U.S. DEPARTMENT OF ENERGY | Nuclear Energy

Nuclear Energy University Programs (NEUP) Fiscal Year (FY) 2013 Annual Planning Webinar

Advanced Structural Materials (RC-3)

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Grand Challenge

Structural Materials Are Critical for Advanced Nuclear Reactor Technologies

- **Development and qualification of advanced structural materials are critical to the design and deployment of the advanced nuclear reactor systems that DOE is developing**
  - High Temperature Gas Cooled Reactors (HTGRs)
  - Sodium Cooled Fast Reactors (SFRs)
  - Fluoride Salt Cooled High Temperature Reactors (FHRs)
  - Lead and Lead-Bismuth Cooled Fast Reactors (LFRs)
- **Structural materials must perform over design lifetimes for pressure boundaries, reactor internals, heat transfer components, etc. Where applicable, appropriate codes and standards must be developed and met to facilitate regulatory compliance**
- **Performance of metallic alloys for the long times and high operating temperatures associated with advanced reactor system development is supported by three separate reactor programs**

Program Overview and High Level Workscope

NGNP, ARC and SMR Programs Include High Temperature Metals R&D Activities

- **Next Generation Nuclear Plant (NGNP) performs fuels & materials qualification and reactor systems R&D for high-temperature gas cooled reactors**
  - ASME Codification and environmental effects of Alloy 617 for heat exchanger applications and A533B/508 for pressure boundaries
- **Advanced Reactor Concepts (ARC) performs R&D on fundamental nuclear technologies that enable new uses of nuclear energy**
  - Development of improved ferritic-martensitic and austenic alloys for SFRs and materials compatibility for SFR and LFR systems
- **Small Modular Reactor (SMR) supports smaller, advanced designs with longer-term licensing horizons through focused R&D**
  - High temperature design methodology issues for ASME Codification (617, 800H, T91, etc.) and environmental effects for LFR systems

NEUP Structural Materials Research Needs for Advanced Reactor Concepts Are Very Focused in FY13

- **RC-3.1 Long Term Emissivity**
  - Addresses the need to ensure the adequate capability for passive heat rejection through the reactor pressure vessel, esp. for HTGRs
  - Primarily applicable to the NGNP program, but same phenomenon is relevant for other reactor systems under ARC and SMR
- **RC-3.2 Negligible Creep**
  - Identifies the bounding conditions under which creep will not impact cyclic performance of reactor components.
  - Particularly applicable to A508/533B steel in pressure boundaries at longer times or higher temperatures than LWRs experience
- **RC-3.3 Creep Fatigue**
  - Improve predictions of creep and fatigue interactions for materials for high temperature reactors (617, 800H, T91, etc.), especially regarding limitations of available supporting experimental data

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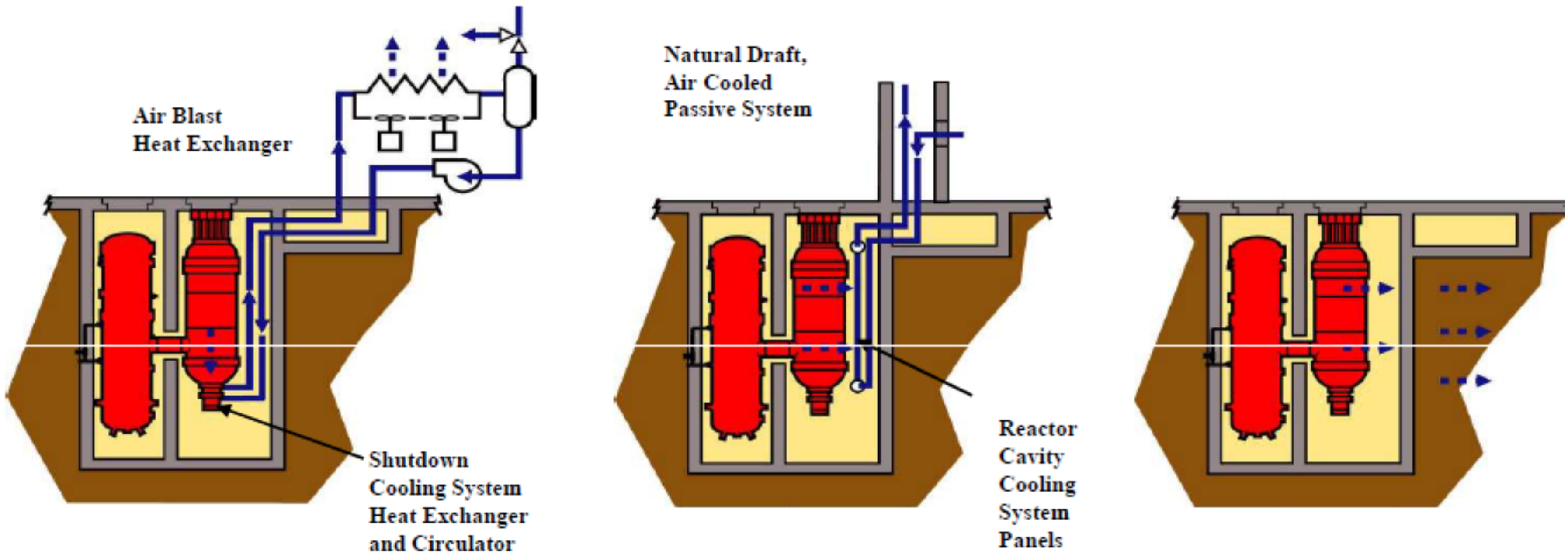
Residual Heat Removal Paths Include Passive Radiation When Active Heat Transport Systems Are Unavailable



A) Active Shutdown Cooling System

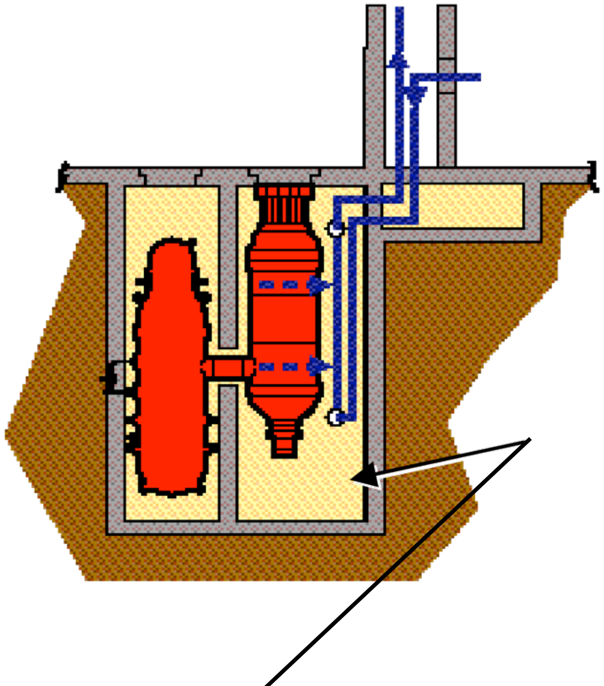
B) Passive Reactor Cavity Cooling System

C) Passive radiation and conduction of residual heat to reactor building (Beyond Design Basis Event)

Adequate Long Term Emissivity of Critical Surfaces Needs to Be Ensured for Full HTGR Lifetime

Adequate Emissivity Needs to Be Ensured for Full Reactor Lifetimes

- Addresses the long-term need to ensure the adequate capability for passive heat rejection through the reactor pressure vessel
- Primarily applicable to the NGNP program but the same phenomenon may be relevant for other reactor systems under ARC or SMR
- Passive heat rejection through the RPV provides ultimate heat sink assumed by PRA for HTGRs



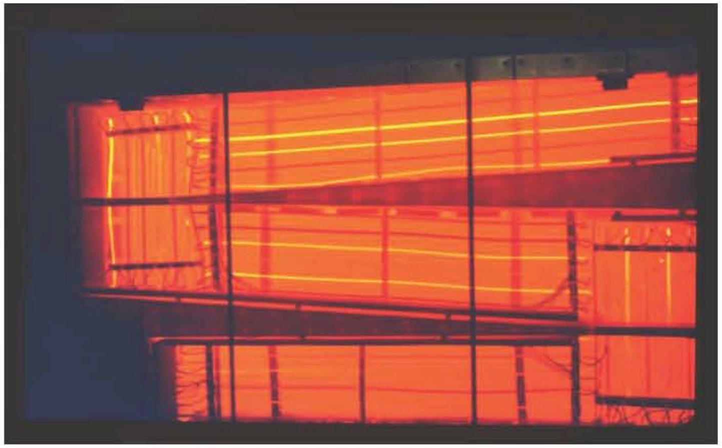
Passive cooling by radiation to water or air panels or ground

Long Term Heat Rejection Capability Through RPV Surface Must Be Ensured

- **Provides ultimate rejection of decay heat under accident conditions (e.g., PLOFC & DLOFC)**
- **Short-term studies have demonstrated emissivities from 0.3 for cleaned to 0.9 for oxidized surfaces (> ~ 0.8 required)**
- **Composition and long-term stability of corrosion products must be evaluated**
  - RPV and core barrel surfaces (inner and outer)
  - Oxide type and morphology critical
  - Effects of environment and potential long-term, near-surface elemental depletion must be considered
  - Additional surface treatments or coating may be needed

Validated, Long Term Predictive Models of Emissivity Are Needed

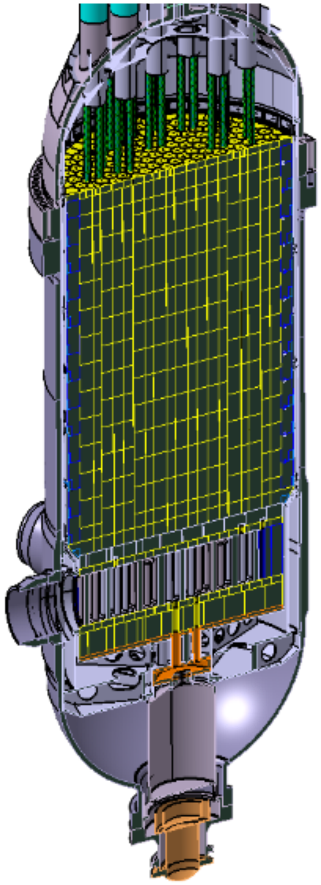
- Proposals are solicited to develop and validate models of emissivity for relevant RPV and internals materials for up to 60 years of operation in air and relevant reactor coolants



- Emissivity for HTGR materials and coolant is of primary concern, though other advanced systems (e.g., SFR, FHR, etc.) should be considered
- Materials of interest include A533B/A508, Alloy 800H, 304 and 316 stainless steel, 9Cr-1MoV, and other alloys for advanced nuclear reactor systems

Negligible Creep Limits for RPV Materials Are Not Fully Defined

- Negligible creep is condition where creep does not impact cyclic performance
- In LWR RPV design, no time-dependent deformation is considered, hence no creep-fatigue interactions are required
- For advanced reactors, RPVs will operate for longer times (60-year design) and possibly both higher operating and excursion temperatures, hence negligible creep is a potential concern

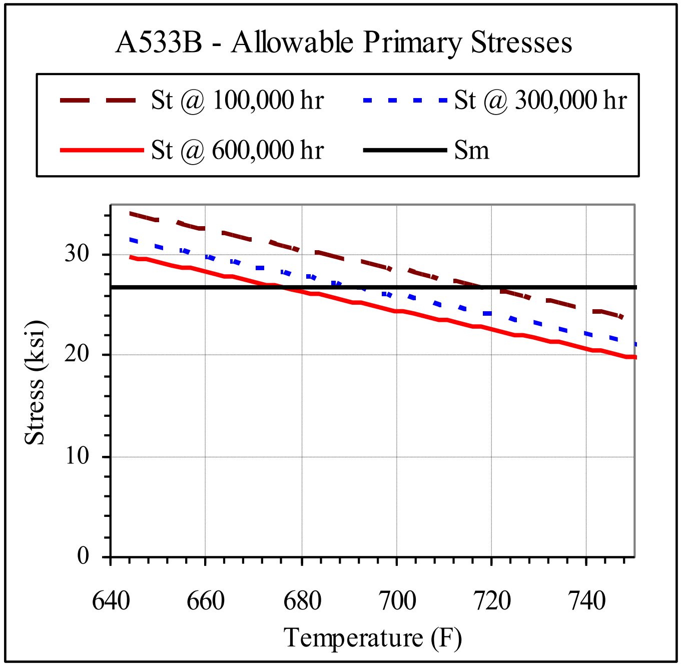


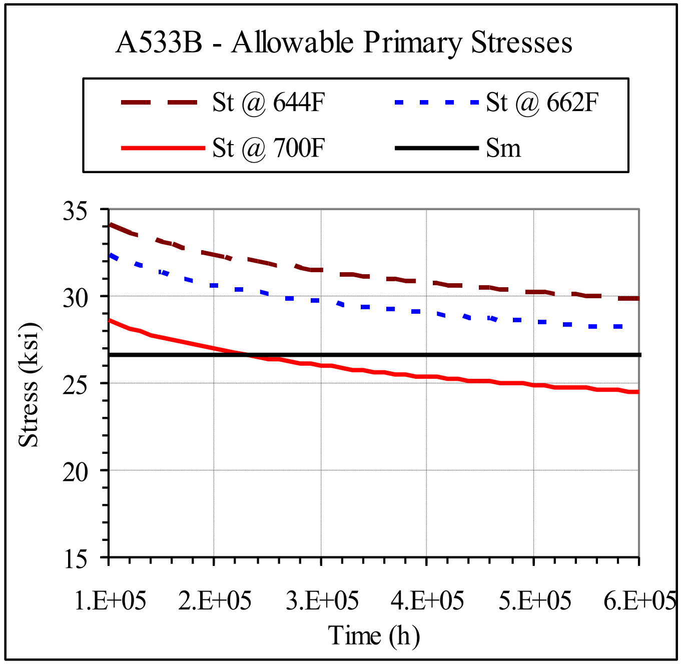
HTGR RPV

ASME Code Rules Developed for LWRs May Not Be Conservative for HTGRs

- **A533B/A508 currently used in light water reactor vessels**
  - Nominal operating temperatures from 500 to 575ºF (260 to 302ºC)
- **Proposed vessel design conditions for HTGRs**
  - Nominal operating temperature up to approximately 662ºF (350ºC) for 500,000 to 600,000 hours service life
- **Below 700ºF (371ºC) ferritic materials are usually assumed to behave in a time-independent fashion, but the very long times and potentially higher temperatures for HTGRs bring this into question**
- **ASME rules are extrapolated from a small database with relatively short creep rupture times**
- **BUT, limited data show that creep and stress relaxation CAN occur at low temperatures – “Cold Creep”**

Margin for Current ASME Time-Dependent vs Time-Independent Rules Is Slim for HTGR Vessel Designs





S_m = Time Independent Allowable Stress S_t = Time Dependent Allowable Stress

- **At 662ºF (350ºC) & 600,000 hr, ASME's allowable stress is time-independent (no creep effects), but at 700ºF (371ºC), time-dependent stresses currently govern designs for lifetimes beyond 220,000 hr—MORE DATA IS NEEDED**

Long-Term Prototypical Creep Data Is Needed to Assess Code Margins

- **Most A533B/508 available data was developed for LWR excursions**
  - Short term, high stress, and higher temperature creep rupture data
    - Generated to support ASME Code Case N499, a Code Case to set limits on short term elevated temperature excursions above 700ºF (371ºC)
- **Data is needed to evaluate potential creep effects under HTGR pressure vessel design conditions**
  - (up to ~ 662ºF/350ºC)
  - 500,000 to 600,000 hrs
- **It is impractical to generate creep data under the time/ temperature conditions of relevance**
- **Now what?**

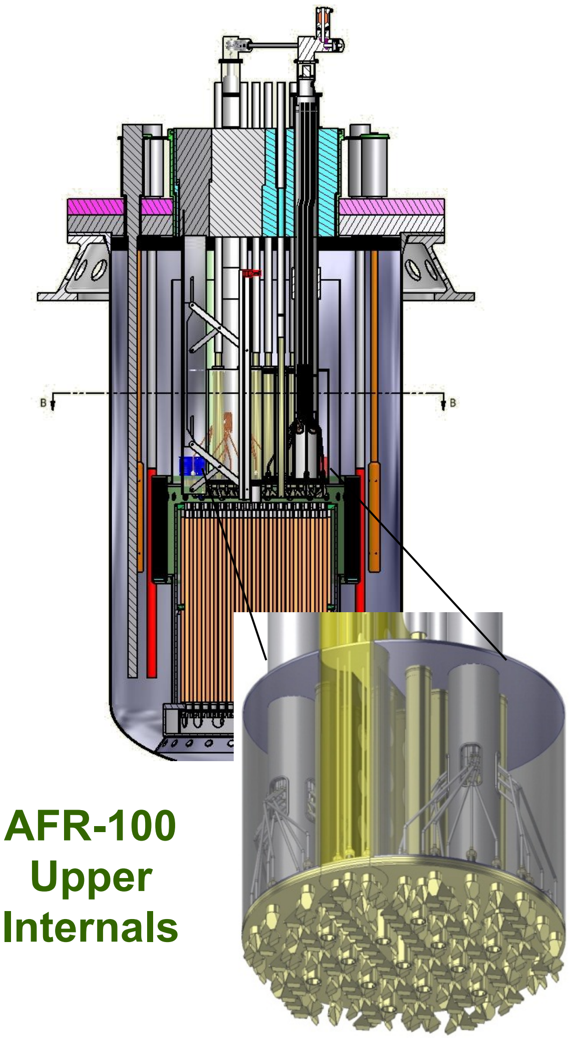
What is the Influence of HTGR Conditions on Negligible Creep?

- **Proposals are solicited to generate a mechanistic understanding of the influence of the more demanding service conditions envisioned for HTGR RPVs on negligible creep limits in A533B/508 pressure vessel steels**

- **Approaches might include:**
  - Accelerated experimental methods for exploring the boundary of negligible creep limits or developing representative data under long-term, low temperature conditions
  - Simulations of negligible creep with substantial experimental validation

More Accurate Predictions of Creep Fatigue Interaction Are Needed

- Understanding creep-fatigue interactions are critical for design and safety analysis of components in high temperature reactors operating under inelastic conditions of time, temperature, and loading
- Cumulative damage from creep and fatigue must be understood and predicted
- Many service applications generate more creep damage than fatigue damage, but most experimental CF data includes more fatigue damage
- Relative amounts of damage from either mechanism can affect the metallurgical failure processes as well as the numerical analysis correlations of the results



AFR-100 Upper Internals

Relative Creep and Fatigue Loading Fractions Influence Fracture Mechanisms

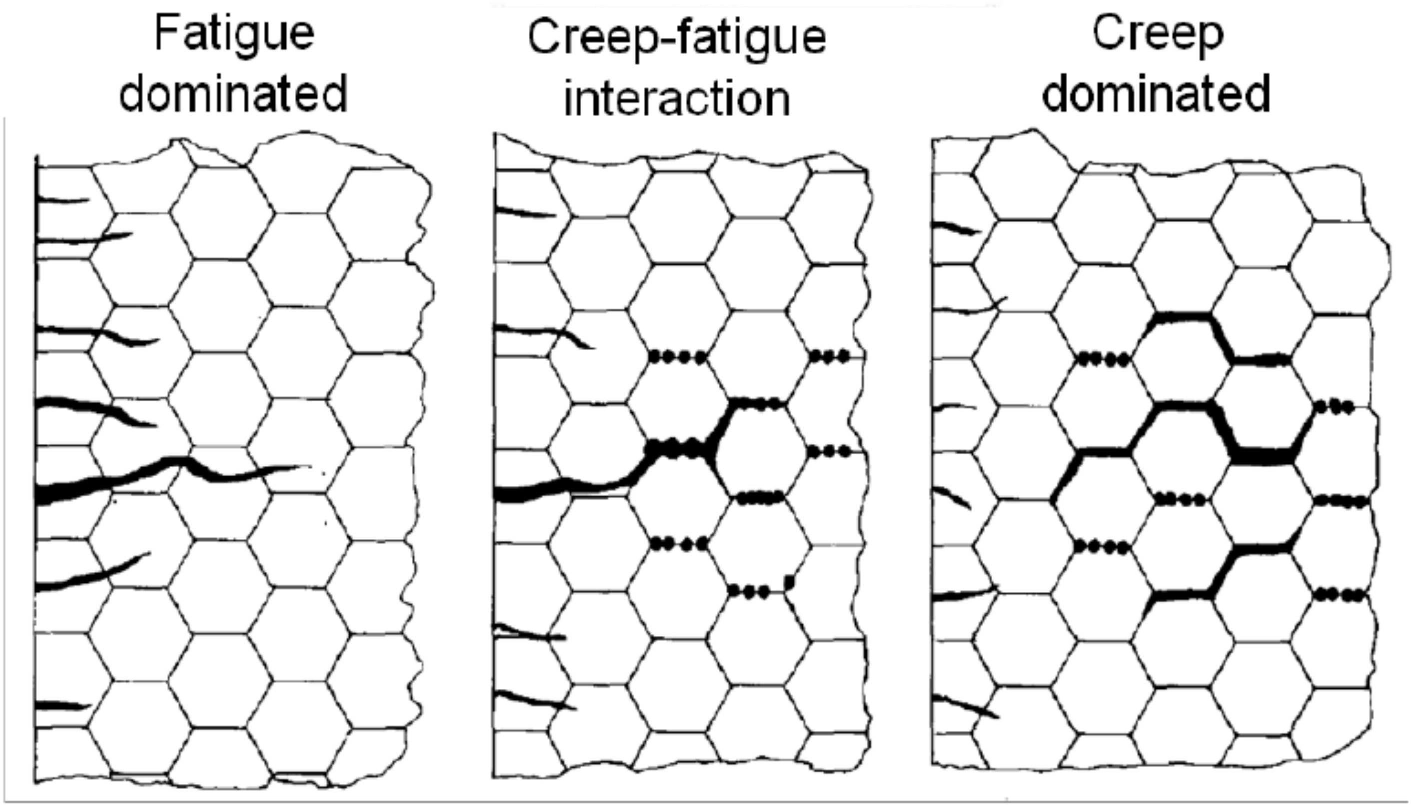
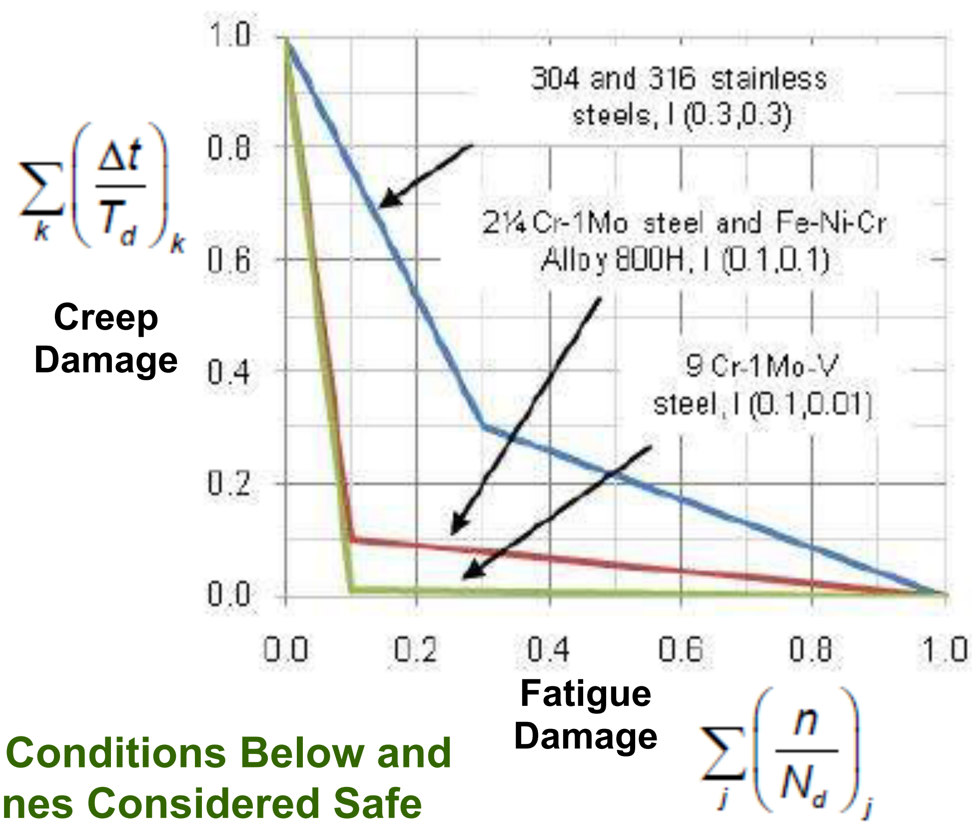


Figure from Miller, Hamm, Phillips, *Materials Science & Engineering*, vol. 53, p.234.

ASME Accounts for Creep Fatigue Interactions Using D-Diagrams

- In ASME Section III Subsection NH, creep-fatigue design analysis is based on evaluating creep damage (in terms of time-fraction) and fatigue damage independently
- Inadequate data and analysis approaches can lead to overly conservative D-Diagrams

$$\underbrace{\sum_j \left(\frac{n}{N_d}\right)_j}_{\text{Cyclic Damage}} + \underbrace{\sum_k \left(\frac{\Delta t}{T_d}\right)_k}_{\text{Creep Damage}} \leq D$$



Operating Conditions Below and Left of Lines Considered Safe

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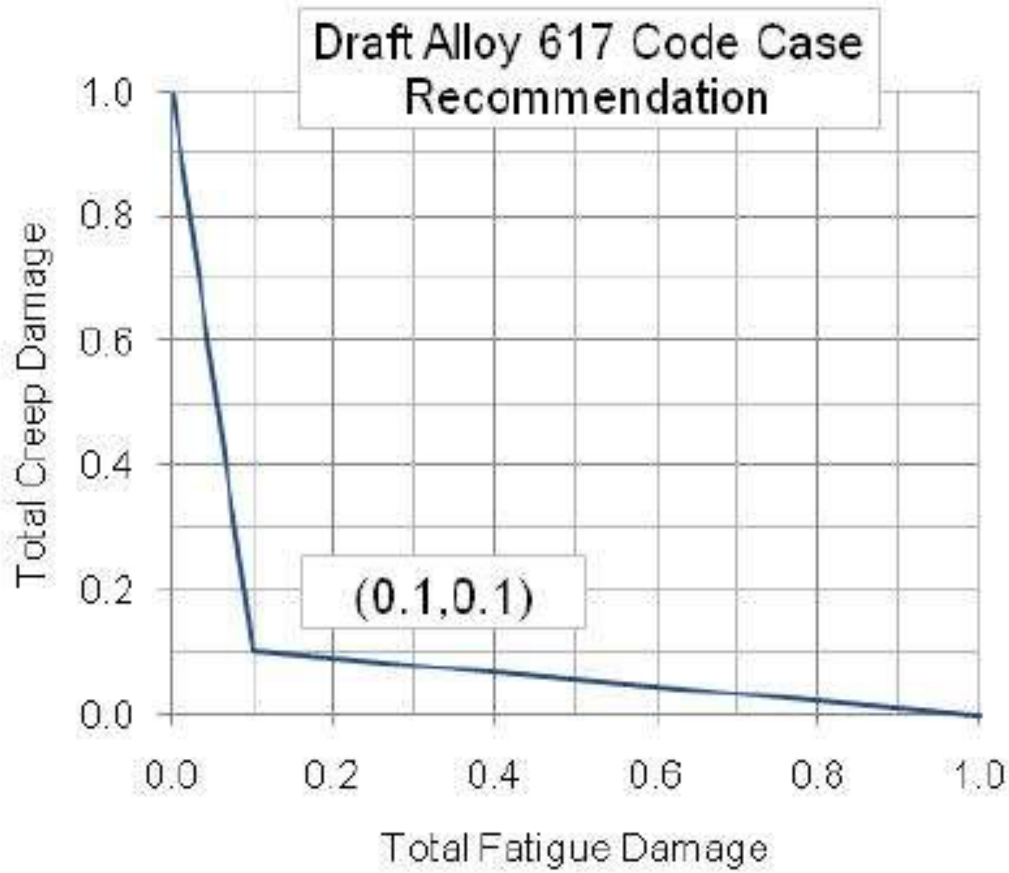
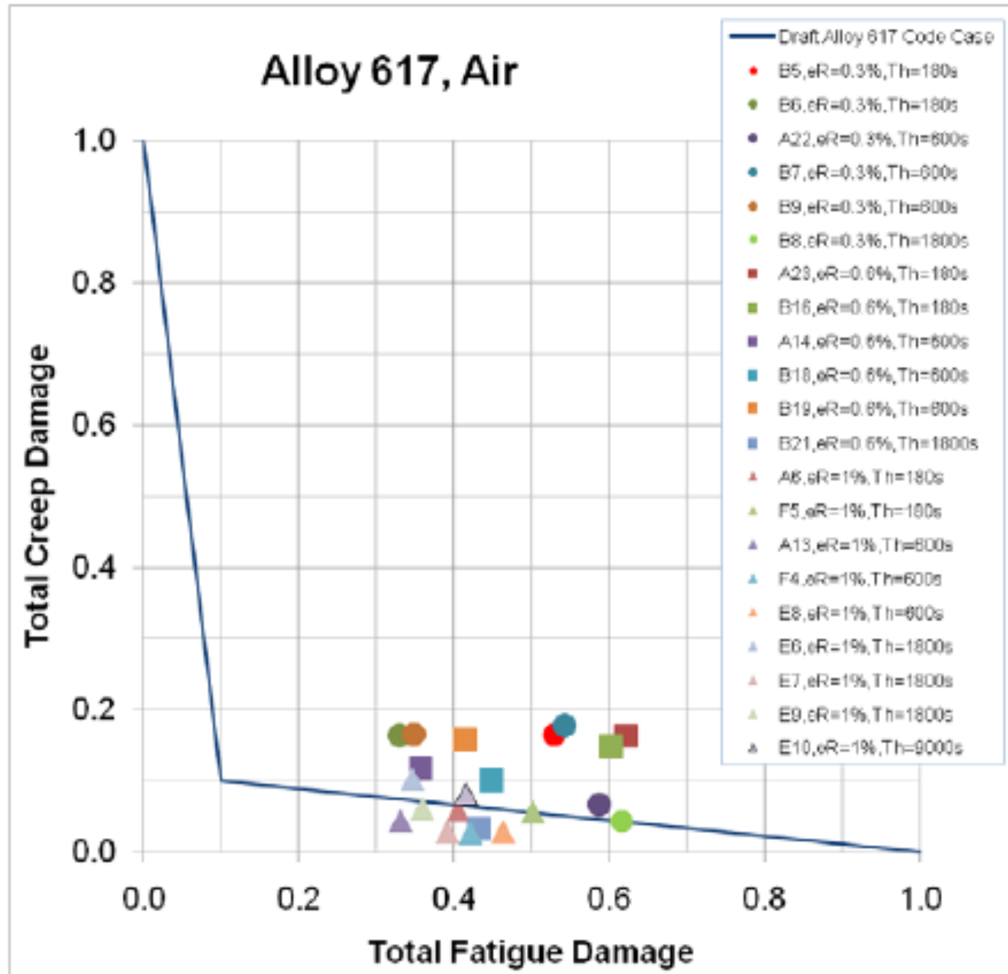
The Creep Fatigue Diagram for Alloy 617 Is Being Developed

- The ASME Draft Alloy 617 Code Case has recommended an intersection of (0.1,0.1) in the D-Diagram for Alloy 617
- Experimental validation of the creep-dominated portion of the creep-fatigue interaction diagram is a challenge



Most Available Experimental Data Is Dominated by Fatigue Damage

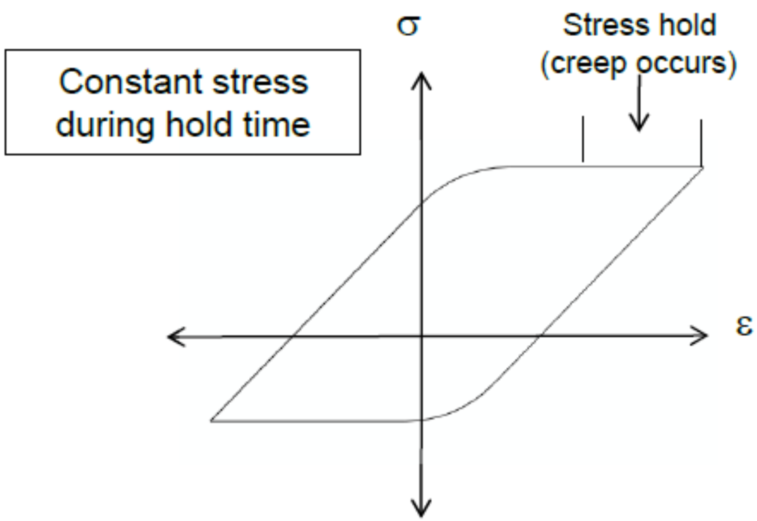
- Typical strain- or load-controlled fatigue tests with a tensile hold period result in mostly fatigue damage for “reasonable” test durations
- Alternate test methods for developing creep-dominated CF data are of particular interest



More Accurate Predictions of Creep Fatigue Interaction Are Needed

- **Proposals are solicited to generate a better mechanistic understanding of creep fatigue interactions, especially where creep is the dominant mechanism**

- **Areas of interest include:**
  - Novel experimental methods to generate creep fatigue damage dominated by creep
  - Creep fatigue model development with substantial experimental validation



- **Materials of interest include Alloy 617, Alloy 800H, 9Cr-1MoV, and possibly other high temperature alloys for advanced nuclear reactor systems**