Impact of fuel assembly transportation on zirconium alloys: toward a mechanistic understanding

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INDUSTRIAL BACKGROUND: END OF LIFE OF THE FUEL ASSEMBLY

Spent Nuclear Fuel Assembly

Temperature from 400°C up to 450°C

In-reactor use  Dry transportation  In-pool storage

Neutron irradiation of the Zr alloy cladding

Post-irradiation creep of the Zr alloy cladding and radiation damage recovery

Room temperature mechanical properties at retrieval?
Recrystallized Zr-1%Nb alloy

Chemical composition (% wt)

<table>
<thead>
<tr>
<th>%wt</th>
<th>O</th>
<th>Sn</th>
<th>Fe</th>
<th>Cr</th>
<th>Nb</th>
<th>Zr</th>
</tr>
</thead>
<tbody>
<tr>
<td>RXA Zr-1%Nb</td>
<td>0.12</td>
<td>0</td>
<td>0.02</td>
<td>0</td>
<td>1</td>
<td>Bal.</td>
</tr>
</tbody>
</table>

Cladding in Zr alloy

→ creation of a high density of small loops
Heat treatment → to simulate transportation

After heat treatment at 400°C during 250 h

- As-irradiated material
  - $\langle \rho_b \rangle = 1.2 \times 10^{22} \text{ m}^{-3}$
  - $\langle d \rangle = 14 \text{ nm}$

- After heat treatment at 400°C during 250 h
  - $\langle \rho_b \rangle = 2.5 \times 10^{21} \text{ m}^{-3}$
  - $\langle d \rangle = 18 \text{ nm}$

The loop size increases while the density decreases

After heat treatment at 450°C during 960 h

- $\langle \rho_b \rangle = 1.2 \times 10^{20} \text{ m}^{-3}$
  - $\langle d \rangle = 176 \text{ nm}$

Radiation damage recovery
Loop nature using inside / outside contrast method

- Only vacancy loops at the end of the recovery
- Interstitial loops recover faster than vacancy loops

What is the loop recovery mechanism?
As-irradiated microstructure

Shrinking of the vacancy and interstitial loops

Shrinking of the smallest loop at the expense of the biggest loop

Annealed microstructure
Predictive modelling approach: cluster dynamic

Single-vacancy evolution:
\[ \frac{dC_v(i)}{dt} = \text{production} - \text{annihilation} \]
No creation of point defects during annealing

Cluster evolution:
\[ \frac{dC(n)}{dt} = a_{n-1}C(n+1) - b_nC(n) + c_{n-1}C(n-1) \]

Mean-field modeling
Sinks: clusters, dislocation lines, grain boundaries / free surfaces
Vacancy emission
Vacancy absorption
Vacancy cluster
Cluster dynamic modelling $\rightarrow$ computation of loop size and density

Correct agreement between experiment and modeling
Evolution of vacancy and interstitial loop size distribution at 400°C

Evolution of the proportion of vacancy loops during heat treatment

<table>
<thead>
<tr>
<th>Annealing temperature (°C)</th>
<th>Annealing time (h)</th>
<th>Experiment Number of analysed loops</th>
<th>Percentage of vacancy loops (%)</th>
<th>Modelling Percentage of vacancy loops (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>As-irradiated</td>
<td>As-irradiated</td>
<td>-</td>
<td>50 [23]</td>
<td>50</td>
</tr>
<tr>
<td>350</td>
<td>250</td>
<td>19</td>
<td>50</td>
<td>45</td>
</tr>
<tr>
<td>350</td>
<td>500</td>
<td>24</td>
<td>50</td>
<td>43</td>
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<tr>
<td>400</td>
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<td>37</td>
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<td>100</td>
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<tr>
<td>400</td>
<td>500</td>
<td>38</td>
<td>71</td>
<td>100</td>
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<tr>
<td>450</td>
<td>500</td>
<td>38</td>
<td>71</td>
<td>100</td>
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<tr>
<td>450</td>
<td>960</td>
<td>13</td>
<td>100</td>
<td>100</td>
</tr>
</tbody>
</table>

→ Correct agreement between experiment and modeling
High density of small loops -> pinning of dislocations -> radiation hardening

\[ \Delta \tau_c = \alpha \mu b \sqrt{Nd} \]

Hardness as a function of the loop density & size

\[ \rightarrow \text{Radiation hardening recovery during heat treatment} \]

\[ \rightarrow \text{Correct prediction of radiation hardening recovery} \]
Post-irradiation creep → to simulate transportation

As-irradiated

After post-irradiation creep:
→ decrease of the loop density
( + increase of the loop size)
→ Radiation damage recovery during creep
→ Effect on the tensile mechanical behavior?
- Irradiation induced hardening (increase of the yield stress and ultimate tensile strength)

- Loss of macroscopic ductility (decrease of the uniform elongation), but the failure remains ductile (strong necking)

→ What are the deformation mechanisms?
RT tensile test on non-irradiated material

Ring tensile test at room temperature after irradiation

→ homogeneous glide of dislocations in the prismatic planes mainly

Channels in the basal plane

Channels in the prismatic planes

- Heterogeneous deformation inside the grains (dislocation channeling)
- both basal and prismatic slip can be activated (depending on the grain orientation)
- easier basal slip than before irradiation
High density of small loops
-> pinning of dislocations
-> radiation hardening

Clearing of loops by gliding dislocations
-> microscopic strain softening
-> early localization of the deformation at the specimen scale
-> decrease of the Uniform Elongation observed during ring tensile tests

-> Why is the basal slip more easily activated after irradiation than before irradiation?
Loop Burgers vector:
\[ \mathbf{b} = \frac{1}{3} [11\bar{2}0] \]

Dislocation Burgers vector:
\[ \mathbf{b} = \frac{1}{3} [11\bar{2}0] \]

B dislocation - loop

P dislocation - loop

Glissile junction

\( \rightarrow \) Easy Basal channeling / difficult Prismatic channeling

Sessile junction
- Recovery of the radiation induced hardening after creep test
- Recovery of the macroscopic ductility (uniform elongation)

→ What are the deformation mechanisms?
After ring tensile test following post-irradiation creep:

-> very few remaining loops → lower radiation hardening

-> homogeneous prismatic glide mainly, no channel

→ recovery of the uniform elongation
CONCLUSIONS

Radiation damage recovery:
- Occurs by exchange of vacancies between loops
  → Shrinking of Int loops and of small Vac loops at the expense to bigger Vac loops

Cluster dynamic modeling:
→ Correct prediction radiation damage recovery & radiation hardening recovery

Mechanical behavior after neutron irradiation:
- Decrease of the uniform elongation due to the dislocation channeling
- Failure occurs after a strong necking → ductile failure
- Evolution of activated slip systems explained by junction between dislocations and loops

Impact of transportation (post-irradiation creep):
- Radiation hardening recovery due to the loop annealing during creep
- Recovery of the uniform elongation due to homogeneous glide of dislocations

→ Better knowledge and understanding of the effects of transportation on the mechanical properties of the fuel assembly at retrieval
Thank you!