ASME NTB-3-2020

Gap Analysis for Addressing Adequacy or Optimization of ASME Section III, Division 5 Rules for Metallic Components

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GAP ANALYSIS FOR ADDRESSING ADEQUACY OR OPTIMIZATION OF ASME BPVC SECTION III, DIVISION 5 RULES FOR METALLIC COMPONENTS

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FOREWORD

The goal of this publication is to provide an assessment of an integrated list of 39 issues that have been assembled from three prior reviews of various forerunners of ASME Boiler and Pressure Vessel Code ("BPVC"), Section III *Rules for Construction of Nuclear Facility Components*, Division 5 *High Temperature Reactors*, Code rules for metallic coolant boundary components and core supports. The focus of the assessment has been on whether the current BPVC Section III, Division 5 Code rules provide reasonable assurance of adequate protection against identified structural failure modes with respect to these issues.

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EXECUTIVE SUMMARY

This gap analysis report provides an assessment of an integrated list of 39 issues assembled from three prior reviews of various forerunners of BPVC Section III, Division 5 Code rules for metallic coolant boundary components and core supports. The focus of the assessment has been on whether the current BPVC Section III, Division 5 Code rules provide reasonable assurance of adequate protection against identified structural failure modes with respect to these issues. If gaps are identified, an attempt was made to assess whether they are related to the adequacy or the optimization of the Code rules. For example, extension to cover longer design lifetimes that does not affect the adequacy of the underlying technical basis of the rules corresponds to optimization. The 39 gaps and issues evaluated in this report originated from four references: O'Donnell, Hull and Malik's 2008 paper [1]; O'Donnell and Griffin's 2007 STLLC STP-NU-010 report [2]; the 1985 paper by Griffin [3]; and the 1993 Nuclear Regulatory Commission (NRC) NUREG report by Huddleston and Swindeman [4].

The comprehensive STP-NU-010 report by O'Donnell and Griffin [2] itemized 25 items that are listed in Table 1. Each of the items is assigned a distinct item number for ease of reference. O'Donnell, Hull and Malik's paper in 2008 [1] summarized these issues and is largely based on the STP-NU-010 report [2].

Table 1: Safety Issues for Structural Design of Very High Temperature Reactors and Gen IV Systems in O'Donnell, Hull and Malik 2008 [1] and O'Donnell and Griffin 2007 [2]

Original Item number	Item title	Corresponding Issue number in this report
OG- 1	Transition joints	**III.1**
OG- 2	Weld residual stresses	**III.2**
OG- 3	Design loading combinations	**VII. 1**
OG- 4	Creep-rupture and fatigue damage	**I.4**
OG- 5	Simplified bounds for creep ratcheting	**I.2**
OG- 6	Thermal striping	**I.8**
OG- 7	Creep-fatigue analysis of Class 2 and 3 piping	**I.5**
OG- 8	Are limits of Case N-253 for elevated-temperature Class 2 and 3 components met?	**VI.1**
OG- 9	Creep buckling under axial compression – design margins	**I. 11**
OG- 10	Identify areas where Appendix T rules are not met	**I.1**
OG- 11	Rules for component supports at elevated-temperature	**V.1**
OG- 12	Strain and deformation limits at elevated-temperature	**I.3**
OG- 13	Evaluation of weldments	**III.3**
OG- 14	Material acceptance criteria for elevated-temperature	**II. 2**
OG- 15	Creep-rupture damage due to forming and welding	**II. 1**
OG- 16	Mass transfer effects	**VII. 2**
OG- 17	Environmental effects	**VII. 3**
OG- 18	Fracture toughness criteria	**VII. 5**

Original Item number	Item title	Corresponding Issue number in this report
OG- 19	Thermal aging effects	**II. 4**
OG- 20	Irradiation effects	**VII. 4**
OG- 21	Use of simplified bounding rules at discontinuities	**I. 9**
OG- 22	Elastic follow-up	**I.6**
OG- 23	Design criteria for elevated-temperature core support structures and welds	**V.2**
OG- 24	Elevated-temperature data base for mechanical properties	**II. 3**
OG- 25	Basis for leak-before-break at elevated temperatures	**VII. 6**

Note: The issues were not ranked in any particular order by the authors.

In the 1985 paper by Griffin "Elevated Temperature Structural Design Evaluation Issues in LMFBR Licensing" [3], nine issues associated with elevated temperature structural design identified by the NRC licensing review of Clinch River Breeder Reactor Plant (CRBRP) for a construction permit were described. These nine items are listed in Table 2. It was noted in the paper that "the design criteria and basic approach to design evaluation were accepted and that no major inadequacies were discovered."

Table 2: Elevated Temperature Structural Design Issues in Liquid Metal Fast Breeder Reactor Licensing Identified by Griffin [3]

Original Item number	Item title	Corresponding Issue number in this report
G-3. 1	Weldment Safety Evaluation	**III.2; III.3**
G-3. 2	Notch Weakening	**IV.3**
G-3. 3	Design Analysis Methods, Codes and Standards	**II. 7**
G-3. 4	Steam Generator	**I. 10**
G-4. 1	Elevated-temperature seismic effects	**VI. 5**
G-4. 2	Elastic follow-up in elevated-temperature piping	**I.6**
G-4. 3	Creep-fatigue evaluation	**I.5**
G-4. 4	Plastic strain concentration factors	**I.7**
G-4. 5	Intermediate heat transport system transition weld	**III.1**

Note: The issues were not ranked in any particular order by the authors.

In addition, 23 items were identified in the 1993 NRC NUREG report by Huddleston and Swindeman [4]. The perspective of the Huddleston-Swindeman report is somewhat different in that it is intended "to identify any code design basis issues that could negatively impact (delay) the design certification process." Many of the identified issues are taken from Volume 1 of the four-part Welding Research Council (WRC) series edited by A. K. Dhalla, *Recommended Practices in Elevated Temperature Design* [5], which is discussed in more detail in the ASME NTB-2-2019 report *Background Information for Addressing Adequacy or Optimization of ASME BPVC Section III, Division 5 Rules for Metallic Components* [6]. The WRC report is quoted extensively in the Huddleston-Swindeman report. However, in terms of reactor types considered, the scope of Huddleston-Swindeman [4] is much broader and the operating conditions potentially more demanding. Also, significantly, the full list of issues is narrowed to ten major issues in the report. The 23 items identified by Huddleston-Swindeman are listed in Table 3, and those identified as major issues by the authors are marked.

Table 3: Material and Design Bases Issues in ASME Code Case N-47 Identified by Huddleston and Swindeman [4]

Original Item number	Item title	One of the 10 major issues	Corresponding Issue number in this report
HS- 1	Lack of Material Property Allowable Design Data/Curves for 60-Year Design Life	Yes	**II. 2; II. 3**
HS- 2	Degradation of Material Properties at High Temperatures due to Long-Term Irradiation	Yes	**VII. 4**
HS- 3	Degradation of Material Properties due to Long-Term Thermal Aging	No	**II. 4**
HS- 4	Degradation of Material Properties due to Corrosion Phenomena	Yes	**VII. 3**
HS- 5	Lack of Property Allowables Based on Current Melting and Fabrication Practices	No	**II. 2; II. 3**
HS- 6	Degradation Effect of Small Cyclic Stresses	No	**VI.2**
HS- 7	Creep Induced Failures at Temperatures Below CCN47 Limits	No	**VI.3**
HS- 8	Use of Average vs Minimum Material Properties in Design	No	**II. 6**
HS- 9	Lack of a Design Methodology for Modified 9Cr-1Mo Steel	No	**II. 5**
HS- 10	Lack of Understanding/Validation of Effects of Short-Term Overload Events on Subsequent Mechanical Properties	No	**VI. 4**
HS- 11	Lack of Validated Thermal Striping Materials and Design Methodology	Yes	**I. 8**
HS- 12	Lack of Reliable Creep-Fatigue Design Rules	Yes	**I.5**
HS- 13	Difficult, Overly Conservative Ratcheting Design Rules	No	**I.2**
HS- 14	Lack of a Validated Weldment Design Methodology	Yes	**III.2; III.3**
HS- 15	Lack of Flaw Assessment Procedures	Yes	**VII. 7**
HS- 16	Uncertainty of Multiaxial Stress State Effect	No	**IV.1**
HS- 17	Uncertainty of Nonradial (Nonproportional) Loading Effect	No	**IV.2**
HS- 18	Lack of Understanding/Validation of Notch Weakening Effects	Yes	**IV.3**
HS- 19	Lack of Conservatism in Code Rules for Simplified Fatigue Evaluations Based on Plastic Strain Concentration Factors	No	**I.7**
HS- 20	Lack of Validated Rules/Guidelines to Account for Seismic Effects at Elevated, Temperature	Yes	**VI. 5**
HS- 21	Lack of Inelastic Design Procedures for Piping	Yes	**II. 7**
HS- 22	Overly Conservative Buckling Rules	No	**I. 11**
HS- 23	Need for Thermal Stratification Design Guidelines	No	**VI. 6**

Note: The issues were not ranked in any particular order by the authors.

There is considerable overlap in the identified issues among these sources [1], [2], [3], [4]. Combining all the items from these references, a total of 39 issues are recognized and are further grouped into 6 categories: (1) Issues Relating to Strain, Deformation, and Fatigue Limits; (2) Issues Relating to Materials Properties; (3) Welds and Core Supports; (4) Multiaxiality; (5) Miscellaneous Issues; and (6) Issues Outside of Division 5 Scope. Table 4 lists these 6 categories and the issues under each category. The origins of these issues are also identified in Table 4.

Table 4: Issues for Assessing the Adequacy or Optimization of the Current BPVC Section III, Division 5 Rules and Code Cases in Construction of High-Temperature Reactors

Issue Number	Origin of Issue*	Issue Title
Category I: Relating to Strain, Deformation, and Fatigue Limits		
Issue I. 1	OG-10	Identify areas where Appendix T rules are not met
Issue I. 2	OG-5, HS-13	Simplified bounds for creep ratcheting
Issue I. 3	OG-12	Strain and deformation limits at elevated-temperature
Issue I. 4	OG-4	Creep-rupture and fatigue damage
Issue I. 5	OG-7, G-4.3, HS-12	Creep-fatigue analysis of Class 2 and 3 piping
Issue I. 6	OG-22, G-4.2	Elastic follow-up
Issue I. 7	G-4.4, HS-19	Plastic strain concentration factors/Lack of Conservatism in Code Rules for Simplified Fatigue Evaluation Based on Plastic Strain Concentration Factors
Issue I. 8	OG-6, HS-11	Thermal striping
Issue I. 9	OG-21	Use of simplified bounding rules at discontinuities
Issue I. 10	G-3.4	Steam generator tubesheet evaluation
Issue I. 11	OG-9, HS-22	Creep buckling under axial compression – design margins
Category II: Relating to Materials Properties		
Issue II. 1	OG-15	Creep-rupture damage due to forming and welding
Issue II. 2	OG-14, HS-1, HS-5	Material acceptance criteria for elevated-temperature
Issue II. 3	OG-24, HS-1, HS-5	Elevated-temperature data base for mechanical properties
Issue II. 4	OG-19, HS-3	Thermal aging effects
Issue II. 5	HS-9	Lack of a design methodology for Modified 9Cr-1Mo steel
Issue II. 6	HS-8	Use of average vs. minimum material properties in design
Issue II. 7	G-3.3, HS-21	Material property representation for inelastic analysis/Lack of inelastic design procedures for piping
Category III: Welds		
Issue III. 1	OG-1, G-4.5	Transition joints
Issue III. 2	OG-2, G-3.1, HS-14	Weld residual stresses
Issue III. 3	OG-13, G-3.1, HS-14	Evaluation of weldments
Category IV: Multiaxiality		
Issue IV. 1	HS-16	Uncertainty of multiaxial stress state effects
Issue IV. 2	HS-17	Uncertainty of non-radial (non-proportional) loading
Issue IV. 3	G-3.2, HS-18	Notch weakening/Lack of understanding/validation of notch weakening effects
Category V: Components and Core Supports		
Issue V. 1	OG-11	Rules for component supports at elevated-temperature
Issue V. 2	OG-23	Design criteria for elevated-temperature core support structures and welds
Category VI: Miscellaneous Issues		
Issue VI. 1	OG-8	Are limits of Case N-253 for elevated-temperature Class 2 and 3 components met?
Issue VI. 2	HS-6	Degradation effect of small cyclic stresses
Issue VI. 3	HS-7	Creep-induced failures at temperatures below Code Case N-47 limits
Issue VI. 4	HS-10	Lack of understanding/validation of effects of short term overload events on subsequent material properties.
Issue VI. 5	G-4.1, HS-20	Elevated-temperature seismic effects/Lack of validated rules/guidelines to account for seismic effects at elevated temperature
Issue VI. 6	HS-23	Need for thermal stratification guidelines

Issue Number	Origin of Issue*	Issue Title
Category VII: Issues Outside of Division 5 Scope		
Issue VII. 1	OG-3	Design loading combinations
Issue VII. 2	OG-16	Mass transfer effects
Issue VII. 3	OG-17, HS-4	Environmental effects
Issue VII. 4	OG-20, HS-2	Irradiation effects
Issue VII. 5	OG-18	Fracture toughness criteria
Issue VII. 6	OG-25	Basis for leak-before-break at elevated temperatures
Issue VII. 7	HS-15	Lack of flaw assessment procedures

Note: (1) * Origin of Issue refers to the item numbers in Table 1, Table 2 and Table 3.
(2) The issues are not ranked in any particular order.

The issues discussed in this report, as well as those identified in the four references, were not ranked in any particular order by the authors.

The discussions of these 39 issues in this document use the O’Donnell, Hull and Malik 2008 paper [1] as the baseline, and the text explaining the issues is taken from that report. Duplication from the Griffin and Huddleston-Swindeman document is noted and discussed as appropriate. Additional issues beyond those identified by O’Donnell, Hull and Malik [1] are discussed.

In addition to the Summary of each tabulated issue, there is a General Assessment of its significance, Material Specific Remarks as applicable, Required Actions, if any, and Conclusions regarding gaps or actions addressing the adequacy or optimization of BPVC Section III, Division 5. Thus, for each tabulated issue, there are five subheadings expanding on the issue and its determination from a gap analysis perspective.

ABBREVIATIONS AND ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
CC	Code Case
CRBR	Clinch River Breeder Reactor
CRBRP	Clinch River Breeder Reactor Plant
DMW	Dissimilar Metal Weld
DOE	Department of Energy
EPP	Elastic Perfectly Plastic
FEA	Finite Element Analysis
FFTF	Fast Flux Test Facility
Gr91	Modified 9Cr-1Mo
HTGR	High Temperature Gas-cooled Reactor
HTR	High Temperature Reactor
ISI	In-Service Inspection
JSME	Japan Society of Mechanical Engineers
LMFBR	Liquid Metal Fast Breeder Reactor
LWR	Light Water Reactor
MHTGR	Modular High Temperature Gas-cooled Reactor
N/A	Not Applicable
NGNP	Next Generation Nuclear Plant
NIMS	National Institute for Materials Science
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
R&D	Research and Development
SMT	Simplified Model Test
STLLC	ASME Standards Technology, LLC
UK	United Kingdom
VHTR	Very High Temperature Reactor
WRC	Welding Research Council

1 RELATING TO STRAIN, DEFORMATION, AND FATIGUE LIMITS

1.1 Issue I. 1 – Identify Areas Where Appendix T Rules are not Met

1.1.1 Summary

O'Donnell, Hull and Malik [1] summarized this issue in Item OG- 10 as follows:

> Appendix T in NH [7] provides three expressions for determining strain range [8] using elastic analysis and, if these rules cannot be satisfied, additional rules are provided, presumably less conservative, based on the results of inelastic analyses which require detailed constitutive models of material behavior under time varying loading conditions. For the CRBR, these behavioral models were a contractual provision based on RDT Standards. These applicable standards are no longer maintained and there have been numerous technical developments in this area since then [9]. Appendix T rules cover strain, deformation, creep and fatigue limits at elevated temperatures for 304SS/316 SS (816°C), Alloy 800H (760°C), 2.25Cr-1Mo (593°C), 9Cr-1Mo-V (649°C). Development of material models for materials not currently covered or for temperatures beyond their original range of verification will be a considerable effort. Modifications in Appendix T rules for higher temperatures and additional materials (e.g., Alloy 617, Hastelloy X/XR) may be needed.

1.1.2 General Assessment

Appendix HBB-T of BPVC Section III, Division 5 provides procedures to evaluate strain limits and creep fatigue damage using elastic analysis. Alternatively, for some Class A materials, recent code cases provide procedures based on Elastic-Perfectly plastic (EPP) analysis. If these rules cannot be satisfied, additional rules are provided which are based on results of inelastic analyses. However, inelastic analysis requires detailed constitutive models of material behavior under time varying loading conditions. For the CRBR, these behavioral models were based on Nuclear Standard NE F9-5T. This standard is no longer maintained, and numerous technical developments have been made since. However, ASME has established a working group to develop inelastic analysis methods and constitutive models for Class A materials for incorporation into BPVC Section III, Division 5. Models for several materials have been developed and are ready for ASME Code action [10], and the development process continues in the working group for the remaining materials.

1.1.3 Material Specific Remarks

None.

1.1.4 Action Required

(1) Complete the extension of the EPP methods to the remaining Class A materials, and (2) proceed to ballot the recommended constitutive equations developed in the Working Group on Analysis Methods.

1.1.5 Conclusion

Categorized as Optimization.