Project Number: 945 041

D4.2 GFR Needs for Nuclear Standardization and Codes

Authors	Due date:	30.9.2023
Brian Daniels (Jacobs) Petr Vácha (ÚJV Řež)	Actual release date:	20.09.2023

Contributors	Approved by MST	Coordinator
Peter James (Jacobs) Petr Hájek, Lubor Žežula (ÚJV Řež)	Jakub Heller	Branislav Hatala
	Heller	Hatala

Version number: 1.1
Initially released on: 30/09/2023
Final version released on: 26/09/2023

Project start date: 01/10/2020 Project duration: 48 months

Dissemination level			
PU	Public	X	
RE	Restricted to specific group		
CO	Condifential (only for SafeG partners)		

Version control table

Version number	Date of issue	Author(s)	Brief description of changes made
1.0	20.09.2023	Brian Daniels Petr Vácha, Peter James, Petr Hájek, Lubor Žežula	Draft
1.1	26.09.2023	Jakub Heller	Reviewed by partners and MST Final version

Project information

Project full title:	Safety of GFR through innovative materials, technologies and processes
Acronym:	SafeG
Funding scheme:	Research and innovation action
ECGA number:	945041
Programme and call	Horizon 2020 Framework Programme for Research and Innovation (2014-2020) NFRP-2019-2020 (Nuclear Fission and Radiation Protection Research)
Coordinator:	Dr. Branislav Hatala
EC Project Officer:	Cristina Fernandez Ramos
Start date – End date:	01/10/20 – 30/09/2024 i.e. 48 months
Coordinator contact:	+421 905 567 985, Branislav.hatala@vuje.sk
Administrative contact:	+420 602 771 784, jakub.heller@evalion.cz
Online contacts (website):	www.safeg.eu

„This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945041".

Executive Summary

This report has been developed to meet deliverable D4.2 within the "safety of GFR though innovative materials, technologies and processes" (SafeG) project, under Task 4.3. The report provides a review of available codes and standards applicable to high temperature reactors, with specific focus on Gas-cooled Fast Reactors (GFR) to meet the stated aims of Task 4.3. Therefore, the aim of this report is to understand both the applicability of these codes to GFRs, and to identify where gaps may exist, which may further impact different regulatory regimes.

A review of available codes, standards and procedures that could be feasibly applied to the GFR has been performed. This has included an understanding of code basis, ongoing code development activities and adaptability of the codes. Significant understanding is well captured in prior work performed by Jacobs and EDF in the UK (as part of the EASICs project); as such a review of the EASICs project and key findings has also been performed. These reviews have helped inform current understanding in a general sense. A more specific review focused on the potential challenges for a GFR is then provided.

The review included in this report has identified a number of key challenges and the potential areas which could benefit from further standards and codes development. The identified potential areas are:

- Metallic components
  - Some through-life assessment methods include viable alternatives for elements of the design codes where the current design codes are not sufficient. Some example aspects include:
    - Ratcheting due to large temperature excursions leading to excessive distortion;
    - High cycle fatigue due to thermal fluctuations (e.g. thermal striping);
    - Creep rupture at elevated temperature;
    - Thermal ageing at elevated temperature;
    - Crack growth by a combination of creep, fatigue and environmental factors;
    - Buckling and/or creep buckling in thin-section components.
  - The requirement for the application of post-weld heat treatment to all welds where possible.
  - Reduction of the number of highest reliability locations and minimise integrity claims on welds.
  - The influence of environment (chemical) on the structural integrity claims that can be made are considered as an important issue.
  - Long-term testing is considered in a representative environment at the stresses, strains and temperatures relevant to plant loading conditions.
  - Ensure that the different manufacturing techniques (including weldments) do not degrade the material properties and have associated long-term materials data to support their deployment.
  - Impact of the helium environment with the expected impurities content on creep, tensile and fracture properties should be specifically examined.
  - Existing materials within the codes may prove challenging for high temperatures at longer durations.
  - Very high temperature data of the existing materials within the codes are limited and not sufficient for the full qualification of GFR technologies under operation, transient, and accident conditions
  - For the GFR, the approach to component classification should be clearly detailed.
- Ceramic composite components
  - Definition of the component classification and acceptable probability of failure.

- Sufficient testing to characterize the material reliability and possible failure modes such that provide design curves can be produced. The testing should also look to support the development of materials models mature enough to allow design by analysis methods to be used.
- Qualification of identified ceramic material behaviour with irradiation.
- Monitoring and NDT inspection of the ceramic core.
- Design ceramic core for decommissioning.

▪ Reactor core considerations

- Minimising and looking to reduce any impact of flow induced vibrations should be considered.
- The effects of deformation within fuel sub-assemblies and reactor core and the subsequent impact on the reactor internals may need to be considered further.

▪ Non-destructive testing

- Development of small remotely operated high-temperature NDT equipment for GFR applications.
- Revision of existing NDT requirements in international nuclear codes and standards to include remotely operated NDT equipment requirements.

It is also considered a worthwhile activity to review ongoing developments for other high temperature reactors, in particular the HTGR, to see where any learning can be made or where co-development opportunities may exist.

Contents

Acronyms and Abbreviations

AFCEN	English translation as French Association for Design, Construction and in-service inspection rules for Nuclear Island Components
AGRs	Advanced Gas-cooled Reactors
ARs	Advanced Reactors
AMRs	Advanced Modular Reactors
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ATLAS	EPRI Advanced Technology for Large Scale project
BEIS	Department for Business, Energy and Industrial Strategy
BPVC	Boiler and Pressure Vessel Code
BEPU	Best Estimate Plus Uncertainty
B&PV	Boiler and Pressure Vessel
CMC	Ceramic Matrix Composite
CO_2	Carbon Dioxide
CSD	Control and Shutdown Devices
DSD	Diverse Shutdown Devices
dpa	Displacement per atom
EASICS	Establishing AMR Structural Integrity Codes and Standards
EC	Eddy Current
ESNII	European Sustainable Nuclear Industrial Initiative
FFS	Fitness For Service
FSRF	Fatigue Strength Reduction Factor
FFTF	Fast Flux Test Facility
GDA	Generic Design Assessment
GFR	Gas-cooled Fast Reactor
Gen IV	Generation IV
He	Helium
HIP	Hot Isostatic Pressing
HT	High Temperature
HTGRs	High-Temperature Gas-cooled Reactors
ISI	In-Service Inspection
JSME	Japan Society of Mechanical Engineers
LFR	Lead-cooled Fast Reactor
LMRs	Lead-Metal Reactors
MI	Manufacturing inspection
MMA	Manual Metal Arc
MSR	Molten Salt Reactor
NDT	Non-Destructive Transition Testing
NIP	Nuclear Innovation Programme
NQA-1	Nuclear Quality Assurance Part 1
NPP	Nuclear Power Plant

ODS	Oxide Dispersion Strengthened
OPEX	Operating Plant Experience
ONR	Office for Nuclear Regulation
PLEX	Plant Life Extension
PM/HIP	Powder Metallurgy and Hot Isostatic Pressing
POF	Probability of Failure
PSI	Pre-Service Inspection
PWHT	Post Weld Heat Treatment
PWR	Pressurised Water Reactor
RPs	Requesting Party
RPV	Reactor Pressure Vessel
RT_{NDT}	Nil-Ductility Temperature
R&D	Research & Development
SafeG	Safety of GFR though innovative materials, technologies and processes
SAPs	Safety Assessment Principles published by UK ONR
SCWR	Supercritical Water Reactor
SFR	Sodium-cooled Fast Reactor
SI	Structural Integrity
SiC	Silicon Carbide
S_r	Rupture stress
S_{mt}	Primary membrane stress intensity limit
SNETP	Sustainable Nuclear Energy Technology Platform
S_t	Temperature and time-dependent stress intensity limit
SS	Austenitic Stainless Steel
TASCS	Thermal Stratification, Cycling, and Striping
TIG	Tungsten Inert Gas
T_m	Melting point temperature (in Kelvin)
T_i	Irradiation temperature (in Kelvin)
UK	United Kingdom
USR	Upper Shelf Region
VHTR	Very High Temperature Reactor
ΔV	Volume change
V_0	Original volume
WPs	Work Packages
WSEF	Weld Strain Enhancement Factor

1. Introduction

This report has been developed to meet deliverable D4.2 within the "Safety of GFR though innovative materials, technologies and processes" (SafeG) project, under Task 4.3 [1]. The report provides a review of available codes and standards applicable to high temperature reactors, with specific focus on Gas-cooled Fast Reactors (GFR) to meet the stated aims of Task 4.3. Therefore, the aim of this report is to understand both the applicability of these codes to GFRs, and to identify where gaps may exist, which may further impact different regulatory regimes. It is noted that for European application, the RCC-MRx [2] code is typically applied for high temperature reactors but was developed for liquid metal-cooled reactors; likewise, the R5 procedure in the UK was developed alongside CO_2 cooled reactors. As such, it is anticipated that there will be gaps within the available codes, standards and procedures when applied to GFRs.

It is noted that high temperature reactors belong to the family of so called Advanced Reactors (AR), Advanced Modular Reactors (AMR) or Generation IV (Gen IV) reactors. Other terms are used depending on the source reference material, however high-temperature or ARs have typically been used. Here, it is noted that there is not a consistent definition of "high temperature" across codes and standards (and procedures). The approach commonly considered is that high temperatures are where creep effects cannot be discounted, which may alter for different materials (for instance the ASME limits suggest approximately 370°C for carbon and low alloy steels, and 425°C for austenitic stainless steels).

It is noted that the report makes use of prior work performed by Jacobs and EDF in the UK as part of the EDF Energy led EASICs project [3]. The EASICs project looked to identify good practice application for assessment of high temperature AR for the generic design assessment (GDA) process in the UK.

This report has been separated to consider:

- Section 2 - Available codes, standards and procedures that are applicable to the GFR. This first introduces the codes, standards and procedures, then discusses some ongoing code development activities and finally discusses the adaptability of the codes for GFR plant.
- Section 3 - An overview of the EASICs project is provided which reviews, operational experience (OPEX) of high temperature reactors, comparison of high temperature design codes, pertinent findings and recommendations for high temperature reactor build.
- Section 4 - A short overview of the ALLEGRO GFR design is provided. This helps inform the potential materials and structural integrity challenges which could be expected for GFR plant. Where possible, this is related to codes and standards guidance to understand any potential gaps.
- Section 5 – The main challenges and potential code development areas are presented.
- Section 6 – Main conclusions from the work are outlined.

2. Codes, Standards and Procedures

This section provides a high-level overview of the background to the review and available codes and standards that are likely to be applicable to the ALLEGRO GFR design. As such, this has focused on the high temperature parts of ASME (ASME III, Division 5, but also on the through-life parts in ASME XI), AFCEN codes (RCC-MRx, but also comments on RSE-M) and UK specific codes (R5 and R6). Low temperature design codes such as Japan Society of Mechanical Engineers (JSME) have not been included. Also included within this section is a high-level overview of how easily these codes and standards, and procedures, can be updated for new reactor designs.

2.1 Overview of Available High Temperature Codes and Standards and other Applicable Procedures to Demonstrate Integrity

Design codes and standards can be defined as a set of technical definitions, guidelines, and instructions for designers, manufacturers, vendors and operators. The aim of design codes and standards is to promote safety, reliability, productivity, and efficiency. As such, design codes are intentionally general, and therefore can relate to almost every industry that relies on engineering components or equipment.

Design codes and standards are considered voluntary in some industries and countries because they serve as guidelines, but do not themselves have the force of law. Conversely, design codes and standards can become mandatory, when they have been incorporated into a business contract or incorporated into regulations (such as ASME). For the design of a nuclear power plant, it is necessary to consider both pressurised equipment and nuclear specific codes. This provides further complexity in terms of the materials, procedures, manufacturing routes, inspection requirements, etc. that are required for the design of new nuclear plant.

The following sub-sections provide a high-level overview of the main design codes and assessment procedures. This is split to consider a summary of the design codes themselves, an historical review of application to existing and planned nuclear power plants (NPP), and a specific commentary to the expected use of high temperature design codes.

2.1.1 ASME III

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) design codes are the most widely adopted nuclear design code internationally, and forms the basis for many other codes. ASME itself is an accredited Standards Developing Organisation that meets the due process requirements of the American National Standards Institute (ANSI). Design codes and standards that are developed under an accredited program may be designated as American National Standards.

ASME develops and revises the B&PV design codes (generally referred to as ASME) based on market needs through a consensus process, whose meetings dealing with standards-related actions are open to all members of the public [4]. The ASME consensus committees are comprised of volunteer subject matter experts from a diverse range of interests, including manufacturers, operators, the American government, and general industry interest. ASME B&PV design codes and subsequent revisions are based upon a review of reliable technical data by the consensus committee and its sub-tier committees.

The ASME process includes a broad public review for all of its ASME B&PV design codes actions. Any interested member of the general public may review and comment on proposed ASME B&PV design codes or revisions, as well as initiate an appeal based on previously submitted concerns.

2.1.1.1 Conventional (Low Temperature) Plant

ASME has been developed and extended over many years to consider conventional sized pressurised water reactors (PWR). Maximum temperature limit defined for conventional plant is between 370°C (i.e. carbon & low alloy steel) and 425°C (i.e. austenitic stainless steel). The ASME code covers a range of topics, including wider aspects such as civil requirements. The main areas of the code relating to the design of the nuclear island are included within ASME Section III [5], with some further information (e.g. materials data) in ASME Section II [6] and through-life approaches in ASME XI [7].

The ASME classification of vessels and piping into Class 1, Class 2 and Class 3 is specific to nuclear applications, together with the requirement to design pressure equipment for a specifically quantified lifetime transient loading. The allowable levels of stress intensities also vary according to the reactor 'service loadings' planned. These different levels are for key safety related or high integrity components through to those which are less safety critical.

ASME also inherently includes a quality assurance and quality control process, based on generic pre-qualification assessments, which enable manufacturers to obtain a specific "Nuclear Component (N)" stamp. There are also similar stamps for nuclear installers, nuclear parts, and nuclear safety valve manufacturers. These stamps certify that companies have the right quality systems and work control practices to make the quality grade of components required to obtain the stamp. ASME III Article HAB-4000 [8] refers the users to the Nuclear Quality Assurance Part 1 (NQA-1) standard [9] which provides requirements and guidelines for the establishment and execution of quality assurance programs during siting, design, construction, operation and decommissioning of nuclear facilities. Nuclear facilities can include nuclear power plants, small modular reactors, and advanced reactors.

2.1.1.2 ASME III Division 5

ASME Boiler and Pressure Vessel Code (BPVC) Section III Division 5 (ASME III Division 5) [8] provides construction rules for high-temperature reactors, including both high-temperature gas-cooled reactors (HTGRs) and liquid-metal reactors (LMRs). To guard against failure whilst operating at higher temperatures, the rules for Class 1 components detailed in ASME III (Subsection NB), should not be used for operating temperatures that exceed the temperature limits of ASME Section II [6], Part D (370°C for carbon and low alloy steel, and 425°C for austenitic stainless steel). Above those temperatures, the creep and stress rupture characteristics of materials permitted to be used become significant factors that are described in more detail within Division 5. The upper temperature limits for application are dependent on the available materials data in the codes, which differs depending on the material.

The ASME III Division 5 design code covers the material, design, fabrication, examination, installation, testing, overpressure relief, marking, stamping and preparation of reports. All material data required for design calculations are included in the code. It is noted that these requirements are linked to the general PWR requirements elsewhere within ASME III, which should also be met unless the ASME III Division 5 rules say otherwise.

ASME III Division 5 [8] applies to components exceeding specified material-dependent temperatures where creep effects are significant; generally termed "high temperature". In addition to the requirements elsewhere in ASME III, the information contained in ASME III Division 5 also considers the time-dependent material properties and structural behaviour by guarding against the seven failure modes:

- Ductile rupture from short-term loadings,
- Creep rupture from long-term loadings,
- Creep fatigue failure,
- Gross distortion due to incremental collapse and ratcheting,
- Loss of function due to excessive deformation,
- Buckling due to short-term loadings,
- Creep buckling due to long-term loadings.

Non-mandatory Appendix HBB-T of Section III Division 5 provides rules for strain, deformation and fatigue limits at elevated temperatures. HBB-T-1710 defines special requirements at welds. HBB-T-1800 provides the isochronous stress-strain data required for simplified strain limits, relaxation and relaxation creep damage calculations.

2.1.2 AFCEN (RCC-M, RCC-MRx)

2.1.2.1 Overview of Code

AFCEN (English translation as French Association for Design, Construction and in-service inspection rules for Nuclear Island Components) is the governing body, which provides oversight to the codes and standards developed and used in France. AFCEN extended the range of technical fields covered, with three codes for mechanical components: RCC-M (fabrication), RSE-M (in-service inspection) and RCC-MRx (high-temperature reactors, experimental reactors and fast-neutron reactors); one code for electricity and I&C systems (RCC-E); one code for nuclear fuel (RCC-C); one code for civil engineering works (RCC-CW), and one code for fire protection systems (RCC-F).

The main aspects related to this review are included within RCC-M and RCC-MRx.

RSE-M [10] is a through-life code and, therefore, has not been directly considered here. In many aspects the information contained in RSE-M is similar to that in RCC-M and RCC-MRx, with added areas such as that for the fracture assessment. However, a review of the information contained within RSE-M [11], particularly for the fracture mechanics approaches contained, has been conducted from a UK perspective for the EPR GDA process to help address questions raised by the UK Office for Nuclear Regulation (ONR). The review conducted demonstrated strong similarities to R6.

2.1.2.2 RCC-M (PWR)

The first RCC-M design code, published in 1980, was based on the ASME III design code (under an adaption agreement) with deliberately similar title headings and clause numbering system to aid correspondence and ease of use [12]. The current RCC-M design code reflects the OPEX gained from the fleet of French PWRs over many years. In some areas there are optional routes provided through the code, such as in the calculation of fatigue damage, where more responsibility is placed on the designer to justify design choices and assumptions that are made in the design calculations rather than rely on a prescriptive set of rules. The RCC-M code is not mandatory but embodies practices, procedures and criteria to satisfy the regulations, and is recognised and recommended by the French safety authorities [12].

RCC-M failure modes considered include: excessive deformation, instabilities (e.g. plastic, elastic, & elastic-plastic), progressive deformation (or ratcheting), fatigue, and fast fracture. RCC-M design code applies to pressure equipment in nuclear islands in safety Classes 1, 2 and 3, and certain non-pressure components, such as vessel internals, supporting structures for safety class components, storage tanks and containment penetrations.

The RCC-M [13] classification system is similar to ASME, having Class 1, Class 2 and Class 3 components where the allowable levels of stress (or stress intensity) vary according to the reactor ‘service loadings’ that are under consideration. However, the service limits and material allowable stress levels in RCC-M are different to ASME code. Unlike the ASME code, RCC-M does not require manufacturers to attain any accreditation before they can manufacture equipment to RCC-M.

2.1.2.3 RCC-MRx

RCC-MRx [2] was developed specifically for sodium cooled fast reactors, research reactors and fusion reactors. Therefore, the RCC-MRx design code rules build upon those for RCC-M to include additional rules for design and construction of mechanical components involved in areas subject to significant creep and/or significant irradiation. In particular, the code incorporates an extensive range of materials (in particular materials with low absorption cross sections such as aluminium and zirconium alloys), sizing rules for thin shells and box structures, and more modern welding processes such as, electron beam, laser beam, diffusion bonding and brazing. The RCC-MRx code was adopted for the MYRRHA research reactor [14] with a lead-bismuth coolant and also the sodium-cooled fast reactor concepts ASTRID in France [15] and Prototype Fast Breeder Reactor in India [16].

The RCC-MRx [2] is a design code dedicated to high temperature reactors, fusion reactors and research reactors. RB3000 rules in RCC-MRx [2] are provided to ensure that the components are sufficiently safe under the various mechanical damages (and the time dependent damage equivalents where creep or fatigue effects increase the damage accumulation) including: excessive deformation, plastic instability, fracture, progressive deformation, and fatigue. No specific temperature range is defined for using RCC-MRx, but it is assumed to be similar to ASME III Division 5 where temperatures above PWR conditions (above approximately 400°C) will need to consider high temperature effects and the upper bound is dependent on the materials data available.

All material data required for design calculations are included in the code, and in addition, a guideline has been published for non-codified materials. The design principles are similar to ASME III, with the definition of loads to be considered for the integrity demonstration (RB 3130 and RB 3140) associated to criteria levels (RB 3150, level A to level D). Damages to be prevented (RB 3120) are the followings:

- Type P (primary) damages;
    - Immediate excessive deformation,
    - Immediate plastic instability,
    - Time-dependent excessive deformation,
    - Time-dependent plastic instability,
    - Time-dependent fracture,
    - Elastic or elastoplastic instability.
- Type S (secondary) damages;
    - Progressive deformation,
    - Fatigue (progressive cracking).
- Buckling,
- Fast fracture.

Different sets of rules have to be applied depending on the results of the following tests:

- Negligible creep tests (RB 3216.1),
- Negligible irradiation test (RB 3216.2),
- Negligible ageing tests (RB 3216.3).

RCC-MRx follows a similar approach to ASME in that fracture is not considered in the design phase beyond basic checks for brittle fracture. If a fracture assessment should be required, the approaches in RCC-MRx Appendix A16 or RSE-M [10] would be called upon. The approaches in RSE-M and Appendix A16 include conventional elastic-plastic fracture approaches as considered in R6 [17].

2.1.3 R5 and R6

2.1.3.1 Overview of Procedures

The R5 [18] and R6 [17] assessment procedures are the UK structural integrity assessment procedures, which have been continuously developed within the UK power generation industry since the 1970's and contain a well validated approach to consider the defect tolerance of a component. Flaw tolerance arguments are a key part of the safety case for operation of many components, and the challenges to such arguments become more demanding as the plant ages. Hence, there is a drive to improve existing methods by incorporating developments in fracture mechanics and high temperature creep understanding, including novel and more advanced methodologies.

It is noted that these are procedures (not codes) and provide alternative routes and guidance depending on the complexity of the assessment being performed and are therefore not prescriptive nor mandatory.

2.1.3.2 R6 (Quasi-Static Loading) Assessments

The "Assessment of the Integrity of Structures Containing Defects", known as the R6 procedure [17] is comprised of the following Chapters:

Chapter I: Basic Procedures,

Chapter II: Inputs to Basic Procedures,

Chapter III: Alternative Approaches,

Chapter IV: Compendia,

Chapter V: Validation and Worked Examples.

The R6 procedure [17] has been developed to assess the integrity of nuclear and conventional plant operating at low temperature (i.e. resulting in negligible creep) such that the loading can be considered quasi-static. The limiting condition of a structure is evaluated by reference to two criteria, fracture and plastic collapse. Structural integrity relative to the limiting condition is evaluated by means of a failure assessment diagram that corrects the elastic stress intensity factor assessed for elastic-plastic conditions (and is therefore an elastic-plastic J-estimation scheme, similar to that considered in the RSE-M code, although presented differently).

The R6 methodology can be applied to the defect tolerance assessment of components subject to primary (such as pressure) and secondary stresses (such as thermal and residual). The approach contained with R6 forms the basis for many other approaches such as that included in BS7910 [19] and RSE-M [10].

2.1.3.3 R5 (High Temperature) Assessments

The R5 procedure [18] has been developed to assess the integrity of nuclear and conventional plant operating at high temperature, such that time-dependent creep conditions are included, with a preliminary focus on the UK Advanced Gas Reactors (AGR) design. Within R5, there are specific procedures for assessing creep-fatigue initiation in initially defect-free components (Volume 2/3) and for assessing components containing defects (Volume 4/5).

The approaches contained within R5 have been developed partly under the UK fast reactor programme and improved following continual application to AGR in the UK and hence they incorporate a significant level of high temperature OPEX. As with R6, the approaches are not within the design basis but would be used for through-life considerations and, therefore has potential to be applied in the GDA process for Gen IV reactor designs in the UK.

The R5 procedure is comprised of the following volumes:

Volume 1: The Overview,

Volume 2/3: Creep-Fatigue Crack Initiation Procedure for Defect-Free Structures,

Volume 4/5: Procedure for Assessing Defects under Creep and Creep-Fatigue Loading,

Volume 6: Assessment Procedure for Dissimilar Metal Welds,

Volume 7: Behaviour of Similar Weldments: Guidance for Steady Creep Loading of Ferritic Pipework Components.

It is worth noting that the integrity assessment procedure used in design codes are approximately equivalent to that found in R5 Volume 2/3, which considers the initiation of a defect in an uncracked structure (noting some that R5 Volume 2/3 addresses a number of issues which are not addressed in design codes, see Section 5). R5 Volume 4/5 considers crack growth under creep and creep-fatigue loading. R5 Volume 6 considers dissimilar metal welds and is predominantly focused on pressure-loaded, butt-welded components. R5 Volume 7 outlines specific application of the R5 procedure to ferritic weldments, where Type IV cracking must be considered.

Development of the high temperature R5 structural integrity assessment procedure [18] to account for the environmental interactions; both crack initiation (Volume 2/3) and subsequent crack growth (Volume 4/5). The key requirement here will be the availability of the appropriate test data to support the relevant materials being considered for future GFRs.

2.2 Ease of Code Modification

The approaches to modify the codes and procedures differ for those identified above, depending on if they are a code or procedure.

The codes (i.e. ASME and RCC-MRx) follow a process where significant changes are first introduced via a "code case" in ASME or a "probationary rule" in RCC-MRx. These are initially proposed by the respective code committees (the groups that are responsible for that part of the code) and then get reviewed by the wider code development organisations. To introduce a code case or probationary rule can take anywhere between a year to more than 10 years to get accepted (a review of the current code cases [20] show some from prior to 2010) depending on the complexity and scope of the changes. The main challenges for such changes are the knock-on effects to other parts of the code, the verification and validation performed, and the potential legal implications when adjusting something previously accepted for use. A further complexity, at least for ASME, is that any objections to the changes from the reviewing board will mean the proposal is not accepted and fed back to the proposing committee; as such any more contentious changes could be very challenging to make. Less complex changes, explanations, or simple editorial changes, will still need to go through the same review process but are normally accepted in less than a year.

In contrast, changes to the R5 and R6 procedures are relatively more straight-forward. Changes are proposed from within the development programme and supplied to the R5[1] and R6[2] committees for comment. After committee acceptance these are incorporated to the procedures and a final review performed by EDF prior to issue. A rolling programme of targeted updates is included in the programme oversight meaning each section of R5 and R6 will be reviewed and re-issued every 5-10 years (some more frequent).

The R5 and R6 procedures do not include material properties and can be applied to any available well defined materials data. Conversely, the design codes include specific materials data that should be applied within the code. As noted above, it is normally this materials data which limits the application of the codes (in terms of

[1] The R5 panel currently consists subject matter experts from EDF, Frazer-Nash Consultancy (until the end of 2023), Rolls-Royce, Australian Nuclear Science and Technology Organisation (ANSTO) and Jacobs.

[2] The R6 panel currently consists subject matter experts from EDF, Engineering Analysis Services Limited (EASL), NRG (the Dutch Nuclear Research and Consultancy Group), Rolls-Royce, TWI (until the end of 2023) and Jacobs.

upper bound temperature or lifetime for instance). As a result, there may be a desire to introduce new materials to the codes for some ARs. This process can take a long time to both generate the respective data needed (see Section 5.6 below) and the time to gain acceptance from the code committee. In 2020, ASME introduced Alloy 617 to the code after 12 years of development and $15 million investment, which is also the first material to be added to the main section in over 30 years [27] (noting some adjustments for high temperature materials are ongoing). RCC-MRx are more proactive with introducing more materials, with a few planned (noting there is still time and expense to test the materials).

3. Review of EASICs Findings

This section provides an overview of the work performed within the "Establishing AMR Structural Integrity Codes and Standards for UK GDA" (EASICS) project as part of the Phase 2 Nuclear Innovation Programme (NIP) funded by BEIS. This part summarises the draft Guidance Document currently under review by BEIS [3].

3.1 Background

The EASICS project was established to help define the requirements for codes and standards for the design of AMR needs in order to ensure that state-of-the art knowledge will be brought to bear on developing the required design and assessment methodologies. The EASICS project was carried out between July 2019 and December 2021 and was project managed by EDF, in partnership with Rolls-Royce and NNL.

The project builds on three of the UK Government's Department for Business, Energy and Industrial Strategy (BEIS) Nuclear Innovation Programme (NIP) Materials and Manufacturing Phase 1 Projects (specifically those detailed in References [21] and [22] looking at potential gaps in design codes), which highlighted a number of shortfalls and opportunities in the structural integrity codes and standards area for AMRs. Several of the high priority areas were selected for consideration within EASICS based upon: 1) the potential to reduce costs; 2) shortfalls in existing codes and standards; and 3) generic applicability to a number of AMR designs to avoid vendor specific issues.

The following Work Packages (WPs) were undertaken within EASICS:

- WP1 - Probabilistic Design: This has been led by Rolls-Royce and has looked to provide specific case examples using the Best Estimate Plus Uncertainty (BEPU) approach to design specific cases. This package also looked to understand how to more effectively use probabilistic approaches in plant design.
- WP2 - Thin Section Defect Tolerance: This has been led by EDF with the support of Frazer Nash Consultancy and Jacobs. The aim of this programme has been to look at the assessment of thin structures, specifically at the assessed fracture toughness. This has included performing and assessing specimens that have an out-of-plane constraint loss.
- WP3 - Creep-Fatigue Behaviour: This has been led by EDF with the support of Jacobs. This has included more complex creep-fatigue testing of well-characterised material (316H) and also compared materials relevant to AMR designs to ensure current understanding can be read-across.
- WP4 - Codes and Standards Guidance: This has been led by EDF and NNL with the support of Jacobs. This is discussed further below.

WP4 draws all of the findings from the other work packages together to provide the headline project deliverable: A high-level UK specific guidance to help define the requirements for codes and standards for the design of AMRs (typically based on high temperature Gen IV reactors). This should therefore help ensure that state-of-the art knowledge can be brought to bear on developing the required design and assessment methodologies. In addition, high temperature mechanical testing and assessment work was undertaken in support of WP4. The work package was broken-down into four tasks:

- WP4 - Task 1: Welded Tube Validation Testing (which is applicable to R5 Volume 2/3 Appendix A4),
- WP4 - Task 2: Weldment Assessment Route Development through Inelastic Analysis (which is applicable to R5 Volume 2/3),
- WP4 - Task 3: Comparison of ASME Section III Division 5, RCC-MRx and R5 procedures,
- WP4 - Task 4: AMR GDA Guidance Document Development.

3.2 Regulatory Framework and Available Design Codes

The Guidance Document developed under EASICs was intended for use within the UK GDA process and includes detailed sections on both the GDA process and available design codes. Subsequently, this has not been summarised here as this may not be applicable for the ALLEGRO GFR.

3.3 Operating Experience of High Temperature Reactors

A particularly pertinent part of the Guidance Document examined the available OPEX of early advanced reactors. This included a specific focus on the adequacy of current design codes. The information included within the Guidance Document [3] is extensive and too detailed to repeat here but is considered worth reviewing in the context of this report. The headline approach and main conclusions from the OPEX review are included here.

Over the past 60 years, OPEX was gained on four classes of Gen IV or advanced nuclear systems: helium cooled high temperature gas cooled reactors (HTGR); sodium cooled fast reactors (SFR); molten salt reactors (MSR) and molten lead and lead-bismuth eutectic cooled reactors (LFR). For HTGRs and SFRs, quite substantial OPEX was gained with the construction of both experimental small reactors and larger prototype or demonstration reactors with electricity generation. There is much less experience for LFRs and MSRs, see Table 1. This table is only approximate as the operating histories of the reactors were complex.it is In addition, it was difficult to decide what experience is relevant. For example, the LFR experience is with reactors that have an intermediate spectrum rather than a fast spectrum, a lead-bismuth coolant and with very different configuration and fuel compared to current designs for civil power applications.

Despite this, some very valuable lessons were learned on the management of reactors with molten lead alloy coolants. AGRs have been included in the tables below as they are comparable to the other systems discussed in many ways, but do not count as a Gen IV reactor system.

Table 1 - Estimates of relevant operating experience with Gen IV reactor systems (taken from [3])

System	Operating experience including other activities than electricity production (reactor operational years)	Operating experience from electricity production registered on IAEA PRIS [23] (reactor full power years equivalent)
High Temperature Gas Cooled Reactor (HTR/HTGR/VHTR)	~150	20.1
Sodium Cooled Fast Reactor (SFR)	Oxide fuel ~250 Metal fuel ~25	Oxide fuel 61.4 Metal fuel 5.6
Lead Cooled Fast Reactor (LFR)	~70 in submarines Not fast flux, Pb-Bi coolant	0
Molten Salt Reactor (MSR)	Thermal ~2, Fast 0	0 - No electricity generation
Advanced Gas-cooled Reactors (AGR)	~530	~365

Interest in the development of these advanced systems waned after the Chernobyl accident in 1986, partly due to safety concerns but mainly due to the low price of oil and gas, so most of the OPEX discussed here is prior to 1996, by which time nearly all of the advanced reactors built had been shut down. The exceptions were in Russia and India, where SFRs continued to operate, and new examples built. Japan also maintained its development of HTGRs and commissioned an experimental HTGR in 1999. China started work in the 1990's with construction of a small Russian designed fast reactor and a HTGR.

It worth noting that all the Gen IV reactor classes are operated at higher temperatures than the water-cooled reactors that are currently used to generate the bulk of nuclear electricity, see Table 2. Some of the ARs will have similar inlet and outlet temperatures as the AGRs and will experience similar issues with primary circuit and steam generator structural materials. The AGR OPEX has been included for comparison, as degradation mechanisms may be similar and therefore the OPEX of AGRs is expected to be hugely beneficial when considering AMRs. A more extensive review of AGR OPEX is included in the Guidance Report [3].

Table 2 - List of the 6 Gen IV Forum reactor systems [24], with similar data for water and AGR reactors (taken from [3])

System	Neutron Spectrum	Coolant	Primary inlet/ outlet T (°C)	Primary circuit pressure (MPa)
Sodium Cooled Fast Reactor (SFR)	Fast	Sodium	280 - 400/ 430 - 565	Low pressure
Lead Cooled Fast Reactor (LFR)	Fast	Lead or Pb-Bi	350 - 420/ 445 - 540	Low pressure
Molten Salt Reactor (MSR)	Fast/ Thermal	Fluoride or Chloride Salts	560 - 660/ 650 - 800	Low pressure
Gas Cooled Fast Reactor (GFR)	Fast	Helium	260 - 550/ 530 - 900	7-13
High Temperature Gas Cooled Reactor (HTGR)	Thermal	Helium	250 - 650/ 750 - 1000	5-7
Supercritical Water Reactor (SCWR)	Thermal/ Fast	Water	270 - 300/ 510 - 625	~25
Gen III Water Reactors, PWR, BWR, Candu, etc.	Thermal	Water/ Heavy Water	250 - 290/ 280 - 330	7 - 16
Advanced Gas-cooled Reactors (AGR)	Thermal	CO_2	275 - 292/ 634 - 675	3.4 - 4.4

The amount of information on the OPEX of advanced systems is large but can be difficult to find and for some reactors there are almost no publicly available documents. In addition to specific reports and publications for individual reactors, the IAEA has published reports summarising the status of development in the TECDOC series. Furthermore, the US NRC engaged consultants to carry out a review of OPEX for SFR and HTGR reactors, which was completed in 2019 [25] [26]. There was also a related US NRC study on molten salt reactors [27] [28].

The main conclusions from the more extensive OPEX review in the Guidance Document [3] are repeated below:

- The four Gen IV systems that have been explored to date share issues on dealing with higher temperatures that impact both on reactor structural materials and energy conversion systems that overlap with experience on AGR reactors. This is reflected in the issues found on thermal ageing, fatigue and creep in primary circuit materials and especially in the welds that are left with significant residual stress.
- The only significant experience on fast neutron damage to structural materials has come from the OPEX on the prototype and large sized SFRs. This is an important body of work for future reactor systems with higher fast fluxes, and has shown that austenitic stainless steels have limited operating lives in such environments.
- Fast neutron damage results in enhanced irradiation induced swelling and material embrittlement. To date, helium embrittlement, which is more important at higher temperatures, has mainly impacted on component life from the presence of boron in alloys or steels.
- The largest area of uncertainty for structural integrity is corrosion, which is reflected by the different materials solutions required for each Gen IV reactor type. OPEX has shown the importance of choosing structural materials that are matched to the working environment.
- Control of the environment is important and many of the issues with the OPEX so far have been the result of contamination of the primary circuit with oxygen, water and carbon.
- Similarly, problems are found on the waterside, with chloride ion and caustic contamination. Care also has to be taken to protect pipework and vessels from external contamination and corrosion.

- Many of the issues with ARs were related to auxiliary systems. For example: decay heat removal loops; coolant monitoring and purification systems; storage and transfer of new and used coolant and fuel; control rod drives; instrumentation: etc. In the case of MONJU failures in such systems contributed to their low availability this resulted in its early closure. Superphénix did also have multiple issues but these had been resolved by the time it was closed down; its closure was more politically driven.

The main conclusions from the AGR specific OPEX review within the Guidance Document [3] are:

- Structural integrity challenges, including creep-fatigue, carburisation, reheat cracking, oxidation, thermal ageing and stress corrosion cracking in the AGRs has led to billions of pounds in lost revenue for the AGR operators.
- In all cases of significant plant degradation, significant research and development (R&D) programmes have had to be established to develop suitable understanding of the degradation mechanisms to demonstrate it was safe to continue operating.
- In several cases (creep-fatigue, reheat cracking and carburisation), this has led to improved assessment methods which have been included in the R5 assessment procedure. Other mechanisms have been assessed using other “in-house” procedures.
- In cases where the AGR owners were driven to repair or replace components, the level of access made a significant difference to how easy a challenge was resolved. In many cases replacement may have been a preferable solution rather than lengthy outages developing repair solutions, but this was rarely an option in the AGR designs.
- The oxidation monitoring scheme provided a hugely valuable source of information, to mitigate emergent risk experienced later in life due to environmental effects.

3.4 Technical Comparison of Design Codes

A technical comparison of the main high temperature design and assessment codes (ASME III Division 5, RCC-MRx and R5) was included within the Guidance Document. The similarities and differences within these codes and standards were identified over the following aspects:

- Classification of Components,
- Material Properties,
- Stress Analysis,
- Stress Classification,
- Plastic Collapse,
- Ratchetting/Shakedown,
- Creep Rupture,
- Fatigue Initiation,
- Creep-Fatigue Crack Initiation,
- Environmental Factors,
- Reheat Cracking,
- Defect Tolerance/Avoidance of Fracture.

More specific information is included in Reference [3]. However, from the technical comparison of a number of structural integrity considerations pertaining to the approaches in ASME III Division 5, RCC-MRx and R5 are repeated below as:

- There are no set criteria for component classification and that further guidance may be required.
- Material properties (i.e. thermal and mechanical) are code specific, but there are possible limitations for applying existing material properties to AMR designs, especially for higher temperatures or longer durations than currently considered.

- The basic stress analysis approach is similar between codes, although some specific differences exist for welds.
- The approach to classifying stresses is essentially the same for all approaches. The main differences here relate to how prescriptive the codes/approaches are for given components, with ASME being the most prescriptive.
- The general approach to plastic collapse between the codes/approaches is very similar, although there are some subtle variations on how the limiting stress is calculated. A high level comparison of the approaches is included in Table 3 below.
- Additional design limits on secondary and peak stresses limit excessive deformation but do not necessarily provide a demonstration of shakedown. R5 is the only approach to explicitly consider this.
- The basic approaches for creep rupture are similar, but the definition of the stress used as input to the stress rupture curves differs between the approaches. It is likely that the reference stress in R5 will be the lowest of these stress values. However, the potential differences in the rupture curves used will also have an impact on the results.
- There are a number of differences in the approaches for fatigue and creep-fatigue in the approaches. The three case studies detailed in Reference [29] were considered to support the comparison of creep-fatigue initiation within ASME Section III, Division 5, to that in Section RB of RCC-MRx and that in Volume 2/3 of R5. From this, a high-level summary of the approaches and the main differences is included in Table 4 below. In terms of the actual creep-fatigue approaches, it is clear that they are generally based on the same underlying methodology; to calculate a fatigue damage and a creep damage term. The approaches to calculate the strain range and creep strain are then more similar within R5 and RCC-MRx than ASME, although some differences do remain (see Table 4).
- Environmental effects are not specifically included in ASME or RCC-MRx as these are considered outside of the code, although warnings are included that they should be considered. R5 includes some general guidance on how the environment may impact an assessment. More specific guidance is planned on carburisation effects.
- Neither ASME nor RCC-MRx include methods to consider re-heat cracking (as the influence of weld residual stresses in their design or creep-fatigue analysis are neglected). Although not included in R5 specifically, there are proprietary guidance documents that provide a means to adapt the rules in R5 for assessing reheat cracking.
- Defect tolerance assessments are not generally considered in the design codes, although a check on the prevention of fact fracture is included. R6 would be considered for a defect tolerance assessment in the UK, where this is typically applied during GDA. For high temperature AMRs the approaches in R5 (for crack growth mechanisms) and R6 (for limiting defect size) would both be needed for a defect tolerance assessment.

Table 3 - Summary of main similarities and differences to plastic collapse and stress limits in ASME, RCC-MRx and R5 (taken from [3])

	ASME (III, Div. 5)	RCC-MRx (RB)	R5 (Volume 2/3)
Plastic collapse (primary loads)	$P_m \leq S_o$ $P_L + P_b \leq 1.5S_o$ S_o values are tabulated but generally follow the $\frac{2}{3}\sigma_y$ and $\frac{1}{3}\sigma_{UTS}$ rules.	$P_m \leq S_m$ $P_L + P_b \leq 1.5S_m$ S_m is adjusted to minimum of $\frac{2}{3}\sigma_y$ or $0.9\sigma_y$ (for Stainless Steels at temperature) and $\frac{1}{3}\sigma_{UTS}$ or $\frac{1}{2.7}\sigma_{UTS}$ depending on material.	$P_m \leq \frac{2}{3}S_y^{\prime}$ $P_L + P_b \leq S_y^{\prime}$ $S_y^{\prime} = \sigma_y$ for $n \geq 2$, else $S_y^{\prime} = 3n\sigma_y/2(n+1)$.
Design limits (secondary stresses)	$P_m \leq S_{mt}$ $P_L + P_b \leq KS_m$ $P_L + P_b/K_t \leq S_t$ $(P_L + P_b/K_t) + Q_R \leq S_a$ $(P_L + P_b/K_t) + Q_R \leq \sigma_{y\_ave}$ $\sum_i \frac{t_i}{rt_{id}} \leq 0.1$, & $\sum_i \varepsilon_i \leq 0.2\%$ For "Tests" 1, 2 and 3. Equivalent stresses based on Tresca criteria.	$P_m + Q_m \leq S_{em}$ $P_L + P_b + Q + F \leq S_{et}$ $\Delta(P_L + P_b)_{max} + \Delta Q \leq 3S_m$ Stress range based on von-Mises combination.	$\Delta(P_L + P_b + Q) \leq 2S_y^{\prime}$ $\Delta(P_L + P_b + Q) \leq 2.7S_y^{\prime}$ For ferritic and stainless steels, respectively. Stress range based on von-Mises combination.
Shakedown	Assumed to be covered in design limits for most cases.	Assumed to be covered in design limits for most cases.	A simple shakedown test is available to show $\bar{\sigma}_{el,lin}(x,t) \leq K_s\sigma_y$ Some refinement and advanced methods also available.
Creep rupture endurance	Based on time fraction approach. Limiting time evaluated for a factor of the stress $P_L + P_b/K_t$ at a given temperature.	Based on time fraction approach. Limiting time evaluated for a given temperature and equivalent stress.	Based on time fraction approach. Limiting time evaluated for a given temperature and reference stress.

Table 4 - Summary of main similarities and differences of the assessment methods related to creep-fatigue initiation of defects (taken from [3])

	ASME (III, Div. 5)	RCC-MRx (RB)	R5 (Volume 2/3)
Elastic stress/strain inputs	Linearised strain distribution	Stress range at the assessment location (not linearised).	Stress range at the assessment location (can be linearised).
Elastic stress/strain range	von-Mises equivalent range	von-Mises equivalent range	von-Mises equivalent range
Elastic-plastic strain	Three options available with different levels of conservatism.	Neuber construction (from stress-strain hysteresis loop construction)	Neuber construction (from stress-strain hysteresis loop construction)
Creep dwell position	At peak stress determined from shakedown figures.	At peak stress (recent changes to allow intermediate dwells).	Any position (intermediate dwells allowed).
Creep strain	Isochronous stress strain curves – assumes primary stress	Relaxation equations via integration of creep law	Relaxation equations via integration of creep law
Creep damage	Time fraction	Time fraction	Ductility exhaustion, no damage from compressive dwells.
Fatigue	Fatigue curve	Fatigue curve	Fatigue curve, modified for size effects.
Weld Fatigue Properties	Parent curves modified by a factor of 2 (see WSEF).	Either directly measured or by multiplying the parent curve by a factor.	Modified for the potential presence of micro-defects (Weld Endurance Reduction).
Weld Creep Properties	Parent curves modified by material weld strength reduction parameters.	Multiplication terms to rupture life provided.	Based on material specific ductility.
Weld Strain Enhancement Factors (WSEF) or Fatigue Strength Reduction Factor (FSRF)	WSEF of 2 applied to the strain range. Some additional enhancements with FSRF = 4 for specific welds (a partial penetration weld for instance).	Factor of 1.25 for stainless steels to account for the weld material. Enhancements with FSRF = 1, 2 or 4 depending on the weld and examination applied.	Weld type specific WSEF (for Type I, II and III welds) of 1.16, 1.23 and 1.66 for stainless steels.
Multiaxial Effects on Creep.	No adjustment to creep.	"Cr" factor included to account for multiaxial effects but no clear guidance on how to calculate.	Approaches to reduce ductility included.
Residual Stresses	No specific need to PWHT for all materials. No accounting for residual stress in-service.	No specific need to PWHT for all materials. No accounting for residual stress in-service.	PWHT not recommended in R5. Need to account for a residual stress included.
Crack-Like Defects	Only to protect against cleavage fracture (elastic). No assessment of creep-crack growth.	Able to assess cleavage and ductile cracks. No specific assessment of creep-crack growth but some creep-crack growth laws included.	R6 considers cleavage and ductile crack assessments. R5 Vol. 4/5 includes assessment of creep cracks.
Materials Data	Only from the data included within the code.	Only from the data included within the code.	No data in procedure. Able to use external data.

3.5 Recommendations from EASICS Project

A detailed review of the various topics reviewed from the code comparison, when also compared to the GDA requirements and discussion with industrial experts, was performed to provide a relatively extensive list of recommendations. The recommendations are detailed below. More context and background can be found in Reference [3].

The recommendations are as follows:

- Recommendation 1 - The GDA process is adopted for AMRs in the UK and that to take full benefit from the approach, that the Requesting Party (RP) are flexible in their design approach. Early engagement with the ONR prior to undertaking the GDA may be beneficial for novel approaches being considered, so they can be applied within the GDA process with confidence.
- Recommendation 2 - An active approach to selection of AMR design be pursued to minimise the range of reactor types being proposed for the UK.
- Recommendation 3 - RPs appreciate the importance of producing an "adaptation document" that sets out the use of codes and standards (and materials data) they intend to use.
- Recommendation 4 - R5 and R6 are considered as a viable alternative for elements of the design codes in cases where the latter are considered overly conservative, incompatible with reactor type being considered, or contain gaps for the UK regulatory expectations.
- Recommendation 5 - R5 and R6 procedures are maintained and accessible to support AMR deployment.
- Recommendation 6 - Skills in the relevant high temperature codes and standards be protected and/or developed to support AMR deployment.
- Recommendation 7 - Perform R&D to address specific issues relevant to AMR plant highlighted in the report and for RPs to identify known shortfalls as part of GDA.
- Recommendation 8 - Consider PWHT to all welds where possible to minimise the impact of residual stress on cracking mechanisms driven by a residual stress (such as reheat cracking).
- Recommendation 9 - The design be constructed to reduce the number of highest reliability locations and minimise integrity claims on welds.
- Recommendation 10 - The safety significance and commercial significance of non-highest reliability items be explicitly considered by the RP.
- Recommendation 11 - Basis of methods to consider test data interpretation and extrapolation are examined and verified.
- Recommendation 12 - The influence of environment (chemical) on the structural integrity claims that can be made are considered as a priority due to the perceived gaps in knowledge and uncertainties which exist (such as impurity levels in coolants).
- Recommendation 13 - Long-term testing is considered in a representative environment at the stresses, strains and temperatures relevant to plant loading conditions. The tests should be representative of the expected loading level and be cognisant of potential synergistic loading effects expected on plant. This may take the form of a surveillance scheme.
- Recommendation 14 - There is an associated investment and development of the test facilities and supply chain to take advantage of the need to support AMRs.
- Recommendation 15 - As far as possible, ensure that the different manufacturing techniques (including weldments) do not degrade the material properties and have associated long-term materials data to support their deployment.
- Recommendation 16 - For High Temperature (HT) AMR plant, the magnitude and number of transients be reduced where possible (consider value of load-following vs energy capture etc.).
- Recommendation 17 - For HT AMR plant, the design based transient set is as accurate as possible or appropriately conservative prior to performing an assessment.

- Recommendation 18 - The plant include a monitoring and categorisation system to record the transient, loading and operational history to ensure design assumptions are appropriate and to update through-life damage assessments.
- Recommendation 19 - A pragmatic approach would be for a RP to design an AMR according to the "design lifetime" (say 60yrs), using the best available information available. However, the initial "safety case lifetime" at GDA will be limited by the availability of materials data (say 20 years), with the supporting work in place (e.g. surveillance schemes and test programmes) to substantiate the design lifetime through plant lifetime extension (PLEX) as new data becomes available.
- Recommendation 20 - A corresponding NDE guidance document for GDA be developed.
- Recommendation 21 - Ensure the design should consider access, inspection needs, plant monitoring and even exchangeability.
- Recommendation 22 - Probabilistic approaches be developed and applied more regularly to help optimise a design and provide more robust justification to the lifetime assessment. Here the Probabilistic Guidance Document can be applied, where the Level 2 and 3 approaches outlined are considered mature enough for application to AMRs.
- Recommendation 23 - The equivalent result to a deterministic calculation from the probabilistic assessment be included to allow comparison to the historic GDA approaches. This will then allow the assessment margin to be demonstrated and benefits of the approach to be shown.
- Recommendation 24 - TAGSI to provide updated guidance on Structural Integrity (SI) demonstration of Structures, Systems and Components in the context of a new design with a highest reliability claim in mind, taking into account ONR's Safety Assessment Principles (SAPs) and past precedent in the UK.
- Recommendation 25 - A UK High Temperature Code Application Community be established to allow developments to be shared and provide links to codes and procedures.
- Recommendation 26 - UK participation in international research programmes to be enhanced to provide feedback from UK experience, provide feedback from the UK approach and to help ensure forewarning of changes.

3.6 Summary of EASICS Main Findings

Some of the key findings from the EASICS project are repeated [3] below:

- The GDA process has never been undertaken for a HT reactor and therefore there is no benchmark. GDA is an opportunity to ensure the RP approach to design substantiation is consistent with the expectations of the UK regulators. The UK operates in non-prescriptive regulatory regime and the UK regulators have extensive experience of regulating HT reactors. Therefore, undertaking the GDA process with an AMR is expected to significantly benefit the RP with regards to reducing the risks to achieving licencing in the UK and may result in a more robust safety case and/or design (if the RP is flexible in their design approach).
- At present no single code or standard is adequate to undertake the structural integrity substantiation for AMRs. It is expected that each RP present an "adaptation document" which sets out an RPs approach on the use of codes, standards and materials data for demonstrating the structural integrity of the design in accordance with UK regulatory expectations. It is expected that each "adaptation document" will be unique to the reactor technology and design being proposed. A unified approach for AMRs is not considered practicable.
- The international structural integrity design codes that are available (ASME III Division 5 and AFCEN RCC-MRx) serve an essential purpose, which is not replicated by any existing UK code or standard. However in some areas the codes may be considered overly conservative, incompatible with the reactor type being considered, or contain gaps for the UK regulatory expectations. In these circumstances, the UK assessment procedures, R5 and R6 are considered a viable alternative for elements of the design codes.
- There are still notable technical structural integrity challenges for AMR technology that are not well addressed by the existing codes and standards, which need attention. Operational experience from AGRs and Gen IV plant emphasise unique challenges of HT operation and the commercial/lifetime costs of not addressing such risks. Areas of note include:

- the synergistic impact of environment, creep and fatigue on both crack initiation and crack growth,
- the impact of thermal ageing on material properties,
- the treatment of weldments (especially welding residual stresses),
- the treatment of loading due to thermal expansion (resulting in significant secondary stresses and mechanisms such as ratchetting, high cycle thermal fatigue and significant creep-fatigue interaction).

▪ The GDA process focusses on the safety case for a nuclear power plant. Attention is therefore focussed on the highest reliability systems, structures and components, the number of which should be minimised through good design. The safety case should also consider the holistic safety of the plant, considering non-highest reliability systems, structures and components. However, a safety case does not consider the commercial risks. In short, a safety case demonstrates a reactor is safe to operate because it can shut down safely under all postulated scenarios; it does not consider whether the reactor can restart.

▪ Long-term materials data availability and assessment methodology challenges may limit safety case lifetime claims that can be demonstrated at the start-of-life (below the intended design lifetime). The intended design lifetime could then be achieved through a planned plant lifetime extension programme, supported by a well-designed lifetime materials testing programme and surveillance scheme. Ensuring the plant design considers access, inspection and even exchangeability of potentially life limiting components would significant reduce long-term commercial risk.

▪ Novel materials, manufacturing techniques and welding techniques, require suitable long-term materials data and verified assessment methods to substantiate any design lifetime. Considering the conclusion above, the inclusion of novel solutions may introduce significant challenges in substantiating the initial safety case lifetime as well as increase the risk to achieving the design lifetime (note this could be mitigated through access and exchangeability consideration).

▪ Probabilistic approaches are now well established and are regularly used in operating nuclear plant in the UK. The EASICS probabilistic work package has proposed a hierarchy of methods that may be adopted to support and optimise design, inspections and lifetime management. Such methods are expected to be of particular value to AMRs due to the inherent variability and systematic uncertainties that are expected as a result of operating novel high temperature plant.

▪ Whilst a unified solution to structural integrity codes and standards is not viable, open collaboration will be important to share best practice within the UK and internationally. Due to the operating experience of AGRs and associated skills and tools developed in the UK (including within the UK regulators) and due to the non-prescriptive regulatory regime in the UK, the UK is in a unique position to successfully develop, support and deploy AMR technology.

4. ALLEGRO Specific Considerations

4.1 ALLEGRO GFR Design Features

ALLEGRO is one of the fast reactor technologies supported by ESNII (European Sustainable Nuclear Industrial Initiative) within the SNETP (Sustainable Nuclear Energy Technology Platform) [30]. ESNII addresses the need for demonstration of Gen IV Fast Neutron Reactor technologies, together with supporting research infrastructures, fuel facilities and R&D work.

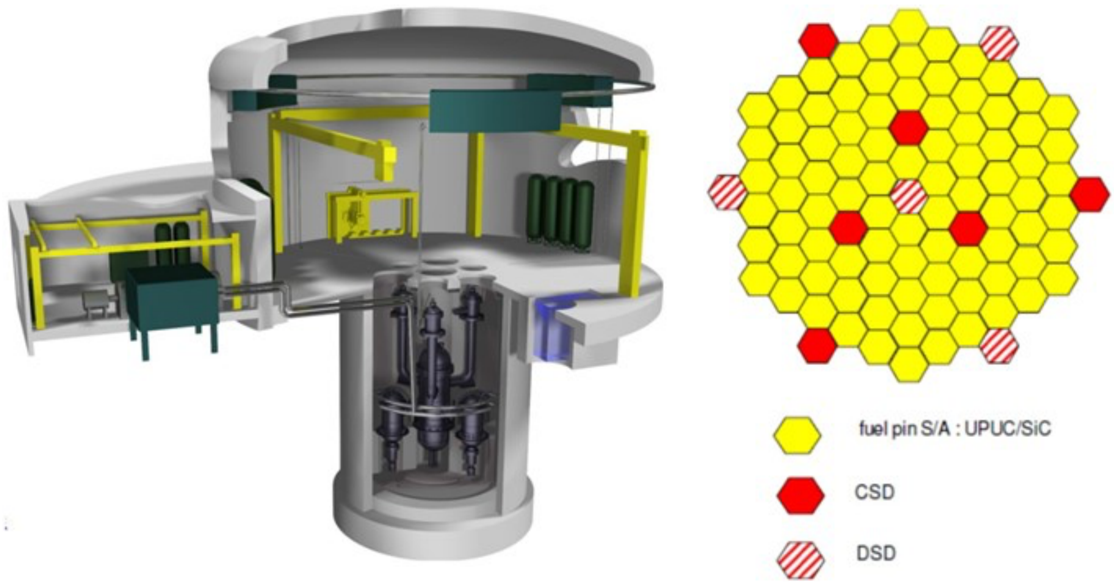
ALLEGRO is a concept of a demonstration unit of the GFR (gas-cooled fast reactor) technology, developed in Europe with the aim to prove viability, safety, and reliability of the whole concept of a high temperature, gas-cooled, fast spectrum reactor. Unique design features noted on the ALLEGRO website [31] include:

- Aimed at minimizing nuclear waste and closing the fuel cycle.
- Helium heated to up to 850°C to demonstrate high-potential heat applications.
- With the aim to reach unprecedented levels of safety.
- First prototype of the gas fast reactor (GFR) technology.

The current strategy of ALLEGRO includes finishing of qualification of a GFR-specific fuel in pile. Therefore, there are two different configurations of ALLEGRO envisaged. The so-called “driver core” is fuelled with a “standard” fast reactor fuel (oxide pin-type in stainless steel cladding) with 6 experimental positions for qualification of GFR fuel. The outlet temperature is limited due to safety concerns to 530°C. The ALLEGRO GFR reactor “refractory” configuration has a maximum fuel cladding operating temperature of 990°C [30]. After transfer to the Helium gas and heat loss, Section 2 of [30] uses an assumed inlet and outlet helium gas temperatures of 400°C and 800°C. Noting there may be some uncertainty over the temperature of ALLEGRO, to bound potential conditions, the maximum assumed Helium temperature is therefore taken to be 850°C for comparison to available code data herein. An overview of the concept of the ALLEGRO nuclear power plant and refractory core layout is depicted in Figure 1. The core layout is shown as:

- 87-off Fuel sub-assemblies (i.e. fuel pin S/A : UPUC/SIC in Figure 1),
- 6-off Control and shutdown devices (CSD in Figure 1),
- 4-off Diverse shutdown devices (DSD in Figure 1).

The core is surrounded by 4 rings of radial reflectors (i.e. 174 reflector sub-assemblies) and by 3 rings of neutron shielding (198 neutron shielding sub-assemblies).



ALLEGRO Concept Design

Core Layout

Figure 1 - ALLEGRO Concept Design and Core Layout [30]

Within the Czech national project TK03030121 KOBRA, a design of power conversion system for GFRs, with its main components, and a new safety system for gas-cooled nuclear reactors have been developed. The aim is to develop a robust safety-related system that will both prolong the rundown period of the whole turbomachinery in SBO accidents and that will improve the cooling of the reactor core, and, therefore, prevents the development of a severe accident in gas-cooled nuclear reactors. The selected solution for the power conversion system is to combine a simple Brayton cycle with nitrogen as the secondary circuit, and to utilize the still high-potential heat in a supercritical water cycle in the tertiary circuit. The resulting net efficiency of such a layout exceeds 44 %.

The safety concept of ALLEGRO is based on utilization of passive safety systems, with the aim to implement as much fully passive systems as possible. Since transients without SCRAM has proven to be the most challenging for GFRs due to combination of very low thermal inertia of the coolant and a relatively high power density, a special attention has been paid to reliability of the core shutdown device (CSD). Furthermore, very low thermal inertia of the coolant dictates the need for a constant flow of the coolant through the core in all possible scenarios. Dedicated decay heat removal (DHR) system has been developed for this purpose, based on a patented solution for a passive DHR system with increased reliability. To increase its performance in depressurized scenarios, two additional safety systems are added. The primary one being a fully-passive leak-tight primary containment vessel keeping residual pressure in the primary circuit well above the atmospheric pressure. It is supported by an emergency core coolant injection system (ECCS). Moreover, a special system for prolongation of rundown period of the main cooling loop has been under development to provide necessary diversification for the single main cooling loop.

4.2 Additional Review Topics for ALLEGRO GFR

Based on the information on ALLEGRO provided in 4.1, and information obtained as a part of this review, the potential GFR challenges has been broken down into the following topics:

- Helium coolant (Section 4.3)
  - Helium impurities
  - Corrosion Issues
  - Helium properties

    - Impact of He on metallic components
- Ceramic components (Section 4.4)
    - General Requirements for CMC Component and Assemblies
    - Technical Requirements for CMC Component and Assemblies
    - CMC Material Requirements, Specifications, and Test Standards
- Irradiation damage levels (Section 4.5)
    - Interaction between radiation damage and microstructure
    - Pressure boundary
        - Irradiation-induced shift of transition temperature
        - Irradiation embrittlement
    - Reactor internals and fuel
        - Swelling
        - Flow induced vibrations.
- Higher operating temperatures (Section 4.6)
    - Larger thermal transients
    - Thermal striping and stratification
    - Thermal shock
    - Thermal ageing embrittlement
- Material properties (Section 4.7)
- Welding and Manufacturing Processes (Section 4.8)
- Non-Destructive Testing (Section 4.9)

These items are briefly discussed over the following sub-sections.

4.3 Helium Coolant

The following section aims to capture and review the existing information available regarding the potential impact of a helium primary coolant gas on the structural integrity of materials used for high temperature GFR. Focussing upon the interaction of the He environment with the structural materials used in the reactor containment, which are predominantly metallic in nature.

Design codes are not prescriptive over the treatment and effect of helium on the design requirements. As noted below, there are effects on the material properties, material selection, corrosion and loading etc. However, these are considered as through-life properties and environmental effects where the codes suggest these should be accounted for, but provide no insight to the testing or design requirements thereof.

4.3.1 Helium Impurities

The corrosion in gas-cooled reactors report [32] focuses on degradation issues unique to HTGRs. Corrosion is a concern for both the ceramic core and metallic components. Helium is an inert gas but impurities (small amounts of CO_2, CO, H_2O, H_2, CH_4, etc.) can have implications for the metallic components resulting in surface oxidation, carburization or de-carburization depending on the impurity levels and temperature.

Several mitigation strategies for helium impurities [32] have been suggested to limit environmental effects and thereby extend component lifetimes:

- **Controlled impurities** to achieve the required more favourable gas compositions to avoid rapid carburisation or decarburisation to minimise detrimental mechanisms.
- **Material selection** balancing between mechanical properties, environmental resistance, and alloy cost.
- **Oxidation-resistant surface coatings** to mitigate the environmental effects.

It is difficult to determine which is the most effective mitigation strategy because of the limited information available. Reducing the impurity effects may minimize detrimental effects but may not be practical for operation. Stronger alloys may be more expensive, more difficult to fabricate, reduced material ductility, and made not be code compliant. Coatings can have attractive performance, particularly in laboratory studies, but may be difficult to fabricate and reliably deploy on a commercial scale. However, experience from the S-ALLEGRO integral facility show very promising results in using protective coatings in its primary (helium) high-temperature circuit. Some limited guidance on surface coatings is included in RCC-MRx but this is not specific to a helium atmosphere.

4.3.2 Helium Properties

Although the thermo-physical properties of helium environments are not directly applicable to the interaction of this atmosphere with the structural materials (and hence not included in the codes), it is worth noting that one outcome of this review of the historic helium environment literature has been to identify a significant database of helium thermo-physical properties. These properties include molecular weight, gas constant, specific heat, compressibility factor, density, viscosity, thermal conductivity, and Prandtl number [33]. This data is generally available either in tabular form or as equations. Such data is clearly key to support modelling in several areas of development, including thermal hydraulics and heat transfer, since, for the relevant operation and transient conditions, they tend to differ from the ideal gas law values in orders of several %.

4.3.3 Effect of He on Metallic Components

A range of structural metals have been exposed to high temperature helium environments within historic tests (1960-1980's), including mild steels, low alloy steels, austenitic stainless steels, iron-nickel alloys and nickel-based alloys. Regardless of the material, it was suggested that the quality of the sample surface, moisture content, duration, temperature and pressure of the test all had an influence on the extent of corrosion or oxidation occurring. In general, it was highlighted that increasing the temperature, duration and pressure of the test resulted in an increased extent of both surface and sub-surface oxidation. In addition, the sample geometry as well as surface finish appeared to have a significant influence on the extent of oxidation occurring.

Depending on the activities of oxygen and carbon in the gas, regimes have been established for optimal compatibility with structural alloys. The type of reaction also can impact alloy mechanical properties. Carburisation [32] can embrittle structural alloys while decarburisation or selective oxidation of Cr, Al or Ti can dissolve strengthening phases, thereby impacting creep resistance. The normal sources of contaminants are those, which are desorbed from reactor components, residual air and air in-leakage, fission products that migrate from the fuel, moisture from steam generator leakage and contaminants from new helium supply. Helium purification system must be designed to reduce the quantity of chemical impurities in the primary coolant helium and to remove the gaseous radionuclide fission products.

Influence of helium environment on strain vs time curves for 316 stainless steel tests can be seen in Figure 2. The improvement in creep performance with the larger grain size is apparent, when compared with the smaller (60 μm) grained material. These tests appeared to show an initially higher creep deformation rate in the helium environment, but this effect was limited in the large grain size material and extended test times (i.e. >4,000 hrs). Similar to other studies, metallography again suggested that the higher creep deformation rates in impure helium were associated with oxidation and grain boundary attack, which was not observed in the tests performed in air [34].

The fatigue test results within air and helium environments can be seen in Figure 3[a] and Figure 3[b], for strain and stress control respectively. Only limited tests in helium had been completed at the time of the progress report [35] but the available data suggests limited effects of the helium environment and possibly an enhancement in fatigue life by helium. This may be expected from the relatively protective nature of the helium environment in the short term, compared with the oxidation damage that would be expected at 750°C in air for 316 stainless steel. Thus, even the longest strain controlled fatigue test at a test frequency of 0.0167 Hz was <38 hrs, whilst for the stress controlled tests the longest time under test was ~406 hrs. Although, from a mechanical testing viewpoint, 406 hrs appears to be a relatively long test duration, from an oxidation and carburisation perspective this remains a relatively short time period. This is especially true as it is known that the AGRs did not start to display carburisation issues until times approaching 10 years (i.e. 87,600 hours) in some cases, although in others it was nearer to 1 year. Subsequently, some care is required in interpreting this fatigue data (as with the creep tests) because at the test times evaluated the atmosphere was probably unable to interact with any growing fatigue crack and may actually appear protective compared with the air environment.

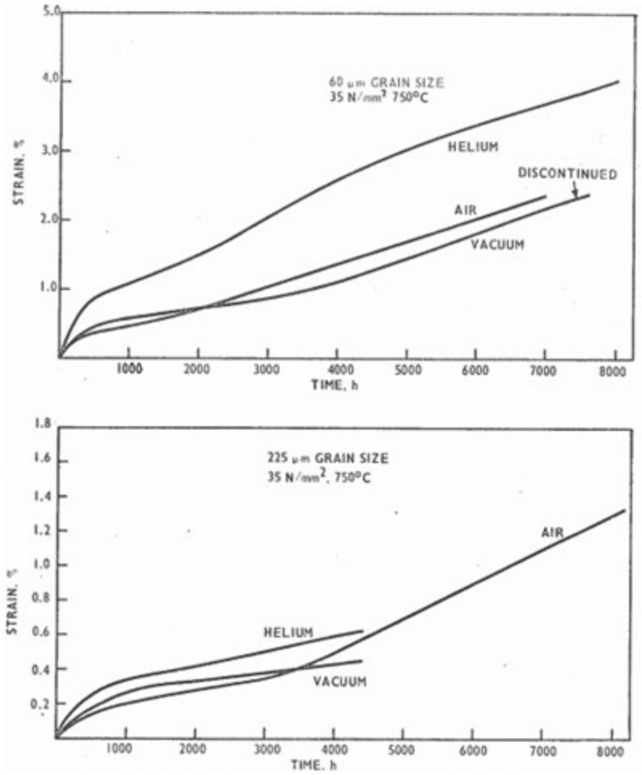
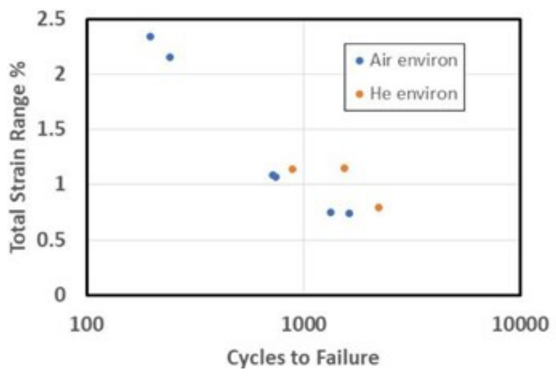
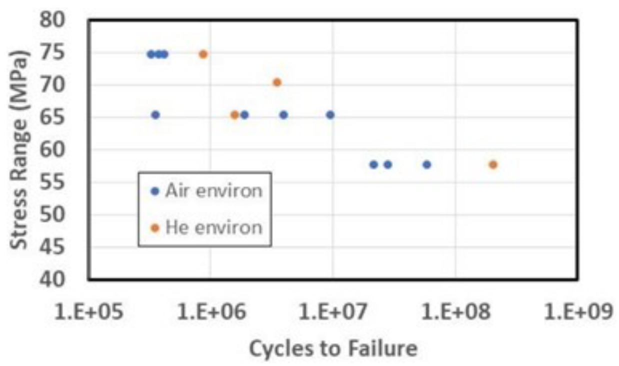


Figure 2 - Creep strain vs time curves for 316 stainless steel tested in various environments at 750°C [34]



[a] Strain Control Fatigue Endurance



[b] Stress Control Fatigue Endurance

Figure 3 - Strain and load controlled fatigue endurance of Type 316 stainless steel at 750°C in both air and helium atmospheres [35]

Despite the caveats regarding the interpretation of the creep rupture and fatigue data for austenitic stainless steels, it is clear that there can be an interaction between the helium environment and the subsequent mechanical behaviour of the austenitic stainless steel. This interaction does not have a clear-cut impact on the austenitic stainless steels because at high stress and large levels of strain the gaseous impurities appear to be detrimental in creep, whereas at low stresses and strains there is little if any negative impact on the creep rupture behaviour. In fatigue, the helium environment actually appeared to be beneficial in the tests performed on 316 stainless steel at 750°C compared with tests in air. It is worth noting that many of the tests undertaken on the austenitic stainless steels were operated at relatively high temperatures compared with their normal, and accepted, operating temperatures.

It is recognised that the structural integrity implications of the helium environment interactions with structural materials may not necessarily occur in the early phases of future reactor operation but could be at 5-10 years after start of life or even later in life, as was observed with the AGRs and the carbon dioxide (CO_2) carburisation issues. Currently available test information indicates the following factors [32] will influence degradation of structural materials:

- Alloy composition,
- Impurity levels within the helium gas, can affect mechanical properties, therefore, a goal is to minimize negative effects:
  - Oxidation can decrease lifetime by depleting Cr from the alloy and affecting strengthening phases (e.g. dissolution of Cr-rich carbides),
  - Carburisation can increase creep strength but decreases ductility,
  - Decarburisation can decrease lifetime by removing carbide strengthening phases,
- Temperature, duration and pressure,
- Specimen geometry and surface finish.

Whilst a substantial amount of work has been conducted in understanding the influence of impure helium environments on the structural integrity of a range of metals, there are still significant gaps in knowledge. For example, none of the historic work conducted any micro-hardness measurements after testing. As such, it is unclear whether any surface hardening and hence embrittlement has occurred due to carburisation. This phenomenon could cause premature cracking and hence pose a risk to the structural integrity of the materials employed. Similarly, the role the helium environment on the creep and fatigue crack growth of structural materials has not been studied previously.

There have been several experimental studies on the effect of helium pressure and velocity on corrosion rates, which could also impact the environmental impact on mechanical properties. Most studies were conducted at ambient pressure, but the primary circuit of ALLEGRO is at pressurized at 70 bars [30], which could significantly affect reaction rates. In [36] the effects of high temperature exposure in air as well as in impure He on mechanical properties of 316L and P91 steels were investigated. After the exposure both in air and He, the ultimate tensile strength of P91 decreased significantly more than that of 316L. After the exposure in He, the fracture toughness of 316L was reduced to 60% while fracture toughness of P91 showed no significant changes. In [37], high temperature corrosion and degradation of alloys (800 H, SS 316 and P91) in helium containing minor impurities (H2, CO, CH4, H2 O) at temperatures up to 760°C were studied, and results between HTHL helium loop and static furnace experiments compared. The most susceptible of the tested alloys to corrosion in impure helium was Alloy 800H. Fall of hardness and microhardness after exposure was recorded in case of P91, Nicrofer weld metal and Alloy 800 H – heat-affected zone, hardness of base metal of Alloy 800H slightly increases after exposure. Hardness of SS316 was almost constant. The corrosive layer on Alloy 800 H after 260 hours exposure in HTHL loop was found to be thicker than that after exposure in HTF after 760°C/1500 hours.

Figure 4 shows a basic understanding of how impurities affect oxidation and carburisation but does not include the effects of temperature, alloy composition and other impurities, such as H_2O.

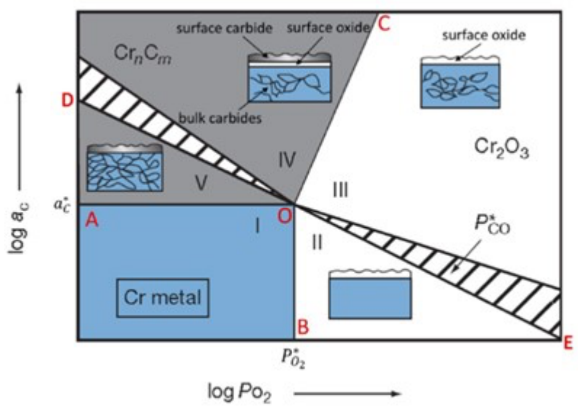


Figure 4 Cr-C-O stability diagram, including a schematic representation of the corrosion regimes expected to be observed in each condition [38]

Because helium hardly dissolves in solids, it gathers in the form of helium bubbles in the material and has a great influence on swelling, creep, and embrittlement behaviours. Helium embrittlement is attributed to the nucleation of helium bubbles on grain boundaries, their subsequent growth by helium absorption or coarsening, followed by their stress-induced transformation to unstably growing cavities. The authors are not aware of existing guidance for helium embrittlement within a nuclear context.

4.4 Ceramic Core Components

Fiber-reinforced ceramic matrix composites (CMCs) have many desirable properties for high-temperature nuclear applications, including excellent thermal and mechanical properties at elevated temperatures and reasonable to outstanding radiation resistance [39]. Over the last 20 years, the use of ceramic composite materials has already expanded in many commercial non-nuclear industries as fabrication and application technologies mature.

The ASME CMC design and construction rules were first published in the ASME BPVC Section III, Division 5 in the 2019 edition under Subsection HA, Subpart A, and Subsection HH, Subpart B. It is an attractive material because of its low thermal expansion, superior high-temperature performance, light weight, and moderate to outstanding radiation resistance. CMCs are complex material structures with variability in material properties, which requires probabilistic assessment methods for composite component design. The code addresses the construction requirements for nuclear reactor applications of silicon carbide matrix with silicon carbide fibers and carbon-based matrix with a carbon fiber composite material.

The 2023 ASME design and construction rules under Section III [8], Subsection HH, Subpart B lay out the requirements and criteria for materials, design, machining and installation, inspection, examination, testing, and the marking procedure for ceramic composite core components, which is similar to the established graphite code under Section III, Subsection HH, Subpart A. Moreover, the general requirements listed in Section III, Subsection HA, Subpart B have been expanded to include ceramic composite materials. The code rules rely heavily on the development and publication of standards for composite specification, classification, and testing of mechanical, thermal, and other properties. These test methods are developed in the American Society for Testing and Materials Committee C28 on Advanced Ceramics with a current focus on ceramic composite tubes.

Subsection HH, Subpart B is structured to allow for multiple applications and continual development because it is process based. Most developed ASTM standards align with material testing and support the requirements of the material data sheet.

4.4.1 General Requirements for CMC Component and Assemblies

The general requirements for non-metallic core components (Subsection HH subpart A in ASME III Division 5 [8]) are applicable to graphite and ceramic composites, core components, and assemblies. They provide the rules for component classification, as well as the roles and responsibilities for the owner, designer, and the material organization. They further provide the requirements for a quality assurance program and the appointment of an authorized inspector. They elaborate on how to obtain certificates and data reports from ASME and reference the applicable standards that must be applied.

The owner or owner designee is responsible for preparing the design specification, but compliance of the code remains the responsibility of the owner. The material organization is the party responsible for supplying the materials, machining the components, and installing the core components in the core assembly.

4.4.2 Technical Requirements for CMC Component and Assemblies

The design rules for CMCs (Subsection HH subpart B in ASME III Division 5 [8]) specifically focus on composites materials and lay out the requirements and criteria for materials, design, machining and installation, inspection, examination, testing, and the marking procedure for ceramic composite core components.

They are similar to the graphite rules in that they use a probability of failure (POF) approach over the component's lifetime. However, they are distinctly different in that they do not use the maximum deformation energy, or equivalent stress, theory that allows stresses to be combined as it is applied for graphite. Instead, they use the maximum failure mode for stress analysis, which differentiates between primary and secondary stresses.

The rules provide two design approach options. Designers can either follow a design-by-analysis method or a design-by-test method. For the design-by-analysis method, two key concepts are noteworthy. First, the failure mode related to the allowable stress should be applied to determine the POF. Second, the design margin is implied from the statistical analysis of the material test data, and no specific safety factor is used in the stress assessment. The design-allowable stress is based on the statistically determined margin from the proportional limit and ultimate limit strengths. The minimum value for both limits will be considered. The design-by-analysis method defines several steps:

1) Identify the potential failure modes and loading criteria (either static or time-dependent),
2) Define the component classification and acceptable POF,
3) Develop models to determine the component stress and derive the maximum mode stress from the application specific failure modes,

4) Statistically characterize the material reliability,
5) Determine the design allowable stresses based on component POF,
6) Perform the structural reliability assessment.

If the design loading or geometry is too complex to rely on analysis, it is possible to follow the design-by-test method. The design-by-test method requires multiple components and a demonstration of the requirements stipulated in the design specification.

As such, for any new material there is a need to better understand the failure mode and associated endurance data for the composite identified. There is therefore a need to perform testing to detail the potential failure modes (to allow probabilistic interpretation) and define endurance. The experimental data would also support the development of a suitable materials model would also be required to help enhance the design via iterative design-by-analysis within computational models.

4.4.3 CMC Material Requirements, Specifications, and Test Standards

Composites are a "new" material system tailored for a specific component. Composites have different design rules and failure mechanisms than metals and monolithic ceramics. The rules for materials are addressed in Article-2000 [8].

CMCs have some key issues. Both SiC/SiC and C/C composites are complex in fibers, matrix, and porosity with a wide range of constituents, different properties, and many distinctly different densification techniques. The reinforcement architectures can vary widely with marked anisotropy, giving anisotropic physical and mechanical properties. The component properties can vary widely based on the constituents, architecture, and processing. Moreover, the component requirements can vary widely, depending on the design requirements and composite material architectures.

For this reason, the material must be specified in the early stages of component development. The component geometry and component primary and secondary loads must be well understood. The two ASTM standards, C1783 [40] and C1793 [41], were adopted and applied in Subsection HH, subpart B [8]. The standards provide guidance on how to specify the constituents, structure, desired engineering properties, methods of testing, manufacturing process requirements, quality assurance requirements, and traceability for composites needed for nuclear reactor applications.

The rules also require designers to obtain the necessary composite design data and provide material data sheets that must be populated, which include as-manufactured, irradiation and oxidation, or chemical attack properties, as well as material behaviour being subjected to stress-time-temperature effects.

Since the initiation of code rule development for composite component nuclear application, it was realized that another important undertaking was required to develop standards to collect material properties. This effort was led by Committee C28, the Committee on Advanced Ceramic Standards-specifically subcommittee C28.07, which focuses on CMCs. A recent development includes the new standard, ASTM C1899, to perform flexure strength tests on tubes [42].

4.5 Irradiation Levels

Since 1942, it was recognized that high-energy neutrons [32] would have the ability to disrupt the crystal lattice of metals through which they might pass and that this disruption might lead to serious changes in the mechanical and physical properties of structural materials used in the construction of nuclear reactors. On theoretical grounds and on the basis of fundamental experiments it is seen that clusters of radiation-produced vacancies or interstitials, by interacting with dislocations already present and additional dislocations produced during plastic deformation, are primarily responsible for the increased strength and decreased ductility of irradiated metals [32].

As noted earlier, the effect of through-life and environmental effects are not necessarily included in the codes. There is therefore a need to understand the potential evolution of material properties under irradiation beyond code guidance. Where select data is included within the codes, this is likely to be for pressurised water reactor conditions and the specific fluence these reactors experience. For the GFR and the different materials likely to be considered there are therefore some aspects, noted below, that may need further consideration. It is noted that the higher operating conditions may be beneficial to some aspects and provide self-annealing of irradiation damage accumulated such that it may be possible to apply current best estimate understanding from existing reactors (if that metal has been tested/used).

4.5.1 Interaction between Radiation Damage and Microstructure

Different radiation damage mechanisms occur at different temperatures [43] roughly defined by proximity to the melting temperature, as shown in Figure 5, that demonstrates the relationships. At low temperature, embrittlement due to radiation damage or due to the build-up of embrittling transmutation gases such as helium and tritium (3^{+H}) may cause a loss of toughness at low temperature. At intermediate temperatures, radiation creep and void swelling cause dimensional instabilities that must be understood for proper reactor operation. In addition, high-temperature helium embrittlement is likely unless the helium is properly managed. Because ALLEGRO has operating temperatures above ~600°C, it will need to consider effectively strengthened alloys with higher melting temperature metals for structural components. Using these higher melting temperature alloys does not eliminate the possibility of similar radiation damage mechanisms as those found in construction materials (i.e., Fe- and Ni-base alloys) of current generation reactors. All of the higher temperature alloys need to be investigated to understand the effect of radiation damage and define performance limits in Gen IV reactor environments.

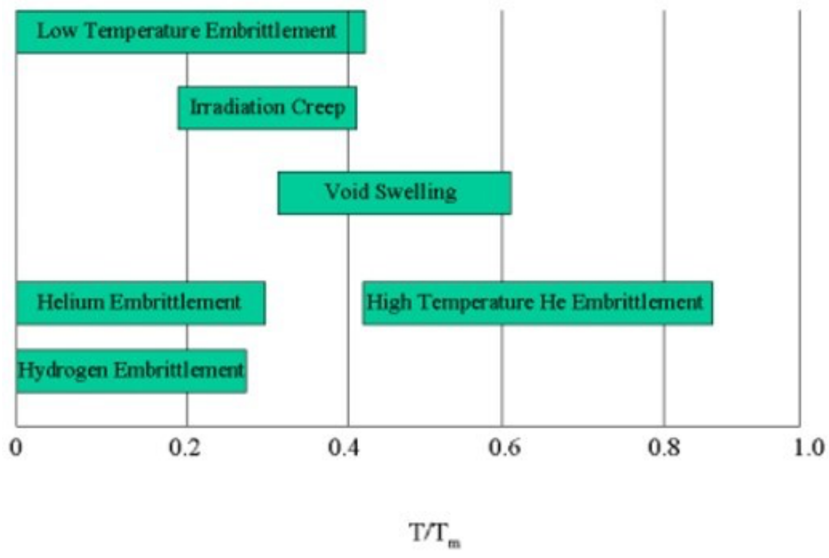


Figure 5 - Temperature Ranges over which Radiation Damage Occurs [43]

4.5.2 Pressure Boundary

Normally, a large proportion of the radiation-produced crystal lattice defects responsible for the changes in mechanical properties in steels can be removed by heating in the range 289°C to 482°C. Likewise, irradiation at temperatures above 260°C usually results in less damage than irradiation at lower temperatures, presumably because of concurrent thermal annealing of the damaged regions. Under certain conditions, radiation may accelerate phase transformation or aging phenomena, and changes in mechanical properties [32] as a result of such secondary effects may not be removed by the annealing treatments at 289°C to 482°C.

Intergranular low-ductility fractures attributed to helium effects are observed in tensile tests of austenitic stainless steels for irradiation temperatures $T_i/0.5T_m$, where T_m is the melting temperature (temperatures in Kelvin). Such elevated-temperature helium embrittlement in austenitic stainless steels can occur with as little as 1 ppm helium or less, depending on the composition, thermomechanical processing, irradiation conditions, and test conditions (temperature, strain rate, etc.). Indications are that the ferritic/martensitic steels are relatively immune to this type of embrittlement [44].

It is noted that the microstructure changes due to radiation damage affect the macroscopic, mechanical properties of the material. These effects happen for a variety of reasons, but are generally less noticeable at higher temperatures as the damage caused by radiation is constantly being annealed out: at higher temperatures vacancy and interstitial mobility are increased so they are removed from the lattice faster. Table 5 gives an overview of the effects observed.

Irradiation-induced shift of this transition temperature has been of the greatest concern for conventional RPV design. Pressure vessels for non-nuclear applications operate in the 21 to 260°C temperature range, above the steel's brittle-to-ductile transition temperature, and, therefore, the steel's ductility is maintained. However, if irradiation increases the ductile-to-brittle transition temperature into the operating temperature range, then a possibility of catastrophic brittle failure of the pressure vessel is introduced. The ASME BPVC specifies that the RPV should operate 15°C above the ductile-to-brittle transition temperature.

Temperature during irradiation is an important factor in influencing the degree of irradiation embrittlement [45]. Maximum embrittlement has been found to be caused by irradiation temperatures below 232°C. The nil-ductility temperature (RD_{NDT}) shift progressively decreases with increasing irradiation temperature because the less stable defect clusters anneal out at the higher irradiation temperature.

At the higher irradiation temperatures, only the most stable defect clusters remain so and, consequently, only minor changes in the transition temperature can be expected. ALLEGRO outlet temperature of 800°C-850°C will result in minor changes in the transition temperature for reactor internals, but the inlet temperature of 400°C is likely to result in RPV transition temperature changes.

The embrittlement due to irradiation of RPV steels results in a decrease in Charpy absorbed energy at upper shelf region (USR), in addition to the shift of ductile-brittle transition temperature.

To evaluate the integrity of RPV steels after long-term operation (assuming 60 year operation), it is necessary to predict the material toughness through the operation period. NRC Regulatory Guide 199 Rev.2 [46] provides the method to predict the decrease in upper shelf due to neutron irradiation. It is recognised that copper has the largest influence on the decrease in USR among candidate affecting chemical compositions such as copper, nickel, silicon and phosphorus [47]. Japanese experimental results indicate the decrease in upper shelf for irradiated materials depends on neutron fluence, copper and nickel contents. However, this is independent of the initial silicon and phosphorus content.

Table 5 - Known Effect of Radiation Damage on Material Properties

Material Property	Effect of Radiation Damage
Yield strength	Increases with irradiation, along with a decrease in plastic flow range.
Ultimate tensile strength	This also increases with irradiation, but less than the yield strength.
Ductile-brittle transition temperature	This marks the transition between a material exhibiting ductile behaviour at higher temperatures and brittle behaviour at lower temperatures. It increases significantly with irradiation, which can present a problem when the reactor vessel cools on shut down when internal pressure within the reactor is still high, and so fracture can occur if this is not taken into account.
Creep	Irradiation appears to cause a slight decrease in the creep rate but the rupture times are somewhat reduced by irradiation. This reduction in rupture life is attributed to irradiation induced decrease in elongation at rupture [45].
Fatigue	Irradiation increased the endurance limit, performance in the elastic range, while it decreased the low-cycle life, performance in plastic range. The improvement of the endurance limit takes place after about 10^4 to 10^5 cycles [45].
Upper-Shelf Fracture Toughness	Decrease in Charpy absorbed energy at upper shelf region [47].
Young's modulus	Small increase with irradiation.
Hardness	Increase with irradiation.
Ductility	Decrease with irradiation.
Stress-rupture strength	Decrease with irradiation.
Density	Decrease as the material swells with irradiation.
Impact strength	Decrease with irradiation.
Thermal conductivity	Decrease with irradiation since lattice disorder increases, thus increasing phonon scattering.
Electrical conductivity	Decrease with irradiation for similar reasons to thermal conductivity.

Note NASA document [45] summarises information about the effects of radiation on: zirconium alloys and steel alloys.

4.5.3 Reactor Internal & Fuel

ALLEGRO design based on ceramic core (i.e. oxide core or carbide core). The refuelling is managed at frequency of five, namely 1/5th of fuel sub-assemblies are extracted and replaced with fresh fuel sub-assemblies at each beginning of cycle. Each cycle lasts for 341 equivalent full power days. Fuel pin wrappers are manufactured from silicon carbide composite (SiC-SiCf) in Figure 6.

Cross-section of Fuel Sub-assembly Cross-section of Fuel Pin

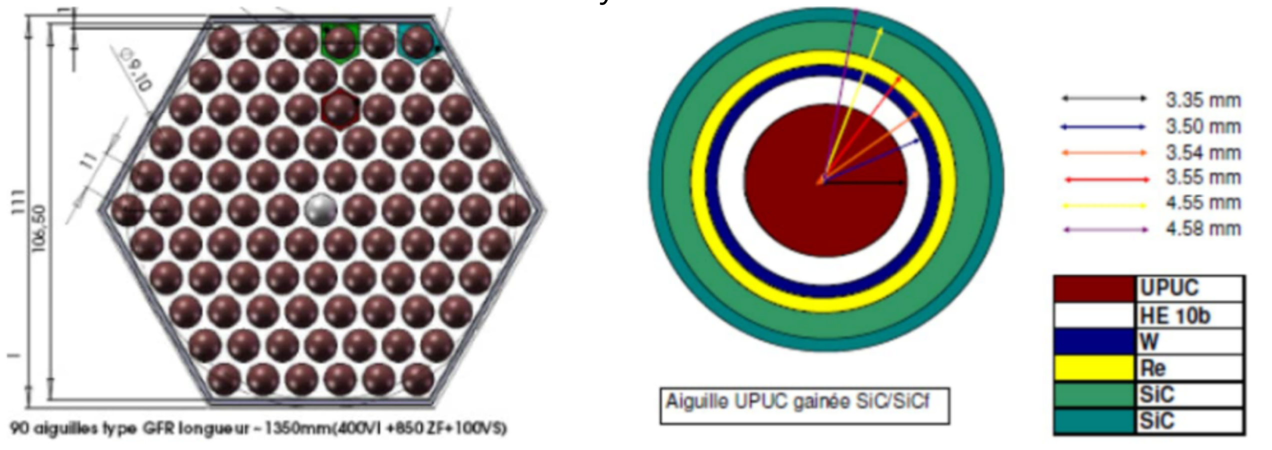


Figure 6 - Fuel Sub-assembly and Fuel Pins

General information about radiation embrittlement due to helium generation and fission product interactions have been studied in general but experimental studies under prototypical GFR conditions are needed to generate more specific and relevant information. Fission product interactions also need to be considered but the values are very low so their impacts might only be realized after long exposures.

These aspects are generally outside the scope of the design codes, but some aspects are noted below.

4.5.3.1 Swelling

The effects of damage caused by neutron irradiation include swelling (volume increase), irradiation hardening, and irradiation embrittlement (the influence of irradiation hardening on fracture toughness) [48]. These effects are primarily associated with high-energy (greater than 0.1 MeV) neutrons. High-energy neutron irradiation in a fast reactor displaces atoms from their normal matrix positions to form vacancies and interstitials; it is the “displacement damage”, measured as displacements per atom (dpa).

Consequently, irradiation damage from neutrons is of considerable importance in fast reactors, which produce a significant flux of high-energy neutrons during operation. Irradiation embrittlement must also be considered in the development of ferritic steels for fast reactors and fusion reactors. Although ferritic steels are more resistant to swelling than austenitic steels, irradiation may have a more critical effect on the mechanical properties of ferritic steels [48].

The general progressive change in microstructure with irradiation dose and temperature involves the agglomeration of vacancies and interstitials into voids and dislocation loops that cause swelling [44]. Loops form below 400–450°C; loop size increases and loop number density decreases with increasing temperature, eventually becoming unstable. In ferritic/martensitic steels, agglomeration of vacancies can lead to void swelling up to about 500°C [44]; beyond this temperature creep will help counter void swelling. Ferritic steels first became of interest for the fast reactor program because they are low swelling compared to conventional austenitic stainless steels (e.g., type 304 or 316 stainless steels) when irradiated in the Experimental Breeder Reactor EBR-II (Figure 7).

Swelling is defined as $\Delta V/V_0$, where ΔV and V_0 are volume change and original volume, respectively. At the maximum swelling temperature of around 400 - 420°C, less than 2% swelling was observed for HT-9 martensitic steel and modified 9Cr-1Mo (T91) irradiated to 200 dpa in the Fast Flux Test Facility (FFTF) [44]. Irradiation-induced precipitate changes can also affect properties. Precipitates formed in the 9 - 12% Cr steels during irradiation include α', G-phase, M6C, and chi-phase. For most of the 9 - 12% Cr Cr-Mo steels investigated, Laves phase, which forms during thermal aging at ~400 to 600°C, can cause embrittlement; it does not form if irradiation is above ~600°C [48].

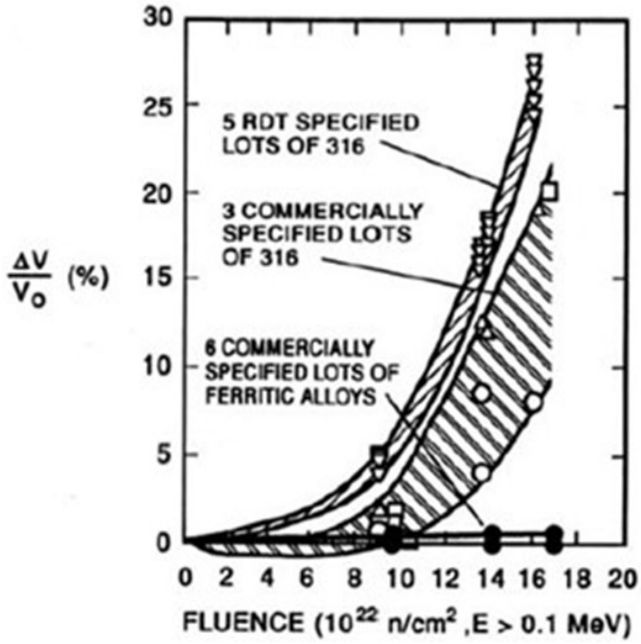


Figure 7 - Swelling behaviour of six commercial heats of ferritic/martensitic steels compared to type 316 stainless steel after irradiation in EBR-II at 420°C to ~80 dpa (from D. S. Gelles, unpublished research) [44].

4.5.3.2 Flow-Induced Vibration

In nuclear reactors, flow-induced vibrations are nearly always the result of the reactor coolant impinging on flexible reactor internals, fuel rods, shielding, or heat exchanger tubes. The resultant vibrations are undesirable and often unanticipated. They are frequently the unfortunate by-product of the tendency of structures to become more flexible and flows to increase in velocity as reactors are scaled up.

4.5.3.3 Fuel Sub-Assembly and Reactor Core Distortion

Because of irradiation the fuel sub-assemblies are likely to distort and become bowed, this was first observed on UK PFR fuel sub-assemblies in the 1980's [49]. Distortion of the fuel sub-assemblies is likely to increase mechanical loads required to operate both the 'Control and Shutdown Devices (CDS)' and 'Diverse Shutdown Device (DSD)'. In addition, the distortion of the fuel sub-assembly is likely to increase the mechanical load required to remove spent fuel sub-assemblies from the reactor core.

Any radial distortion of the reactor core is likely to be generate radial loading on the reactor internal components (i.e. shear stresses) and core restraint system (i.e. tensile loading). Mechanical and radiological analysis of the reactor core distortions, reactor internal component stresses and core restraint system loading to support the reactor design process and the operational safety case may be required. It is not clear how relevant this is for a ceramic core, meaning mechanical testing of ceramic components and a core restraint system might be required to confirm if this is of concern and to help aid predictions.

Some analytical and/or experimental work will be required to simulate the influence of fuel sub-assemblies distortion on the insertion CDS/DSD loads and the refuelling loads to ensure safe operation can be maintained.

International design code and standards (e.g. ASME III Division 5) currently only mention distortion of welds. Distortion of the reactor core or fuel sub-assembly is not currently mentioned, although it can play a non-neglectible role in fuel mechanical design, especially for high-temperature fast reactors.. Moreover, OPEX suggests core distortion may be of greater relevance for fast reactors but the potential impact on ceramics within GFRs is less clear.

4.6 Loading at Higher Operating Temperatures

In general, larger thermal transient loads in GFRs play a large role in structural performance and possible cracking, as pressures are relatively low (i.e. 70 bar, which is significantly lower than in a typical PWR). Potential thermo-hydraulic concerns, including the following:

1) Fluctuation and thermal striping risks in the mixing zones with flows at different temperatures,
2) Complex flow zones and interactions with flow patterns between subassemblies,
3) Thermal stratification of helium and its consequences on the structures, including the inner vessel,
4) Cold shocks or hot shocks during transient conditions.

4.6.1 Larger Thermal Transients

Existing design codes and standards tend to promote thicker sections or higher strength materials to meet the design requirements. However, thicker sections generally result in higher thermal stresses, but using higher strength materials would likely result in:

- Reduced fracture toughness (i.e. smaller critical crack sizes),
- Reduced creep rupture strength,
- Reduced fatigue life.

As such, there may be a need to consider a balanced approach to the vessel design. This may look to make use of design by analysis approaches or an optimisation process for design.

4.6.2 Thermal Striping and Stratification

Thermal fatigue (as a result of thermal striping) was responsible for excessive cracking in components such as pump inlets, T-junctions, and control rod guide tubes in reactors such as Superphénix, the UK Prototype Fast Reactor, and the Russian BN 600 (i.e. sodium-cooled fast reactors) [50]. Thermal striping phenomenon is a random temperature fluctuation that may cause thermal fatigue damage of components until their break. These situations can cause the failure of important mechanical systems. For nuclear industry, it is a serious safety concern; however, the severity of the consequences depends on technology.

The pre-existence of a macro-crack upon structures, which are already suffering from thermal striping, at high temperature, can complicate further the situation. The initiation and propagation of crack can lead to the failure, before the thermal fatigue does. The component damage is then accelerated until its rapid breakdown. Therefore, the thermal membrane and bending stresses induced by thermal expansion, particularly in areas of constraint, must be carefully considered. These thermal stresses have often been the source of structural integrity issues in fast breeder reactor operation.

A number of nuclear power plants worldwide have shown that Thermal Stratification, Cycling, and Striping (TASCS) in piping can cause excessive thermal stress and fatigue on the piping materials, potentially leading to fatigue cracking. These phenomena are diverse and complicated because of the wide variety of geometry and thermal-hydraulic conditions encountered in reactor piping systems.

It may be necessary to optimise the design of junctions and joints to prevent or minimise such mixing locations as far as possible. It is not clear if existing designs in the codes sufficiently address this.

4.6.3 Thermal Shock

During an emergency shutdown, the intermediate heat exchanger may experience thermal shock caused by the influx of cold helium. For a reactor operating at higher temperatures such as ALLEGRO, this may lead to excessive stresses and strains well in excess of the materials yield stress. This could then lead to rupture, buckling and other structural issues. Future GFR projects might investigate thermal shock mechanisms and modelling of the transient process to predict the structural response to see if the shock levels can be substantiated within the codes and procedures used.

4.7 High Temperature Material Properties

4.7.1 Review of Existing Materials Data

A review of the allowable primary membrane stress intensity limit (S_{mt}), the temperature and time-dependent stress intensity limit (S_t), and the rupture stress limit (S_r) in ASME III Division 5 is summarised in Table 6. All the materials in ASME III Division 5, noted in Table 6, have maximum temperature values greater than 425°C, and only the Type 316 stainless steel has limiting S_{mt}, S_t and S_r values at 800°C. No material properties are currently available above the maximum temperature. Noting the maximum temperatures of the RPV and reactor internals could be up to 850°C (depending on location within the primary circuit) there is a clear gap in terms of the available materials for the higher temperatures in the ALLEGRO GFR.

A review of the S_m, S_t and S_r in RCC-MRx is summarised in Table 7, where all the materials have maximum temperature values greater than 425°C. Unfortunately, the maximum temperature of material properties is below 800°C.

Graphical comparison of the allowable primary membrane stress intensity limit (S_{mt}), the temperature and time-dependent stress intensity limit (S_t), and the rupture stress limit (S_r) between ASME III Division 5 and RCC-MR at 425°C and 800°C is provided in Figure 8 and Figure 9, respectively.

In order to minimise the thermal transient stress, it is necessary to minimise the wall thicknesses, it is likely that the RPV will be manufactured from SA-533 Type B ferritic steel, 9Cr-1Mo_V steel or X6NiCrTiMoVB25-15-2 austenitic stainless steel (SS) because of the higher allowable stresses at 425°C, as presented by Figure 8.

The data shown indicates that it may be challenging to adopt existing materials (depending on the lifetime) within the higher temperature regions of the ALLEGRO GFR reactor without further justification.

Table 6 - Maximum temperatures for steel properties in ASME III Division 5

Material	Primary Membrane Stress Limit (S_{mt})	Temperature & Time-Dependent Stress (S_t)	Rupture Stress (S_r)
Type 304 SS	675°C in HBB-I-14.3A	800°C in HBB-I-14.4A	800°C in HBB-I-14.6A
Type 316 SS	800°C in HBB-I-14.3B	800°C in HBB-I-14.4B	800°C in HBB-I-14.6B
Ni-Fe-Cr (Alloy 800H)	750°C in HBB-I-14.3C	750°C in HBB-I-14.4C	900°C in HBB-I-14.6C
$2^1/_4$Cr-1Mo	650°C in HBB-I-14.3D	650°C in HBB-I-14.4D	650°C in HBB-I-14.6D
9Cr-1Mo-V	650°C in HBB-I-14.3E	650°C in HBB-I-14.4E	800°C in HBB-I-14.6F
SA-533 Type B	538°C in HBB-II-3000-1	538°C in HBB-II-3000-2	538°C in HBB-II-3000-4

Table 7 - Maximum temperatures for steel properties in RCC-MRx

Material	Maximum Allowable Stress (S_m)	Temperature & Time-Dependent Stress (S_t)	Rupture Stress (S_r)
X2CrNiMo17-12-2(N) SS	700°C in Table A3.1S.41	700°C in Table A3.1S.52	700°C in Table A3.1S.53
X6CrNi18-10 SS	600°C in Table A3.2S.41	700°C in Table A3.2S.52	700°C in Table A3.2S.53
X2CrNiMo17-12-2 SS	550°C in Table A3.3S.41	600°C in Table A3.3S.52	600°C in Table A3.3S.53
X2CrNi18-9 SS	550°C in Table A3.4S.41	650°C in Table A3.4S.52	650°C in Table A3.4S.53
Ni-Fe-Cr Alloy	550°C in Table A3.5S.41	575°C in Table A3.5S.52	575°C in Table A3.5S.53
X15CrNiW22-12 SS	550°C in Table A3.6S.41	700°C in Table A3.6S.52	700°C in Table A3.6S.53
X6NiCrTiMoVB25-15-2 SS	600°C in Table A3.10S.41	650°C in Table A3.10S.52	650°C in Table A3.10S.53
10CrMo9-10 Alloy Steel	550°C in Table A3.14AS.41	550°C in Table A3.14AS.52	550°C in Table A3.14AS.53
13CrMo4-5 Alloy Steel	500°C in Table A3.15AS.41	500°C in Table A3.15AS.52	500°C in Table A3.15AS.53
$2^1/_4$Cr1Mo Alloy Steel	500°C in Table A3.16AS.41	550°C in Table A3.16AS.52	550°C in Table A3.16AS.53
X10CrMoNbV9-2 Alloy Steel	600°C in Table A3.17AS.41	600°C in Table A3.17AS.52	660°C in Table A3.17AS.53

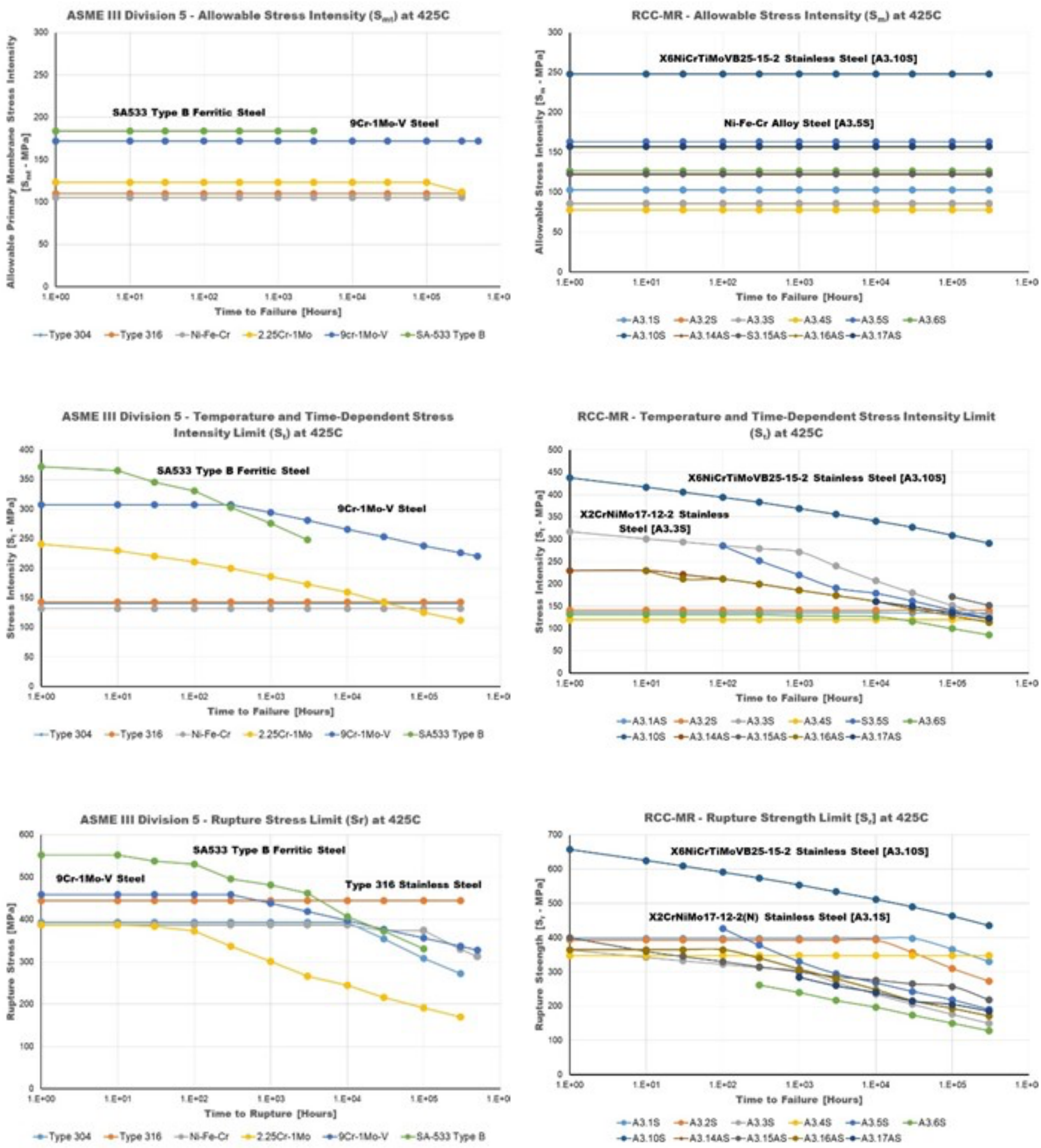
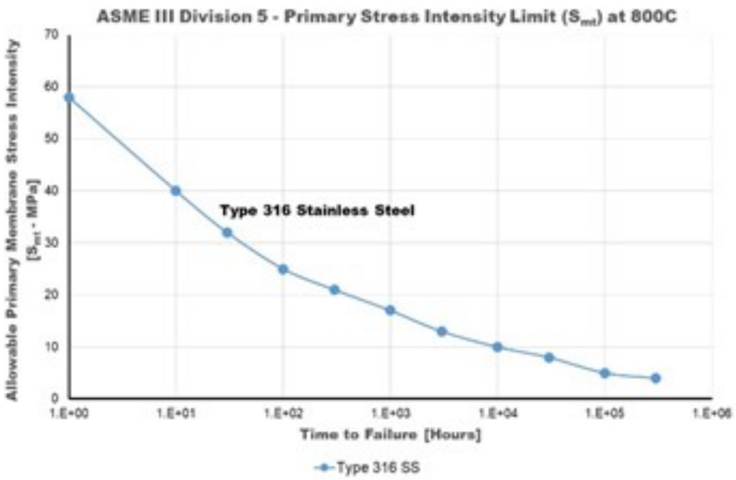
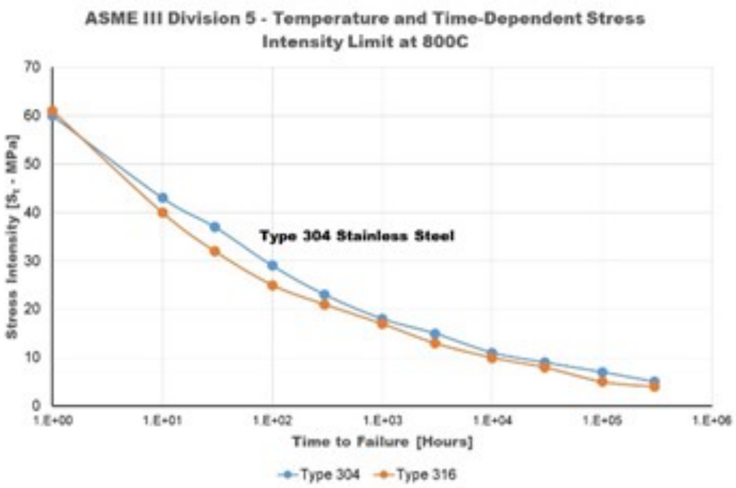


Figure 8 – ASME III Division 5 & RCC-MR Stress Intensity Limits at 425°C





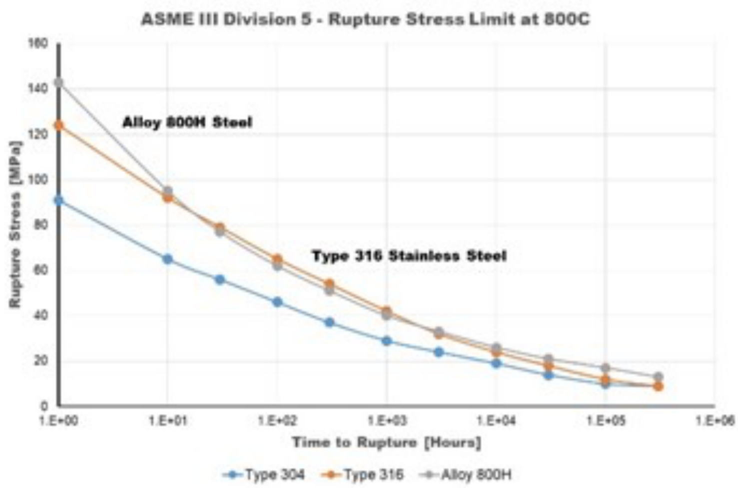


Figure 9 – ASME III Division 5 & RCC-MR Stress Intensity Limits at 800°C

4.7.2 Knowledge Gaps in Material Properties

Clearly, there are gaps in the material properties currently contained in either ASME or RCC-MRx for new high temperature reactors (as indicated within Table 8 which is extracted from an EPRI review of ARs). Here, the EPRI review suggests there is a need to develop materials resistant to swelling and understand the long-term time dependent properties of the structural material, and understand the endurance data for ceramics, for GFRs. In order to be considered for use these will need further materials testing and characterisation. As noted, for use in some regulatory processes (such as the UK GDA process), sufficient testing should also be performed to demonstrate that the materials are suitable for the temperature and conditions being considered. This will include consideration of environment where possible as well as properties required to perform defect tolerance assessments.

Table 8: EPRI Identified Materials Needs for Different ARs [51].

Reactor	Component	Material	R&D Needs
High Temperature Gas Reactor	Core Support / Structural Materials	316 and Austenitic Alloys	Time dependent properties
		316 FR	Time dependent properties
		800H	Summary of properties, code inclusion, improved weld filler
	Vessel	Low alloy steels	Time dependent and fatigue properties
	Moderator	Graphite	Long-term behaviour of specific graphite to be used
Gas Fast Reactor	Core Support	Ferritic-Martensitics	Resistance to swelling, time dependent properties
	Cladding and Reflector	Ceramics	Endurance data
Sodium Fast Reactor	Vessel and Core Support	316 Stainless Steel	Time dependent properties
		Alloy 709 SS	Summary of properties, radiation tolerance, qualification
		D9 Stainless Steel	Properties testing, resistance to swelling
	Core Support and Cladding	Ferritic-Martensitics	Resistance to swelling, fabrication technology and weld endurance
Lead Fast Reactor	Structural Material / Vessel	316	Extend creep-fatigue data, demonstrate corrosion resistance to lead
		Type 15-15T	Swelling resistance, generation of data for code inclusion
	Near-core structure and cladding	Ferritic-Martensitics	Resistance to swelling, time dependent properties, demonstrate corrosion resistance to lead, fabrication and joining methods
High Temperature Lead Reactor	Structural Material / Vessel	Alumina Forming Austenitic Stainless Steels	Resistance to irradiation/swelling, demonstrate corrosion resistance to lead, fabrication and joining methods
	Cladding	SiC-SiC	Development of structures, demonstrate resistance to lead corrosion, generate properties for code inclusion

There is also a possibility that some materials data already in the codes could be extended to consider higher temperatures and longer durations. However, it is also accepted that long-term materials data will not be available in the time required for the design process. Subsequently, it may be necessary to consider a design to the available materials data and look to consider long-term testing to substantiate any plant life extension decisions.

As highlighted by the carburisation concerns within the AGRs, there are also potential longer-term environmental effects that could impact on the safety justification for a reactor. These effects are not covered by the design codes, and to some extent are unknown at the design stage. Consequently, design considerations to allow easy sampling (and possible sacrificial components) and accelerated (if possible) testing are possibly the best route to help alleviate such concerns.

An overview of the identified testing requirements to get a new material included within ASME and RCC-MRx are described in the following sub-section.

4.7.3 New Data Testing Requirements

Guidelines for design data needs for "New Materials" is given in ASME III Division 5, Appendix HBB-Y. The intent of Appendix HBB-Y is to raise such awareness that provisions are made for an adequate data package to support successful codification. As material behaviours at elevated temperatures, particularly under cyclic service, are complex and material specific, the judgment on the adequacy of a data package is subject to deliberation by cognizant Code committees on a case-by-case basis.

The following time-independent data (Article HBB-Y-2000) are required for characterization, including physical properties:

- Monotonic tensile stress–strain curves,
- Coefficient of thermal expansion from room temperature to maximum use temperature,
- Thermal conductivity from room temperature to maximum use temperature,
- Thermal diffusivity from room temperature to maximum use temperature,
- Density,
- Poisson's ratio,
- Young's modulus from room temperature to maximum use temperature.

Data on yield strength and ultimate tensile strength (HBB-Y-2110) are needed to demonstrate that a new material is not susceptible to thermal ageing over the intended time and temperature range of applications. If the material is susceptible to thermal ageing, yield and ultimate tensile strengths data from thermally aged materials are needed to establish tensile reduction factors for ageing as functions of exposure time and exposure temperature. For design lives of 40 to 60 years, extrapolation of shorter-term data would be necessary to establish these reduction factors.

The isochronous stress-strain curves, relaxation strength at temperature and creep-fatigue data are also required when examining the time-dependent conditions.

The materials data requirements in RCC-MRx are provided in [52], which suggest the below data are required for each material (where minimum requirements are shown in *italic*):

- Physical Properties – *Coefficient of thermal expansion, specific heat capacity, thermal conductivity, thermal diffusivity, density, Young's modulus, Poisson's ratio*,
- Border lines – *Negligible creep curve, thermal ageing curve*, negligible irradiation curve, maximum irradiation curve.
- Tensile properties – Mechanical properties such as *yield stress, elongation*, necking, *stress-strain curve* (with uniform elongation).
- Fatigue Behaviour – Cyclic materials curves and associated assessment coefficients, *fatigue curves*.
- Viscoplastic Behaviour – Creep rupture stress, creep strain rate (for primary and secondary creep), fatigue-creep interaction diagram.
- Fracture – *Fracture toughness*.
- Ageing – Tensile properties, fatigue behaviour, short-term viscoplastic behaviour, fracture toughness.
- Irradiation – As for ageing but with further irradiation phenomena (swelling, irradiation creep etc).

4.8 Welding and Manufacturing Processes

ASME III Division 5 [8] fabrication and installation requirements are defined in Article HBB-4000 for Class A metallic components, HCB-4000 for Class B metallic components, HGB-4000 for core support structures, and HHB-B-3000 for ceramic composite materials. It is noted that any process may be used to form or bend pressure-retaining materials, including weld metal, provided that the post fabrication heat treatment requirements in ASME III Division, HBB-4212 subparagraphs are met. Any induced strain due to fabrication may be deleterious to material properties, which effect their subsequent service life. Thus, HBB-4212 states that a post fabrication heat treatment is required unless locally induced fabrication strains do not exceed a limit of 5%.

The code does allow for a written technical justification for not performing this heat treatment or the use of alternative heat treatment to those specified in the code. This justification should provide the assurance that the material properties are adequate for the intended service and should include the variability of properties through the ASME II [6]. However, this option is not permitted for certain materials, especially if the components are subjected to short-time high temperature excursions that result in an accumulation of temperature exposures that would negate any heat treatment. In addition, this option is not permitted for any austenitic material that is subjected to strains > 20%. The post fabrication heat treatment requirements are specified for ferritic and austenitic materials.

RCC-MRx deals with the fabrication and examination of Class N1Rx components in Section RB 4000. This contain the minimum requirements to be met and are used in conjunction with Tomes 2 (Materials), 3 (Examination Methods), 4 (Welding) and 5 (Manufacturing Operations other than welding) and Subsection A (General Provisions for Section III). Additional requirements and special provisions are presented in Section II.

Tome 5, the Manufacturing Operations guide, includes marking, cutting, forming, surface treatment, cleanliness, bolted assemblies and heat treatment. Section RB 4300 provides supplementary provisions to Tome 5.

4.8.1 Welding

ASME IX [53] includes a qualification standard for welding procedures and welding operators. ASME IX is divided into four parts; general requirements, welding, brazing and plastic fusing. Each part addressing a material joining process is then divided into Articles. These Articles deal with:

- General requirements specifically applicable to the material joining process,
- Procedure qualifications; each material joining process that has been evaluated and adopted by ASME Section IX is listed separately with the essential and nonessential variables as they apply to that particular process,
- Performance qualifications; these list the various processes with the essential variables that apply to the qualifications of each process,
- Data; these include the variables grouped into categories such as joints, base materials and filler materials, positions, pre-heat/PWHT, gas, electrical characteristics and technique,
- Standard welding procedure specifications.

Permissible welding materials are specified in ASME III Division 5, Table HBB-I-14 for Types 304 and 316 stainless steel, Alloy 800H, 2¼Cr-1Mo and 9Cr-1Mo-V steels. Guidelines for the approval of new welding and brazing materials are provided in ASME II Part C. For some environments, the welds associated with the stainless steels are likely to be suitable, whereas others may need further development (in line with the comments for the environmental aspects noted above). It is considered that the welds associated to given materials within the codes can be used up to the equivalent temperature limits of the parent material, but this is not clear.

Alongside the conventional welding technologies, like tungsten inert gas (TIG) and manual metal arc (MMA) welding, there has been growing interest in the use of power-beam welding technologies (i.e. either electron beam or laser) for rapid thick section welding of pressure vessels and other components. These potentially offer major reductions in manufacturing time, from a month to a few hours. These processes are generally autogenous and thus component fit-up tolerances are a key requirement to enable successful joints to be produced. Since these technologies permit thick section welding in a single pass, the opportunity for weld inspection during production of a conventional multi-layered weld is not possible and thus inspection technology must be applicable to the full thickness of the proposed weldment.

In addition, the very localised focus and high speed of these power beams tends to lead to high local residual stresses, and it has been found that typical ASME recommended post-weld heat treatments (PWHTs) are inappropriate for stress relieving power-beam welds in low alloy steels for example. Thus, the adoption of such techniques may require specific PWHT development, along with code case submissions to the design codes. It is worth noting that a code case submission for electron beam welding of low alloy steel has previously been submitted to ASME.

Special consideration (HBB-2160) shall be given to the influence of elements such as copper and phosphorus on the effects of irradiation on the properties of material (including welding material) in the core belt line region of the RPV to reduce the impact of irradiation embrittlement.

In satisfying the requirements of Article HBB-3000, particular considerations shall be given to the design, analysis, and construction of welded and compression contact junctions between two materials that have different physical and mechanical properties (HBB-3139.1). Such properties at elevated temperatures include thermal expansion, creep rate, creep ductility, and fatigue life. Examples of such junctions are bimetallic welds, brazed joints, compression or shrink fits, bolted flanges, and other types of mechanical joints. When temperatures cycle between low temperatures and elevated temperatures, the inelastic strains can result in significant localised strain and hence damage accumulation near an abrupt change in mechanical properties.

Operational experience suggests that weldments are key regions for issues with structural integrity. Full penetration and butt piping welds must comply with the requirements of HBB-3350 and HBB-3337. As noted in [54], the diffusion bonding process is not included in ASME, but a code case is under development.

Given the potential requirement to use novel materials in high temperature reactors, it should be noted that creep-fatigue data for the weldments (HBB-Y-3000) of a new material are needed to assess the adequacy of the assessment of welds per the Non-mandatory Appendix HBB-T procedures under creep–fatigue conditions. The weldment creep-fatigue data are to be developed from applicable filler metals and welding processes. Both deposited filler metal and cross-weld specimens will need to be tested. Weldment creep–fatigue test data are needed, with hold times sufficient to assess the reduction in cyclic life as hold time increases. ASME HBB-Y-3400 noted that, ideally, the hold time should be long enough to capture "saturation" where the decrease in cycles to failure becomes negligible as the hold time increases. However, this may not be achievable for all weldments and test temperatures.

RCC-MRx welding requirements are broken-down into: full penetration welding (RB4440); repair welding (RB 4450); and NDE of welds (RB 4460). Welding in areas that are exposed to significant irradiation during operations is to be reduced to a strict minimum (REC 3263.3).

Tome 4 in RCC-MRx covers the rules regarding welding operations and their implementation:

- RS 3000: Qualification of the welding procedure, including: electron beam (RS 3560), laser beam (RS 3570), diffusion welding (RS 3580), and friction welding (RS 3590),
- RS 4000: Qualification of welders and operators for that welding procedure,
- RS 5000: Qualification of the filler materials,
- RS 6000: Qualification of the workshop,
- RS 7000: Welding operations during the manufacture and erection of the component,
- RS 9000: Destructive tests and examinations on welds.

Technical qualification of production workshops (RS 6000) is to evaluate the capacity and technical resources of the workshop for carrying out welding operations.

4.8.2 Additive Manufacturing

These technologies are often related to the welding processes noted above, however, the term “additive manufacturing” can relate to a range of techniques that all build up a metallic structure in a step-wise fashion. The metal may be supplied in the form of either powder or wire (usually) and these can then be melted in-situ with a range of technologies, ranging from TIG to electron beam and laser. Scanning of the metal feedstock and melting system permits complex geometries to be constructed, opening the way for incorporating integral cooling passages for example. However, it must be remembered that wherever component surfaces cannot be subsequently machined or finished in some way, then the as-deposited surface finish must be tolerable for the loadings expected on the components. This is obviously particularly pertinent to components experiencing some form of fatigue loading, where work has already shown a significant debit in fatigue strength for as-deposited 316L stainless steel (laser melting of powder feedstock), compared with wrought or surface ground additively manufactured 316L material. Another potential issue with this material is the tendency for the microstructure to be strongly aligned with the thermal gradients occurring in the deposition process, similar to those seen in weldments but potentially here all encompassing throughout the component. Such aligned microstructures can lead to differences in mechanical response in the different orientations, although these effects may be second-order in comparison with typical design allowable.

A range of materials have been investigated with additive manufacturing, within the nuclear engineering context but especially materials like 316L austenitic stainless steels, zirconium alloys and nickel-base alloys, for example Inconel 718 [55]. There remains opportunities for other materials but there could be a significant amount of effort required to qualify materials produced using these techniques.

In some additively manufactured materials, small oxide inclusions have been noted and these also can affect the fatigue crack initiation behaviour of the material. The source of these oxide inclusions in uncertain but may be related to the powder feedstock and it would be helpful to compare the performance of material produced using wire feedstock, rather than powder. Until these oxide inclusions are eliminated from deposits there will remain a limit on the fatigue performance possible with some techniques within this overall additive manufacturing sphere.

Although whole components may be produced by this technology, it is also possible that additive manufacturing could be used to produce features (e.g. nozzles) on a conventionally produced component (e.g. a forging) and this could be a route to significantly reduce the amount of metal required on forgings to allow for subsequent machining of such features. This presumably will require both the forging, and any subsequent additively manufactured element of the overall component, to be assessed against the design code, where required by the safety classification of the component.

Code case development for additive manufacturing is underway but presumably it will have to identify the specific technique – e.g. powder + laser, or wire + arc and the associated feedstocks and environmental control during deposition. Thus, there could be a raft of code cases for the different techniques, unless a comparative exercise is able to show similar properties across a number of approaches. It may be that the allowances made for weldments may be capable of covering the performances measured for additively manufactured material and could provide something of a short-cut to code-case acceptance. However, it is recognised that some developers in the additive manufacturing community dislike the association with weldments but comparing with weldments could be a pragmatic approach to introducing such material.

4.8.3 Powder Metallurgy and Hot Isostatic Pressing (PM/HIP)

The use of metal powders and hot isostatic pressing (HIP) has had a considerable level of effort devoted to it over the past decade and components are beginning to be produced using this manufacturing approach. Here metal powder is placed into an appropriately shaped metal can, evacuated and then sealed. It is then placed inside the HIP vessel where it is subjected to argon gas pressures up to 300 MPa (although more commonly up to ~100 MPa) and heating to temperatures for steels and nickel-base alloys typically in the range 1100-1200°C for times of up to ~8 hours. This combination of pressure and temperature forces the powder particles into intimate contact with each other, where they undergo solid state diffusion bonding to produce a solid shape. Usually it is then necessary to remove the external metal can, which is typically made from a cheaper material, to give the final component.

The final shape produced is dependent on the original can geometry, the packing density of the metal powder and the collapse behaviour of the powder at the HIP temperature; noting that consolidation and can deformation will commence prior to the full target HIP temperature being achieved. There has been considerable developments in the field of modelling prediction for the powder consolidation and the can shape production.

Thus, it is possible to achieve an approximate net shape capability using this route, although there is always a balance to be struck between the generation of the can geometry and the closeness to net shape achievable and the reduction of subsequent machining time.

The is a requirement for the PM/HIP component to be produced, to fit into a HIP furnace, which therefore provides an upper limit on the size of components which can be manufactured from this technology and currently the maximum HIP furnace diameter available internationally is 2.0 m (in Japan), although there is a plan to potentially develop a larger furnace of 3.1 m diameter in the US and the UK are engaged in some of these discussions (as part of the EPRI Advanced Technology for Large Scale (ATLAS) project).

A potential performance advantage for PM/HIP materials can be the good chemical composition homogeneity because of the very short solidification distances in the powders, compared with those opportunities for chemical segregation in large scale conventional ingot metallurgy. This good chemical homogeneity of the materials can lead to reduced mechanical property scatter, compared with similar ingot produced material.

PM/HIP processing has been developed and demonstrated for austenitic stainless steel in nuclear applications and a code case has been presented to ASME BPVC, although this has been largely aimed at lower temperature, PWR, applications. There has been some mixed experiences with the mechanical performance of PM/HIP austenitic stainless steels, with some demonstration of 600°C fatigue and creep equivalence with wrought material [56] but also ongoing issues with reduced creep-fatigue performance [57] [58]. Thus, it appears that excellent PM/HIP austenitic stainless steel mechanical performance may be possible, although careful processing and control over the input powder stock is clearly required to ensure adequate performance.

Some work has also been undertaken on evaluating the performance of low alloy steel (e.g. SA508 Gr3) PM/HIP components and this has also started to show some promise in terms of the toughness performance, compared with wrought materials. Some larger scale SA508 Gr3 PM/HIP components have been produced in the UK, as part of an EPRI funded sub-size Nuscale reactor pressure vessel demonstrator at the Nuclear AMRC in Sheffield. However, it appears that there remains further work to both characterise and optimise the mechanical performance of low alloy steel produced in this manner.

Most of the oxide dispersion strengthened (ODS) steel materials discussed elsewhere have been generally produced using the PM/HIP route, although this has tended to be only part of a wider thermo-mechanical processing route, for example including post-HIP forging or rolling. It is not clear that the use of PM/HIP for near net shape production in ODS steels has been evaluated in any depth at this time.

There has been some interest in using PM/HIP to produce nickel-base alloy components, for example with Inconel 625 and Inconel 617 alloys. These alloys have been selected for either corrosion resistance in PWR environments or potential high temperature reactor applications. To date the mechanical evaluation of these materials has shown disappointing performance, considerably lower than found with conventional wrought materials. Thus, there remains some fundamental powder production, composition and processing to be developed for this class of alloy with PM/HIP. However, with the cost of nickel-base alloys, routes to minimise material wastage may still be attractive and make the continued development of this technology worthwhile.

4.9 Non-Destructive Testing

The link between the defect tolerance assessment and Non-Destructive Testing (NDT) is crucial in justifying the components integrity for the remainder of the components lifetime. This link between the NDT and the defect tolerance assessment means the two disciplines are inter-dependent to the over-arching NPP safety case. However, it is likely that the Fitness for Service (FFS) codes will inherently be used as a guideline in many respects: volumetric coverage, inspection interval, scanning specification, beam angles etc. FFS assessment for high temperature GFR plant could be undertaken using either: ASME XI code; RCC-MRx/RSE-M codes; and R5/R6 procedures.

Both the ASME XI and the RSE-M inspection requirements cover the same NDT methods, which are discussed further in the following sub-sections:

1) Ultrasonic examination,
2) Radiographic examination,
3) Penetrant examination,
4) Magnetic particle testing,
5) Eddy current examination,
6) Visual examinations and video examinations,
7) Acoustic emission.

Literature review of the maximum operating temperature of the nine common types of NDT methods are summarised below:

1) Ultrasonic testing (i.e. Phased array up to 300°C),
2) Eddy current testing (i.e. up to 300°C),
3) Infrared thermal testing (i.e. up to 300°C),
4) Radiography testing (i.e. X-ray & Gamma, up to 300°C),
5) Visual inspections (i.e. up to 80°C),
6) Magnetic particle inspection (i.e. up to 65°C),
7) Liquid penetrant testing (i.e. up to 65°C),
8) Acoustic emission testing (i.e. unknown).

Only four NDT methods are suitable to In-Service Inspection (ISI) of high temperature components (i.e. up to 300°C): Ultrasonic testing; Eddy current testing; Infrared thermal testing; and Radiography. As such, the existing NDT technology will be unable to perform ISI of the ALLEGRO GFR because the operating temperatures for the inlet gas, outlet gas and reactor core are 400°C, 800°C and 990°C respectively. Therefore, the ISI will have to be undertaken during shutdown and refuelling periods and during the reactor start-up procedure, when the operating temperature is less than 300°C.

A number of potential concerns and limitations for the four NDT ISI methods noted above are included as:

- Ultrasonic Testing - Conventional ultrasonic transducers will tolerate temperatures up to approximately 50 °C. At higher temperatures, they will eventually suffer permanent damage due to internal disbonding caused by thermal expansion. However, ultrasonic probes for use at elevated temperature levels up to 250°C are available, but require active cooling. Influence of temperature and irradiation on performance of both ultrasonic and electro-magnetic inspection probes. Maximum normal operating temperature of 60°C or more will require probe cooling.

  Most common ultrasonic couplants such as propylene glycol, glycerin, and ultrasonic gels will quickly vaporize if used on surfaces hotter than approximately 100°C. Thus, ultrasonic testing at high temperatures requires specially formulated couplants that will remain in a stable liquid or paste form without boiling off, burning, or releasing toxic fumes. At very high temperatures, even specialized high temperature couplants must be used quickly since they will tend to dry out or solidify and no longer transmit ultrasonic energy.

Dried couplant residue should be removed from the test surface and the transducer before the next measurement.

- Eddy Current Testing - Eddy current (EC) can only be used to measure materials that support the flow of electrical current. One parameter known to affect the EC and thickness gauge sensors is temperature. Since both sensor accuracies depend on the ambient temperature of the system, the plate checker has been characterized for these sensitivities.
- Infrared Thermal Testing - Thermal infrared imaging technology could potentially be used in pressure vessel and pipeline on-line detection in high temperature and pressure environments. Potential disadvantages of thermal imaging:
  - Thermal imaging products require high initial investment cost,
  - Images are difficult to interpret in specific objects having erratic temperatures,
  - Accurate temperature measurements are hindered by differing emissivities and reflections from surfaces.
  - Unlike visible light, infrared radiation cannot go through glass.
- Radiography Testing
  - Relatively slow inspection process.
  - Sensitive to flaw orientation.
  - Usually not possible to determine depth of indications.
  - Two-sided access to test object is required.
  - Ineffective for sizing of planar and surface defects.

In addition, the inspection data provides important information throughout the entire safety case to underpin assessments made of core and component behaviour and provides validation of the material behaviour models.

The list also considers potential challenges for in-service inspection (ISI):

- The ALLEGRO inlet/outlet gas temperatures for the GFR are 400°C and 800°C-850°C respectively. At such high temperature, the ISI will require very high temperature probes, which bring a number of additional challenges. It is noted that this problem is not so significant for Manufacturing Inspection (MI) as this will be at lower temperatures. However, the Pre-Service Inspection (PSI) that provides the inspection fingerprinting for later ISI should be at high temperatures.

- The reactor and reactor internals may not be inspectable because of limited optical access. It is likely that radiography would probably also be ineffective due to the high operating temperature. Ultrasound propagation may be sufficient, but will require a large phased array to give a meaningful image. It is also likely that the resolution will still be too poor to approach the capability for crack/damage detection.

- Infrared Thermal Testing also remains a viable option for the high-temperature environment inside the ALLEGRO reactor pressure vessel, although it carries a number of challenges as listed above, especially the correct calibrations of surface emissivities will require a substantial further R&D using GFR prototypic materials and conditions.

5. Overview of Potential Code Development Areas for GFRs

This report has provided a high level overview of the available codes and standards, and some relevant design codes, for design substantiation of a high temperature reactor. The information available on ALLEGRO GFR has been reviewed and information such as nominal operating conditions, temperatures, coolant etc. have been considered to review experience and data in the literature to identify a number of potential challenges and gaps in the design codes. This section provides a high levels overview of a number of these aspects that are considered to have greater impact (not presented in any specific order). It is noted that only a few of these address specific gaps within codes and standards; most aspects noted are additional considerations that would need to be examined beyond the existing scope of the design codes. Where suitable recommendations for how to address the identified gaps/topics are noted below in *italic*.

It is also noted that the GFR design and requirement challenges are not unique and are seen, to varying levels, for other high temperature reactors (where the high-temperature gas reactor, HTGR, is probably most similar). It is therefore prudent to review ongoing developments for these other reactors, in particular the HTGR, to see where any learning can be made or where co-development opportunities may exist.

5.1 Metallic Components

For existing and decommissioned reactors, life-limiting failure mechanisms have included embrittlement due to neutron irradiation, creep-fatigue initiation and growth, stress corrosion cracking and environmental fatigue. The basic design approaches are considered suitable for GFRs in terms of basic features such as wall thicknesses etc. However, for the GFR conditions further life-limiting failure mechanisms that are less defined within the codes (or potentially considered in a very conservative way) are likely to include:

- Ratcheting due to large temperature excursions leading to excessive distortion;
- High cycle fatigue due to thermal fluctuations (e.g. thermal striping);
- Creep rupture due to longer operating times (i.e. 40 to 60 years) and high stress constraint conditions (i.e. lower ductility) at higher temperatures;
- Thermal ageing, again due to longer operating times at higher temperatures;
- Crack growth by a combination of creep, fatigue and environmental factors;
- Buckling and/or creep buckling in thin-section components, possibly resulting from thermal loadings.

The ability to specifically assess against most of these aspects is currently outside of the ASME III Division 5 and RCC-MRx codes and standards. However, the R5 procedure specifically includes a method to protect against ratcheting and assess for creep rupture and creep-fatigue crack growth (noting the effect of different GFR environments on crack growth will still need to be examined). There are also methods to adjust the creep rupture strain for high constraint (i.e. high stress triaxiality) conditions in R5 whereas the design codes do not adjust the rupture time for stress triaxiality (see Table 4); this may be more influential for long term, low stress conditions. It is noted that there is a plan to allow RCC-MRx to refer to R5 for some methods within the next revision of RCC-MRx, as such there may soon be a more direct method to consider these aspects.

However, there are a number of potential GFR gaps in the high temperature codes, standards and procedures including:

- A subset of the EASICs recommendations [3] are considered applicable to the GFR in terms of design basis and how to best supplement the codes:
  - Recommendation 4 - R5 and R6 are considered as a viable alternative for elements of the design codes in cases where the latter are considered overly conservative, incompatible with reactor type being considered, or contain gaps for the UK regulatory expectations.
  - Recommendation 8 - Consider PWHT to all welds where possible to minimise the impact of residual stress on cracking mechanisms driven by a residual stress (such as reheat cracking).

- Recommendation 9 - The design be constructed to reduce the number of highest reliability locations and minimise integrity claims on welds.
- Recommendation 11 - That basis of methods to consider test data interpretation and extrapolation are examined and verified.

  To examine the extrapolation methods a number of approaches could be considered including artificial intelligence methods. It is, however, observed that greater micro-mechanic understanding helps inform these models meaning that research and development activities to understand creep, fatigue and irradiation mechanisms would be useful. There are currently various activities ongoing in these areas and, at least short term, there is a need to understand recent projects and developments.

- Recommendation 12 - The influence of environment (chemical) on the structural integrity claims that can be made are considered as a priority due to the perceived gaps in knowledge and uncertainties which exist (such as impurity levels in coolants).

  There is a need to perform tests in representative conditions (linked to recommendation 13 below) to try to confirm if there are potential environmental influences on the materials of choice. This should examine the potential impact of potential impurities also (and consider how to minimise these). This is likely to be longer-term than the plant design acceptance so approaches so, where possible, accelerated tests should be considered or increased use of surveillance specimens. It is noted that this is outside of code requirements.

- Recommendation 13 - Long-term testing is considered in a representative environment at the stresses, strains and temperatures relevant to plant loading conditions. The tests should be representative of the expected loading level and be cognisant of potential synergistic loading effects expected on plant. This may take the form of a surveillance scheme.
- Recommendation 14 - There is an associated investment and development of the test facilities and supply chain to take advantage of the need to support AMRs.

  The requirement for additional testing to meet through-life test conditions should promote the need to develop new facilities. Ideally, the long-term testing programme should be developed and shared such that the facilities have the confidence to develop and/or modify their testing capability to the new operating conditions.

- Recommendation 15 - As far as possible, ensure that the different manufacturing techniques (including weldments) do not degrade the material properties and have associated long-term materials data to support their deployment.

  Linked to the need for further testing and outlining the full test scope.

- Recommendation 17 - For HT AMR plant that the design based transient set is as accurate as possible or appropriately conservative prior to performing an assessment.

  Ensure the design and structural assessment are linked to the core physics modelling so that the temperature conditions and fluid flow are modelled without loss of accuracy when transferring between disciplines. This may then require an iterative approach to design as changes to either the core modelling or design requirements would impact one-another.

- Recommendation 18 - The plant include a monitoring and categorisation system to record the transient, loading and operational history to ensure design assumptions are appropriate and to update through-life damage assessments.
- Recommendation 19 - A pragmatic approach would be for a RP to design an AMR according to the "design lifetime" (say 60yrs), using the best available information available. However, the initial "safety case lifetime" at GDA will be limited by the availability of materials data (say 20 years), with the supporting work in place (e.g. surveillance schemes and test programmes) to substantiate the design lifetime through plant lifetime extension (PLEX) as new data becomes available.

▪ As noted within this review a number of further specific topics have been noted as:

  - Impact of the helium (He) environment on creep, tensile and fracture properties should be specifically examined. This may also need to include the effect on fatigue initiation curves and crack growth rates. This would require testing in representative conditions (and ideally under representative loading) to understand any impact, especially for longer durations (extending on EASICs Recommendation 13). This would be considered outside of code requirements as this relates to through-life environmental conditions beyond initial design.

- There are clearly limitations for applying existing materials and their associated properties from within the existing design codes for higher temperatures (i.e. >400°C) or longer durations (i.e. >20 years). The review identified that the maximum potential temperature within ALLEGRO is above the temperatures currently included within the codes.

  There may be a need to test materials currently included within the codes to slightly higher temperatures (say 900°C to be bounding) where appropriate. This testing should be considered to meet both the requirements of ALLEGRO and the data required by the codes (see Section 4.7.3) for consideration within the codes where possible. This would necessitate down-selection of the materials to be used as a priority. If a new material (to the codes) is being considered there is potentially a large range of tests required to provide suitable confidence. Depending on the regulatory view, there may also be a need to gain code acceptance of the material before the design is accepted (noting this is not a requirement in the UK as long as sufficient evidence is provided to the materials suitability, which is converse to the approach in America).

- For the GFR the approach to component classification should be clearly detailed.

  Effective design will depend on a robust approach to define the methodology to classify the components being assessed. As such, an outline approach should be developed and shared with the regulatory body at an early stage of design.

- Creep rupture due to longer operating times and high stress constraint conditions (i.e. lower ductility) at higher temperatures should be examined for the ALLEGRO material of choice.

  Ensure creep rupture tests are included in any test programme over different constraint levels.

- Attempts to design against high cycle thermal fatigue (e.g. thermal stripping) should be made as this has the potential to rapidly degrade a material when at very high temperatures.

  Where possible the basic approaches included within the design codes should be used to provide an initial design. Ideally, this should then be optimised through coupled computational fluid dynamics and structural modelling to minimise the levels of high cycle thermal fatigue.

- Impact of thermal ageing should be examined. It is noted that because of self-annealing, the impact of irradiation is considered to be less significant at higher operating temperatures, although this would need further confirmation.

- Impact of cyclic stress-strain behaviour on creep-fatigue may need further consideration for the material and temperatures of choice.

- Approaches to weld and join sections together should be examined with suitable justification for the through-life properties and assessment methods included. Linked to the EASICs Recommendations 8 and 9, there is also a need to understand the impact of the joining method (for instance if HIP was used) on the through-life properties. The number of transition joints should also be minimised and located where the operating conditions are less onerous.

5.2 Ceramic Components

The design rules are structured to allow for multiple applications and the continual development of CMCs. The general ceramic requirements are addressed in Article-3000 of ASME [8] but this focusses on testing requirements as limited ceramic properties are included. The design rules do, however, allow for future applications because they are not limited to the current material systems. They are also process based because they provide guidance for the permissibility of the materials and specify how to qualify the material systems. As mentioned, the rules are probabilistic because failure is derived from the variability in the material strength and accommodates the material changes due to environment exposure, such as irradiation, chemical attack, and/or stress-time-temperature effects.

ASME III Division 5 provide general and technical requirements for ceramic composite components and assemblies. However, the approach included provides a method based on R&D experimental work will be required to determine the mechanical, physical, fracture, and thermal properties of composites and radiation induced property changes. Currently, none of the international design code and standards provide any ceramic material properties.

> *The ceramic core material should undergo testing commensurate to the approach in ASME (even if ASME is not the code of choice) to ensure the material is well defined and the failure modes are known to allow probabilistic methods to be used for core integrity. The design could also then be optimised based on design by analysis methods.*

5.3 Reactor Core

The materials and approach to assess the core fuel is nominally considered beyond the scope of this report. It is noted, however, that the different operating and coolant conditions for the GFR may lead to a different response to the operational experience from PWR conditions. It may therefore be necessary to further consider the metallic interaction of the fuel and assembly under GFR temperatures and fluence.

> *Examine available fast reactor data where possible to understand variation in material properties under irradiation when compared to available PWR conditions. It may be necessary to consider some limited irradiation testing of sample materials. It may also be sufficient for surveillance specimens to provide sufficient forwarding of any differences such that such testing is not necessary.*

The potential for high cycle fatigue from flow-induced vibration should be considered. This is likely to require detailed computational fluid dynamics modelling coupled with thermo-mechanical modelling but could possibly be considered at a later stage.

OPEX on potential deformation and bowing of the fuel assembly, and the potential for gross expansion of the core, should be considered within the design. Distortion of the reactor core or fuel sub-assembly is not currently mentioned in design codes. It may be that this has been designed and assessed away for current experience, and is likely to be less relevant for ceramic cores, but the higher GFR temperatures and operating conditions may mean it is worth further consideration.

> *Examine any tests performed on the ceramic of choice (linked to recommendation above) to see if gross expansion and deformation is possible for GFR conditions.*

5.4 Non-Destructive Testing

Outline structural integrity condition monitoring of the RPV using existing NDT technology is currently impossible, because of the high operating temperature (i.e. 400°C to 800/850°C). NDT inspection of GFR will only be possible during the refuelling periods (<300°C). Therefore, the design of the RPV, reactor core and reactor internals must take into account the inspection access requires for the components.

Any approach to consider online monitoring methods would mean that R&D will be required. It may also be necessary to develop small remotely operated high-temperature NDT equipment for GFR applications. Similarly, existing NDT requirements in international nuclear codes and standards will need to be revised.

5.5 WS64 project

At the European level, there exist only a very limited amount of national codes according to which a fast or a high-temperature reactor was build, significant part of the fleet has been constructed using either the French AFCEN codes or the German KTA codes. Both sets of codes are dedicated to nuclear facilities and are independent of conventional industry equipment, covering the essential parts of these facilities and referring to International and European standards

Taking such set of codes as a basis, stakeholders in the nuclear energy sector have given consideration to developing a European set of codes that would take advantage of the lessons learnt from the whole European fleet of reactors. This gave rise to CEN Workshop 64, Phase 1 (CEN/WS 64) of which took place between 2011 and 2013, Phase 2 (CEN/WS 64-II) between 2014 and 2018 and Phase 3 (CEN/WS 64-III) between 2018 and 2022. The overall objective of the CEN Workshop on Design and Construction Codes for Gen II, III and IV nuclear facilities (CEN/WS64) is to develop and harmonize nuclear design codes in a European context in support of design, construction and licensing of new nuclear reactors as well as plant life management of existing reactors as a key element in a future low carbon system. To this end, the AFCEN code is used as the reference code. CEN/WS 64-Phase 3 was established in anticipation of further developing three AFCEN codes, taken as pilot cases, in order to comply as widely as possible with European reactor fleet construction and maintenance requirements.

The workshop is organized with four specialized “prospective groups”, each of them covering a specific technical area addressed by an AFCEN code and based on the underneath mentioned structures:

- PG1: mechanical equipment for GEN II-III reactors (with reference to the RCC-M code);
- PG2: mechanical equipment for GEN IV reactors (with reference to the RCC-MRx code);
- PG3: civil works (with reference to the RCC-CW code), covering Generation II and III as well as Generation IV;
- PG4: Electrical Equipment for Gen II to IV reactors.

The PG2 addresses mechanical components for non-water cooled reactors and other nuclear installations and uses the AFCEN code RCC-MRx for the Design and Construction of Mechanical Components in high-temperature structures, experimental reactors and fusion reactors (RCC-MRx) as the reference . The RCC-MRx code was developed for sodium-cooled fast reactors (SFR), research reactors (RR, e.g. Jules Horowitz) and fusion reactors (FR-ITER). Special emphasis has been given to the rules for the design of mechanical components subjected to significant creep and/or significant irradiation. RCC-MRx is the selected design code for the European Generation IV fast reactors, in particular the lead-cooled fast reactors (LFR) or accelerator driven systems (ADS) such as ALFRED and MYRRHA respectively and the gas cooled fast reactor (GFR) demonstrator ALLEGRO. The extension of RCC-MRx to GFR and LFR requires special attention to material environmental degradation induced by the coolants (in particular lead, but also helium).

Within the PG2, UJV raised a number of topics, addressing some of the identified gaps in the RCC-MRx code from the perspective of GFR. Namely:

- New and innovative materials will be used in GFR – the process to add them to the code is currently is too complicated
- High-temperature performance data are missing or incomplete for some of the already codified materials
- More attention should be paid to composite and ceramic materials

For the first topic, the process of adding a new codified material to Czech national standards was presented and generally agreed upon as a good example for future work on simplification of the process within the RCC-MRx code.

One of the main conclusions of Phase 3 was, that SMR /AMR designers, including GFR ones, will be more prone to use new material solutions or change the operational conditions, and that it would need to be associated with evolution of the design codes, and this issue will to be addressed in Phase 4, which is planned to 2024 – 2028.

6. Conclusions

This report has provided a high level overview of the available codes and standards, and some relevant design codes, for design substantiation of a high temperature reactor. The information available on ALLEGRO has been reviewed and information such as nominal operating conditions, temperatures, coolant etc. have been considered to review experience and data in the literature to identify a number of potential challenges and potential gaps in the design codes. Some specific items have been noted from the review as:

- Metallic components
  - Some through-life assessment methods include viable alternatives for elements of the design codes where the design codes are not sufficient. Some example aspects include:
    - Ratcheting due to large temperature excursions leading to excessive distortion;
    - High cycle fatigue due to thermal fluctuations (e.g. thermal striping);
    - Creep rupture at higher temperatures;
    - Thermal ageing at higher temperatures;
    - Crack growth by a combination of creep, fatigue and environmental factors;
    - Buckling and/or creep buckling in thin-section components.
  - Need to consider PWHT to all welds where possible.
  - Reduce the number of highest reliability locations and minimise integrity claims on welds.
  - The influence of environment (chemical) on the structural integrity claims that can be made are considered as a priority.
  - Long-term testing is considered in a representative environment at the stresses, strains and temperatures relevant to plant loading conditions.
  - Ensure that the different manufacturing techniques (including weldments) do not degrade the material properties and have associated long-term materials data to support their deployment.
  - Impact of the helium environment on creep, tensile and fracture properties should be specifically examined.
  - Existing materials within the codes may prove challenging for high temperatures at longer durations.
  - For the GFR the approach to component classification should be clearly detailed.
- Ceramic composite components
  - Definition of the component classification and acceptable probability of failure.
  - Sufficient testing to characterize the material reliability and possible failure modes such that provide design curves can be produced. The testing should also look to support the development of materials models mature enough to allow design by analysis methods to be used.
  - Qualification of identified ceramic material behaviour with irradiation.
  - Monitoring and NDT inspection of the ceramic core.
  - Design ceramic core for decommissioning.
- Reactor core considerations
  - Minimising and looking to reduce any impact of flow induced vibrations should be considered.
  - The effects of deformation within fuel sub-assemblies and reactor core and the subsequent impact on the reactor internals may need to be considered further.

- Non-destructive testing
  - Development of small remotely operated high-temperature NDT equipment for GFR applications.
  - Revision of existing NDT requirements in international nuclear codes and standards to include remotely operated NDT equipment requirements.

It is also considered a worthwhile activity to review ongoing developments for other high temperature reactors, in particular the HTGR, to see where any learning can be made or where co-development opportunities may exist.

7. References

[1] SafeG Consortium, "SafeG - Safety of GFR Through Innovative Mateirals, Technologies and Processes - Proposal ID: NFRP-2019-2020," Horizon 2020, 2020.

[2] AFCEN, "RCC-MRx - Design and Construction Rules for Mechanical Components of Nuclear Installations," 2018.

[3] M. Chevalier and P. James, "Guidance for the Structural Integrity Demonstration of Advanced Modular Reactors in the UK," EASICs Project, December 2021.

[4] American Society for Mechanical Engineers, "BPVC Resources," ASME, [Online]. Available: https://www.asme.org/about-asme/standards/bpvc-resources. [Accessed January 2018].

[5] American Society for Mechanical Engineers, "ASME III Rules for Construction of Nuclear Facility Components Division 1 - Subsection NB, ASME BPVC.III.NB," 2021.

[6] American Society for Mechanical Engineers, "ASME Section II Part D Properies (Metric), ASME BPVC.II.D.M," 2021.

[7] American Society for Mechanical Engineers, ASME XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 2015.

[8] American Society for Mechanical Engineers, "ASME III Rules for Construction of Nuclear Facility Components, Division 5 - High Temperature Reactors," 2019.

[9] "Quality Assurance Requirements for Nuclear Facility Applications, NQA-1," American Socity of Mechanical Engineers, 2022.

[10] AFCEN, "RSE-M: In-Service Inspection Rules for Mechanical Components of PWR Nuclear Islands," AFCEN.

[11] M. Nielson and A. Goodfellow, "Independent Review of the French RSE-M Code for Application to the UK EPR Fracture Mechanics Assessment, Paper ID 826," in *SMiRT 23*, Manchester, 2015.

[12] N. G. Smith, "A Comparative Review of International Design and Construction Codes for Advanced PWRs," in *BNES Seminar, Pressure Component Standards for APWRs*, 1995.

[13] AFCEN, "RCC-M - Design and Construction Rules for Mechanical Components of PWR Nuclear Islands," AFCEN, 2020.

[14] "2022 RCC-MRX Code Edition: Content, Overview, On-going Developments, SMiRT-26, Division VI," T. Lebarbé; C' Pétesch; L. Vaillant de Guélis; C. Primault; M. Blat-Yrieix, 2022.

[15] "Evolution brought to RCC-MRx Code in relation to ASTRID project, FR13: International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable; IAEA-CN--199/145," International Atomic Energy Agency, 2013.

[16] J. E. M. Garacia, C. Petesch, T. Lebrarbe, D. Bonne, C. Pascal and M. Blat, "Design and construction rules for mechanical components of high-temperature, experimental and fusion nuclear installations: the RCC-MRx Code last edition", Mechanical Engineering Journal, Paper No.20-00052, Vol.7, No. 4," Japan Society of Mechanical Engineers, 2020.

[17] EDF, "R6 - Assessment of the Integrity of Structures Containing Defects, Revision IV, Issue 12," 2019.

[18] EDF, R5 - Procedures for Assessing Structural Integrity of Components under Creep and Creep-Fatigue Conditions, Issue 3, 2014.

[19] British Standards, BS 7910 - Guide to Methods for Assessing the Acceptability of Flaws in Metallic Structures, 2013.

[20] American Society for Mechanical Engineers, "C&S Connect > Code Case Search," [Online]. Available: https://cstools.asme.org/CodeCases.cfm. [Accessed August 2023].

[21] P. James, J. Sharple and N. Underwood, "UK Programme on Codes, Standard and Procedure needs for SMR and GEN IV Reactors," 2018.

[22] P. James, N. Underwood and J.K.Sharples, "UK Programme on Codes, Standard and Procedure needs for SMR and GEN IV Reactors – Phase 1 Output," in *Proceedings of the ASME 2019 Pressure Vessels & Piping Conference, PVP2019-93861*, 2019.

[23] IAEA, "IAEA Power Reactor Information System PRIS," [Online]. Available: https://pris.iaea.org/PRIS/home.aspx.

[24] "A Technology Roadmap for Generation IV Nuclear Energy Systems, GIF-02-00," USDOE and GIF, 2002.

[25] R. Turk et al, "Advanced Non-Light Water Reactor Materials and Operational Experience," USNRC Technical Letter Report, TLR-RES/DE/CIB-2019-01, March 2019.

[26] F. W. Brust, "Summary of Operational Experience for Advanced Non-light Water Reactors: Materials and Structural Integrity Issues," in *SMiRT-25*, Charlotte, NC, USA, 2019.

[27] J. Busby et al, "Technical Gap Assessment for Materials and Component Integrity Issues for Molten Salt Reactors," US NRC Project Report ORNL/SPR-2019/1089, 2019.

[28] D. Sulejmanovic et al, "Materials integrity considerations and technical gap assessment for molten salt reactors," in *SMiRT-25, Carlotte, NC, USA*, 2019.

[29] P. James, C. Meek and I. Morris, "EASICS: Code Comparison Report for Creep-Fatigue Damage Calculations (Draft)," 2021.

[30] "Deliverable D6.1.1-2 ALLEGRO Core Specification," ESNII+ project, 2023.

[31] ALLEGRO, "ALLEGRO," [Online]. Available: https://allegroreactor.cz/. [Accessed August 2023].

[32] Oak Ridge National Laboratory, "Corrosion in Gas-cooled Reactors, TLR-RES/DE/CIB-CMB-2021-04," 2021.

[33] H. Petersen, "Tables of thermo-physical properties of helium, DRAGON report no. 734," 1971.

[34] R. Cook, "Effects of impure helium upon the creep behaviour of some austenitic stainless steels, Central Electricity Laboratories note no. RDL/N/3/72," 1972.

[35] R. Suhr, "Progress report on Mk III fatigue work in helium, English-Electric-AEI report no. RPC-CM/P(71)53," 1971.

[36] K. Kunzova et. al, "Effect of thermal exposure in helium on mechanical properties and microstructure of 316L and P91," *Journal of Nuclear Materials,* vol. 472, no. 15, pp. 47-54, 2016.

[37] J. Berka et al., "Corrosion tests of high temperature alloys in impure helium," *Proceedings of HTR 2014,* 2014.

[38] G. Gulsoy, "Mechanism of internal oxidation of alloy 617 in controlled impurity helium environments at high temperatures, PhD Thesis, University of Michigan, Ann Arbor, MI," 2014.

[39] J. Geringer, Y. Katoh, S. Gonczy, T. Burchell, M. Mitchell, M. Jenkins and W. Windes, "ASME Code Rules and ASTM Standards intergration for Ceramic Composite Core Materials and Components, Proceedings of HTR 2021, Paper HTR 2021-53," HTR, 2021.

[40] "ASTM Standard C1783-15: Standard Guide for Development of Specifications for Fiber Reinforced Carbon-Carbon Composite Structures for Nuclear Applications," American Society for Testing and Materials, 2015.

[41] "ASTM Standard C1793-15: Standard Guide for Development of Specifications for Fiber Reinforced Silicon Carbide-Silicon Carbide Composite Structures for Nuclear Application," American Society for Testing and Materials, 2015.

[42] "ASTM Standard C1899-21: Standard Guide for Development of Specifications for Fiber Reinforced Silicon Carbide-Silicon Carbide Composite Structures for Nuclear Applications," American Society for Testing and Materials, 2021.

[43] US Department of Energy, "Higher Temperature Reactor Materials Workshop, Sponsored by the Department of Energy Office of Nuclear Energy, Science, and Technology (NE) and the Office of Basic Energy Sciences (BES), ANL-02/12," 2002.

[44] American Socity of Mechanical Engineers, "ASME Handbook – Properties and selection: Irons, Steels, and High-performance Alloys, Volume 1," 1990.

[45] M. Kangilaski, “Radiation Effect Design Handbook - Section 7: Structural Alloys, NASA CR-1873,” 1971.

[46] US Nuclear Regulatory Commission, “Radiation Embrittlement of Reactor Vessel Materials, Regulatory Guide 199 Revision 2,” 1988.

[47] M. Osaki and T. Otaka, “Reduction of Upper Shelf Energy of Highly Irradiated RPV Steel,30th MPA-Seminar in conjunction with the 9th German-Japanese Seminar, Stuttgart,” 2004.

[48] L. Porter, “Radiation Effects in Steel, ASTM STP39595S,” 1960.

[49] A. Brook and R. Anderson, “A Review of PFR Core Distortion Experience,” International Atomic Energy Agency, 1984.

[50] P. Chellapandi, S. Chetal and B. Raj, “Thermal Striping Limits for Components of Sodium Cooled Fast Spectrum Reactors, Nucl. Eng. Des., 239(12), pp. 2754–2765,” 2009.

[51] M. Albert, “Advanced Reactor Material Development Roadmap (Draft Presentation),” EPRI, 2022.

[52] AFCEN, “Guide for Introducing a New Material in the RCC-MRx (Draft),” AFCEN, January 2021.

[53] American Society for Mechanical Engineers, “ASME Section IX Welding, Brazing, and Fusing Qualifications, ASME BPVC.IX,” 2021.

[54] G. Selby, S. Aakre and Z. Fan, “Non-Destructive Examination of Diffusion-Bonded Compact Heat Exchangers, ASME PVP2020-21293,” 2020.

[55] C. Sun, Y. Wang, M. D. McMurtrey, N. D. Jerred, F. Liou and J. Li, “Additive Manufacturing for Energy: A Review, Applied Energy, vol. 282.,” 2021.

[56] M. Callaghan, M. Chatterton, D. Coon, O. Wallace and W. Kyffin, “WP 3.3 Mechanical Performance of PM/HIP Type 316L Austenitic Stainless Steels, Wood Report 207970-TR-006,” 2019.

[57] R. Wright and R. Rupp, “The Elevated-Temperature Cyclic Properties of Alloy 316L Manufactured Using Powder Metallurgy Hot Isostatic Pressing, Idaho National lab report no. INL/EXT-20-59327,” 2020.

[58] R. Rupp, “The Elevated-Temperature Cyclic Properties of Alloy 316H Fabricated by Powder Metallurgy Hot Isostatic Pressing, Idaho National Lab. Report no. INL/EXT-21-62993,” 2021.

[59] J. L. King, A. Shahsafi, K. Blomstrand, K. Sridharan and M. A. Kats, “Impact of corrosion on the emissivity of advanced reactor structural alloys. Journal of Nuclear Materials, 508, pp. 465-471,” 2018.

[60] W. Young and R. Budynas, “Roark’s Formulas for Stress and Strain, Seventh Edition,,” McGraw-Hill, 2002.