Meet the presenter

Stuart Maloy is a Team Leader for MST-8 (materials at radiation and dynamic extremes) at Los Alamos National Laboratory (where he has worked for 28 years) and is the advanced reactor core materials technical leader for the Nuclear Technology Research and Development’s Advanced Fuels campaign and the NEET Reactor Materials Technical Lead for DOE-NE. He earned his Bachelors Degree ('89), Masters Degree ('91) and PhD ('94) in Materials Science from Case Western Reserve University and is a registered PE in Metallurgy. He has applied his expertise to characterizing and testing the properties of metallic and ceramic materials in extreme environments such as under neutron and proton irradiation at reactor relevant temperatures. This includes testing the mechanical properties (fracture toughness and tensile properties) of Mod 9Cr-1Mo, HT-9, 316L, 304L, Inconel 718, Al6061-T6 and Al5052 after high energy proton and neutron irradiations using accelerators and fast reactors. Characterization of materials after testing includes using transmission electron microscopy for analyzing defects such as dislocations, twins and second phases, using high resolution electron microscopy to characterize defects at an atomic level and nanoscale mechanical testing. Stuart has >190 peer reviewed technical publications and numerous presentations.

Email: maloy@lanl.gov
Outline

- Radiation Effects in Materials
- Materials in Nuclear Reactors – FCC, BCC alloys
- Reactor Conditions/Materials Performance Issues
  - LWR (BWR/PWR)
  - Typical materials
  - Advanced Reactors (VHTR, SCWR)
  - Advanced Fast Reactors (SFR, GFR, MSR)
- Summary of Reactor Operating Conditions
- Summary of Performance Issues
Materials in nuclear systems can fail

Fast Reactor Duct Failure

Grid-to-Rod Fretting

CRUD

Davis-Besse Reactor Vessel Head Degradation
Displacement damage occurs when enough energy (approximately 25 eV) is transferred to an atom producing a or many Frenkel defects. Energy can be transferred from many different particles (neutrons, protons, fission products, etc).

Particles with higher energies (>5 MeV) can cause spallation in materials.
Background on radiation damage

1st Transmutation (changing elements)
- Activating the material
- Producing new elements leading to He bubble formation and embrittlement e.g. $^{56}\text{Fe}(n,a)^{53}\text{Cr}$ or $^{58}\text{Ni}(n,a)^{55}\text{Fe}$

2nd Materials property changes due to atomic displacements:
- Creation of Frankel pairs can lead to:
  - Increase of dislocation density → embrittlement
  - Formation of voids → swelling
  - Increased diffusivity → local segregation
  - Amorphisation or crystallization → unexpected phase changes
Example of cascade by low energy PKA’s

0.5 nm
Radiation effects are governed by the ultimate fate of point defects (& transmutants)

Voids, I-loops & precipitates in stainless steel

**EVENT**
- Particle-atom collisions (displacement production in cascades)
- Diffusion of vacancies (V), self-interstitials (I) and solutes/impurities
- V-I clustering: formation of (bubble, void, loop)
- V/I absorption at sinks (e.g., dislocations, clusters)

**ENERGY (eV)**
- $10^6$ (neutrons)
- $10^4$ (PKAs)
- $10^2$ (recoils)

**TIME (s)**
- $10^{-13} - 10^{-11}$
- $10^{-9} (\text{I}), 10^{-4} (\text{V})$
- $> 10^7$

**Nascent cascade**
- Number of defects: $N_i = N_v = \nu(T)$

**“Quenched” cascade**
- Number of defects: $N_i' = N_v' < \nu(T)$
A wide range of materials properties are determined on the mesoscale.

- **High melting phase** (e.g., MgS)
- **Coherent precipitates**
- **Incoherent precipitates**
- **Edge dislocations**
- **Substitution atoms**
- **Interstitial**
- **Screw dislocations**
- **Unit cell**
- **Grain boundary precipitates**
- **Grain diameter**

**Legend:**
- **Green** = nanoscale defects
- **Blue** = nanoscale-meso scale
- **Red** = mesoscale defects (actually defect dependent)

**Unit cell**
- e.g. alpha iron
  - 0.2867 nm
Typical Alloy Compositions for Nuclear Applications

- **Austenitic Steels (Face Centered Cubic, FCC)**
  - 316L (Fe, 18Cr, 10Ni, 2 Mo), 304L Stainless steel (Fe, 18Cr, 8 Ni, 1.5 Mo)
  - Inconel 718 (55 Ni, 21Cr, 24Fe, 5Nb, 3Mo, 1Ti, 0.8Al)
  - Alloy 600 (72 Ni, 17Cr, 9Fe, 1 Mn)

- **Ferritic Steels (Body Centered Cubic, BCC)**
  - Mod 9Cr-1Mo (Fe, 9Cr, 1Mo, 0.1C, 0.5Mn)
  - HT-9 (Fe, 12Cr, 1Mo, 0.2C, 0.5W, 0.6Mn, 0.3V, 0.5Ni)

- **Zirconium Alloys (Hexagonal Close Packed, HCP)**
  - Zr-Sn alloys (Zircaloy 2, Zircaloy 4)
  - Zr-Nb alloys (Zr-1Nb; Zr-2.5Nb, M5 (Zr-1Nb))
  - Zr-Sn-Nb-Fe alloy (ZIRLO)
Radiation Effects in Metals

- Defect formation
  - Basically Frenkel Defects (self interstitials and vacancies)
  - No charge compensating defects
  - Very little effect from gamma irradiation
  - Amorphization is uncommon at typical irradiation temperatures 25 to 600°C

- Close Packed Structures
  - FCC (face centered cubic)-close packed plane is (111)-Frank loops- atomic packing factor is 0.74
  - BCC (body centered cubic)-close packed plane is (110)-atomic packing factor is 0.68
  - HCP (hexagonal close packed)-for ideal c/a ratio of 1.633, atomic packing factor is same as FCC, 0.74
At lower temperature (blue region) vacancies are immobile and interstitials are mobile resulting in interstitial clusters and loops and small vacancy clusters.

At medium temperature (gray region) vacancy mobility increases resulting in more self annihilation of defects (vacancy finds interstitial) and possibility of swelling.

At higher temperature (red region) vacancy and interstitial mobility are high leading to problems with creep or helium embrittlement.

Irradiation Effects in Metals
A. 316L/304L Stainless Steel (FCC)
B. Alloy 718 (FCC)
C. Ferritic Steels (BCC)
Irradiation Effects in 316L/304L Stainless Steel (FCC) at 50-200°C
TEM images showing the growth of Frank loops in 304L.

Irradiated with 800 MeV proton beam at 30-50°C

- no helium clusters are observed
- loops mainly from collection of interstitials

50 nm

0.7 dpa
3.8 dpa
9.8 dpa
Dislocation Loops Show a Saturation in Density around 4 dpa

<table>
<thead>
<tr>
<th>Dose (dpa)</th>
<th>Loop Number Density (m⁻³)</th>
<th>Mean Loop Diameter (nm)</th>
<th>Total Dislocation Density (m²)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.7</td>
<td>1.6x10^{22}</td>
<td>1.8</td>
<td>9.0x10^{13}</td>
</tr>
<tr>
<td>3.8</td>
<td>5x10^{22}</td>
<td>9.6</td>
<td>1.51x10^{15}</td>
</tr>
<tr>
<td>9.8</td>
<td>2.1x10^{22}</td>
<td>20.1</td>
<td>1.32x10^{15}</td>
</tr>
</tbody>
</table>
Stress/Strain Curves show increase in yield stress and decrease in elongation in 316L Stainless steel after irradiation.

\[ \text{T}_{t} = \text{T}_{\text{irr}} = 50 \text{C} \]
Irradiated Materials Suffer Plastic Instability due to Dislocation Channeling

$T_{test} = T_{irr} = 270^\circ C$

$g = 011$
Change in the Tensile Properties with Dose for 304L and 316L Stainless Steel for Tirr=50-160C
SEM Images of Fracture Surfaces of 316L Stainless Steel (dose = 9dpa) tensile specimen
Irradiation Effects in Inconel 718 (FCC) at 50-200C
Disordering of \( \gamma' / \gamma'' \) Precipitates under Proton Irradiation

\[ \gamma' \text{ and } \gamma'' \text{ disorder under proton irradiation at all dpa levels in Inconel 718.} \]
Inconel 718 (precipitation hardened) exhibits small change in yield stress during irradiation.
Loss of Ductility is Quick in Inconel 718 (precipitation hardened) after irradiation.
At higher doses fracture appearance changes to intergranular failure in Irradiated Alloy 718

Total elongation is zero at 19.8 dpa
Can we vary alloy composition to improve radiation tolerance (e.g. add precipitates or solutes)?
Do other metal alloys show promise?

Irradiation Effects in Mod 9Cr-1Mo and HT-9 (BCC) at 50-400°C
Stress/Strain Curves for Mod 9Cr-1Mo after irradiation in a Spallation Environment
The Change in the Tensile properties with dose for Mod9Cr-1Mo
Fig. 6. Test temperature dependence of the SP energy for specimens of different irradiation doses.

From Dai et al., JNM, v. 318 (2003), 192-199
The ACO-3 duct was analyzed after irradiation in the Fast Flux Test Facility (FFTF), Hanford site, WA.
Mechanical Test Results on ACO-3 Duct Show Strong Effects of Irradiation Temperature
TEM analysis of ACO-3 Duct Material (B.H. Sencer, INL, O. Anderoglu, J. Van den Bosch, LANL)

T=384°C, 28 dpa
- G-phase precipitates and alpha prime observed
- No void swelling observed.

T=450°C, 155 dpa
- Precipitation observed
- Dislocations of both \( a/2 <111> \) and \( a <100> \)
- Loops of \( a <100> \)
- Void swelling observed (~0.3 %)

T=505°C, 4 dpa
- No precipitation or void swelling observed.
Nanostructured Ferritic Alloys Show Promise as Advanced Radiation Tolerant Materials

- Strength & damage resistance derives from a high density Ti-Y-O nano-features (NFs)
- NFs complex oxides ($Y_2Ti_2O_7$, $Y_2TiO_5$) and/or their transition phase precursors with high M/O & Ti/Y ratios (APT)
- MA dissolves Y and O which then precipitate along with Ti during hot consolidation (HIP or extrusion)
- Oxide dispersion strengthened alloys also have fine grains and high dislocation densities

UCSB, LANL, ORNL
Ductility Retention observed in MA957 after irradiation to 6 dpa at 290°C
Reactor Conditions/Materials Performance
BWR/LWR
Gen IV Reactors
  VHTR- SCWR
  SFR-LFR-MSR
### Pressurized water reactors (PWR) and Boiling Water Reactors (BWR)- Present Reactor Fleet

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Purpose</th>
<th>Reactor Conditions</th>
<th>Materials Issues</th>
</tr>
</thead>
</table>
| PWR     | Power production (1.5GWe/reactor) | Water coolant (~288-360°C) | Cladding:  
  • Fuel clad chemical interaction  
  • Hydride formation  
  • Zircaloy corrosion  
  Water coolant piping:  
  • Stress Corrosion Cracking /IASCC  
  Pressure Vessel:  
  • Aging effects |
| BWR     |         | Thermal and Fast neutron spectrum 2-4 dpa/year |                |

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**Materials Issues**

**Cladding:**
- Fuel clad chemical interaction
- Hydride formation
- Zircaloy corrosion

**Water coolant piping:**
- Stress Corrosion Cracking /IASCC

**Pressure Vessel:**
- Aging effects
Construction materials for current reactor designs are diverse

<table>
<thead>
<tr>
<th></th>
<th>LWR</th>
<th>SFR</th>
<th>GFR/VHTR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coolant</td>
<td>Water</td>
<td>Sodium</td>
<td>Helium</td>
</tr>
<tr>
<td>Temperature</td>
<td>288-360°C</td>
<td>400-550°C</td>
<td>550-1100°C</td>
</tr>
<tr>
<td>Cladding</td>
<td>Zirconium-based</td>
<td>9 or 12Cr steels</td>
<td>SiC/SiC</td>
</tr>
<tr>
<td>Core Internals</td>
<td>304/316 SS</td>
<td>316 SS</td>
<td>SiC/Alloy 800H</td>
</tr>
<tr>
<td>Vessel</td>
<td>Steel/308 SS</td>
<td>316 SS</td>
<td>Steel/316 SS</td>
</tr>
<tr>
<td>Heat Exchanger</td>
<td>Alloy 600/690</td>
<td>9-12Cr/316 SS</td>
<td>Alloy 617</td>
</tr>
<tr>
<td>Piping</td>
<td>304/316 SS</td>
<td>9-12Cr/316 SS</td>
<td>Alloy 617</td>
</tr>
</tbody>
</table>

- Despite considerable differences in operating parameters, there are common material uses between LWR and SFR applications.
Very High Temperature Reactor
VHTR- NGNP

Reactor Purpose:
More Efficient Power production
Inherent passive safety features

Reactor Conditions
He coolant
950C outlet temperature
600 MWt
Solid graphite block core
2-4 dpa/year

Materials Issues
Improved metallic materials for VHTR pressure vessels (operating temperature ~450C).
Improvements in graphite properties (oxidation resistance and structural strength)
High Temperature Mechanical properties of coolant piping (e.g. Inconel 617)
Development of materials for the intermediate heat exchanger
Supercritical Water Reactor

Reactor Purpose:
More Efficient Power production

Reactor Conditions
supercritical water coolant
550°C outlet temperature
1700 MWe
>20 MPa
2-4 dpa/year

Materials Issues
corrosion and stress corrosion cracking,
radiolysis and water chemistry
dimensional and microstructural stability and strength,
embrittlement and creep resistance of fuel cladding and
structural materials.
temperature range of 280–620°C and irradiation damage
dose ranges of 10–30 displacements per atom (dpa)
(thermal spectrum) and 100–150 dpa (fast spectrum)
Why Fast reactors?

- Fast reactors are able to address the back end of the fuel cycle
- Produce energy out of waste
- Extend the fuel resources

![Diagram showing Relative Radiotoxicity over time with different stages of fuel cycle such as Spent Fuel, Fission Products, and Natural Uranium Ore. The diagram also highlights different timeframes for transmutation of actinides and separation of Pu and U.](image-url)
Sodium cooled fast reactor/ Lead Fast Reactor

Reactor Purpose:
High level nuclear waste Transmutation
Power production
Actinide management

Reactor Conditions
Na, Pb or Pb/Bi coolant
550°C to 800°C outlet temperature
20-30 dpa/year

Materials Issues
In Core-
- High dose irradiation effects
- FCCI
- Liquid metal corrosion
  - Lead corrosion of materials
  - Liquid metal embrittlement

Swelling in 316L SS

FFTFT research reactor in Hanford (1980-1993)
Gas cooled fast reactor

Reactor Purpose:
- High level nuclear waste Transmutation
- More efficient Power production
- Actinide management

Reactor Conditions
- He or Supercritical CO\(_2\) coolant
- 850°C outlet temperature
- Several Fuel options and core configurations
- 20-30 dpa/year

Materials Issues
- Fuel development (must achieve high-power density and retain fission gases at high burnup and temperature)
- Proposed fuel is a composite ceramic (CERCER) with closely packed and coated actinide carbide kernels or fibers.
- Alternative fuel concepts
  - fuel particles with large kernels and thin coatings and ceramic-clad solid solutions.
  - Nitride compounds, enriched 99.9% in N-15
Reactor Purpose:
High level nuclear waste
Transmutation
Fast Reactor power
Liquid fuel core removes radiation effects concerns in the fuel

Reactor Conditions
Fuel: liquid Na, Zr, U and Pu fluorides or chlorides
700-800°C outlet temperature
1000 Mwe
Core materials - Ni-based alloys (pressure vessel), graphite, SiC (solid fuel))
Low pressure (<0.5 MPa)
20-30 dpa/year for solid fuel clad

Major Materials Issues
- Materials compatibility testing in a controlled chemistry test loop
- Materials compatibility testing in a controlled chemistry test loop under irradiation.
- Radiation damage to pressure vessel and coolant piping
## Summary Reactor Operating Conditions

<table>
<thead>
<tr>
<th>Reactor Type</th>
<th>Fuel Materials</th>
<th>Fuel Temperature</th>
<th>Pellet to Clad bond</th>
<th>Coolant Type</th>
<th>Structural Materials for Core Internals</th>
<th>Lifetime Dose (dpa)</th>
<th>Structural Temperatures</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gen IV/ Lead Fast Reactor LFR</td>
<td>U/PuN, TRUN (enriched to N³⁵)</td>
<td>500-600°C</td>
<td>Lead</td>
<td>Pb or LBE</td>
<td>FeMnCoMartensitic Steel alloys</td>
<td>150-200</td>
<td>400-600°C</td>
</tr>
<tr>
<td>Gen IV/ Sodium Fast Reactor SFR</td>
<td>Metal(U-TRU - 10%Zr Alloy), MOX/TRU (heating)</td>
<td>600-800°C</td>
<td>Sodium</td>
<td>Sodium</td>
<td>FeMnCoMartensitic Steel alloys</td>
<td>150-200</td>
<td>400-550°C</td>
</tr>
<tr>
<td>Gen IV/ Gas cooled Fast Reactor GFR</td>
<td>UPuC/SiC (50/50%) with 20% Pu content ; Solid Solution fuel with SiC/SiC cladding</td>
<td>2000 +</td>
<td>Helium</td>
<td>Helium</td>
<td>Nickel Superalloys/Ceramic Composites</td>
<td>80</td>
<td>500-1200°C</td>
</tr>
<tr>
<td>Fusion Energy</td>
<td>N/A</td>
<td>N/A</td>
<td>Pb-Li</td>
<td>F/M steels; Vanadium alloys, Ceramics</td>
<td>150</td>
<td>300-1000°C</td>
<td></td>
</tr>
<tr>
<td>LWR – PWR, BWR</td>
<td>UO₂</td>
<td>800-1600°C</td>
<td>Helium</td>
<td>Water</td>
<td>316L Ferritic pressure vessel, Zircaloy cladding</td>
<td>Cladding ~10 dpa, Internals up to 80 dpa</td>
<td>200-300°C</td>
</tr>
<tr>
<td>Very High Temperature Reactor (VHTR, NGNP)</td>
<td>TRISO</td>
<td>800-2000°C</td>
<td>Intimate contact</td>
<td>Helium</td>
<td>Ni-based alloys, ceramis and graphite</td>
<td>~10 dpa</td>
<td>700-1000°C</td>
</tr>
<tr>
<td>Supercritical Water Reactor (SCWR)</td>
<td>UO₂</td>
<td>800-2000°C</td>
<td>Helium</td>
<td>Water</td>
<td>F/M steels, austenitic steels</td>
<td>10-30 thermal 100-150 Fast</td>
<td>300-600°C</td>
</tr>
<tr>
<td>Mottten Salt Reactor (MSR)</td>
<td>Na₂, Zr, U, Pu fluoreides</td>
<td>700-800°C</td>
<td>N/A</td>
<td>N/A</td>
<td>Ni-based alloys, graphite</td>
<td>100-150 dpa</td>
<td>600-800°C</td>
</tr>
</tbody>
</table>
## Summary of Materials Performance Issues

<table>
<thead>
<tr>
<th>Reactor type</th>
<th>Primary Materials</th>
<th>Performance Issues</th>
</tr>
</thead>
<tbody>
<tr>
<td>Light Water Reactors (PWR/BWR)</td>
<td>Ferritic pressure vessel steels, Fe-based austenitic stainless steels, zirconium alloys</td>
<td>IGSCC, IASCC, Fuel clad mechanical interaction, hydriding, Radiation embrittlement (DBTT), hydrogen embrittlement</td>
</tr>
<tr>
<td>Very High Temperature Reactor (VHTR)</td>
<td>Ni-based superalloys, Graphite, ferritic/martensitic steels, W/Mo Alloys, SiC/SiC composites</td>
<td>Helium embrittlement, creep strength, swelling, RIS, transmutation, toughness, oxidation</td>
</tr>
<tr>
<td>Sodium Fast Reactor (SFR)</td>
<td>Fe-based austenitic SS, Ferritic/martensitic steels,</td>
<td>Radiation Embrittlement (DBTT), toughness, helium embrittlement, swelling, RIS, corrosion, FCCI</td>
</tr>
<tr>
<td>Lead Fast Reactor (LFR)</td>
<td>Fe-based austenitic SS, Ferritic/martensitic steels,</td>
<td>Radiation Embrittlement (DBTT), toughness, helium embrittlement, swelling, RIS, corrosion, FCCI, liquid metal embrittlement</td>
</tr>
<tr>
<td>Supercritical Water Reactor (SCWR)</td>
<td>Ferritic pressure vessel steels, Fe-based austenitic stainless steels, zirconium alloys, ferritic/martensitic steels</td>
<td>IGSCC, IASCC, Fuel clad mechanical interaction, hydriding, Radiation/helium embrittlement (DBTT), swelling, RIS, corrosion, toughness</td>
</tr>
<tr>
<td>Gas Fast Reactor</td>
<td>Ceramics (carbides, nitrides), ceramic composites, nickel superalloys</td>
<td>Helium embrittlement, creep strength, swelling, RIS, transmutation, toughness, oxidation</td>
</tr>
<tr>
<td>Molten Salt Reactor</td>
<td>Ni-based alloys, graphite, coatings</td>
<td>Corrosion, Helium embrittlement, creep strength, swelling, RIS, transmutation, toughness, oxidation</td>
</tr>
</tbody>
</table>
Questions???

Grid-to-Rod Fretting

Fast Reactor Duct Failure

BY-97
53 dpa,
$\Delta V/V=28\%$

BY-92
52 dpa,
$\Delta V/V=30\%$

U-796
34 dpa,
$\Delta V/V=14\%$

CRUD
Upcoming webinars

21 March 2018  SCK•CEN’s R&D on MYRRHA  Prof. Dr. H.C. Hamid Ait Abderrahim, SCK-CEN, Belgium

18 April 2018  Russia BN 600 and BN 800  Dr. Iiuri Ashurko, Institute of Power and Engineering, Russia

23 May 2018  Proliferation Resistance of Gen IV Systems  Dr. Robert Bari, Brookhaven National Laboratory, USA
Call for abstracts  Extended Deadline - 31 March 2018

Track 1 & 2: Progress on Gen IV systems
Track 3: Human capital development
Track 4: Research infrastructures
Track 5: Safety and security
Track 6: Fuels and materials
Track 7: Advanced components and systems for Gen IV reactors
Track 8: Integration of nuclear reactors in low carbon energy systems
Track 9: Decommissioning & Waste Management
Track 10: Operation, Maintenance, Simulation & Training
Track 11: Construction of nuclear reactors

The symposium has two major objectives:

• to review the progress achieved for each system against the R&D goals of the 2014 Technology Roadmap Update.

• to identify the remaining challenges and associated R&D goals for the next decade necessary for the demonstration and/or deployment of the Gen IV systems, and the goal of establishing nuclear energy as a necessary element in the world’s long-term sustainable carbon-free energy mix.

MSc and PhD students, young professionals, policy makers and nuclear stakeholders are encouraged to participate