are therefore necessary to justify life extension across the AGR fleet. 3. All reactor designs that incorporate a secondary circuit require heat exchanges/boilers. The nature of these components always results in complicated 3D flow (to enhance heat transfer). The ability to model and validate these complex flows and resulting heat transfer with confidence would improve initial design optimisation and through life performance.	
AGR_T_02	The ability to undertake tests to validate turbulence models and associated wall modelling for momentum and energy for forced, natural and mixed convection. This will include detailed measurements of the near wall flow structure and resulting heat transfer.	1. The behaviour of the near wall layer presents a number of challenges to numerical methods and modelling. Improvements in the accuracy of wall modelling would result in more reliable models that can be applied to a wider range of scenarios.	UC2
AGR_T_03	A facility to produce measurements to enable validation to improve accuracy in 3D thermal CFD modelling of large structures (recirculation zones, swirl effects, interaction and impingement of jets and plumes); for example to support improvements in turbulence modelling for flows including heat transfer.	1. Despite significant progress over recent decades, turbulence modelling remains an area of uncertainty in all general purpose CFD software. 2. Reducing uncertainty in CFD is key to using its benefits to improve nuclear safety and optimise designs.	UC1, UC2
Reactor Technology: Liquid Metal Fast Reactor			
LMFR_T_01	The ability to measure heat transfer to/from solid surfaces during experiments, including natural, forced and mixed convection, to allow the derivation or validation of heat transfer correlations	1. Heat transfer data enables numerical model development and validation, which can be used to assist reactor designers and support the acceptance of modelling outputs in safety case evidence.	UC1, UC2

UR No	Statement of Need	Justification/Benefit	Linked Use Case
	suitable for prototypical liquid metal reactor components.		
LMFR_T_02	The ability to perform large scale integral effect tests of flow and heat transfer in liquid metal reactor pool configurations.	1. The complex and interrelated components and phenomena that occur within reactors cannot be studied satisfactorily by separate effect tests. Testing them under representative conditions allows, in particular, transients and accidents to be evaluated. 2. Large scale pool tests provide the ability to test whole-core or whole-system behaviours and validate system level and coupled multi-scale models. 3. Includes configurations representative of reactor upper and lower plenums. 4. Reduced scale or more convenient liquids (i.e. water) can be used, but robust arguments can be hard to create to scale to the operational size or fluid, given the inability to match the momentum, viscosity, heat transfer, buoyancy and free surface related dimensionless numbers concurrently.	UC1
LMFR_T_03	The ability to test pressure drop and flow mixing through and heat transfer from fuel assemblies and sub-assemblies under normal operating conditions and fault conditions.	1. Testing of fuel assembly components allows their performance under normal and accident conditions to be quantified and model validation data to be gathered. 2. Tests allow the identification of phenomena that may not be possible to reliably predict with models, such as flow induced vibration, or wear interactions between fuel elements and the surrounding structure.	UC1
LMFR_T_04	The ability to test the behaviour of passive decay heat removal from a pool type reactor, with natural convection operating within the decay heat removal heat exchanger.	1. Natural circulation as a Passive Safety System (PSS) is expected to be a key feature of LMFRs. The ability of passive decay heat removal systems to be relied upon to self-start their natural convection flow and prevent blockage by freezing under all circumstances requires demonstration.	UC1

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LMFR_T_05	The ability to perform tests that replicate flow and equipment configurations that cause gas entrainment into liquid metals, and its accumulation and transport. This should include the ability to test equipment capable of removing the gas.	1. A large quantity of gas crossing the core can cause a positive reactivity change (for example accumulating under the core and all crossing at once) and so is a safety concern. 2. Vortex entrainment of gas occurs in compact configurations for swirling, downward flows. 3. Entrained gas disturbs ultrasonic instrumentation. 4. CFD modelling is possible, but hard to accurately capture the effects and range of entraining conditions. System codes can transport bubbles, but with imperfect modelling. 5. Experiments need to match the dimensionless numbers associated with free surface flow.	UC2
LMFR_T_06	The ability to accurately measure unstable flow mixing processes in pipe junctions, particularly the transient evolution of the inner wall temperatures, and perform post-experiment metallurgical evaluations to identify the life of the components exposed to thermal fatigue.	1. Validation evidence is needed for the varying temperatures experienced near to mixing flows, and data is required for the associated fatigue damage for the pipework caused by this process. These processes are known to cause pipework failures, and this evidence for the life or reliability of pipe systems can guide their design and operation, and provide data for safety assessments. 2. Tests of this process with water and air are of limited value because the effect of the very high thermal conductivity of liquid metals cannot be replicated.	UC2
LMFR_T_07	The ability to test the flow and heat transfer distribution of heat exchangers under representative conditions, including under natural circulation and their interactions with pumps.	1. Primary and secondary heat exchangers need to be optimised, particularly for the uniformity of flow and heat transfer. 2. The effect of buoyancy driven flows is significant for safety critical passive heat removal and natural circulation operation, and predicting how this affects the distribution of flow and heat transfer is important. 3. Flow induced vibration can occur during high flow operation.	UC1

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		4. Freezing of liquid metal coolants is possible during low flow conditions.	
LMFR_T_08	The ability to make measurements of turbulence related quantities in low Prandtl number fluids.	1. Further detailed data are required to help derive and validate CFD turbulence models, particularly related to heat transfer in low Prandtl fluids such as liquid metals.	UC2
LMFR_T_09	The ability to test conventional shaft and impeller liquid metal pumps within the reactor core in representative environments.	1. Liquid metal pumps are integrated within pool type reactors and can be closely coupled to the heat exchangers, making the details of their performance tightly coupled to the core and heat exchanger. This information is needed to demonstrate the performance and reliability of the pumps. 2. The performance and material compatibility of pumps, bearings and seals needs to be demonstrated in representative conditions.	UC1
LMFR_T_10	Tests of main core components of specific design of a candidate reactor design (rather than a generic pool) are necessary to provide details necessary for underwriting the performance of particular designs for operation.	1. Significant testing has been carried out for core components, heat transfer, pumps, water/steam interaction, freezing, core flows in large and small facilities, of loop and pool type. These are not exhaustive for all types of LMFR design however, and design specific tests will be required.	UC1
LMFR_T_11	The ability to test the forces induced by a large displacement of heavy liquid metal within a pool.	1. Displacement and sloshing can occur in response to a seismic event or an explosive steam release, for example steam generator rupture. The forces generated (especially by heavy liquid metals) can be large and are needed to ensure structural integrity of the vessel.	UC2
LMFR_T_12	In a test where the response of fuel assemblies to flow and temperature changes is performed, the ability to measure changes to their relative positon with high accuracy (to within 1/10 mm) is required.	1. Displacements of components within the core must be controlled and measured very accurately because there is a strong coupling between spacing and reactivity. This provides evidence for the passive safety of designs where the core expands with increased temperature, giving negative reactivity feedback.	UC1

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LMFR_T_13	The ability to perform tests on the insertion of control rods, both from above and below into representative coolants.	1. Testing the operation and speed of actuation of safety-critical components in representative conditions would facilitate the qualification of new designs for use. 2. For heavy metals in particular, the buoyancy forces involved are significant. For example, in lead, forcing rods into the coolant from above may be difficult because of its high density.	UC1
LMFR_T_14	The ability to test a 10 MW, 1/6th sector electrically heated representation of the LFR-AS-200 core.	1. 10 MW allows the test to reproduce the decay power. 2. Symmetry of the reactor assembly makes it possible to represent a 1/6 sector without significant loss of information on its behaviour. 3. Full-scale testing and qualification of Spiral Tube Steam Generator (STSG), pump and impeller, instrumentation and decay heat removal at representative conditions (temperature, flows, coolant composition).	UC1
LMFR_T_15	The ability to perform tests that produce plugging of components, such as heat exchangers and pipes, due to the deposition of contaminated coolant.	1. Liquid metal coolants can become contaminated by oxygen corrosion or by alloying with structural materials, and this can deposit in low temperature locations such as heat exchanger tubes – this is known as 'plugging'.	UC3
LMFR_T_16	The ability to test the behaviour of the free surface of a reactor pool and the gas filled void above it. In particular, measurement of gas flow direction, velocity, gas entrainment into liquid metal coolant and heat transfer through the gas void are of interest.	1. The space above the free surface, which is filled with argon gas, is reduced in the new designs of SFRs to reduce the size of the core. This increases the velocity of the free surface, generating stronger vortices and potentially increasing gas entrainment into the sodium. The temperature of the free surface is significantly higher than that of the reactor roof, and hence radiation heat transfer is expected to be strong. 2. These tests will help support the design and development of future pool type liquid metal reactors.	UC1, UC2

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Reactor Technology: Molten Salt Reactor			
MSR_T_01	The ability to undertake performance tests to deliver an improved understanding of the thermofluid properties of molten salts (such as density, viscosity, heat capacity and conductivity). Specifically, the quantification of these parameters across all relevant MSR operating ranges. This includes the development of understanding in the equipment needed to measure these properties accurately.	1. The chemical composition of molten salts is very complicated and the thermofluid properties are not as comprehensively studied as conventional fluid and molten metals. Comprehensive information on the properties of coolant salts is necessary to accurately predict the heat transfer for new designs. This is needed to be able to predict the detailed performance, and demonstrate the safety, of new designs. 2. This information would constitute IP of value to international MSR programmes, thereby increasing the potential contribution of the UK.	UC2
MSR_T_02	The ability to undertake performance tests to deliver an improved understanding of how the thermofluid properties of molten salts are affected by the presence of fuel, fission products and activated coolant. Specifically, the quantification of these parameters across all relevant MSR operating ranges.	1. The presence of the fuel and fission products is known to modify important thermofluid properties in molten salt coolants and introduces further complication. Comprehensive information on the properties of the coolant in the presence of fissile material is necessary for the development of new designs. 2. This information would constitute IP of value to international MSR programmes, thereby increasing the potential contribution of the UK.	UC2
MSR_T_03	An understanding of how the absorption coefficients of molten salts are affected by fuel, coolant and fission products. Specifically, the quantification of these parameters across all relevant MSR operating ranges.	1. Molten salts have significant grey-body emissivity. Though the methodology exists to model this, there is significant variation in absorption coefficient as a function of activated fluid species and this is currently not well understood. Experimental derivation of coefficients is needed to enable an analytical approach to be used to predicting radiative heat transfer. This is needed to enable accurate modelling to further the development of new designs. 2. This information would constitute IP of value to international MSR programmes thereby increasing the potential contribution of the UK.	UC2

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MSR_T_04	Testing to validate the modelling of 3D complex flows in molten salts including turbulence, mixing and melting/solidification.	1. This is required to accurately predict the flow and heat transfer within the reactor for the development and optimisation of the design and to demonstrate key safety functions.	UC1, UC2
MSR_T_05	Testing to produce experimental data for the development of heat transfer correlations for the modelling of molten salt mixtures.	1. The heat transfer behaviour of molten salts is not as comprehensively studied as conventional fluid and molten metals, and the dataset is therefore not as rich. New correlations will be required to accurately predict heat transfer behaviour, develop sub-channel and system level models to underpin safety and design assessments.	UC2
MSR_T_06	Testing to provide data to develop and/or validate models to predict the flow of heat generating fluids.	1. Historically, very little research has been done on the flow of heat generating fluids, and there is a need to better understand the combined effects of buoyancy forces and internal heat generation on the flow of fluids. Without this information, there may be phenomena that are not anticipated or accurately predictable. This understanding is important to the development of new designs and to underpin their safety.	UC1, UC2
MSR_T_07	Testing to provide data to develop and validate models to predict the natural convection behaviour of molten salts.	1. Historically, very little research has been done on the natural convection behaviour of molten salts. Without this information, there may be phenomena that are not anticipated or accurately predictable. Even in a pumped system, prediction of natural convection is required to support nuclear safety assessments, especially under fault conditions.	UC1, UC2
Reactor Technology: Supercritical Water Reactor			
SCWR_T_01	The ability to test a large-scale Supercritical Water Reactor (SCWR) fuel bundle and additional reactor components, 12m total in height and consisting of 25 rods typically, under supercritical conditions in a vertical orientation.	1. The SCWR is an advanced reactor design and one of the six concepts down-selected by the Gen IV International Forum. Confirmatory analyses and experiments have been started on a small scale. Performing representative tests at full-scale will increase confidence in the design and predictions; in particular those associated	UC1

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		with natural circulation. As a result, this should enable the economy of the design to be improved and support regulatory acceptance.	
SCWR_T_02	Ideally, the ability to test a large-scale SCWR fuel bundle, 4.5m long and consisting of 25 rods typically, under supercritical conditions in a horizontal orientation.	1. This requirement is necessary to remove the effects of buoyancy from the tests such that effects can be examined separately. By performing such tests, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved.	UC1
SCWR_T_03	The ability to perform heat transfer tests at supercritical water pressures.	1. Due to the large variation in the thermophysical properties of supercritical water, it is challenging to predict the heat transfer coefficient by conventional methods. Experimental data is needed to support the development of reliable prediction methods to support new designs and underpin safety.	UC2
SCWR_T_04	The ability to perform tests at low pressures, temperatures and flow rates as well as high pressures, temperatures and flow rates.	1. This will increase the flexibility of the test facility, making it potentially of benefit to equivalent Gen III or Gen III+ light-water designs. For example, they could use the data collected to improve fuel economy or support further design iterations. 2. Low pressures, temperatures and flow rates would extend the flexibility of the rig to cover fault conditions and natural circulation.	UC1, UC2
SCWR_T_05	The ability to test the performance of components, e.g. pumps and valves, in a vertical orientation.	1. The SCWR is an advanced reactor design and one of the six concepts down-selected by the Gen IV International Forum. Components, including pumps and valves, will be positioned and orientated in such a way that performance is optimised. By performing these tests, the reactor designer will better understand the performance of each test article, which will lead to more informed design decisions and improved reactor performance.	UC1
SCWR_T_06	The ability to test the performance of components, e.g. pumps and valves, in a horizontal orientation.	1. The SCWR is an advanced reactor design and one of the six concepts down-selected by the Gen IV International Forum.	UC1

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		Components, including pumps and valves, will be positioned and orientated in such a way that performance is optimised. By performing these tests, the reactor designer will better understand the performance of each test article, which will lead to more informed design decisions and improved reactor performance.	
SCWR_T_07	The ability to measure the turbulence structure in complex fuel bundle geometry during tests.	1. Heat transfer from fuel bundles to coolant is an area of reactor performance that is still subject to uncertainty. One of the contributing factors to this is uncertainty in the prediction of turbulence. By measuring the turbulence structure within complex fuel bundle geometries, designers will increase their understanding of the flow within the fuel bundle geometry. In turn, this will lead to the development of more representative turbulence models, more informed design decisions and improved reactor performance.	UC1, UC2
Reactor Technology: High Temperature Gas-cooled Reactor			
HTGR_T_01	Experiments and tests to give improved understanding of the phenomenology of air-ingress accident scenarios.	1. A critical event in the safety analysis of VHTRs and GFRs is the air-ingress scenario, where there is a primary circuit break, loss of coolant and the high temperature components are exposed to air, leading to structural damage. Limited experimental data is available to validate models and understand the phenomenology of the accident scenario. 2. The behaviour of stratification during the air-ingress process is not well understood and additional data would be beneficial to the design and safety analysis of this reactor type.	UC1
HTGR_T_02	The ability to undertake integral tests to determine the conditions for core structural failures due to reactions and material degradation during an air-ingress accident.	1. Following an air-ingress accident, oxidation of core components (for example) can cause structural failure. Predicting the likelihood or time taken for this to occur allows the reactor safety analysis to be performed.	UC1

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HTGR_T_03	The ability to perform tests on components exposed to high velocity gas flows under accident conditions.	1. During LOCA conditions, where there is a rapid depressurisation of the reactor, the coolant flow velocities will be very high, potentially damaging components with high aerothermal loads. Tests of this would provide direct evidence of the loads, and would provide validation evidence for models.	UC1, UC2
HTGR_T_04	The ability to provide test data for the development of improved heat transfer correlations for specific HTGR geometries (core, dedicated heat exchangers) using relevant coolants (for example helium or supercritical CO_2) under operational conditions.	1. System and sub-channel codes use heat transfer correlations to predict fuel temperatures in the core. Test data is needed to improve correlations specific to the reactor geometries, and will enable improved designs of fuel elements and flow channels, and result in better safety analyses. 2. Test results would provide validation evidence for high fidelity CFD codes.	UC2
HTGR_T_05	The ability to provide test data to support the development of improved predictions of friction loss in the core and coolant circuits under operational conditions.	1. System and sub-channel codes use friction factor correlations to predict flows in the core. Improved correlations specific to the reactor geometries will enable improved designs of fuel elements and flow channels, and result in better safety analyses. 2. Gas reactors typically have high flow velocities, hence high frictional losses, leading to high circulator power requirements and difficulties in obtaining a desired flow balance between core components, increased confidence in the friction factors would improve this. 3. Test results provide validation evidence for high fidelity CFD codes.	UC2
HTGR_T_06	The production of heat transfer and friction factor test data in flow conditions relevant to natural convection in HTGR components. This is expected to be in the 'mixed' convection regime, where buoyancy plays an important role, and affects the process of transition from laminar to turbulent flow.	1. The ability to understand and predict heat transfer across all relevant flow regimes is required for accurate prediction of thermal hydraulic behaviour and phenomena. This will form the basis for safe operating and design limits for the reactor and its components, and will inform the development of the most efficient designs.	UC2

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		2. In particular, the peak clad temperature occurring during a loss of flow accident requires accuracy in the transition, mixed convection regime. 3. Existing correlations are known to be inaccurate by up to a factor of 2 in some flow/fluid regimes, leading to errors of approximately 100°C in predicted peak clad temperatures in loss of flow accidents. 4. This data can be used to provide data to develop correlations and as validation data for thermal hydraulic models. 5. A representation of the uncertainties in the correlation would allow this to be reflected in the correlation derived from it.	
HTGR_T_07	The ability to undertake tests to measure the flow patterns (in terms of velocity and temperature distribution) in prototypic configurations of plenums where the hot gas from fuel channels mix.	1. Due to complex flow patterns at the core outlet, it is possible to induce hot spots in the plenum structure. Inadequate mixing of turbulent jets that impinge on core support structures can generate unacceptable thermo-mechanical loadings on them. 2. Tests can provide validation data for CFD simulations of these configurations. These are known to be challenging phenomena for CFD to predict.	UC1, UC2
HTGR_T_08	The ability to investigate the effect on heat transfer and friction of fuel assembly and flow control/support structure roughness and erosion in a test loop that represents system effects.	1. The use of gas coolants generally requires artificial roughening of the cladding to enhance heat transfer and maintain acceptable cladding temperature, resulting in an increased pressure drop over the core, and a higher requirement on circulator power. 2. To maintain uniform core outlet temperatures, either each fuel assembly needs to have an adjustable flow gag, or the amount of roughening of the cladding needs to be varied. 3. The friction factor and heat transfer on surfaces is substantially affected by roughness, and its inclusion in correlations or	UC1, UC2

UR No	Statement of Need	Justification/Benefit	Linked Use Case
		representations by CFD is an important contributor to model accuracy and the reduction of uncertainty. 4. Dust particles in high speed flow can erode structural components.	
HTGR_T_09	The ability to undertake natural convection tests at full scale in order to improve knowledge of the stability, reliability and performance of natural convection in core flows and decay heat removal systems.	1. Following the removal of coolant and forced cooling from a reactor during an air-ingress accident, the time taken for natural convection to be established will allow assessment of the transient temperature response of the fuel. 2. The operation of decay heat removal systems is ensured by natural convection. This is a key safety function, so it is necessary to understand the reliability and performance. This will form the basis of safety assessments, and design and operating margins. 3. Flow instability and stagnation in hot channels under natural circulation can result from increasing coolant viscosity with increasing temperature. This can disrupt the successful performance of systems that depend on natural circulation and are therefore key areas for investigation. 4. A sufficiently representative prototypical geometry would also allow system code validation data to be obtained for flow performance in core and bypass flow paths.	UC1
HTGR_T_10	The ability to perform long term tests of the erosion and contamination of pipe networks due to the presence of entrained particulates.	1. The presence of dust in the primary circuit leads to deposition and potentially to damage by erosion in components exposed to high velocity flow. 2. The deposition of dust can lead to reduced performance and/or heat transfer (potentially affecting safety margins). The evaluation of deposition and re-entrainment, especially in areas of non-uniform flow, will enable improved plant performance and reduced maintenance.	UC2

UR No	Statement of Need	Justification/Benefit	Linked Use Case
Cross-Cutting			
CC_T_01	The ability to test advanced materials at sub-assembly and component levels under prototypical thermal hydraulic conditions.	1. To support qualification of advanced materials in nuclear design. 2. To provide through-life assurance of modular build.	UC3
CC_T_02	The ability to test new joining process (e.g. new types of weld) at sub-assembly and component levels under prototypical thermal hydraulic conditions.	1. To support qualification of alternative manufacturing processes in nuclear design. 2. To provide through-life assurance of modular build.	UC3
CC_T_03	The ability to test advanced manufacturing processes (e.g. additive layer manufacturing techniques) at sub-assembly and component levels under prototypical thermal hydraulic conditions.	1. To support qualification of advanced manufacturing processes in nuclear design.	UC3
CC_T_04	The ability to test large-scale machined components under prototypical thermal hydraulic conditions.	1. To support qualification of large-scale machined components in nuclear design.	UC3
CC_T_05	The ability to test measurement techniques for fluid collapsed level and void distribution by multiple methods under prototypical thermal hydraulic conditions for light water reactors.	1. To support the development of reactor test measurement and monitoring equipment. 2. To assist with generic design assessment of light water reactors.	UC3
CC_T_06	The ability to capture data on material degradation at sample and component level in pressurised-water reactors and high-temperature reactors under prototypical thermal hydraulic conditions.	1. To increase understanding of compatibility of components and coolants.	UC3
CC_T_07	The ability to test components that have been repaired using additive layer manufacturing	1. To support qualification of repaired components and understand their effects on thermal hydraulics	UC3

UR No	Statement of Need	Justification/Benefit	Linked Use Case
	processes under prototypical thermal hydraulic conditions.		
CC_T_08	The ability to capture data using advanced sensing technology under prototypical thermal hydraulic conditions.	1. To support the qualification of advanced sensing technology from other industries. 2. To understand the effects of advanced sensing technologies on thermal hydraulics.	UC3
CC_T_09	The ability to test accident-tolerant fuel cladding material under prototypical off-normal thermal hydraulic conditions.	1. To support the qualification of accident-tolerant fuels.	UC3
CC_T_10	The ability to perform corrosion-driven-fatigue endurance testing consisting of thermal cycling with rapid rates of temperature change under flowing water conditions on 4" nominal bore pipe.	1. The integrity of the validation data for this phenomenon has recently been brought into question by the US Nuclear Regulatory Commission (NRC). The Electrical Power Research Institute (EPRI) has a working group looking at addressing the resulting shortfalls in safety cases. Performing such tests at the UK facility would provide an opportunity for international cross-cutting collaboration with an international research institute and regulator. In addition, shortfalls in active safety cases would be addressed.	UC3
CC_T_11	The ability to perform thermal-striping-driven-fatigue endurance testing	1. Thermal-striping could be an issue for some Gen IV designs, particularly fast reactors with liquid metal coolants which have high thermal conductivity. By performing such tests, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved.	UC3
CC_T_12	The ability to test the capture of diagnostic data on the performance of components and sub-assemblies under prototypical thermal hydraulic conditions.	1. This type of technology is used in gas turbines to look for real-time symptoms and correlations. It has not yet been adopted in the nuclear industry. The UK facility could be used to qualify diagnostic and data monitoring technology for use across the nuclear industry.	UC3

5.2 Facility Requirements

UR No	Statement of Need	Justification/Benefit
Reactor Technology: Pressurised Water Reactor		
PWR_F_01	The development of UK test service expertise for the design, construction, development and operation of test facilities, rigs and prototypes.	1. Accessibility and availability of thermal-hydraulic test facilities and appropriately skilled and experienced engineers is a major challenge for new PWR SMR designs.
PWR_F_02	A test facility that can accommodate multiple temporary, integral effects test rigs to support reactor design and substantiation. The test bays should be high enough to be able to accommodate full scale components.	1. All reactor designs need integral effects test rigs to substantiate important areas of the design. These rigs are design specific and are therefore only of value to one manufacturer. However, they are an expensive and important part of the reactor design process. A multi-purpose built facility would help to reduce costs and encourage development in the UK.
PWR_F_03	A test facility containing physical services with standardised connections: steam, water, air, electricity configured to enable a 'plug and play' environment. Water pressure of up to 22MPa and water temperature of up to 370°C required.	1. The supply of these services reduces overall costs for those wishing to use the facility (i.e. they do not have to design and build the equipment to supply these themselves). 2. A 'plug and play' philosophy would greatly improve the usability by enabling other rigs to be designed to 'fit' the bays in the future; reusing the existing infrastructure.
PWR_F_04	A test facility with small scale / laboratory space for multiple scaled separate effects test rigs configured with the same 'plug and play' philosophy as the large scale bays	1. It is often easier to perform separate effects tests on a small scale, enabling detailed measurements to be taken under more easily controlled conditions. 2. Small scale lab tests are good enough for many separate effects tests and are much cheaper to design, build and execute.
PWR_F_05	A test facility including wider services to support testing including: Clean room environment; storage facilities; handling equipment (e.g. gantry cranes); data logging, management and communication; waste management; good transport links (road and rail); accommodation for visiting engineers; infrastructure allowing for physical security and segregation between users.	1. In addition to the thermal hydraulic technical requirements of the tests, other facility attributes need to be considered. Some are essential, but other just make it more attractive to potential users.

UR No	Statement of Need	Justification/Benefit
PWR_F_06	A test facility with the ability to capture high frequency data (10Hz) from advanced instrumentation.	1. High frequency data logging is needed to capture evolution of transient or unsteady phenomena.
PWR_F_07	Computing hardware and software to enable thermal hydraulic modelling and visualisation of results on site.	1. Analysis and visualisation capability will enable direct comparison with analytical methods and a more integrated process of testing and model development.
PWR_F_08	A test facility with on-site rig design and construction capability including rapid prototyping equipment.	1. This would enable modification to rig design and test samples to be made on site in an interactive manner.
PWR_F_09	A test facility with on-site instrumentation services such as a calibration and quality assurance team.	1. State of the art instrumentation is not enough in itself to ensure high quality measurements. Instruments need to be maintained and calibrated to be useful.
Reactor Technology: Advanced Boiling Water Reactor		
BWR_F_01	Test facility infrastructure that can supply 15MW of power to a test rig.	1. The safety margins associated with fault scenarios that achieve the critical power ratio for BWRs will be a significantly greater than one. This constraint requirement is necessary to perform representative tests at full-scale. By performing such tests, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved. 2. By creating a UK test facility with this capability, the UK will increase its practical expertise and understanding of thermal hydraulics. 3. By creating a UK test facility with this capability, the UK will increase its expertise and understanding of the ABWR design in the UK.
BWR_F_02	Test facility infrastructure that can supply water at a pressure of 10MPa to a test rig.	1. The safety margins associated with fault scenarios that achieve the critical power ratio for BWRs will be a significantly greater than one. This constraint requirement is necessary to perform representative tests at full-scale. By performing such tests, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved. 2. By creating a UK test facility with this capability, the UK will increase its practical expertise and understanding of thermal hydraulics.

UR No	Statement of Need	Justification/Benefit
		3. By creating a UK test facility with this capability, the UK will increase its expertise and understanding of the ABWR design in the UK.
BWR_F_03	Test facility infrastructure that can supply water at a flow rate of 90 tonnes per hour to a test rig (equivalent to 25 kg/s).	1. The safety margins associated with fault scenarios that achieve the critical power ratio for BWRs will be a significantly greater than one. This constraint requirement is necessary to perform representative tests at full-scale. By performing such tests, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved. 2. By creating a UK test facility with this capability, the UK will increase its practical expertise and understanding of thermal hydraulics. 3. By creating a UK test facility with this capability, the UK will increase its expertise and understanding of the ABWR design in the UK.
BWR_F_04	Test facility infrastructure that can house a 15m high test rig.	1. The safety margins associated with fault scenarios that achieve the critical power ratio for BWRs will be a significantly greater than one. This constraint requirement is necessary to perform representative tests at full-scale; in particular to demonstrate natural circulation. By performing such tests, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved.
BWR_F_05	Test facility infrastructure that can install a 15m high test rig.	1. The safety margins associated with fault scenarios that achieve the critical power ratio for BWRs will be a significantly greater than one. This constraint requirement is necessary to locate a rig that is able to perform representative tests at full-scale; in particular to demonstrate natural circulation. By performing such tests, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved.
Reactor Technology: Advanced Gas-cooled Reactor		
No facility-specific requirements were identified.		

UR No	Statement of Need	Justification/Benefit
Reactor Technology: Liquid Metal Fast Reactor		
LMFR_F_01	The ability to melt, store, transfer and filter liquid metal coolants (including heavy metals such as lead), and control impurities and dissolved oxygen.	1. Liquid metal coolants will need to be introduced and removed from a test rig as part of the process of configuring tests. 2. Maintaining the composition and purity of the coolant dissolved oxygen is important to obtain useful and representative results.
LMFR_F_02	The ability for a facility crane to lift an LFR-AS-200 assembly (15 ton, 10m lift)	1. The lifting capability allows the easy removal of the key components of the test, for modifications, replacement with a different variant.
Reactor Technology: Molten Salt Reactor		
MSR_F_01	A test facility infrastructure that can accommodate high fluid temperatures up to 900°C (higher temperatures may be required for other designs).	1. This would enable test data to be obtained under representative conditions that can be used to underpin key safety functions.
MSR_F_02	A test facility infrastructure that can inert and redox control molten salt fluids.	1. The redox control of molten salts in a reactor environment is needed to manage corrosion. This would also be needed to manage the fluid in a test environment to prevent damage to the test facility and contamination of the fluid thereby changing its thermophysical properties and compromising the tests. 2. The redox control of molten salts is important in a test environment as dissolved atmospheric constituents have the potential to modify the thermophysical properties thereby compromising the tests. 3. The redox control of salts in a test facility gives the opportunity to test new methods of redox control.
MSR_F_03	A test facility that can accommodate large scale test rigs (up to 10.5m high, 3m by 6m area)	1. Full-scale testing of key reactor components is currently required for regulator acceptance. 2. Test facilities that can accommodate the height required for full scale testing are unusual and this would increase the appeal of the UK test facility to national and international users.

UR No	Statement of Need	Justification/Benefit
MSR_F_04	A test facility with an infrastructure that can supply power at 10MW	1. This would enable testing of a representative section of the core at power.
MSR_F_05	A test facility that can handle and store HA and Pu based fuels in rig and on bench and reprocess/synthesise artificial fuel types on site.	1. This would enable tests involving true fuel/salt mixtures and would prevent regulatory difficulties associated with transporting material to and from the test facility. 2. This capability would also be of benefit to support other fast reactor programmes thereby increasing the appeal of the facility to a wider range of users.
MSR_F_06	A test facility that can be applicable to a broad range of technologies and which is specified to meet the needs of the future.	1. Specifying the facility for a specific technology could skew the economic landscape. By the time the facility is operational, the technological challenges will have moved on. Considering flexibility and future needs in the test facility design will mitigate against it being out of date before it is built and maximise the scope of potential users.
Reactor Technology: Supercritical Water Reactor		
SCWR_F_01	Test facility infrastructure that can supply power up to 14MW to a test rig.	1. This constraint requirement is necessary to perform representative tests at full scale, in particular to supply sufficient energy to the rig to reach supercritical conditions and supply power to electrically heated test fuel bundles. 2. This constraint requirement bounds the requirement of an equivalent Gen III or Gen III+ light-water rig. Therefore, by creating a UK test facility with this capability, Gen III and Gen III+ designers and operators could benefit.
SCWR_F_02	Test facility infrastructure that can supply water at pressures up to 28MPa to a test rig.	1. This constraint requirement is necessary to perform representative tests at full scale, in particular to supply sufficient pressure to the rig to reach supercritical conditions. By performing such tests, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved. 2. This constraint requirement bounds the requirement of an equivalent Gen III or Gen III+ light-water rig. Therefore, by creating a UK test facility with this capability the potential number of users is increased.

UR No	Statement of Need	Justification/Benefit
SCWR_F_03	Test facility infrastructure that can supply water at flow rates up to 25 kg per second to a test rig.	1. This constraint requirement is necessary to perform representative tests at full scale, in particular to supply a flow rate that is representative of reactor conditions. By performing such tests, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved. 2. This constraint requirement bounds the requirement of an equivalent Gen III or Gen III+ light-water rig. Therefore, by creating a UK test facility with this capability the potential number of users is increased.
SCWR_F_04	Test facility infrastructure that can house a 15m high test rig.	1. This constraint requirement is necessary to perform representative tests at full scale, in particular to supply sufficient pressure to the rig to reach supercritical conditions. By performing such tests, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved. 2. This constraint requirement bounds the requirement of an equivalent Gen III or Gen III+ light-water rig. Therefore, by creating a UK test facility with this capability, Gen III and Gen III+ designers and operators could benefit. For example, they could use the data collected to improve fuel economy or support ageing management.
SCWR_F_05	Test facility infrastructure that can install a 15m high test rig.	1. This constraint requirement is necessary to perform representative tests at full scale, in particular to accurately recreate the conditions under which natural circulation is expected or required. By performing such tests, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved. This constraint requirement bounds the requirement of an equivalent Gen III or Gen III+ light-water rig. Therefore, by creating a UK test facility with this capability the potential number of users is increased.
Reactor Technology: High Temperature Gas-cooled Reactor		
No facility-specific requirements were identified.		

UR No	Statement of Need	Justification/Benefit
Cross-Cutting		
CC_F_01	The ability to visualise test rig assemblies using Virtual Reality (VR).	1. To demonstrate VR assembly flow modelling capability in nuclear design.
CC_F_02	The ability to share knowledge with the proposed Safety Centre of Excellence.	1. To promote dissemination of information to industry stakeholders. 2. To focus new or follow-on R&D programmes within the test facility on aspects which enhance nuclear safety. 3. Associated with Use Case 7.
CC_F_03	The opportunity to perform site tours of test facility with members of the public.	1. To support public engagement with the civil nuclear industry. 2. Associated with Use Case 7.

5.3 Modelling and Simulation Requirements

UR No	Statement of Need	Justification/Benefit	Linked Use Case
Reactor Technology: Pressurised Water Reactor			
PWR_M_01	Improved ability to accurately model multiphase flows using CFD. This includes the ability to model gas and liquid films and flow involving droplets in steam. Parameters that need to be accurately predicted are velocity and temperature distribution, pressure drops over components and heat transfer to/from surfaces.	1. Multiphase flows, although central to the majority of PWR thermal hydraulic safety issues, are not yet completely understood or modelled with sufficient accuracy. Therefore, ongoing research will support nuclear safety and reduce margins.	UC6
PWR_M_02	The ability to predict rapid steam generation i.e. flashing. In particular, the rates of steam generation, over pressurisation and associated heat transfer processes.	1. A number of the fault recovery systems on a PWR involve the spraying of water onto hot surfaces. This causes flashing and rapid steam generation, as does rapid depressurisation of a PWR. A more accurate prediction capability would enhance safety and reduce margins.	UC4, UC5, UC6
PWR_M_03	The ability to mechanistically model boiling and predict CHF on fuel bundle surfaces and within porous crud structures.	1. Significant effort is currently underway across the community to predict boiling and the critical heat flux. Improvements in this area are to be expected in the coming years, and such advances are required to design safer light water reactors. 2. Correlations are only applicable within the range of conditions for which validation data is available. A more mechanistic approach would enable more accurate predictions of CHF to support nuclear safety under conditions where physical testing is difficult/impossible.	UC6
PWR_M_04	Improvement in the ability to predict the total water inventory and the reactor liquid level following a LOCA.	1. Water level is a key parameter in fault recovery under LOCA conditions and it is not feasible to test large scale, whole reactor faults directly. Improvement in the accuracy of predictions improves safety and reduces margins.	UC4

UR No	Statement of Need	Justification/Benefit	Linked Use Case
PWR_M_05	Improvement in the modelling of 3D, single phase turbulent flow, especially with regard to heat transfer and mixing through geometrically complicated areas of the core (e.g. spacer grids) and in plenums. Key predictions would include transient evolution of velocity and temperature distribution.	1. The accurate prediction of single-phase flow and mixing is important to many aspects of a PWR including: fuel bundle heat transfer and spacer grid performance, upper and low plenum uniformity, thermal streaming potentially leading to solid component fatigue. 2. At present, most models included in CFD codes are empirical and this limits their applicability to specific ranges of parameters. Further development of turbulence modelling will provide more accurate predictions under a full range of reactor conditions, to support reactor design and nuclear safety.	UC6
PWR_M_06	Improvement in the ability to predict 3D PWR core flows on a large scale i.e. prediction of mixing and crossflow between channels in a PWR core.	1. Whole core modelling is normally carried out with reduced order codes that do not fully predict the 3D flow. Improvement in the ability to predict mixing between channels would improve the accuracy of whole core modelling and help identify factors causing instability and flow induced vibration. 2. Whole core modelling is essential to the assurance of both the performance and safety of a PWR, as tests can only be conducted on smaller scales.	UC4, UC5, UC6
PWR_M_07	Improvement in the ability to model multi-species flow mixing, using both 3D and lower fidelity methods.	1. Fault recovery in a PWR can involve the injection of boron. The distribution of boron is important for reactor control under fault conditions and therefore nuclear safety.	UC4, UC5, UC6
PWR_M_08	Improvement in the modelling of the mixing of streams of fluid (water or steam) of different temperatures.	1. In a PWR primary circuit there are places where hot and cold fluid streams interact. Inadequately mixed flow can lead to thermal fatigue of reactor components.	UC6
PWR_M_09	The ability to accurately model/predict DNB and CHF (using correlations) for PWR fuel bundles under	1. Circular tubes have been used as a basis for most of the CHF models, correlations and experiments. However, the thermohydraulic behaviour in sub-channel systems is influenced by numerous other	UC4, UC5

UR No	Statement of Need	Justification/Benefit	Linked Use Case
	an extended range of PWR conditions (i.e. beyond those for which data is currently available).	complicated effects and interactions. This cannot yet be accurately modelled and limits the confidence in those correlations. 2. Accurately predicting CHF at low flow/low pressure regimes is required for fault scenarios (such as LOCA) and for operating regimes relevant to SMRs. This is needed to support nuclear safety.	
PWR_M_10	The ability to accurately model and predict condensation heat transfer in the presence of non-condensable gases.	1. Accurate modelling of condensation heat transfer is required for reliable predictions of two-phase vessel passive cooling system performance. This is required to underpin safety functions, and would define safe operating and design limits for the core. 2. A mixture of non-condensable gases and steam represents a particular challenge, as the non-condensable gases impede the access of the steam to the cool surfaces.	UC4, UC5, UC6
PWR_M_11	The ability to model coupled two-phase thermal hydraulics and species concentration to improve predictions of drivers of crud deposition throughout the primary circuit; particularly deposition within the core.	1. The formation of crud can cause detrimental effects in a reactor (e.g. uneven transfer of heat across fuel rods and other components). Ultimately this can cause failure of fuel and other components. It is therefore important the drivers for the formation of crud, and its effects, can be accurately modelled.	UC6
PWR_M_12	The ability to modelling flow induced vibration and fluid structure interaction with fuel bundles and upper core components under a range of 'normal' operating conditions.	1. Flow induced vibration is often observed in upper core components and fuel bundles within a PWR. 2. This capability would improve predictions of component vibration and consequent through-life wear (e.g. grid-to-rod fretting). This will help inform design and operating limits for reactor components and potentially reduce maintenance costs.	UC6
PWR_M_13	Improvements in the methods used to predict conjugate heat transfer across multi-layered components including fluid layers.	1. Aspects of plant, especially the fuel, have multiple layers of different materials, often separated with narrow gas/fluid gaps. The accurate prediction of heat transfer through these layers is important for both nuclear safety and plant performance.	UC5, UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
PWR_M_14	The ability to model coupled multi-physics i.e. thermohydraulics, neutronics and mechanical effects in a coupled manner such that there is the capability to accurately model flow, temperature, power and structural response during transients.	1. Accurate, coupled, multi-physics, multi-scale tools would enable improved estimates of the performance of key areas of the reactor (e.g. fuel). These results would help optimise in-service inspection, maintenance and refuelling outage frequency and could serve to inform plant diagnostics tools. 2. Integration would allow for quicker design iterations and additional detail to be analysed at an early stage in the development of new designs. 3. Specifically the coupling of thermal and neutronics modelling enables the effect of temperature on the reaction rate to be predicted, increasing the accuracy with which key safety predictions can be made. 4. The net result would be greater plant efficiency, reduced operation and maintenance costs, increased profitability (reduced cost of electricity).	UC6
PWR_M_15	The ability to accurately predict phenomena of interest to low flow systems, such as: fluid stratification, counter-current flow, natural-convection and heat transfer in large pools.	1. The ability of modelling codes to predict naturally driven, slow flow (especially at the limits of its stability) is less well developed than that of faster, pumped flow. As it has advantages for reduced complexity and enhanced nuclear safety to use passive flow systems, research in this area would have a direct benefit on current and future reactor types.	UC5, UC6
PWR_M_16	The ability to predict two-phase natural convection in prototypical containment volumes and pressures, including measurement of condensation heat transfer.	1. In fault recovery, some PWR systems include the ability to cool the outside of the pressure vessel by means of natural circulation of boiling water and condensing water. However, as it is difficult to predict or validate the heat transfer achieved, it is difficult to claim the benefits of the system for nuclear safety purposes. More confidence in	UC4, UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
		the performance of this type of natural circulation would have benefits for passive safety.	
PWR_M_17	Development of further correlations to accurately predict heat transfer and tube dryout in steam generators.	1. Empirical correlations can only be reliably applied to the range of scenarios from which experimental data was obtained. Predicting performance for new designs therefore requires obtaining more experimental data and developing new correlations for the specific design. 2. Heat transfer phenomena impose limits in the design and operation of boiling heat transfer equipment. Improved accuracy and confidence in modelling would enable reductions in safety margins and more efficient designs.	UC4
PWR_M_18	To improve ability to accurately model and predict specific phenomena relevant to heat transfer in steam generators with CFD.	1. Steam generators often include highly complex geometry and 3D flows. The ability to predict these flows with CFD would improve understanding of the design and how to improve it (rather than just characterising the design with correlations). 2. Heat transfer phenomena impose limits in the design and operation of boiling heat transfer equipment. Improved accuracy in modelling would enable reductions in safety margins and more efficient designs.	UC6
PWR_M_19	The development of system codes with the ability to model transient behaviour for 'non-power' reactor applications, such as desalination and hydrogen production.	1. The ability to accurately model and predict transient behaviour is required to define safe design and operating limits for the reactor, and to underpin a suite of safety functions. 2. This capability is an enabler for the development of innovative non-power applications such as a clean source of hydrogen production and nuclear desalination, which could potentially be used to provide portable, clean water to communities.	UC4
PWR_M_20	Best practice guidelines for nuclear engineers on the most appropriate thermal hydraulic modelling	1. This would enable thermal hydraulic modelling tools such as CFD to be applied more widely and with increased confidence, allowing a	UC4, UC5, UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
	approach and the validated solution space for the codes that are available.	wider design space to be explored with the associated benefits of reduced programme cost and duration and increase regulator acceptance.	
PWR_M_21	The development or adaption of uncertainty evaluation procedures for nuclear thermal hydraulics engineers to enable consistent characterisation of the uncertainty in the inputs and outputs of an analysis.	1. Currently, margins are increased to allow for the unknown effects which if reduced could lead to a more optimised design. 2. All thermal hydraulic modelling codes contain empiricism and approximation. This can undermine confidence in the results if the impact of areas of uncertainty and approximation are unknown. A clearer, more transparent evaluation method is needed for advanced modelling methods to be more widely used and accepted by nuclear regulators.	UC4, UC5, UC6
PWR_M_22	Ability to predict the process of clad ballooning, including the transient variation in fuel clad temperature, in the event of severely overheated PWR fuel.	1. In the event of a LBLOCA, the fuel clad temperatures can become hot enough to enable relatively rapid ductile creep. Due to the high internal pressure of the fuel, this creep results in the fuel clad 'ballooning' radially outward and there is concern that this could result in contact between adjacent pins and areas of the clad which are then effectively starved of cooling flow. The ability to predict this with confidence would improve the evidence available for LBLOCA safety cases. 2. Despite numerous experimental investigations into clad ballooning, there has been little work in developing a state-of-the-art modelling methodology in recent years. This work would represent an opportunity for international collaboration with the owners of the experimental programmes.	UC5, UC6
Reactor Technology: Advanced Boiling Water Reactor			
BWR_M_01	Improvements to state-of-the-art dryout models.	1. Dryout of the liquid film in a BWR core represents a major fault in the fuel channel and can lead to significant safety and performance	UC4, UC5

UR No	Statement of Need	Justification/Benefit	Linked Use Case
		issues. Greater confidence in model predictions is required in order to reduce safety margins and improve the economy of the design.	
BWR_M_02	The development of microscopic boiling and film development models with CFD.	1. The improvement of the mechanistic modelling of two-phase phenomena is an important step in being able to predict DNB, CHF and dryout without the need for empirical correlations. This in turn will lead to a reduction in future development costs and a reduction in safety margins.	UC6
BWR_M_03	Modelling to predict critical power in BWR fuel bundles.	1. The safety margins associated with fault scenarios that achieve the critical power ratio for BWRs will be significantly greater than one. By continuing to improve the modelling capability in this area, confidence in the design and predictions will enable the safety margin to be reduced and the economy of the design to be improved.	UC4, UC5
BWR_M_04	The development of a detailed ABWR core simulator, coupling the modelling of neutronics and thermal hydraulics.	1. In reality, the thermal hydraulics and neutronics of a reactor are coupled and analysing them separately introduces inaccuracies. These inaccuracies are taken into account in the margins specified to assure nuclear safety. If these inaccuracies were reduced the safety margins could also be reduced, improving reactor economic performance. 2. This capability would also improve nuclear fuel designs.	UC4, UC5
BWR_M_05	The improvement of transient and accident scenario analysis codes for ABWR fuel bundles.	1. Such improvements are required in order to satisfy the UK Regulator's generic design assessment.	UC4
BWR_M_06	The development and verification of a critical power evaluation method for new, innovative RBWR fuel bundles. The axial power distribution may be uniform or non-uniform.	1. The RBWR uses a hexagonal lattice fuel bundle design with narrow gaps between fuel rods. This new geometry means that current critical power evaluation methods are inadequate. By developing such methods, the design performance can be evaluated and improved as required.	UC4, UC5

UR No	Statement of Need	Justification/Benefit	Linked Use Case
BWR_M_07	The development of CHF and DNB correlations for input into BWR systems codes to characterise the performance of new/novel fuel assemblies.	1. DNB and CHF are important parameters affecting reactor design and operation. For any new fuel type, these correlations need to be developed to enable reactor and fuel performance to be predicted.	UC4
BWR_M_08	The development of a detailed core flow-rate prediction method, including void fraction distribution in the chimney cell and coupling of two-phase flow with core power, for the Double MS (Modular Simplified & Medium Small) Reactor) Reactor (SMR).	1. The chimney cell is a key design feature of the Double MS Reactor as it enhances the natural circulation within the core. Such predictions and correlations are necessary in order to provide a level of clarity in the predictions that will help develop the understanding of and confidence in the performance characteristics of the reactor. This could enable design improvements or reductions in safety margins.	UC6
BWR_M_09	Improved confidence in the ability to predict the non-linear dynamic behaviour of an ABWR including the coupling between variation in void and reactivity which can lead to oscillatory conditions.	1. An improved understanding of the level of uncertainty in existing modelling methods would have key economic and nuclear safety benefits. 2. Ultimately an improved predictive capability could be developed which would enable reduced uncertainty and improved plant performance.	UC4
Reactor Technology: Advanced Gas-cooled Reactor			
AGR_M_01	The ability to model gas flow in highly complex geometries such as boilers. In particular prediction of heat transfer is important.	1. The ability to accurately model gas flow in ageing boilers is required to predict the temperature of boiler components, reduce operating margins, increase reactor power operation and underpin the safety case. 2. Ageing boilers are one of the main issues associated with AGR lifetime extension. Accurate models and predictions of the behaviour of ageing boilers are therefore necessary to justify life extension across the AGR fleet. 3. All reactor designs that incorporate a secondary circuit require heat exchanges/boilers. The nature of these always results in complicated 3D flow (to enhance heat transfer). The ability to model these complex	UC4, UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
		flows and resulting heat transfer with confidence would improve initial design optimisation and through life performance.	
AGR_M_02	Improvement in the ability of 3D modelling codes to accurately model fluid flow through porous media.	1. Porous material modelling enables CFD models to be built without having to explicitly represent the detailed geometry associated with various plant components, such as boiler tube banks, banks of standpipes and insulation. However, the accuracy of predictions obtained using these models is often lower than required. Increased confidence in these models would enable their benefits to be fully realised.	UC6
AGR_M_03	An increase in the speed of analysis solutions without compromising the accuracy of the result (by means of solver/ modelling software improvement).	1. The run time of solutions can be significant, particularly for complex models. Speeding up the run time will reduce the computational power required to produce results, and will reduce the overall timescales for validation and verification.	UC6
AGR_M_04	Improvements in the accuracy of eddy viscosity turbulence schemes to a wider range of flow types.	1. Eddy viscosity turbulence models are widely used throughout industrial CFD simulations and they are less computationally expensive than other methods. However, they contain a lot of approximations. Improvements would result in more reliable models that can be applied to a wider range of scenarios.	UC6
AGR_M_05	The development of agreed standards and best practise for applying Large Eddy Simulation (LES) and Detached Eddy Simulation (DES) CFD simulations that resolve turbulence.	1. The application of LES and DES is more specialised than conventional industrial CFD, and so more limited resources are available that can produce useable results. For LES/DES to become widely adopted, agreed standards and best practice for using the method must be developed. This will enable training on correct use of LES/DES to be put in place. 2. While LES/DES is much more costly than RANS CFD modelling, there are some situations where good results can only be obtained by LES/DES.	UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
AGR_M_06	Improved accuracy and reliability of wall modelling approximations to use both in association with eddy viscosity turbulence models and large eddy simulation or detached eddy simulation.	1. The behaviour of the near wall layer presents a number of challenges to numerical methods and modelling. Improvements in the accuracy of wall modelling would result in more reliable models that can be applied to a wider range of scenarios.	UC6
AGR_M_07	Access to fast, high performance computing facilities (HPC) for use of detailed modelling methods, or performing a large number of simulations for Monte Carlo/statistical analyses of simpler models to quantify uncertainties.	1. Advanced thermal hydraulic modelling methods often require significant hardware to reduce run times to something that is useful to industry.	UC4, UC5, UC6
AGR_M_08	Improvement of interfacing between system codes and detailed component modelling, potentially tools which couple the approaches.	1. Currently most system codes are based on a 1D or 2D methodology with pessimistic approximations of more complex flow phenomena. This restricts the usefulness for 'off-design' calculations and can result in operational restrictions (due to high levels of pessimism). The ability to explicitly model important areas of detail in conjunction with a system code's 'whole circuit' approach would give increased accuracy in the results, decreasing margins.	UC4, UC5, UC6
AGR_M_09	Development of sub-channel models that can predict thermal hydraulic conditions with local flow circulation.	1. If, for example, the flow path of an AGR fuel stringer is blocked, the natural circulation within the blocked flow stringer cannot be predicted. 2. This is significant for safety case justifications under some fault conditions within the fuel route. Normally very pessimistic methods are used for this situation but this leads to restrictions for normal operation. Improved prediction capabilities able to take credit for recirculation would allow conservatism to be reduced.	UC5
Reactor Technology: Liquid Metal Fast Reactor			
LMFR_M_01	The development of parameterised heat transfer correlations from high fidelity modelling, covering	1. Correlations and high fidelity results enable numerical model development and validation, which can be used to assist reactor	UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
	cases of natural, forced and mixed convection within prototypical liquid metal reactor components.	designers and support the acceptance of modelling outputs in safety case evidence.	
LMFR_M_02	The development of more accurate methods to model coupled multi-species and multi-phase coolant mixtures in a liquid metal reactor pool.	1. These methods enable confident prediction of phenomena such as freezing/solidification. 2. Under severe accident conditions, fissile material relocation can occur by molten fuel transport and/or dispersion in the coolant. These analyses are required for improved reactor safety.	UC6
LMFR_M_03	The ability to embed higher fidelity, more resolved scale models (e.g. CFD) within higher level system codes to reduce the reliance on parameterised or empirical correlations, which have limited applicability or limited validation envelopes.	1. The complex and interrelated components and phenomena that occur within reactors cannot be studied satisfactory by separate effect simulations. Combining them allows relevant transients and accidents to be evaluated. 2. The use of system codes, with adjustment based experimental results, is the traditional way to design a reactor: available codes (systems codes and CFD codes), modified for use with liquid metals (for example RELAP5 and CATHARE), and models adjusted based on the results of the tests on dedicated test facilities can be used for representation of the dynamics of the reactor. 3. Significant increases in computation capabilities since the last substantial generation of design presents new opportunities for the development of multi-physics codes, therefore it is possible that there can be a transition in design practice. 4. Classical design methods still have a role, but there is an incentive to pursue a new design approach which has the potential to be more accurate and more efficient. 5. For some cases higher fidelity simulations are becoming essential, for example in the case of passive systems.	UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
LMFR_M_04	The ability to use simulations to predict the pressure drop, flow mixing and temperature of fuel assemblies and sub-assemblies with sufficiently accurate and validated methods. This includes fuel internal, clad and support temperatures.	1. The performance of fuel and fuel components in terms of pressure drop, pin clad, support structure, wrapper temperatures, deformation due to thermal expansion (intrinsic and engineered expansion for reactivity feedback control for example) is necessary for including in system level models, but can be included in a parameterised form, rather than as detailed models. These analyses are required for improved reactor design and safety justifications. 2. Features that could be studied using this capability include steady-state and transient responses, natural convection, deformation, transverse flow and mixing.	UC6
LMFR_M_05	The improvement of the modelling of two-phase flow in liquid metals, including modelling of bubble transport and the onset of boiling.	1. Although two-phase flow is not relevant for ideal, normal operating regimes. The modelling of two-phase flow is relevant to some fault conditions including both the entrainment of gas bubbles and the onset of boiling. Improvements in modelling methods to predict two-phase flow therefore has the potential to reduce safety margins and improve designs.	UC5, UC6
LMFR_M_06	The demonstration of the use of highly resolved, massively parallel high fidelity CFD simulations (e.g. LES, DNS) as a substitute for or supplement to experimental data, and the production of best-practice and referenceable results.	1. Experiments are not able to make measurements that are highly resolved in time and space at arbitrary locations, and of all quantities. Massively parallel high fidelity CFD (up to approximately 1 billion cells, and 100,000 processors) is able to do this to provide validation or correlations to use in lower fidelity models.	UC6
LMFR_M_07	The ability to model the interaction between the simultaneous flow and thermal effects that occur within pool type reactor upper plenums with sufficiently accurate and validated methods. The thermal hydraulic phenomena of relevance include jet impingement, flow mixing/thermal striping, free	1. Within the upper plenum of pool type LMFRs, the interaction of thermal hydraulic phenomena are known to lead to thermal fatigue. This is driven by rapidly varying high temperatures, adjacent to structural components and by flow induced vibration. The large range of time and length scales involved, the sensitivity of the phenomena to small inaccuracies and their intrinsic coupling make this a challenging modelling prediction.	UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
	surface instability, and static and transient stratification.	2. In the core outlet region, evaluating the distribution of core outlet temperatures allows the demonstration that they are representative of the channel, and helps to validate and interpret instrumentation readings. These analyses are required for improved reactor design and safety justifications.	
LMFR_M_08	The ability to model the flow and temperature distribution within pool type reactor lower plenums with sufficiently accurate and validated methods.	1. The requirement is separate from upper plenums because it does not involve free surface flows, and is strongly dependent on the operation and performance of the primary heat exchangers. 2. The outlet from the heat exchanger can have temperature inhomogeneities, driving thermal fatigue of structural components. 3. Similarly, under transient operation, buoyancy driven flows and unstable thermal stratification can lead to thermal fatigue. 4. Asymmetric operation, where not all heat exchanger circuits are operating can create thermal stresses that need to be predicted. These analyses are required for improved reactor design and safety justifications. 5. Highly resolved CFD simulations of heat exchanger and plenum interactions are needed to determine amplitude and frequency of instabilities. These were not practical in previous generations of reactor analysis, but are now.	UC6
LMFR_M_09	The ability to confidently simulate the behaviour of passive decay heat removal from a pool type reactor, with natural convection driven by the decay heat removal heat exchanger.	1. The interaction of decay heat removal heat exchangers with pool configurations, that include stratification, occurs via mixed convection (natural and forced combined). This is a complex 3D phenomena and cannot be captured in 1D system codes. 2. These analyses are required to improve the design and safety justifications of the heat exchangers and their connection to the ultimate heat sink.	UC4, UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
LMFR_M_10	The ability to predict mixing at junctions in pipe systems where the merging streams can have > 100°C differences in temperature in timeframes suitable for use by industry.	1. Unstable mixing flows can cause rapid temperature fluctuations of adjacent pipe walls, leading to thermal fatigue. Predicting the onset of this behaviour allows for better confidence in the reliability of pipe systems and guidance in their design and operation. 2. High resolution CFD can predict this process in detail - previous generations of design did not have access to the computing power needed to use detailed simulations. However, these simulations are currently very specialist and time consuming and would therefore not be appropriate to a fast moving industry design project.	UC6
LMFR_M_11	Improvement in the ability to accurately include thermal stratification in the prediction of flows and heat transfer in pipes and bends.	1. Low flow rates can lead to stratification in horizontal pipes or U-shaped sections (for example) and differential thermal expansion of the pipe top and bottom, with the associated stress. 2. Stratification can affect the initiation of natural convection flows for passive decay heat removal. Predicting these effects allows for better confidence in the reliability of passive heat removal systems and guidance in their design and operation.	UC4, UC6
LMFR_M_12	The ability to model the details of flow and heat transfer in heat exchangers with sufficiently accurate and validated methods.	1. Heat exchangers often have highly complex geometry and flow. Primary and secondary heat exchangers need to be optimised, particularly for the uniformity of flow and heat transfer. The ability to predict flow and heat transfer accurately under more conditions would lead to improved performance and reliability of the heat transfer systems within a reactor. 2. The effect of buoyancy driven flows is significant for safety critical passive heat removal and natural circulation operation, and predicting how this affects the distribution of flow and heat transfer is important. 3. Flow induced vibration can occur during high flow operation.	UC4, UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
		4. Freezing of liquid metal coolants is possible during low flow conditions.	
LMFR_M_13	The ability to accurately simulate turbulent flows using CFD in liquid metals with large temperature differences. Although considerable progress has been made in recent years this challenge is still not completely resolved.	1. The prediction of all flows relevant to reactor systems must account for the presence of turbulence on heat transfer and flow. For lower Peclet number flows, the Reynolds analogy used in typical industrial (RANS) CFD models is not applicable for highly conductive (low Prandtl number) fluids, such as liquid metals. Improvements or alternative approaches are needed. 2. The interaction between buoyancy forces and the evolution of turbulence is not well characterised for low Prandtl number fluids. 3. Modern computing facilities can overcome these restrictions either directly simulating the missing physics in the system under consideration via higher resolution methods (e.g. LES) or by numerical research into improved models (e.g. LES or DNS).	UC6
LMFR_M_14	The ability to model liquid metal pumps within the reactor core with sufficiently accurate and validated methods.	1. Liquid metal pumps are integrated within pool type reactors and can be closely coupled to the heat exchangers, making the details of their performance tightly coupled to the core and heat exchanger. 2. The dynamics of the pump under transients, such as power failure are important, where their inertia is used to maintain cooling flow for a short time during the transition to natural circulation.	UC4, UC5
LMFR_M_15	The ability to model the temperature of transported fuel modules within fuel handling systems and operations with sufficiently accurate and validated methods.	1. Demonstrating safe fuel handling will be a key requirement for any reactor. Similarly, minimising the time between stopping a reactor and being able to handle fuel is an important operational and economic consideration. Fuel handling systems often handle fuel elements in a gas environment, and need to be tolerant of faults in the transport and cooling equipment. Therefore, accurate simulations of fuel cooled by natural convection and by radiation are important.	UC5, UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
LMFR_M_16	The ability to model the free surface of the reactor pool and the gas void above it. This should include simulation of instability, gas space flow patterns and radiation heat transfer through the gas.	1. The space above the free surface, which is filled with argon gas, is reduced in the new designs of SFRs to reduce the size of the core. This increases the velocity of the free surface, generating stronger vortices and potentially increasing gas entrainment into the sodium. The temperature of the free surface is significantly higher than that of the roof and hence radiation heat transfer is expected to be strong.	UC4, UC6
LMFR_M_17	The ability to incorporate models of supercritical CO_2 power cycles and turbomachinery into the analysis of high temperature reactors.	1. Supercritical CO_2 turbomachinery can offer improved thermal efficiency compared to steam power cycles and result in reduced plant size, giving lower capital cost. 2. They are particularly advantageous for LMFRs, especially molten sodium, because they remove the danger of an explosive reaction with water.	UC4
Reactor Technology: Molten Salt Reactor			
MSR_M_01	The ability to model coupled neutronics, chemical species and thermohydraulics in a single code.	1. The understanding of thermophysical variation as a function of irradiation and species evolution, is necessary to fully predict the primary circuit behaviour of an MSR. Good theoretical agreement is required to demonstrate to a regulator that all key safety functions (control of reactivity, control of heat removal, control of containment) can be achieved throughout all operating states of a reactor. 2. MSR designs, where the fuel is dissolved in the coolant, present additional challenges with regard to movement of fissile material through the system. For example, the ability to predict the movement of delayed neutron precursors is important to reactor control and safety.	UC4, UC5, UC6
MSR_M_02	An improved capability to model thermo-mechanical coupling such as fuel-cladding interactions under	1. This is required to demonstrate the structural integrity of fuel and clad in designs where this is relevant. The ability to predict mechanical coupling effects dynamically during processes such as the	UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
	transient conditions and swelling for solid fuel designs.	transition of fuel assemblies from one in-power assembly position to the next would enable demonstration of this integrity as reactor designs move away from campaign style refuelling.	
MSR_M_03	The ability to accurately predict the flow of molten salts in complex geometries. This would include development/verification of eddy diffusion kinetics and best fit turbulence models for halide salt flows in coolant channels.	1. This is required to accurately predict the flow and heat transfer within the reactor for the refinement/ optimisation of the design and to demonstrate key safety functions.	UC5, UC6
MSR_M_04	Determination of mass transport scaling laws for halide salt flows.	1. This could be used to enable the use of smaller scale, less expensive tests to support nuclear safety and design.	UC2
MSR_M_05	The development of heat transfer correlations for the modelling of molten salt mixtures.	1. The heat transfer behaviour of molten salts is not as comprehensively studied as conventional fluid and molten metals, and the dataset is therefore not as rich. New correlations will be required to accurately predict heat transfer behaviour, develop sub-channel and system level models to underpin safety and design assessments.	UC4, UC5
MSR_M_06	An improved understanding of and the ability to accurately predict the flow of heat generating fluids.	1. Historically, very little research has been done on the flow of heat generating fluids, and there is a need to better understand the combined effects of buoyancy force and internal heat generation on the flow of fluids. Without this information, there may be phenomena that are not anticipated or accurately predictable. This understanding is important to the development of new designs and to underpin their safety.	UC6
MSR_M_07	The ability to predict natural convection behaviour of molten salts.	1. Historically, very little research has been done on the natural convection behaviour of molten salts. Without this information, there may be phenomena that are not anticipated or accurately predictable. Even in a pumped system, prediction of natural convection is required	UC4, UC5, UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
		to support nuclear safety assessments, especially under fault conditions.	
Reactor Technology: Supercritical Water Reactor			
SCWR_M_01	The development of Direct Numerical Simulation (DNS) as a useable prediction method for turbulence at higher Reynolds numbers. This is currently limited by computing hardware.	1. High uncertainties remain in analysing supercritical heat transfer with industrially available, general purpose CFD tools, particularly those using RANS turbulence models. Improving existing turbulence models is an important step to enhance the confidence of the developed concept. DNS is one potential method of increasing the understanding of turbulence characteristics.	UC6
SCWR_M_02	The further development of conventional turbulence modelling appropriate to a SCWR fuel bundle.	1. DNS gives one route to better understanding turbulent flow structures. However, it is not likely to be practical to perform many analyses in this way. More conventional turbulence models are therefore needed to realise the benefits of detailed modelling in reactor design and safety assessment.	UC6
SCWR_M_03	The ability to reliably predict the heat transfer coefficient of supercritical water under relevant conditions, taking into account the large variation in material properties.	1. Due to the large variation in the thermophysical properties of supercritical water, it is challenging to predict the heat transfer coefficient by conventional methods. The development of new methods (both empirical and numerical methods), which take into account variable fluid properties as functions of pressure and temperature, is needed to develop reliable prediction methods to support new designs and underpin safety.	UC4, UC5, UC6
SCWR_M_04	The ability to predict the flow instability and large flow structures in the fuel bundle and the mixing factors between sub-channels	1. The pitch-to-diameter ratios in SCWRs are normally small, resulting in a situation where the flow cross-section area varies significantly (large gaps sit alongside small ones) leading to flow instability. Unsteady flow structures may influence heat transfer or cause pin vibrations. The inter-sub-channel turbulent mixing factors are influenced by these flow structures. The prediction of these is	UC4, UC5, UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
		therefore important to reactor design, performance and safety assurance.	
SCWR_M_05	The development of methods to predict heat transfer deterioration in supercritical water channels.	1. Heat transfer deterioration is a phenomenon observed in some fluids, including supercritical water, where the temperature difference between the bulk fluid and a solid surface increases rapidly over a particular range of heat flux. This has been found to be related to the strong variation in material properties and significant buoyancy induced flows. The prediction of heat transfer deterioration is extremely challenging, but necessary to determine the performance of a SCWR under all relevant conditions. 2. The ability to predict conditions where heat transfer deterioration will occur with confidence will contribute to reactor design and the assurance of safety.	UC5, UC6
SCWR_M_06	Ability to understand and better predict similarities/scaling between different fluids at supercritical pressure.	1. The critical pressure and temperature of water are very high and hence experiments under these conditions are very challenging. Much research has therefore been carried out using surrogate fluids such as CO_2 whose critical pressure/temperature values are much lower and hence easier to work with. However, the property variations are different for different fluids, and the correlations/data produced using one fluid is not necessarily directly applicable to other fluids. Understanding the similarities of the behaviours of different fluids and deriving scaling laws between them is hence important to make use of the large body of data/correlations from various non-water fluids. 2. Improved scaling laws would enable existing validation data to be used for the development of modelling tools for SCWRs.	UC5, UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
Reactor Technology: High Temperature Gas-cooled Reactor			
HTGR_M_01	A programme of simulations of the phenomena that affect the mixing and distribution of air in an air-ingress accident to identify the major contributory effects. Key thermal hydraulics phenomena of interest are: turbulent mixing, buoyancy driven flow and stratification.	1. A critical event in the safety analysis of VHTRs and GFRs is the air-ingress scenario, where there is a primary circuit break, loss of coolant and the high temperature components are exposed to air, leading to structural damage. The mixing of air with coolant is influenced by the complex flow pattern in the coolant ducts and the vessel during air-ingress. This is governed partially by mixing by flow and by diffusion, interacts with stratification patterns and varies with time and the location of the break in the reactor. Buoyancy dominated flows such as gravity currents are relevant to large breaks, where flow may occur simultaneously in and out of a large aperture, causing air to replace coolant. 2. This simulation work will improve understanding of and provide evidence to support nuclear safety relating to an air-ingress fault.	UC4, UC5, UC6
HTGR_M_02	The ability to model the loads on components exposed to high velocity gas flows under accident conditions.	1. During LOCA conditions, where there is a rapid depressurisation of the reactor, the coolant flow velocities will be very high, potentially damaging components with high aerothermal loads. The ability to predict these loads would provide a key input to the design of these components.	UC4, UC5, UC6
HTGR_M_03	Improved heat transfer and friction factor correlations covering the mixed convection, transition flow regime. In particular, the prediction of peak fuel clad temperature occurring during a loss of flow accident requires accuracy in the transition, mixed convection regime.	1. The ability to understand and predict heat transfer across all relevant flow regimes is required for accurate prediction of thermal hydraulic behaviour and phenomena. This will form the basis for safe operating and design limits for the reactor and its components, and will inform the development of the most efficient designs. In particular, the flows encountered in natural circulation operation are of high significance and are subject to large uncertainties.	UC4, UC5, UC6

UR No	Statement of Need	Justification/Benefit	Linked Use Case
		2. Improvements in correlations will improve the accuracy of thermal hydraulic models.	
HTGR_M_04	Improvements to the computational speed of high fidelity CFD and multi-physics simulations.	1. High fidelity simulations are still slow for routine design level analysis of power reactors. Because of this, simulations often simplify aspects of the physics to gain speed. Methods for reducing computer runtime by optimising algorithms and implementation would allow more detailed analysis and more simulations to be performed. 2. High fidelity CFD offers benefits in terms of accuracy and enhanced understanding. Removing barriers to its use will improve both the economics and safety of reactor design.	UC6
HTGR_M_05	The ability to accurately model graphite or silicon carbide dust motion, deposition and suspension in reactor and heat exchanger geometries.	1. Fine dust particles that are released from the moderator or fuel clad are picked up by the primary coolant flow and deposit on the tube and equipment surfaces. Because the particles are radioactive, the graphite dust complicates maintenance and repair work. In addition, the ejection of radioactive dust from broken pipes is a threat to the environment in potential depressurisation accidents. 2. Improvements in dust transport modelling will result in reduced cost of maintenance and improvements to nuclear safety.	UC4, UC6
HTGR_M_06	The ability to accurately estimate how much tritium is generated in the system and transferred to the environment.	1. In VHTRs, tritium, posing a nuclear safety risk, can permeate through the heat transfer surfaces from the primary coolant. In non-power applications, this may radioactively contaminate the coupled process (hydrogen or steam production). 2. This is a thermal hydraulics issue because the rate that tritium is transported from the core to other areas by the coolant, and its permeation into and through solids is expected to be predicted by thermal hydraulics tools.	UC4

UR No	Statement of Need	Justification/Benefit	Linked Use Case
HTGR_M_07	Improvements to the rigour of methods to predict the balance between oxidation power released and convective cooling in graphite channels.	1. When high temperature graphite (as a moderator in a VHTR) is exposed to air flow, the air can cool the graphite by convection, but any oxygen present oxidises exothermally with the graphite, generating heat. The conditions where the cooling effect is greater than the heating needs to be predicted. Some experimental data and calculation codes (system and CFD) are available, and theoretical knowledge exists on air-ingress in specific configurations, but only simplified tools are currently able to couple the thermal hydraulics and chemistry. 2. Prediction of this key phenomena relevant to reactor air-ingress is important for nuclear safety.	UC5, UC6
HTGR_M_08	The ability to model and predict the coupled effects of thermal hydraulics and chemistry, specifically the reactions of air, nitrogen and steam with the reactor core and fuel assemblies at high temperatures.	1. During a fault scenario, air, nitrogen or steam could enter the reactor. Therefore, the chemical reactions with these gases and the reactor core at high temperature must be understood in order to assess the risk of core degradation. Oxidation, nitriding and material interactions coupled with the core flow pattern govern the possible core degradation in the early phase of a severe accident.	UC4, UC5, UC6
HTGR_M_09	The development of a whole-core transient system code representing solid heat conduction, graphite thermo-mechanics, gas dynamics, including sufficient representation of 3D effects, and other coupled physics, such as neutronics, with sufficient local resolution to investigate and understand the flows in the core.	1. A whole core model is needed as conduction heat transfer to the outer surface of the core is a significant contributor to cooling under loss of flow accidents. Existing computational thermal hydraulic codes often solve only a piece of the core. 2. Existing system codes often assume a prescribed portion of the flow bypasses the fuel channels and passes through reflector structures, rather than predicting this explicitly. 3. Three-dimensional effects are not easily captured with system codes, and CFD models are too complex for routine transient analyses.	UC4, UC5

UR No	Statement of Need	Justification/Benefit	Linked Use Case
		4. The distribution of flowrate in each fuel channel needs to be predicted to allow the flow distribution across the reactor to be adjusted to achieve a uniform outlet temperature (given the radial variation in core power). 5. Multi-physics capabilities allow for the assessment of faults such as steam ingress, which leads to a change in moderation and reactivity. 6. A whole core system model would be an important tool in analysing faults and supporting nuclear safety as a key enabler for this technology.	
HTGR_M_10	The ability to couple non-power generation dynamics (for example, a hydrogen plant) with reactor dynamics in a system code.	1. The interaction of the dynamics of rotating plant is a consideration in the design and operation of safety systems for power producing reactors. For non-power producing applications, such as those envisioned for high temperature reactors, the dynamics of the attached plant (variations in load, response to emergency shutdown) must be coupled to the reactor model.	UC4
Cross-Cutting			
CC_M_01	The ability for software relevant to the UK thermal hydraulics modelling capability to ‘plug into’ the Virtual Engineering Environment.	1. A parallel BEIS funding work stream is currently developing a virtual engineering environment. It would be of benefit to ensure that any new tools are developed to interface with this environment.	UC4, UC5, UC6

6 Discussion of Requirements

The requirements listed in Section 5 represent a picture of the current challenges in the nuclear thermal hydraulic research landscape, across a range of reactor technologies. By inspection, it is easy to see how complicated the full set is and the interlinking between different technologies, thermal hydraulic phenomena and use cases. Whilst full review of the requirements is required to understand their technical detail, this section provides scope for some qualitative review of the themes and principles allowing unifying features to be picked out.

The section also provides further discussion of the requirements in the context of the use cases highlighting what is needed, beyond simple scientific research, for qualification of a specific design.

6.1 Technology Differences

In many cases, the requirements raised are specific to the reactor technology under consideration, and in some cases to the specific reactor design. Given how the information was captured, this is neither surprising nor problematic; technology developers tend to be focused on their own issues and not on the areas of commonality across reactor types.

A reactor type's Technology Readiness Level (TRL) fundamentally changes the depth and maturity that is observed in the requirements. For example, there are many operational PWRs worldwide and many organisations with millions of hours of design and operational experience. Therefore, the PWR requirements that have been gathered are focused towards specific improvements or detailed investigations of areas which offer a specific economic benefit. In contrast, most Generation IV reactor technologies are based on only a small number of previous test reactors, many of which are no longer operating. As a result, it is extremely unlikely that the designers of these systems have identified all of the challenges and promising areas for investigation and the requirements that have been expressed are often raised at quite a high level.

The priority of many of the organisations involved in the development of Generation IV reactors is to create a working demonstrator at the minimum possible cost (even if it is quite a long way from an optimised design). The consideration of thermal hydraulic challenges is focused around the need for integral testing of design specific systems to achieve this end.

The differences in primary circuit fluid between different reactor technologies raises a variety of technology specific requirements. For example, in some cases the relevant thermal hydraulic properties of the fluid are well understood, in others this is still an area of need. The complete list of technology specific requirements are too numerous to list here and reference should be made to Section 5 for further details of requirements relating to a specific technology.

6.2 Common Themes

Systematic review of the requirements, in combination with the discussions during the capture process, have enabled the identification of a number of themes common to more than one reactor technology.

6.2.1 The Use of Modelling in Nuclear Thermal Hydraulics

An overriding concern for many stakeholders and across all technologies was related to 'trust' in the results of thermal hydraulic modelling. Modelling and simulation, for the purposes of demonstrating design performance and key nuclear safety parameters, has become an essential part of reactor development and licensing. Physical testing is also an essential part of the evidence needed for any new nuclear installation; but there are significant costs involved in building and operating test rigs across all technologies, this has driven the development of increasingly advanced modelling tools in order to replace this significant development cost sink. However, in common with all calculation methods, thermal hydraulic models contain inherent assumptions and approximations that limit their ability to simulate real plant. Technology developers sometimes meet challenges in their interactions with the regulator, where there is a desire to use modelling predictions to support nuclear safety claims.

Counterintuitively, it is the simpler models (system and sub-channel codes) with more approximations which are most extensively used by industry. The reasons for this are, in part historical and a full exploration is outside the remit of this document[4]. More modern and advanced modelling methods (e.g. CFD) offer significant potential benefits, but their complexity and flexibility make validating their use more challenging. However, the ability to make more extensive use of advanced 3D modelling methods was an almost universal requirement across all industry contributors and the consensus of opinion was that this was of at least as high a priority as the further development of advanced methods.

The need to improve the way that existing modelling techniques are used was raised by representatives of a number of different reactor technologies. For example, the development of best practice guidelines and a consistent understanding of the use of more advanced (i.e. CFD) modelling methods would add value across all technologies, but is especially important for new reactor designers. Similarly a need to actively develop methodology to quantify uncertainties of modelling was identified as having benefit across the whole UK civil nuclear industry.

The coupling of models into a tool chain which makes predictions across a broader range of physical phenomena, was identified as an underexploited approach. Coupling could be explored in a number of different ways: with coupling methods for models of different fidelities and different physics identified as key. The embedding of higher fidelity, more detailed models within the framework of a low fidelity, 'whole system' model offers benefits over current industry best practice. The aim is a "best-of-both" approach: increasing the accuracy of the low fidelity model and improving the speed of model execution by limiting high fidelity approaches to areas where they bring most benefit. The coupling of models predicting different physics phenomena and behaviours (i.e. multi-physics modelling) was also identified as having significant benefit. It was noted by several stakeholders that the coupling of thermal hydraulics models with neutronics and/or chemistry models has been done on numerous occasions in an academic environment, but is not commonly used in industry.

With the advancement in computer technology, advanced CFD techniques, such as Direct Numerical Simulation (DNS) and Large Eddy Simulation (LES), are expected to play an increasing and important role in thermal hydraulics modelling. In the research community, they are already widely used. Within industry however, the large computational costs and timescales often make using these methods prohibitively expensive under most conditions. Despite this, there is a desire to make these methods more accessible and potentially use them to supplement testing (which also has a high cost and long timescales).

It was clear across all industries that experimental data with which to validate all types of modelling are needed. In many cases, models have been developed (or partially developed) and it is the validation data that are lacking. This largely stems from the high costs associated with testing. Supporting the development of a national facility to produce suitable data for model

[4] Section 4 touches on a number of technical reasons for the use of low fidelity models.

validation and development is a key objective of this project. However, the more advanced CFD techniques require more sophisticated testing and faster data acquisition and processing.

6.2.2 Thermal Hydraulic Phenomena

The different primary circuit fluids employed in the various Generation IV reactor technologies result in significant differences in the requirements for thermal hydraulic testing and model improvements suggested by each community. However, despite these differences, areas of common thermal hydraulic phenomena were raised:

- Many developers and researchers are interested in the effects and behaviours associated with single and two-phase natural convection; it is vital to the safety claims made for many of the Gen III+ and Gen IV reactor designs and accurately predicting heat transfer and flow under natural convection conditions is important for underwriting the safety systems that depend on these mechanisms. The adoption of an increasing number of passive cooling features in advanced designs was given as a key justification of the need in many cases.
- The dominant thermal hydraulic concerns for all types of LWR revolve around two-phase flow and boiling. An enormous amount of research work has been expended in this area, but, because it is relatively complex and vital for reactor performance, safety and accident response; there is still the need for further improvement. Beyond LWRs, many reactor technologies use water within a secondary circuit (for the generation of steam). Two-phase flow is also an important consideration in all types of boiler and the performance of heat exchangers and steam generators has been identified as an area of common ground.
- Interestingly, the area of single-phase turbulent mixing was raised by many industry contributors. Although initial impressions may be that this is a well-studied area in terms of both model development and testing, it remains a key area of uncertainty in many modelling tools and is therefore of concern to both reactor designers and regulators. Examples include: heat transfer of all regimes, but particularly, mixed convective heat transfer and natural circulation; the impact of complex geometry such as fuel spacers; and conjugate heat transfer. Increased confidence in modelling results, or at least a well-defined validation envelope was highlighted as the main requirement.
- Reactor component fatigue (caused by flow induced vibration or fluid temperature fluctuation) was raised as a concern across a number of technologies. Although this is not entirely a thermal hydraulics issue, the driver for the fatigue is often an unsteady or inadequately mixed flow. Identifying these flows is key to mitigating the risk of component failure and aspects of thermal hydraulic model improvement or testing that contribute to this are of benefit[5].

[5] Although this is raised as an area of common ground, the thermal hydraulic drivers behind the fatigue can be very technology and/or design dependent. This should be taken into consideration when considering future work.

7 Abbreviations

Acronym	Definition
ABWR	Advanced Boiling Water Reactor
AGR	Advanced Gas-cooled Reactor
ANL	Argonne National Laboratory (USA)
INL	Idaho National Laboratory (USA)
ARIS	Advanced Reactor Information System
BEIS	Department for Business, Energy and Industrial Strategy
BWR	Boiling Water Reactor
CEA-DEN	Commissariat à l'énergie atomique et aux énergies alternatives, la Direction de l'Énergie Nucléaire (France)
CFD	Computational Fluid Dynamics
CHF	Critical Heat Flux
CORDIS	COmmunity Research and Development Information Service (European Commission)
CPD	Continuous Professional Development
CSNI	Committee on the Safety of Nuclear Installations (OECD NEA)
DECC	Department of Energy and Climate Change
DES	Detached Eddy Simulation
DNB	Departure from Nucleate Boiling
DNS	Direct Numerical Simulation
DOE	Department of Energy (USA)
Double MS	Modular Simplified & Medium Small
FIV	Flow Induced Vibration
GenIV	Generation IV
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum
HTGR	High Temperature Gas-cooled Reactor
IAEA	International Atomic Energy Agency
IP	Intellectual Property
ITT	Invitation to Tender
JAEA	Japan Atomic Energy Agency
KAERI	Korea Atomic Energy Research Institute
LBE	Lead-Bismuth Eutectic
LBLOCA	Large Break Loss-Of-Coolant Accident
LES	Large Eddy Simulation
LFR	Lead-cooled Fast Reactor
LFR-AS-200	Lead-cooled Fast Reactor, Amphora-Shaped, 200 MW
LMFR	Liquid Metal-cooled Fast Reactor
LOCA	Loss-Of-Coolant Accident
LOD	Line Of Development
LWR	Light Water Reactor
MSR	Molten Salt Reactor

Acronym	Definition
NAMRC	Nuclear Advanced Manufacturing Research Centre
NEA	Nuclear Energy Agency (OECD)
NIRAB	Nuclear Innovation and Research Advisory Board
NNL	National Nuclear Laboratory (UK)
NNUF	National Nuclear User Facility (UK)
NURETH	International Topical Meeting on Nuclear Reactor Thermal Hydraulics
OECD	Organisation for Economic Co-operation and Development
ONR	Office for Nuclear Regulation (UK)
PSS	Passive Safety System
PWR	Pressurised Water Reactor
R&D	Research and Development
RANS	Reynolds-Averaged Navier-Stokes
RBWR	Resource-renewable Boiling Water Reactor
SCK•CEN	Studiecentrum voor Kernenergie • Centre d'Étude de l'énergie Nucléaire (Belgium)
SCWR	Supercritical Water Reactor
SFR	Sodium-cooled Fast Reactor
SIG	Special Interest Group
SMR	Small Modular Reactor
STEM	Science, Technology, Engineering and Mathematical
STSG	Spiral Tube Steam Generator
TNA	Training Needs Analysis
TRL	Technology Readiness Level
UC	Use Case
UKAEA	United Kingdom Atomic Energy Authority
UR	User Requirements
VHTR	Very High Temperature Reactor
VR	Virtual Reality

8 References

1. ‘UK Nuclear Innovation and Research Programme Recommendations’, March 2016, NIRAB-75-10.

2. ‘Prioritisation of UK Nuclear Innovation and Research Programme Recommendations’, November 2016, NIRAB-124-1.

3. ‘Nuclear Research and Development Programme: Invitation to Tender for R&D on Digital Nuclear Reactor Design’, October 2016, TRN 1210/09/2016(3).

4. ‘NURETH-17 International Topical Meeting on Nuclear Thermal Hydraulics’ September 2017.

5. C. Howlett, G. Macpherson, D. McNaught, ‘Project FORTE - Nuclear Thermal Hydraulics Research and Development: Specification Supporting a UK Nuclear Thermal Hydraulic Modelling Capability’, Frazer-Nash Consultancy, FNC 53798/47483R, Issue 1, 2019.

6. J. McIntosh, A. Strange, ‘Project FORTE - Nuclear Thermal Hydraulics Research and Development: National Nuclear Thermal Hydraulic Facility Specification’, Frazer-Nash Consultancy, FNC 53798/47366R, Issue 1, 2019.

7. ‘Technology Roadmap Update for Generation IV Nuclear Energy Systems - GenIV International Forum’, January 2014.

8. J. McIntosh, ‘Project FORTE - Nuclear Thermal Hydraulics Research and Development: Potential cross-cutting requirements’, Frazer-Nash Consultancy, FNC 53798/47834R, Issue 1, 2019.

ANNEX A - REQUIREMENTS QUESTIONNAIRES

This Annex contains two examples of the questionnaires completed by contributors to make a contribution to the User Requirements process.

A1 Academic Expert Questionnaire

Project Forte

Nuclear Energy Thermal Hydraulic Requirements Questionnaire

1. INTRODUCTION

1.1 THE UK VISION

The worldwide nuclear industry is expected to grow significantly in the coming decades through the development, build and operation of new large nuclear installations and small modular reactors. The UK Government recognises the importance of nuclear energy in providing a reliable and affordable source of low carbon electricity. Both national ambition and international growth therefore bring the potential for the UK to make a more significant contribution to new reactor design.

The UK Government's 2013 Nuclear Industrial Strategy described these significant growth ambitions. It proposed a series of shared commitments between Government and industry to grow the UK's industrial capability and enable a future where nuclear energy is a key pillar of the UK's 2050 energy mix. One of the features of the strategy was establishing the Nuclear Innovation and Research Advisory Board (NIRAB) to co-ordinate and advise on innovation in the nuclear energy sector.

NIRAB developed a series of research recommendations[1], grouped into five key programmes. One of these programmes, Digital Reactor Design, includes two recommendations that relate to nuclear thermal hydraulics:

- The development of a major new UK Nuclear Thermal Hydraulic Test Facility;
- The development of new Nuclear Thermal Hydraulic Modelling Techniques and Tools.

1.2 PROJECT BACKGROUND

To address these recommendations, the Government's Department for Business, Energy and Industrial Strategy (BEIS) has tasked Frazer-Nash Consultancy and our partner organisations (the University of Manchester, the University of Sheffield, STFC Daresbury, EDF Energy and Westinghouse Electric Company) with delivering the first phase of a programme of research and development (R&D). The scope of work includes the following two work packages:

- Developing a Specification for an Innovative UK Thermal Hydraulic Modelling Capability

  The Modelling Capability Specification will define the thermal hydraulic knowledge, understanding and modelling tools that are required within the UK to support reactor design and operation and to underpin safety cases. The output will assess the modelling challenges associated with new and existing technologies, use these to set the overall goals of UK model development and recommend strategies for achieving those targets.

  It is envisaged that the outcomes of this task will shape the direction of UK Government funded nuclear thermal hydraulic modelling research for the foreseeable future.

- Developing a Specification for a UK National Nuclear Thermal Hydraulic Test Facility

  The Test Facility Specification will define the system requirements of a specific test rig, or rigs (referred to herein as 'the test facility'). The requirements for the test facility will be sufficiently mature to form part of a tender for the design and build of the test facility.

[1] NIRAB-75-10, March 2016

> The test facility will provide a much needed resource to support the UK nuclear industry. However, it is also intended to meet global needs for state-of-the-art test facilities, thereby making a positive contribution to the worldwide nuclear industry.

BEIS anticipate the test facility will be completed in 2022.

The current 'Phase 1' modelling specification and parallel research activities will be completed in 2019, along with a proposed plan for a more extensive 'Phase2', to perform further UK Government funded research and development.

2. YOUR CONTRIBUTION

It is important to the UK Government that any R&D that they fund is clearly linked back to benefit to the nuclear energy industry. We are currently undertaking a programme of engagement with UK and international nuclear stakeholders to identify the requirements that will drive the specifications for the modelling capability and the test facility.

We have contacted you as a subject-matter-expert in nuclear thermal hydraulics and we would value your thoughts and opinions on the future of nuclear thermal hydraulic R&D. The current focus of the work is capturing the needs of industry, prior to identifying ways forward. We are aware that, through your research and contacts, you have insight into both the current status of thermal hydraulic modelling and testing, and where developments would be of specific benefit to commercial power generation.

To include your insight and to help us to understand your research activities, we would be grateful if you could complete the questionnaire in Section 3. We are eager for a wide response to this questionnaire and encourage you to answer as much as possible. However, please disregard any questions that you are unsure how to answer. We would also appreciate any supplementary information or evidence that you would like to provide to support your answers.

If you are interested in contributing, please provide your questionnaire responses by email to Jordan McIntosh (j.mcintosh@fnc.co.uk).

2.1 TIMESCALES

We are seeking a response by 30th September 2017. We acknowledge this is a tight timescale and, if you are unable to meet the initial deadline, we would still encourage you to submit your response as soon as possible.

2.2 PUBLIC DOMAIN INFORMATION

The final output of this study is likely to be a public domain document. It would therefore be beneficial to all parties if you could exclude any sensitive information and only include responses that you are happy to have in the public domain.

3. QUESTIONNAIRE

3.1 YOUR RESEARCH INTERESTS

Question 1: What technical areas do you and your group/collaborators work in that are relevant to nuclear thermal hydraulics?
Click here to enter text.

Question 2: Which other national and international research programmes, institutions or companies do you collaborate with in your research?
Click here to enter text.

Question 3: Which conferences do you attend and which journals do you publish in, or read, that you think are most relevant to nuclear thermal hydraulics?
Click here to enter text.

Question 4: Do you conduct or know of research in other fields that you think could be applied to nuclear thermal hydraulics and is currently not being fully exploited?
Click here to enter text.

3.2 RESEARCH AND DEVELOPMENT PRIORITIES

Question 5: Looking ahead between 5 and 10 years, could you define any specific nuclear thermal hydraulic challenges that would be addressed by improved thermal hydraulic modelling capabilities? *Please feel free to recommend any review articles or seminal papers that contain further information to expand on your ideas.*
Click here to enter text.

Question 6:
Looking ahead between 5 and 10 years, could you define any specific nuclear thermal hydraulic challenges that could be addressed by a future UK test facility? *Please feel free to recommend any review articles or seminal papers that contain further information to expand on your ideas.*
Click here to enter text.

Do you have a clear idea of what is needed to address these challenges? If so, we would like to hear your more detailed requirements and suggestions.

For example:

- Which thermal hydraulic phenomena need to be better understood or modelled?
- What conditions do you need a test facility to replicate and what parameters would you wish to measure?
- What is needed to increase regulator acceptance of thermal hydraulic modelling methods?

Please be as specific as possible and include how you hope these more detailed requirements/ suggestions will address the higher level challenges listed in your responses to Questions 5 and 6.

Question 7:
Please describe any more detailed requirements or suggestions.
Click here to enter text.

3.3 YOUR WIDER VIEW OF THE FIELD

Question 8:
Are there any particular international technical programmes that you think the UK Government should join, or increase their involvement in, and why?
Click here to enter text.

We kindly ask that you respond by email to Jordan McIntosh (j.mcintosh@fnc.co.uk).

Thank you for taking the time to complete this questionnaire!

A2 Industry Questionnaire

Project Forte

Nuclear Energy Thermal Hydraulic Requirements Questionnaire

1. INTRODUCTION

1.1 THE UK VISION

The worldwide nuclear industry is expected to grow significantly in the coming decades through the development, build and operation of new large nuclear installations and small modular reactors. The UK Government recognises the importance of nuclear energy in providing a reliable and affordable source of low carbon electricity. Both national ambition and international growth therefore bring the potential for the UK to make a more significant contribution to new reactor design.

The UK Government's 2013 Nuclear Industrial Strategy described these significant growth ambitions. It proposed a series of shared commitments between Government and industry to grow the UK's industrial capability and enable a future where nuclear energy is a key pillar of UK's 2050 energy mix. One of the features of the strategy was establishing the Nuclear Innovation and Research Advisory Board (NIRAB) to co-ordinate and advise on innovation in the nuclear energy sector.

NIRAB developed a series of research recommendations[1], grouped into five key programmes. One of these programmes, Digital Reactor Design, includes two recommendations that relate to nuclear thermal hydraulics:

- The development of a major new UK Nuclear Thermal Hydraulic Test Facility;
- The development of new Nuclear Thermal Hydraulic Modelling techniques and tools.

1.2 PROJECT BACKGROUND

To address these recommendations, the Government's Department for Business, Energy and Industrial Strategy (BEIS) have tasked Frazer-Nash Consultancy and our partner organisations with delivering a programme of research and development. The scope of work includes the following two work packages;

- Developing a Specification for an Innovative UK Thermal Hydraulic Modelling Capability;

  The Modelling Capability Specification will define the thermal hydraulic knowledge, understanding and modelling tools that are required within the UK to support reactor design and operation and to underpin safety cases. This will include the overall goals and targets of UK model development and recommend strategies for achieving those targets by assessing the potential and challenges associated with new and existing technologies.

  It is envisaged that the outcomes of this task will shape the direction of nuclear thermal hydraulic modelling research in the UK for the foreseeable future.

- Developing a Specification for a UK National Nuclear Thermal Hydraulic Test Facility;

  The Test Facility Specification will define the system requirements of a specific test rig, or rigs (referred to herein as 'the test facility'). The requirements for the test facility will be sufficiently mature to form part of a tender for the design and build of the test facility.

[1] NIRAB-75-10, March 2016

The test facility will provide a much-needed resource to support the UK nuclear industry. However, it is also intended to target some of the gaps between the global state-of-the-art and the industry capability needs, thereby making a positive contribution to the worldwide nuclear industry.

BEIS anticipate the test facility and model development activities will be completed in 2022.

2. STAKEHOLDER INPUT

2.1 AIM

It is important to the UK Government that any R&D that they fund is clearly linked back to benefit to the nuclear energy industry. As an organisation with an interest in the UK nuclear industry, you have been identified as a potential stakeholder in our project.

We are therefore seeking your contribution to help steer future UK Government-funded development work. The success of both thermal hydraulic work packages requires accurately capturing the end users' needs. Input into our requirements capture process will ensure that your needs are taken into account in future decisions.

Broadly speaking, the areas we are hoping to receive your input on include:

- The most challenging thermal hydraulic predictions affecting you now and in the future.
- Your current thermal hydraulic modelling methods and where you feel there would be the most to gain from development.
- Your current access to test/validation data and which additional data would be of most benefit to you.
- Your ideas and opinions on the scope and nature of future UK Government funded R&D into thermal hydraulic modelling.
- Your ideas and opinions on the scope and nature of a future UK nuclear thermal hydraulic test facility.

2.2 YOUR CONTRIBUTION

To provide this input, we would welcome you completing and returning the questionnaire in Section 3. With your permission, we will then initiate a follow-up discussion to ensure that we have captured your specific requirements accurately. We are eager for a wide response to this questionnaire and encourage you to answer as much as possible. However, please disregard any questions that you are unsure how to answer.

We would appreciate any supplementary information or evidence you can provide to support your answers.

2.3 TIMESCALES

We are seeking a response by 8th September 2017. We acknowledge this is a tight timescale and, if you are unable to meet the initial deadline, we would still encourage you to submit your response as soon as possible.

2.4 CONFIDENTIALITY

The final output of this study is likely to be a public domain document. It would therefore be beneficial to all parties if you could exclude any business sensitive information and only include responses you are happy to have in the public domain.

3. QUESTIONNAIRE

3.1 THE STATE OF THE ART

Question 1: Looking ahead between 5 and 10 years, what are the top 5 challenges you face in thermal hydraulic prediction and testing and what aspects of those challenges are most important to you?
Click here to enter text.

Question 2: Why are these predictions or tests important to you? Which design choices, safety cases or operational decisions do/will they support?
Click here to enter text.

Question 3: What modelling tools/methods do you currently have available to obtain the thermal-hydraulic information you need? What level of confidence do you have in these methods?
Click here to enter text.

Question 4: What thermal hydraulic tests currently support your design and where do you perform them? Why do these tests need to be performed (e.g. model validation, model uncertainty)?
Click here to enter text.

Question 5: What restricts the usefulness of current modelling methods, test results and test facilities? If possible, please briefly explain the cause of these limitations and their significance to you. *Examples may include: scaling challenges associated with small-scale test facilities; significant uncertainty in prediction of a key parameter under particular conditions; cost; regulator acceptance.*
Click here to enter text.

3.2 R&D HIGH-LEVEL PRIORITIES

Question 6: What improvements in thermal hydraulic modelling capability would make the biggest contribution to your project/organisation? *Please list your top 5 requirements in order of priority.*
Click here to enter text.

Question 7: What specific capabilities would this future UK based thermal hydraulic test facility need for you to make use of it?
Click here to enter text.

Question 8: What benefits would be gained, both direct and indirect, if the additional capability specified in your responses to Questions 6 and 7 were realised? *Please quantify benefits where possible.*
Click here to enter text.

3.3 DETAILED REQUIREMENTS AND IDEAS

Do you have a clear idea of what is needed to achieve these benefits? If so, we would like to hear your more detailed requirements and suggestions.

For example,

- Which thermal hydraulic phenomena need to be better understood or modelled?
- What conditions do you need a test facility to replicate and what parameters would you wish to measure?
- What is needed to increase regulator acceptance of the modelling methods you need?

Please be as specific as possible and include how you hope these more detailed requirements/ suggestions will address the higher level needs listed in your responses to Questions 6 and 7.

Question 9: Please describe any more detailed requirements or suggestions?
Click here to enter text.

Question 10: Is there any current research, publications or papers related to your requirements that you think we should be aware of? *Please provide references and/or locations where we can view the information.*
Click here to enter text.

Question 11: Are you happy for us to contact you with any questions that may result from our processing your answers?
Click here to enter text.

We kindly ask that you respond by email to Jordan McIntosh (j.mcintosh@fnc.co.uk).

Thank you for taking the time to complete this questionnaire!

DOCUMENT INFORMATION

Project :	Project FORTE - Nuclear Thermal Hydraulics Research & Development
Report Title :	Thermal Hydraulic Capability User Requirements
Client :	Department for Business, Energy and Industrial Strategy (BEIS)

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