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SiC_f-SiC_m Composite for BWR Channel Application

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DOE NEET WEBINAR
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Research Objectives & Industry Value

Objectives

- Evaluate the feasibility of using SiC_f-SiC_m composites as a nuclear fuel structural material
  - Thorough screening of design requirements against known properties of SiC
  - Generate data/evaluate and disposition issues identified
  - Evaluate material in-core neutronic and economic impacts
  - Evaluate different composite structure and fabrication processes

Value

- Stability of SiC in a reactor environment and resistance to high temperature steam degradation offer revolutionary performance improvements
  - Eliminates BWR channel bow and improves safety system performance
  - High temperature steam oxidation rate a fraction of existing materials
  - Coolable geometry maintained much longer in a loss of coolant accident
  - Multiple lifetime capability
  - Lower neutron capture cross-section can lead to $3 million in savings per reload
- Could be applied to PWR structures as well

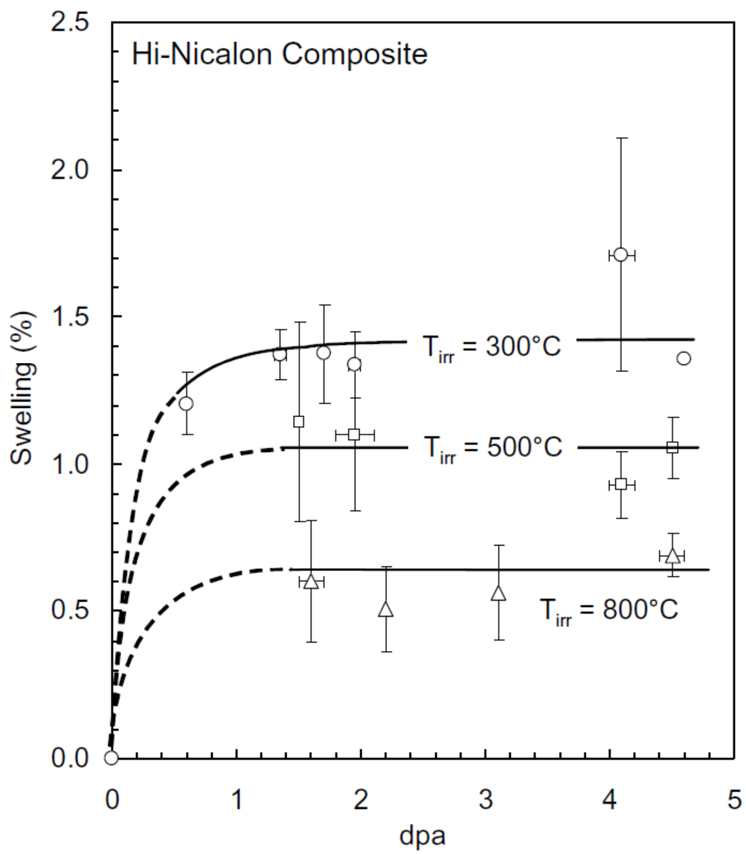
Scope

- Initial design screening (complete)
  - Fragmentation resistance and Irradiation induced swelling issues resolved
  - Eliminated polymer infiltration pyrolysis (PIP) process
- Material property characterization and issue resolution
  - Generate data to support the insertion of a commercial lead channel
  - Resolve two remaining issues (LOCA quench survivability and silica release)
- 1/3 Scale integral sample demonstration in HFIR
- Post irradiation mechanical testing
- Data evaluation and commercial demonstration decision

Initial Volumetric Swelling Concerns

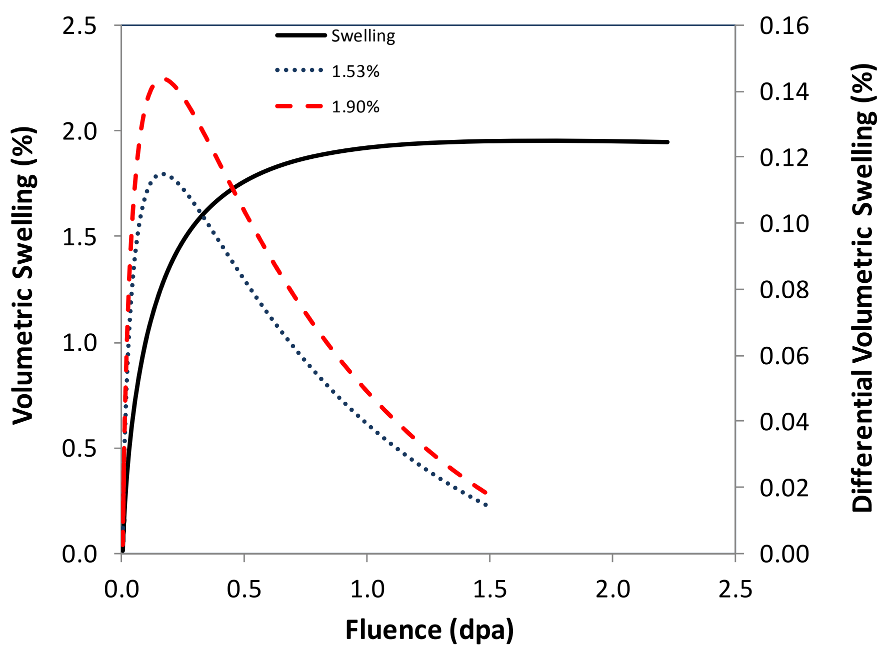
- Initial irradiation induced volumetric swelling
  – ~1.5% by volume
  – Saturates within ~6 months of operation and remain stable to very high fluence

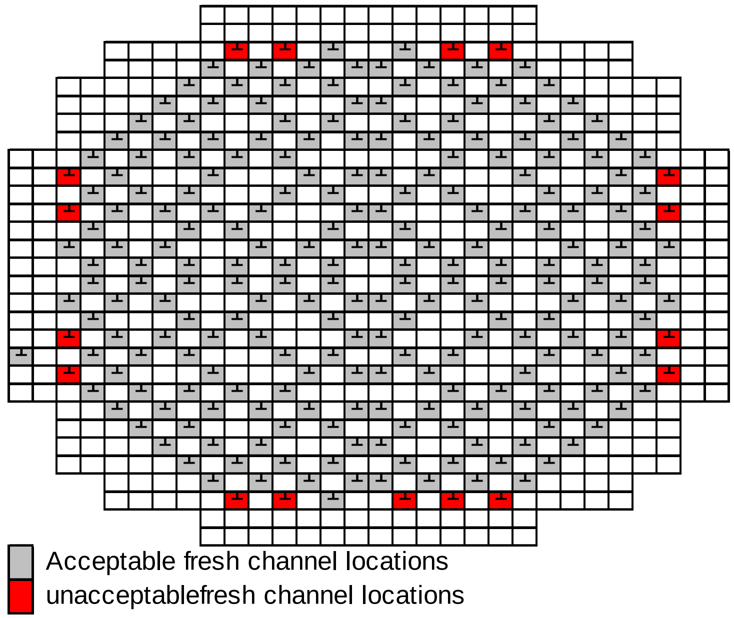
G. Newsome et al., J. of Nuc. Mat., 371 (2007) 76-90



Impact of Irradiation Induced Swelling

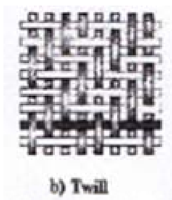
- Initial irradiation induced volumetric swelling
  - Data indicates maximum differential volumetric swelling is around 0.12%
  - Only a handful of core locations pose a challenge now and could be mitigated with core design changes (25% flux gradient shown)

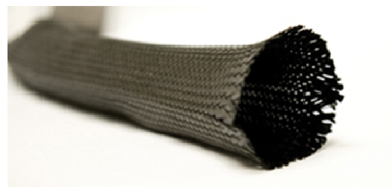




Fragmentation Resistance Evaluation Test Article

- SiC fiber reinforced SiC matrix composite fabricated for impact testing (4”x4”x24”, 0.05-0.06” wall) via CVD process





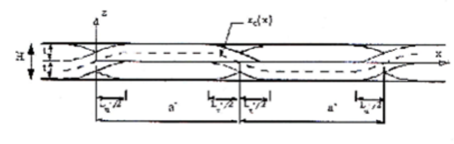


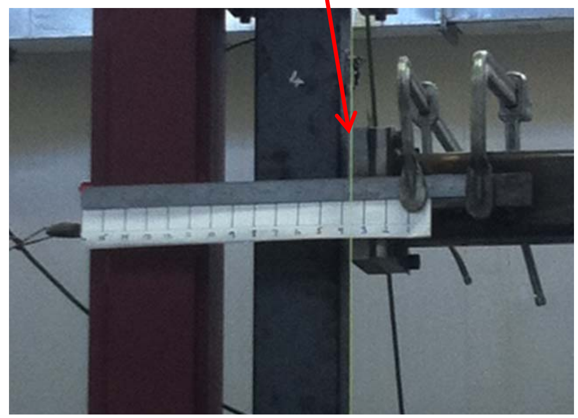
Fig. 3.2 Cross section of the RUC along the + θ_2 braider yarn

  – Hi-Nicalon Type-S fiber
  – Chemical vapor deposition process

- Joint effort with Idaho National Laboratory on fabrication via polymer infiltration & Pyrolysis process
  – Task abandoned due to poor material performance during irradiation testing

Impact Test Apparatus

Flat Plate



6 ft/minute
60 ft/minute

12 ft drop

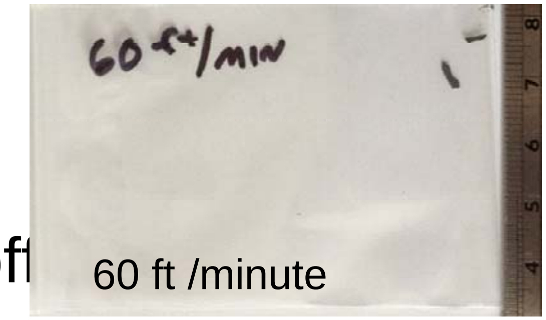
Channel Handle

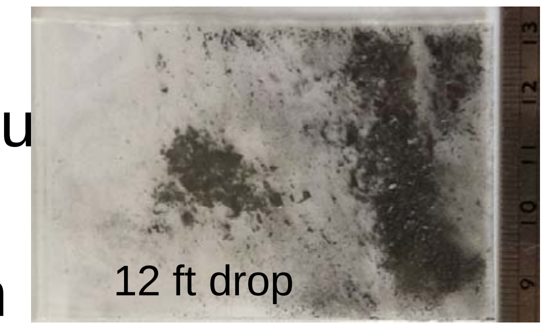
Scenarios: 6 ft/minute
60 ft/minute
12 ft drop equivalent

Impact Test Results

- 6 ft/minute
  - No visible damage
- 60 ft/minute
  - Small pieces came off
- 12 ft drop
  - Part punctured withou fragmentation
  - No crack propagaton

Debris Generated



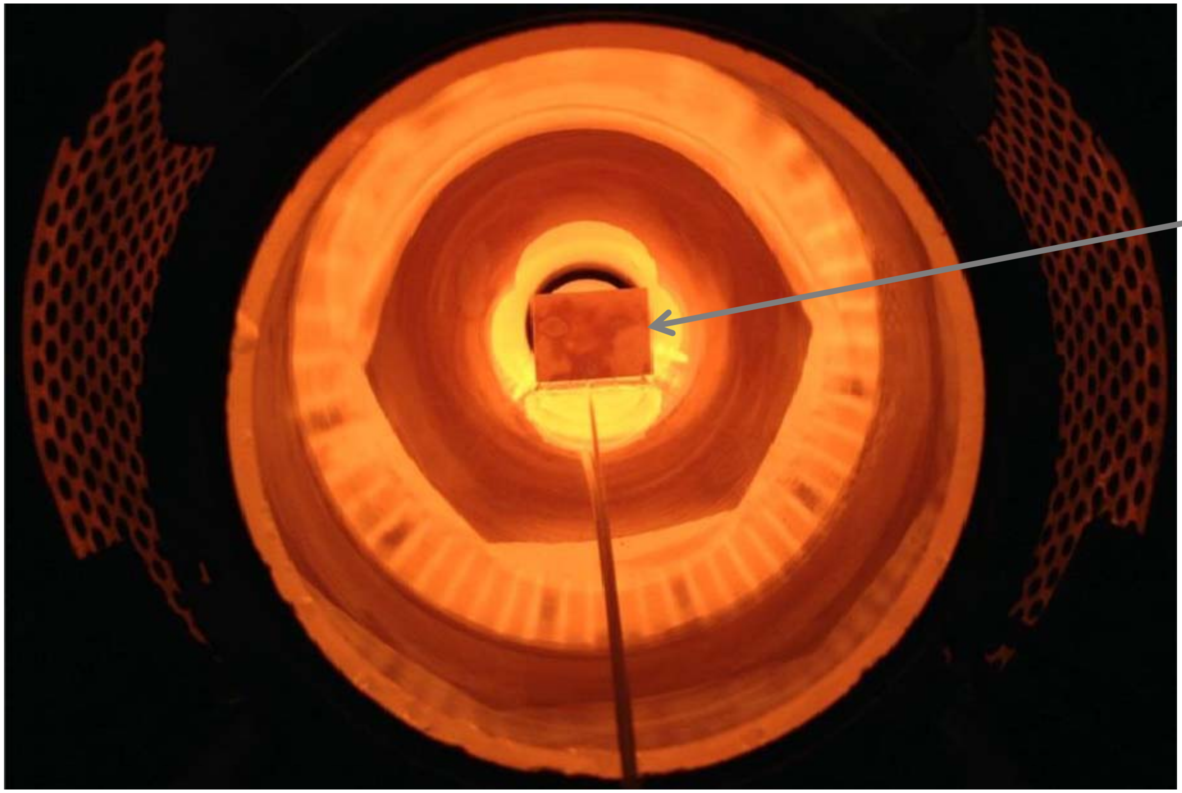


After 12 ft drop

Composite is resistant to fragmentation and crack propagation

Quench Survivability 1/2

- Test sample 4”x4”x3” heated to ~1150°C and quenched in water



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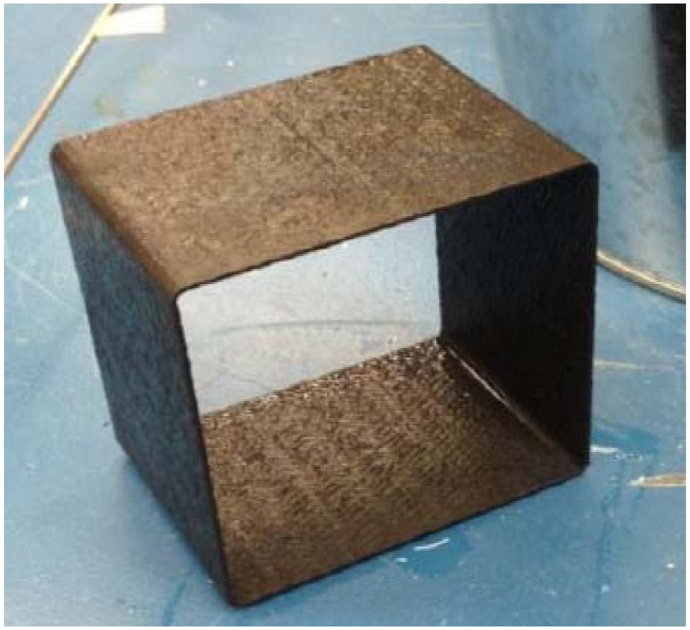
Quench Survivability 2/2

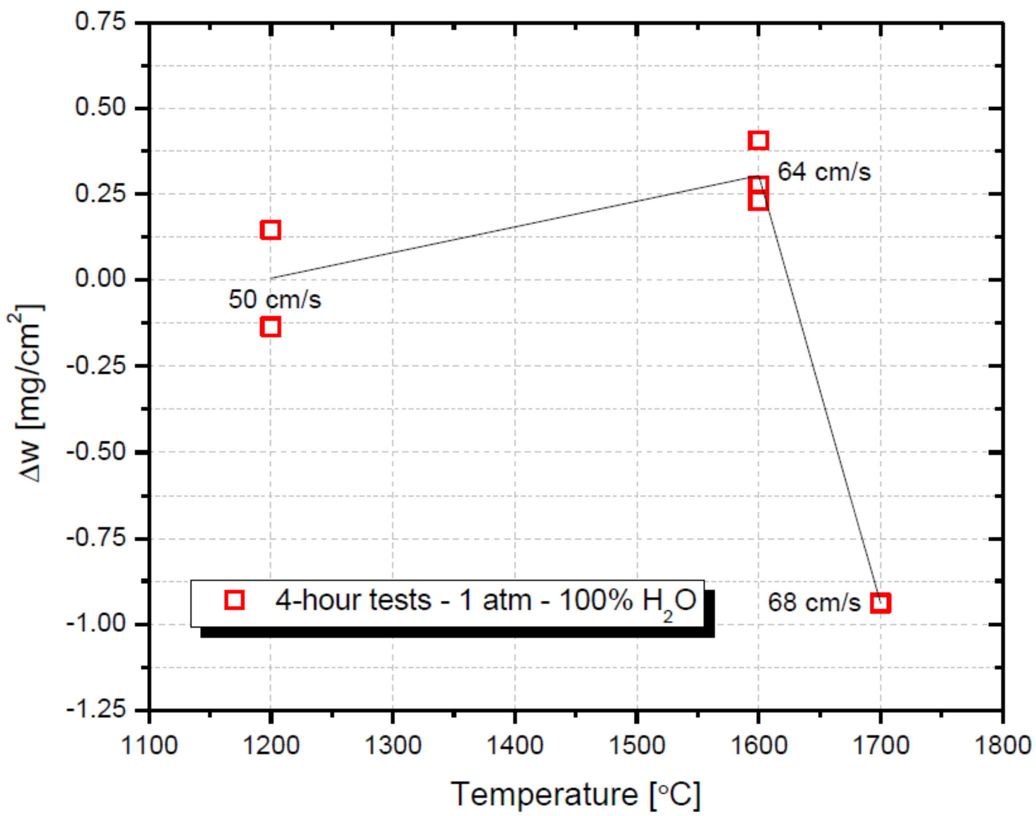
- Test article remained intact without cracking



- Different post test resonant acoustic response indicates micro-structural damage
  - Local layer separation suspected
  - Component maintains load bearing capability

High Temperature Steam Oxidation

- Negligible material reaction ,results consistent with literature data



MIT Irradiation Test

- Evaluate corrosion and creep
  - Obtain irradiation induced swelling data from corrosion coupons, complements ORNL dataset
- Test details
  - Pressurized water loop with BWR-like water chemistry
  - Start with low ppb level coolant oxygen concentration, then increase to ppm level middle cycle
  - Silica content in coolant monitored via Si^{31} gamma emission, coolant sample archived for verification
- Reactor started July 29th, 2013

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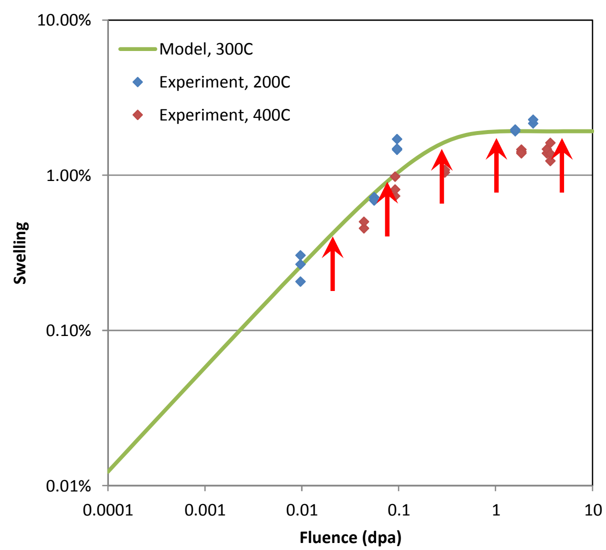
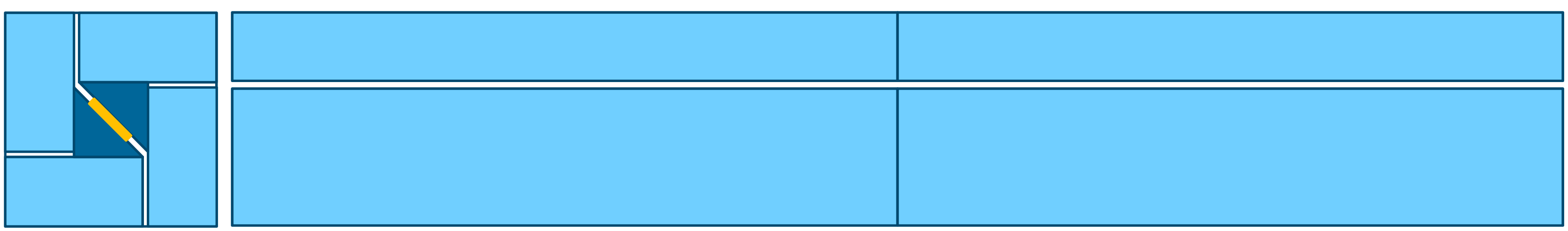
HFIR Irradiation

- Generate irradiation swelling data at several fluence and temperatures

Temp (°C)	Damage (Displacement per atom)				
	0.03	0.1	0.4	1	6
260	X	X	X	X	X
280		X	X		





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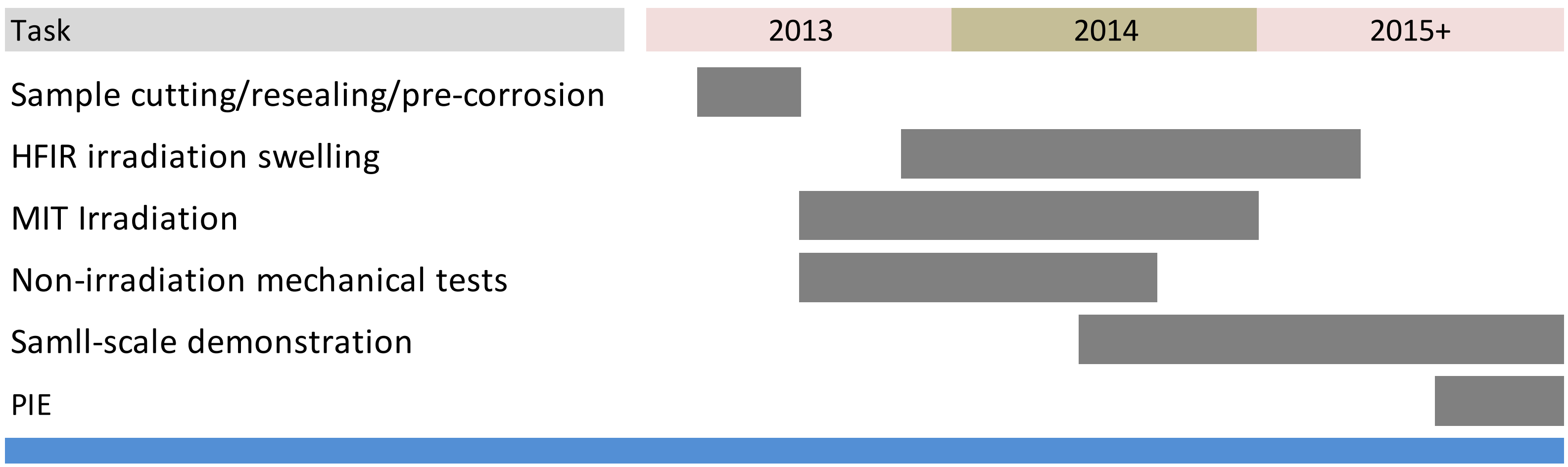
Project Schedule

- Irradiation at ORNL to start around January-March 2014
- Decision to proceed with 1/3 scale demonstration pending MIT corrosion test results
  - Schedule at risk if a decision can not be made early



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