ASME Code Section III Division 5: Rules of Construction for High Temperature Reactors

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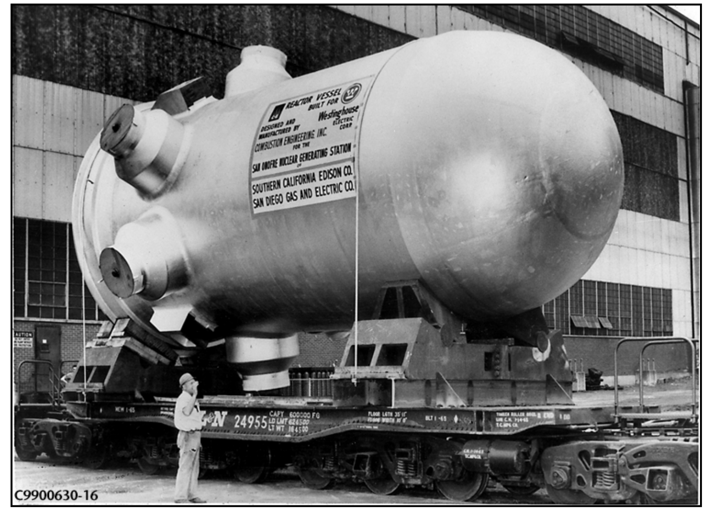
ASME Codes & Standards Are Integral Parts of Nuclear Plant Construction and Licensing

- **ASME Codes & Standards**
  - Provide rules for safe construction & in-service inspection
- **Owner/Operator**
  - Applies for plant license
  - Responsible for Code Implementation
  - Provides input on needed Code rules
- **Department of Energy**
  - Develops technical input for Code updates & improvements
- **Nuclear Regulatory Commission**
  - Reviews and may endorse Code rules to facilitate licensing

ASME Codes & Standards Provide Valuable Pathway to Apply Results from Advanced Reactor R&D

ASME Section III Treats Metallic Materials for Low & High Temperatures Separately

- **Allowable stresses for LWR & low-temperature advanced reactor components not time dependent**
  - **< 700°F (371°C) for ferritic steel and < 800°F (427°C) for austenitic matls**



PWR RPV

THTR Steam Generator

- **At higher temps, materials behave inelastically and their allowable stresses are explicit functions of time & temperature**
  - **Must consider time-dependent phenomena such as creep, creep-fatigue, relaxation, etc.**

Division 5, Specifically Developed to Address High Temperature Reactors, Was Issued in November 2011

- Sec III Div 5 contains construction and design rules for high-temperature reactors, including gas-, metal- & salt-cooled reactors
- Covers low temperature metallic components, largely by reference of other portions of Sec III
- Covers high-temperature metallic components explicitly, including former
  - Sec III, Subsections NG (Core Supports) & NH (Elevated Temperature Components)
  - Relevant Code Cases addressing time-dependent behavior
- Also includes rules for graphite & ceramic composites for core supports & internals for first time in any international design code
- Numerous technical issues in Div 5 have been identified. Some have been and others are being addressed

Materials and High Temperature Design Methods in Div 5 Need Updating*

- **Weldments**
  - Weldment evaluation methods, metallurgical & mechanical discontinuities, transition joints, tube sheets, validated design methodology
- **Aging & environmental issues**
  - Materials aging, irradiation & corrosion damage, short-time over-temperature/load effects
- **Creep and fatigue**
  - Creep-fatigue (C-F), negligible creep, racheting, thermal striping, buckling, elastic follow-up, constitutive models, simplified & overly conservative analysis methods
- **Multi-axial loading**
  - Multi-axial stresses, load combinations, plastic strain concentrations

**Based on Multiple DOE, NRC & National Lab Reviews of High Temperature Reactor Issues over Past 40 Years*

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Div 5 Materials and High Temperature Design Methods Need Updating *(cont)*

- **Materials allowables**
  - Elevated temperature data base & acceptance criteria, min vs ave props, effects of melt & fab processes, 60-year allowables
- **Failure criteria**
  - Flaw assessment and LBB procedures
- **Analysis methods and criteria**
  - Strain & deformation limits, fracture toughness, seismic response, core support, simplified fatigue methods, inelastic piping design, thermal stratification design procedures
- **NRC Endorsement of Div 5 & associated Code Cases**
  - Alloy 617
  - Strain Limits for Elevated Temp Service Using E-PP Analysis
  - Creep-Fatigue at Elevated Temp Using E-PP Analysis

DOE Advanced Reactor Technology R&D Supports Resolution of These Issues Plus Development & Qualification of Data Required for Design

High Priority ASME Code Committee Actions Endorsed by BNCS and DOE

Topics	2017 Edition	Beyond 2017
New simplified analysis methods (EPP) that replace current methods based on linear analysis	X	
Adequacy of the definition of S values used for the design of Class B components, which is based on extrapolated properties at 100,000 hours, in light of application to 500,000 hours design	X	
Construction rules for “compact” heat exchanges		X
Incorporation of new materials such as Alloy 617 and Alloy 709 (austenitic stainless)	A617	A709
Pursuit of “all temperature code”		X
Complete the extension of Alloy 800H for 500,000 hr-design	X	
Complete the extension of SS304, 316 for 500,000 hr-design	X	
Complete the extension of Grade 91 for 500,000 hr-design	X	
Thermal striping		X
Develop design by analysis rules for Class B components (including compact HX)		X
Component classification (Refer back to ANS 53 classification rules), including assessment of: Is Class B really necessary?	X	
Add non-irradiated and irradiated graphite material properties		X

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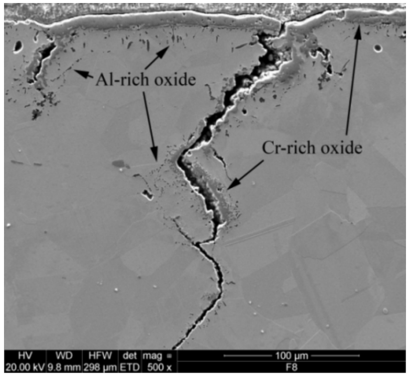
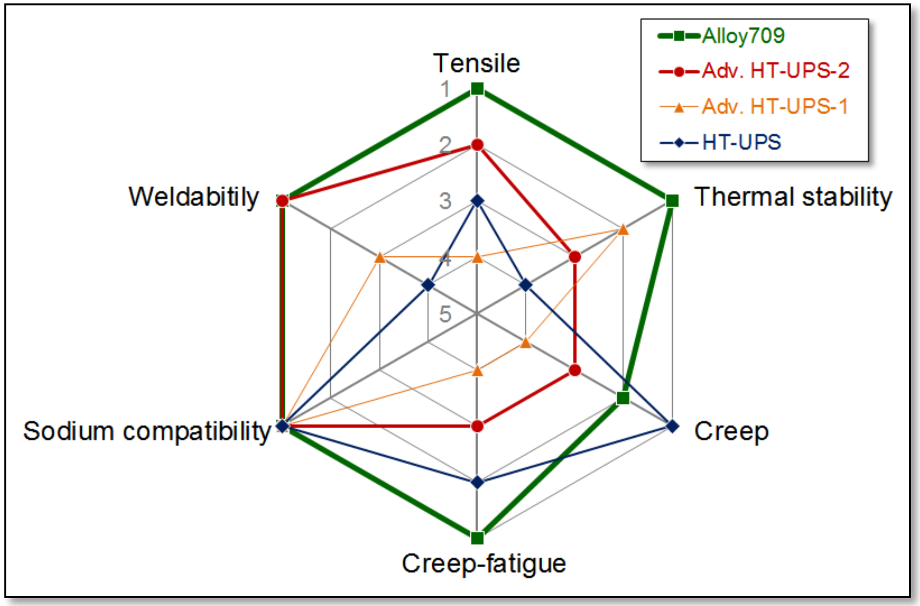
Additional High-Temperature Alloys, Now Being Qualified, Will Provide Additional Options for Nuclear Construction

- **Alloy 617 Code Case currently in review process**
  - Advanced gas reactor heat exchangers & steam generators up to 950°C and 100,000 hrs
  - Low-temperature Code Case (T < 427°C) submitted May 2014 and high-temperature in Sept 2015
  - Anticipate inclusion in 2019 edition of Sec III Div 5



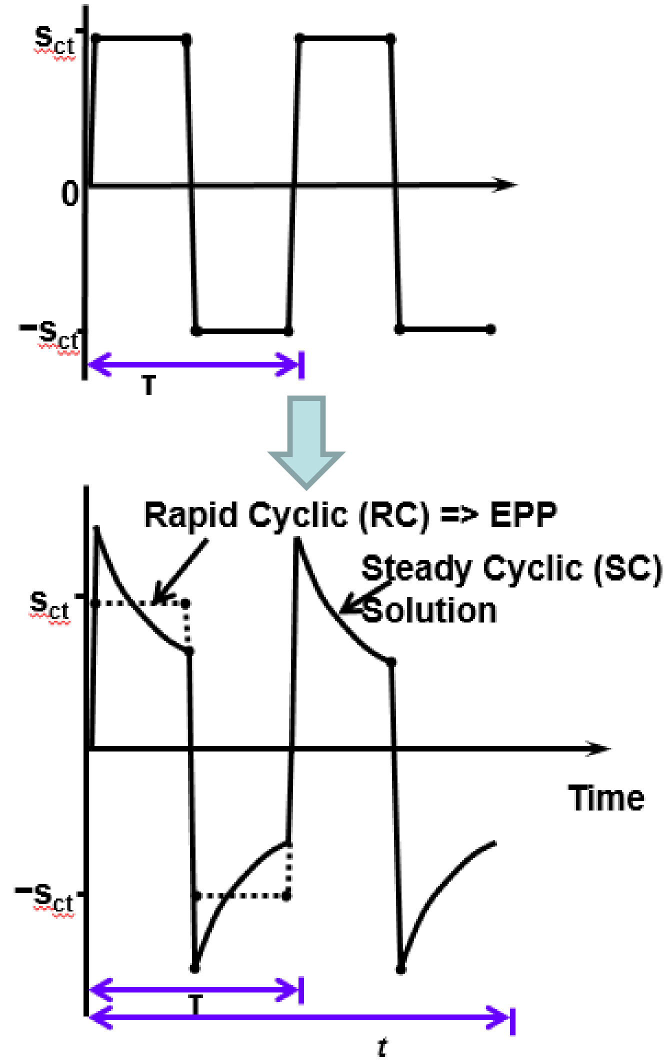
Creep-fatigue crack 617

- **Alloy 709 selected for Code qualification**
  - Will provide improved performance, design envelop, and cost reduction for LMRs
  - Roughly double existing creep strength of existing stainless steels in Sec III Div 5
  - Detailed qualification plan prepared and testing begun



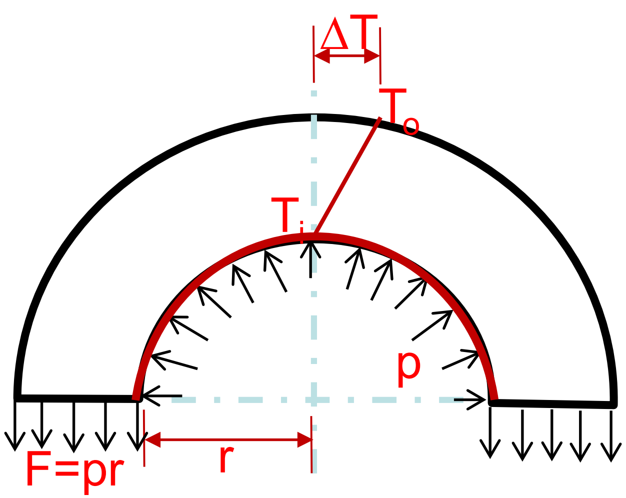
Improved Components of High-Temperature Design Methodology Being Developed

- **Improved design rules, based on elastic-perfectly plastic analysis, proposed for strain limits & creep-fatigue**
  - **Critical for very high temperatures where no distinction exists between creep and plasticity**
    - **Current rules invalid at very high temperatures**
    - **Will enable simplified methods for Alloy 617 > 1200°F (649°C)**
  - **E-PP analysis addresses ratchetting & shakedown**
  - **Avoids stress classification**
- **Yield strength is a "pseudo" strength given by the limiting design parameter, e.g. stress for 1% inelastic strain**
- **The Rapid Cycle (RC) is limiting case that bounds the real Steady Cyclic (SC) solution**

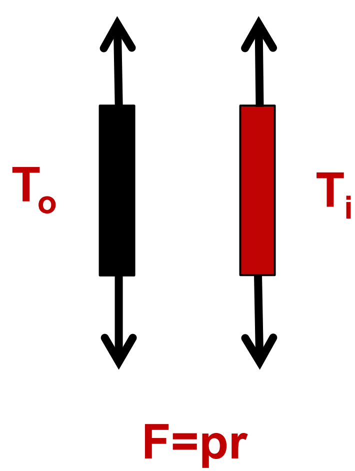


Advances in High Temperature Design Methodology Are Being Validated through Key Features Tests

- **Two-bar tests can simulate combined thermal transients and sustained pressure loads that can generate a ratchet (progressive deformation) mechanism during creep-fatigue, relaxation, elastic follow-up, etc.**
  - **Validation of the E-PP model under varying effects of thermal path and mean stress**



Pressurized cylinder with radial thermal gradient

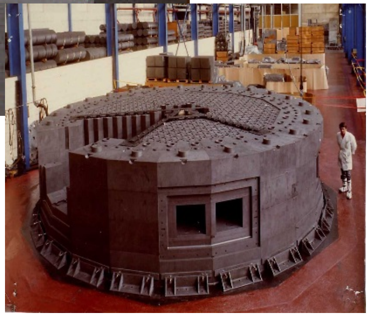


Equivalent Two-bar model

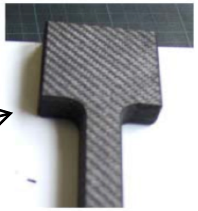
- **Equal deformations**
- **Pressure stress in vessel wall represented by total load on bars;**
- **Through-wall temperature gradient represented by temperature difference between bars**

Technical Bases for Code Rule Development of Graphite and Ceramic Composites Continuing to Expand

- Graphite used for core supports in HTGRs, VHTRs and FHRs
  - Maintain core geometry and protect fuel
  - Includes current and future nuclear graphites
- Special graphite considerations for Code rules
  - Lack of ductility
  - Need for statistically set load limits
  - Requires irradiation and oxidation data
- Ceramic composites (e.g. SiC-SiC) for internals & controls for gas, liquid-metal & salt cooled systems
  - Very high temperature and irradiation resistance
  - $Dose_{max}$ > 100 dpa, $T_{max} \geq 1200°C$
  - Materials specification, design, properties, testing, examination, and reporting rules developing



Graphite Core Supports



Core Tie Rod

ASME Nuclear Code Activities Are Critical for Advanced Reactor Development

- **ASME Roadmaps for sodium- and gas-cooled reactor materials and design code needs developed**
- **Sec III Div 5 Construction Rules for HTRs were issued in 2011**
  - **Covers high temperature metals and design methods, as well as graphite and ceramic composites**
- **ASME and DOE jointly established priority list for needed Div 5 rules improvements in 2015 and are pursuing them**
- **NRC has begun to assess ASME Div 5 for endorsement**
  - **Very important since predecessor ASME docs never endorsed**
  - **Will facilitate HTR design process and enhance regulatory surety**
- **DOE materials program providing technical basis to address ASME Code improvements and NRC endorsement concerns**

Points of Contact for Additional ASME Div 5 Information

- **DOE Activities**
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  - **Metals and Design Methods - Sam Sham: ssham@anl.gov**
  - **Graphite - Tim Burchell: burchelltd@ornl.gov**
  - **Composites - Yutai Katoh: katohy@ornl.gov**
- **NRC Review for Endorsement**
  - **George Tartal: george.tartal@nrc.gov**
  - **Matthew Mitchell: matthew.mitchell@nrc.gov**
  - **Steven Downey: steven.downey@nrc.gov**

QUESTIONS?

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Reviews for Advanced Reactors Found Shortcomings in High-Temp Metals & High-Temp Design Methodology (HTDM)

- **NRC/ACRS Review of Clinch River Breeder Reactor in mid-1980's [1]**
- **GE's PSID for PRISM 1986 – NRC Generated PSER in 1994 [2]**
- **ORNL Review for NRC of ASME Code Case N-47 (now NH and Div 5A) in 1992 [3]**
- **NRC Review and Assessment of Codes and Procedures for HTGR Components in 2003 [4]**
- **DOE-funded ASME/LLC Regulatory Safety Issues in Structural Design Criteria Review of ASME III NH in 2007 [5]**
- **NRC-sponsored Review of Regulatory Safety Issues for Elevated Temperature Structural Integrity for Next Generation Plants in 2008 [6]**

These reviews cumulatively identified over 40 individual concerns, but can be summarized under 8 key areas

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References for High Temperature Reactor Materials and Design Methods Reviews

1. Griffen, D.S., “Elevated-Temperature Structural Design Evaluation Issues in LMFBR Licensing,” Nuclear Engineering and Design, 90, (1985), pp. 299-306
2. NUREG-1368 “Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor,” Feb. 1994
3. NUREG/CR-5955, Huddleston, R.L. and Swindeman, R.W., “Materials and Design Bases Issues in ASME Code Case N-47,” ORNL/TM-12266, April 1993
4. NUREG/CR-6816, Shah, V.N., S. Majumdar, and K. Natesan,“Review and Assessment of Codes and Procedures for HTGR Components,” ATL-02-36, June 2003.
5. O’Donnell, W. J., and D. S. Griffin, “Regulatory Safety Issues in the Structural Design Criteria of ASME Section III Subsection NH and for Very High Temperatures for VHTR and Gen-IV,” ASME-LLC STP-NU-010, Dec. 2007
6. O’Donnell, W.J., Hull, A.B., and Malik, S., “Historical Context of Elevated Temperature Structural Integrity for Next Generation Plants: Regulatory Safety Issues in Structural Design Criteria of ASME Section III Subsection NH,” Proceedings of 2008 ASME Pressure Vessel and Piping Conf., PVP2008-61870, July 2008

ASME LLC Tasks Funded by DOE-NE

- **Task 1 Verification of Allowable Stresses in ASME Section III, Subsection NH with Emphasis on Alloy 800H and Modified 9Cr-1Mo Steel**
- **Task 2 Regulatory Safety Issues in Structural Design Criteria of ASME Section III Subsection NH and for Very High Temperatures for VHTR and GEN IV**
- **Task 3 Improvement of Subsection NH Rules for Modified 9Cr-1Mo Steel**
- **Task 4 Updating of ASME Nuclear Code Case N-201 to Accommodate the Needs of Metallic Core Support Structures for High Temperature Gas Cooled Reactors Currently in Development**
- **Task 5 Collect Available Creep - Fatigue Data and Study Existing Creep - Fatigue Evaluation Procedures**
- **Task 6 Review of Current Operating Conditions Allowable Stresses in ASME Section III Subsection NH**

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ASME LLC Tasks Funded by DOE-NE (cont)

- Task 7 Evaluate ASME Code Considerations for High Temperature Reactor Intermediate Heat Exchangers
- Task 8 Creep and Creep - Fatigue Crack Growth at Structural Discontinuities and Welds
- Task 9 Update Section III Division 1 Subsection NH – Simplified Elastic and Inelastic Methods
- Task 10 Update Section III Division 1 Subsection NH – Alternative Simplified Creep – Fatigue Design Methods
- Task 11 New Materials for Section III Division 1 Subsection NH
- Task 12 NDE and ISI Technology for High Temperature Gas-Cooled Reactors (NRC-Funded)
- Task 13 Recommend Allowable Stress Values for 800H
- Task 14 Correct Allowable Stress Values for 304 and 316 Stainless Steel