INL/MIS-21-62440-Revision-0

High Temperature Alloys Session 1

April 2021

Changing the World's Energy Future

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High Temperature Alloys Session 1

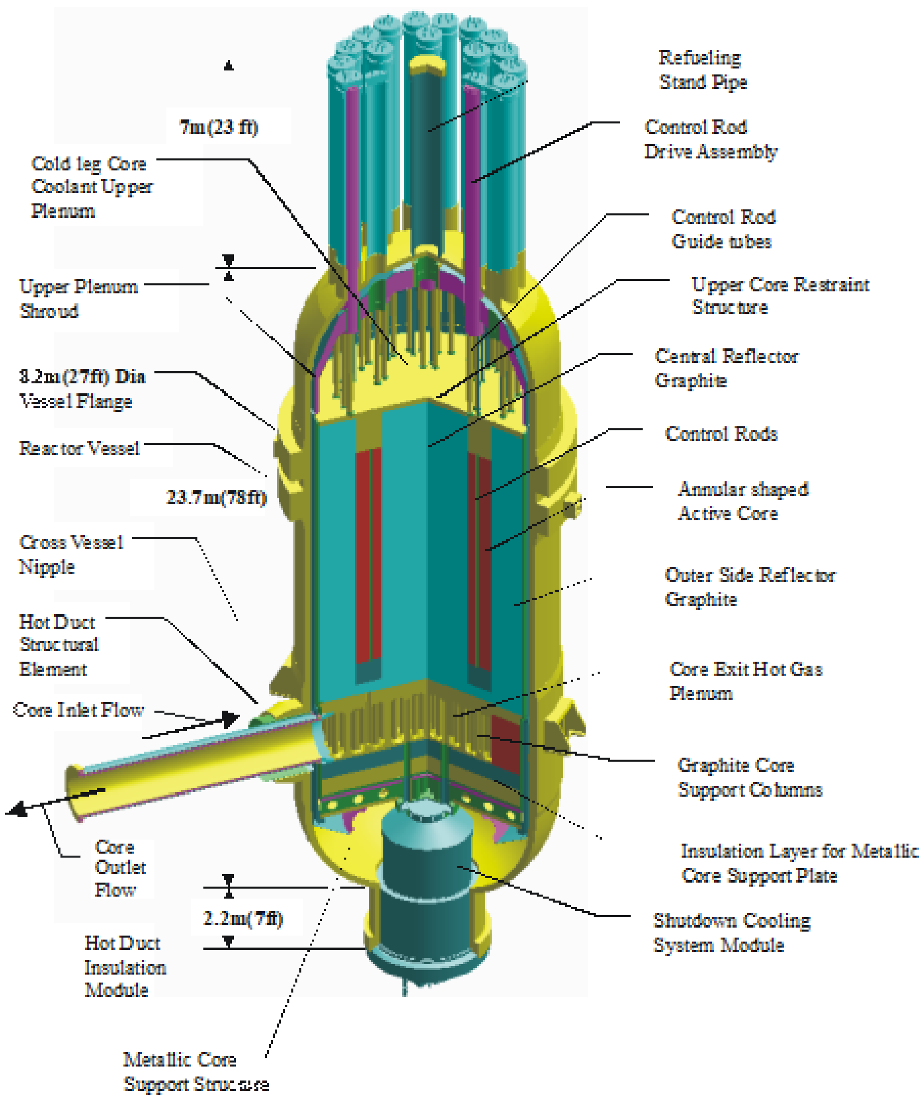
CNSC Seminar

Idaho National Laboratory

Outline of Topics in This Presentation

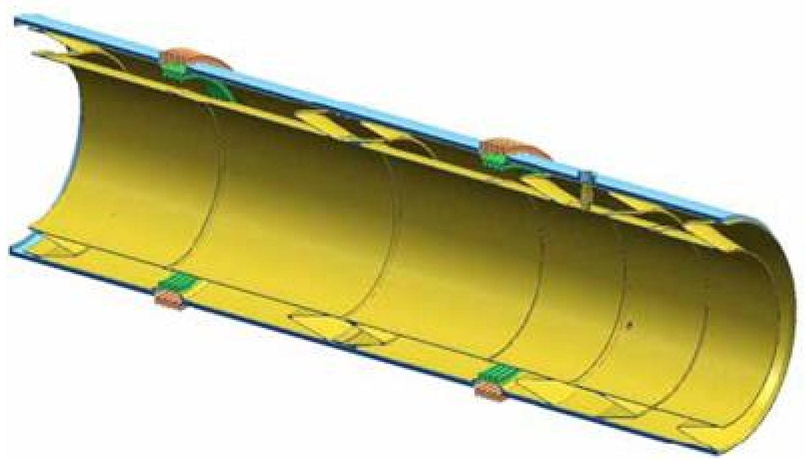
- Section III Division 5 overview
- Code materials
- VHTR materials 617 and 800H
- Qualification of additional materials
- Corrosion effects
- Operating plant experience
- Additive/advanced manufacturing
- Control rod sleeves
- Pressure Vessel Steels
- Allowed materials
- Elevated temperature limits and properties
- Supply chain
- Welding
- Radiation damage

Preconceptual Design for 600MW$_{Thermal}$ VHTR



GA-MHR

- Demonstration high temperature or very high temperature plants have been operated since the 1960s
- The intent of these plants was to demonstrate technology for large reactors for process heat or electricity generation
- The Generation IV International Forum program considered 600MW thermal designs
- Recent activity has focused on small modular designs or mobile micro-reactors. This might impact material and fabrication choices



Cross-duct with high temperature alloy liner and insulated pressure vessel steel structural component

ASME Section III Division 5 Framework for Component Design (Part I)

- Section III Division 1 rules cover light water reactor systems
  - These rules do not allow time dependent deformation
  - Upper temperature limit for ferritic materials is 375°C and for austenitic materials is 425°C
- Section III Division 5 “Rules for Construction of Nuclear Facility Components – High Temperature Reactors” has replaced Section III Division 1 for construction of high temperature reactors
  - Section III Division 1 Subsection NH was first included in the 1995 with 1996 and 1997 Addenda version of the ASME BPVC Code.
  - Section III Division 5 was added in the 2010 with 2011 Addenda version of the Code and considered separate from Section III Division 1 Subsection NH
  - ASME BPVC 2017 is the first version of the code to come without Section III Division 1 Subsection NH
- These rules are applicable to high temperature reactor systems, including HTGR, LMR and MSR
  - ASME BPVC does not consider environment effects for metals
  - For example, Alloy 617 contains up to 15% Co and would not be appropriate in a neutron environment, but the Code would not specifically prohibit it. (Note Alloy 617 is being explored for use in the secondary heat exchanger. As such, it will not experience neutron radiation and the cobalt level is not a concern)

ASME Boiler and Pressure Vessel Code, 2019 Edition, Section III Division 1 and Section III Division 5

ASME Section III Division 5 Framework for Component Design (Part II)

- Only six alloys are allowed for nuclear components under these rules:
  - Annealed 2.25Cr-1Mo and V modified 9Cr-1Mo ferritic steels
  - Type 304 and Type 316H stainless steels and Alloy 800H
  - Sixth alloy, Inconel 617, was recently added as Code Case N-898 in Fall 2019

Material	Fe	Ni	Cr	Co	Mo	Al	C	Mn	Si	S	Ti	Cu	B	P	V	N	Nb
304/304H	Bal	8.0-10.5	18.0-20.0		-	-	0.04-0.08/0.10	2.0 max	0.75 max	0.03 max	-	-	-	0.045 max	-	0.10 max	-
316/316H	Bal	10.0-14.0	16.0-18.0	-	2.0-3.0	-	0.04-0.08/0.10	2.0 max/ 0.04-0.10	0.75 max	0.03 max	-	-	-	0.045 max	-	0.10 max	-
800H	39.5 min	30.0-35.0	19.0-23.0	-	-	0.15-0.60	0.05-0.10	-	-	-	0.15-0.60	-	-	-	-	-	-
2.25Cr-1Mo	Bal	-	2.0-2.5	-	0.90-1.1	-	0.07-0.15	0.30-0.60	0.50 max	0.025 max	-	-	-	0.025 max	-	-	-
9Cr-1Mo-V	Bal	0.40 max	8.0-9.5	-	0.85-1.05	0.04 max	0.08-0.12	0.30-0.60	0.20-0.50	0.010 max	-	-	-	0.020 max	0.18-0.25	0.30-0.70	0.06-0.10
617	3.0 max	44.5 min	20.0-24.0	10.0-15.0	8.0-10.0	0.8-1.5	0.05-0.15	1.0 max	1.0 max	0.015 max	0.6 max	0.5 max	0.006 max	-	-	-	-

ASME Boiler and Pressure Vessel Code, 2019 Edition, Section III Division 1 and Section III Division 5

ASME Section III Division 5 Framework for Component Design (Part IV)

- Material classes allowed in Subsection HA, and maximum temperature allowed (T_{max})

Materials	T_{max}, °F (°C)
Carbon steel	700 (370)
Low alloy steel	700 (370)
Martensitic stainless steel	700 (370)
Austenitic stainless steel	800 (425)
Nickel-chromium-iron	800 (425)
Nickel-copper	800 (425)

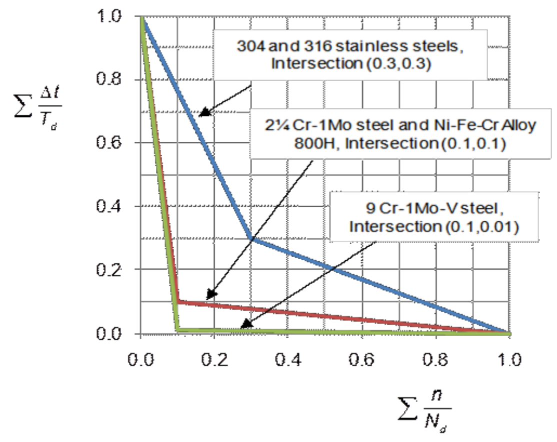
- When safety-related components exceed the appropriate temperature limits from Subsection HA, then Subsection HB is used

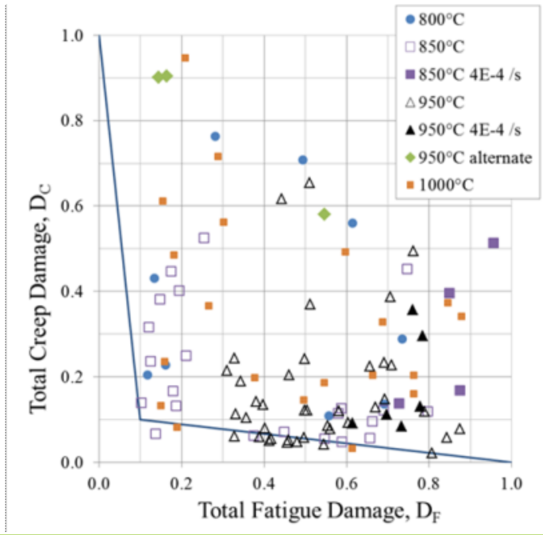
Materials	Temp. not exceeding, °F (°C)
304 SS	1500 (816)
316 SS	1500 (816)
800H	1400 (760)
2.25Cr-1Mo	1100 (593)
9Cr-1Mo-V	1200 (650)

ASME Boiler and Pressure Vessel Code, 2019 Edition, Section III Division 1 and Section III Division 5

ASME Section III Division 5 for Nuclear Qualification (Part III)

- For each allowed material, limits are set for upper temperature and time, e.g., for Alloy 800H 750°C and 300,000 hours
- In addition to time dependent deformation, design rules accounting for creep-fatigue are incorporated
  - The creep-fatigue interaction model takes into account the deleterious effects of creep and fatigue together
  - If creep and fatigue were solely considered separately, design models would be non-conservative, as creep-fatigue interactions cause failure earlier in life than would be expected
- Note: All temperature in degrees Celsius are rounded off per ASME metric convention. Maximum use temperature are expressed in degrees Fahrenheit in Division 5





ASME Boiler and Pressure Vessel Code, 2019 Edition, Section III Division 1 and Section III Division 5

Welding, Diffusion Bonding, Aging and Cold Work

- Gas-tungsten arc welding (GTAW) and submerged arc welding processes (including weld process qualification and qualified filler metals) and inspection requirements are incorporated in the ASME Code for pressure vessel steels and Alloy 800H
- Only GTAW welding is included in the Alloy 617 Code Case currently in the approval process
- Weld strength reduction factors are specified in Section III Division 5 and are applied to creep rupture properties as specified in appropriate sections of the design rules
- Diffusion bonding has been proposed for fabrication of compact heat exchangers for VHTR use – this process is not approved in Section III Division 5 for nuclear construction, though a DOE-NE IRP project is developing Division 5 construction rules for compact heat exchangers
- Reduction factors on the tensile properties are required for some Section III Division 5 materials to be used in seismic analysis of components after long time aging in service; where those factors are required they are specified in appropriate sections of the design rules
- Since VHTR components are expected to experience long-time, elevated temperature service cold worked materials are generally not allowed for the Section III Division 5 materials
- Up to 5% incidental cold work associated with fit-up strain is typically allowed

ASME Boiler and Pressure Vessel Code, 2019 Edition, Section III Division 1 and Section III Division 5, Section IX Welding and Section XI Non-Destructive Examination

Materials Issues for Steam Generator and Heat Exchanger Applications

- The Fe-Ni-Cr material Alloy 800H is fully Section III Division 5 Code qualified for use up to 750°C and times up to 300,000 hours
- Alloy 800H has adequate properties for proposed VHTR steam generator tubes up to the maximum Code qualification temperature
- Above 750°C for gas-to-gas heat exchangers an additional material – Ni-Cr-Co-Mo Alloy 617 is Code qualified
- The Alloy 617 Code Case is for an upper temperature limit of 950°C and time of 100,000 hours
- Both Alloy 800H and Alloy 617 were extensively characterized for the gas reactor programs in Germany, Japan and the US in the 1970s and 1980s
- Alloy 800H was used in the steam generator of the German pebble bed demonstration reactors and in the US Fort St. Vrain plant
- Additional alloys – Hastelloy X and Haynes 230 have been considered for high temperature structural applications, but neither was judged by the US program to have sufficient technical maturity and creep properties to proceed with Code qualification
- The Japanese demonstration reactor has used a modified Hastelloy X in the cross-duct and heat exchanger; this alloy is little known in the US and is not Code qualified

Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan INL Document PLN-2804, Rev. 1, 2010
J. Wright, 2015, Draft ASME Boiler and Pressure Vessel Code Case for Use of Alloy 617 for Class A Elevated Temperature Service Construction, INL/EXT-15-36305

ASME Code Qualification

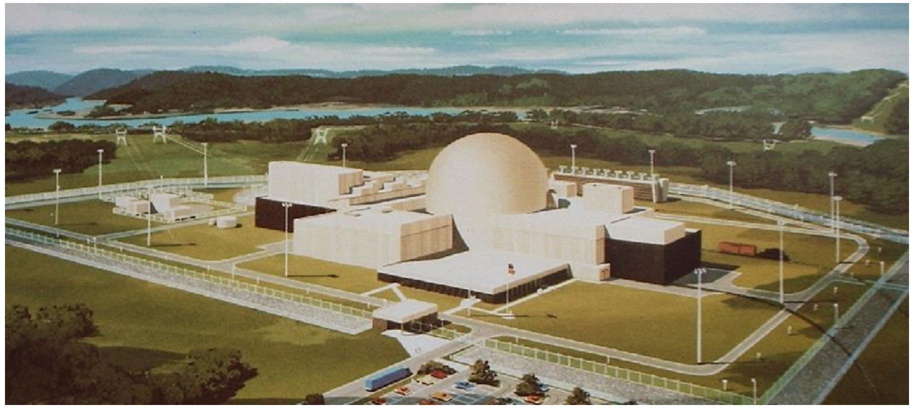
- Higher temperature design of VHTR systems might require structural alloys with elevated temperature properties exceeding those of the six Code qualified alloys; new materials would need to be qualified
- Section III Division 5, Appendix HBB-Y, “Guidelines for Design Data Needs for New Materials” describes required properties
  - Technical basis established through DOE Advanced Reactor Technology base program on the Alloy 617 Code Case in support of HTGR/VHTR applications

Required testing to introduce a new structural material into Section III, Division 5, or a Division 5 Code Case

- HBB-Y-2100 Requirement For Time-independent Data
- HBB-Y-2110 Data Requirement for Tensile Reduction Factors for Aging
- HBB-Y-2200 Requirement for Time-Dependent Data
- HBB-Y-2300 Data Requirement for Weldments
- HBB-Y-3100 Data Requirement for Isochronous Stress-Strain Curves
- HBB-Y-3200 Data Requirement for Relaxation Strength
- HBB-Y-3300 Data Requirement for Creep-Fatigue
- HBB-Y-3400 Data Requirement for Creep-Fatigue of Weldments
- HBB-Y-3500 Data Requirement for Cyclic Stress-Strain Curves
- HBB-Y-3600 Data Requirement for Inelastic Constitutive Model
- HBB-Y-3700 Data requirement for Huddleston multiaxial failure criterion
- HBB-Y-3800 Data Requirement for Time-Temperature Limits for External Pressure Charts
- HBB-Y-4100 Data Requirement for Cold Forming Limits
- Validation of Elastic-Perfectly Plastic (EPP) Simplified Design Methods for the new alloy

Issues Identified in NRC Assessment of the Clinch River Breeder Reactor

- Nine areas of concern were identified in the NRC assessment of the Clinch River Breeder Reactor in the late 70’s and early 80’s that are still under evaluation for elevated temperature components:
  - Weldment cracking
  - Notch weakening
  - Materials property representation for inelastic analysis
  - Steam generator tubesheet evaluation
  - Elevated temperature seismic effects
  - Elastic follow-up in piping
  - Creep-fatigue evaluation
  - Plastic strain concentration factors
  - Intermediate piping transition weld



NUREG-0968, Vol. 1 Main Report, “Safety Evaluation Report Related to the Construction of the Clinch River Breeder Reactor Plant,” March 1983, ML082380939

NRC Evaluation of High Temperature Power Reactors

- In the 90’s, the NRC sponsored a reevaluation of the design issues for high temperature reactors
- 23 issues needed to be resolved, most importantly
  - Lack of material property allowable design data/curves for 60 year design life
  - Degradation of material properties at high temperatures due to long-term irradiation
  - Degradation of material properties due to corrosion phenomena
  - Lack of validated thermal striping materials and design methodology
  - Lack of reliable creep-fatigue design rules
  - Lack of validated weldment design methodology
  - Lack of flaw assessment procedures
  - Lack of understanding/validation of notch weakening effects
  - Lack of validated rules/guidelines to account for seismic effects at elevated temperatures
  - Lack of inelastic design procedures for piping

W. O’Donnell, A. Hull, S. Malik, 2008, “Structural Integrity Code and Regulatory Issues in the Design of High Temperature Reactors,” Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology

Further Review of High Temperature Reactor Regulator Requirements

- Mid 2000’s, NRC updated the licensing needs for next generation power plants
  - General issues related to high temperature stability
  - Ability to withstand service conditions
  - Long-term thermal aging
  - Environmental degradation (impure helium)
  - Issues associated with fabrication and heavy-section properties
  - Further development of Section III of the ASME code needed (for higher temperatures – up to at least 900^{o}C), including Alloy 617 and Hastelloy X
  - Creep behavior models and constitutive relations are needed for cyclic creep loading
  - Models must account for the interaction between the time independent and time dependent material response

W. O’Donnell, A. Hull, S. Malik, 2008, “Structural Integrity Code and Regulatory Issues in the Design of High Temperature Reactors,” Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology

VHTR Phenomena Identification and Ranking Tables (PIRT)

- Safety relevant phenomena were considered for potential degradation concerns and ranked according to importance and current state of knowledge
- High temperature structural materials issues were evaluated for major structural components such as the reactor pressure vessel, control rods, reactor internals, primary circuit components, heat exchangers, etc.
- The PIRT was created as there are major design changes for high temperature reactors from the current LWR reactors and both the industry and NRC have very little experience with HTGRs (there is very little existing data)
- 58 phenomena were identified, with 17 of high importance and low/medium state of knowledge

Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs}, Volume 4: High-Temperature Materials PIRTs, NUREG/CR-6944, Vol. 4, ORNL/TM-2007/147, Vol. 4, March 2008

Operating Experience with VHTR Technology

	Dragon	AVR	Peach Bottom	Ft. St. Vrain	THTR-300	HTTR
Country of Origin	OECD/Britain	Germany	U.S.	U.S.	Germany	Japan
Thermal Power MWt	21.5	46	115	842	750	30
Net Electric Power MWe	—	13	40	330	300	10
Maximum Core Outlet Temp °C	750	950	725	775	750	950
Helium Pressure MPa	2.0	1.1	2.25	4.8	3.9	4
Steam Temp °C	—	505	538	538	530	
Reactor Type	Sleeve	Pebble	Sleeve	Block	Pebble	Prism
Vessel Material	Steel	Steel	Steel	PCRV[a]	PCRV	Steel
Date of Operation	1964-1975	1966	1967	1979 – 1989	1985	1997

a. Pre-stressed concrete reactor vessel.

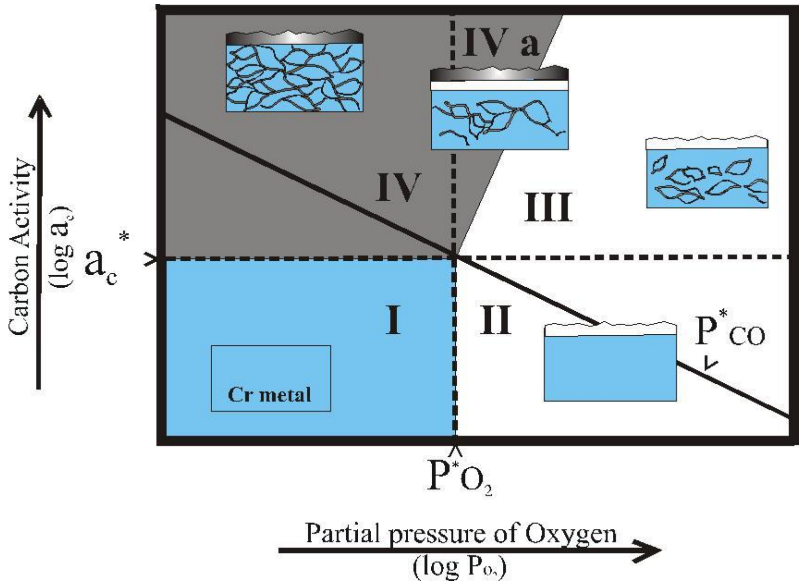
Alloys Used in Demonstration VHTR Applications

Alloy	Ni	Fe	Cr	Co	Mo	Al	W	Ti	C	Si	Mn
Inconel 617	44.5	3	20-24	10-15	8-10	0.8-1.5		0.6	0.05-0.15	1	1
Alloy 230	Bal	3	20-24	5	1-3	0.2-0.5	13-15		0.05-0.15	.25-.75	0.3-1
Alloy 800H	30-35	39.5	19-23			0.15-0.6		0.15-0.6	0.05-0.1		
Alloy XR	Bal	20	23	1	10	0.1	1	0.03	0.15	0.5	1

Chemistry of Coolant for Operating Plants (Composition in ppm)

	H_2O	H_2	CO	CO_2	CH_4	O_2	N_2
Dragon	0.1	0.1	0.05	0.02	0.1	0.1	0.05
Peach Bottom	0.5	10	0.5	<0.05	1.0	—	0.5
Fort St. Vrain	1	7	3	1	0.1	—	—
AVR	0.15	9	45	0.25	1		22
THTR	<0.01	0.8	0.4	0.2	0.1		0.1

Aging and Environmental Effects



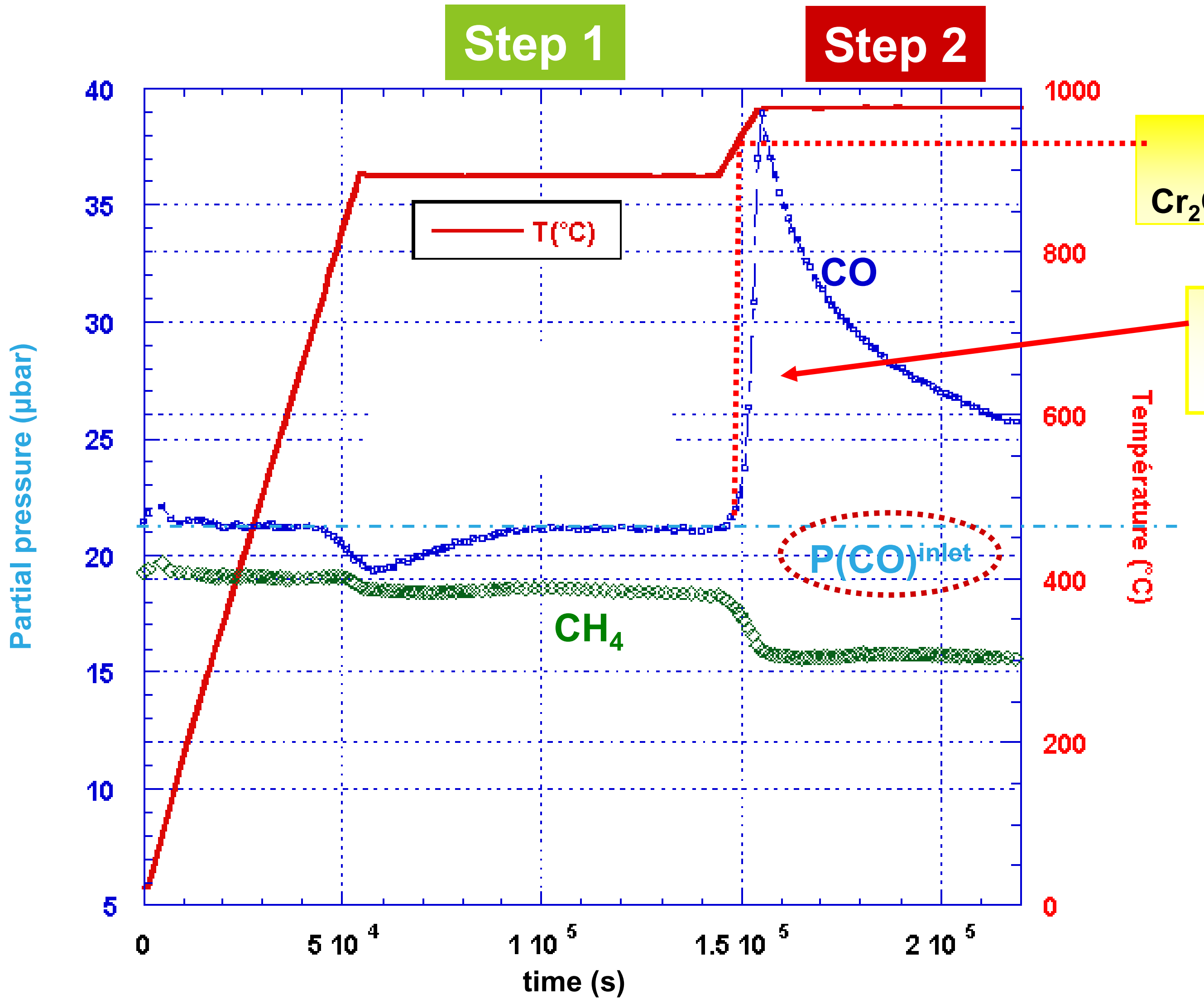
Assessments of Inconel 617 stability at various gas concentrations. Five conditions are represented:

- I. Reducing
- II. Oxidizing
- III. Stable external oxide with stable internal carbides
- IV. Strongly carburizing internally and externally
- IVa. Strong external carburization with stable oxide layer

- There is no environment that is inert with respect to the alloys; oxidation or carburization will always occur to some extent depending on the coolant gas chemistry and temperature
- Environmental effects maps will help in specification of He impurity content of primary coolant for long-term stability of heat exchangers
- A slightly oxidizing gas chemistry is preferred (region II in the figure); the protective oxide scale prevents either rapid oxidation or carburization
- The large volume of graphite was shown in the German AVR demonstration reactor to provide a chemical buffer on the coolant such that the preferred impurity content was maintained
- The mechanical properties of Alloys 800H and 617 are not significantly affected by long-term exposure to typical VHTR gas chemistry

C. Cabet, A. Mannier, and A. Terlain, 2004, “Corrosion of High Temperature Alloys in the Coolant Helium of a Gas Cooled Reactor,” *Materials Science Forum,* Vols. 461-464, pp. 1165-1172.

Dynamic test: critical temperature T_A



T_A

$Cr_2O_3 + 3[C] \rightarrow 3CO + 2Cr$

Production of CO(g) & scale destruction

Haynes 230®

He /200 H_2 /21 CO / 19 CH_4 /0.5 H_2O (µbar)

Dynamic Test: Surface Reactions

Oxidation by H_2O (&CO)

$T<T_A$

$$3H_2O + 2Cr \rightarrow \underbrace{Cr_2O_3}_{\text{surface}} + 3H_2$$

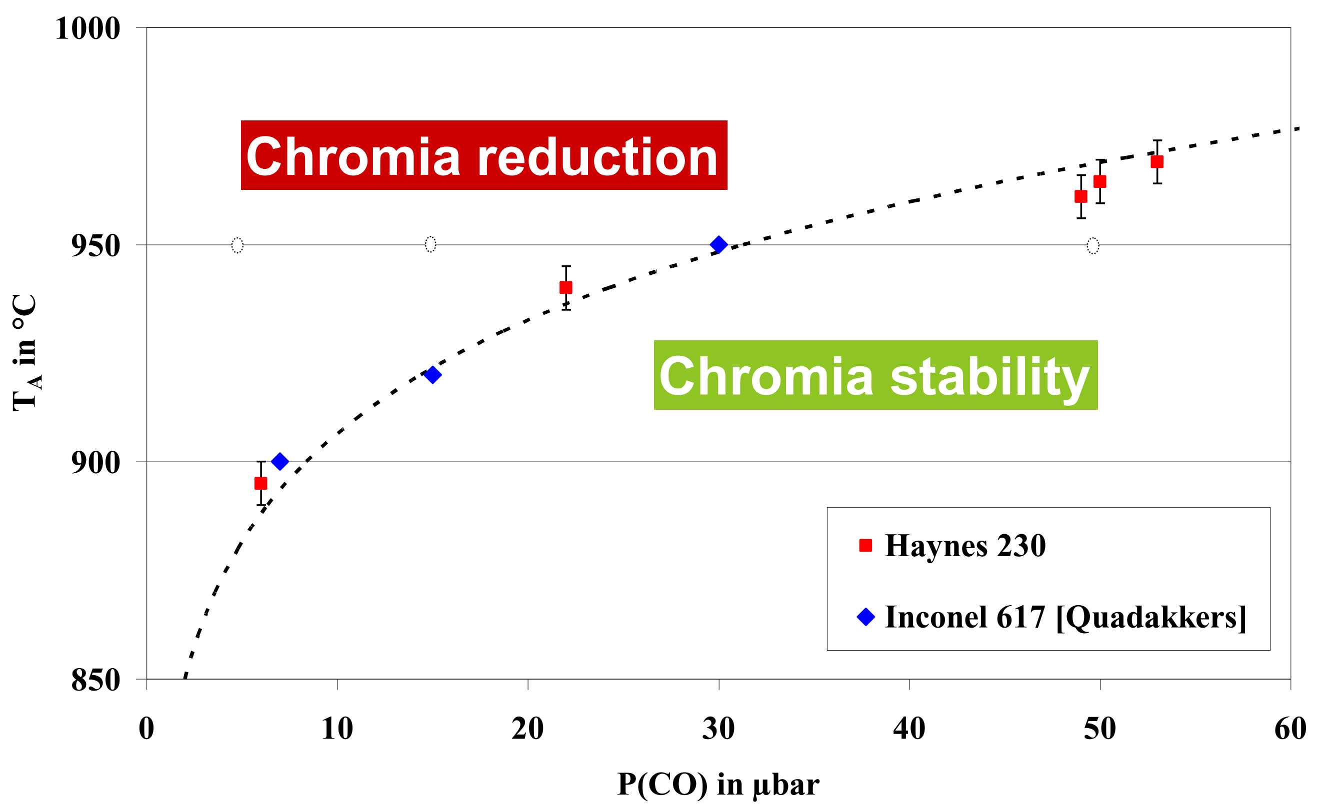
$$3CO + 2Al \rightarrow \underbrace{Al_2O_3}_{\text{internal}} + 3C_{Solution}$$

Cr-oxide reduction by carbon

$T>T_A$

$$Cr_2O_3 + 3C_{Solution} \rightarrow 3CO(g) + 2Cr$$

Stability of the Surface Chromia vs. P(CO)



Rouillard F., Cabet C. et al. Oxid Met 68 (2007) 133
data on Inconel 617 after [4] W. J. Quadakkers, Werkstoffe und Korrosion 36 (1985) 335

Control Rod Assessment Pebble Bed Modular Reactor

- 24 replaceable control rods located in the side graphite reflector blocks
- Half of the rods are used for control and the other half are used for shutdown
- The rods consist of a B4C rings between two coaxial cladding tubes
- Alloy 800H as the most suitable material for the control rods for the following reasons:
  - Adequate high temperature strength at the normal operating temperature of 700°C
  - Creep resistance sufficiently qualified for long-term operation at 700°C
  - Limited operation at 850°C under abnormal events is allowed as per available data
  - Irradiation response has been characterized to high levels of fast fluence.
  - Extensive qualification of Alloy 800H control rods in previous German HTR programs

Overview of Pressure Vessel Steels (Part I)

- VHTR pressure vessels tend to be large – 600MW thermal design concept specified 8m diameter and 250 mm thickness
- VHTR goal outlet temperatures between 700-950 °C
- Conventional SA 508 Grade 3 Class 1 (forging grade) and SA 533 Type B Class 1 (rolled product form) low alloy bainitic steel commonly used in light water reactors can be used if the vessel temperature is held to 370°C or less
  - Mandatory Appendix HBB-II (Of Section III Division 5) allows for use of these steels and their weldments for Class A nuclear components with metal temperatures above 370°C during operating conditions associated with Level B (upset), C (emergency) and D (faulted) service limits
  - Temperature shall not exceed 425°C for Level B and 540°C for Level C and D
  - Component design shall be based on a maximum cumulative time of 3,000 hr at metal temperatures above 370°C

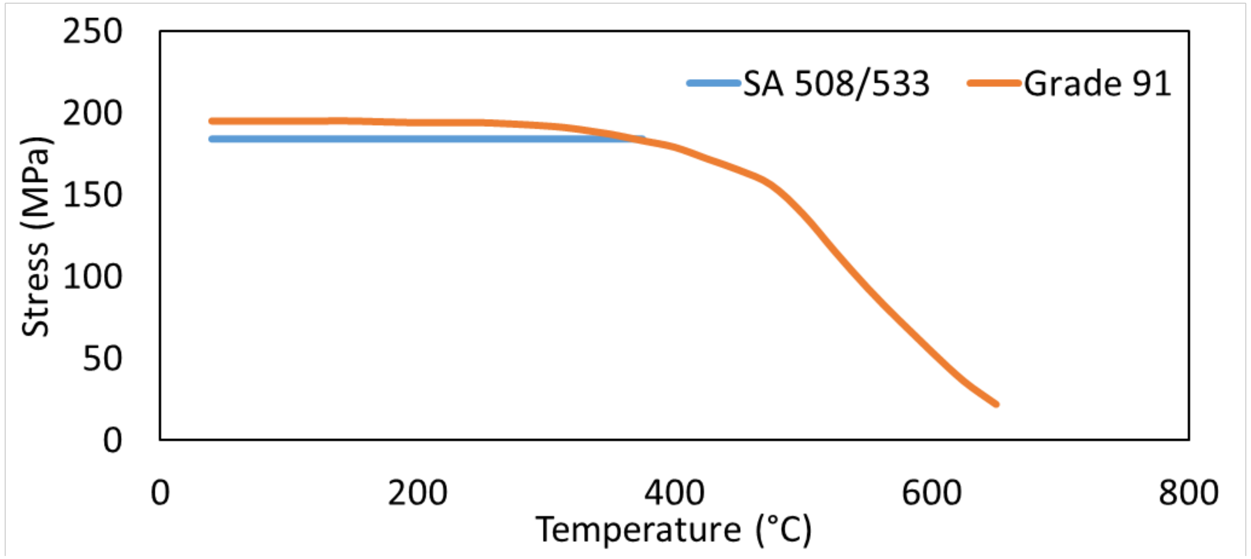
ASME Boiler and Pressure Vessel Code, 2017 Edition, Section III Division 5
Next Generation Nuclear Plant Reactor Pressure Vessel Materials Research and Development Plan INL Document PLN-2803, Rev. 1, 2010
R. Wright, 2014, Creep of A508/A533 Pressure Vessel Steel, INL/EXT-14-32811

Overview of Pressure Vessel Steels (Part II)

- A508/A533B is approved for Class 1 components under Section III, Subsection NB
- NGNP RPV operating temperatures may be higher than LWR, but fluence for 60-year lifetime is comparable or less than LWR beltline for 30 years
- ASME Code rules exist for welding and inspection of heavy section vessels using these steels
- Properties of these materials are largely insensitive to heat treatment and welding; there is a large base of experience resulting from use in light water reactor systems
- For passive cooling of VHTR systems the emissivity of the vessel needs to be high and stable over long operating periods; the native oxide on these steels has been shown to be adequate for passive cooling from accident conditions that have been considered
- Capacity for forging and rolling required section sizes is available for these steels in Japan, Korea and France

Overview of Pressure Vessel Steels (Part III)

- V modified 9Cr-1Mo (Grade 91) steel could be used at higher temperature and is allowed in Section III Division 5 for elevated temperature design
- Grade 91 steel has been considered for use in new French and Japanese fast reactor applications and widely used in tubing in fossil plants; there is currently no capacity to melt or forge sections sizes typical of VHTR vessels



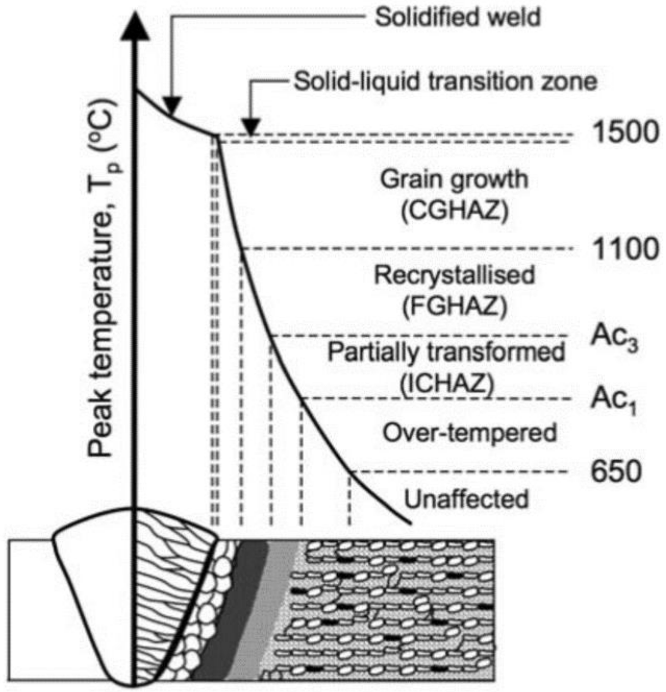
Allowable Stress Intensity Values (100,000 hr life for Grade 91 which affects values above 475°C)

ASME Boiler and Pressure Vessel Code, 2017 Edition, Section II Part D and Section III Division 5

Next Generation Nuclear Plant Reactor Pressure Vessel Materials Research and Development Plan INL Document PLN-2803, Rev. 1, 2010

Overview of Pressure Vessel Steels (Part IV)

- Grade 91 steel is susceptible to Type IV cracking in the heat affected zone of the base metal, above a certain temperature where creep damage occurs
  - This is a form of creep cracking in fine grained recrystallized material in the base metal adjacent to welds (HAZ)
  - Cracks form from creep damage and can be rapid as the crack links voids from creep damage



Schematic of welded ferritic alloys like Grade 91. Type IV cracking occurs in the FGHAZ region.

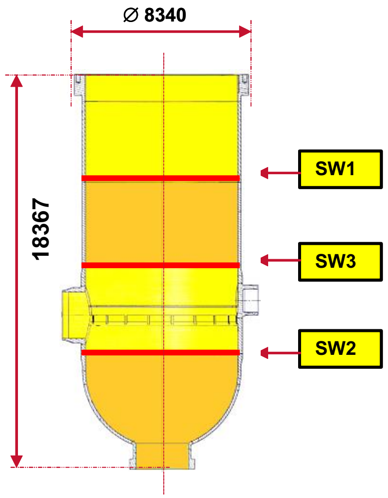
Francis, et al, 2006, Mat. Sci. and Tech. Vol. 22 No. 12, pp. 1387-1395

Overview of Pressure Vessel Steels (Part V)

- Type 4 cracking can only be avoided by a re-normalizing heat treatment after welding or by reducing the temperature of the vessel below the creep regime
- Properties of Grade 91 steel are very sensitive to austenitizing temperature and subsequent tempering treatment; there is currently no NDE method that can assure proper heat treatment was achieved through-thickness in heavy sections
- The US VHTR program made the determination that use of conventional steels (SA 508 and SA 533) was the only feasible near-term option

Next Generation Nuclear Plant Reactor Pressure Vessel Materials Research and Development Plan INL Document PLN-2803, Rev. 1, 2010

RPV Example: AREVA Design and JSW Forging Press



- Welding Grade 91 is well known
- AREVA proposes on-site welding of four rings
- Rings ideally to be forged – the only forge shop on the planet with large enough capacity has no experience with larger than 130 tons Grade 91 – segregation may limit size of ingot
- Can fabricate from rolled semi-circles with longitudinal welds
- Heat affected zone associated with welds will have degraded properties and as-deposited weld metal will be brittle
- On-site welding will require on-site heat treatment

Advanced Manufacturing in the ASME BPVC

- ASME Special Committee on Advanced Manufacturing for Pressure Retaining Components was formed in 2017
  - Is in the process of releasing a report on the "Criteria for Pressure Retaining Metallic Components Using Additive Manufacturing"
  - "…the additive manufacturing build cycle relies on the results of the tensile test of the witness sample to provide the final measure of component quality"
    - Witness specimens shall be constructed and tested with each production build
    - Witness specimens are material test specimens generated during the production build cycles to measure and ensure on-going process stability
    - Used for room temperature tensile tests and metallographic analysis

"The ASME AM Committee did not investigate data for AM components operating in the material creep regime. Creep data was discussed but sufficient material property data was not available to accept AM components operating at elevated temperature in the scope of the current AM criteria"

AM Qualification for High Temperature Use

- There is an ASME Task Group examining AM for Division 5 applications (formed February 2020)

Charter

Fabrication of nuclear components for elevated temperature service using advanced manufacturing (AM) methods is of increasing interest from the vendor community. These methods can include, for example, hot isostatic pressing near net shape components from powder, powder bed fabrication, wire feed methods and diffusion bonding. This Task Group will determine appropriate approaches for qualifying materials processed by AM methods and specifying acceptance criteria for components produced by these methods. The goal of the Task Group is to develop Code Actions for incorporating AM materials and components in Division 5 for elevated temperature nuclear construction.

A path forward for qualification is still not clear

Challenges Facing AM in High Temperature Nuclear Applications

- Little data on elevated temperature testing of AM material
  - Typical existing data does not fully capture all properties of interest
  - Often short timeframes (100-hour creep tests)
- Can witness specimens still be used
  - Room temperature tensile does not adequately describe creep, fatigue or other elevated temperature characteristics of interest
  - Elevated temperature testing is more difficult, expensive and time consuming
  - In a typical qualification, creep data can only be extrapolated by a factor between 3 to 5
- High diversity between manufacturing techniques
  - New qualification plan needed for each technique?
  - New qualification plan needed for each material within each technique?
  - This is a rapidly changing landscape (both techniques and desired materials)

Powder Metallurgy Hot Isostatic Pressing (PM-HIP)

- PM-HIP is a mature advanced manufacturing method that is used by many non-nuclear industries to fabricate structural components
  - Elimination of inspectability issues and concerns
  - Production of homogenous microstructures
  - Enabling components using near-net shape (NNS) technologies
  - Enabling new alloys systems and targeted chemistries
  - Enhancing weldability through stringent “tramp” element controls
  - Alternate supply route for long-lead time components
  - Elimination of re-work or repair of large cast components
  - Production of smaller, individual heats (lots) of material as opposed to several ton heats in NNS form or as ingots
- PM-HIP attributes are also attractive for fabricating reactor components
  - **Particularly timely for microreactors**

LWR reactor head – Courtesy EPRI

316L PM-HIP component for ASME Section III Code Case N-834

Kyffin, W., Gandy, D., Burdett, B. (2020)

Section III, Division 1 Applications - Code Case N-834

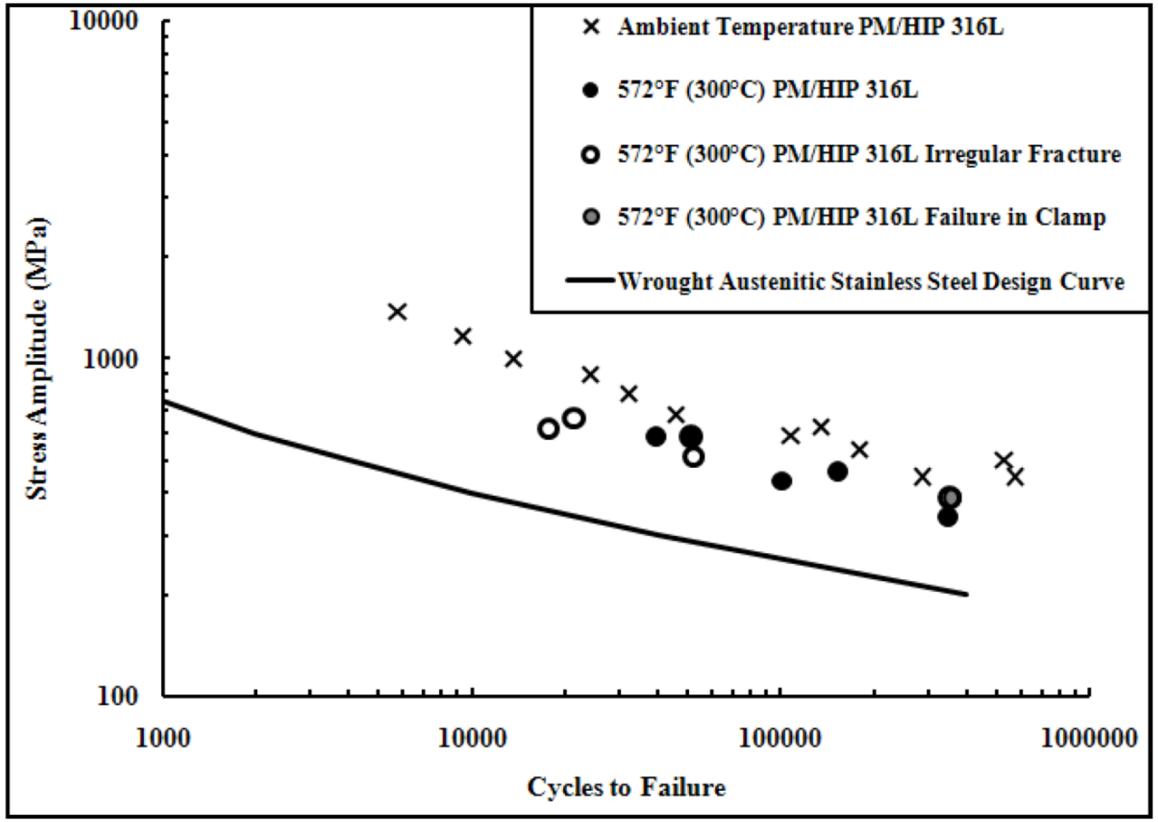
For Section III, Division 1 Subsection NB, Class 1, **316L** components fabricated by PM HIP, the wrought design methodology shall be used if the following requirements are met:

- Conform to ASTM A988/A988M-11: Standard Specification for Hot Isostatically-Pressed Stainless Steel Flanges, Fittings, valves, and Parts for High Temperature Service
- Fabrication requirements:
  - The maximum allowable powder particle size shall be 0.5 mm or less
  - Following atomization, powders shall be stored under a positive nitrogen or argon atmosphere
  - All surfaces exposed to process fluids shall be removed
- Post-fabrication requirements:
  - Density measurements and microstructural examination shall be performed in accordance with ASTM A988 paras. 8.1.1 and 8.1.2
    - 8.1.1: Measured density greater than 99% of the density typical of the wrought grade of the same heat-treated condition
    - 8.1.2: Microstructure shall be reasonably uniform and free of voids, laps, cracks, and porosity at 20-50x, 100-200x, and 1,000-2,000x
  - Meet minimum requirements for tensile strength, yield strength, elongation, and reduction of area
  - Chemical composition analysis of the final blend powder and component
  - Intergranular corrosion tests shall be performed in accordance with ASTM A262 Practice E
  - Ultrasonic examination over 100% its entire volume

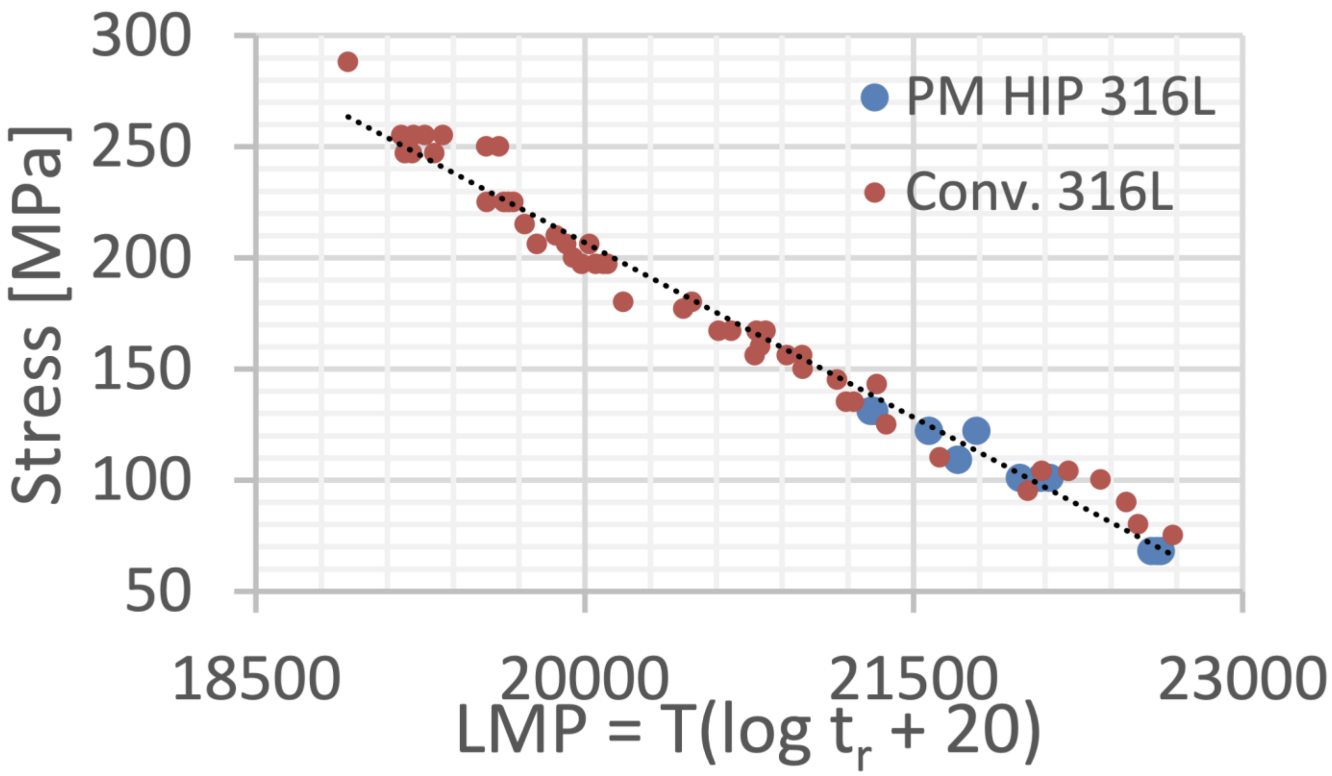
Data Package Supporting Code Case N-834

- Chemical composition
- Grain size measurements
- Hardness measurements
- Drawings and images
- Microstructure
- Density
- Inclusion content
- Toughness
- Tensile properties (70–1000°F [21–538°C])
- Yield stress-strain curves
- Weldment properties
- Fatigue properties
- Corrosion results

300°C Tests



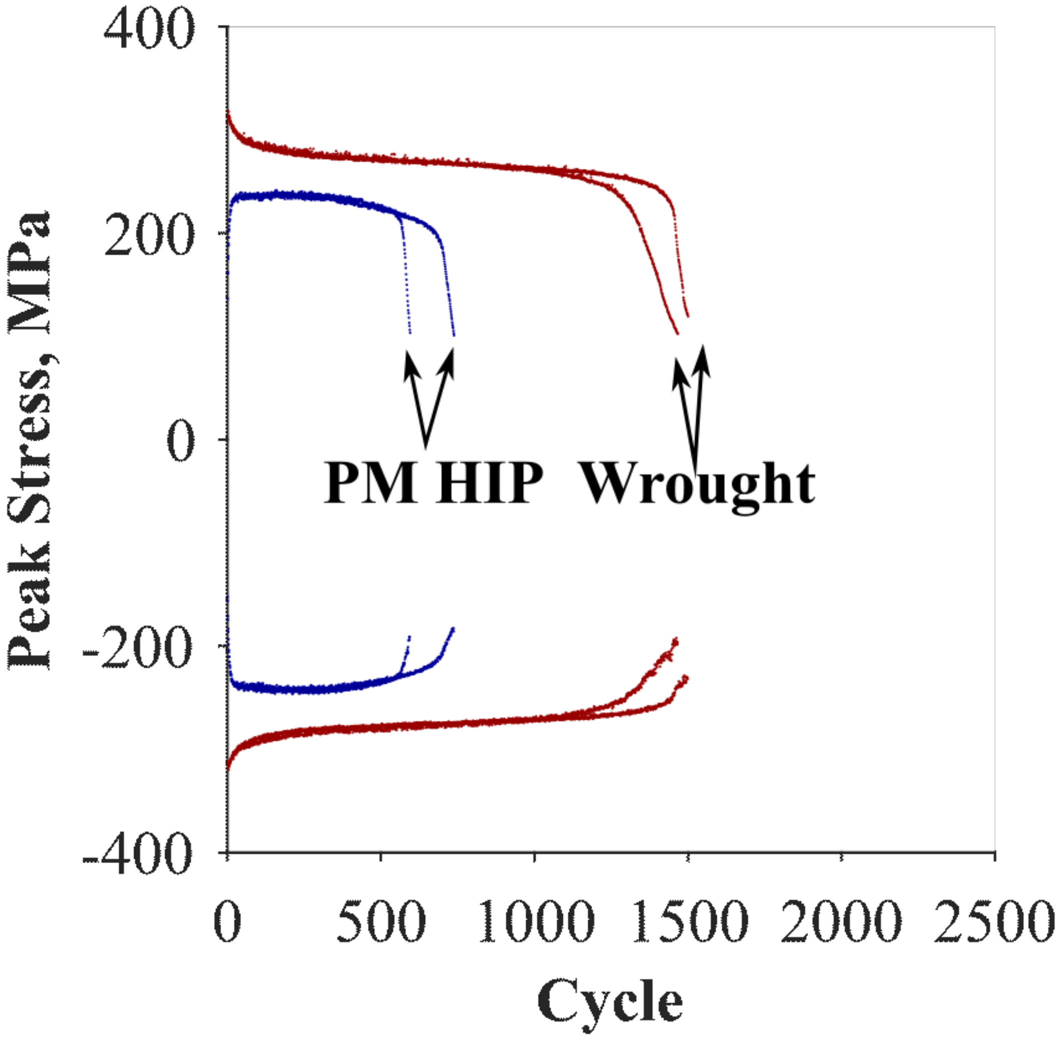
Tensile and Creep Properties Have Been Shown to be Equivalent to Wrought Material for 316L



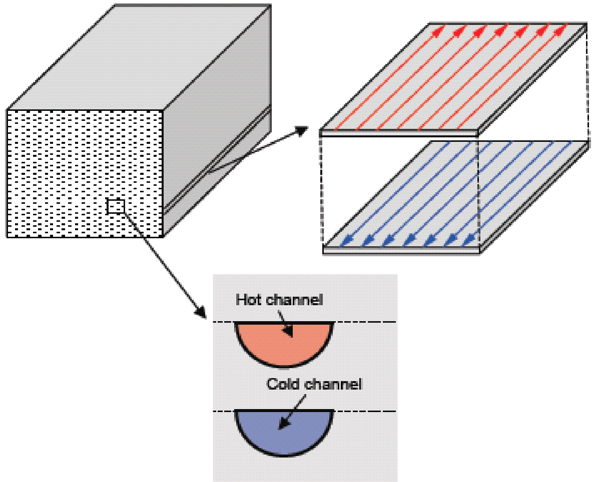
Kyffin, W., Gandy, D., & Burdett, B. (2020). A Study of the Reproducibility and Diversity of the Supply Chain in Manufacturing Hot Isostatically Pressed Type 316L Stainless Steel Components for the Civil Nuclear Sector. *Journal of Nuclear Engineering and Radiation Science*, *6*(2).

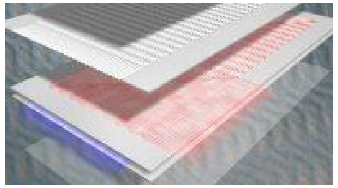
High Temperature Cyclic Properties of PM-HIP 316L – INL Data Creep-Fatigue at 650°C

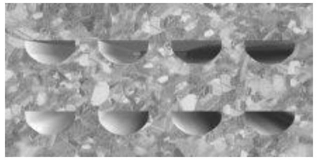
30 min. hold



Diffusion Bonded (Welded) Compact Heat Exchanger





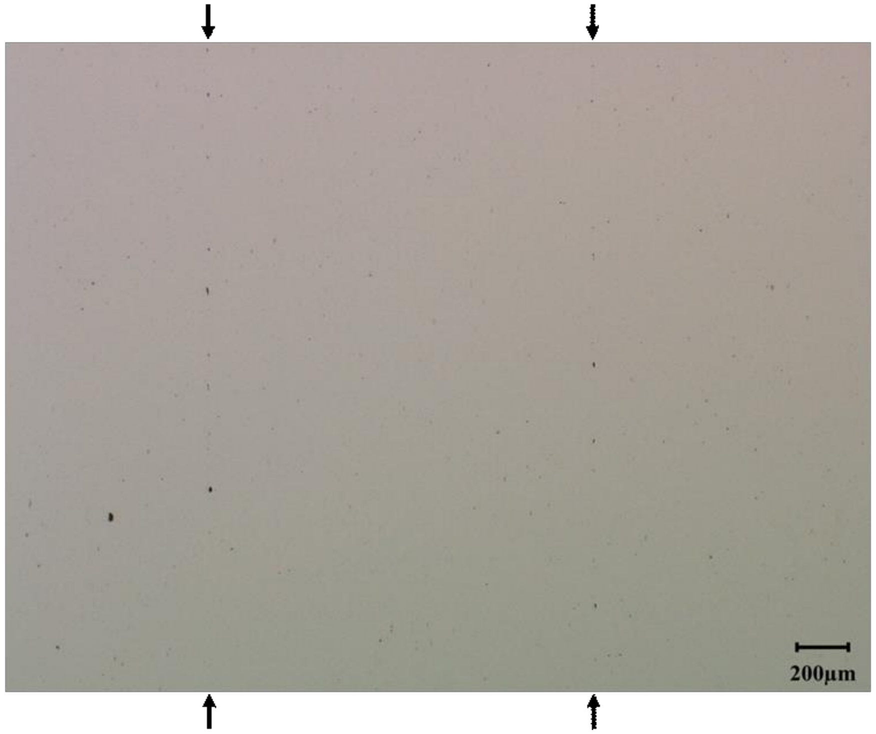


The diffusion-bonded heat exchanger in the foreground undertakes the same thermal duty, at the same pressure drop, as the stack of three shell and tube exchangers behind.

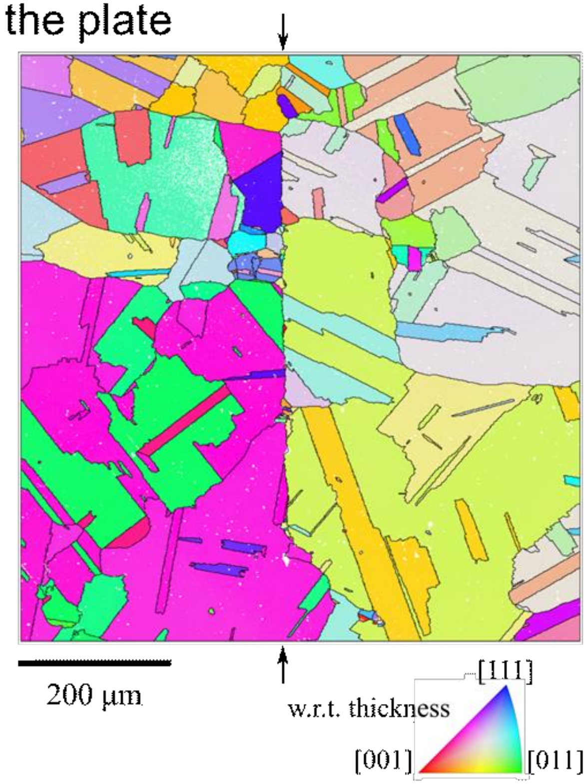
Diffusion Welded Alloy 617 Sheets

- Porosity is present at the DW interfaces of BP8I
- Grain growth across the DW interface occurred in some regions

Optical microscopy image of BP8I from the middle of the plate at 100x

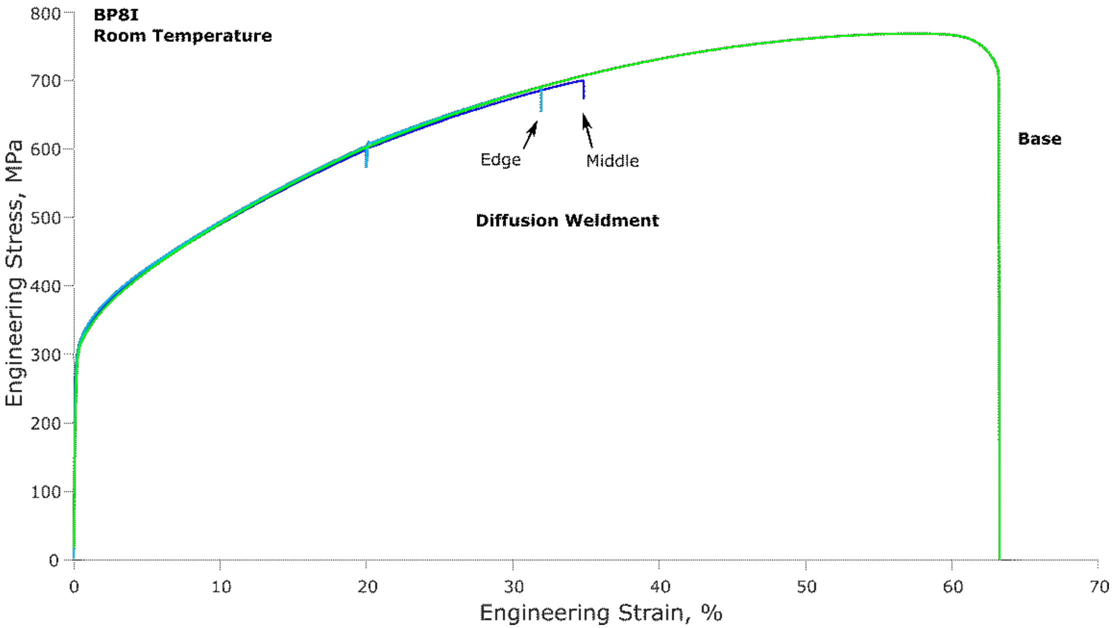


IPF map from EBSD data of BP8I from the middle center of the plate



Tensile Properties of Diffusion Welded Alloy 617

- Room-temperature tensile tests met the requirements of QW-153 in Section IX of the Code
- Tensile facture occurs in a brittle manner at a diffusion weld interface



Longitudinal axis is perpendicular to DW interfaces

	Tensile Strength (MPa)	Yield Strength (MPa)	Elongation (%)
BP8K[1]	730 ± 15	318 ± 4	37.1 ± 1.4
BP8I edge	690	310	30.9
BP8I middle	700	310	33.4
Base	765	302	63.3
Sec. IX QW-185	655	-	-

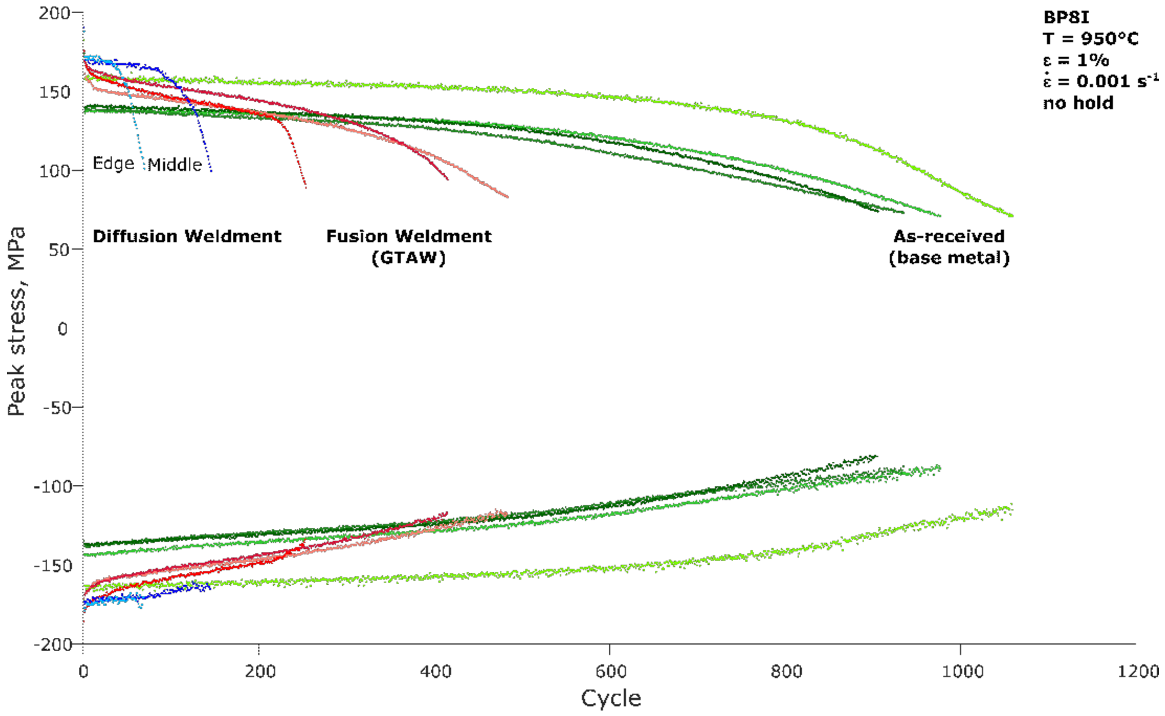
Edge

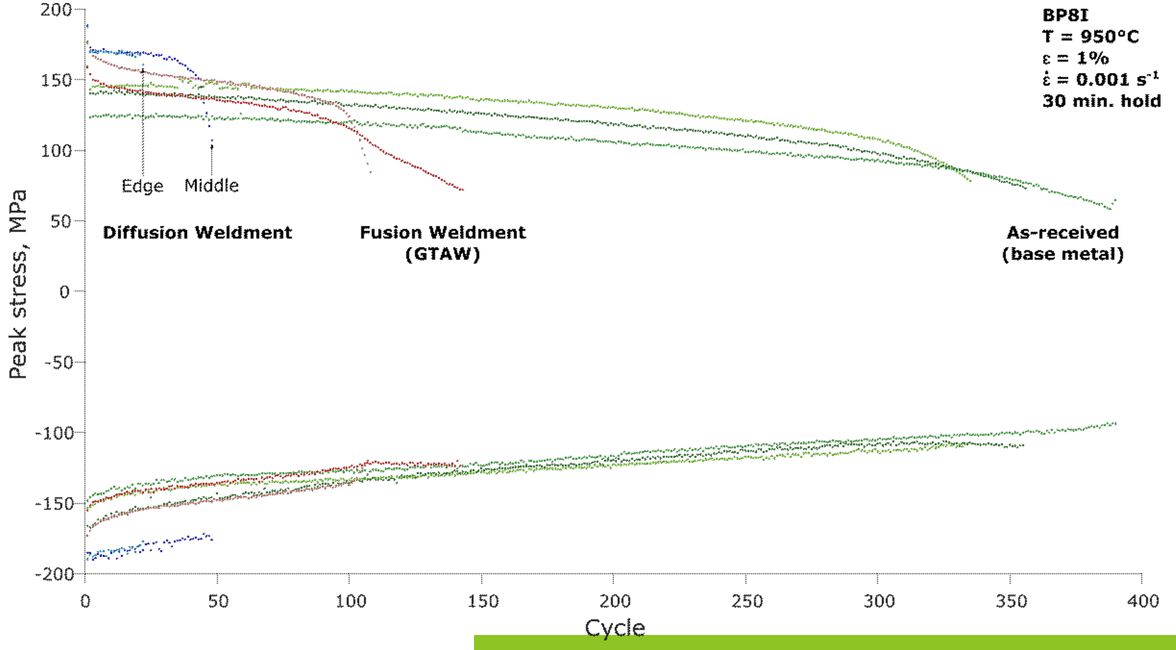
Middle

[1] Sah, I., Hwang, J. B., Kim, W. G., Kim, E. S., & Kim, M. H. (2020). High-temperature mechanical behaviors of diffusion-welded Alloy 617. *Nuclear Engineering and Design*, *364*, 110617.

Fatigue Properties of Diffusion Welded 617 Compared to Fusion Welded and Base Metal

- The number of cycles to failure of the DW metal was lower than the base and GTAW fusion-welded metal
- Specimens from the edge of the DW plate has a lower number of cycles to failure than specimens from the middle of the plate





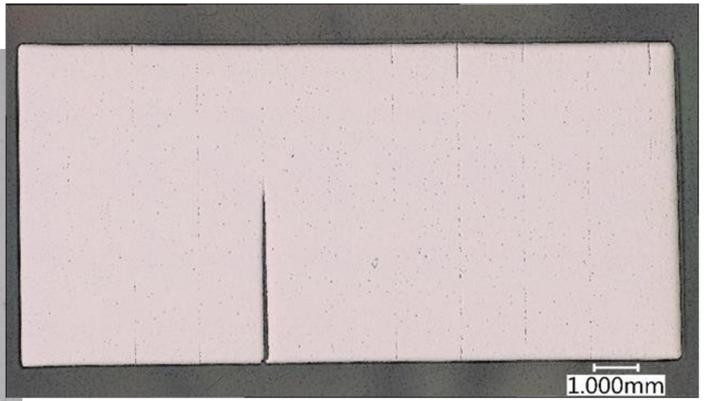
Fractography of Failed Fatigue Samples

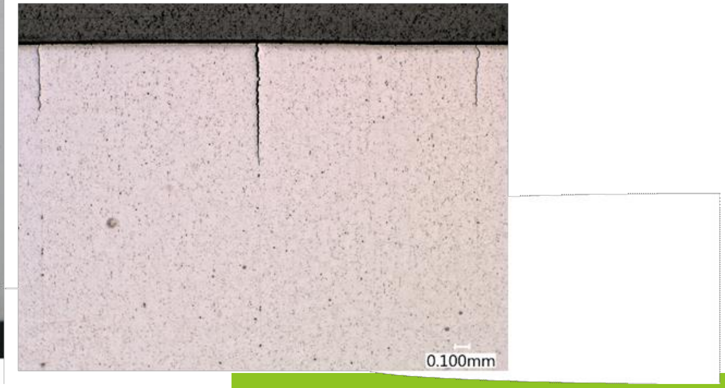
- The cyclic life is limited by the weakest interface.
- Interface quality is improved by grain growth across the interface.

Edge (N_f = 61)

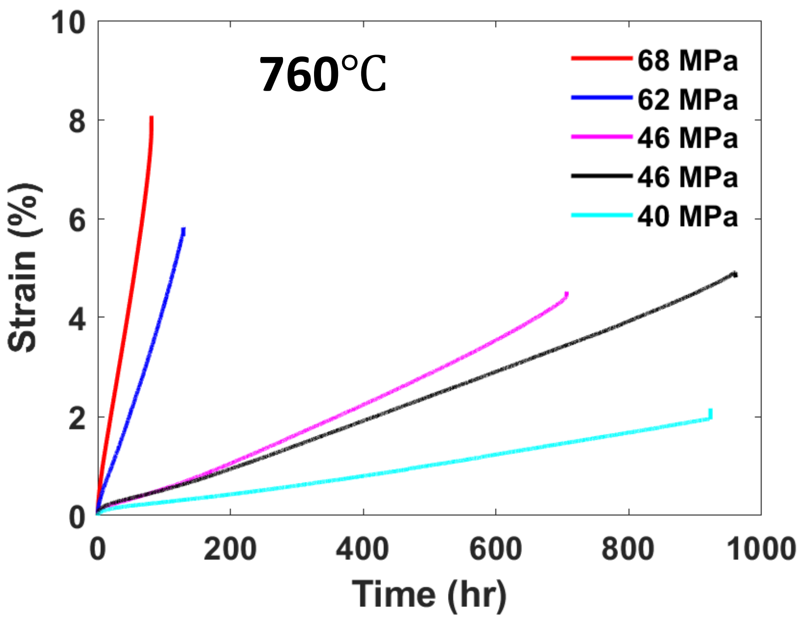
950°C, ε = 1%, no hold

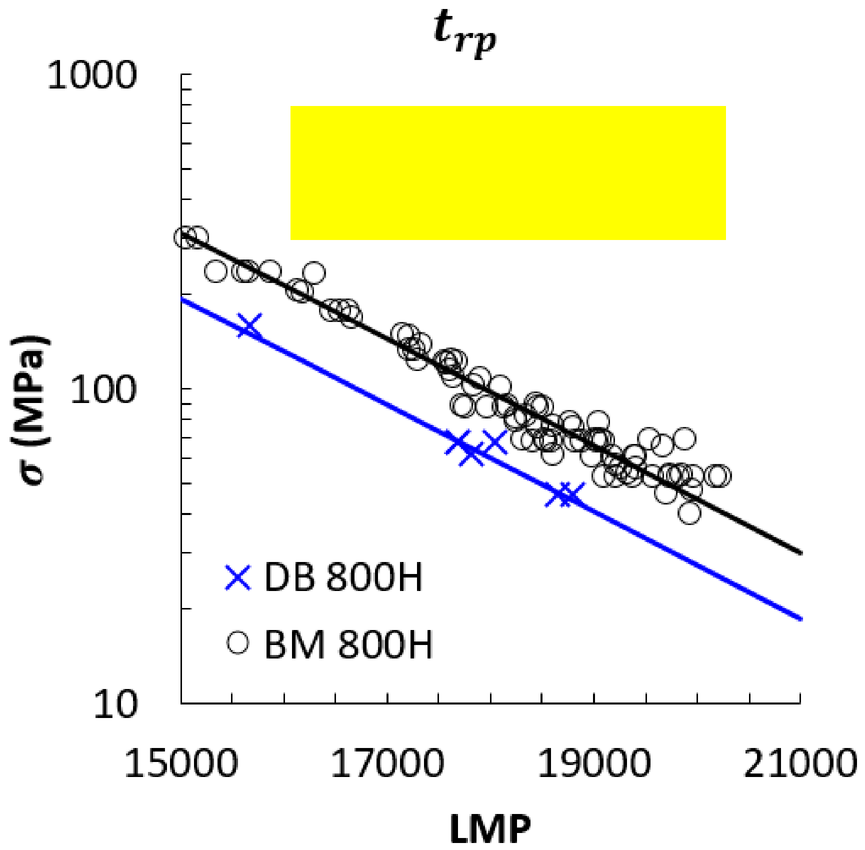
Middle (N_f = 136)



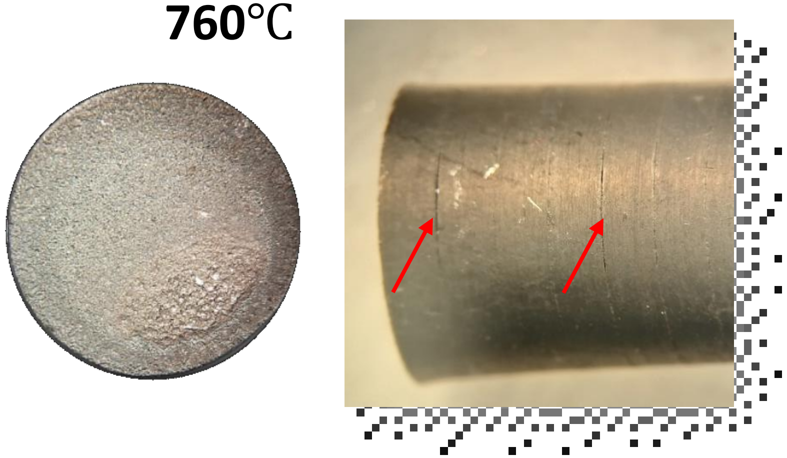


Creep Performance of Diffusion Welded Alloy 800H Sheets



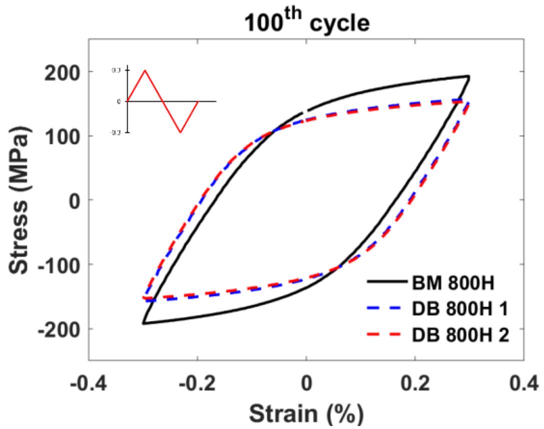


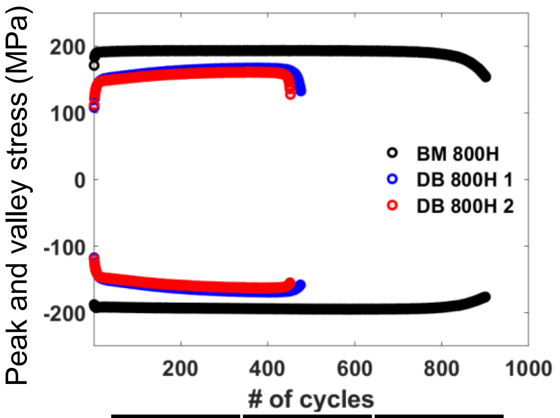
760℃



Fatigue Tests of Diffusion Welded Alloy 800H Sheets

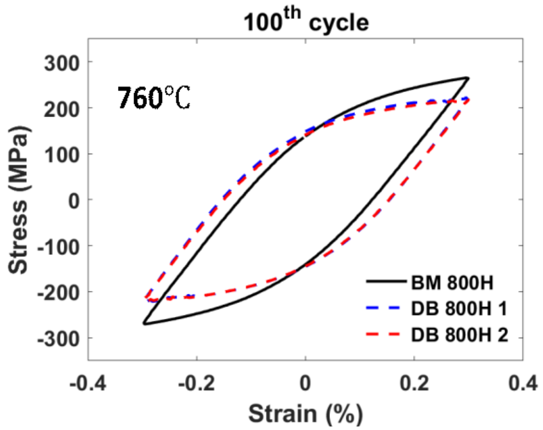
760℃

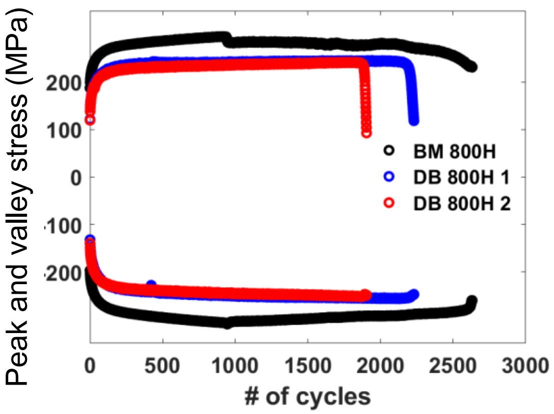




BM	DB1	DB2
901	476	452

650℃

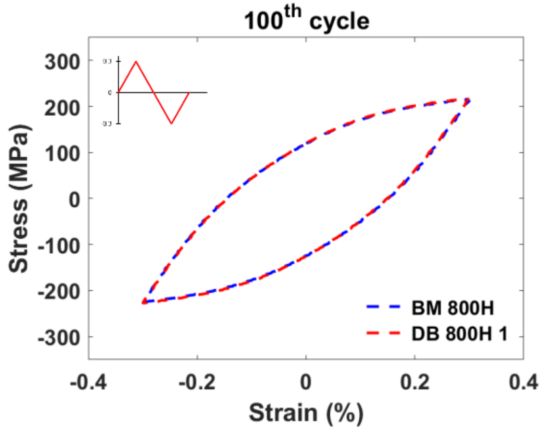


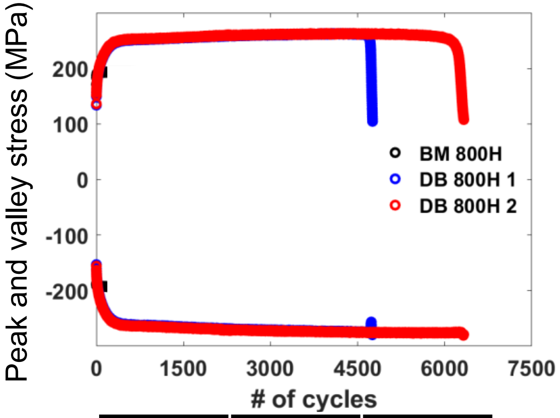


BM	DB1	DB2
259 7	221 3	189 4

Due to noise in BM800H test, it will be repeated.

550℃





BM	DB1	DB2
--	473 2	625 4

Suggested Reading

- ASME Boiler and Pressure Vessel Code, 2019 Edition, Section III Division 1 and Section III Division 5, Section IX Welding and Section XI Non-Destructive Examination.
- C. Cabet, A. Mannier, and A. Terlain, 2004, "Corrosion of High Temperature Alloys in the Coolant Helium of a Gas Cooled Reactor," Materials Science Forum, Vols. 461-464, pp. 1165-1172.
- Next Generation Nuclear Plant Reactor Pressure Vessel Materials Research and Development Plan INL Document PLN-2803, Rev. 1, 2010.
- Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan INL Document PLN-2804, Rev. 1, 2010.
- R. Wright, 2014, "Creep of A508/533 Pressure Vessel Steel," INL External Report 14-32811, Rev. 0
- X. Yan, 2016, "Very high-temperature reactor", Handbook of Generation IV Nuclear Reactors, pp. 5-90
- NGNP High Temperature Materials White Paper INL/EXT-09-17187, Rev. 1, 2012
- W. O'Donnell, A. Hull, S. Malik, 2008, "Structural Integrity Code and Regulatory Issues in the Design of High Temperature Reactors," Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology
- W. O'Donnell, D. Griffin, 2007, "Regulatory Safety Issues in the Structural Design Criteria of ASME Section III Subsection NH and for Very High Temperatures for VHTR & GEN IV," Final Report for ASME Gen IV Materials Project
- NUREG/CR-6944, "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), Vol. 4, High Temperature Materials PIRTs," 53 pp, (March 2008).
- J. Wright, 2015 "Draft ASME Boiler and Pressure Vessel Code Case for Use of Alloy 617 for Class A Elevated Temperature Service Construction", INL/EXT-15-36305
- J. Wright et al, 2016 "Determination of the Creep-Fatigue Interaction Diagram for Alloy 617," Proceedings of the ASME 2016 Pressure Vessels and Piping Conference

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