ARCHER

Collaborative Project (CP)

Co-funded by the European Commission under the Euratom Research and Training Programme on Nuclear Energy within the Seventh Framework Programme

Grant Agreement Number : 269892

Start date : 01/02/2011 Duration : 48 Months
www.archer-project.eu

Advanced Reactor for Cogeneration of Heat & Electricity-R&D

Material data collation and evaluation for material codes, gap analysis and mitigating measures

Authors : BUCKTHORPE Derek (AMEC)

ARCHER - Contract Number: 269892
Advanced High-Temperature Reactors for Cogeneration of Heat and Electricity R&D
EC Scientific Officer: Dr. Panagiotis Manolatos

Document title	Material data collation and evaluation for material codes, gap analysis and mitigating measures
Author(s)	BUCKTHORPE Derek
Number of pages	24
Document type	Deliverable
Work Package	WP 42
Document number	D42.14
Issued by	AMEC
Date of completion	13/02/2015
Dissemination level	Confidential, only for members of the consortium (including the Commission Services).

Summary

The ARCHER Collaborative Project, running over the period 2011 to 2015, has brought together several key European stakeholders in the development of the technology for the High Temperature Reactor, HTR, and Very High Temperature Reactor, (V)HTR; and has contributed to the international development of these nuclear power plant through the GenIV programme. At the end of the project an important task has been to survey the existing state of the art; and to identify the work to be done in order to develop such reactors to the point at which they can be ordered, built and operated. As part of WP42, several organisations have been particularly tasked in considering the availability of high temperature materials and their properties for use in association with design codes. Such documents are crucial for the design of HTRs / (V)HTRs, and several have been available for several years or even decades (sometimes directed at other reactor systems) yet they need critical evaluation to ensure they meet the current requirements. This is particularly the case for (V)HTRs where temperatures are generally much higher than considered before. Within this report, consideration has been given to the availability of materials data in existing codes, the recent work to update them based on newer testing programmes and assessments, and the specific requirements of (V)HTRs. Gaps in the materials data have been identified, with recommendations given for work to fill those gaps, thereby permitting the effective future design and safe operation of (V)HTR reactors.

Approval

Date	By
16/02/2015	Derek BUCKTHORPE
16/02/2015	Derek BUCKTHORPE
16/02/2015	Steven KNOL

Distribution list

Name	Organisation	Comments
P. Manolatos	EC DG RTD	Through the EC participant portal
All beneficiaries	ARCHER	Through the private online workspace

Table of contents

1 Introduction

The ARCHER Collaborative Project, running over the period 2011 to 2015, has brought together several key European stakeholders in the development of High Temperature Reactor, HTR, and Very High Temperature reactor (V)HTR technology; and has contributed to the international development of these nuclear power plant through the Gen-IV programme. Within WP42, several organisations have been particularly tasked in considering the availability of high temperature materials and their properties for use in association with design codes. Such code documents are crucial for the design of HTRs / (V)HTRs, and many have been available for several years or even decades (sometimes directed at other reactor systems) yet they need critical evaluation to ensure they meet the current requirements. This is particularly the case for (V)HTRs where temperatures are generally much higher than considered before.

The impetus for HTRs has changed considerably since the early work in the 1960s to 1980s. As the EUROPAIRS project, amongst others has pointed out, there is considerable potential demand for nuclear cogeneration [1]. Thus, the demand for cogeneration HTRs appears to be differentiated into those with the potential for hydrogen production, with primary coolant temperature up to 850°C, and those targeted at primary material manufacturing processes, up to 1000°C, Table 1.

Whilst the codes have received much attention over the last decade or so, in a similar timeframe there has been considerable international activity on thermal power systems to achieve higher thermal efficiencies at steam temperatures of 700-760°C. For the first time, extensive use of nickel base alloys is being considered in boilers and steam turbines; a change in philosophy which has a potential mutually beneficial impact on the use of such components for power generation in respect of HTRs. This is not least in the field of materials, where extensive testing has taken place within programmes such as AD700, AD700-II, US A-USC and similar [2],[3].

The design of reactors, steam generators, heat exchangers and the non-nuclear island some 30 years ago would have been predominantly code case based; and that necessarily continues to this day. However, in the intervening period the analytical techniques available for structural integrity assessment of components has improved immeasurably. Similarly, our knowledge of materials and how they might be described in material models has advanced. The net effect is that finite element analysis of complex geometries with materials properties described by equations rather than by tables is becoming a common addition to the tools to ensure efficient and safe design.

At the end of the project it is important to survey the existing state of the art; and to identify the work to be done in order to develop such reactors to the point at which they can be ordered, built and operated. In the following sections we therefore see our responsibility to both look to the past, and to anticipate the future. Our philosophy has been to consider the state of the art in external codes and other publications, the work within ARCHER expressed within the deliverables (particularly SP4, [4] to [9], but also SP2, [10], [11]), and thereby provide a synthesis of gaps in materials data and mitigating measures.

1.1 Requirements for Materials

In the context of the present report, we consider only the non-fuel materials; principally those used for structural, vessel, and out-of-core components. For example, the reactor fuel, graphite moderator rods and insulation are specifically excluded from this overview. A particular aspect of current Gen-IV thinking is that the reactors are already expected to operate for 60 years, leading to additional demands on qualified materials properties for design.

According to Maloy et al [8] a key component of the HTR reactor is the pressure boundary and the properties can be simply summarised as normal/abnormal temperatures of 200-300°C at 5-8MPa coolant pressures; requiring good (non-creep) mechanical strength at low temperatures to 300°C. The material for the vessel should be able to withstand the coolant environment and a moderate neutron fluence of up to 1.10^{19} n.cm^{-2} over a period of 60 years; with little or no corrosion or loss in properties. Moreover, the vessel and other parts of the pressure boundary should be fabricable by forging rings and other parts, and can be assembled by welding on-site.

Within ARCHER the material selected for the pressure boundary of the HTR was 20MnMoNi55 steel [8][13], already well qualified for vessel manufacture under code KTA 3201 [13]. This choice was made on the basis

Table 1. Heat market per type of nuclear reactor technology, according to ARCHER D12-31 [1]

Reactor type	**Max. required steam/heat temperature**	**Relevant applications**	**Minimum temperature range needed**	**Comment**	**Total heat market**
Water reactors	250°C	District heating	70-130°C	Seasonal, nuclear already used	700 TWh/y (≈ 80 GWth)
		Desalination	100-130°C	High growth expected, nuclear already used	
Fast breeders	450°C	Pulp & Paper	100-250°C	High load volatility, nuclear already used	
High temperature reactors	550-650°C (secondary steam) < 750°C (secondary gas)	Oil refining	500-550°C		1 000 TWh/y (≈ 114 GWth)
		Chemicals	500-550°C		
		Fertilizers	500-600°C	Largest H_2 application	
Very high temperature reactors	1000°C	Hydrogen	750-850°C	Sector expected to grow, in particular with new applications in transport, energy, energy storage and Carbon Capture and Utilization	1 300 TWh/y (≈ 148 GWth)
		Other applications	>1000°C	Other sectors could be envisaged for nuclear pre-heating (e.g. lime, aluminium, iron & steel, high-temperature O_2 production, glass)	

that temperatures could be kept below 300°C at the pressure boundary wall, by internal arrangement of the reactor and other major components. Moreover a pipe-in-pipe concept was considered in which the cooler return flow only was in contact with the pressure boundary. In separate work within ARCHER, Type 316 stainless steel was chosen for the shell of the Intermediate Heat Exchanger (IHX) and for the shell of the Steam Generator [8]. In other design concepts, and particularly for the higher primary coolant temperatures of the (V) HTR, then forgings of Grade 91 are generally considered for the pressure boundary [eg. [11],[14]]. In the selection of reactor vessel the intention is to keep below the temperature at which creep needs to be considered in design (the "insignificant creep" or "negligible creep" temperature). For continuous operation, generally this is thought of as 325°C for low alloy steels and 375°C for creep strength enhanced martensitic steels such as Grade 91.

2 Evaluation of Material Codes

2.1 Norms and Standards

Pressure vessel design and construction codes provide mandatory rules for calculation of material thickness, allowable materials of construction, welding, NDE, testing and regulatory documentation.

The pressure vessel code(s) applicable to the location of installation must be followed for mechanical design and construction. There are several commonly accepted ‘global’ design codes:

- ASME BPVC – American Society of Mechanical Engineers, Boiler and Pressure Vessel Code [15]
- ASME - American Society of Mechanical Engineers, Code Case N47 [16]
- KTA 3221 “Metallische HTR Komponenten” Parts 1 to 3 (Draft) [17]
- KTA 3201 Light Water Reactors [13]
- EN 13445 – the European standard, harmonized with the Pressure Equipment Directive (PED) [18]
- RCC-MRx - Design and Construction Rules for mechanical components of nuclear installations [19]

For High Temperature Reactors, HTRs (with maximum coolant temperatures of 750-850℃) and Very HTRs (800-1000℃), the first four of those listed above dominate, and are considered below in more detail.

2.2 ASME N47

The US Nuclear Code, ASME N47 [16] was originally developed for PWR design and construction, hence the effects of creep were largely ignored. During the 80s and 90s it was appreciated that significant work was required to update the ASME codes to meet the requirements of FBR/HTR design, both within the core and in the remainder of the reactor, see eg [20]. During the 1990s, ASME Section III N47 was essentially migrated into ASME BPV Section III Subsection NH.

2.3 ASME BPV

The ASME Boiler and Pressure Vessel Code, at the time of this report, was most recently released in 2012, with a revision scheduled for 2015. As noted in section 2.2, apart from the reactor core, most other matters are covered, or intended to be covered within the BPVC. In the first decade from the start of the century, although most of the technical work had been started, the BPV codes (principally Subsection III “NH”) had not been modified. In the “NGNP High Temperature Materials White Paper” published in 2010 [**] the list of tasks completed (identified with Standards Technology Publications of the form “STP-NU-xxx”) and still planned was given, Table 2. Essentially, considerable progress has been made, but there still remain several tasks to be fully developed. Of particular note is the document: STP-NU-045-1– 2012 Roadmap to Develop ASME Code Rules for The Construction of High Temperature Gas Cooled Reactors (HTGRS) [21]. Section 8 of that document essentially updates the earlier review of Huddlestone and Swindemann [20], and the NGNP HTR White Paper [23] identifying the future tasks to be completed to update the BPV.

Issued in November 2011, the ASME BPV has a new Division 5, which essentially brings together the existing parts of the code relating to HTRs, together with Subsections requiring development. This development was presented at the PVP2012 conference by Morton et al [24]. A summary of the changes is also given in the Appendix (supplied by D Buckthorpe).

2.4 KTA 3221 and 3201

The German Code, Draft KTA 3221 [17], represents something of an anomaly in HTR design: whilst it is widely referenced in development programmes and, at least until recently, was the only source of design data at sufficiently high temperatures for HTRs, as it has not been formally released. Moreover, the significant effort that went into developing the code by organisations including KFA Julich and others, the decision by the German government soon after the 2011 earthquake in Japan, and the problems at the Fukushima plant, means that support of that code may not be forthcoming. In particular, it is becoming increasingly difficult to either locate or obtain the source assessment documents that are the origin of the values within KTA 3221. That is not to say that there is any doubt about the efforts undertaken by German workers to develop the data, as illustrated by publications by Nickel, Diehl [25], [26] and others.

Table 2. (Upper) Tasks required and their progress in updating the ASME BPVC, adapted and updated in *red italic* from the NGNP HTR White Paper [23].
(Lower) Listing of Relevant STP documents.

Task 1 Allowable Stresses in Section III, Subsection NH on Alloy 800 H and Grade 91 Steel (STP-NU-020)
Task 2 Regulatory Safety Issues in Structural Design Criteria (STP-NU-010)
Task 3 Improvement of Subsection NH Rules for Grade 91 Steel (STP-NU-013)
Task 4 Updating Nuclear Code Case N-201
Task 5 Creep-Fatigue Data and Evaluation Procedures for Grade 91 Steel and Hastelloy XR (STP-NU-018-2009)
Task 6 Operating Condition Allowable Stress Values (*STP-NU-037*)
Task 7 ASME Code Considerations for the IHX (*STP-NU-038*)
Task 8 Creep and Creep-Fatigue Crack Growth (*STP-NU-039*)
Task 9 Update NH – Simplified Elastic and Inelastic Methods (*STP-NU-040*)
Task 10 Update NH – Alternative Simplified Creep-Fatigue Design Methods (*STP-NU-041*)
Task 11 New Materials for NH (*STP-NU-042*).
Task 12 Nondestructive examination and ISI Technology for high temperature reactors (HTRs; Funded by NRC)
Task 13 Recommend Allowable Stress Values (planned for 2010) (*STP-NU-035*).
Task 14 Corrections to Stainless Steel Allowable Stress (*STP-NU-059, STP-NU-063*).

Listing of STP documents relevant to Section III, Subsection NH / Section V

STP-NU-010 Regulatory Safety Issues in Structural Design Criteria of Sec III NH for VHTR & GENIV Reactors
STP-NU-013 Improvement of ASME Section III-NH for Grade 91 Negligible Creep and Creep-Fatigue
STP-NU-018-2009 Creep-Fatigue Data and Existing Evaluation Procedures for Grade 91 Steel and Hastelloy XR
STP-NU-019-1 Verification of Allowable Stresses in ASME Section III, Subsection NH for Grade 91 Steel
STP-NU-020 – Verification of Allowable Stresses in ASME Section III, Subsection NH for Alloy 800H
STP-NU-035 – 2012 Extend Allowable Stress Values for Alloy 800H
STP-NU-037 – 2010 Operating Condition Allowable Stress Values in ASME Section III Subsection NH
STP-NU-038 – 2010 ASME Code Considerations for the Intermediate Heat Exchanger (IHX)
STP-NU-039 – 2011 Creep and Creep-Fatigue Crack Growth at Structural Discontinuities and Welds
STP-NU-040 – 2012 Update and Improve Subsection NH-Simplified Elastic and Inelastic Design Analysis Methods
STP-NU-041 – 2011 Update and Improve Subsection NH--Alternative Simplified Creep-Fatigue Design Methods
STP-NU-042 – 2011 New Materials for ASME Subsection NH
STP-NU-045-1– 2012 Roadmap to Develop ASME Code Rules for The Construction of High Temperature Gas Cooled Reactors (HTGRS)
STP-NU-059 – 2013 Corrections to Stainless Steel Allowable Stresses
STP-NU-063 – 2013 Correct and Extend Allowable Stress Values for 304 and 316 Stainless Steel

For example, it is required to periodically reassess materials property data in the light of new testing programmes. At present it would seem difficult or perhaps impossible to obtain the test data used in the derivation of the KTA 3221 values. By default, then, unless there is the political will to develop HTR reactor technology in Germany, there remains some doubt as to the full value that can be placed on KTA 3221 documents.

On the other hand, the Light Water Reactor code, KTA 3201 [13], is fully maintained and available from the German regulators. Indeed, it has to be, since LWR are still running in Germany and in other countries designed to those codes. Except for the pressure boundary material, and the associated design rules however, there is relatively limited use that could be made of KTA 3201, bearing in mind it is for reactor systems at considerably lower temperatures, where the design is below the negligible creep temperature, and so creep can essentially be ignored.

2.5 EN 13445

The European standard EN13445 in many ways has a similar scope to the non-nuclear parts of ASME BPV code. However, *in terms of materials data* there are substantial differences. Whilst the ASME BPV provides tables of materials properties, including material strength tables and derived allowables; in the case of EN13445 these are omitted, with the user referred to the product standards instead (eg EN 10028). Moreover, in EN13445 two Annex's are provided which allow the designer to extrapolate material property data from within the product standard to the conditions of interest:

1 Annex R (informative) Coefficients for creep-rupture model equations for extrapolation of creep rupture Strength

2 Annex S (informative) Extrapolation of the nominal design stress based on time-independent behaviour in the creep range

Whilst this is a laudable attempt to aid the designer if data are unavailable, the philosophy is quite different between BPV and EN13445: if the data are omitted from the BPV, they simply cannot be used from another document, or else generated by extrapolation. Similarly, the data in the EN product standards, are already extrapolated (often by x3 in rupture duration, and by – at least – 25°C below the conditions of testing, and sometimes above); further extrapolation entails some additional unquantified risk. In addition, EN13445 contains information concerning design against fatigue conditions, but there are no rules for creep fatigue interaction per se, rather this topic is covered in Annex B (normative) Design by Analysis.

The cross reference from EN13445 to the product standards offers a reasonable degree of flexibility in the latter – the informative annex containing the strength properties can reflect the needs of that semi-finished product class. On the other hand, it is known that the strength values in such standards have been derived from older national standards, and the degree of traceability to the original assessments must be questionable. Some of those values have been updated by the independent but quasi-standards authority the European Creep Collaborative Committee [27]. The ECCC datasheets containing assessed creep properties however generally represent the state-of-the-art in terms of collated data, method of assessment and technical review, and can be relied upon. Nevertheless, the datasheets have no formal authority within the standards world unless and until they are adopted within the relevant product standard.

2.6 French RCC MRx Code

Within the French nuclear industry (principally CEA, EdF, AREVA), the codes, RCC M, RCC-MR, and latterly RCC-MRx, have been developed to design LWRs, sodium fast reactors and in future (V) HTRs. Most recent experience relevant to Gen-IV has been on fast reactors, though there is currently a strong intention to develop the RCC-MRx codes to be applicable to Gen-IV reactors. Of particular note is the CEN Workshop CEN WS/64 organised by the AFCEN , which in 2014 began its Phase 2, with the Business Plan and other documents available from the CEN website [28]. Its stated objective is:

> "The proposal consists into a voluntary mechanism for a broad set of partners involved with design and construction of nuclear facilities in Europe. It will allow partners not yet using AFCEN codes to learn about these codes. It will also give the opportunity to all participants to express their specific requirements for the long term modifications of the Codes including identification of pre-normative research where necessary."

Essentially the workshop allows members to recommend the future medium term orientation of the codes, identify R&D needs, and look for substitution by existing international standards (or develop them).

It is important to note that the RCC codes were themselves developed from the ASME BPV code, with the licence to use the latter code purchased in the late 1970's; after which the RCC codes were developed based on French (and other) engineering experience and the particular needs of the French nuclear industry. Now, with the focus moving to Gen-IV reactor systems, the RCC-MRx code will be developed formally within AFCEN, but with the input from CEN WS/64 Phase II. An explicit intention is to study and consult other codes including those from KTA, the British R5/R6 codes and the work conducted within ASME.

Of particular interest to the present report is the status of materials property data. It is almost certain that such data will be explicitly included in the RCC-MRx code, and such property data will either be developed specifically for the code, or adopted from other sources, in the case of (V) HTRs, KTA 3221 is an obvious source of such data. However, as will be discussed in later sections of this report, such data are potentially different to other, more recent published data sets. It may also be difficult or impossible to trace the original assessments, raising questions about the traceability about the property data in KTA 3221.

2.7 Other Codes and Discussion

Regulatory bodies in other parts of the world have access to design codes for HTR reactors. Notably, these include the Chinese NNSA (National Nuclear Safety Administration), and the Atomic Energy Commission of Japan. However, the Chinese and Japanese codes are not readily accessible, and therefore have not been reviewed here. Taking the measure of recently published work, it would seem that the ASME BPV currently dominates in respect of HTR design. Currently, the ASME BPV is the only code that is readily accessible and recognised in the Europe and the US, which already has the range of materials property data and related stress allowables appropriate to the HTR, or has a specific development plan in order to incorporate such data. In theory, the KTA 3221 (Draft) could be approved and released, but this seems highly unlikely given the German government's announced nuclear phase-out, accelerated after the 2011 Fukushima nuclear incident in Japan. Whilst the KTA 3221(Draft) now seems unlikely to be updated, it is under review for incorporation into the RCC MRx code under the auspices of AFCEN and the CEN WS/64 Phase II, described above.

The ASME organisation, through the Standards Technology LLC organisation, seems well-funded and motivated to cover the gaps in design data, as evinced by the relatively large number of "STP" documents aimed at updating the relevant code sections, Table 1. On the other hand, there are some substantial gaps in terms of materials, for example bolting materials are not covered explicitly (though can be designed according to other accepted codes). Similarly, the environmental effects of the HTR coolant are not yet included, as neither are the properties of thin sections or alternative joining methods (though these matters are noted in documents such as STP-NU-038). Nevertheless, the momentum in updating ASME BPV seems unstoppable. In terms of materials, at least, there still remains a high capacity of expertise in Europe, the Far East and elsewhere that could and should contribute data, assessment expertise and understanding to ensure that the material property values in European, ASME BPV and other codes are accurate and produced in timely fashion. Most of the data used to assess such property values are typically collated and assessed from international sources. Presently, this is done separately for each code's requirements. It would seem to make good sense to consider if such activities could be co-ordinated worldwide, not least because such R&D on such topics is considered by many industrial organisations to be pre-competitive.

3 Material Data Collation and Gaps

Data on high temperature materials area available from a wide variety of sources, are of varying quality, and must be distinguished as being either test data on particular heats, or material property data describing the behaviour of any material purchased to a specification. It is only the latter that are used in the calculation of design allowables. Generating test data can be expensive, creep testing costs can amount to 3-4M Euro for well characterised materials [29]. Thus it is imperative that such testing is performed accurately, efficiently and appropriate to industrial needs. This generally means that such programmes are collaborative, though alloy producers and end users may also run their own programmes.

Sources of creep and fatigue data include:

- Material suppliers' own testing programmes
- Power plant equipment manufacturers' own testing programmes
- National data generation programmes, such as have been or are being conducted within the UK (ERA Project 2021 Creep of Steels), France (CEA), Germany (German Creep Group, AGH and related) and Japan (notably the NIMS datasheet series)
- Industry or standards bodies such as the Materials Properties Council, European Creep Collaborative Committee.
- EU funded Joint Research Centres (eg JRC Petten) and US national laboratories (eg Oak Ridge National Laboratory – US Department of Energy).
- EU funded research programmes (eg EU BRITE-EURAM, EU Framework 5-7 programmes, COST activities).

- US funded research programmes (eg. A-USC, Advanced – UltraSuperCritical Steam Technology Programmes, U.S. Department of Energy)
- Collations and Compilations, such as the ASM Metals Handbook Series, Metallic Materials Properties Development and Standardization handbook (Battelle)

Owing to the high cost of creep and fatigue data, it is often not published, and access to such data may only be given on formal application, noting that the data are to be kept confidential. Generally, though access by standardisation bodies to such data are forthcoming and it is notable that in recent years, many standards development organisations have had access to the same test data, but developed different data assessments, and hence allowable stresses. As reviewed in detail - within ECCC publications eg [41] - that is because differences in assessment methodologies can lead to different mean and minimum properties being declared, a matter which is often exacerbated when the properties can only be obtained by extrapolation. This points is considered further in Section 3.2.

3.1 Pressure Boundary Materials

3.1.1 20MnMoNi55

In the German KTA Code 3201, 20MnMoNi55 is the preferred pressure vessel material for PWR construction and, noting the design requirement to keep the normal vessel temperature at or below 325°C should have sufficient properties to operate in the non-creep regime. A similar approach is followed in the ASME BPVC Code III Subsection NB for the US widely used alternatives grade SA-508 Grade 3 Class 1/ SA-533 Type B Class 1. The steel SA 508 Grade 3 Class 1 approved for use up to 371°C temperatures within ASME BPVC Section NB, essentially for non-creep use; generally this implies that a vessel cooling system will be used to limit the maximum temperature of the vessel. Recently ASME Code Case N-499-2 was approved to permit limited high temperature excursions up to 427°C for 3000 hours and 538°C for 1000 hours for accidents.

1.1.1 Grade 91

The steel, modified 9%Cr1%Mo – "Grade 91" - is the favoured, but not the only ferritic/martensitic steel being considered for the pressure boundary of the (V)HTR. Originally developed as a boiler tube and pipe material, it now also sees extensive use in plate and other forms. A cast variant exists in both ASTM and EN standards, and forging variants also exist. To date, however, it's development as a *thick-section vessel* forging material has been somewhat restricted to the nuclear industry, and some development will be required to demonstrate manufacturing aspects. Similar to Grade 20MnMoNi55 considered above, the same gaps need to be considered. During normal operation the (V)HTR pressure boundary is likely to be designed to be below the negligible creep temperature for Grade 91, but excursions during abnormal operation are likely to take the material, at least for short durations into the creep regime.

3.1.2 Gap Analysis

Perceived gaps in the data for the pressure boundary materials 20MnMoNi55, and for Grade 91, are principally concerned with the response of such steels to the 60 year planned service. Taking the key requirements in turn:

1. Resistance to substantial change in fracture toughness properties owing to the (V)HTR's slightly harder (higher energy compared to PWR) neutron fluence of 10^{19} n.cm^{-2}. It is likely that no special programmes are necessary during design stages, but surveillance programmes during operation will be required.

   The general tendency for higher alloyed steels to have higher ductile to brittle transition temperatures, and the higher normal operating temperature of Grade 91, would in the absence of test data, suggest that it should be more susceptible to embrittlement. However, present studies show that this is unlikely, and even in the presence of substantial radiation damage, Grade 91 remains relatively resistant to a change in fracture toughness properties, Figure 1 after Boutard et al [21].

2. The definition of the negligible creep temperature needs to be reconsidered for both materials (see also STP-NU-045, Section 8 [21]). In terms of stress allowables, it is likely to be sufficient to keep to the current temperature of 371°C. However, in terms of local creep damage, particularly at welds,

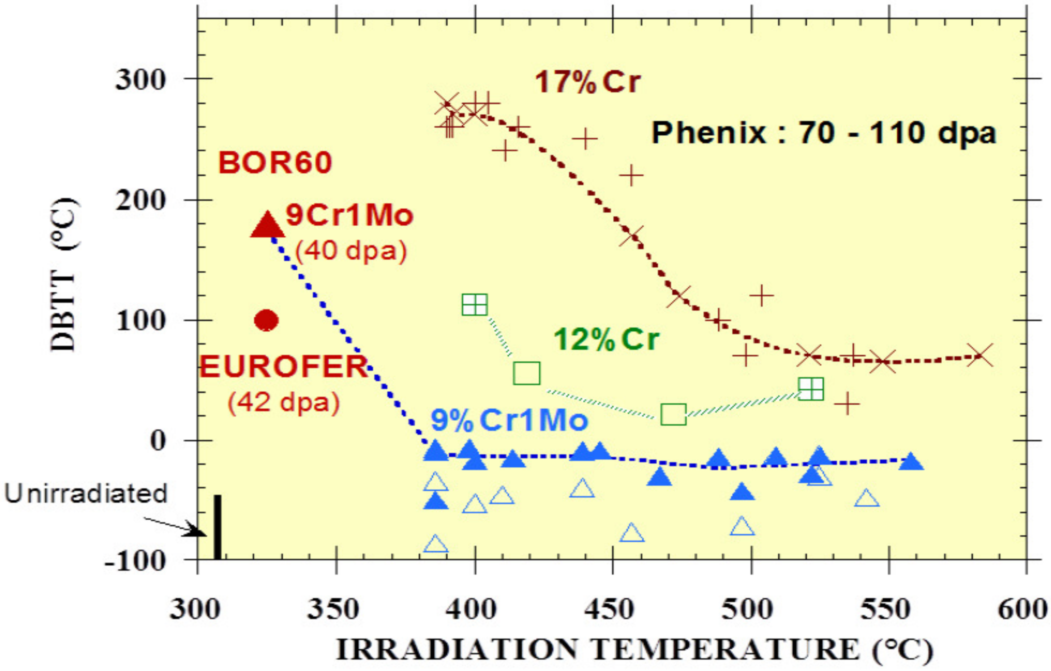


Figure 1. Ductile to brittle transition temperature for F/M steels with different Cr contents as a function of irradiation temperature, from Boutard et al, 2008 [21].

stress relaxation is likely at these and even lower temperatures of normal HTR operation (300-325℃). Such relaxation is often beneficial in terms of creep life, but where local strains approach or exceed the equivalent of 0.5% total plastic strain, alternative design approaches may be required.

The negligible creep temperature for Grade 91, for the (V)HTR, is stated to be 370℃ in ASME BPV Section III Division 1 Subsection NB, but this needs to be reconsidered, in the light of recent assessment activities in ECCC and elsewhere. A more conservative and likely more accurate assessment of negligible creep can be obtained from assessment [30]. Further background to the ASME code requirements on negligible creep is considered in [31], which also described a programme of creep tests close to the negligible creep limit. As with 20MnMoNi55, the potential for relaxation, should be considered.

a. Related to negligible creep, and particularly for 20MnMoNi55, understanding of the creep-fatigue behaviour above and below the nominal negligible creep temperature needs to be improved; leading to a creep fatigue interaction diagram at relevant temperatures for design.

3. In thermal power systems, the presence of Type IV cracking (an intercritically annealed region with lower strength near the edge of the HAZ closest to the parent) and its effect on weld strength factors (WSFs), is under active consideration for all advanced 9-12% chromium steels, including Grade 91 and Grade 92. WSFs of 0.65 or below are being considered at long creep durations (200kh). Although this is primarily a creep phenomenon, the presence of a “metallurgical notch” cannot be avoided in weldments of the reactor vessel and cross vessel, and should be fully understood during the qualification of this material. This is particularly the case if it can be shown that the presence of such a “notch” can exacerbate creep-fatigue interaction.

4. Compared to the ferritic/martensitic grades, the low alloy grades 20MnMoNi55 and the US alternatives are more likely to oxidise in the HTR coolant environment, and further work is possibly required to understand the oxidation thicknesses, and under what conditions the scale is likely to exfoliate. This will be particularly important in the light of the planned 60 year service.

5. Generally the 9-12% ferritic/martensitic grades should exhibit better oxidation resistance compared to the low alloy grades. However, there is some industry evidence that steels with less than 9%Cr (as is permitted in some Grade 91 specifications) are only slightly more resistant to oxidation (in air

or steam) compared with steels with 1% to 2¼%Cr. A transition region at or slightly above 9%Cr seems to exist, above which oxidation is substantially reduced. Such behaviour, may need to be considered in relation to raising the minimum %Cr required in the specification, if it proves relevant to degradation in HTR environments.

6. In the realm of the ARCHER project, the designs for both the SGU and the IHX are based on austenitic steel shells. To date, no work has yet been undertaken in order to characterise the dissimilar metal welds. Similarly, it is unlikely but not yet certain that expansion bellows will be needed (they can probably be designed against at least on the primary coolant circuit). Should they be needed, the materials used in their manufacture, and the design methodology, has still to be decided upon and ratified.

7. Finally, in many designs of the SGU, it is envisaged that bolting will be employed in order to fix the dome to the top of the shell (although welded designs also exist). Currently Type 316H and Type 321H are nominally identified as bolting material for such purposes, but the capability of these materials as bolts at temperatures approaching the primary or secondary coolant temperatures must be questioned. If bolts are to be used at relatively high temperatures, then for the HTR and particularly the (V)HTR, superalloy bolting alloys will need to be developed and characterised.

3.2 Tubing, Pipework, Hot Duct Liner, Structural Members

3.2.1 Alloy 800H

Alloy 800H is one of the most tested alloys used in HTRs, having been extensively reviewed in ARCHER deliverables eg. [6],[7]. Essentially it is an austenitic alloy, sometimes referred to as a super-austenitic steel owing to its high levels of chromium and nickel. Its use in future HTRs is assured, but for (V)HTRs operating above 800-850°C it is likely to be replaced by Alloy 617, Alloy X (or its derivatives ...XR, …XR-II), or Alloy 230. Expected major uses include the hot-liner of the cross vessel, steam generator tubing and plate, intermediate heat exchanger tubing or thin plate (for the compact IHX).

Of primary interest is the creep rupture behaviour. Within KTA3221, a full table of creep properties is reported, with those properties likely to be related to the work of Diehl and co-workers [26]. Recent work by Swindemann and co-workers was undertaken in the US in support of code case development [32], building on the earlier work of Booker et al [33]. In 2012, the work of Swindemann and co-workers was extended and reported within STP-NU-035 [34], with materials properties and stress allowables derived for temperatures up to 900°C, and to 500kh (but not at the highest temperatures). That work should allow extension of the ASME BPVC III Subsection NH to be updated to permit (V)HTR design using Alloy 800H.

3.2.2 Alloy 617

Within the context of US DOE and ASME activities, Wright and co-workers [35] has reported on Alloy 617 parent and weldments, based on tensile, creep, creep fatigue tests, and TEM investigations. The effect of γ' strengthening at 750°C was considered, along with the phenomenon of serrated yielding at intermediate strain rates. Generally, Alloy 617 behaves largely as anticipated, with no barriers foreseen to the introduction of the material into the BPVC Code III Subsection NH in 2015-2016.

In general, the specification for Alloy 617 is more fluid than for Alloy 800H. Recently, there has been much interest in Alloy 617 with controlled Al+Ti levels, leading to improved creep properties in the range 700-800°C [38]. Such materials are sometimes given the nominal designation Alloy 617B, or Alloy 617CA ("<u>C</u>ontrolled <u>A</u>dditions") to distinguish them. Clearly, the matter of which specification to use needs to be decided upon prior to codification. For pragmatic reasons, it seems likely that the original specification will be initially chosen, with modified specification introduced at a later date.

3.2.3 Other Alloys

As described in most detail in STP-NU-038 [39], a number of other alloys are being considered for these components, notably Alloys X, XR and XR-II (tradename "Hastelloy X" and related derivatives), and Alloy 230 (tradename "Haynes 230"). In terms of creep rupture strength, Alloy 617 and Alloy 230 have the best resistance to creep at high temperatures (though Alloy 230 has the advantage of no cobalt); Alloy 800H is limited by its creep response to applications to below 800°C; whilst alloy X, XR and XR-II have an intermediate temperature capability; the restricted compositional ranges of XR and XR-II conferring a slight advantage over Alloy X.

In cooler sections of the IHX and SGU then the 300 series steels, notably Type 321 or 321H and Type 316L or 316H could be used for structural parts, and the ARCHER designs for the compact IHX and the SGU include these materials, including for casings. In all cases, the properties listed in Subsection NH will need to be updated to include these materials up to the maximum temperatures of application.

3.2.4 Gap Analysis

The most thorough assessment of shortcomings in the design Codes is given in STP-NU-038 [39]. Whilst that document necessarily concentrates on the ASME BPV and the IHX, it also includes a survey of historical code development, and surveyed requirements of industrial organisations. A summary of the current experience is given in Table 2.

The perceived gaps or shortcomings in respect of Alloy 800H and Alloy 617 are as follows:

1 In general, the European Creep Collaborative Committee provides guidelines for creep rupture data assessment, in essence restating the common "rule of thumb" that data should not be extrapolated more than three times the longest duration failed test point, and within 25°C from that test point's temperature. That rule of thumb must of, course, be broken in the ASME work to extend the database of Alloy 800H to 500kh (otherwise test durations of 167kh would be required over the whole temperature range of expected use). In the case of STP-NU-035, it would be useful to examine the assessed dataset in respect of the required extent of extrapolation.

Considering this point in further detail, a comparison of the strength values is considered in Figure 2. In the upper part of the figure, the strength values from the new assessment of STP-NU-035 [34], the tabular values from KTA 3221.1 and from the ECCC datasheet are plotted as mean creep rupture strengths at specified durations vs temperature (the >300kh "ECCC" data has been obtained by extrapolating the parametric equation; that is essentially prohibited by the ECCC rules, but has been done to afford a comparison. In the lower figure, the data are shown as isothermal plots. Of most interest are the properties in the range 700-900°C; at stresses above 10MPa, and durations of $\geq$100kh. In those regions, it can be seen that:

- the new assessment in STP-NU-035 is reasonably similar to the KTA 3221 assessment (indeed the latter has generally slightly higher strengths).
- The ECCC datasheet values are similar to the STP-NU-035 values, except at longer durations where the ECCC datasheet values are lower (a deficit of up to ~30% in strength at 500kh)

Although the dataset assessed in STP-NU-035 is larger than preceding assessments, the extent of *long-term data* is likely to be the same. Therefore, such differences are potentially an artefact of the assessment procedures, particularly the choice of time-temperature-parameter. Similarly, the ratio of mean to minimum strengths varies slightly between assessments. For example at 700°C and 100kh, STP-NU-035 has a ratio of 78.3%, KTA 3221 it is 80%, and for the ECCC datasheet it is not stated (likely to default to 80% in EN standards). At 900°C and 100kh, the ratios are 72.9% and 70% for STP-NU-035 and KTA 3221, respectively; different, but not by much. Those points should be considered further before the values are accepted into codes.

2 In STP-NU-035 it is further noted that Alloy 800H datasets contributed from Petten report shorter times for the same test conditions. Moreover, the shape of the creep curve gave some problems in assessing the <u>time to the onset of tertiary creep</u>: some tests gave a "classical" shape, with well-regulated primary, secondary and tertiary regions; whilst others were less regular, sometimes showing an apparent tertiary region occurring soon after the 1% creep strain level, followed by a time of reduced creep rate, and a second tertiary region. The authors note that those data form the basis of the calculated stress allowables; and as such further consideration of the unusual shape of the Alloy 800H creep curve needs to be considered further.

3 Discontinuous yielding of Alloy 617 has been identified by Wright and his co-workers [35] [36]. Also variously called "serrated yielding" and "dynamic strain ageing" this occurs in tensile tests at temperatures of around 650-750°C and at strain rates of about 10^{-4}. A similar phenomenon is seen in tensile curves of Alloy 800H, albeit at lower temperatures, perhaps 500-600°C. This behaviour is

usually associated with strain ageing, where the dislocations necessary for plastic strain become decorated with interstitial atoms. Under certain conditions, they become “locked” and unable to move except with sudden and (it is thought, though not proven) localised “bursts” of strain. The net effect is generally small and manageable within current design procedures, though Wright cautions that lower ductility can result.

4 Wright also described the low cycle fatigue behaviour, with and without hold time. Whilst data of other authors is also discussed, in general, there is an insufficiency of LCF data on Alloy 617, certainly insufficient it appears to produce a credible creep-fatigue interaction diagram. Whilst on this matter, it is noted that there are differences in philosophy in combining creep and fatigue, either based on life fraction (as in BPV) or on strain fraction (UK R5 code), see also STP-NU-041 (Table 2). Similar work has been undertaken within the EPRI project [37], which should be studied in order to arrive at best practice.

5 Within the ARCHER programme, the IHX has investigated Alloy 800H in machined thicknesses of less than 3mm. Deliverable D43.51 reviewed the consequence in terms of mechanical and other properties. Although there are several reports that thin sections should have worse creep properties (eg, to due oxidation, surface effects remaining from manufacturing, fewer grains across the section etc) in fact in testing in WP43 the properties of thin samples was similar to the KTA mean values taken for the IHX design.

6 STP-NU-038 (Clause 4.2.1.3) has further raised the question as to whether very thin sheet (1.5mm or less) can be used since the diffusion of tritium through such thin sections may cause excessive contamination of the secondary circuit (although it may be possible to develop “tritium traps” to mitigate such a problem). Nevertheless measurements of tritium diffusion through IHX sheet materials may be advisable to quantify any contamination effect.

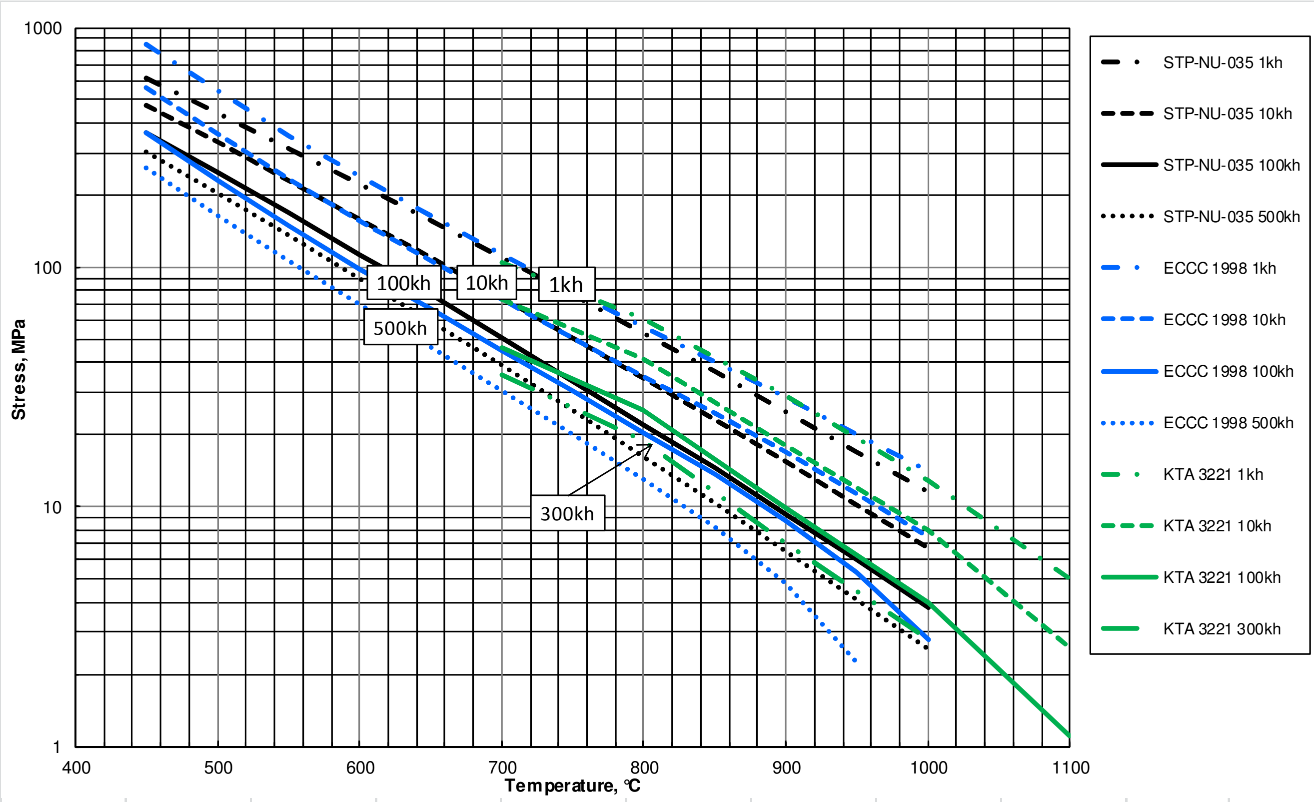
7 The effect of environment – impure helium - on Alloy 800H was similarly reported in ARCHER deliverables [6] and [7]. Generally, the effect depended on the relative levels of impurities; such that the balance between oxidising and reducing impurities was key. In any given HTR environment and for a particular alloy, there appears to be a characteristic temperature, T_A, above which the effect of the environment becomes particularly detrimental. It will be essential to follow up that work, not least to set an upper safe temperature limit for Alloy 800H and Alloy 617. Even below that temperature, testing is required to determine the corrosion/erosion allowance of several materials for design of thin section components in impure helium.

8 Related to this point is the possibility of using coatings to improved reduced environmental degradation, see also STP-NU-038 (4.2.1.3). The benefits could be considerable in terms of lifetime of thin sections, but there are potential drawbacks including the risk of exfoliation, reduced heat transfer and difficulties with either joining a coated sheet, or coating a sheet assembly once joined.

9 A success of the ARCHER IHX programme has been the development of laser beam welding for the compact plate heat exchanger. The welds are applied without subsequent post weld heat treatment, or machining of the flash from the interior of the assembly (which is inaccessible). Even though there is a considerable change in hardness, the presence of sharp corners, it is a credit to the WP43 participants that the welds could be created. Creep rupture tests demonstrate that the weldment has similar (slightly reduced) lives compared to the parent metal, with reduced but still acceptable ductility. Noting the limitations of the small testing programme, weld strength factors of about 0.90 were derived with testing out to about 30kh [9]. Nevertheless, further testing on other heats to longer durations will be required to further qualify this manufacturing procedure and determine longer term WSFs.

Table 2. Materials Issues Identified in STP-NU-038 "ASME Code Consideration for the Intermediate Heat Exchanger (IHX)" Part 1Review of Current Experience on Intermediate Heat Exchanger

Sections	Issue	Comments
4.1.1.4 Material Degradation 4.1.2.1 Materials Issues	- 20 year lifetime of IHX of tubes ~2.3mm thick (helical tube) or ~1.5mm thick (plate) at risk due to lack of information on corrosion	- Long-term tests in representative environments required (Alloy 800H, Alloy 617) - Mitigation suggested: coating of long tubes/ weldments
6. Materials Candidates	- Alloy 617 identified by vendors as primary candidate with Alloy 230 (or Alloy X; .. XR) as alternatives	- Alloy 617B noted as future candidate - Alloys 230, … XR, incompletely characterised for (V)HTR - ODS alloys, ceramics mooted for the future
6.2 Materials Selection	- Creep Rupture - Use of tertiary creep for allowables - Creep Fatigue – relatively uncharacterised; could be more servere for compact IHX	- Availability of data on candidate alloys including use of Alloy 800H. - Implications for existing evaluations – revisions needed? - Creep fatigue interaction to be characterised for all candidate alloys
	- Weldability of Alloy 230 possible but difficult, especially in thicker section - Other candidates easier to weld, but need precautions to avoid hot cracking	- Weldments to be included in characterisation programmes
	- Microstructural stability to be studied - Alloy X /XR potentially most at risk of TCP phases - Alloys 617, 800H precipitate γ' with potential reduction in ductility	- Further studies needed; small associated risk of dimensional instability in γ' formers
	- Corrosion depends on exact He coolant composition	- Tests in coolant composition required to characterise any debit, especially for thin sections.
6.3.1 Acceptable Materials for Construction	- Only 6 materials (were) currently listed in Subsection NH; others to be included - Use of inelastic analysis permits tighter specifications to limit scatter/reduce safety margins	- Alloys 617, 230, X, XR require substantial testing/assessment for inclusion (see also STP-NU-035) - Tighter specifications yet to be defined.
6.3.2 Maximum Temperature	- Current 760°C limit to be addressed	- Work in other STP documents to extend Alloy 800H and 617 to higher temperatures; work on other alloys not yet published
6.3.3. Maximum Design Life	- 300kh current limit insufficient for commercial plant - Practical life of some components may be less (IHX possibly)	- Extension to 500kh required
6.3.4 Nonclassical Creep	- Creep shape not classical 3 region; needs some care in determining onset of tertiary creep.	- Consideration of separating time-independent and time-dependent plasticity may be advisable, requiring adaptations in the Code.
6.3.5 Environmental Effects	- Possible degradation due to carburisation, decarburisation and oxidation	- Effect on mechanical properties to be determined.
6.3.6 Possible New Failure Mode	- Reduction after long-term exposure of tensile and fracture toughness properties due to embrittlement	- Risk of such failure to be determined through testing



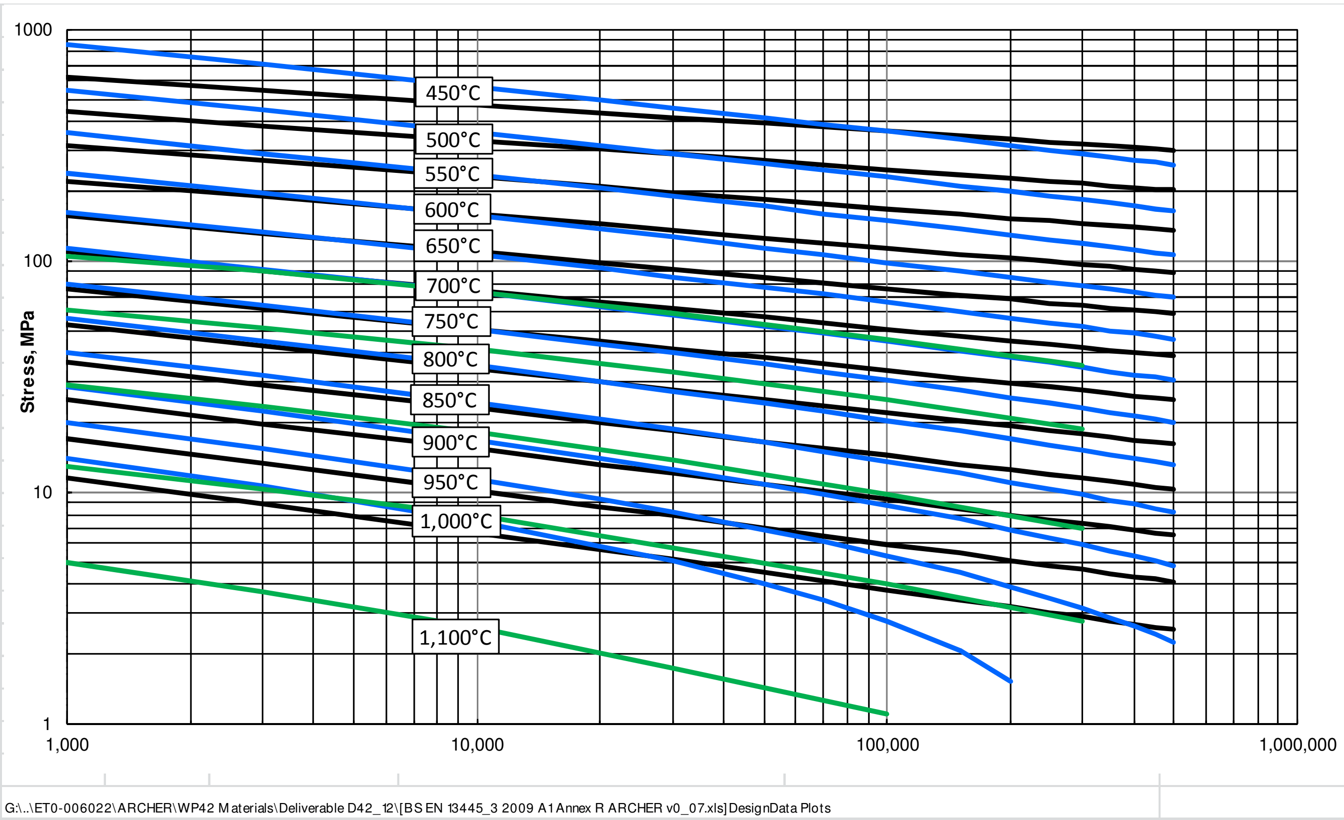


Figure 2. Comparison of the creep rupture mean strength of Alloy 800H predicted by KTA 3221.1[17], ECCC 2006 [27] and STP-NU-035 [34].
Upper – mean strength at specified durations vs temperature
Lower – mean strength at specified temperatures vs duration

3.1 General Gap Analysis

In preparing this section, the ASME documents STP-NU-045 has been referred to particularly, and considered in the light of the ARCHER results. Besides those matters already discussed above:

1 The matter of negligible creep will arise for several other components, not only the pressure boundary material. In particular, it will be inconvenient but necessary to declare different negligible creep temperatures for the same material when it is used for two different purposes. Take Type 316 as an example. When it is used for a shell material, it may be perfectly acceptable to declare a negligible creep temperature to permit it to be designed below that temperature without referring to creep or creep fatigue. On the other hand, some creep deformation may occur at that negligible creep temperature which is important eg for bolts and flanges, for bellows, and other components (including coated materials) where even very small or localised strains may be unacceptable. This topic warrants further study.

2 Rules for design and in service evaluation of flaws are recommended in STP-NU-045, and it is recommended that an international effort be brought to bear, since several specific developments have been made in for example UK and French codes. Validation of these methodologies and specific crack growth data on the materials of interest are required.

3 The safety case of key components and the efficiency of heat exchange surfaces depend greatly on the emissivity of materials, both in the virgin and after degradation in service in impure helium. Specific tests will be required.

4 Over a 60 year period substantial changes could take place microstructurally. Besides the ever present risk of precipitate phases causing loss of properties, other phenomena may occur. Some examples are the formation of γ' in Alloy 800H and Alloy 617 (more should form, and at lower temperatures). There is also a slight risk of long-range ordering of nickel based alloys that though unlikely in these particular alloys should be investigated. Steels may also suffer from α/α' decomposition at temperatures of 400℃ at long-times. Finally, the microstructural changes mentioned above are also often associated with dimensional changes. Work as part of the German AGH [42] has clearly shown contraction strains of 0.05% are possible (though this can approach 0.20% if long-range ordering occurs). Such small strains are generally irrelevant for many components; on the other hand they cannot be ignored where there is a constraint or in long pieces. It is suggested that such matters be studied by beginning long-term ageing of materials, whilst monitoring their strains periodically.

5 Several of the key materials that are likely to be used in future within the (V)HTR have only been tested to a few tens of thousands of hours in creep, and their creep fatigue interaction is currently unknown. Significant data development activities are necessary to bring them to a maturity to enable them to be listed in the design codes.

4 Conclusions and Recommendations

Two decades ago it would have been impossible to design a High Temperature Reactor (HTR) with helium up to 800℃ as the primary coolant for several reasons: the design codes had been withdrawn (KTA 3221) or were not sufficiently mature (ASME BPV, RCC-MRx). In most cases they were unable to include or reference properties of materials selected for the build; and the understanding of material behaviour was generally insufficient, or their upper temperature limit was too low for the (V)HTR for operation with primary coolant temperatures approaching 1,000℃. Since that time, significant advances have been made, with well-established roadmaps for development of both codes and materials properties. It now seems plausible that a demonstration HTR could be designed in the next few years, with the (V)HTR to follow shortly after.

Of course, there are some caveats, and the gaps in the materials data identified in this report need addressing, particularly for the (V)HTR. Some are simply of the type “this matter should be considered”, whilst others might require extensive testing programmes to mitigate any risk. The following are the key conclusions and recommendations.

[C1.] A *demonstration* HTR being built by one of the Gen-IV signatories, would no doubt be designed with an anticipated life of 20-30 years. However, *series construction* of HTRs requires a 60 year life, necessitating creep properties up to 500kh. Using the common rule that property data should be supported by testing to one-third of the duration, testing on several materials to 166kh are required. Extrapolation of shorter term data differs across the globe, and for Alloy 800H differences have been shown to exist, depending at least partly on assessment method. An alternative would be to design to shorter durations, but accept higher safety factors.

Risk Level: *medium* / Consequence: *medium*
Mitigation: (1) *Perform technical review, within 3 years, of creep assessment methods, especially extrapolation to 500kh; (2) Review long-term data gathering programmes on key alloys relating to the requirement for 500kh stress allowables.*

[C2.] The HTR primary pressure boundary is generally designed at conditions and with materials so that creep need not be considered, even during reactor excursions. However, on similar materials, parts of the non-nuclear industry are considering lower negligible creep temperatures, particularly to avoid creep *strain* rather than *rupture*. Nevertheless, in the pressure boundary creep-weakened areas will be present at weldments, and an increased understanding of the interaction between creep and fatigue requires a technical review in order to confirm earlier assumptions or raise mitigating actions.

Risk Level: *low* / Consequence: *high*
Mitigation: *Perform technical review, within 3 years, of creep at temperatures approaching or below current negligible creep temperatures occurring at weldments, or contributing to creep fatigue.*

[C3.] For the materials selected for the HTR, with coolant temperatures below 800°C, the methods of joining are generally well understood and characterised. Nevertheless, the properties of thin section fusion welds are not well understood at very long times. Furthermore, laser welding or other welding processes not currently covered by the design codes (eg required by compact plate IHX designs) need further characterisation; particularly to ensure reproducibility and to determine weld strength factors at appropriate durations for design.

Risk Level: *medium* / Consequence: *high*
Mitigation: (1) Characterise fusion welds at very long times; (2) *Perform further development and characterisation of alternative welding processes, to enable codification within 5-7 years.*

[C4.] In general, the reactor coolant may cause variously: oxidation, carburisation, decarburisation, and could have a significant effect on the mechanical properties of thin section components. The loss in section, and perhaps other microstructural effects, could be significant in the <3mm section tubes and ~1.5mm thickness plate in the different heat exchanger designs. It is likely that a debit in mechanical properties will result, requiring data gathering approaching 500kh; or to lesser times, if the component is likely to be replaced during the life of the reactor.

Risk Level: *medium* / Consequence: *medium*
Mitigation: *Within 2 years, commence exposures in reactor coolant of candidate alloys, determine the effective section reduction. Perform associated mechanical tests on coupons after exposure. Consider performing tests directly in flowing coolant.*

[C5.] A number of technical matters regarding the long-term behaviour of materials have also been raised, and are probably best understood by an understanding of underlying phenomena, using an academic approach. In this report we have discussed the evidence for: i) the likelihood of γ' precipitation in Alloy 800H and Alloy 617, at temperatures below their maximum use temperature, which might lead to dimensional instability (contraction) of up to 0.05%; ii) non-classical creep curve shapes in Alloy 617 causing difficulty identifying the onset of tertiary creep; iii) the presence of serrated yielding (common in solution strengthened alloys) and a possible detrimental effect on ductility; and iv) the risk of raising the ductile to brittle transition temperature following long term ageing of the pressure vessel materials.

Risk Level: *low* / Consequence: *medium*
Mitigation: *Commence a series of investigations into these phenomena, using appropriate academic institutions.*

5 References

[1]. M.A. Fütterer, F. Roelofs, N. Kohtz, P. Cuadrado Garcia, D. Knoche, D. Buckthorpe, Report ARCHER Deliverable D-12.31 Rev 3 : “Guidelines and Prioritization for Nuclear Cogeneration System Integration”, 2013

[2]. A. Pirscher, B. Scarlin, R. Vanstone, “Material Development and Mechanical Integrity Analysis for Advanced Steam Turbines”, Advances in Materials Technology for Fossil Power Plants: Proceedings from 5th Conference, Eds R. Viswanathan, D. Gandy, K. Coleman (Eds) EPRI, published by ASM International, 338-352, 2008.

[3]. M. Santella, J. Shingledecker and R. Swindeman. “Materials for advanced ultra-supercritical steam boilers, 24th Annual Conf on Fossil Energy Materials”, DOE, Pittsburgh, PA, (2010).

[4]. C. Bullough, D. Ponca, ARCHER Deliverable D-44.21 (Draft) : “Review of the availability of materials data for SG design above 600 °C steam”, 2015.

[5]. D. Ponca ARCHER Deliverable D-44.41, “Advanced Desgin Technologies” , 2014.

[6]. J. Berka, ARCHER Deliverable D-42.21 “Degradation of metallic materials in HTR helium coolant. Survey of the data generated in previous HTR and related programs”, 2012.

[7]. S. Dubiez-le-Goff, C. Bullough, ARCHER Deliverable D-43.51, “Review of existing data on Alloy 800H (focusing on thin sheet) and produce results applicable to mock-up and IHX.”, 2011

[8]. D. Ponca. ARCHER Deliverable D-44.23 “Manufacturing technologies for the SG concepts”, 2013.

[9]. G. Mutch, ARCHER Deliverable D-43.55, ‘Test report with determination of creep weld coefficients’ 2015.

[10]. ARCHER Deliverable D25.11, “Loading and Boundary Conditions for HTR Primary Pressure Boundaries, Experience from Licensing Processes”, 2013.

[11]. M. Hoffmann, K-H Herter, A. Klenk, N. Kohtz, ARCHER Deliverable D-25.21 “Gap Analysis” Integrity Concept LWR-HTR”, 2013.

[12]. S. Maloy et al, OECD Report LA-UR-13-28251, “Status Report on Structural Materials for Advanced Nuclear Systems”, 2013.

[13]. KTA 3021 “Komponenten des Primärkreises von Leichtwasserreaktoren” Parts 1 to 5, KTA Standards, Der Kerntechnische Ausschuss (KTA). (For further information contact via http://www.kta-gs.de, accessed January 2015)

[14]. G. Brinkmann, D. Vanvor, IAEA Meeting on Materials, AREVA GmbH Erlangen, Wien, 10.06. – 13.06.2014

[15]. ASME BPVC – American Society of Mechanical Engineers, Boiler Pressure Vessel Code, 2013 Edition. (Available from http://www.asme.org accessed January 2015.)

[16]. ASME, "Code Case N47-29, Class I Components in Elevated Temperature Service, Section III, Division 1," Cases of ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers, New York, December 1990'. (Available from http://www.asme.org. January 2015.)

[17]. KTA 3221 “Metallische HTR Komponenten” Parts 1 to 3, Draft Standard. KTA Standards, Der Kerntechnische Ausschuss (KTA). (For further information contact via http://www.kta-gs.de accessed January 2015)

[18]. EN 13445 – “Unfired pressure vessels - Part 3: Design” CEN Standard, 2012.

[19]. Afcen RCC-MRx Code - Design and Construction Rules for mechanical components of nuclear installations, 2012. (Available from http://www.afcen.com.)

[20]. R.L. Huddleston, R.W. Swindeman, Report NUREG/CR-5955 “Materials and design bases issues in ASME Code Case N-47”, 1993. (Available from Oak Ridge National Laboratory under ref ORNL/TM-12266.)

[21]. J. L. Boutard, A. Alamo, R. Lindau, M. Rieth “Fissile core and tritium-breeding blanket: Structural materials and their requirements”, C.R. Physique 9, 287-302, 2008.

[22]. R.Sims, STP-NU-045-1– 2012, “Roadmap to Develop ASME Code Rules for The Construction of High Temperature Gas Cooled Reactors (HTGRS)” ASME, 2010.

[23]. Anon., “NGNP High Temperature Materials White Paper”, Idaho National Laboratory Report INL/EXT-09-17187, June 2010.

[24]. D.K. Morton, R.I. Jetter, J.E. Nestell, T.D. Burchell, T.-L. Sham, “Section III, Division 5 – Development and Future Directions”, Paper PVP2012- 78062, Proceedings of the ASME 2012 Pressure Vessels & Piping Division Conference, PVP2012, July 15-19, 2012, Toronto, Ontario, Canada.

[25]. H. Nickel, F. Schubert and H. Schuster, Evaluation of Alloys For Advanced High-Temperature Reactor Systems” Nuclear Engineering and Design 78, 251-265, 1984.

[26]. H. Diehl and E. Bodmann, "Alloy 800: New Stress Rupture and Creep Data for Pressurized Components in High Temperature Reactors," Nuclear Engineering and Design, Vol. 119, pp. 195-206, 1990.
[27]. "ECCC Data sheets - Rupture strength, creep strength and relaxation strength values for carbon-manganese, low alloy ferritic, high alloy ferritic and austenitic steels, and high temperature bolting steels/alloys" Eds M.W. Spindler, D. Robertson, European Creep Collaborative Committee, Issue 2, Revision 001, May 2014.
[28]. Business Plan for CEN Workshop, CEN WS/64 Phase II, "Design and Construction Codes for Gen II to IV nuclear facilities (pilot case for process for evolution of AFCEN codes)" Document N008, July 2014. Further information from the CEN website at https://www.cen.eu/news/workshops/Pages/WS-2014-006.aspx (accessed January 2015).
[29]. C.K. Bullough, M. Norman, "Reducing engineering risk in power generation by using materials databases and data assessment procedures" 8th International Charles Parsons Turbine Conference, University of Portsmouth, Portsmouth, UK, 2011.
[30]. S. Holmström, N. Luzginova, "Negligible Creep of P91 Steel" ECCC 2014, ECCC Creep and Fracture Conference, Rome, Italy, 2014.
[31]. W. Ren T. Lillo, T. Totemeier, "Assessment of Negligible Creep, Off-Normal Welding and heat treatment of Gr91 Steel for nuclear reactor pressure vessel application", ORNL/GEN4/LTR-06-032, 2006.
[32]. R.W. Swindeman, M.J. Swindeman, B.W. Roberts, BE Thurgood, DL Marriott, Report STP-NU-020 "Verification of Allowable Stresses in ASME Section III Subsection NH for Alloy 800H", ASME, 2008.
[33]. M.K. Booker, V.B. Baylor, and B.L.P. Booker, Report ORNL / TM- 6029 "Survey of Available Creep and Tensile Data for Alloy 800H", Oak Ridge, Tennessee, 1978.
[34]. R. W. Swindeman, D. L. Marriott, J. R. Foulds, STP-NU-035, "Extend Allowable Stress Values for Alloy 800H", ASME, 2012
[35]. R. Wright, Presentation "VHTR Materials Overview Richard Wright Idaho National Laboratory" DOE-NE Materials Crosscut Coordination Meeting, July 2013.
[36]. J. K. Wright, L. J. Carroll, C. J. Cabet, T. Lillo, J. K. Benz, J. A. Simpson, W. R. Lloyd, J. A. Chapman, R. N. Wright "Characterization of Elevated Temperature Properties of Heat Exchanger and Steam Generator Alloys", HTR 2010 Conference Prage Paper 31, 2010. Also, Nuclear Engineering and Design, Vol 251, pp252-260, 2012.
[37]. EPRI "Creep-Fatigue Testing and Assessment Guideline" Technical Report 1019778, Dec 2010
[38]. R. Krein, A. Di Gianfrancesco, A. Gotti, P.Lombardi, W. Gamble, P. Barnard, "ECCC assessment of creep rupture strength for the new nickel alloy Alloy 617B", ECCC 2014, ECCC Creep and Fracture Conference, Rome, Italy, 2014.
[39]. R. Sims, STP-NU-045, "Roadmap to Develop ASME Code Rules for the Construction of High Temperature Gas Cooled Reactors (HTGRS)", ASME, 2010.
[40]. Business Plan for CEN Workshop, CEN WS/64 Phase II, "Design and Construction Codes for Gen II to IV nuclear facilities (pilot case for process for evolution of AFCEN codes)" Document N008, July 2014. Further information from the CEN website at https://www.cen.eu/news/workshops/Pages/WS-2014-006.aspx (accessed January 2015).
[41]. ECCC recommendations – Volume 5 Part IIb [Issue 1] Guidance for the assessment for Creep Rupture, Creep Strain and Stress Relaxation data, recommendations for the assessment of weld creep-rupture data, 0509/MC/102 [Issue 1], 1105/01.
[42]. I. Steller, W. Bendick, T-U. Kern, K. Maile, M. Monsees & A. Scholz "60th anniversary of the German Creep Committee" ECCC 2009, ECCC Creep and Fracture Conference, Dubendorf, Switzerland, 2009.

6 Abbreviations

AFCEN	Association Française pour les règles de Conception, de construction et de surveillance en exploitation des matériels des Chaudières Electro Nucléaires. French Association for Design, Construction and In-Service Inspection Rules for Nuclear Island Components.
COST	European Cooperation in Science and Technology
ECCC	European Creep Collaborative Committee
ENEF	European Nuclear Energy Forum
EUROPAIRS	EU Project, End User Requirements for Industrial Process Heat Applications with Innovative Nuclear Reactors for Sustainable Energy Supply. Website: http://www.europairs.eu/
HTR	High Temperature Reactor (typically primary coolant up to 800°C)
(V)HTR	Very High Temperature Reactor (typically primary coolant up to 1,000°C)
KTA	Kerntechnischer Ausschuss, the German Nuclear Safety Standards Commission. Website: http://www.kta-gs.de/welcome_engl.htm (Accessed January 2015)
NNSA	National Nuclear Safety Administration (Chinese Regulatory Authority).
RCC-M	(AFCEN) RCC-M: Design And Conception Rules For Mechanical Components of PWR Nuclear Islands
RCC-MR	(AFCEN) RCC-MR: Design And Construction Rules for Mechanical Components of Nuclear Installations Applicable To High Temperature Structures and to the ITER Vacuum Vessel
RCC-MRx:	(AFCEN) RCC MRx: Design And Construction Rules For Mechanical Components Of Nuclear Installations Applicable To High Temperature Structures and to the ITER Vacuum Vessel
SNETP	Sustainable Nuclear Energy Technology Platform

Appendix - ASME Boiler and Pressure Vessel Code - the creation of Section III Division 5

D Buckthorpe, AMEC Foster Wheeler

Development of BPV III, Division (III-5)

Within ASME, a new BPV III, Division 5 (III-5) is being developed for high temperature reactors that include both HTGRs and LMRs. This part of the ASME Code provides rules for the design, fabrication, inspection and testing of components exceeding the temperature in Division 1 and are meant for components experiencing temperatures that are equal to or less than 700°F (370°C) for ferritic materials or 800°F (425°C) for austenitic stainless steels or high nickel alloys. This Division 5 also contains the new rules for graphite core components which include general requirements, plus design and construction rules. Irradiation effects on graphite are addressed, as are the features of probabilistic design reflected in the determination of graphite material strength properties. This first edition reflects new, safety-criteria approaches for nuclear power plants with the rules conveniently placed into one, single book format. These rules have been updated and improved over those still in Code Case format. This new Division 5 has an issuance date of November 1st 2011and is part of the 2011 Addenda to the 2010 Edition of the BPV Code.

The layout of Division 5 is one of an editorial compilation of existing Subsections and Code Cases addressing both elevated temperature components and conventional operation below the creep regime. Subsections NB, NC, NF, NG and NH of the ASME III Code are referenced, also the Code Cases for elevated temperature components and the graphite rules. The Division creates three classifications for Systems, Structures and Components (SSCs):

(a) Safety-related
(b) Non-safety related with special treatment
(c) Non-safety related

SSCs that are governed by rules in Division 5 are similar to Class 1 or Class CS rules, and designated as Class A rules in Division 5. SSCs that are deemed to be not safety-related but which require special treatment (alternative safety function capability and contribute to defence in-depth) are addressed by rules that are very similar to Division 1, Class 2 rules (using either design-by-analysis or design-by rule approach as appropriate). These are designated as Class B rules in Division 5. Non-safety related SSCs are treated like balance-of plant items in the current LWR fleet, following the rules of Section VIII of the ASME BPV Code, ASME B31, etc. This classification system currently applies to both HTGRs and LMRs. The code structure for ASME Division 5 is shown below in Table A1 and sections that it refers to and incorporates from are shown in Table A2.

Table A1 - Code Structure for ASME BPVC Section III, Division 5

Class	Subsection	Subpart	Callout	Title	Scope
-	HA	-	-	General Requirements	
Class A & B		A	*HAA*	*Metallic Materials*	Metallic
Class A		B	*HAB*	*Graphite Materials*	Graphite
Class A		C	*HAC*	*Composite Materials*	Composites
Class A	HB	-	-	Metallic Pressure Boundary Components	Metallic
		A	*HBA*	*Low Temperature Service*	
		B	*HBB*	*Elevated Temperature Service*	
Class B	HC	-	-	Metallic Pressure Boundary Components	Metallic
		A	*HCB*	*Low Temperature Service*	
		B	*HCB*	*Elevated Temperature Service*	
Class A & B	HF	-	-	Metallic Supports	Metallic
		A	*HFA*	*Low Temperature Service*	
Class A	HG	-		Metallic Core Support Structures	Metallic
		A	*HGA*	*Low Temperature Service*	
		B	*HG*	*Elevated Temperature Service*	
Class A	HH	-	-	Non-Metallic Core Support Structures	
		A	*HHA*	*Graphite Construction Rules*	Graphite
		B	*HHB*	*Composites Construction Rules*	Composites
Class A & B	Appendices	None	*Appendices*	Mandatory and Non-mandatory Appendices	All

Notes:
1. Subsection ID has three characters for referencing flexibility to address low / elevated temperature concerns or varying materials under the same Subsection.
2. Two Section III classes (A and B) were chosen for high temperature reactors, reflecting (1) safety-related and (2) non-safety related with special treatment classification categories, respectively, for SSCs. Non-safety related SSCs will use non-nuclear standards such as BPV Section VIII, B31, etc.
3. Appendices may be generally applicable (as shown above) or Subsection specific and attached to those subsections (like Subsection NH).

Table A2 - Sources for ASME III BPVC Division 5 Rules

Division 5 Subsection	Refers to Rules in	Incorporates Rules From
Subsection HA General Requirements		
Subpart A – Metallic	Subsection NCA	
Subpart B – Graphite		New Rules
Subpart C – Composite	TBD	TBD
Subsection HB – Class A Metallic		
Subpart A – Low Temperature	Subsection NB	
Subpart B – Elevated Temperature	Subsection NH	
Appendix HBB-I		CC N-499-2
Subsection HC – Class B Metallic		
Subpart A – Low Temperature	Subsection NC	
Subpart B – Elevated Temperature		
HCB-1000		CC N-253-14
HCB-2000		CC N-253-14
HCB-3000		CC N-253-14
HCB-4000		CC N-254
HCB-5000	Subsection NC-5000	
HCB-6000		CC N-467
HCB-7000		CC N-257
Appendix HCB-I		CC N-253-14, Appendix B
Appendix HCB-II		CC N-253-14, Appendix C
Appendix HCB-III		CC N-253-14, Appendix E
Subsection HF – Supports		
Subpart A – Low Temperature	Subsection NF	
Subsection HG – Class A Metallic CSS		
Subpart A – Low Temperature	Subsection NG	
Subpart B – Elevated Temperature		Revised CC N-201, Part B
Appendix HGB-I		Revised CC N-201, Part B, Appendix Y
Appendix HGB-II		Revised CC N-201, Part A
Appendix HGB-III		Revised CC N-201, Part A, Appendix XVIII
Appendix HGB-IV		Revised CC N-201, Part A, Appendix XIX
Subsection HH – Class A Non-metallic CSS		
Subpart A – Graphite		
Subpart B – Composite	TBD	

Notes: CC – Code Case; TBD – to be developed; CSS – Core Support Structures