Initial development for the introduction of provisions for irradiation effects in Division 5 rules

Applied Materials Division

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ABSTRACT

Heat pipe microreactor concept involves the use of a solid core block to encapsulate the nuclear fuel and the passive extraction of heat from the core block to the heat exchanger block by advanced heat pipes. The core block is a key component of the heat pipe microreactor design. It integrates the functions of reactor vessel, core structural components and fuel cladding of traditional Gen IV advanced reactor designs into one single structure.

The effects of irradiation on the structural failure modes are outside the scope of the rules of the ASME Boiler and Pressure Vessel Code, Section III, Division 5, Subsection HB, Subpart B for metallic Class A components. Methods to account for the irradiation effect were investigated in order to develop design methods that can be applied to the core block design in the heat pipe microreactor.

This report presents an initial development of an approach to account for the irradiation effects in the design rules.

TABLE OF CONTENTS

Abstract	i
Table of Contents	iii
List of Figures	v
List of Tables	vii
1 Introduction	1
2 ASME Section III, Division 5	3
2.1 Subsection HB, Subpart B	4
3 Effects of Neutron Irradiation	9
3.1 Design Rules to Address Irradiation	10
4 Summary	13
Acknowledgements	15
Bibliography	17

LIST OF FIGURES

Figure 1.1: (a) Cross-section of a hexagonal core, (b) cross-section of a segmented block showing the fuel holes (solid color) and heat pipe holes (white), (c) cross-sectional view of the thin ligaments between holes. (Part (b) is taken from Ref. [1]).....	1
Figure 2.1: Flow diagram for elevated temperature analysis.	5
Figure 2.2: Sketch of a creep-fatigue interaction diagram. The green point denotes satisfactory creep-fatigue performance. The red point corresponds to failing the creep-fatigue check.	6
Figure 3.1: Approximate temperature regimes where different phenomena caused by irradiation are most prominent in structural alloys, schematic after [2]. The highlighted regions correspond to room temperature to 700F and 700 to 1200F for Grade 91 steel.....	9

LIST OF TABLES

Table 2.1: ASME Section III, Division 5 organization.....	3
Table 2.2: Structural failure modes for Division 5 Class A components.....	4
Table 3.1: Operative temperature ranges for Grade 91 irradiation degradation mechanisms.....	10

1 Introduction

Heat pipe microreactor concept involves the use of a solid core block to encapsulate the nuclear fuel and the passive extraction of heat from the core block to the heat exchanger block by advanced heat pipes. The heat is then further transferred to the energy conversion system to produce electricity or process heat. This results in a compact reactor configuration with minimal moving parts and enables inherent power regulation, autonomous operation and inherit load following. It is well suited for delivering economical and sustainable power to remote locations.

The core block is a key component of the heat pipe microreactor design. It integrates the functions of reactor vessel, core structural components and fuel cladding of traditional Gen IV advanced reactor designs into one single structure. Some designs employ ceramic materials, e.g., graphite, as the construction material for the core block and the heat exchanger block, while others employ metallic materials. This study is focused on the use of alloys as the construction materials for the core block and heat exchanger block in a heat pipe microreactor design.

A sketch of a typical core block configuration is shown in Figure 1.1. It consists of a segmented hexagonal core that have many holes aligned in the axial direction. There are typically two types of holes, those that contain the nuclear fuel - the fuel holes; and those that the evaporator section of the heat pipes are inserted into - the heat pipe holes. The design of the spacing and configuration of these holes depend on a number of factors, such as neutronics and thermal output.

Figure 1.1: (a) Cross-section of a hexagonal core, (b) cross-section of a segmented block showing the fuel holes (solid color) and heat pipe holes (white), (c) cross-sectional view of the thin ligaments between holes. (Part (b) is taken from Ref. [1])

The heat exchanger block is constructed of the same material as the core block. It also has axial holes, but just for the heat pipes. There are no fuel holes in the heat exchanger block. The heat pipe holes in the heat exchanger block are aligned with those in the core block for the placement of the heat pipes. The evaporator section of the heat pipes resides in the core block and the condenser

section in the heat exchanger block. The way that heat is transferred from the fuel to the heat exchanger through the heat pipes is as follows.

The heat generated by the nuclear fuel in the fuel holes of the core block is conducted through the webs, or the ligaments, between the fuel holes and the heat pipe holes to the evaporator section of the heat pipes. The heat would vaporize the working fluid, typically sodium, in the evaporator section of the heat pipes. The sodium vapor flows through the adiabatic region of the heat pipes to the condenser section that resides in the heat exchanger block. As heat is exchanged between the heat pipes and the heat exchanger block, the sodium vapor inside the heat pipes is condensed back to liquid sodium and wicked back to the evaporator section through capillary action. This heat transport mechanism does not involve piping loops and pumps and the inventory of sodium is very small for each heat pipe.

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 5 provides construction rules for structural components in high temperature reactors. As the geometry and the irradiation environment of a core block in a heat pipe microreactor are quite different from, and more challenging than, structural components in traditional Gen IV high temperature reactor systems, a specific consideration through an ASME nuclear code case would be an effective approach to address the needs.

Type 316 stainless steel is the reference construction material for the core block and the heat exchanger block for some heat pipe microreactor design concepts. The identification of additional candidate construction materials would expand the design envelope for higher safety margins and for the exploration of more innovative and economical designs. Since Grade 91 has lower thermal expansion coefficient and higher heat conductivity as compared with Type 316 stainless steel, it would result in lower thermal stresses developed in the core block and heat exchanger block during reactor operations. It also has higher resistance to radiation induced swelling. In order to support the use of Grade 91 in heat pipe microreactors, an integrated plan was established to develop an ASME Section III, Division 5 Code Case for Grade 91 steel that addresses the specific requirements of core block designs, and the design evaluation methods to support the use of the design criteria in the new code case to conduct trade study and design optimization.

In the following sections, the ASME Section III, Division 5 design rules that are applicable to the core block are reviewed. Since environmental effects on structural failure modes, particularly those involved with neutron irradiation, are outside the ASME Code space, other codes that address irradiation are surveyed to gain an insight on the needed considerations. Analysis methods that include the irradiation effects will also be discussed.

2 ASME Section III, Division 5

The Energy Policy Act of 2005 passed by the Congress established the Next Generation Nuclear Plant (NGNP) Program to demonstrate the generation of electricity and/or hydrogen with a high-temperature nuclear energy source. The Idaho National Laboratory (INL) was the lead Department of Energy (DOE) Lab responsible for the execution of the program. In order to reduce regulatory risks for the licensing of the NGNP reactor plant with U.S. Nuclear Regulatory Commission (NRC), it was deemed desirable to streamline the applicable Codes and Standards. INL requested ASME to develop a new Division within Section III to consolidate all high temperature construction rules and to add graphite rules to the new Division to support NGNP. ASME pursued the development effort and issued a new Section III, Division 5 for high temperature reactors in the 2011 Addenda of the ASME Boiler and Pressure Vessel Code.

The ASME Boiler and Pressure Vessel Code, Section III, Division 5 provides rules that govern the construction of vessels, piping, pumps, valves, supports, core support structures and nonmetallic core components for use in high temperature reactor systems and their supporting systems. Construction, as used here, is an all-inclusive term that includes material, design, fabrication, installation, examination, testing, overpressure protection, inspection, stamping, and certification. High temperature reactors covered by Section III, Division 5 include gas-cooled reactors, liquid metal reactors and molten salt reactors with liquid fuel or with solid fuel. The organization of Section III, Division 5 is shown in Table 2.1.

Table 2.1: ASME Section III, Division 5 organization.

Class	Sub-section	Subpart	Subsection ID	Title	Scope
General Requirements					
Class A, B & SM	HA	A	HAA	Metallic Materials	Metallic
Class SN		B	HAB	Graphite and Composite Materials	Nonmetallic
Class A Metallic Pressure Boundary Components					
Class A	HB	A	HBA	Low Temperature Service	Metallic
Class A		B	HBB	Elevated Temperature Service	Metallic
Class B Metallic Pressure Boundary Components					
Class B	HC	A	HCA	Low Temperature Service	Metallic
Class B		B	HCB	Elevated Temperature Service	Metallic
Class A and Class B Metallic Supports					
Class A & B	HF	A	HFA	Low Temperature Service	Metallic
Class SM Metallic Core Support Structures					
Class SM	HG	A	HGA	Low Temperature Service	Metallic
Class SM		B	HGB	Elevated Temperature Service	Metallic
Class SN Nonmetallic Core Components					
Class SN	HH	A	HHA	Graphite Materials	Graphite
Class SN		B	HHB	Composite Materials	Composite

The construction rules for Division 5 Class A components provide a suitable starting point for the design of metallic core block of a heat pipe microreactor. For low temperature service, the rules for Class A components are found in Subsection HB, Subpart A. For temperature service above

700F for ferritic steels and 800F for austenitic alloys, the rules for Class A components are in Subsection HB, Subpart B (HBB).

2.1 Subsection HB, Subpart B

The HBB construction rules have had a long history of development. They were developed specifically for high temperature reactors. In the 1970s, the U.S. NRC issued Regulatory Guide 1.87 (Rev 1, June 1975) to provide interim licensing guidelines to aid applicants in implementing the 10 CFR Part 50 requirements with respect to ASME Class 1 components operating at elevated temperatures for high-temperature gas-cooled reactors, liquid-metal fast-breeder reactors, and gas-cooled fast-breeder reactors. Regulatory Guide 1.87 referenced ASME Code Case 1592 for materials and design, Code Cases 1593, 1594, 1595 and 1596 for fabrication and installation, examination, testing, and overpressure protection, respectively. When the nuclear code cases were separated from the non-nuclear code cases by ASME, the Code Case 1592 series was converted to Code Case N-47, which was used by the Clinch River Breeder Reactor (CRBR) project, with additional DOE requirements, for the structural design of CRBR. Continued improvements of the rules of N-47 were made, and it was subsumed into a new Section III, Division 1, Subsection NH by ASME in 1995. The Division 1, Subsection NH rules were subsumed into Division 5 as Subsection HB, Subpart B in 2011.

2.1.1 Structural Failure Modes

The design philosophy of HBB is to protect reactor components against structural failure modes with reasonable assurance. The HBB design methodologies are based on a design-by-analysis approach. The failure modes can be categorized into time independent failure modes for low temperature operations where creep effect is negligible, and time dependent failure modes for elevated temperature service where the operations are in the creep regime. These failure modes are summarized in Table 2.2.

Table 2.2: Structural failure modes for Division 5 Class A components.

Time Independent Failure Mode	Design Method	Time Dependent Failure Mode	Design Methods
Ductile rupture from short-term loading	Primary Load Check	Creep rupture from long-term loading	Primary Load Check
Gross distortion due to incremental collapse and ratcheting (low temperatures)	Strain Limits Check	Creep ratcheting	Strain Limits Check
Loss of function due to excessive deformation	Strain Limits Check	Creep-fatigue failure	Strain Limits Check
Buckling due to short-term loading	Buckling Check	Creep-buckling due to long-term loading	Creep Buckling Check

The structural failure modes considered in HBB are those caused by loads applied to and temperature transients experienced by components during reactor operations. They can be grouped into (i) load controlled, and (ii) strain and deformation controlled failure modes and are driven by primary stresses and secondary stresses, respectively. The primary stresses are those that do not relax away during creep while the secondary stresses are those that would redistribute during creep. The load controlled failure modes are guarded against by the primary load check and the strain and

deformation controlled failure modes are prevented by the strain limits, creep-fatigue and buckling checks.

2.1.2 Design Limits

The limits associated with these checks can be summarized in the analysis flowchart of HBB shown in Figure 2.1.

Figure 2.1: HBB flow diagram for elevated temperature analysis.

For the primary load design, primary stresses acting on a cross-section of a component due to service loads are linearized into membrane (average), linearized bending and peak stresses and they are limited by the allowable stress intensity, S_{mt} . A life-fraction summation is used when different load cases are involved. The values of the allowable stress intensity are based on the yield strength, ultimate strength, rupture strength and creep deformation data. Thermal aging effects due

to time-at-temperature on the yield and ultimate strengths are accounted for. Thus tensile tests as a function of temperatures and creep rupture tests as functions of temperatures and applied stresses are required to generate these data.

For the strain limits check, the average, linearized bending and maximum of the accumulated inelastic strain across a section are limited to 1%, 2% and 5%, respectively.

For the creep-fatigue check, the total creep damage and fatigue damage due to thermal cycles are evaluated using the creep deformation and creep rupture properties and the fatigue design curves, respectively. The creep-fatigue check is satisfied when the coordinates of the creep damage and fatigue damage are within the creep-fatigue interaction envelope, as illustrated in Figure 2.2. Data generated from strain controlled creep-fatigue tests for different temperatures, strain ranges and hold times are used to construct the creep-fatigue interaction envelope. Data from strain controlled cyclic tests for different temperatures and strain ranges are used to develop the fatigue design curves.

Figure 2.2: Sketch of a creep-fatigue interaction diagram. The green point denotes satisfactory creep-fatigue performance. The red point corresponds to failing the creep-fatigue check.

Time independent buckling check is conducted using buckling charts provided in the ASME Code.

2.1.3 Design Evaluation Options

There are two broad categories of design evaluation options in HBB that can be used to address these design checks. The simplified methods are based on elastic analysis results and are intended as screening tools to check against Code compliance. The design evaluations based on inelastic analysis results provide more accurate but more complex assessments and are intended for checking Code compliance at “hot spots” where the design does not pass the Code checks using the screening methods.

The elastic, perfectly plastic (EPP) methods introduced recently through the Division 5 Code Cases also provide screening tools for primary load, strain accumulations and creep-fatigue checks. But they remove some of the limitations of the elastic analysis approach, e.g., temperature limits, and

do not require stress classification which is complex and requires much engineering judgment. The EPP methods are also more easily adaptable for modern finite element technology.

2.1.4 Environmental Effects

Design procedures and materials data not contained in HBB may be required to ensure the integrity or the continued functioning of the structural part during the specified service life. For example, HBB rules do not provide methods to evaluate material deterioration that may occur in service as a result of corrosion, mass transfer phenomena, radiation effects, or other material instabilities. Owner/operator has the responsibility to demonstrate to US NRC that these effects are accounted for in the design of the component. Concerning the environmental effects, since no coolants are deployed in a heat pipe microreactor design, neutron irradiation effects on structural failure modes would be the main environmental consideration.

3 Effects of Neutron Irradiation

There is an extensive literature on the effects of neutron irradiation on structural materials. The neutron fluence, flux and spectrum, the irradiation temperature, the chemical compositions and the microstructure all have influence on the defects generation and microstructural changes in structural materials. These irradiation mechanisms are complex. However, the phenomena that are important for the structural integrity of components constructed of ferritic-martensitic steels are irradiation creep, void swelling, and irradiation-induced material properties changes. The material properties changes include irradiation hardening and the concomitant loss in ductility, helium embrittlement and other forms of grain boundary embrittlement.

Atomic displacements during irradiation leads to inelastic deformation of the material. Such a deformation is referred to as irradiation creep, in analogy to thermal creep. Similar to thermal creep, irradiation creep can contribute to stress redistribution. However, unlike thermal creep, irradiation creep is considered to be non-damaging and is not limited by creep ductility.

Void swelling is usually considered as stress-free volumetric change of the material. And similar to thermal expansion, void swelling would generate stresses if the volume change is constrained. But like thermal stresses, the void swelling induced stresses are self-limiting and hence are secondary stresses.

The irradiation hardening and the corresponding ductility loss will need to be accounted for in the design procedure as the Division 5 rules assume certain ductility in the materials selected for the Code.

The approximate temperature ranges where these phenomena are dominant are shown in Figure 3.1, following Zinkle et al. [2]. Here T/T_M is the homologous temperature. Generally, ferritic-martensitic steels have much higher resistance to void swelling than austenitic stainless steels.

Figure 3.1: Approximate temperature regimes where different phenomena caused by irradiation are most prominent in structural alloys, schematic after [2]. The highlighted regions correspond to room temperature to 700F and 700 to 1200F for Grade 91 steel.

For Grade 91 steel, the high temperature HBB rules are used when the temperature is above 700F (371C). The maximum use temperature for Grade 91 in HBB is 1200F (650C). With a melting temperature of 2588F (1420C), the homologous temperatures for room temperature, 700F and 1200F are 0.17, 0.38 and 0.55, respectively. These two regions are shown in Figure 3.1.

An overview of the irradiation behavior of the Grade 91 steel is provided by Natesan et al. [3]. References to the irradiation data for Grade 91 are cited in [3]. The approximate temperature ranges where the degradation mechanisms due to irradiation operate are shown in Table 3.1.

Table 3.1: Operative temperature ranges for Grade 91 irradiation degradation mechanisms

Irradiation degradation mechanism	Approximate operative temperature range
Irradiation hardening and loss of ductility	Up to 790F (420C)
Irradiation creep and growth	750 to 1000F (400 to 540C)
Void swelling	570 to 930F (300 to 500C)
Helium embrittlement	840 to 1200F (450 to 650C)

3.1 Design Rules to Address Irradiation

As noted previously, the HBB rules do not provide explicit rules to address irradiation-induced phenomena that affect the structural failure modes covered by HBB. Thus additional consideration is necessary to supplement the HBB rules for the design of the core block for the heat pipe reactor.

The ITER Structural Design Criteria (ISDC) was developed in the 1990's to support the structural design of the components of the fusion reactor for the International Thermonuclear Experimental Reactor (ITER) Project. The ISDC adopted many of the design rules from ASME Boiler and Pressure Vessel Code, Division 1, Subsection NH and the French design code RCC-MR and introduced new rules to account for radiation embrittlement.

3.1.1 Low Temperature Service

For low service temperatures, the irradiated yield and ultimate strengths are added to the criteria for the time-independent allowable stress, S_m , of HBB. Two new rules were introduced in ISDC for materials where the uniform elongation and total elongation are reduced due to irradiation.

The first new rule was introduced to prevent failure due to plastic flow localization. The elastically calculated primary plus secondary membrane stress intensity is limited to a new time independent allowable stress, S_e , based on the irradiated ultimate strength, the irradiated uniform elongation, Young's modulus and an elastic follow up factor.

The second new rule was introduced to prevent local failure due to ductility exhaustion. This is achieved by limiting the elastically calculated primary membrane plus bending plus secondary stress intensity to another new time independent allowable stress, S_d , based on the irradiated ultimate strength, irradiated total elongation, Young's modulus, a triaxiality factor and an elastic follow up factor.

3.1.2 High Temperature Service

For high temperature service, ISDC employs the time dependent allowable stress intensity, S_t , of HBB for primary load design. However, if the creep rupture strain is less than 5% but greater than 1%, the time to 1% strain in the criteria for S_t is replaced by the time to a strain that is the lesser

of 1% and one-fifth of creep rupture strain. If the irradiation dose is large enough that the high temperature helium embrittlement significantly reduces the creep rupture strain to 1% or less, further reduction of the S_t value might be required or the irradiation dose reduced, for example, by shielding.

For strain limits check, ISDC uses elastic analysis results and applies bounding methods similar to the Test No. A-1, No. A-2, No. A-3 and No. B-1 in Appendix HBB-T of HBB, except that the irradiated yield strength is used.

Procedures for creep-fatigue check are not available in the ISDC since the ITER design does not involve significant creep-fatigue interaction. Though creep-fatigue damage would be significant in the next generation fusion reactor design such as DEMO (Sham and Zinkle [4].)

An initial procedure for conducting creep-fatigue check for the core block can be structured as follows. To simplify the procedure, it is first assumed that the irradiation dose is low enough that the creep rupture strain is not significantly reduced by high temperature helium embrittlement. Based on the HBB creep-fatigue procedure for thermal creep using the elastic analysis results, the following modification can be made.

The creep damage can be evaluated a procedure similar to that in HBB. The time fraction is calculated using the creep rupture correlation based on thermal creep but the stress relaxation during the strain hold would be evaluated differently. Instead of using the isochronous stress-strain curves developed from thermal creep, new isochronous stress-strain curves based on the combined thermal and irradiation creep would be used. In addition to temperature and time, the irradiation dose is an additional independent variable for these new isochronous stress-strain curves.

For the fatigue damage evaluation, dose dependent fatigue design curves are required. Further, in addition to the plasticity and creep corrections to the elastically calculated strain range, an additional irradiation creep correction is required using irradiation creep correlation, which is available from the literature for Grade 91.

For the creep-fatigue interaction envelope, an initial approach is to adopt the same envelope from un-irradiated material.

3.1.3 Design Parameters Development

For Grade 91 steel, there are sufficient irradiation data in the literature to develop conservative values for irradiated yield and tensile strengths, irradiated uniform elongation and total elongation. Irradiation creep and void swelling correlations are also available from the literature. However, data to develop fatigue design curves from irradiated Grade 91 materials are not available. Some conservative assumption will need to be employed to enable scoping analyses to be performed. The assumption to neglect the influence of irradiation on the creep-fatigue interaction envelope will also need to be checked.

4 Summary

Methods to account for the irradiation effect in the rules of the ASME Boiler and Pressure Vessel Code, Section III, Division 5, Subsection HB, Subpart B were investigated in order to develop design methods that can be applied to the core block design in the heat pipe microreactor.

By incorporating some rules from the ISDC and the modification of existing HBB rules, an approach is established that could be used to support scoping analyses. Design parameters required in this approach from irradiated Grade 91 material can be developed using literature data. The simplifying assumptions employed in this initial approach will need to be verified by irradiation testing.

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Bibliography

- [1] J. W. Sterbentz, J. E. Werner, M. G. McKellar, A. J. Hummel, J. C. Kennedy, R. N. Wright, J. M. Biersdorf, "Special Purpose Nuclear Reactor (5 MW) for Reliable Power at Remote Sites Assessment Report - Using Phenomena Identification and Ranking Tables (PIRTs)," INL/EXT-16-40741, Revision 1, Idaho National Laboratory, Idaho Falls, ID, 2017.
- [2] S. J. Zinkle, H. Tanigawa, and B. D. Wirth, "Radiation and Thermomechanical Degradation Effects in Reactor Structural Alloys," In Structural Alloys for Nuclear Energy Applications, pp. 163-210, Elsevier, Cambridge, MA, USA, 2019.
- [3] K. Natesan¹, M. Li, S. Majumdar, R.K. Nanstad, and T.-L. Sham, " Code Qualification of Structural Materials for AFCI Advanced Recycling Reactors," ANL-AFCI-244, Argonne National Laboratory, Lemont, IL, 2008
- [4] T.-L. Sham and S.J. Zinkle, "Creep and Fatigue Issues for Structural Materials in Demonstration Fusion Energy Systems," Transactions of The Indian Institute of Metals, Vol. 63, Issues 2-3, pp. 331-337, 2010.

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