Alloy Considerations for Advanced GEN IV Applications

David W. Gandy
Technical Executive, Nuclear Materials
Electric Power Research Institute

RIC 2017
March 14-16, 2017

Presentation Outline

- Consideration of Materials (structural) Requirements
- Gen IV Designs (3 of 6 covered here)
- Outlet Temperature Regimes
- Materials Experience
- EPRI’s Role–Materials Gap Analysis
- Summary

Materials Requirements for GEN IV Applications

- Three primary criteria must be met when operating over a wide range of temperatures, stresses and doses:
  1. Adequate mechanical properties (strength, ductility, and toughness)
  2. Good dimensional stability (resistance to void swelling and thermal/irradiation creep)
  3. Corrosion, SCC, and embrittlement resistance
Six Advanced Reactor Concepts

<table>
<thead>
<tr>
<th>Reactor Concept</th>
<th>Coolant</th>
<th>Outlet Temperature (°C)</th>
<th>Pressure</th>
<th>Neutron Spectrum</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gas-cooled reactor (GFR)</td>
<td>Helium</td>
<td>~750</td>
<td>High</td>
<td>Fast</td>
</tr>
<tr>
<td>Lead-cooled fast reactor (LFR)</td>
<td>Pb (metal) or Pb-Bi (eutectic)</td>
<td>500 - 550</td>
<td>Low</td>
<td>Fast</td>
</tr>
<tr>
<td>Molten salt reactor (MSR)</td>
<td>Fluoride or Chloride salts</td>
<td>650 - 700 – 1000</td>
<td>Low</td>
<td>Fast or Thermal</td>
</tr>
<tr>
<td>Sodium-cooled fast reactor (SFR)</td>
<td>Sodium (metal)</td>
<td>485 – 550</td>
<td>Low</td>
<td>Fast</td>
</tr>
<tr>
<td>Supercritical water-cooled reactor (SCWR)</td>
<td>Water</td>
<td>500 - 625</td>
<td>Very High</td>
<td>Fast or Thermal</td>
</tr>
<tr>
<td>Very high-temperature reactor (HTR &amp; VHTR)</td>
<td>Helium</td>
<td>700 - 750 – 850 – 1000</td>
<td>High</td>
<td>Thermal</td>
</tr>
</tbody>
</table>

1. Sodium-Cooled Fast Reactor (SFR)  
   – 485C-550C (905-1022F)
   - Twenty (20) SFRs – experimental or demonstration units with over 400 yrs of operating experience
   - 500-800MWe (general range)
   - Primary loop is isolated from the steam generator

- Examples:
  - Adv Reactor Corp ARC-100 (510C),
  - GE-Hitachi PRISM (485C),
  - TerraPower TWR (510C)

SFRs–Experimental and Demonstration Units

<table>
<thead>
<tr>
<th>Country</th>
<th>Reactor</th>
<th>MWe</th>
<th>Reactor Type</th>
<th>Operating Life (yrs)</th>
<th>Outlet Temp</th>
</tr>
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<tbody>
<tr>
<td>USA</td>
<td>EBR-II</td>
<td>62.5</td>
<td>Pool</td>
<td>35.9</td>
<td>405</td>
</tr>
<tr>
<td></td>
<td>FTRP</td>
<td>455</td>
<td>Loop</td>
<td>36.5</td>
<td>430</td>
</tr>
<tr>
<td></td>
<td>HFRP</td>
<td>355</td>
<td>Loop</td>
<td>36.5</td>
<td>540</td>
</tr>
<tr>
<td>UK</td>
<td>DFR</td>
<td>60</td>
<td>Loop</td>
<td>54.2</td>
<td>330</td>
</tr>
<tr>
<td></td>
<td>PPRP</td>
<td>355</td>
<td>Loop</td>
<td>29.2</td>
<td>540</td>
</tr>
<tr>
<td>Russia</td>
<td>BOR-60</td>
<td>55</td>
<td>Loop</td>
<td>39.9</td>
<td>480</td>
</tr>
<tr>
<td></td>
<td>BII-300</td>
<td>750</td>
<td>Loop</td>
<td>29.7</td>
<td>415</td>
</tr>
<tr>
<td></td>
<td>BII-600</td>
<td>1470</td>
<td>Loop</td>
<td>29.7</td>
<td>520</td>
</tr>
<tr>
<td>France</td>
<td>Phoenix</td>
<td>583</td>
<td>Post</td>
<td>36.2</td>
<td>555</td>
</tr>
<tr>
<td></td>
<td>SuperPhex</td>
<td>2960</td>
<td>Post</td>
<td>12.4</td>
<td>525</td>
</tr>
<tr>
<td>Japan</td>
<td>Jchoji</td>
<td>140</td>
<td>Loop</td>
<td>32.5</td>
<td>470</td>
</tr>
<tr>
<td></td>
<td>Morita</td>
<td>714</td>
<td>Loop</td>
<td>29.5</td>
<td>470</td>
</tr>
</tbody>
</table>

* No steam generator and only sodium-to-air heat exchanger

Source: A Technology Roadmap for Generation IV Nuclear Energy Systems (GIF-002-00)

SFRs—Materials Experience

- **Vessel materials**
  - 304, 316, 321, 18-18-1
- **Primary/secondary piping**
  - 304, 316, 321, 18-18-1, 9Cr or 12Cr
- **Internal heat exchanger**
  - 304, 316
- **Steam generator**
  - Fe-2.25Cr-1Mo, 304, 321, Alloy 800, 9Cr-1Mo, 12Cr


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2. Molten Salt Reactors (MSR)

**650C-700C (1202F-1292F)**

- Several nickel-based alloys investigated in 1950s/1960s:
  - Hastelloy B and N, INOR-8, and Inconels
- Suitable alloys chosen for the aggressive salt environment:
  - Modified Hastelloy N (<800°C) - w Ti & Nb additions
  - INOR-8 (<815°C)
- For non-graphite core concepts:
  - Ni-based alloys are susceptible to He-induced embrittlement under irradiation conditions resulting in low creep ductility
  - Ti additions limit the embrittlement issue

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2. Molten Salt Reactors (MSR)

**650C-700C (1202F-1292F)**

- **Reactor Vessel materials**
  - Modified Hastelloy N (<800°C)
- **Heat-exchanger alloys**
  - Modified Hastelloy N (<800°C)
  - INOR-8 (<815°C)
- **Primary/secondary piping**
  - Nickel-based alloys
- **Turbine, Compressor, Recuperator**
  - Isolated from primary circuit
  - Will use conventional GT alloys

*Note: As MSRs are moved to higher temperatures, advanced alloys will be required*
3. High Temperature Gas Reactors (HTGR) -- ~750°C (1382°F)

- Most HTGRs operate at ~750°C today
- Goal is to one day achieve 950°C–1000°C operation range (VHTRs)
- Helium is operating fluid which flows directly into the turbine, compressor
- Many HTGRs use an indirect cycle (i.e., coupled via HX to a steam cycle)

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3. High Temperature Gas Reactors (HTGR) -- 700°C-750°C (1292-1382°F)

**Experimental HTGRs**

<table>
<thead>
<tr>
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</thead>
<tbody>
<tr>
<td>Peach Bottom (USA)</td>
<td>1967-74</td>
<td>out of operation</td>
<td>115/40</td>
<td>pin</td>
<td>8.3</td>
<td>377/750</td>
<td>2.5</td>
<td>HEU</td>
<td>Carbide</td>
<td>TRISO</td>
<td>PCRV</td>
</tr>
<tr>
<td>Dragon (UK)</td>
<td>1968-75</td>
<td>safe end.</td>
<td>46/15</td>
<td>pin</td>
<td>14</td>
<td>270/750</td>
<td>3.0</td>
<td>HEU</td>
<td>Oxide</td>
<td>TRISO</td>
<td>PCRV</td>
</tr>
<tr>
<td>AVR (Germany)</td>
<td>1967-88</td>
<td>defuelled</td>
<td>26/10</td>
<td>pin-in-block</td>
<td>2.6</td>
<td>270/950</td>
<td>4.0</td>
<td>HEU/LEU</td>
<td>Carbide/Oxide</td>
<td>TRISO/STISO</td>
<td>TRISO</td>
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<tr>
<td>HTTR (Japan)</td>
<td>1998-xx</td>
<td>in operation</td>
<td>85/35</td>
<td>Spherical</td>
<td>2.5</td>
<td>385/550 and 285/550</td>
<td>3.5</td>
<td>LEU</td>
<td>Oxide</td>
<td>TRISO</td>
<td>PCRV</td>
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<tr>
<td>HTR-10 (China)</td>
<td>2000-xx</td>
<td>in operation</td>
<td>170/950</td>
<td>Spherical</td>
<td>2.</td>
<td>250/350/750</td>
<td>3.0</td>
<td>LEU</td>
<td>Oxide</td>
<td>TRISO</td>
<td>PCRV</td>
</tr>
</tbody>
</table>

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3. High Temperature Gas Reactors (HTGR) -- 700°C-750°C (1292-1382°F)

**Prototype HTGRs**

<table>
<thead>
<tr>
<th></th>
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<th></th>
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<th></th>
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</thead>
<tbody>
<tr>
<td>Fort St. Vrain (USA)</td>
<td>1976-1989</td>
<td>Decommissioned</td>
<td>640/330</td>
<td>Prismatic</td>
<td>405/750</td>
<td>530</td>
<td>550</td>
<td>HEU</td>
<td>Carbide</td>
<td>TRISO</td>
<td>PCRV</td>
</tr>
<tr>
<td>THTR (Germany)</td>
<td>1986-1989</td>
<td>Safe Enclosure</td>
<td>640/330</td>
<td>Spherical</td>
<td>270/750</td>
<td>530</td>
<td>280</td>
<td>HEU/LEU</td>
<td>Oxide</td>
<td>TRISO</td>
<td>PCRV</td>
</tr>
</tbody>
</table>

Source: A Technology Roadmap for Generation IV Nuclear Energy Systems (GIF-002-00)
3. High Temperature Gas Reactors (HTGR)

### Commercial HTGR Projects

<table>
<thead>
<tr>
<th></th>
<th>PNP</th>
<th>HHT</th>
<th>HTR-500</th>
<th>HTR-MHTR</th>
<th>HTR-100</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal/electric power (MWth/MWe)</td>
<td>500/500</td>
<td>1240/500</td>
<td>1250/500</td>
<td>200/80</td>
<td>258/100</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Fuel element type</th>
<th>spherical</th>
<th>block/spherical</th>
<th>spherical</th>
<th>spherical</th>
<th>spherical</th>
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</thead>
<tbody>
<tr>
<td>Power density (MWth/m³)</td>
<td>4</td>
<td>5.5</td>
<td>7</td>
<td>3</td>
<td>3</td>
</tr>
<tr>
<td>He inlet/outlet temperature [°C]</td>
<td>300/950</td>
<td>440/850</td>
<td>280/700</td>
<td>250/750</td>
<td>250/740</td>
</tr>
<tr>
<td>He pressure [MPa]</td>
<td>3.9</td>
<td>5</td>
<td>4.7</td>
<td>5</td>
<td>7</td>
</tr>
<tr>
<td>Enrichment</td>
<td>LEU</td>
<td>LEU</td>
<td>LEU</td>
<td>LEU</td>
<td>LEU</td>
</tr>
<tr>
<td>Fuel</td>
<td>Oxide</td>
<td>Oxide</td>
<td>Oxide</td>
<td>Oxide</td>
<td>Oxide</td>
</tr>
<tr>
<td>Coating</td>
<td>TRISO</td>
<td>TRISO</td>
<td>TRISO</td>
<td>TRISO</td>
<td>TRISO</td>
</tr>
<tr>
<td>Pressure vessel</td>
<td>PORV</td>
<td>PORV</td>
<td>PORV</td>
<td>Steel</td>
<td>Steel</td>
</tr>
</tbody>
</table>

**Source:** High Temperature Gas Cooled Reactor Fuel Materials, IAEA-TECDOC-1645

### Reactor Pressure Vessel
- Modified 9Cr-1Mo (considered for GT-MHR and NGNP core support structures)
- 2.25Cr-1Mo (HTTR in Japan)

### Hot Ductwork/Piping
- Alloy 800H
- Alloys 617, 625, 625-plus
- Alloy 718
- Hastelloys X, XR, N

### Core & core support structures
- Alloys X and XR (<750°C)

### Core barrel assemblies & reactor internals
- 316H
- Above 750°C, other nickel-based alloys are under consideration

**Graphite, carbon fiber, ODS**

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3. High Temperature Gas Reactors (HTGR)

### 700°C-750°C (1292-1382°F)

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  - Alloy 800H
  - Alloys 617, 625, 625-plus
  - Alloy 718
  - Hastelloys X, XR, N

- Core & core support structures
  - Alloys X and XR (<750°C)

- Core barrel assemblies & reactor internals
  - 316H
- Above 750°C, other nickel-based alloys are under consideration

**Graphite, carbon fiber, ODS**
4. Very High Temperature Reactor (VHTR) – 850°C-1000°C (1562°F-1832°F)

- Graphite modulated
- Helium cooled
- Prismatic block- or pebble-bed core
- Intermediate HX
- May be tied to hydrogen production plant
- At 1000°C outlet temperature, the reactor pressure vessel temperature will exceed 450°C (842°F)

Source: M. LaBar, General Atomics HTHX Planned Activities, High Temperature HX Kickoff Meeting, October 2003

4. Very High Temperature Reactor (VHTR) – 850°C-1000°C (1562°F-1832°F)

- Vessel materials
  - Pre-stressed concrete
  - Pre-stressed cast iron
  - Modified 9Cr-1Mo
  - Ferritic-martensitic alloys
- Intermediate heat exchanger
  - Hastelloy XR, Ni-Cr-W alloys, SiC, ODS
- Primary/secondary piping
  - Hastelloy XR, Ni-Cr-W alloys, ODS
- In-Core Materials
  - Si-carbide
  - Zr-carbide
  - Pyrolytic carbon (similar to graphite)

4. Very High Temperature Reactor (VHTR) – 850°C-1000°C (1562°F-1832°F)

Pebble Bed Reactor, ESKOM, South Africa

- Steel pressure vessel
- 6.2m (dia) x 27m (high)
- Lined with 1m thick graphite bricks drilled with vertical holes housing control elements
- 450,000 TRISO-fuel molded in graphite spheres
- Helium cooled
- 900°C outlet temperature
Current Advanced Alloy Development
—EPRI R&D (non-Gen IV)
- Advanced Radiation Resistant Materials (ARRM)
  - Degradation resistant alloys for LWRs & ALWRs
- Creep Resistant Nickel-based Alloys
  - Haynes 282 and Alloy 740H (creep performance for 760C AUSC fossil operation)
- Co-free Hardfacing Alloys
  - NitroMaxx-PM
- Grade 91 (9Cr-1Mo)
  - Considerable fossil experience
  - Specs to enhance performance/life
- CF8C-plus
  - Creep-resistant SS for up to 800C
- Powder Metallurgy-Hot Isostatic Pressing
  - Low alloy, stainless, and nickel-based alloys
  - Dissimilar metals joining
  - Bi-Metallics
  - Irradiation studies
- Si-Carbide Materials
  - Composites for fuel channels; cladding
- Advanced Welding Filler Materials
  - Alloy 52 replacement
  - P97 for DMWs

EPRI’s Role—Materials Gap Analysis for GEN IV Applications (Dec 2017)

I. Materials Considerations/Requirements to be covered.
- Reactor materials
- Reactor internals materials
- Coolant piping, valve, and pump materials
- Secondary piping materials
- Steam generator, pressurizer

II. Materials Property/Environmental Considerations to be covered
- Creep & Creep-fatigue
- General corrosion, oxidation, SCC
- Irradiation damage
- Manufacturing Options (forging, PM-HIP, extrusion, other)
- Weldability, cladding, hardfacing
- Alloy cost

Summary—Materials Knowledge/Experience
- SFRs and MSRs—Good materials knowledge/operation
  - More R&D required for MSRs above 800C however
- HTGRs (750C)—No major concerns—reasonable knowledge/operation
  - Nickel-based alloys available
  - Several experimental & demonstration reactors operated
  - Long-term irradiation effects still unknown
  - Helium environment for turbine, compressor
- VTHRs (850-1000C)—More development required
  - RPV alloys—good knowledge
  - HV and primary/secondary piping materials identified
  - Fabrication and manufacturing concerns; irradiation effects?
  - In-Core—Si-, Zr-Carbides
Irradiation Influences on PM-HIP Alloys

- OEMs are interested in PM-HIP alloys for:
  - near-net shaped capabilities and
  - inspection characteristics
- Evaluation just starting with Purdue U. and NSUF (ATR) to irradiate/evaluate 6 alloys:
  - Austenitic SS: 304L, 316L
  - Nickel-based Alloys 625, 690
  - Ferritic Alloys: 508, Gr 91
  - Compare w/ cast, forged
- Various dpa and temperatures
- Tensile, Charpy, and microstructural examination.