Nuclear Energy University Programs (NEUP) Fiscal Year (FY) 2017 Annual Planning Webinar

Advanced Reactor Components (Subtopics RC-1 & 3)

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August 10, 2016

Structural Materials Are Critical for Advanced Nuclear Reactors

- **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 and Very High Temperature Gas Cooled Reactors (HTGRs and VHTRs)**
  - **Sodium Cooled Fast Reactors (SFRs)**
  - **Salt Cooled High Temperature Reactors (MSR & FHRs)**
  - **Lead and Lead-Bismuth Cooled Reactors (LFRs)**
- **Structural materials must perform over design lifetimes for pressure boundaries, reactor internals, heat transfer components, etc.**
- **Performance of structural materials for the long times, high operating temperatures, and environments required is being examined under the Advanced Reactor Technologies (ART) Program**

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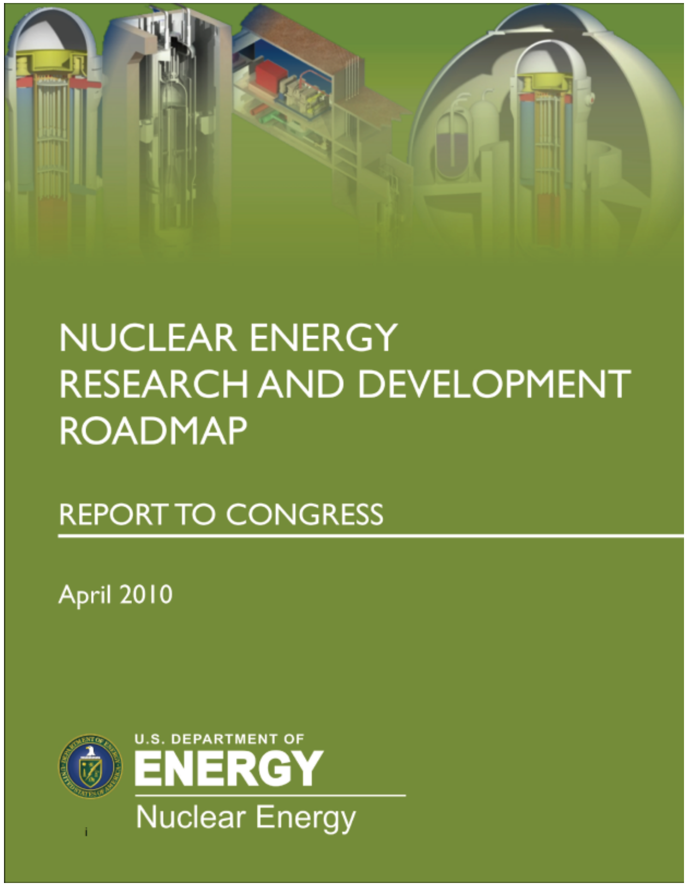
ART Program Supports Advanced Reactor Development

- **ART R&D supports multiple high-level objectives identified in the 2010 Nuclear Energy R&D Roadmap (2 & 3)**

  (2) Develop improvements in the affordability of new reactors to enable nuclear energy to help meet the Administration’s energy security and climate change goals

  (3) Develop sustainable nuclear fuel cycles

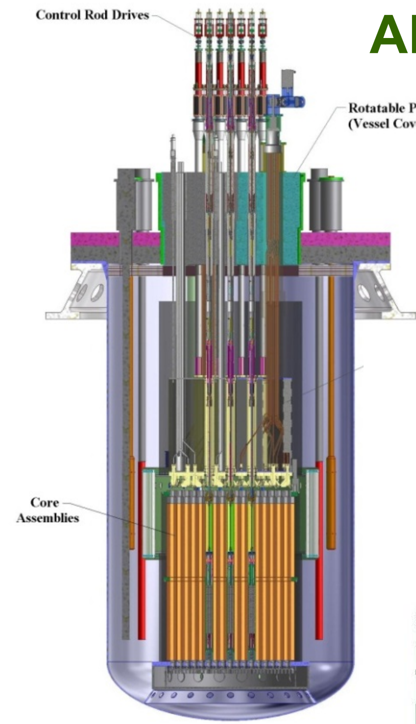
  …overall goal is to have demonstrated the technologies necessary to allow commercial deployment of solution(s) for the sustainable management of used nuclear fuel that is safe, economic, and secure and widely acceptable to American society by 2050.”

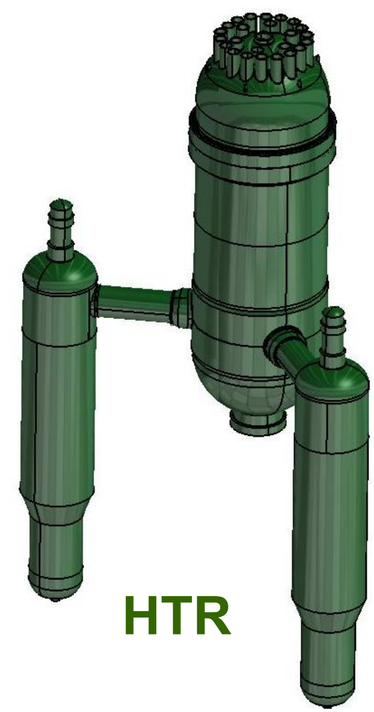


ART Program Includes Advanced Materials R&D Activities

- Development and qualification of graphite, improved high-temperature alloys, and ceramic composites for advanced reactor systems
- Advanced Fast Reactor-100 is an example of fast reactor systems
  - 250MWt/100MWe, sodium-cooled, core life (30 years), plant life (60 years)
- AREVA's High Temperature Reactor is an example of a He-cooled system
  - 625MWt/315MWe, 750°C outlet temperature to steam generator, plant life (60 years)
- Advanced high temperature reactors cooled by liquid lead and salt are also being considered

AFR-100





HTR

Advanced Reactor Technologies Addresses Two Significantly Different Structural Materials Research Topics in FY17

- **RC-1 Materials Compatibility for High-Temperature Liquid Cooled Reactor Systems**
  - Corrosion by liquid coolants in high-temperature reactors may limit lifetimes of approved ASME construction materials and design methods. Corrosion mechanisms and rates must be compared with required lifetimes for critical components. Alternate materials or design methods (e.g. cladding for high temperature structures) must be addressed
- **RC-3 Detection, Evaluation, and Predictive Modeling of Degradation of SiC/SiC Composites**
  - An understanding of failure mechanisms for SiC/SiC composite construction materials for high temperature reactor components is critical. The detection, evaluation, and prediction of slow crack growth and other non-irradiation degradation mechanisms is to be studied to support ASME Code rule development.

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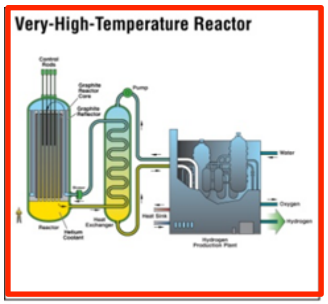
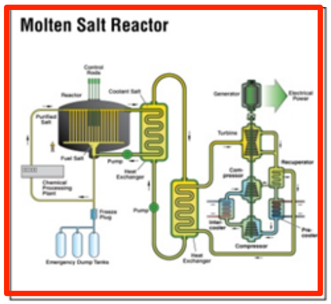
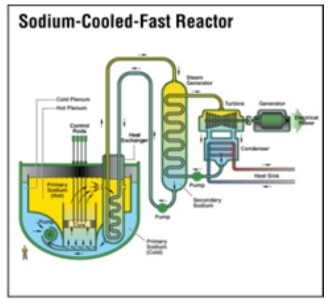
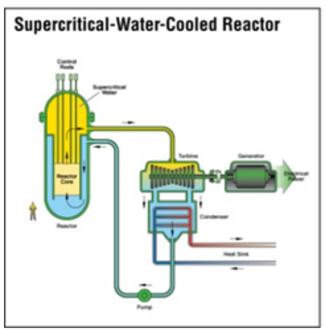
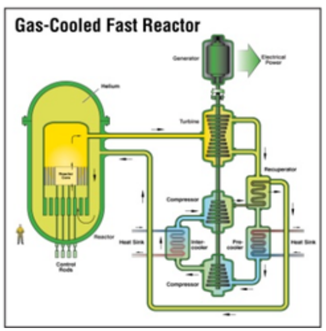
Gen IV Reactors Share Many Materials Challenges and Interests

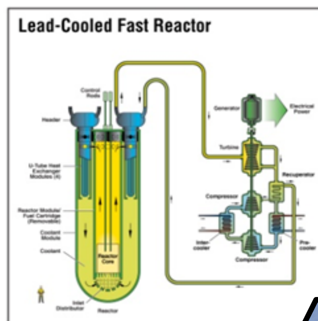






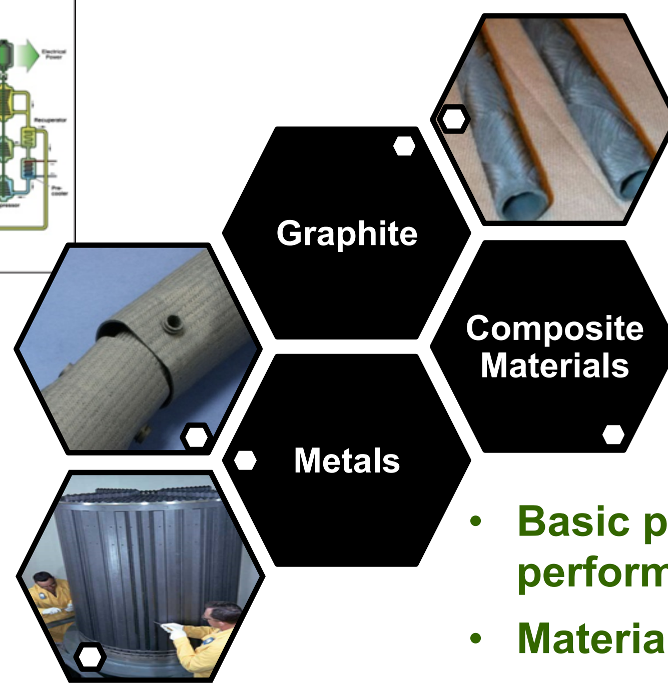






- Long design life at high temperatures
- High dose irradiation and tolerance
- Design methodology development
- Codes, standards, and qualification

- Component design, manufacture & integration
- Failure analysis
- Accident analysis
- Life prediction



- Basic properties and performance
- Material compatibility

RC-1 Materials Compatibility for High-Temperature Liquid Cooled Reactor Systems

- Proposals are sought to assess the potential materials to be used for construction of ASME Code covered components of lead (or lead-bismuth-eutectic) or salt-cooled high temperature reactors and identify preferred candidate materials for their construction

- The proposed research should examine
  - Existing experimental determinations of corrosion and erosion effects, augmented by laboratory experiments as deemed necessary, to extrapolate results to allowable corrosion limits under anticipated service times and conditions
  - Viability of using currently approved materials for ASME nuclear construction to meet anticipated corrosion needs in conjunction with existing Code strength and design requirements
  - Potential for alternate materials, coatings, or cladding and changes to ASME materials and design methods required for their use

Advanced Reactors Need Additional Options for Alloys Approved by ASME for Elevated Temperatures

- Only five alloys are qualified in ASME Sec III Division 5 for service in inelastic (time-dependent) temperature range
  - Two ferritic steels: 2 ¼Cr-1Mo and 9Cr-1Mo steels
  - Two stainless steels: 304 and 316
  - One high-temperature alloy: Alloy 800H
    - Alloy 617 (Ni-based) will be to be added soon to Div 5
- Hastelloy N (and similar foreign alloys, GH3535) are not approved for liquid salt service
- Neither HT-9 nor silicon/aluminum-modified steels are approved for liquid lead/lead-bismuth service
- Efficacy and reliability of coatings or cladding for corrosion prevention in nuclear service requires validation
- ASME III Div 5 does not include rules for design/use of clad structures for high temperature service

Liquid Coolant Corrosion Allowances for ASME Code Approved Materials Must Meet Anticipated Service Lives

- Some components (e.g., RPV) service lives may extend for full life of plant, others (e.g., IHX) may be shorter, only 5-10 years
- ASME approval of new alloys with the required corrosion resistance and elevated temperature strength, will require comprehensive and very long term test data in accordance with the requirements of Division 5, Appendix HBB-Y
- Modifications of existing Sec III Div 5 design rules to include novel materials approaches, such as bimetallic structures or clad structures, may be evaluated. However, Div 5 Para HBB-3227.8(d) requires cladding be considered related to limitations on deformation-controlled quantities (i.e., cyclic loading) but does not provide guidance for that assessment. Thermal stress must also be considered.
- Pathways to modify ASME Code to meet required corrosion allowances must be well defined, if chosen

RC-3 SiC/SiC Composites

- **Proposals are sought to develop a scientific understanding and innovative advanced methods for the detection, evaluation, and prediction of degradation for SiC/SiC composite components in the operating environments of advanced reactor systems**
  - **Neutron irradiation effects are not included in this effort**
- **A predictive capability is the ultimate goal of this work, but development of practical principles of degradation and/or methods for in-service inspections may be considered**
- **The results from this work should support the Code rule development for SiC/SiC composite core components for high temperature reactors in ASME Boiler and Pressure Vessel Code Section III Division 5**

SiC/SiC Composites are Potentially Applicable to Multiple Advanced Reactor Concepts

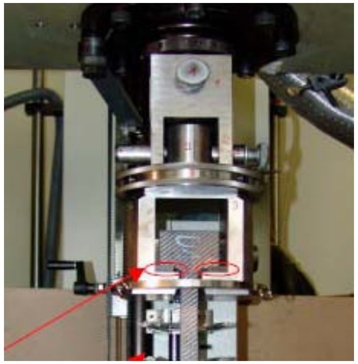
Reactor Concept	Application	Operating Condition	Project / Design Examples	Possible Deployment
Fusion	• Blanket structures • Various functions	• He, Pb-Li • 400-900°C • >50 dpa	• ARIES • EU-PPCS • DREAM	• Long-term
HTGR VHTR	• Reaction control systems • Core support	• He • 600-1100°C • Up to ~40 dpa	• NGNP • PBMR • GT-HTR300C	• Near-term
LWR	• Channel box • Grid spacer • Fuel cladding	• Water • 300-500°C • ~10 dpa	• PWR (WHC • BWR (EPRI)	• Mid-term? (ATF)
FHR AHTR	• Core structures • RCS	• Liquid salt • ~700°C • >10 dpa	• AHTR • DOE IRP • SMR's	• Long-term
SFR	• Core structures • Fuel cladding/support	• Liquid sodium • 500-700°C • >100 dpa	• CEA	• Long-term
GFR	• Core structures • Fuel cladding/support	• He • 700-1200°C • >100 dpa	• CEA • GA EM²	• Long-term

Specific Applications for Composites in Advanced High Temperature Reactor Systems Are Already Being Investigated

Expected applications for composites in high temperature reactor systems.

Reactor Unit

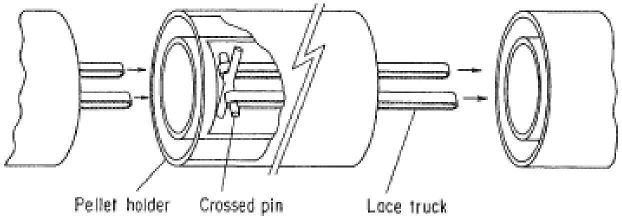
Tie Rods



Restraint Straps

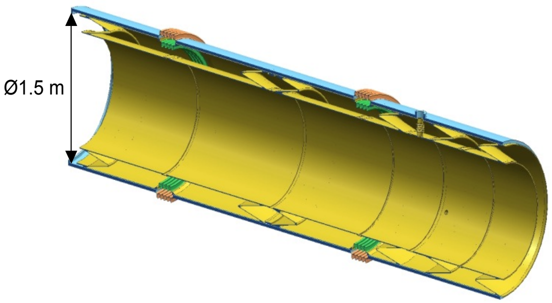
Control Rods

Example HTTR CFRC Control Rod



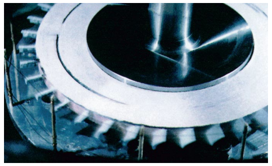
Interconnecting Duct

Hot Gas Duct

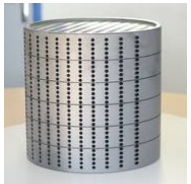


Power Conversion

HP Turbine Blisk

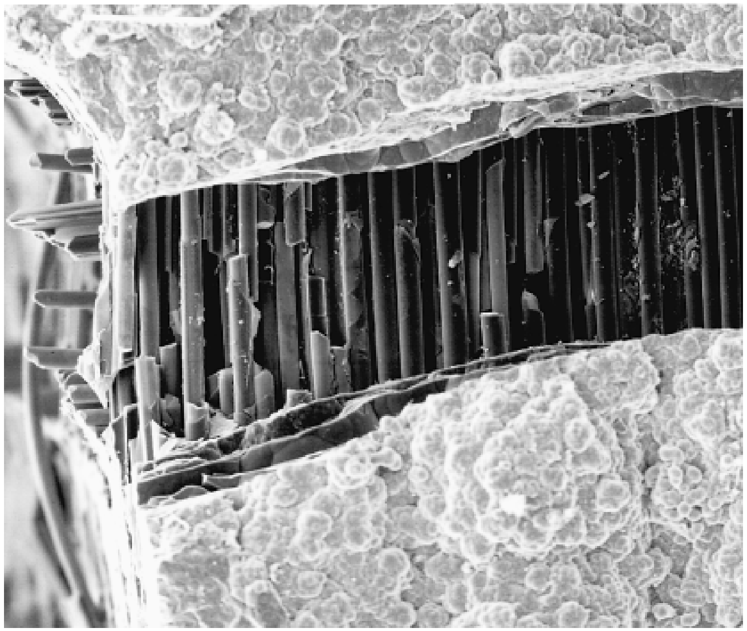


Heat Exchangers



Environment, Load, and Temperature Have Been Shown to Affect Subcritical Crack Growth in SiC-SiC

- **Slow crack growth in SiC-SiC is affected by environmental oxygen content, fabrication processes, residual interfacial layers, operating temperatures, etc.**
- **Validated predictive degradation models and detection methods will be needed for the range of advanced reactor environments**
- **Interim supporting studies determining fundamental effects affecting SiC-SiC behavior are needed to develop required models and detection methods**



Bridged subcritical crack in SiC/SiC composite (from Jones, et. al., PNNL, 2004

ASME Codification of Composite Materials for High Temperature Reactors Is Underway

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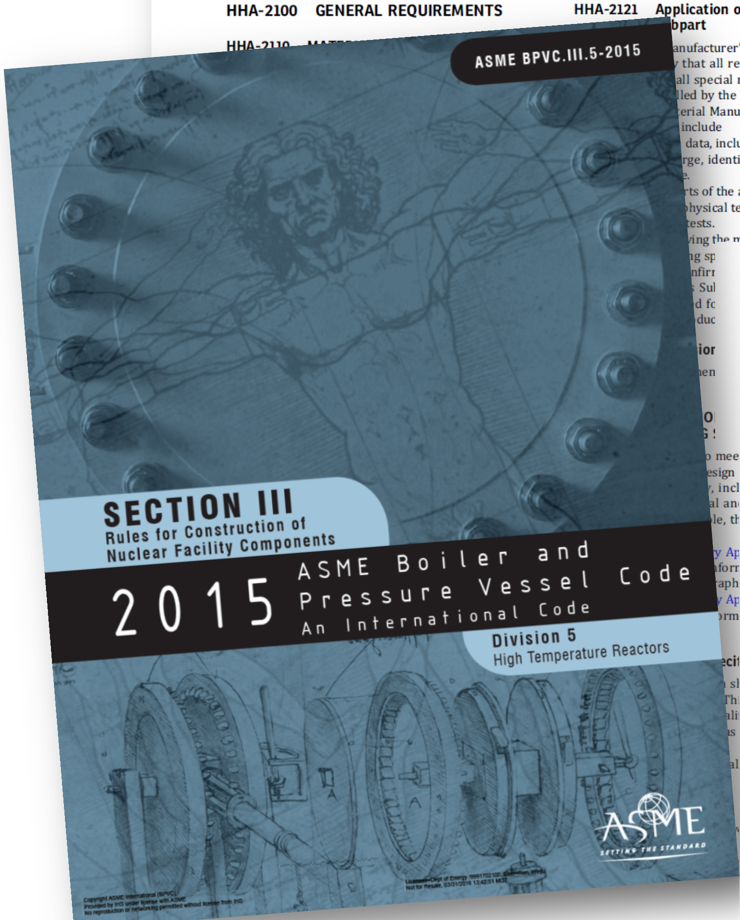
- ASME BPVC Sec. III Div. 5
  - Subsection HH for Class A Nonmetallic Core Support Structures
  - Subpart B for Composite Materials

ARTICLE HHA-2000
MATERIALS

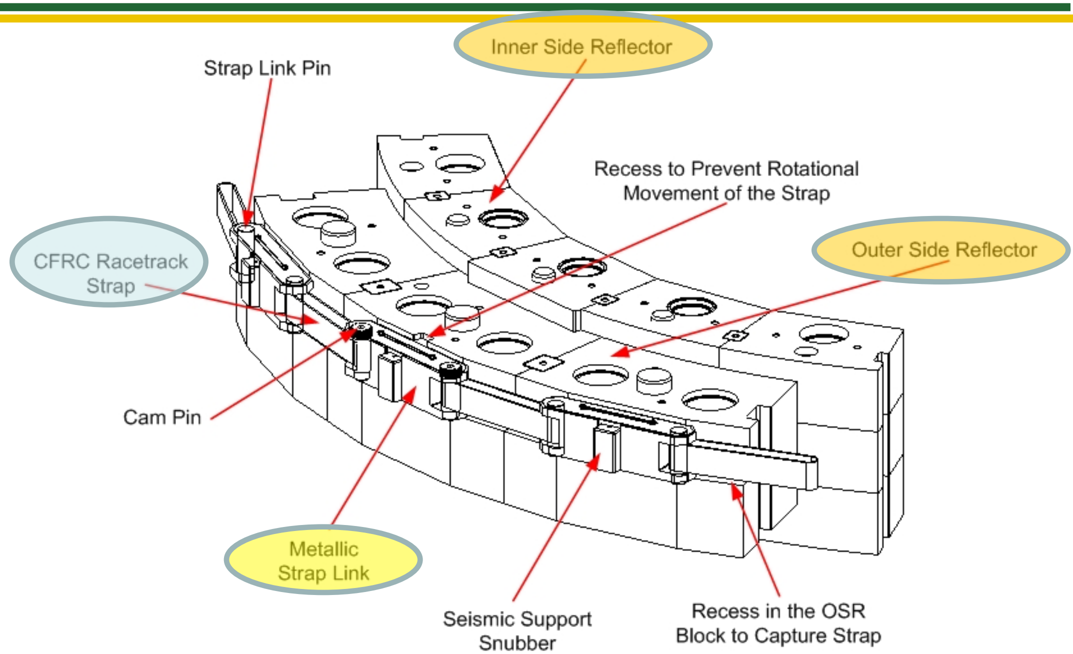
HHA-2100 GENERAL REQUIREMENTS

HHA-2121 Application of the Rules of This Subpart



ASME Sec III Div 5 Is Incorporating Design Rules for Metallic and Non-Metallic Components into the Code

- The ASME Code will provide for mixed assembly of components into a reactor
- PBMR example shows materials combination
- Interactive failure modes must be understood



Core Assembly [HAB / HHA / HG]

Comprises

Graphite Core Component [HAB / HHA]	Composite Core Component [HAC / HHB]	Core Support / Reactor Internal [HAA / HG]

Points of Contact for RC-1 & 3

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