U.S.NRC

UNITED STATES NUCLEAR REGULATORY COMMISSION

*Protecting People and the Environment*

# Hydride Effects in the Cladding Performance for the Safe Geologic Disposal of High-Level Waste


**T. Ahn[1], F. Gaskins[1], D. Pomerening[2], and M. Bailey[1]**

**[1] U.S. Nuclear Regulatory Commission**

**[2] Southwest Research Institute**


**ASTM Subcommittee C26.13, Spent Fuel and High Level Waste, Workshop on Hydride Re-Orientation in Zirconium Alloys, Costa Mesa, CA, January 31 – February 1, 2007**

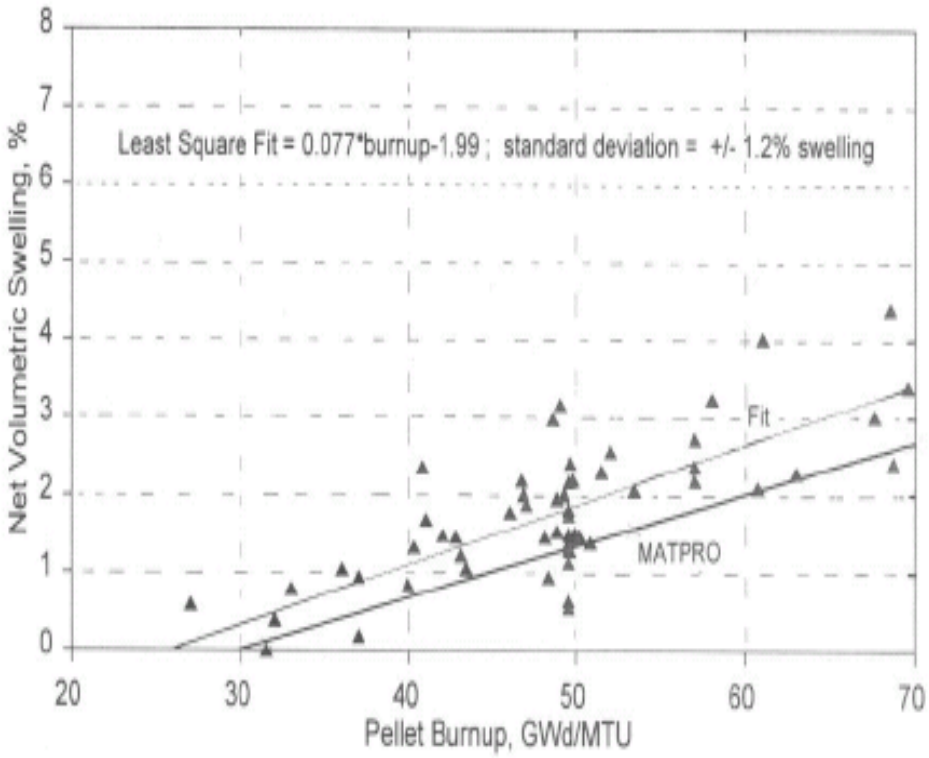
# Outline

- Purpose
- Radiation Damage
- Residual Thermal Expansion
- Cooling Rate
- Applied Mechanical Stress
- Risk Insights

# Purpose

- Present potential issues associated with cladding protection in the long-term repository performance of high-level waste
- Obtain feedback from the Subcommittee on the potential issues
- Participate in the potential development of standards/guidance on the hydride embrittlement of cladding

# Radiation Damage




PWR Swelling Data and Least Square Fit to MATPRO Model (Lanning et al., 1997)

- The matrix swelling by irradiation in the reactor is as big as the thermal expansion of the matrix during the reactor operation. There could be bonding of Zr-$UO_2$.
- This stress could affect the susceptibility of the hydride reorientation at lower temperatures and may cause delayed-hydride cracking (DHC).
- It is likely that there will be more localized effects for high burnup spent nuclear fuel (SNF) (Chung, 2004).

# Residual Thermal Expansion Stress

$$\sigma = E/(1 - 2\nu)\,[\Delta V/(3V)]$$

$\sigma$: stress developed
E: Young's modulus
$\nu$: Poisson's ratio
V: volume
$\Delta$: change

Various more rigorous models are available
(e.g., Retel et al., 2004)

- Localized stress from the matrix thermal expansion could still be high at temperatures of 300 to 400 °C (572 to 752 °F) for the embrittlement.
- The preliminary exercise estimate was made based on models by Retel et al. (2004).
  The estimate considered the matrix expansion in the gap.
- It is likely that there will be more localized effects for high burnup SNF (Chung, 2004).

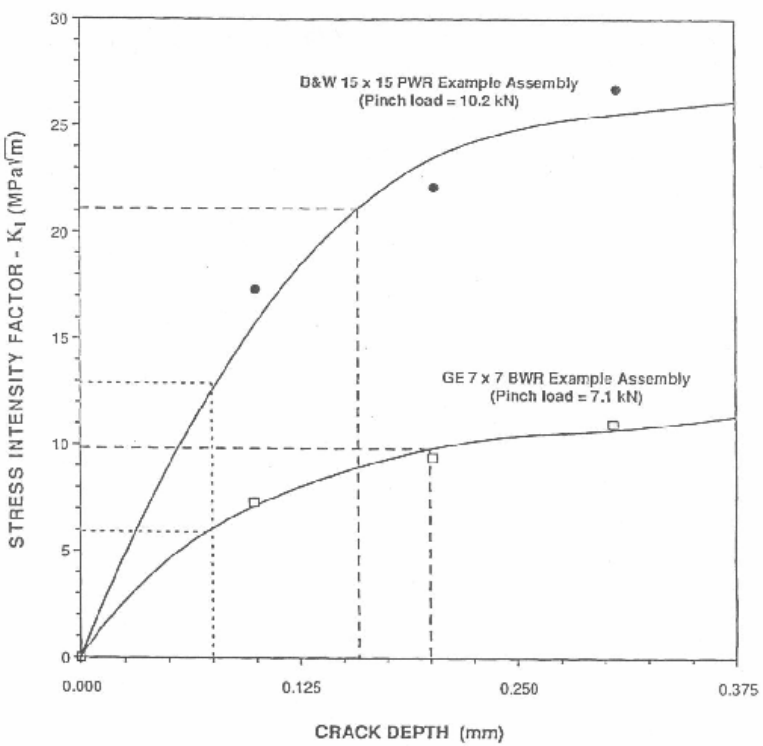
# Cooling Rate

- Substantial ductility loss as a function of cooling rate
- For various levels of radial hydride fraction, hydrogen concentration, and external stress during hydride precipitation

Chan (1996)

- Repository Cooling Rate:
  lower than $10^{-5}$ °C/sec
  ($1.8 \times 10^{-5}$ °F/sec)
- Loading Cooling Rate:
  higher than $10^{-5}$ °C/sec
- Current Repository Temperature Limit in Total System Performance Assessment: 300 - 350 °C
  (572 – 662 °F)

# Applied Mechanical Stress




Sanders et al. (1992)

- The seismic-induced stress is one of the sources of the applied stress in the repository, e.g., rockfalls [Alloy 22 – Nominal Yield Stress: 310 MPa (45.0 ksi), Ultimate Tensile Stress: 690 MPa (100.1 ksi)].
- During vibration, or under accident conditions, any pinch stress could cause high $K_I$.
- Oxide-induced stress.

# Risk Insights

- Operational Safety:

  The radionuclide source term may be modified with the hydride embrittlement of cladding during preclosure operations. The release fraction might be increased from less partial cladding protection with any additional embrittlement. However, other mitigation factors may reduce the release fraction, such as the Transport, Aging and Disposal (TAD) canister or the building containment.

- Repository Isolation:

  The early release of Tc-99 and I-129 appears to be sensitive to the area of the exposed SNF matrix. The SNF matrix exposure may be due to cladding degradation such as gross loss or perforations (e.g., pin holes and hairline cracks). The cladding embrittlement may facilitate cladding degradation, increasing the area of the exposed SNF matrix. Total System Performance Assessment should take into consideration cladding degradation.

# References

K. S. Chan, A Micromechanical Model for Predicting Hydride Embrittlement in Nuclear Fuel Cladding Material, J. of Nuclear Materials, Vol. 227, p. 220-236, 1996

H. Chung, Understanding Hydride- and Hydrogen-Related Processes in High-Burnup Cladding in Spent-Fuel-Storage and Accident Situations, Proc. 2004 Int. Mtg. LWR Fuel Performance, Orlando, FL, p. 470, American Nuclear Society, 2004

D. D. Lanning, C. E. Beyer and C. L. Painter, FRAPCON-3: Modifications to Fuel Rod Material Properties and Performance Models for High-Burnup Application, NUREG/CR-6534, Vol. 1, PNNL-11513, Pacific Northwest National Laboratory, prepared for U.S. Nuclear Regulatory Commission, 1997

V. Retel, F. Trivaudey, M. Boubakar, D. Perreux and P. Thevenin, Comparative Effects of Structural and Material Parameters Variability on Pellet-Cladding Interaction in a PWR Fuel Rod, Nuclear Engineering and design, Vol. 228, p. 35-46, 2003

T. Sanders, K. Seager and others, A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements, SAND90-2406, Sandia National Laboratories, Albuquerque, NM, 1992

# Disclaimer

The NRC staff views expressed herein are preliminary and do not constitute a final judgment or determination of the matters addressed or of the acceptability of a license application for a geological repository at Yucca Mountain. This presentation is also an independent product of the Southwest Research Institute and does not necessarily reflect the views or regulatory position of the NRC.