U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY

Innovative Materials for Advanced Reactor Cladding

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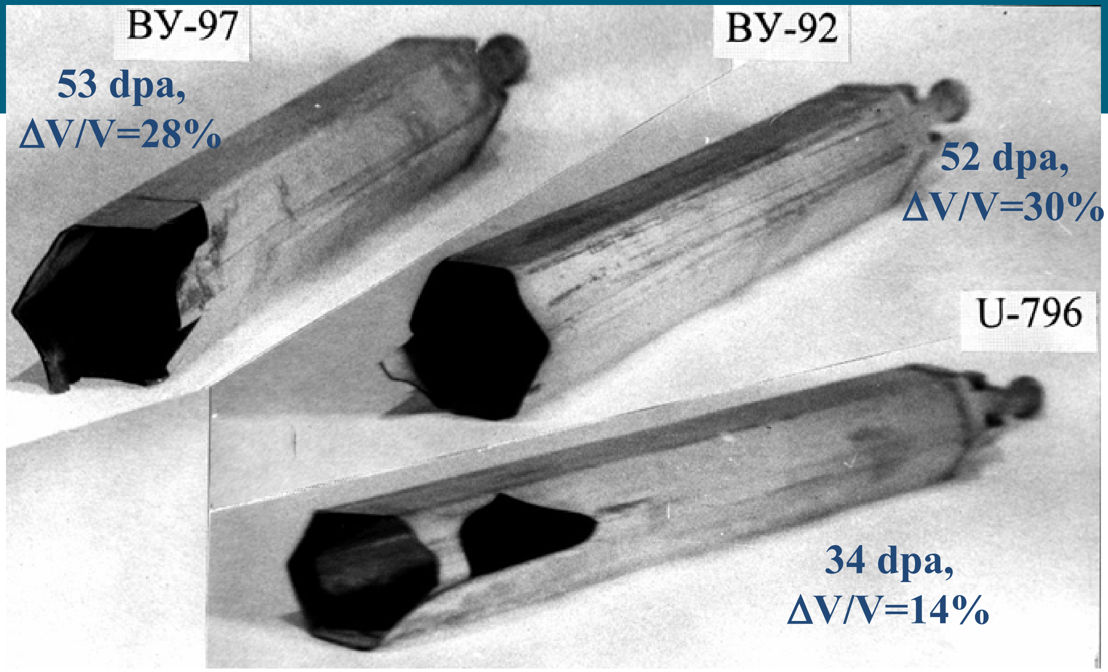
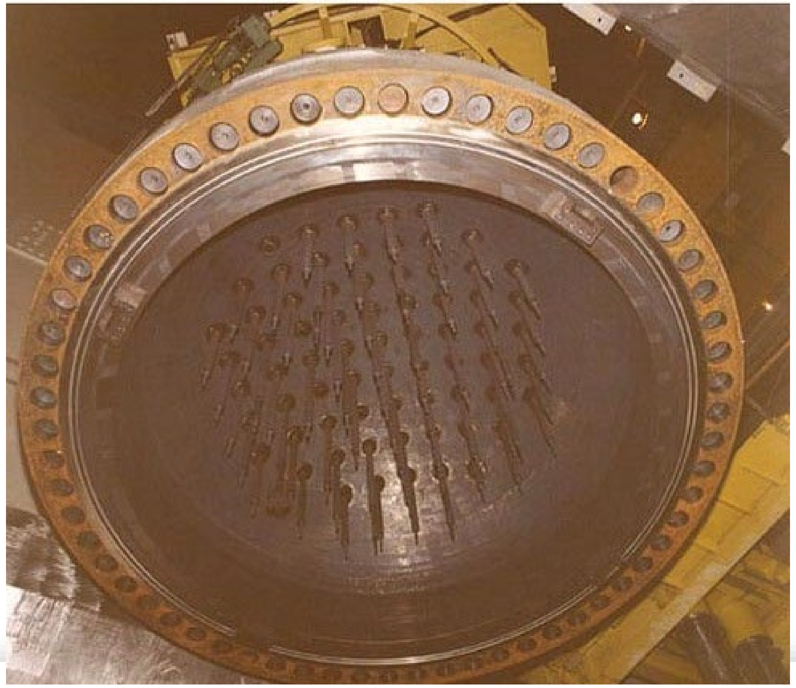
Materials in Nuclear Energy Systems can Fail



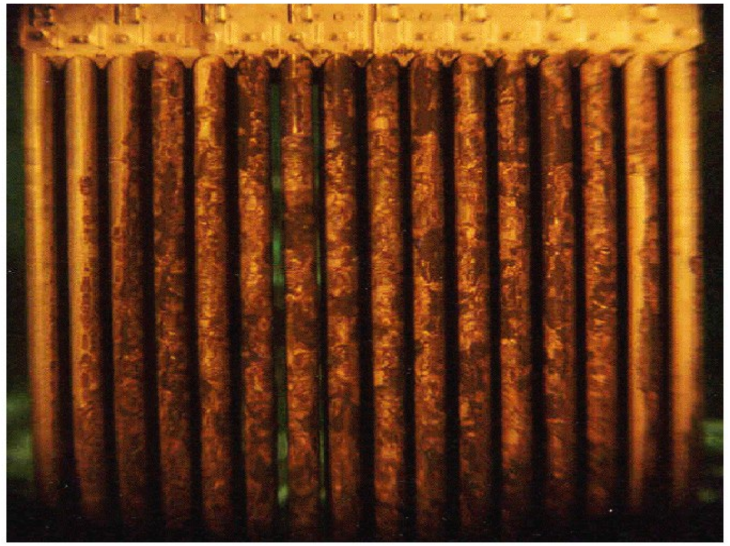
Fast Reactor Duct Failure



Davis-Besse Reactor Vessel Head Degradation

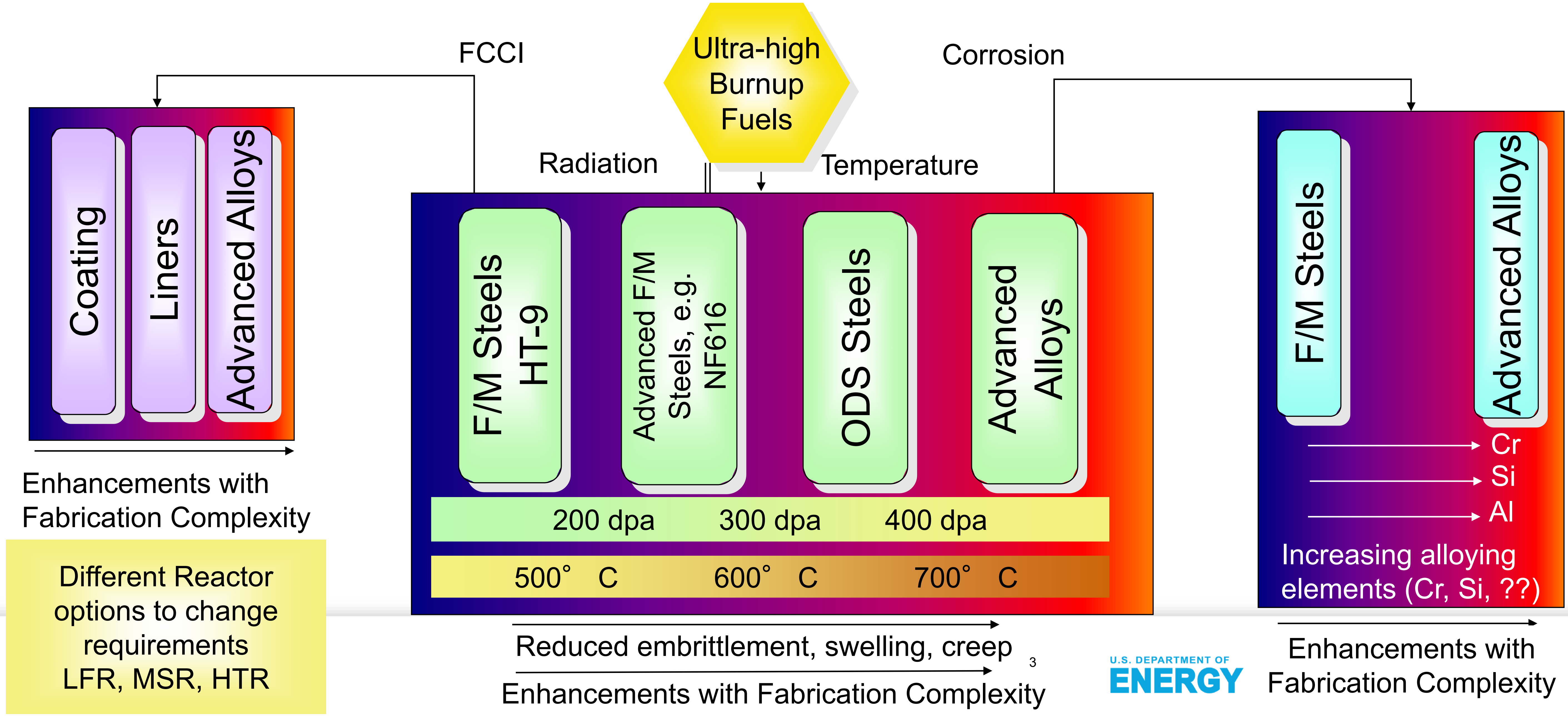
Grid-to-Rod Fretting

CRUD



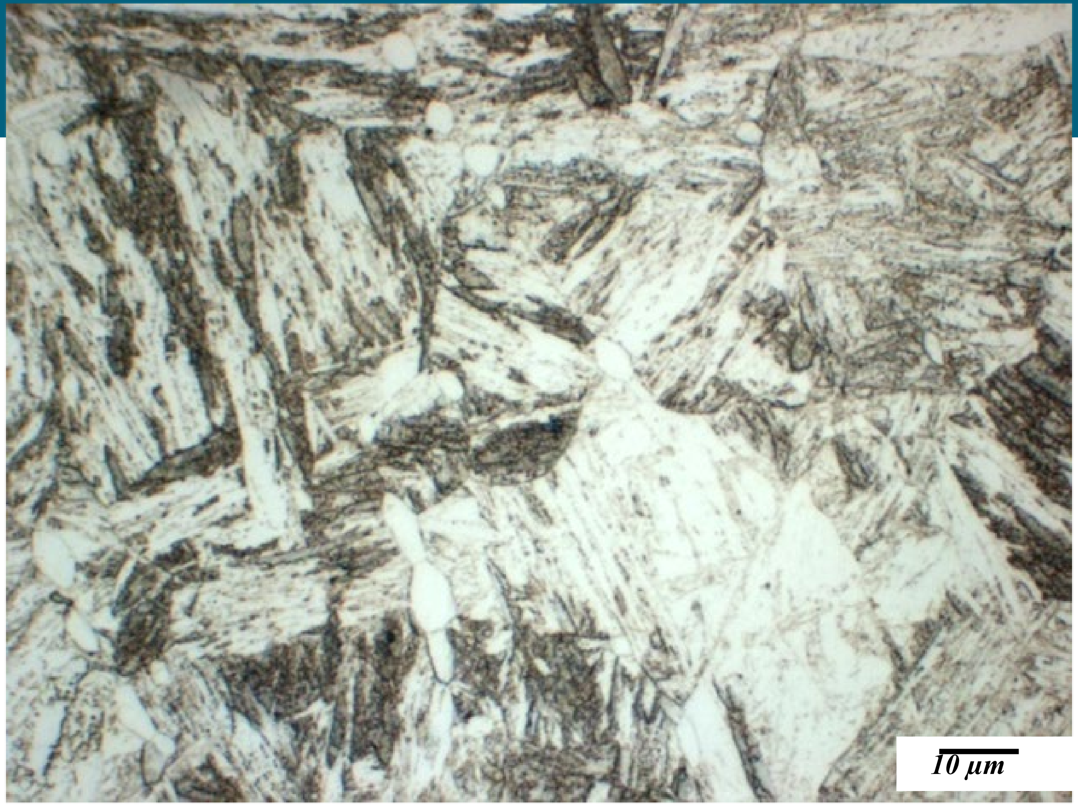
Extreme Environments that Must be Overcome in Developing High Dose Radiation Tolerant Advanced Reactor Cladding Materials

Ultimate goal: Develop and test innovative new cladding materials with the potential to revolutionize or transform future nuclear energy applications



Example of a High Burnup Cladding Material: HT-9

- Tempered Martensitic Steel with a ferritic lath microstructure
- Elemental composition is Fe-12Cr-1Mo-0.2C-0.5W-0.3Si-0.5Ni-0.3V-0.3Mn
- Shows excellent void swelling resistance to >200 dpa but strong hardening for irradiations below 400C.



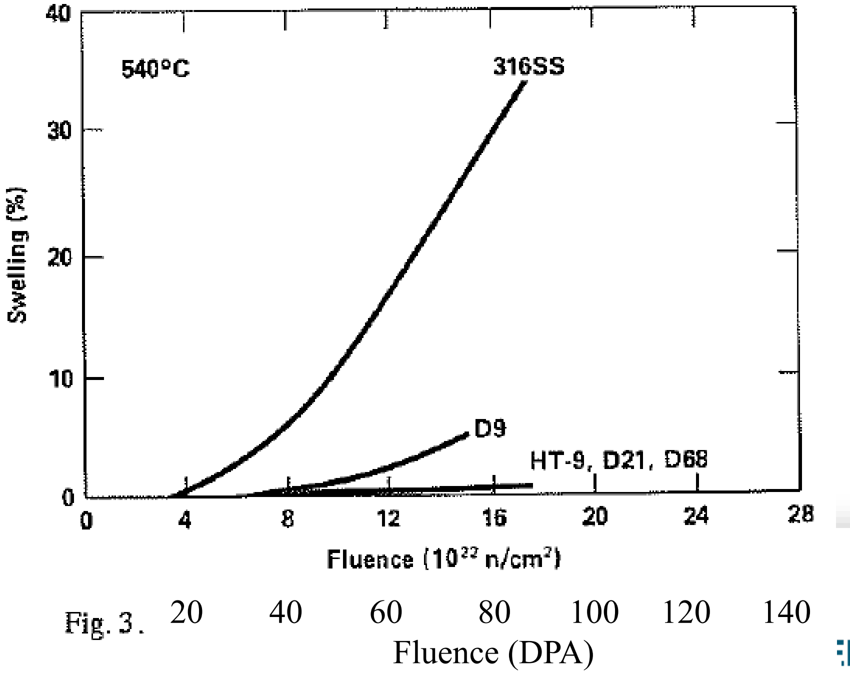
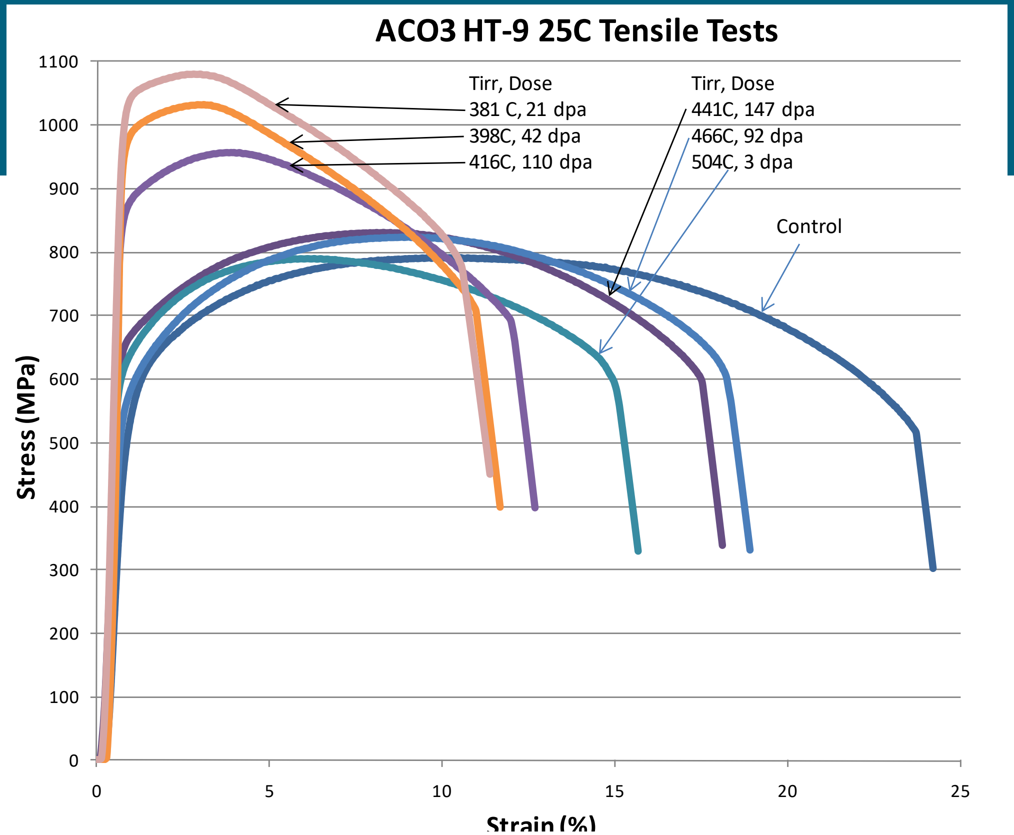


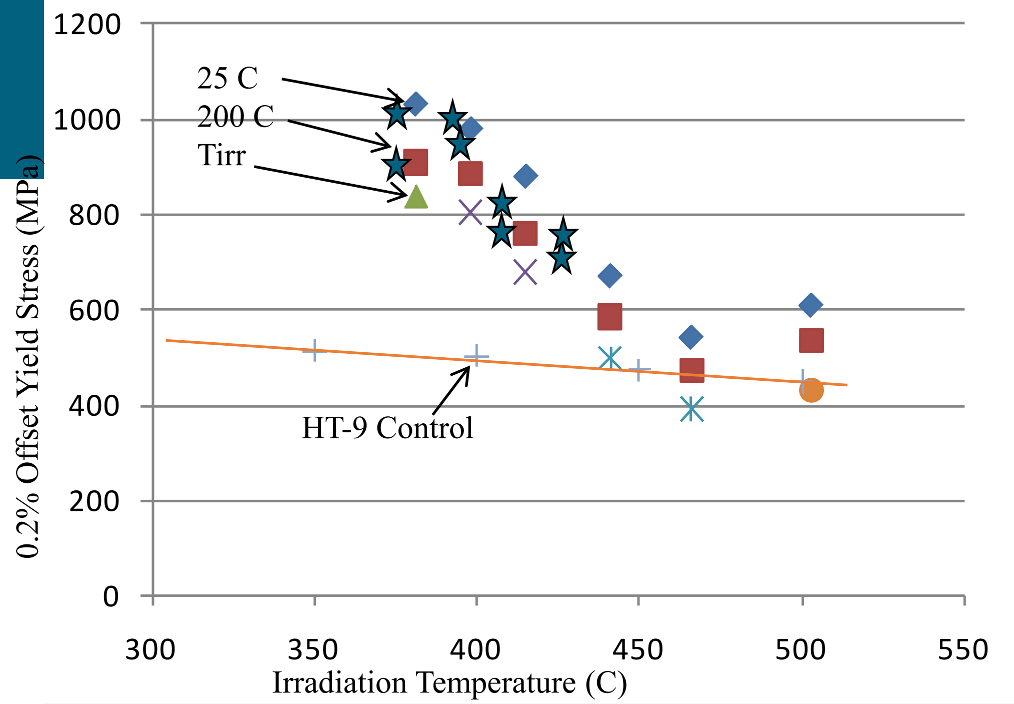
Fig. 3.

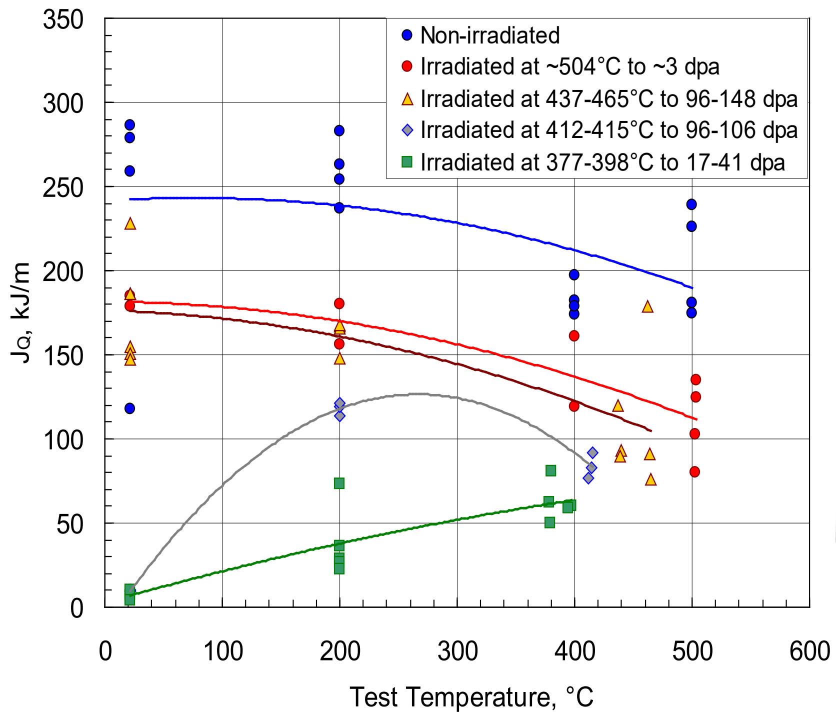
Chin, Neuhold, Straalsund, Nuc. Tech, 1982

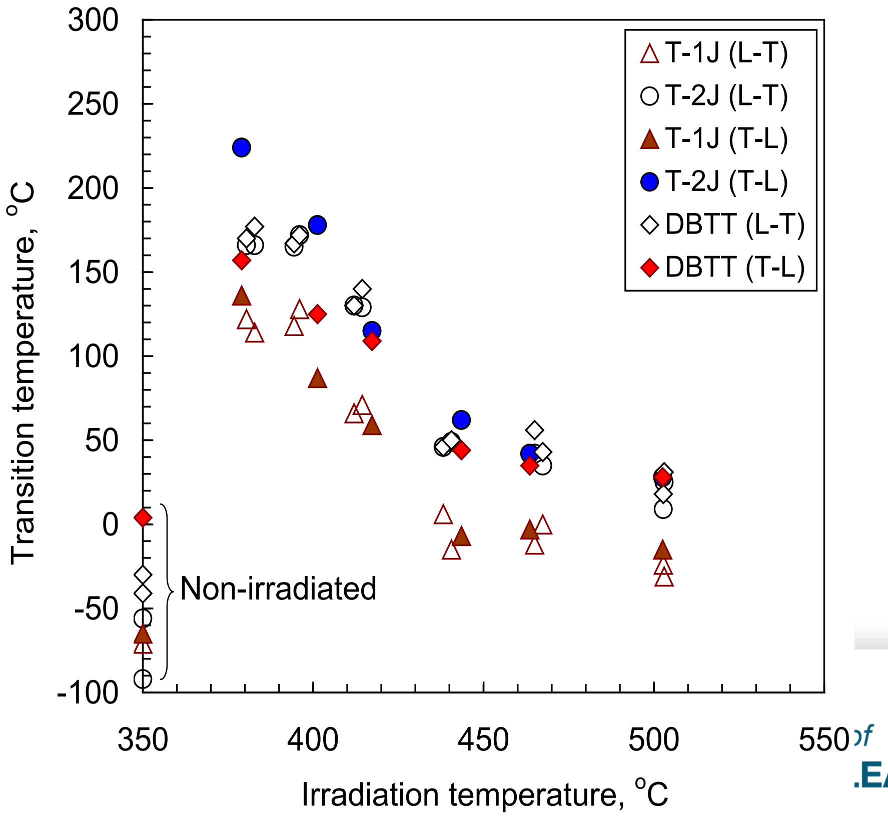
:RGY

Mechanical Test Results on ACO-3 Duct Show strong Effects of Irradiation Temperature

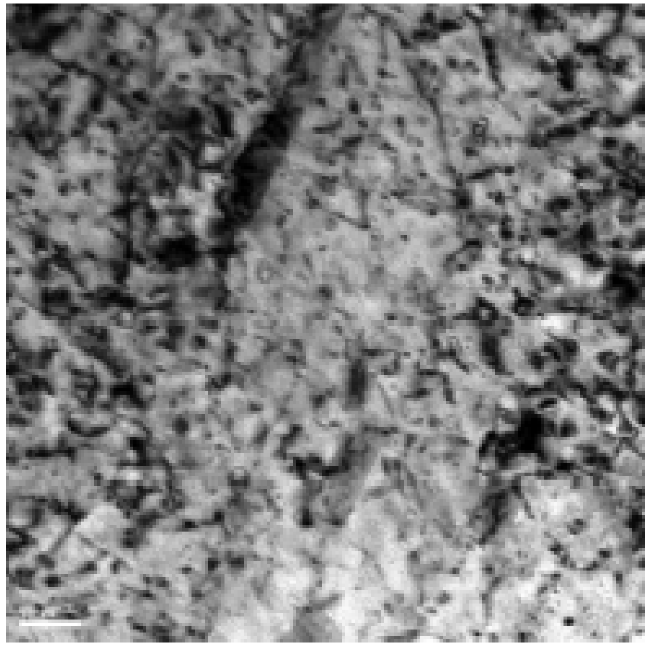


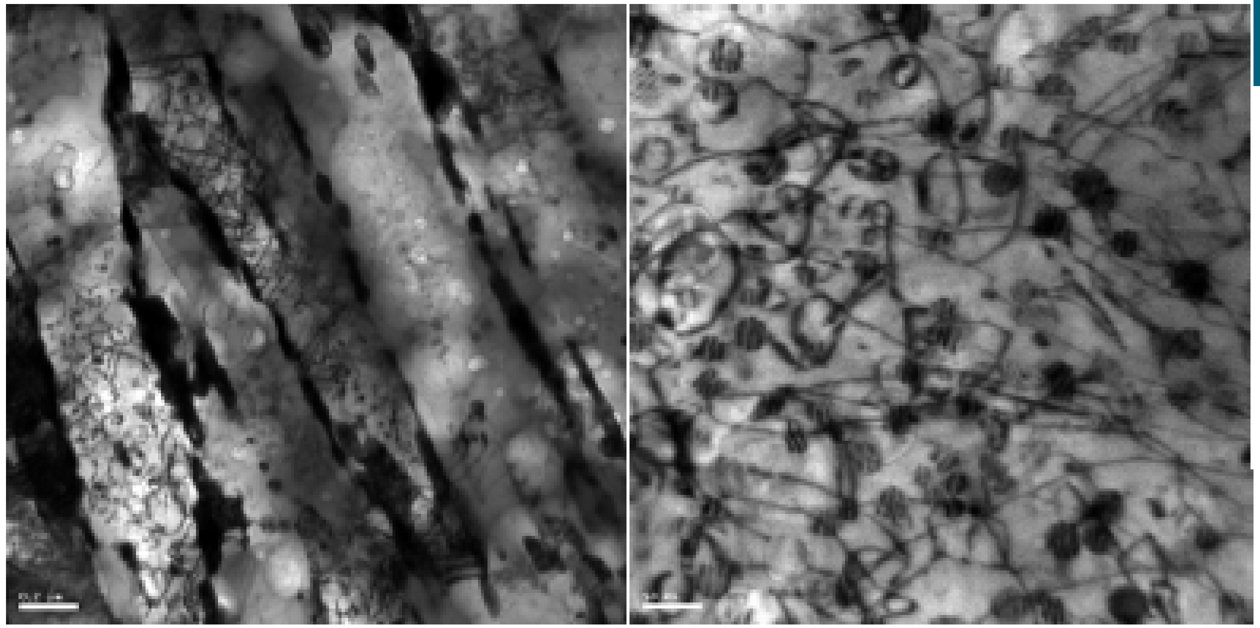


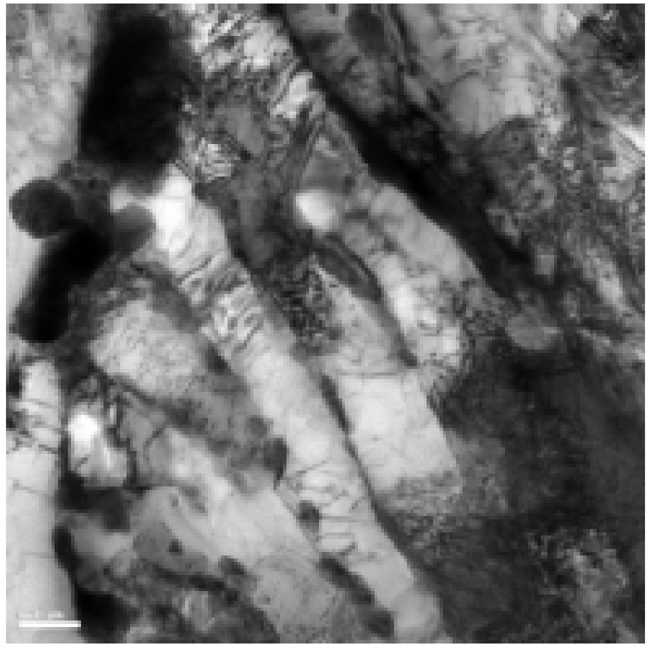




TEM analysis shows variation of irradiation defects with irradiation temperature (B.H. Sencer, INL, O. Anderoglu, J. Van den Bosch, LANL)



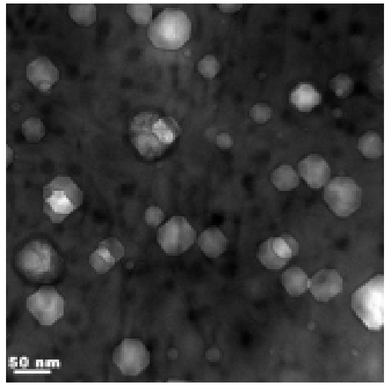


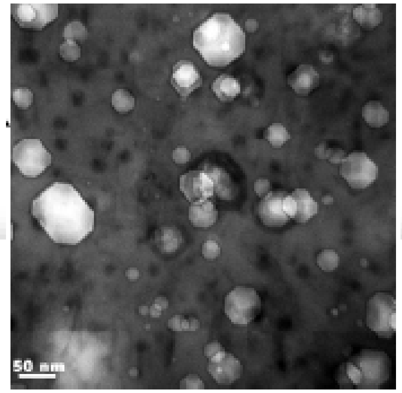


T=384C, 28 dpa

- G-phase precipitates and alpha prime observed

•No void swelling observed.





T=450C, 155 dpa

- Precipitation observed
- Dislocations of both a/2<111> and a<100>
- Loops of a<100>
- Void swelling observed (~0.3 %)

T=505C, 4 dpa

•No precipitation or void swelling observed.

Small Angle Neutron Scattering Measurements
Obtain accurate measurement of α' vs. dose and irr. Temperature
Measurements completed on 5 specimens from ACO-3 duct

Limitations in service life with HT9

Although HT9 shows excellent void swelling resistance to doses over 200 dpa, it has some limitations

- Low Temperature Embrittlement below 400C
- Fuel Clad chemical interaction with metallic fuels
- Low creep strength above 600C
- Radiation induced segregation and second phase precipitation after high dose irradiations
- Corrosion limitations in other coolants (e.g. lead or molten salt)

Vision of IMARC program

- Develop and test innovative new cladding materials with the potential to revolutionize or transform future nuclear energy applications
  - New alloy compositions
  - Coatings to eliminate FCCI and improve corrosion resistance
  - Innovative microstructures with extreme radiation tolerance
  - Innovative manufacturing and joining methods to produce hermetically sealed thin-walled tubing for cladding applications
  - Testing methods to investigate high dose radiation tolerance
    - High dose irradiations
    - Mechanical testing over uniformly irradiated materials
  - Methods to accelerate materials qualification

Materials Research in the Office of Nuclear Energy

NE-4: Fuel Cladding Materials

- Innovative Materials for Advanced Reactor Cladding (IMARC)
- Leading Innovation in Fuel Technology (LIFT)
- Accident Tolerant Fuels (ATF)

- Light Water Reactor Sustainability (LWRS)
- Advanced Materials and Manufacturing Technologies (AMMT)
- Advanced Reactor Technologies (ART)

NE-5: Structural materials

Materials Research in the Office of Nuclear Energy: Complementary Research

NE-4: Fuel Cladding Materials

Innovative Materials for Advanced Reactor Cladding (IMARC)

Leading Innovation in Fuel Technology (LIFT)

Accident Tolerant Fuels (ATF)

Light Water Reactor Sustainability (LWRS)

Advanced Materials and Manufacturing Technologies (AMMT)

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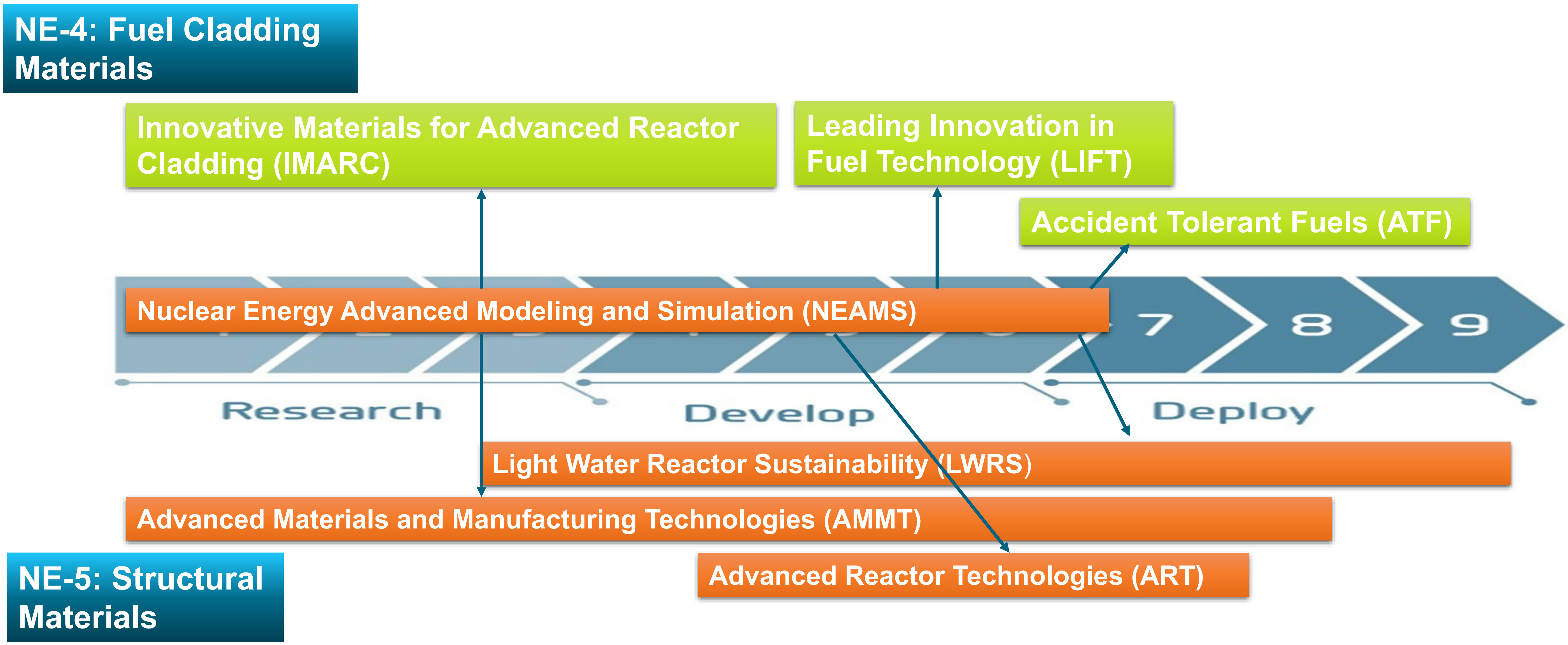
NE-5: Structural Materials

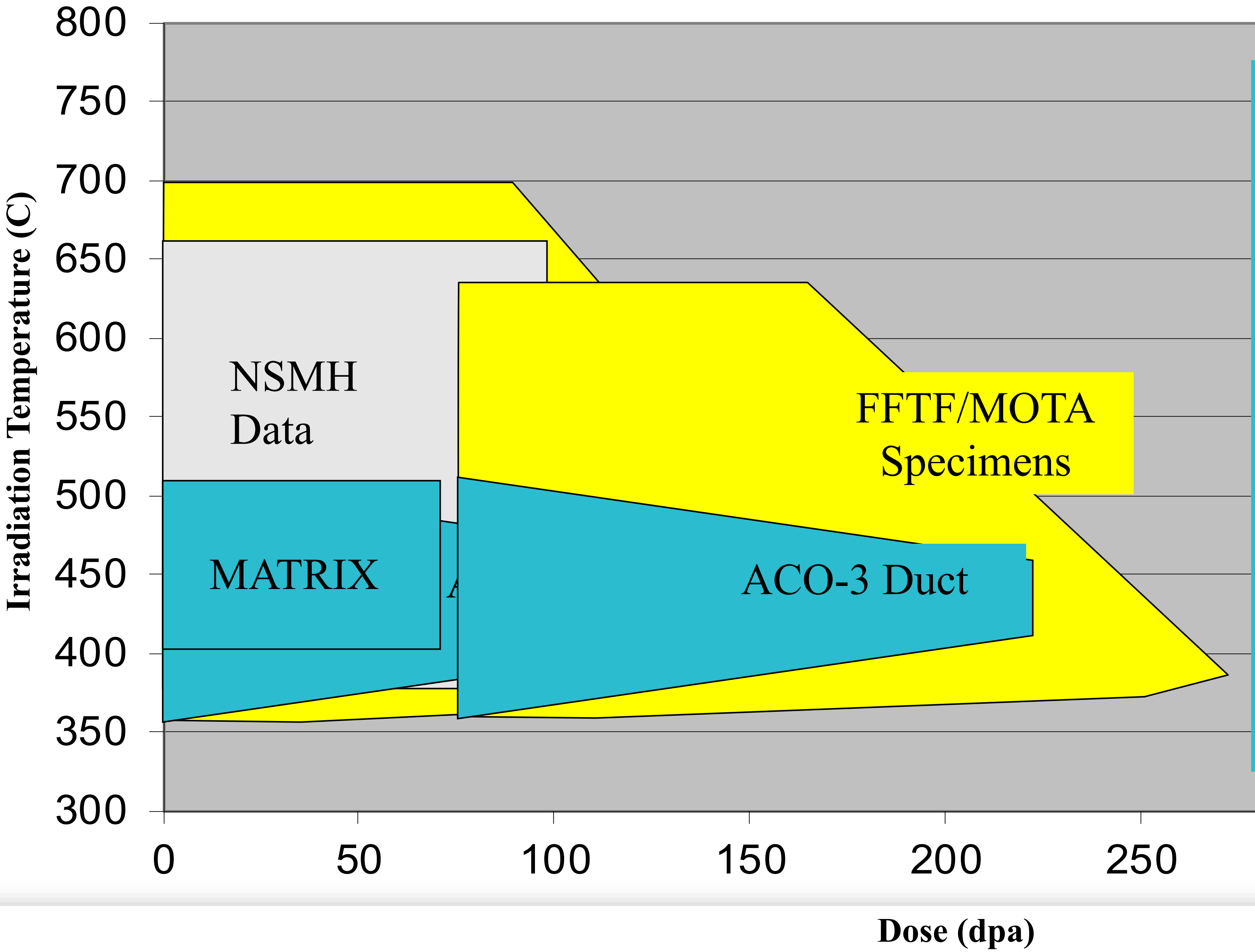
Materials Research in the Office of Nuclear Energy: Advanced Modeling and Simulation



Materials Innovations Required for Development of Advanced Reactor Cladding

- Advanced Alloy development and testing over a wide range of composition space.
- Innovative manufacturing technologies for thin walled tube development and coating techniques over long lengths of tubing (e.g. 9 feet)
- Joining technologies for thin walled tubes of innovative new alloys
- High dose irradiation testing techniques and small scale testing on uniformly irradiated volumes.

How can we obtain high dose irradiation data?



How do we get here??
International Collaborations

FFTF/MOTA extended irradiations in other irradiation facilities (BOR60).

Modeling and Simulation (NEAMS)
High Dose Ion irradiations

Summary

- Overall aim of this workshop is to obtain input from industry, national laboratories and universities leading to priority research directions for this new program on Innovative Materials for Advanced Reactor Cladding
  - Next on the agenda are talks from industry, national lab and universities
  - Encourage questions and suggestions from the audience