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Fuel Cycle Research and Development:

Core Materials Technologies Overview - Fast Reactor and LWR Fuel Cladding

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DOE NE Materials-Cross-Coordination Webinar

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Advanced Fuels Campaign Mission & Objectives in the Fuel Cycle Research and Development Program

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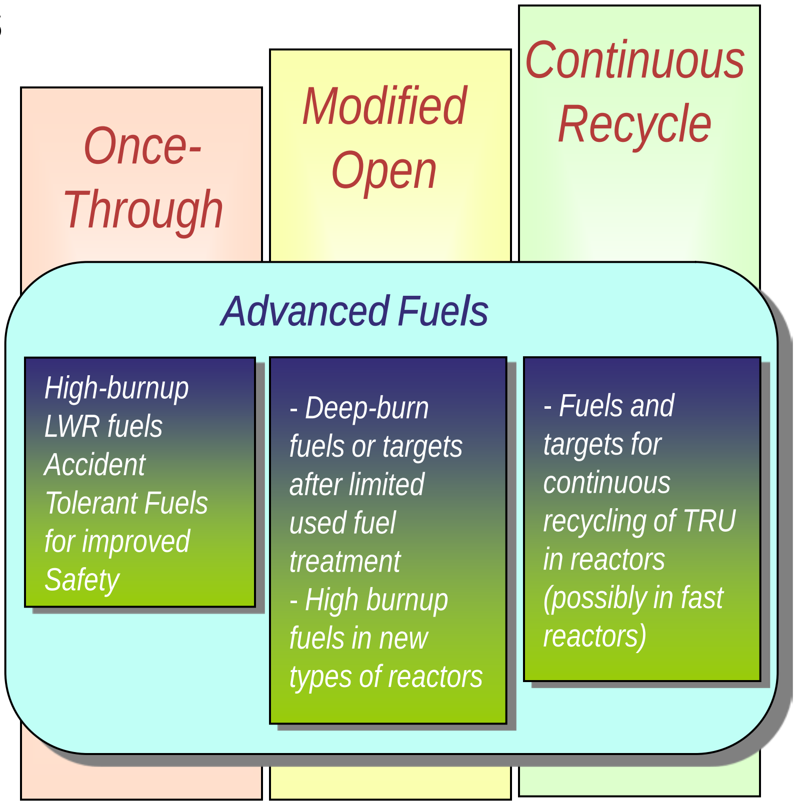
- **Mission**

  Develop and demonstrate fabrication processes and in-pile (reactor) performance of advanced fuels/targets (including the cladding) to support the different fuel cycle options defined in the NE roadmap.

- **Objectives**

  Development of the fuels/targets that
  - Increases the efficiency of nuclear energy production
  - Maximize the utilization of natural resources (Uranium, Thorium)
  - Minimizes generation of high-level nuclear waste (spent fuel)
  - Minimize the risk of nuclear proliferation

- **Grand Challenges**
  - Multi-fold increase in fuel burnup over the currently known technologies
  - Multi-fold decrease in fabrication losses with highly efficient predictable and repeatable processes



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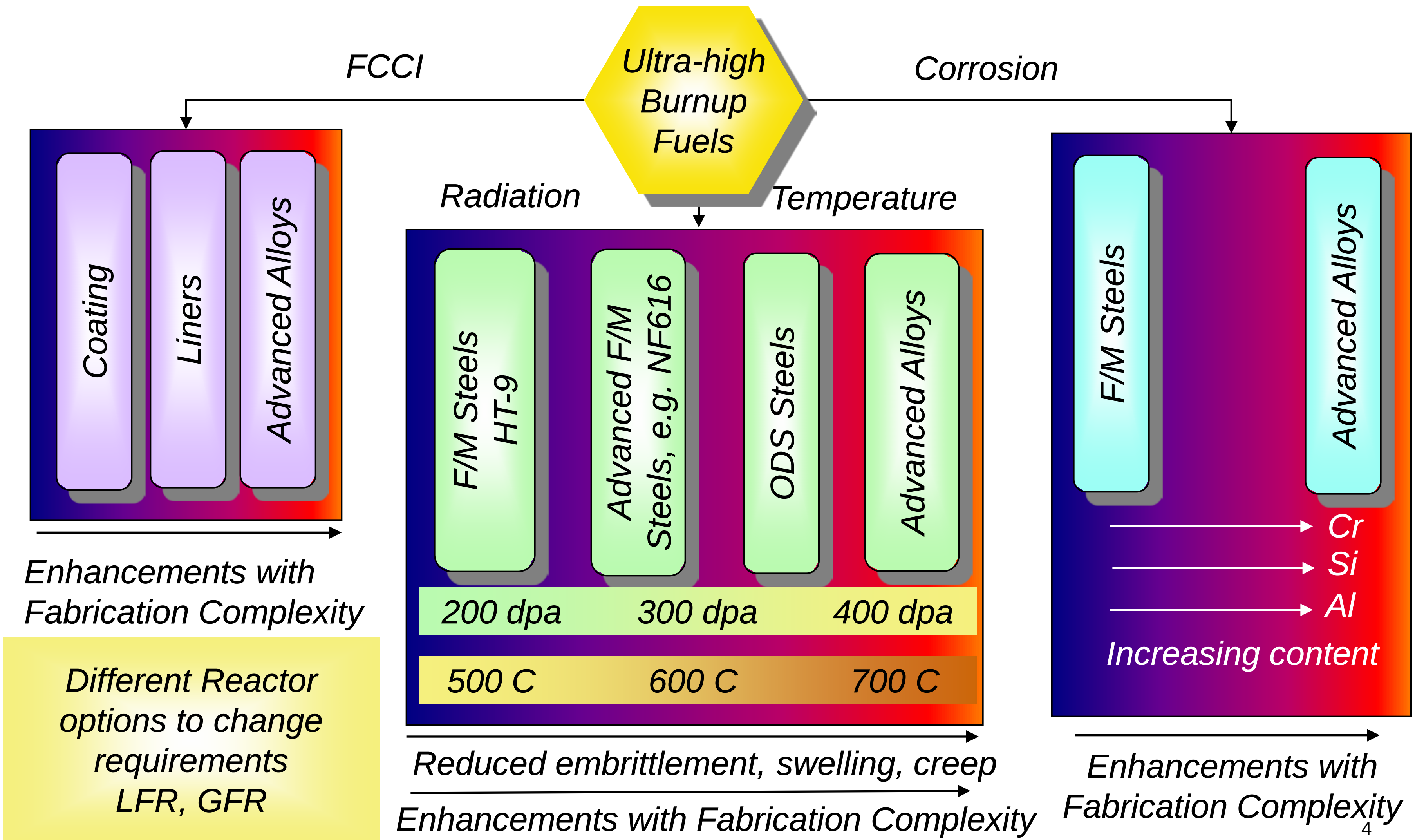
Approach to Enabling a Multi-fold Increase in Fuel Burnup over the Currently Known Technologies

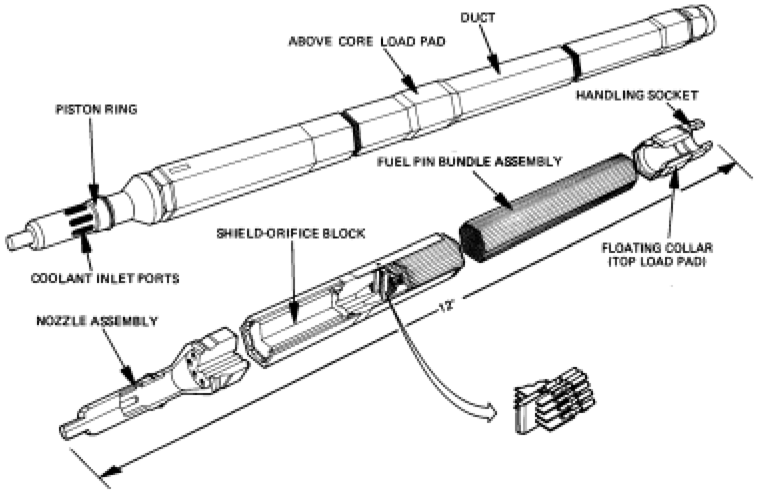
Ultimate goal: Develop advanced materials immune to fuel, neutrons and coolant interactions under specific reactor environments

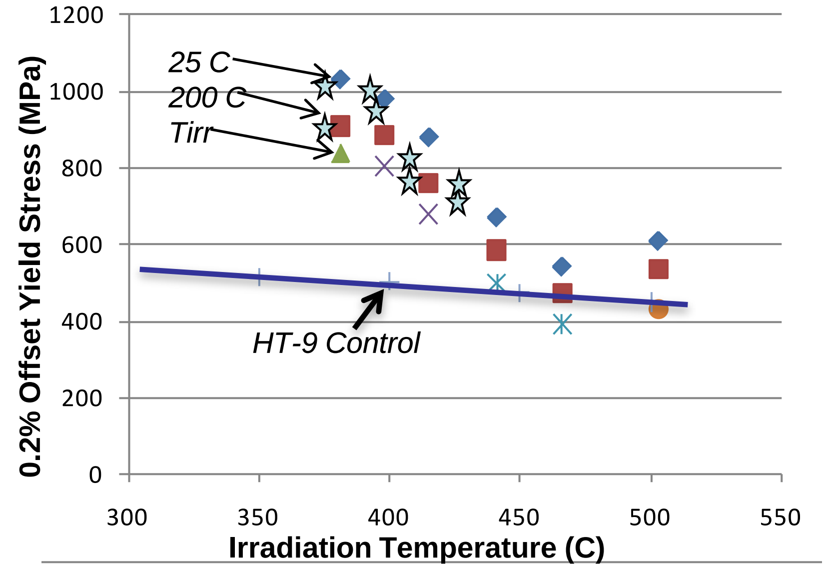


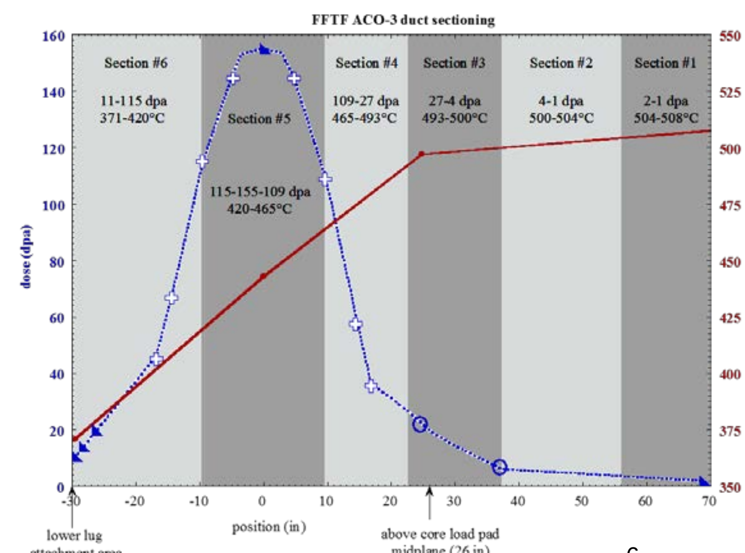
Objectives

- ***Qualify HT-9 to Radiation Doses >250 dpa***
  - *Test previously irradiated materials (ACO3 duct and FFTF/MOTA specimens)*
  - *Measure data for model development – rate jump testing*
  - *Extend irradiation data to higher doses – Re-irradiation of specimens in BOR-60*
- **Develop Advanced Radiation Tolerant Materials**
  - High dose irradiation testing
  - High dose ion irradiation testing
  - Scale up ODS processing (15 kg milling runs complete)
  - Tube production and weld development
- **Develop Coatings and liners to prevent FCCI**
  - Diffusion couple test
  - TiN coating on tube

Analysis of Specimens from ACO-3 Duct



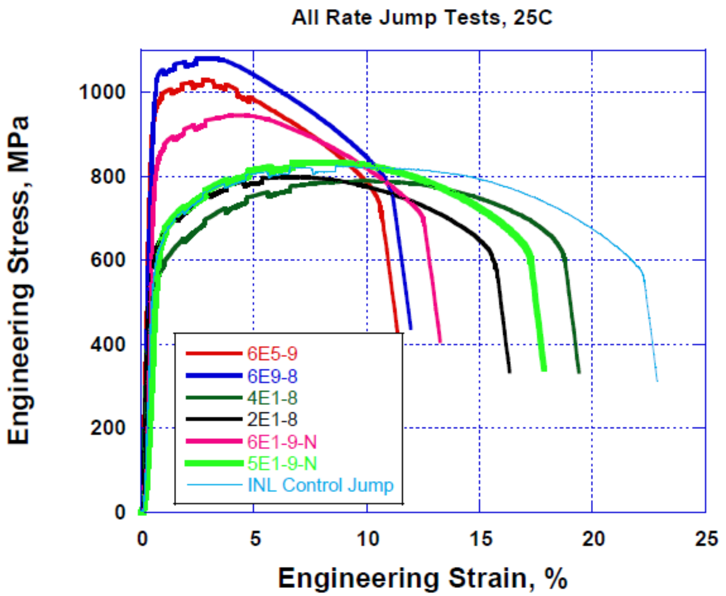




- **Total specimens= 144 Charpy, 57 compact tension, 126 tensile specimens, 500 TEM**
- **Charpy and Compact Tension specimens completed testing at ORNL. Thermal annealing testing completed.**
- **Completed tensile testing from 6 different locations along the duct at 25, 200 and the irradiation temperature.**
- **Completed Rate Jump Testing at 25° C**
- **Detailed microstructural analysis performed.**
- **New specimens EDM machined for BOR-60 Irradiation**

Tensile Testing Completed on High Dose Irradiated F/M Steels

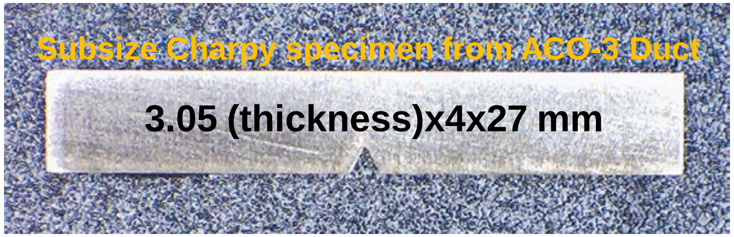
Diagram showing specimen cut plan from ACO-3 duct for re-irradiation in BOR-60



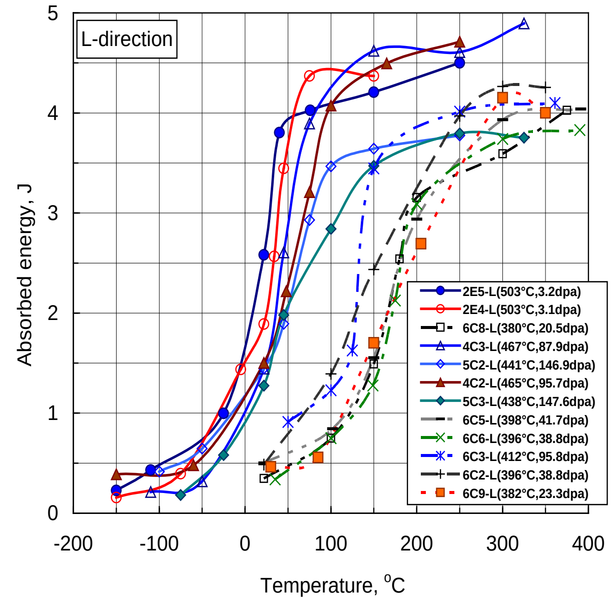
Stress-strain curves measured on irradiated HT-9 while performing rate jump testing

- **LANL recently shipped samples to Russia for re-irradiation in BOR-60 through CRADA with Terrapower.**
- **LANL completed rate jump tests on specimens from the ACO-3 duct. Data is being coordinated with model development.**

Study on Annealing Recovery of Fracture Toughness in ACO-3 Duct HT9 by Specimen Reusing Technique

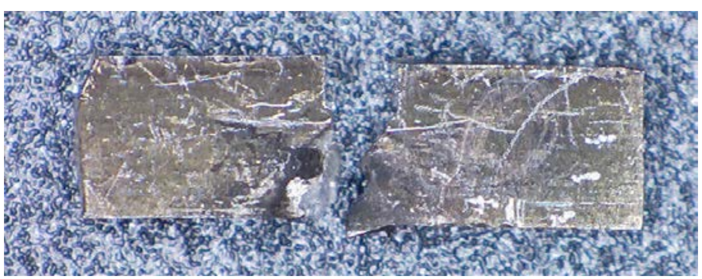


Charpy impact test data



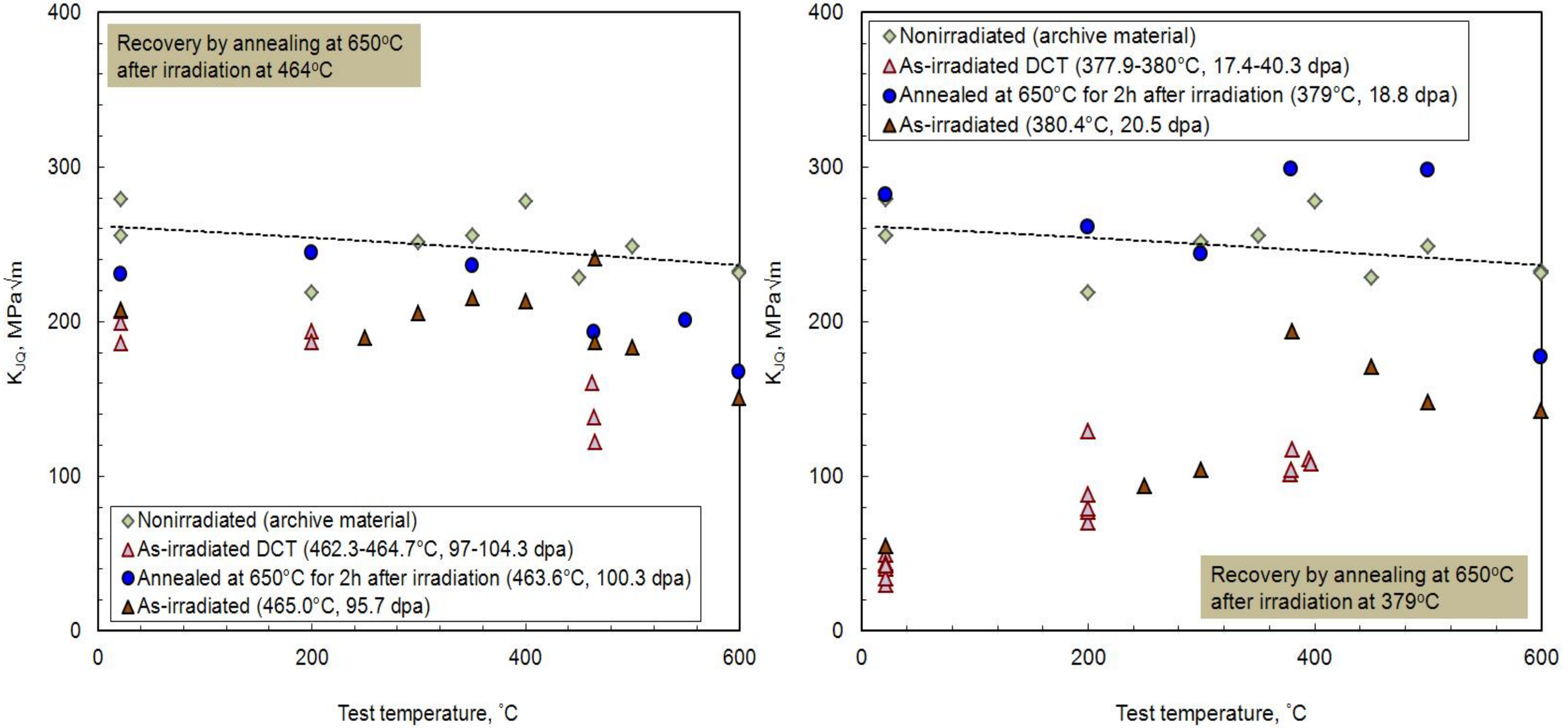
These Charpy specimen halves were reused for J-R tests: annealed, notched, precracked, and fracture (J-R)-tested.

An annealed half



Pieces after fracture(J-R) test

Recovery of K_{JQ} by 650°C Annealing after High Dose Irradiation



- **Complete or near-complete recovery of fracture toughness was observed after 650° C annealing. The toughness recovery was particularly strong after low temperature (~380° C) irradiation.**
- **After 650° C annealing a decrease of K_{JQ} occurred when the test temperature > 500° C although the lowest fracture toughness measured at 600° C was still higher than 170 MPa√m.**

❖ ***Thermal annealing treatment can be a mitigation tool against the radiation-induced embrittlement in HT9 steel core.***

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  - Diffusion couple test
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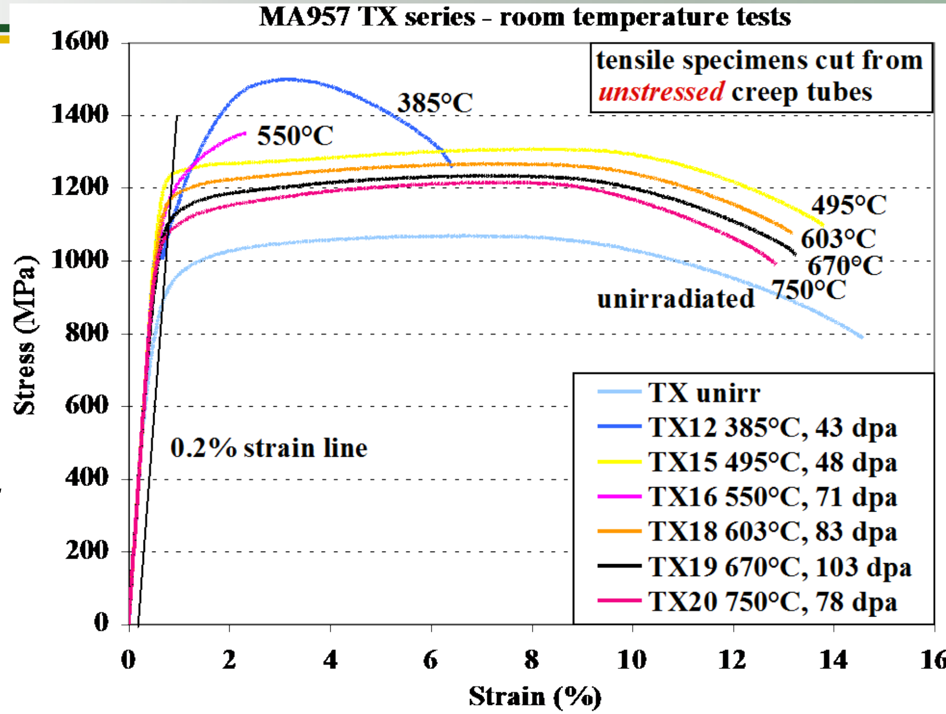
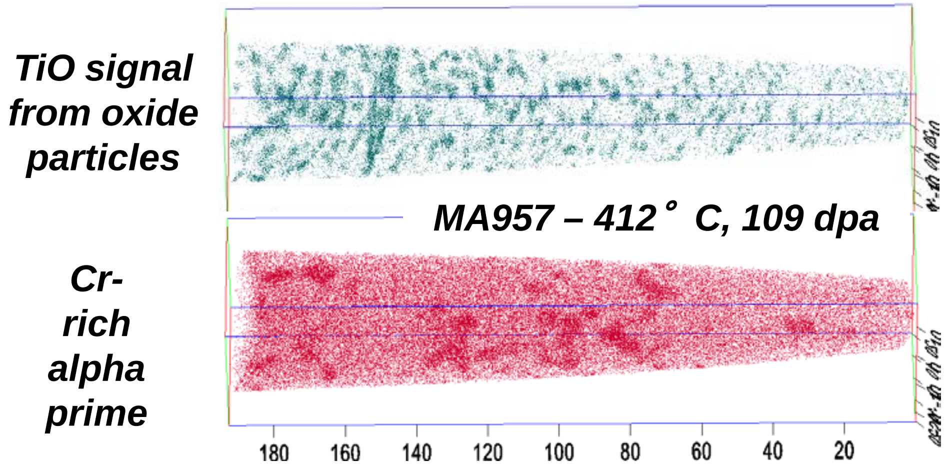
Analysis of High Dose Neutron Irradiated MA957 Tubing Underway at PNNL

- ***Irradiation conditions***
  - *(385° C, 18-43 dpa)*
  - *(412° C, 110 dpa)*
  - *(500-550° C, 18-113 dpa)*
  - *(600-670° C, 34-110 dpa)*
  - *(750° C, 33-120 dpa)*
- ***Status***
  - *First set of room temp tensile tests complete.*
  - *Initial microstructural exams using APT complete.*
  - *In-reactor creep and swelling response analyzed.*
  - *500 dpa ion irradiations complete with measured swelling.*
- ***Current post-irradiation results:***
  - *Tensile: No loss in ductility at all but lowest irradiation temperature exhibits higher strength.*
  - *In-Reactor Creep: Comparable to HT-9 in creep resistance to up to 550° C, much better resistance at 600° C and higher.*
  - *Swelling: No swelling after 110 dpa neutrons. Slight swelling after 100 dpa ions, 4.5% max swelling after 500 dpa ions:*
  - *Microstructure from APT: See next slide.*



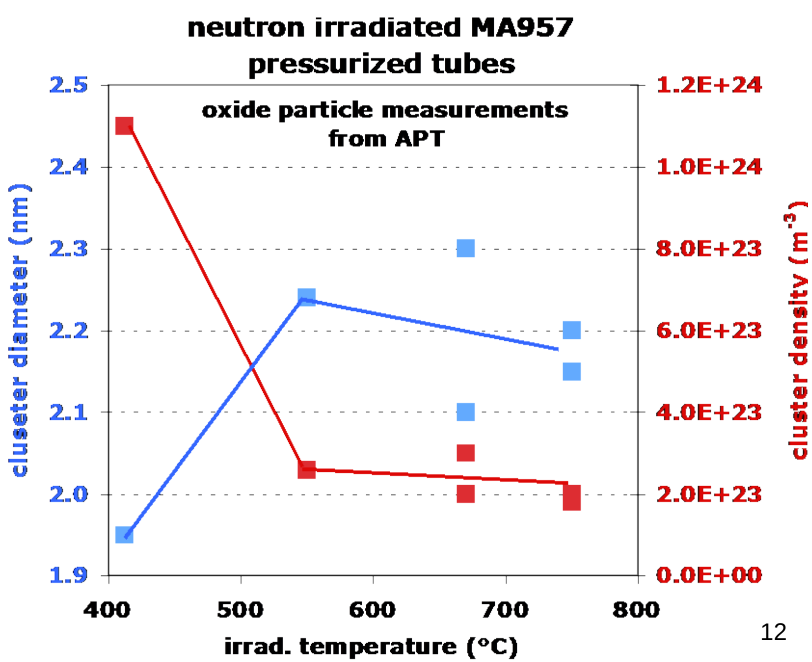
High Dose MA957 ODS Ferritic Alloy APT Exams at UC Berkeley

- Objective – Study microstructure of neutron irradiated ODS ferritic steel, with emphasis on oxide particle morphology.
- Material - MA957 from in-reactor pressurized tube creep specimens.
- Preliminary APT examinations completed on specimens irradiated at 412, 550, 670, and 750° C to 109-121 dpa.



Initial Results

- MA957 pressurized tubes have a small oxide particle size of ~2 nm similar to newer ODS steels such as 14YWT.
- No obvious ballistic dissolution at these irradiation temperatures, but small difference in oxide particle population at 412° C.
- Cr-rich alpha-prime clusters observed at 412° C irradiation temperature consistent with 14Cr composition.
- Some grain boundary segregation.

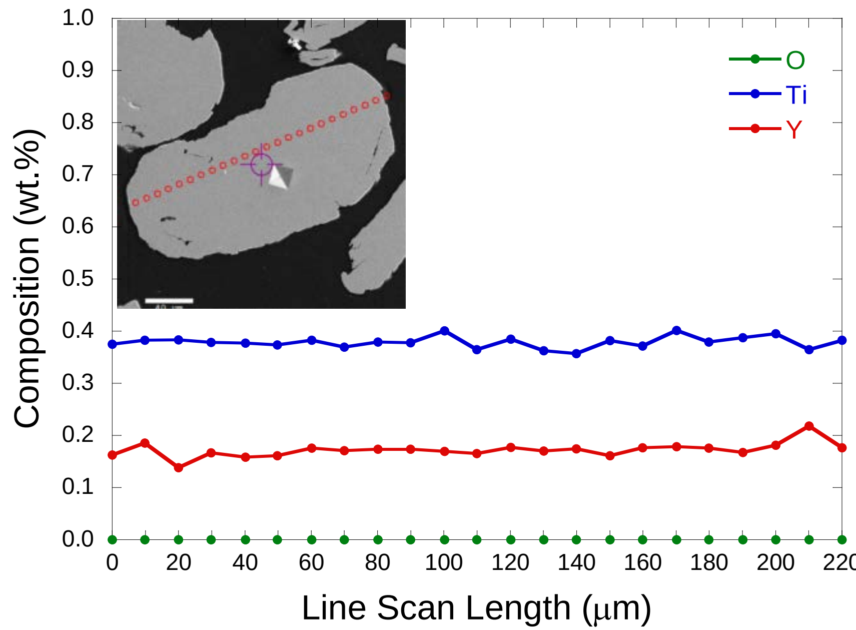


Scale Up Production of 14YWT Ferritic Alloy (Heat FCRD-NFA1)

- 4 of 4 ball milling runs completed by Zoz
  - V540-01: 15 kg of coarse (>150 μm) powder
  - V540-02: 15 kg of medium (45-150 μm) and fine (<45 μm) powder
  - V540-03: 15 kg medium, fine and small amount of V540-01 coarse powder
  - V540-04: 15kg medium, fine powder mixed with yttria for the oxide dispersion.

V540-02 Ball Milled 40 h	
MET. SPECIMEN NO: 12-0581	
LOAD in grams: 200	
Indent no.	HV
1	723.57
2	744.47
3	726.78
4	713.14
5	700.18
6	768.03
AVERAGE =	729.36
STD =	23.99

- EPMA showed 40 h ball milling distributed Y uniformly in fine and medium powders
- 40 h ball milling did not distribute Y uniformly in coarse powders
- Mechanical testing underway.



High Toughness ODS Ferritic Alloy Development in FC R&D (I-NERI)

- ***Development and Characterization of Nanoparticle Strengthened Dual Phase Alloys for High Temperature Nuclear Reactor Applications***
- **To develop high toughness NFAs* for high temperature (700ºC) high dose (>300 dpa) applications: 100 MPa√m over the range of RT - 700°C.**
- **Use grain boundary strengthening/modification techniques.**
- ***ORNL (TS Byun & D.T. Hoelzer) – KAERI (JH Yoon)***
- ***Dec. 1, 2010 – Nov. 30, 2013***

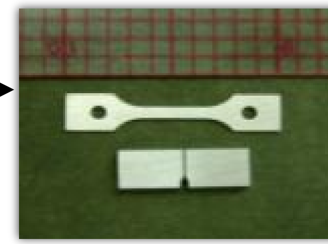
***** *Nanostructured Ferritic Alloys (NFAs) vs. Oxide Dispersion Strengthened (ODS) Alloys*

Production of Base Materials (9YWTV)

Two alloy power heats (8 kg each) have been produced by gas atomization process at ATI Powder Metals:
Fe-9Cr-2W-0.4Ti-0.2V-0.12C+0.3Y_2O_3 &
Fe-9Cr-2W-0.4Ti-0.2V-0.05C+0.3Y_2O_3

Ball milling for 40 hours in Zoz CM08 machine (6 loads)/ Canned & degassed (6 cans, 920g each)

Extruded below 850 °C
Cut into 4 inch long blocks



Characterization

<u>Goals of Yrs 2 & 3:</u>
- ***Post-Extrusion TMT Optimization***
- ***Micro & High Temp. Characterization***
- ***Feedbacks for new processing***

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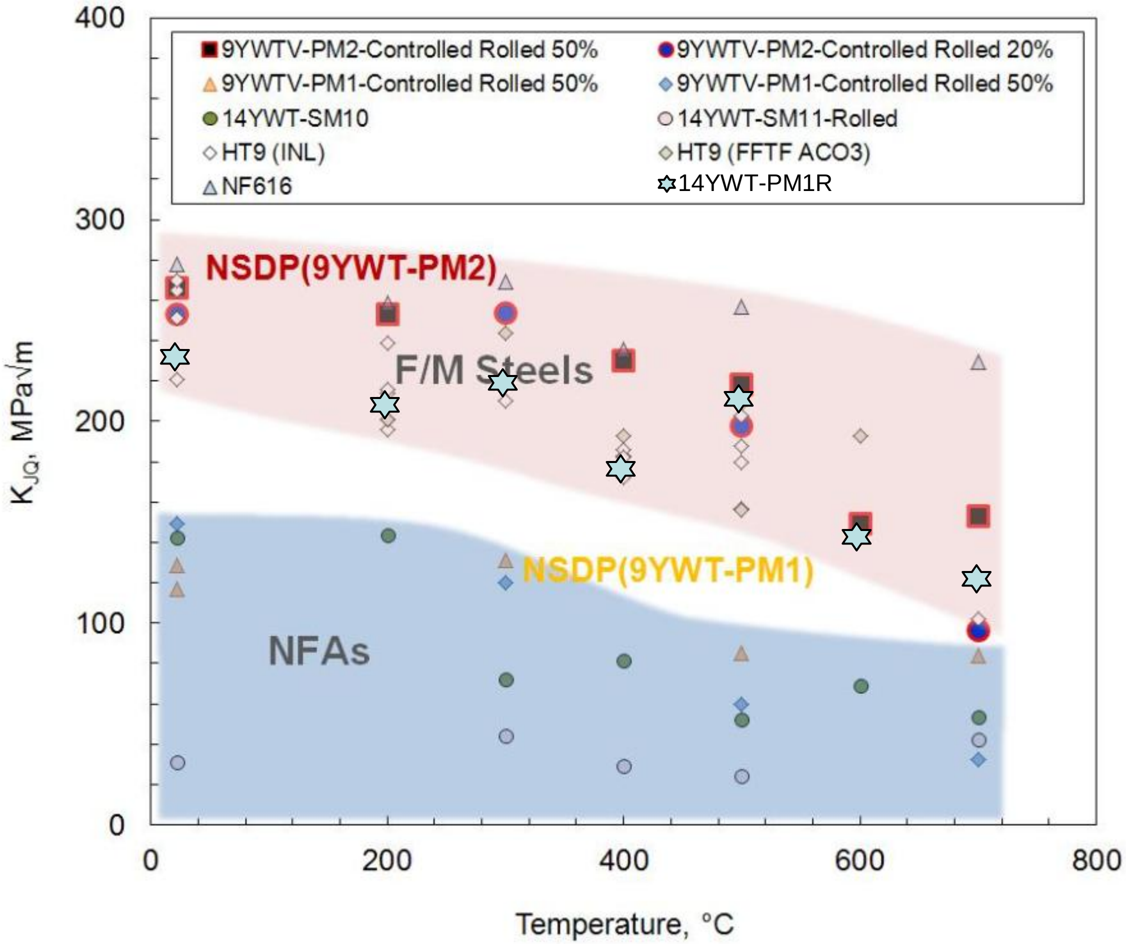
Preliminary Results for Fracture Toughness



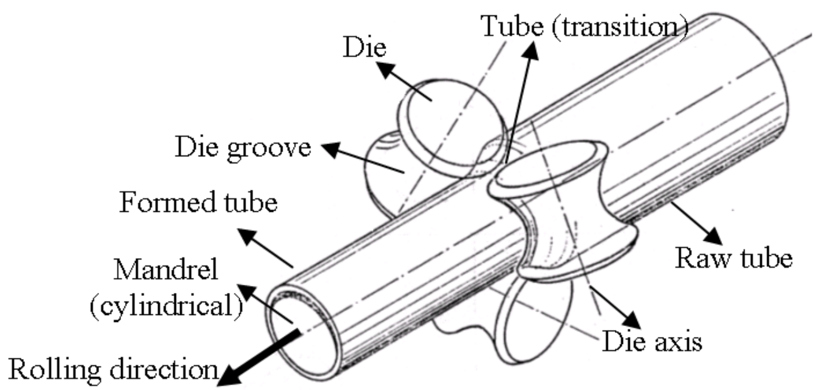
- o 9YWTV-PM2-850C-200m: Annealed at 850°C for 200 minutes.
- o 9YWTV-PM2-850C-20H: Annealed at 850°C for 20 hours.
- o 9YWTV-PM2-900C-50%R: Hot-rolled for multi-step 50% reduction after annealing at 900°C.

- ➤ **Fracture toughness can be significantly improved by some controlled rolling, and the K_{JQ} values are as high as those of FM steels.**
- ➤ Further development/optimization of processing is underway.

Fabrication of Cladding Tubes from ODS alloys

- 3 cans were designed and fabricated for producing thick wall tubes from the ODS 14YWT and 9Cr-ODS alloys
- Powder has been ball milled, canned and hot extruded.
- Potential collaboration with Y. de Carlan, CEA, Saclay to produce cladding tubes

High-Precision Tube Roller Pilger equipment at CEA



L. Toualbi et. al, Journal of Nuclear Materials (2012) In Press DOI:10.1016/j.jnucmat.2011.12.013

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Coating and Diffusion Couple Study for FCCI Mitigation

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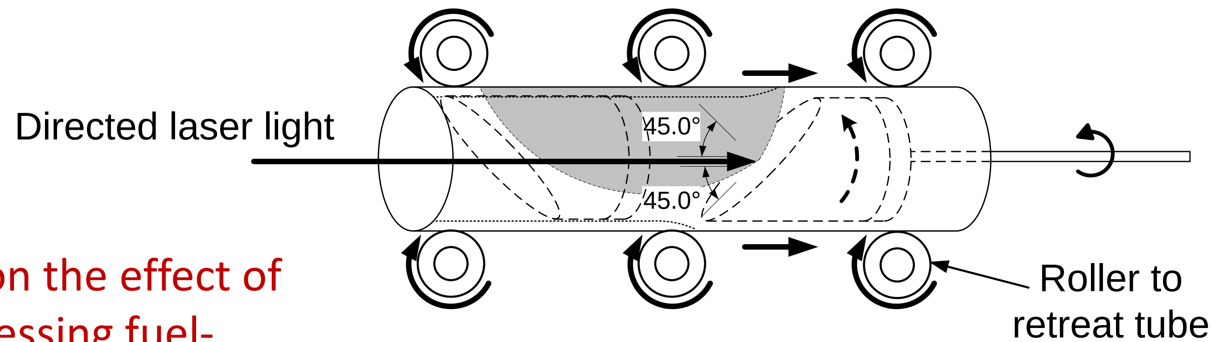
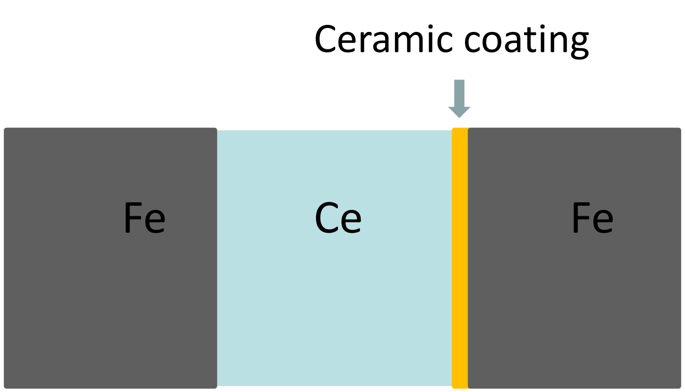
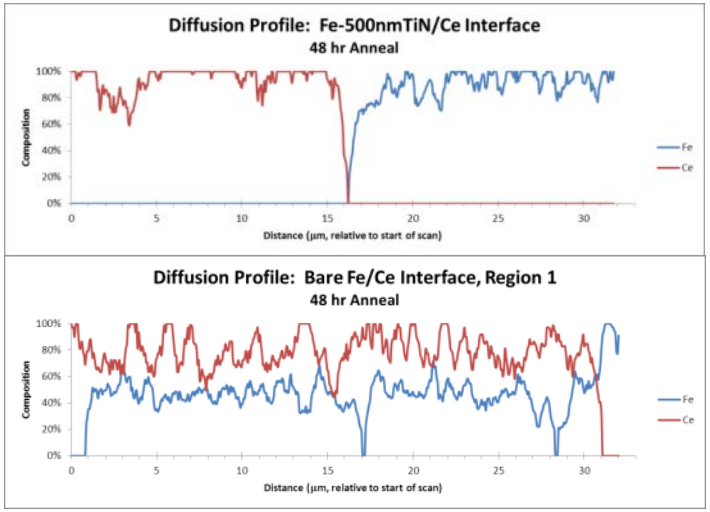
A customer-designed laser deposition system for inner wall coating of long tube is being used at Texas A&M University for the FY2012 coating work



Diffusion couple studies on the effect of ceramic coating on suppressing fuel-cladding-chemical-interaction (550 – 600 °C for 12-24 hours)

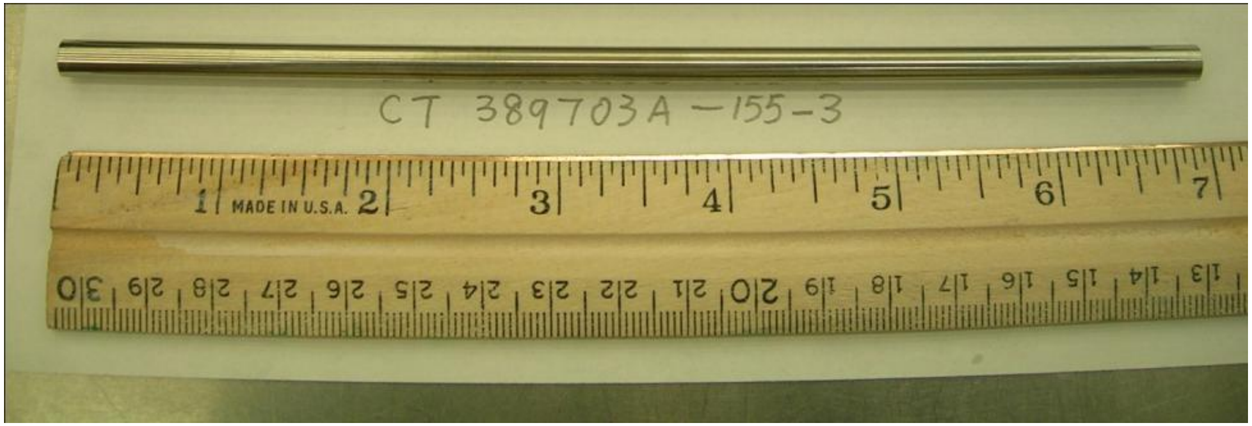




The effect of a 500 nm thin TiN coating on Fe-Ce interaction (550°C/48 hrs)

Met Level 2 Milestone to Fabricate Coated Cladding Tube for Fuels Irradiation in ATR

TiN Inner coating formed on glass tube to test TiN coating process



Successful run performed on HT-9 tube for ATR irradiation.

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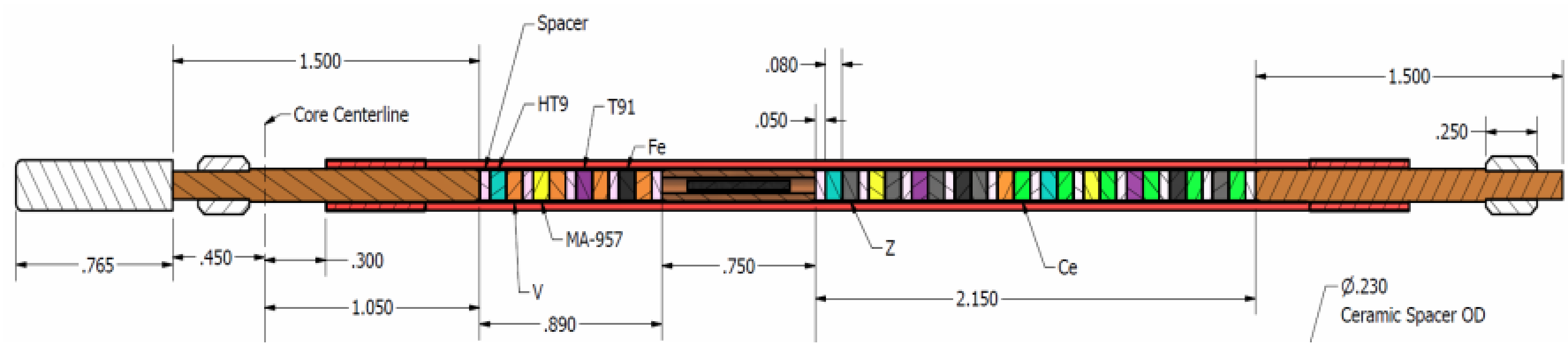
Diffusion Couple Studies on FCCI Mitigation

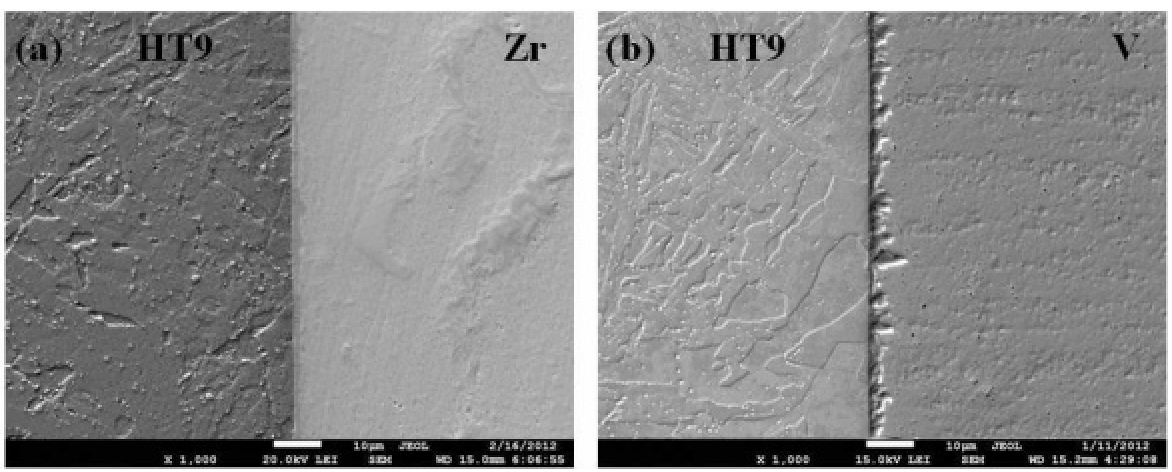
Designed a diffusion couple irradiation experiment in ATR (550 °C for 50 days) to meet the temperature and post-irradiation-examination requirements.



Interaction zone thickness: 0.5 µm
Interdiffusion zone thickness: 18 µm

Interaction zone thickness: 8 µm
Interdiffusion zone thickness: 28 µm

Diffusion couple thermal annealing studies on chemical compatibility at the cladding – liner interface (HT-9 vs. V or Zr). (704 – 815 °C for 50-200 hours)



704 °C for 200 hrs

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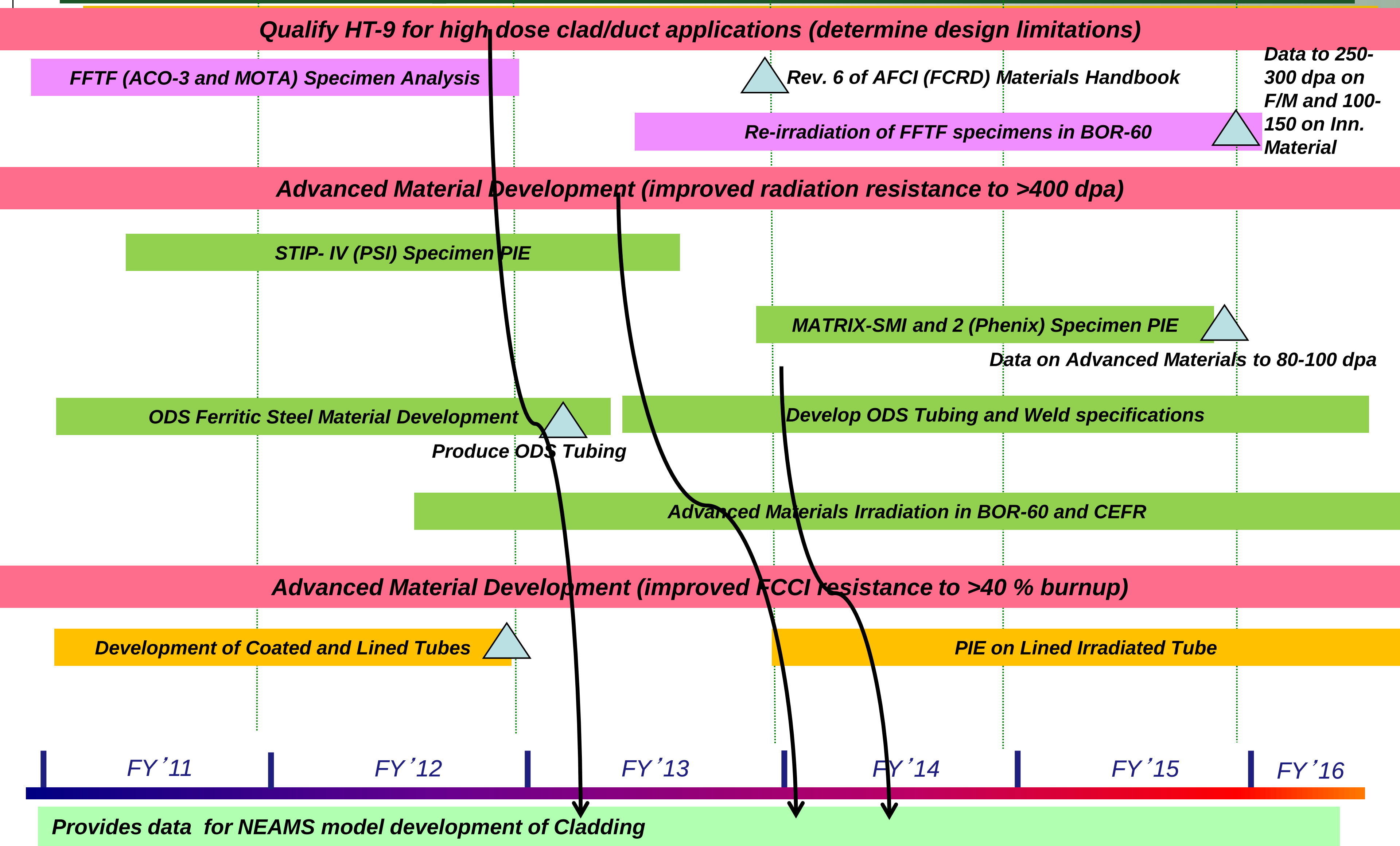
Core Materials Research and Development – 5 Year Plan



Grand Challenge for Core Materials for Next Generation LWR Fuels

Develop and test advanced alloys for Next Generation LWR Fuels with Enhanced Performance and Safety and Reduced Waste Generation

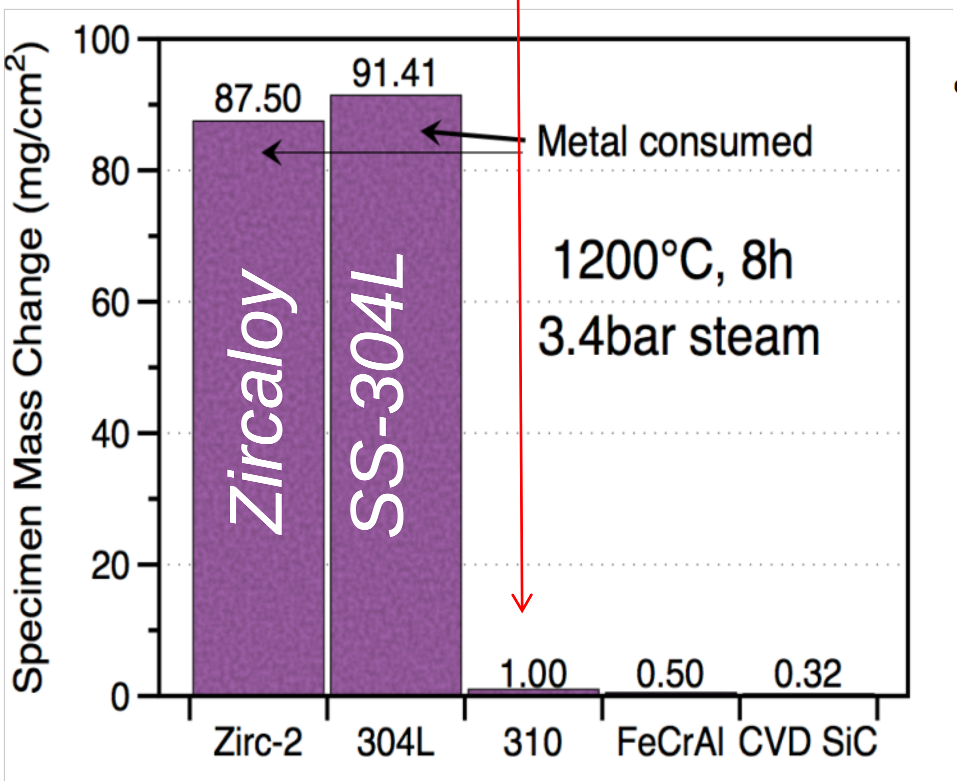
- Low Thermal Neutron Crossection
  - *Element selection (e.g. Zr, Mg)*
  - *Reduce cladding wall thickness*
- Irradiation tolerant to at least 15 dpa
  - *Resists swelling and irradiation creep*
  - *Does not accumulate damage*
  - *Stable microstructure (resists RIS)*
- Mechanically robust under loading and transportation conditions
- Compatibility with Fuel and Coolant
  - *Resists stress corrosion cracking*
  - *Resists accident conditions (e.g. high temperature steam)*
  - *Resists abnormal coolant changes (e.g. salt water)*
- Weldable and Processed into tube form
  - *Maintain hermetic seal under normal/off-normal conditions*

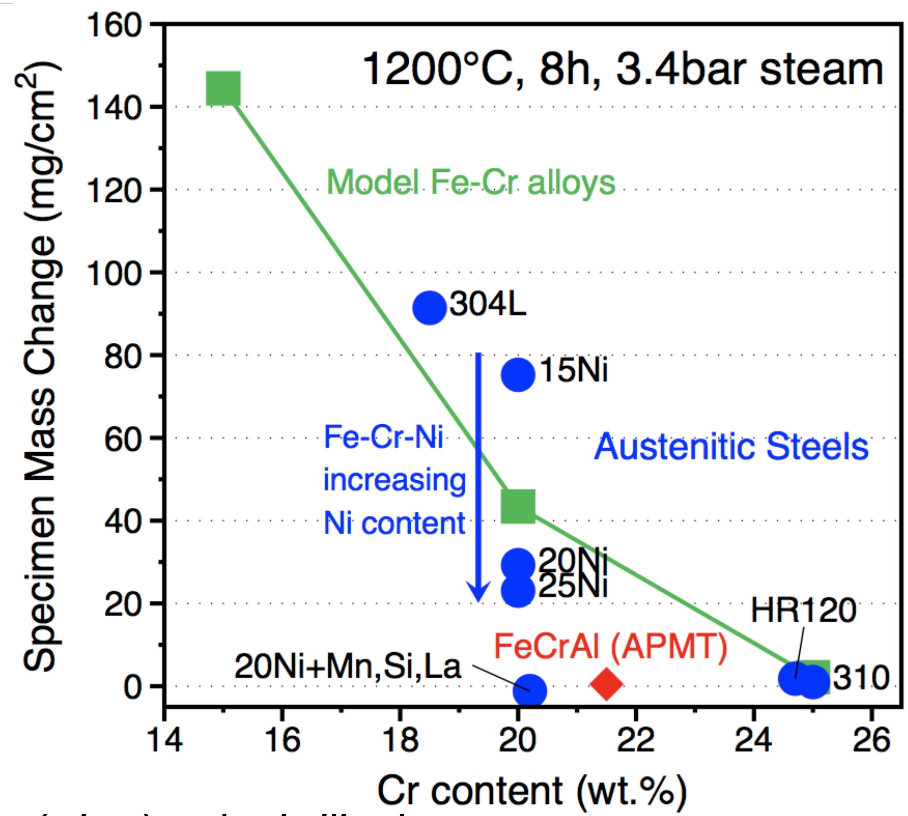
Objectives

- Measure Kinetics of Oxidation in Steam
  - Steam oxidation testing up to 1300C (ORNL)
  - Fundamental oxidation studies in steam (LANL)
- Develop Processing Techniques to produce thin-walled tubing
  - Producing tubing of MA-956 with 250 micron thick walls (LANL)
  - Measuring mechanical properties of thin walled tubes (ORNL)
  - Weld development on thin-walled tubing (INL and ORNL)
- Measure Radiation Tolerance of ATF ferritic alloys
  - Ion irradiated materials (LANL)
  - ATR irradiated materials
    - *Tensile testing (LANL)*
    - *Fracture toughness testing (ORNL)*
- Developing Improved ATF alloy (ORNL)
  - Weld development for improved ATF alloy (INL)
  - Mechanical testing and ion irradiations (LANL)
- Develop Advanced ATF alloy (LANL)
  - Production of Mo tubing using FB-CVD processing

Properly alloyed metals as protective as Si-based ceramics at 1200° C

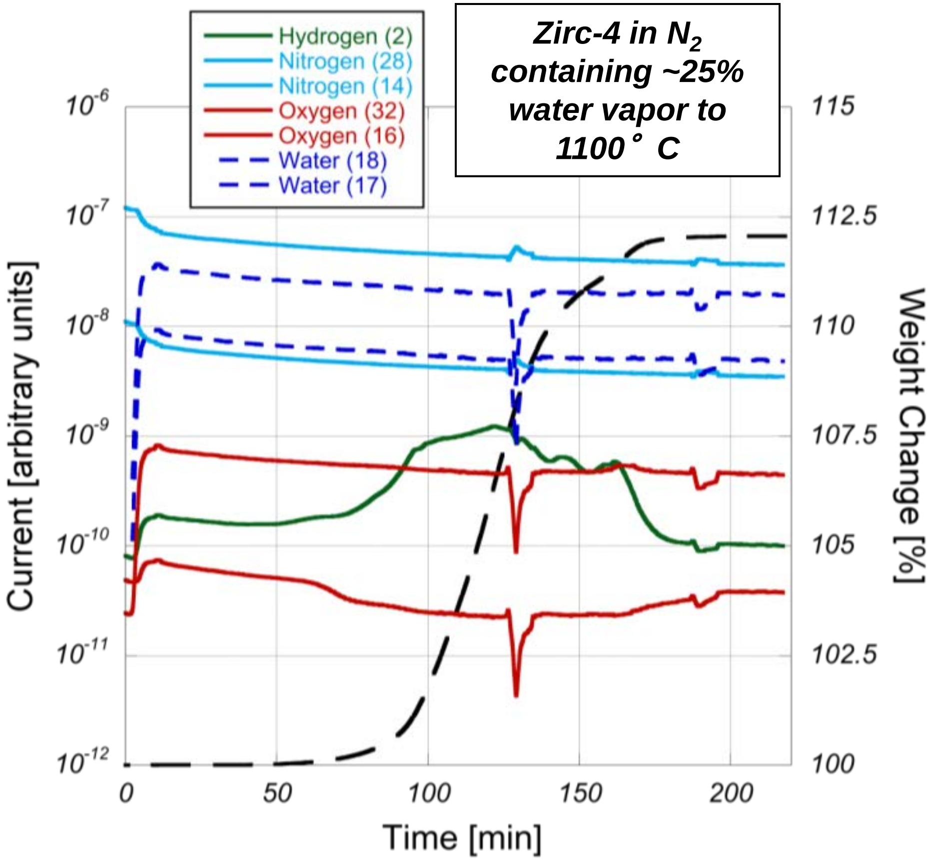
- Example from FCRD experiments at 1200° C in steam at 3.4 bar (50 psia) for 8 h
- All low mass gain: 310SS (Cr_2O_3), FeCrAl, Kanthal APMT (Al_2O_3), CVD SiC (SiO_2)

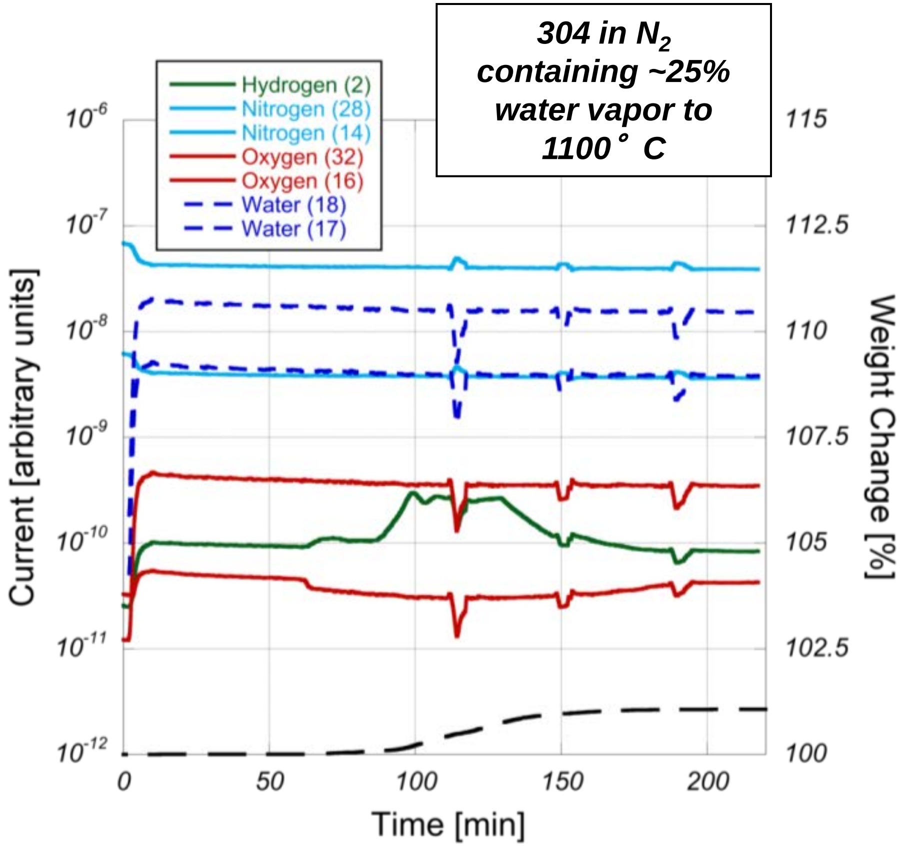




- *Mass change is net value: oxide growth, spallation (minor) and volatilization*
- *Commercial and model alloys included to fundamentally understand role of composition and minimum amount of Cr (and Al) needed for protective behavior*

Measurements on hydrogen evolution performed in steam





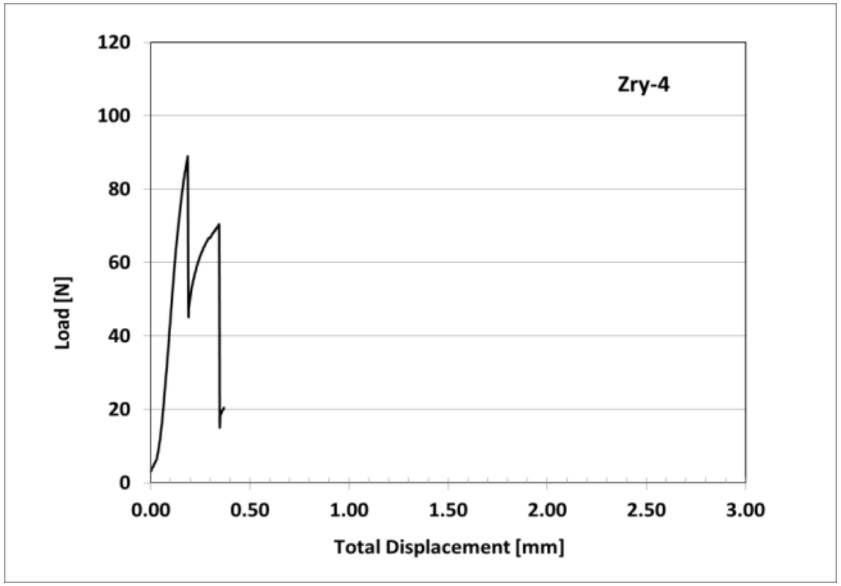
- Hydrogen Production begins in Zircaloy-4 at ~700C and in 304L at ~1000C
- Similar testing will be performed on all advanced alloys in FY13

Mechanical Property Testing

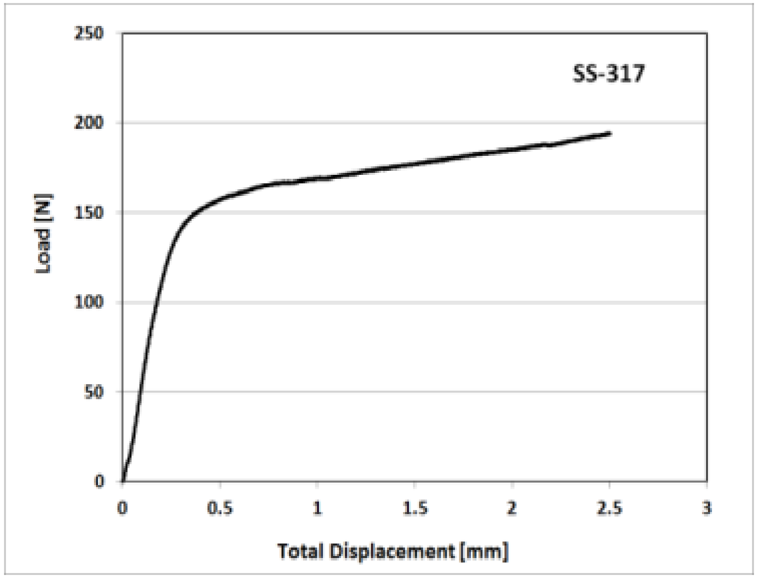
Brittle

Both samples have the same OD and wall thickness, and were oxidized at 1200℃ for ~900s

Ductile



Ring-compression Load-Displacement curves with Zry-4 oxidized at 1200℃ for CP-ECR=30%



Ring-compression Load-Displacement curves with SS-317 oxidized at 1200℃ for CP-ECR=30%

Summary of Oxidation Kinetics Screening Tests

- A series of ~35 conditions has been carried of on silica formers (i.e. SiC) chromia formers (i.e. stainless steels) and alumina former (i.e. alumina forming alloys "AFA's".)
- Results are dependent on temperature, time, pressure, and velocity and therefore the specific beyond LOCA scenario may be critical. However, a subset of attractive materials (CVD and NITE SiC, 310 stainless, and AFA's) have been identified.
- A series of papers have been submitted:

T. Cheng : "Oxidation of Fuel Cladding Materials in Steam Environments at High Temperature and Pressure." submitted Journal of Nuclear Materials.

K. Terrani et. Al. "Protection of Zirconium by Alumina- and Chromia-Forming Steels under High-Temperature Steam Exposure" to be submitted Journal of Nuclear Materials.

J. Keiser High Temperature Oxidation of Candidate Advanced Iron-Based Alloy Cladding Materials in Steam-Hydrogen Environments," *Proceedings of Nuclear Fuels and Structural Materials for the Next Generation Nuclear Reactors*, Chicago, Illinois, June 2012.

B. A. Pint, et. al. "High Temperature Oxidation of Fuel Cladding Candidate Materials in Steam-Hydrogen Environments" 8th International Symposium on High-Temperature Corrosion and Protection of Materials

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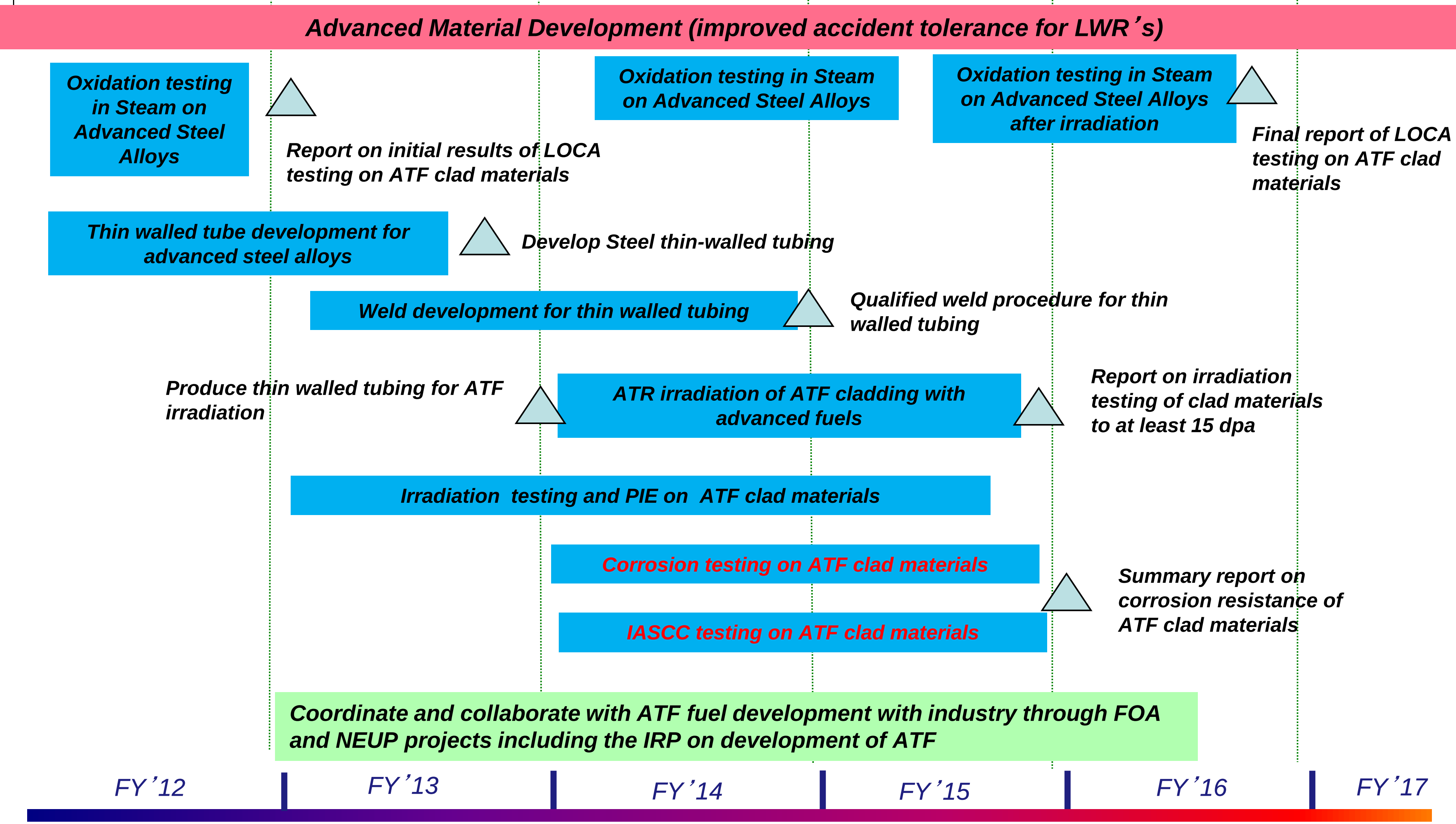
Core Materials Research and Development
ATF Clad Development - 5 Year Plan



Materials Integration and University and International Collaborations

- Integrate FCRD Core Materials Activities
  - Fuels Core Materials Work- (INL, PNNL, LANL, ORNL, LLNL)
    - *Materials teleconferences monthly*
  - University Materials Research (attend university review, review quarterly progress reports)
    - *UCSB- Optimized Compositional Design and Processing-Fabrication Paths for Larger Heats of Nanostructured Ferritic Alloys*
    - *TAMU-Bulk nanostructured austenitic stainless steels with enhanced radiation tolerance*
    - *U. Ill Urb/Champaign–Development of Austenitic ODS Strengthened Alloys for Very High Temperature Applications*
  - ATR Reactor Irradiations (provide materials and preparing to collaborate in testing)
- Working group meetings and Workshops
  - NE Materials Cross-cut Webinars in August 2012 and July-August 2013
- International Collaborations
  - INERI-GETMAT- 14Cr ODS material development
  - INERI-KAERI- 9Cr ODS material development
  - Participant in IAEA Coordinated Research Project on “Benchmarking of Structural Materials Pre-selected for Advanced Nuclear Reactors” – met in Vienna, May 2-6, 2011.
  - DOE-CIAE Collaboration – Proposed irradiation in CEFR
  - DOE-Russia – Proposed irradiation in BOR-60
  - LANL-Terrapower CRADA – proposed irradiation of ACO-3 specimens in BOR-60