ENCLOSURE 2

FLN-2008-015

Final Presentation for Meeting to Discuss Dry Storage Technology

Non-Proprietary Information

IMPORTANT NOTICE

This is a non-proprietary version of Enclosure 1 to FLN-2008-015, which has the proprietary information removed. Portions of the document that have been removed are indicated by white space with an open and closed bracket as shown here [[ ]].

GNF/GE-HITACHI Perspectives on Spent Nuclear Fuel and Dry Storage

Team:
Paul E. Cantonwine (GNF)
Andy Langston (GNF)
Sean Day (GE-HITACHI)
Jin Yan (GE-HITACHI)

Global Nuclear Fuel
A Joint Venture of GE, Toshiba, & Hitachi

Non-Proprietary Information

Outline

- BWR Overview (Cantonwine)
- Spent Fuel Performance (Cantonwine)
- Fuel Particle Barrier Technology (Cantonwine/Langston)
- Tip-Over Analysis (Langston)
- CFD Predictions of Temperature Distributions (Yan/Cantonwine)

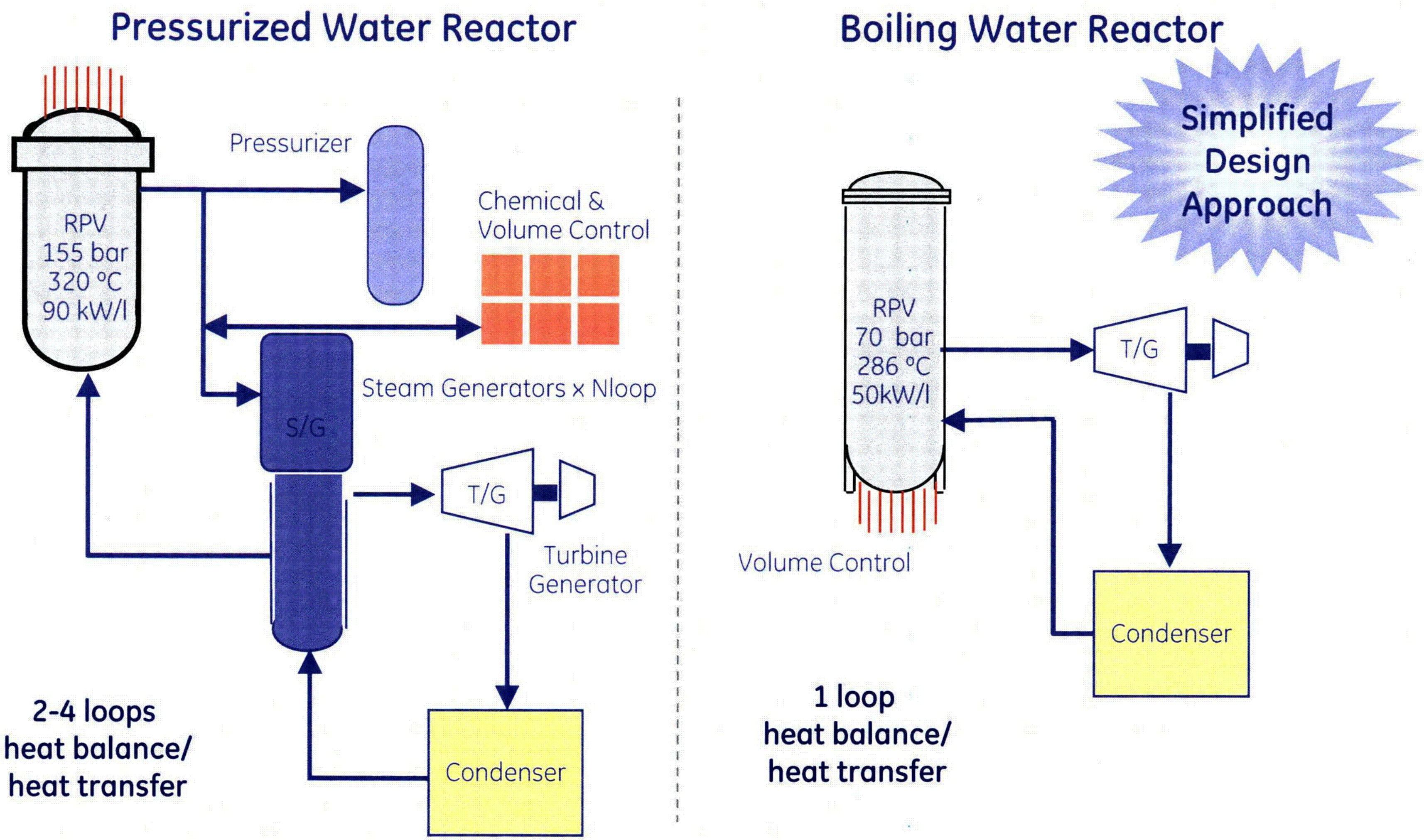
Global Nuclear Fuel

BWR Overview

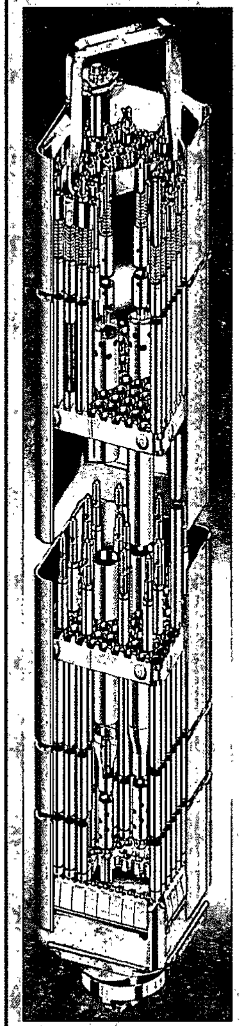
Paul Cantonwine, Lead Engineer
Materials Technology and Fuel Reliability

Non-Proprietary Information

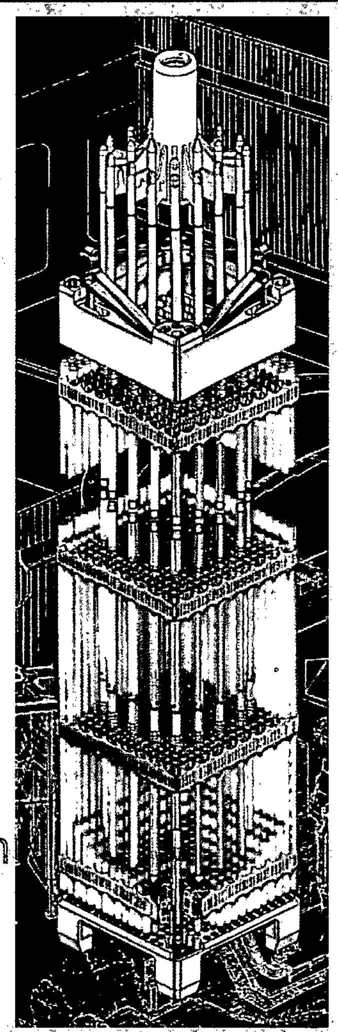
PWR vs BWR ... main differences



Fuel Characteristics

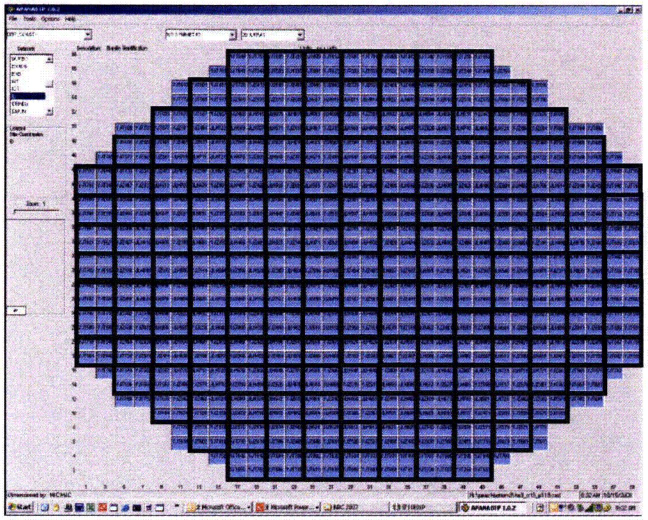


	BWR	PWR
Lattice	10x10	14x14 – 18x18
Lattice size	5.278"	~9"
Height	120"-150"	144"-168"
Fuel	UO_2/MOx	UO_2/MOx
Fuel rods	92	176-300
Part length rods	14	0
Non-fueled rods	2	20-25
Control	Ext. control rod	Int. control cluster
Cladding	Zr2 for PCI, nodular corrosion	Zr4/Zirlo/M5 for uniform corrosion & hydrogen
Channels	Yes	No
Fuel mass	180 kgU	600 kgU



1 PWR bundle ≈ 3 ½ BWR bundles

BWR Core Characteristics



BWR/2 – 308 to 560 fuel bundles

BWR/3 – 400 to 724 fuel bundles

BWR/4 – 240 to 764 fuel bundles

BWR/5 – 444 to 764 fuel bundles

BWR/6 – 624 to 800 fuel bundles

ABWR – 872 fuel bundles

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GNF Fuel Product History

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Fuel Performance History

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Crud-Induced Localized Corrosion (CILC)

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Characteristics of CILC

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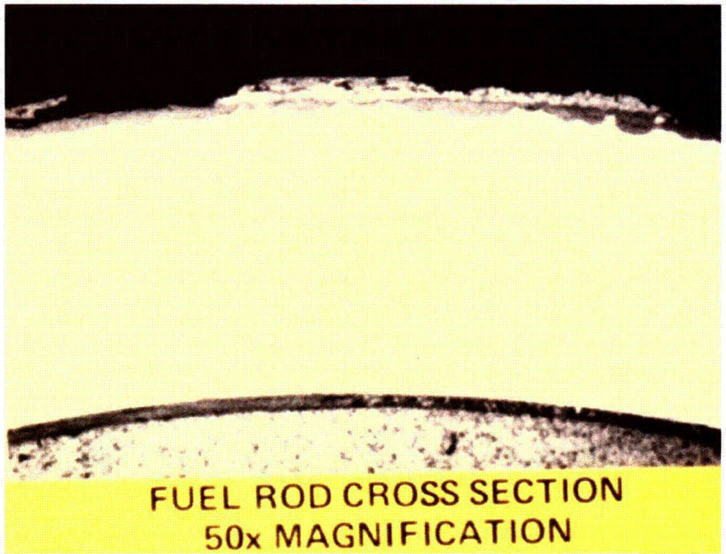

Observations correlated to poor nodular corrosion.

CILC eliminated with processing improvements that improved nodular corrosion

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Low Power Rods Preferentially Affected

Evidence that CILC Preferentially Affected Low Power Rods

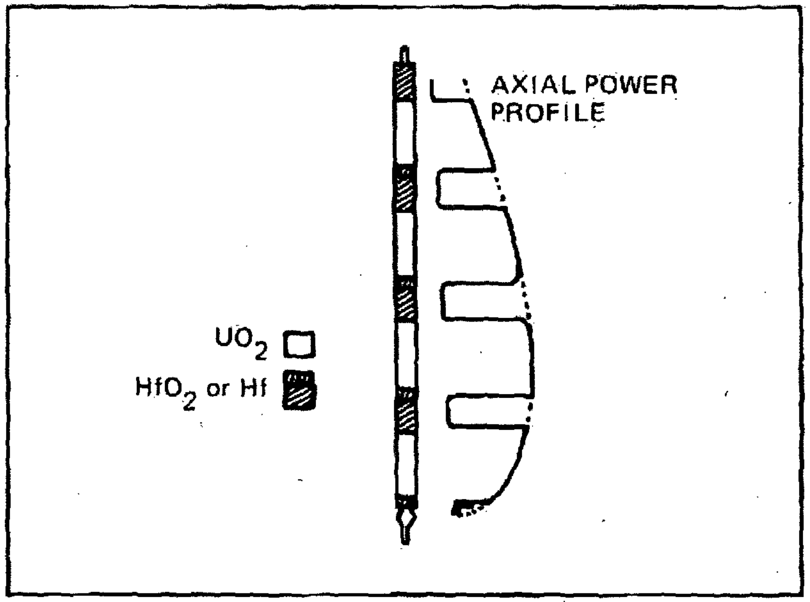

Figure 12. Schematic Diagram Showing Locations of Fuel and HF Bearing Absorber Materials

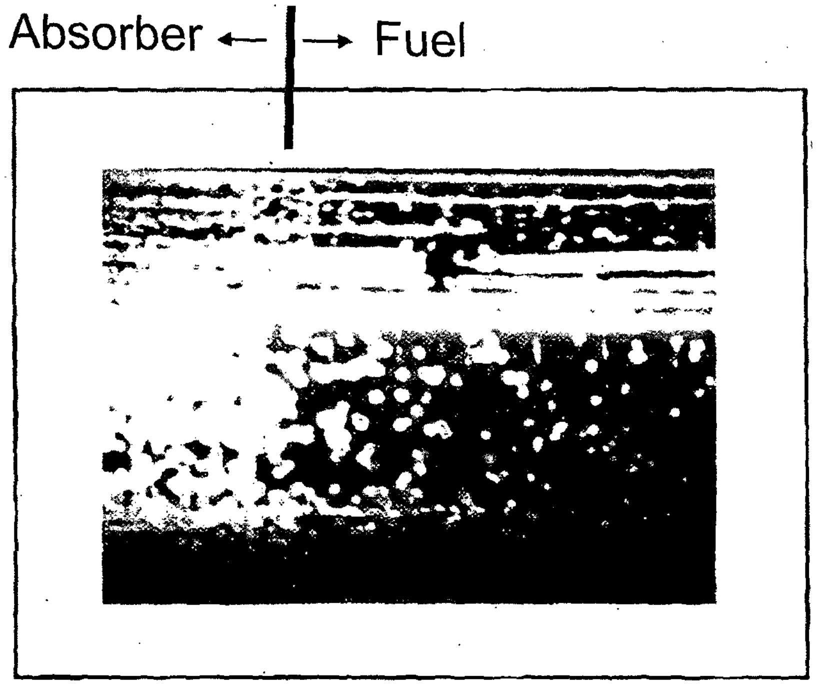

Figure 14. Transition in Nodular Oxide Coverage at Fuel/Absorber Interface

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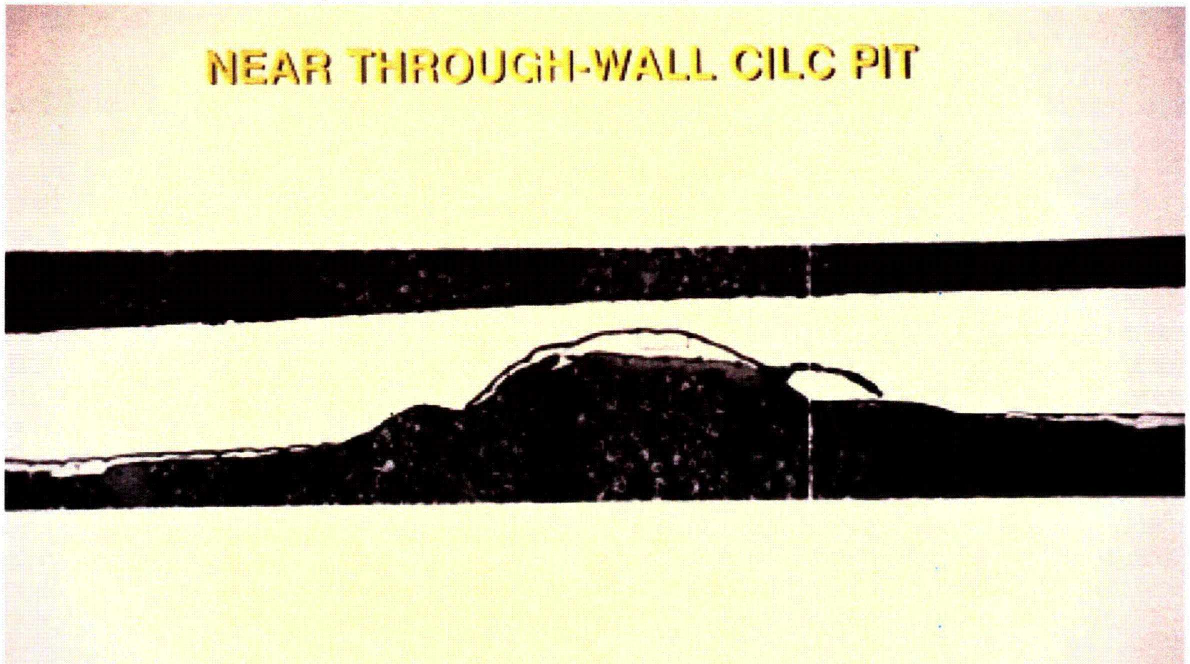

- 90% of inspected failed rods contained $(U,Gd)O_2$
- Extent of sound rods with near through-wall pits uncertain

NRC's ISG-11 Rev. 3

Cladding Considerations for the Transportation and Storage of Spent Fuel

For all burnups, for normal conditions (drying, backfilling and transfers)

- > clad temp on peak rod < 400C
- > 400C minimizes available H for reorientation to ~200ppm
- > Peak rod temp. minimizes number of susceptible rods

For low burnups, higher temp limits acceptable

- > by showing hoop stress < 90 Mpa

Limit thermal cycles to < 10 and $\Delta T < 65C$

- > Avoids re-precipitation and resolution

For off-normal and accident conditions

- > Clad temp < 570C

NRC's ISG-1 Rev. 2

Classifying the Conditions of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function

- 10 CFR 72.122:
  - (h)(1) Fuel must not degrade leading to gross rupture or it must be confined
  - (h)(5) Retrievability required without release of radioactive materials or exposures in excess of Part 20 limits

Summary

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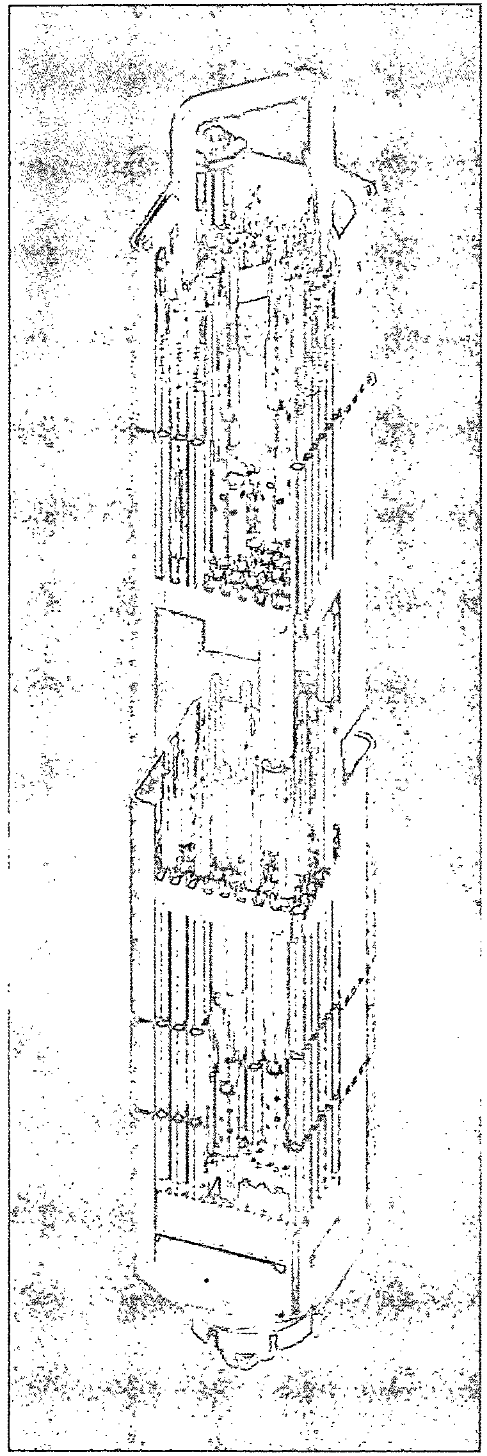
Non-Proprietary Information

Spent Fuel Performance

Paul Cantonwine, Lead Engineer

Materials Technology and Fuel Reliability

November 20, 2008



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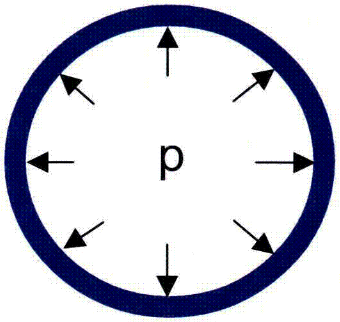
Outline

- Defining Performance limits for Spent Fuel
- Creep distortion
- Delayed Hydride Cracking
- Hydride Reorientation

Potential Stress States in Spent Fuel

Normal Operations



$\sigma =$ Biaxial Tension

$\dot{\varepsilon} =$ Creep Strain Rates

$\varepsilon \leq 0.1\%$

Industry needs well defined tests to develop performance limits

Accident Scenarios

Side Drop

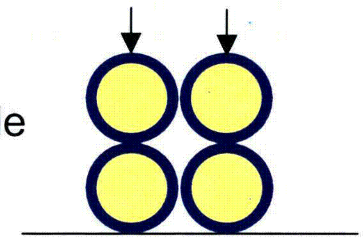
Near upper/lower tie plate

$\sigma =$ Axial Bending Stress

$\dot{\varepsilon} =$ High Strain Rates

$\varepsilon \leq ?$

Within bundle interior

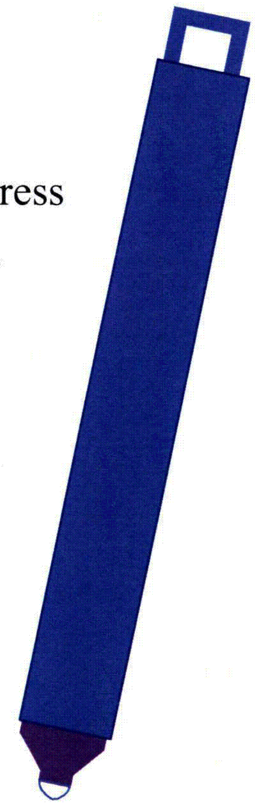


$\sigma =$ Compressive/Tensile Hoop Stress

$\dot{\varepsilon} =$ High Strain Rates

$\varepsilon \leq ?$

Corner Drop



$\sigma =$ Compressive/Tensile Axial Stress

$\dot{\varepsilon} =$ High Strain Rates

$\varepsilon \leq ?$

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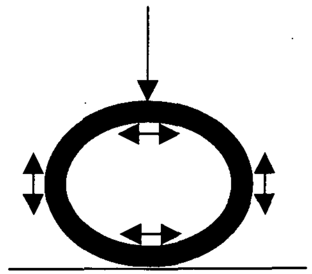
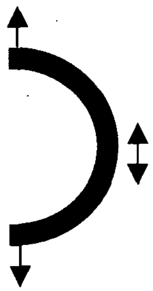
Current Tests Available

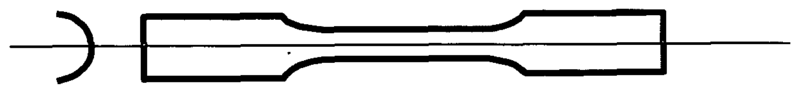
- Ring Compression Tests
  - Hoop Strength
  - Performed without fuel



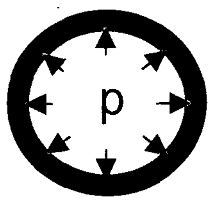
- Ring Tensile Tests
  - Hoop Strength



- Longitudinal Tests
  - Axial Strength



- Burst Tests
  - Hoop Strength



Defining Performance Limit of Spent Fuel

- Should limit be strain or stress based
  - 1% strain independent of strain rate ?
  - 1% strain at high strain rate; 5% strain at low strain rate ?
  - Ultimate tensile stress?
- Should limit be a function of hydrogen or hydride reorientation?

Creep Distortion

- ISG-11 Revision 3
  - Creep distortion is not viewed as a credible mechanism to cause gross fuel failure

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Dry Storage Temperature Profile

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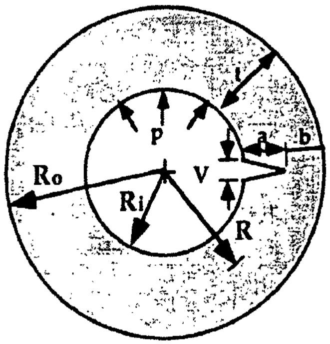
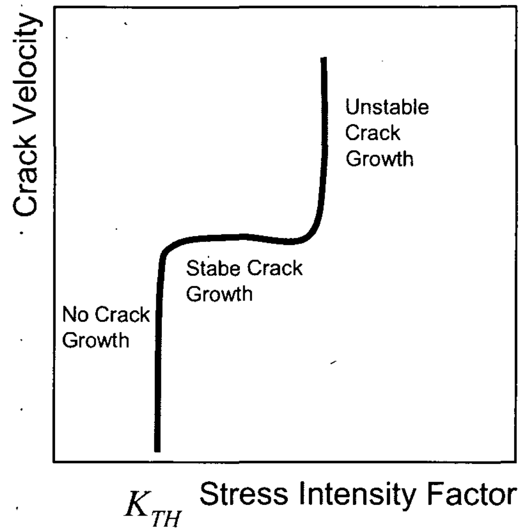
Delayed Hydride Cracking



$$K_I = \frac{2PR_o^2}{R_o^2 - R_i^2}\sqrt{\pi a}F\left(\frac{a}{t},\frac{R_i}{t}\right)$$

$$F = 1.1 + A\left[4.951\left(\frac{a}{t}\right)^2 + 1.092\left(\frac{a}{t}\right)^4\right]$$

$$A = \left(0.125\frac{R_i}{t} - 0.25\right)^{0.25}$$



a/t < 0.75

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Hydride Reoreintation

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Limit thermal cycles to < 10 and ΔT < 65C

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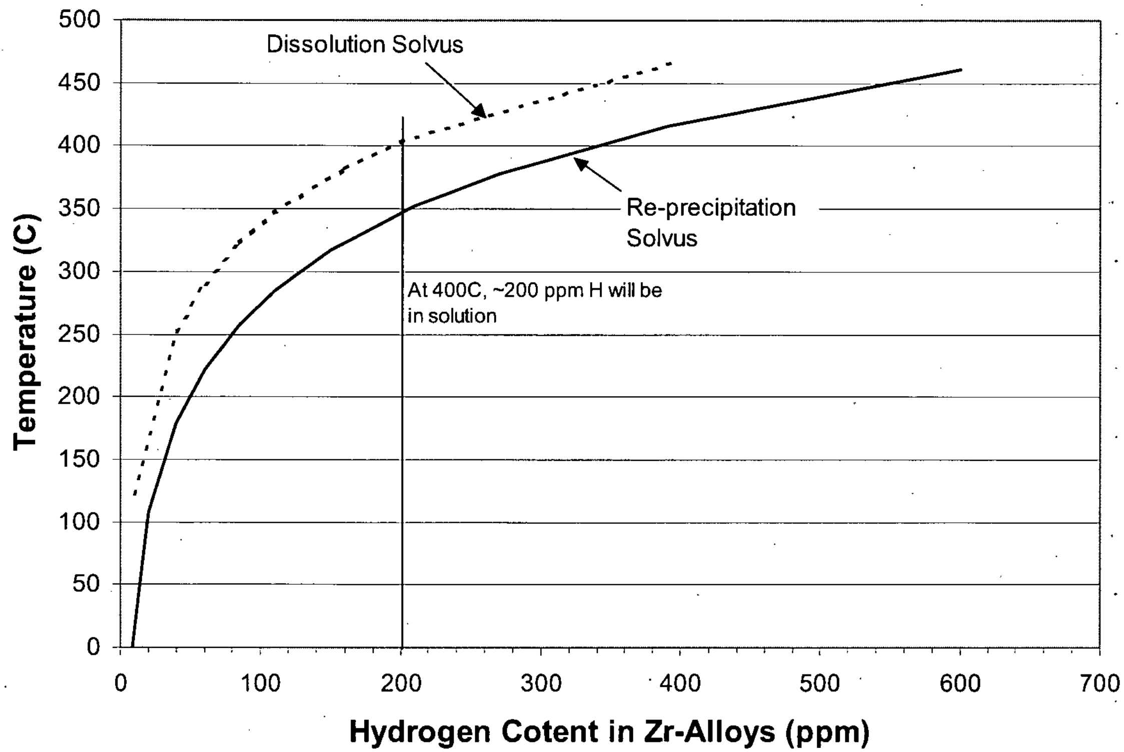
For off-normal and accident conditions

- Clad temp < 570C

Hydride Reorientation May Occur During Re-precipitation



In the Zr-H system, precipitation requires a supersaturation of H that results in a ΔT between the dissolution solvus and the precipitation solvus.

At 400C, ~200ppm H will be in solution and available for stress driven reorientation.

Solvus lines after Kammenzind et al. ASTM STP 1295, 1996, pp. 338-370

Predicted Percent Re-Precipitated Hydride

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Aomi et al., Zirconium in Nuclear Industry, ASTM Sunriver OR, June 2007 "Evaluation of Hydride Reorientation Behavior and Mechanical Property for High Burnup Fuel Cladding Tube in Interim Dry Storage"

Material:

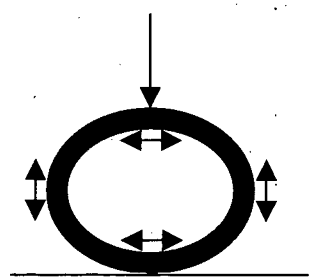
- Zr-2 with and without liner, Zr-4

Irradiation:

- ~50 GWd/t

Reorientation Method:

- Heat Treat Pressure Tube

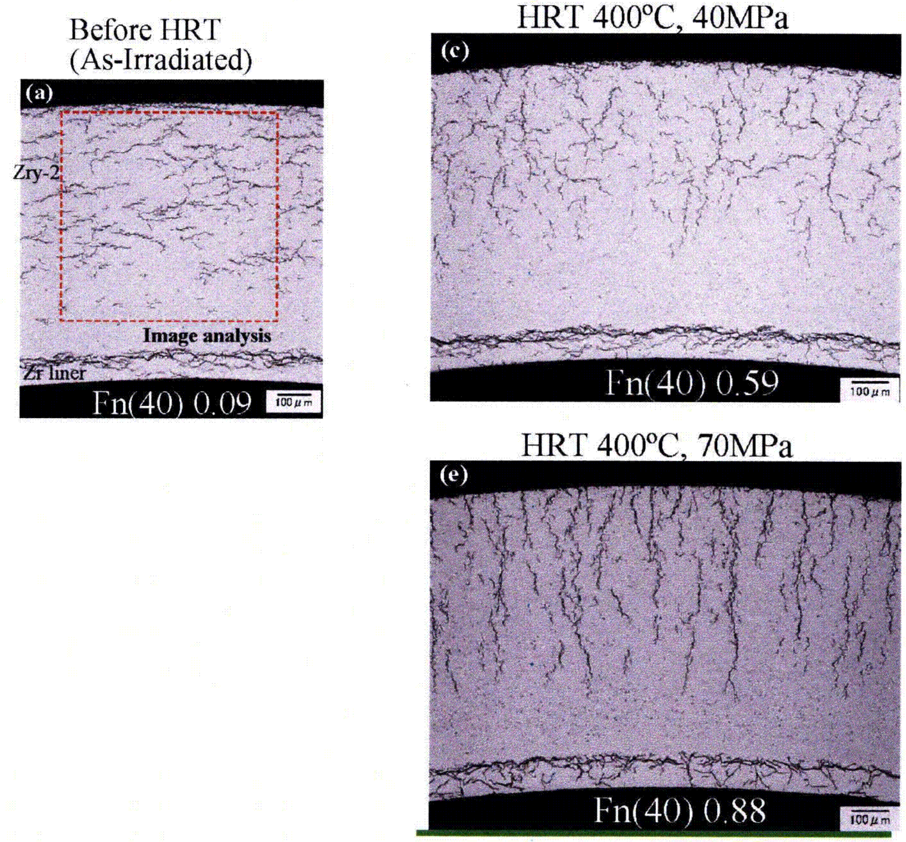


Testing:

- Majority Ring Compression Testing at Room Temperature
- Minority Ring Tensile and Longitudinal Tensile Testing at R.T.

BWR Clad: Radial Stress Threshold Aomi et al.

Constant Stress Pressurized
Tube HR Treatments



BWR Clad

50 GWd type

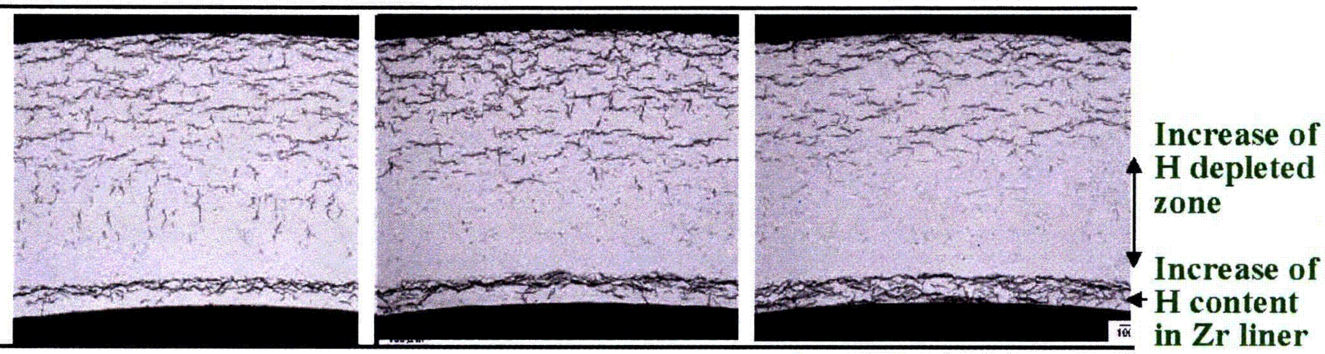
Zr-2 liner

H 133-264ppm

Effect of Cooling Rate: Aomi et al.

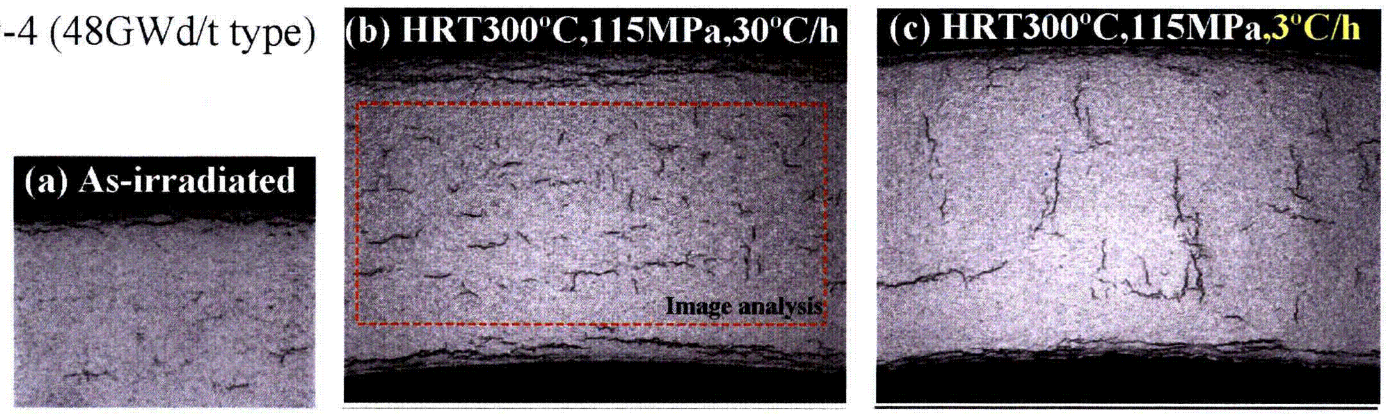
30°C/h | 3°C/h | 0.6°C/h

HRT 300°C, 70MPa
Irradiated BWR Cladding
(50GWd/t type)



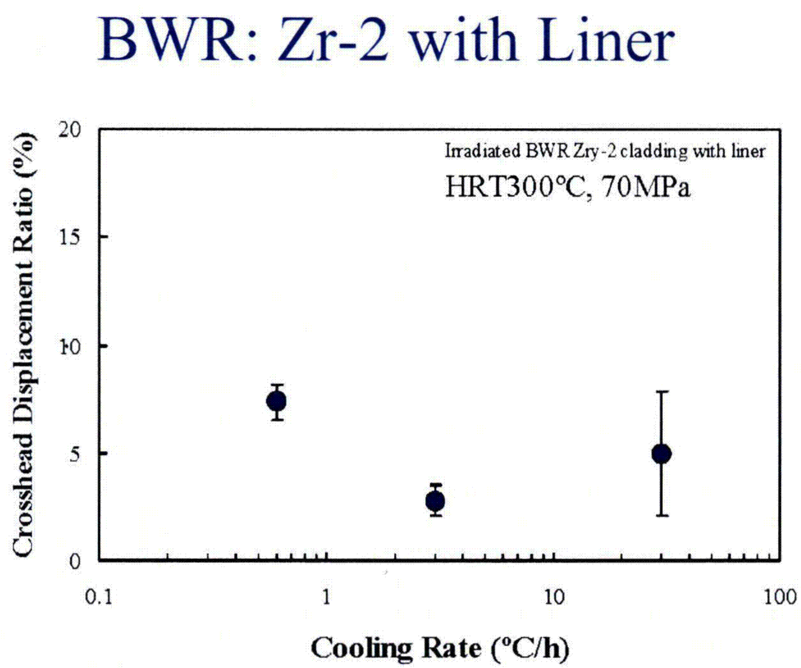
BWR Cladding: As cooling times increase, the H content in the liner increases and the amount of radial hydrides decreases

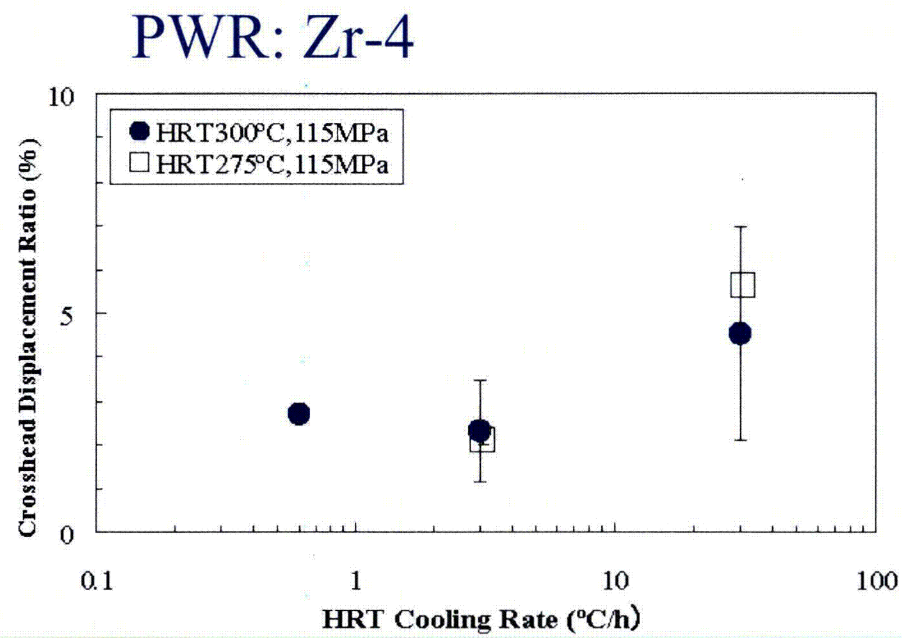
Zry-4 (48GWd/t type)



PWR Cladding: As cooling times increased, radial hydrides become longer and potentially more damaging

Effect of Cooling Rate: Aomi et al.





In BWR clad observe increase in ductility with decreasing cooling rate while in PWR ductility decreases with decreasing cooling rate.

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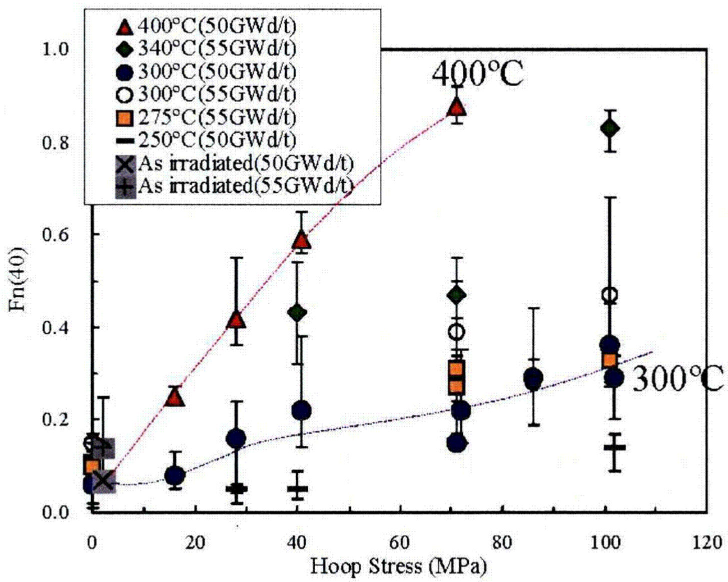
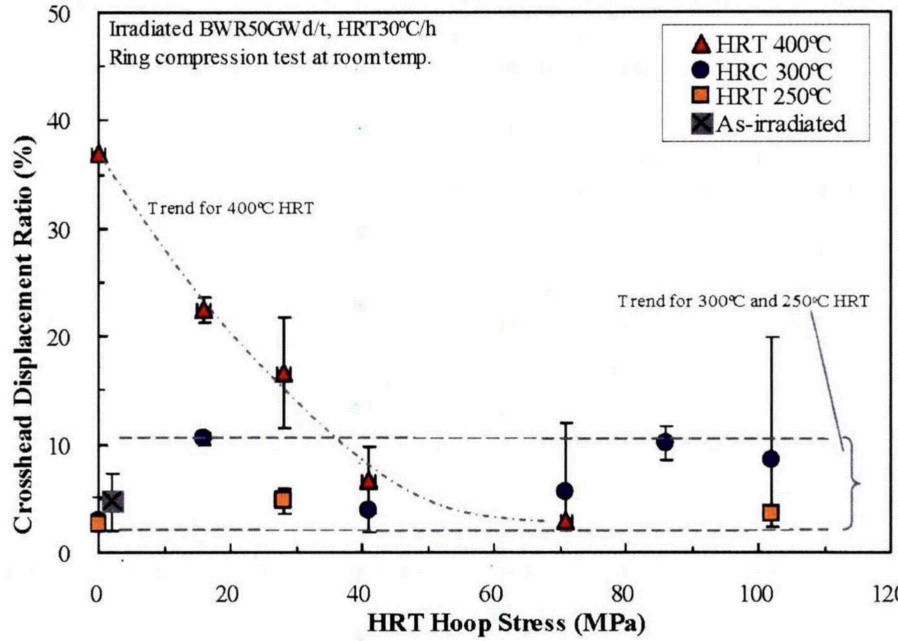
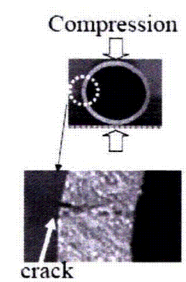
Ductility v. Radial Hydride Characterization Metrics: Aomi et al.

50 and 55GWd/t type Zry-2
with Zr liner .
H content: 98-264 ppm.
Cooling rate: 30°C/h





$$Fn\ (40) = \frac{\text{Sum of the number of hydrides in radial direction} \pm 40°}{\text{Sum of the number of total hydrides}}$$



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Summary Aomi et al.

BWR Cladding: Zr-2 with and without liner

- Zr-2 with liner may benefit from slow cooling rates in dry storage because H will diffuse to Zr liner
- Observed some reorientation at 28 MPa and 400°C but ductility for σ=70 Mpa similar to as-irradiated
- For temperature < 300°C, ductility apparently not sensitive to hoop stress

Characterizing Microstructure

- Correlated microstructural metrics to ductility and hoop stress

Summary

- Industry needs well defined performance limits for transportation and storage of spent fuel

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- Hydride reoreintation is a concern
  - Recent evidence: reorientation stress < 90 MPa
- BWR liner clad
  - H-depleted zone that increases as cooling rate decreases
    - Impact on spent fuel performance uncertain

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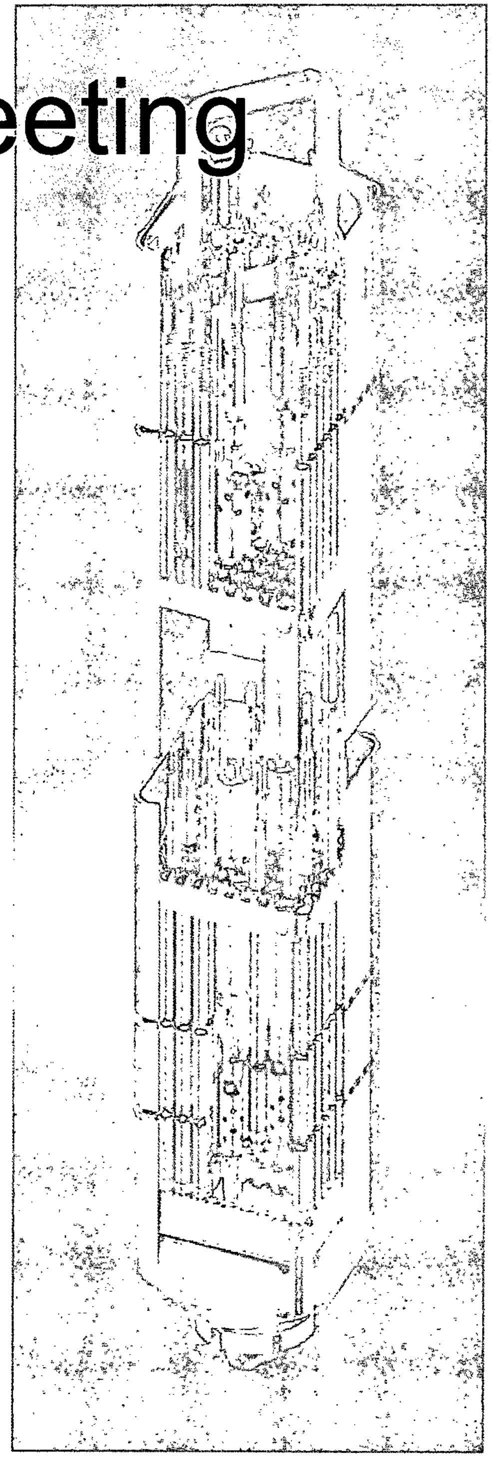
GNF Technology Update Meeting

Dry Fuel Storage
Channel Confinement System



Andy Langston, Packaging Engineer

Fuel Cycle Technology

November 20th, 2008

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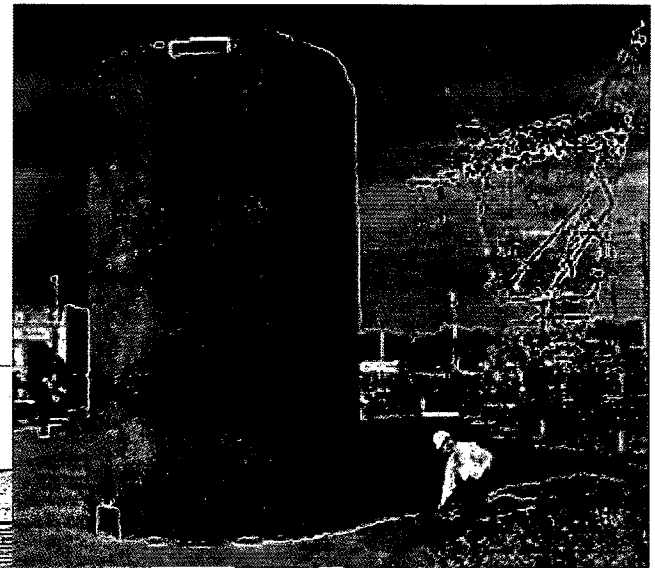
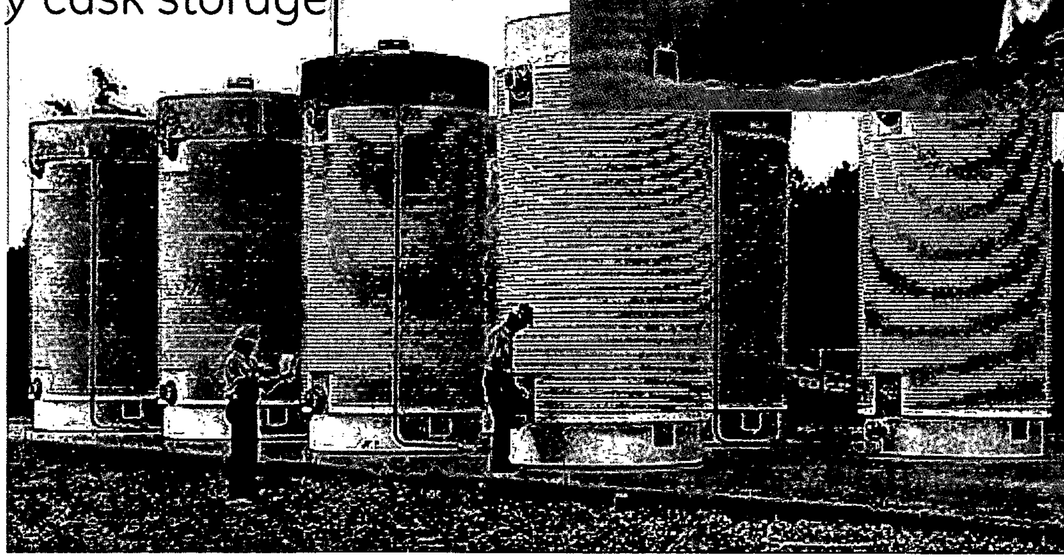
Bundle Cap Structural Evaluation

Perform Structural Analysis Consistent with Damaged Fuel Can

- Treat bundle cap technology as damaged fuel can.
- Evaluate all loading conditions associated with dry storage and transport.
- Bound loading conditions documented in dry cask storage vendors SAR.
- Work closely with dry cask storage vendor and utility to implement technology.





ANSYS Model of Channel

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60g Side Drop / Tip-Over Analysis

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GE Hitachi
Nuclear Energy

CFD Simulation of Heat Transfer Inside a Fuel Bundle

Jin Yan, PHD

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Contents

- Geometry
- Computational Mesh
- Physical Models and boundary conditions
- Preliminary Results
- Conclusion

Geometry

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Geometry

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Computational Mesh

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Computational Mesh with End Cap

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Physical Model & Boundary conditions

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Preliminary Results /Velocity vector at diagonal plane (across water rods) - **Case 1-Natural convection**

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Preliminary Results /Temperature at z=0 plane and fuel rod surfaces - **Case 1-Natural convection**

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Preliminary Results /Temperature z =0 plane and the [[fuel rod surfaces - **Case 2-No natural convection**

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Preliminary Results /Velocity vector at diagonal plane (across water rods) - **Case 3-Natural convection with**

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Preliminary Results /Temperature z =0 plane and the

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Surface temperature on the fuel rods

Sample line locations

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Surface temperature on the fuel rods

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Conclusion

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