Potential Yucca Mountain Repository:

the Materials Behavior in the Risk-informed Performance-based Evaluation

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(4) Korea Institute of Nuclear Safety (KINS), Daejeon, Korea, March 29, 2006

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.

PURPOSE

- **Present U.S. Nuclear Regulatory Commission (NRC)'s risk-informed, performance-based evaluation of materials behavior in the potential licensing of the Yucca Mountain (YM) repository**
- **Present compliance assessment methods and tools for risk assessments, for operation safety, and for long-term waste isolation**
- **Present issues associated with waste form**
- **Present issues associated with waste package (WP) and drip shield (DS) materials**

OUTLINE

- The NRC's Evaluation of the Potential YM Repository of High-Level Waste (HLW)
- Schematic Illustration of the Potential YM Repository
- Compliance Assessment and Tools for Risk Assessment: operation safety, waste isolation

OUTLINE (continued)

- **Waste Form:**

  - initial cladding conditions
  - dose calculation for drop/collision and oxidation of spent nuclear fuel (SNF) assemblies
  - repository radionuclide release and dose calculation
  - rims, hydride embrittlemet, Np solubility and Pu colloids

- **Canister, WP and DS:**

  - Transportation, Aging and Disposal (TAD) Canister
  - WP and DS – failure modes, localized corrosion, long-term passivity, materials stability, creep, fabrication reliability

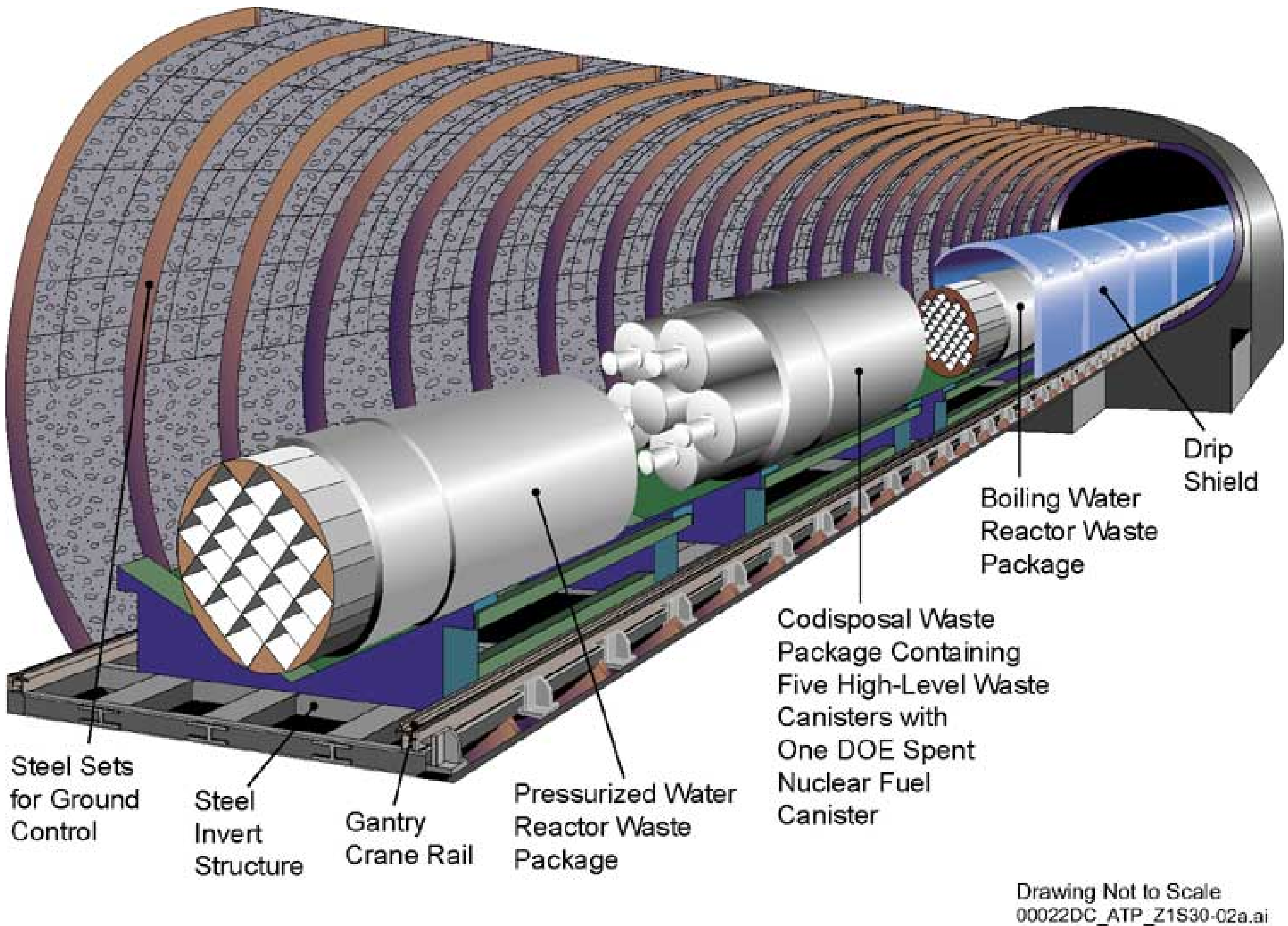
Potential YM Repository of HLW

The NRC's evaluation includes:

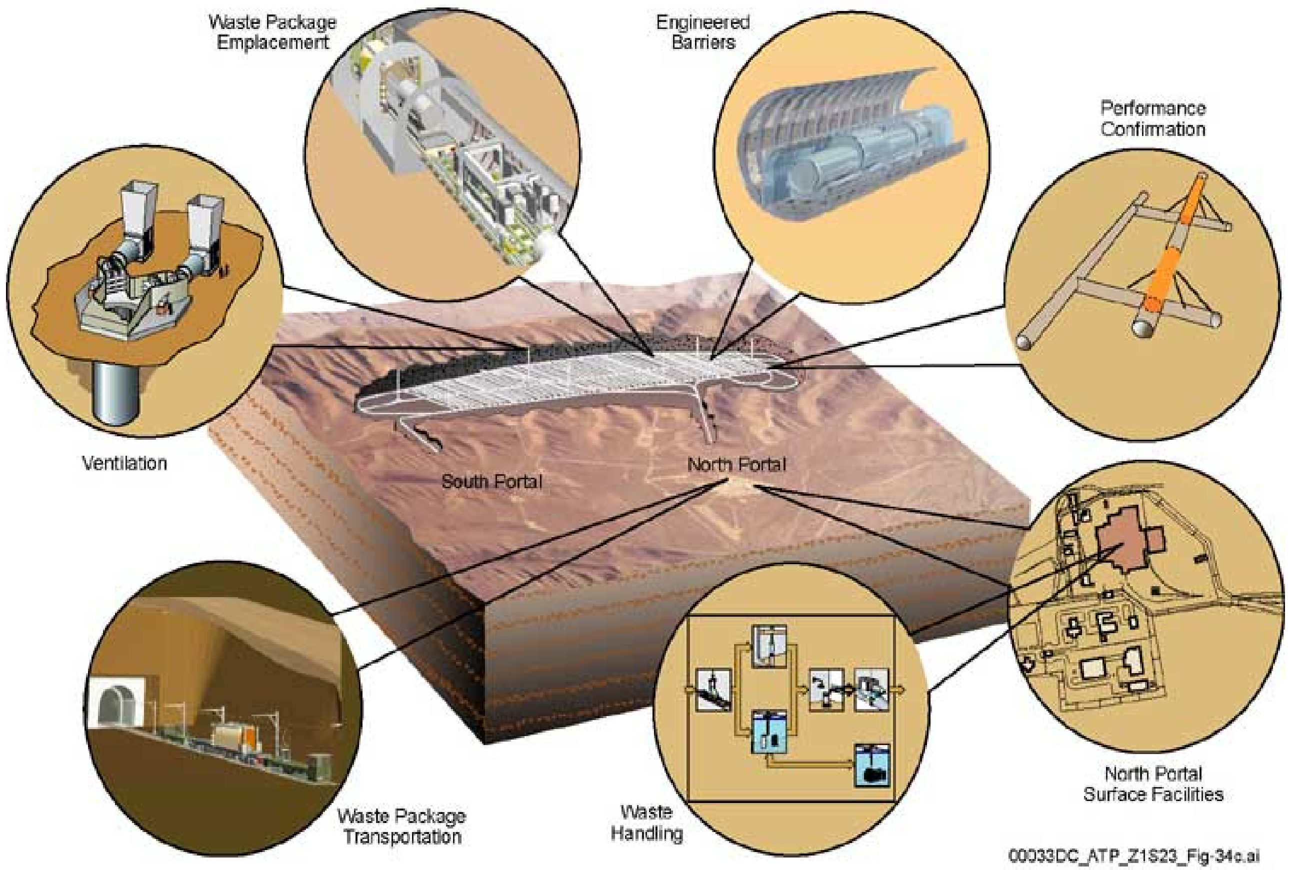
(i) Construction and operation of the pre-closure facilities (i.e., safety)

(ii) Ability of the repository to isolate HLW (i.e., waste isolation)

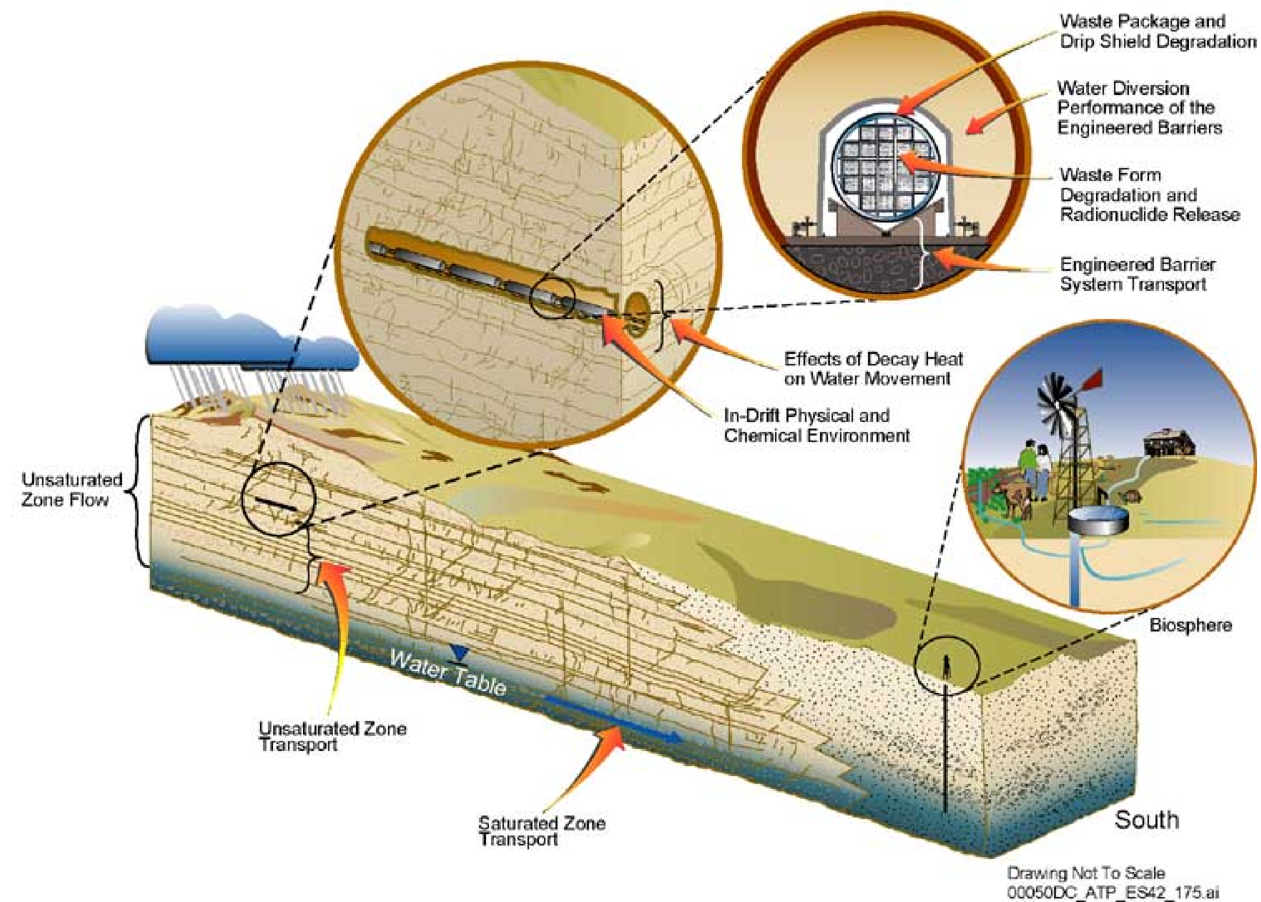
Schematic Illustration of the Emplacement Drift with Cutaway Views of Different Waste Packages (DOE, 2002)



Proposed Monitored Geologic Repository Facilities at Yucca Mountain (DOE, 2002)



Schematic Illustration of the Ten General Processes Considered and Modeled for Total System Performance Assessment (DOE, 2002)



View Looking Down Exploratory Studies Facility (DOE, 2002)

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Compliance Assessment

The radionuclide releases must comply with the performance objectives (i.e., regulatory dose requirements)

(i) safety: for workers and the public under normal operating and accident conditions (e.g., drop/collision, power off, building leakage, seismicity, or aircraft crash)

(ii) waste isolation: for individual protection, human intrusion, and groundwater protection under nominal and disruptive (e.g., seismicity and volcanism) scenarios

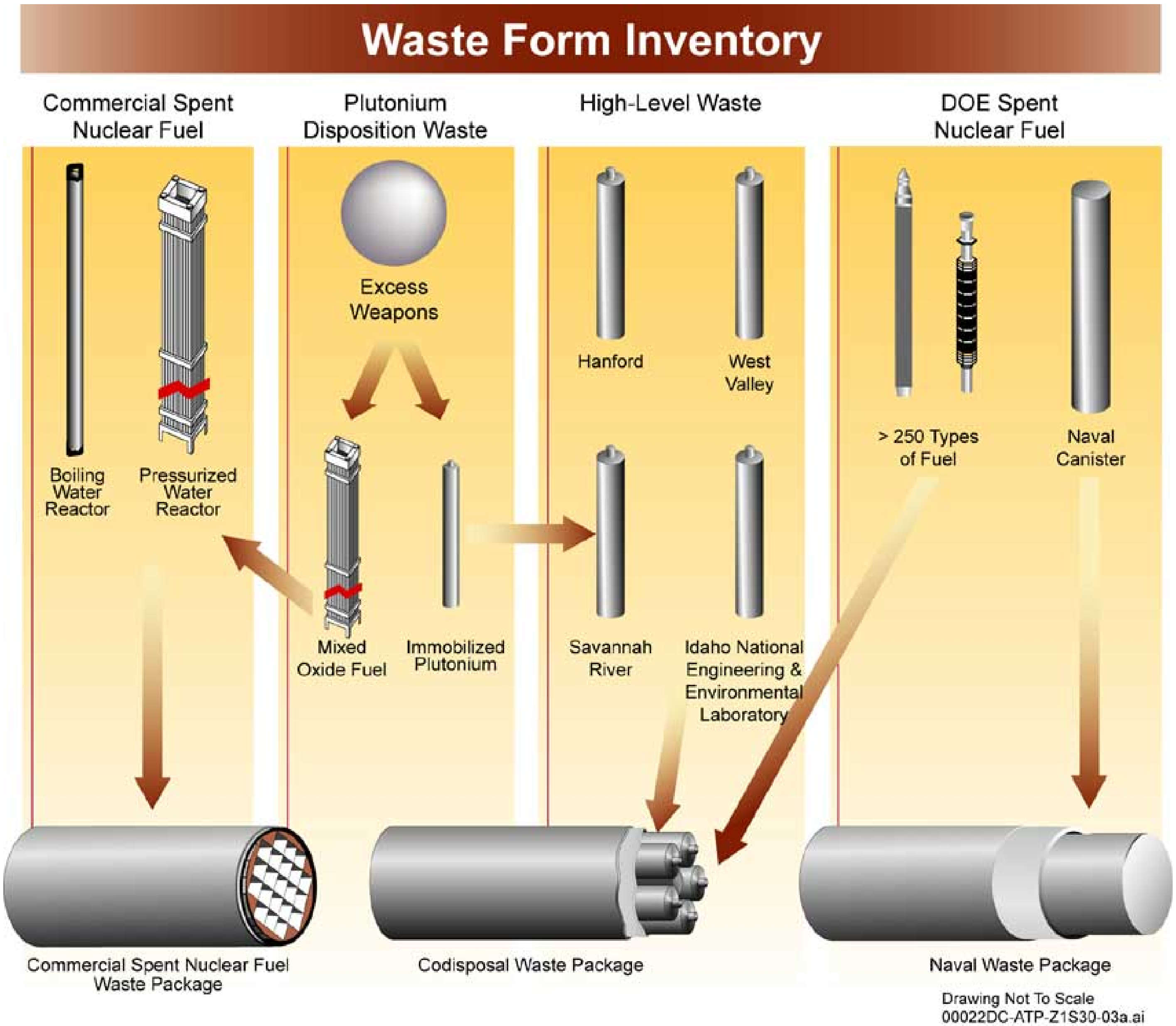
Tools for Risk Assessments

- **Safety: Pre-Closure Safety Analysis (PCSA)**

  - initiating events and event sequences - annual probability cutoff
  - SNF assemblies at risk
  - fault and event tree probabilistic scheme
  - consequence models
  - bases for model supports

- **Waste Isolation: Total-system Performance Assessment (NRC's TPA, DOE's TSPA)**

  - screening of event sequences and scenarios – annual probability cut off
  - Monte Carlo scheme
  - consequence models – engineering, geology, and hydrology
  - bases for model supports


Waste Form Inventory
Commercial Spent Nuclear Fuel
Plutonium Disposition Waste
High-Level Waste
DOE Spent Nuclear Fuel
Boiling Water Reactor
Pressurized Water Reactor
Excess Weapons
Mixed Oxide Fuel
Immobilized Plutonium
Hanford
West Valley
Savannah River
Idaho National Engineering & Environmental Laboratory
> 250 Types of Fuel
Naval Canister
Commercial Spent Nuclear Fuel Waste Package
Codisposal Waste Package
Naval Waste Package
Drawing Not To Scale
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Initial Cladding Condition: pin-holes, hairline cracks, or rupture

Safety:

- could be generated during the normal reactor operation; could be developed more and further during the subsequent handling operations, should the cladding have high stress, vibration or larger flaws, or is subject to embrittlement

- may lead to matrix oxidation should the SNF be exposed to oxygen at elevated temperatures, which may occur at SNF dried for loading. Partial oxidation potentially causes SNF pellets to crumble and generate respirable particulates upon vibration or any stress applied during transportation and storage.

- may lead to matrix oxidation should the SNF be exposed to oxygen at elevated temperatures, which may occur as SNF is transferred from transportation/staging packages to the disposal waste package at the potential pre-closure facilities

- methods for detecting pin-holes and hairline cracks and their sensitivity are being explored

Waste Isolation:

- Failed waste packages (e.g., initially defective, corrosion or mechanical) may allow radionuclide releases from SNF rods with pin-holes and hairline cracks.

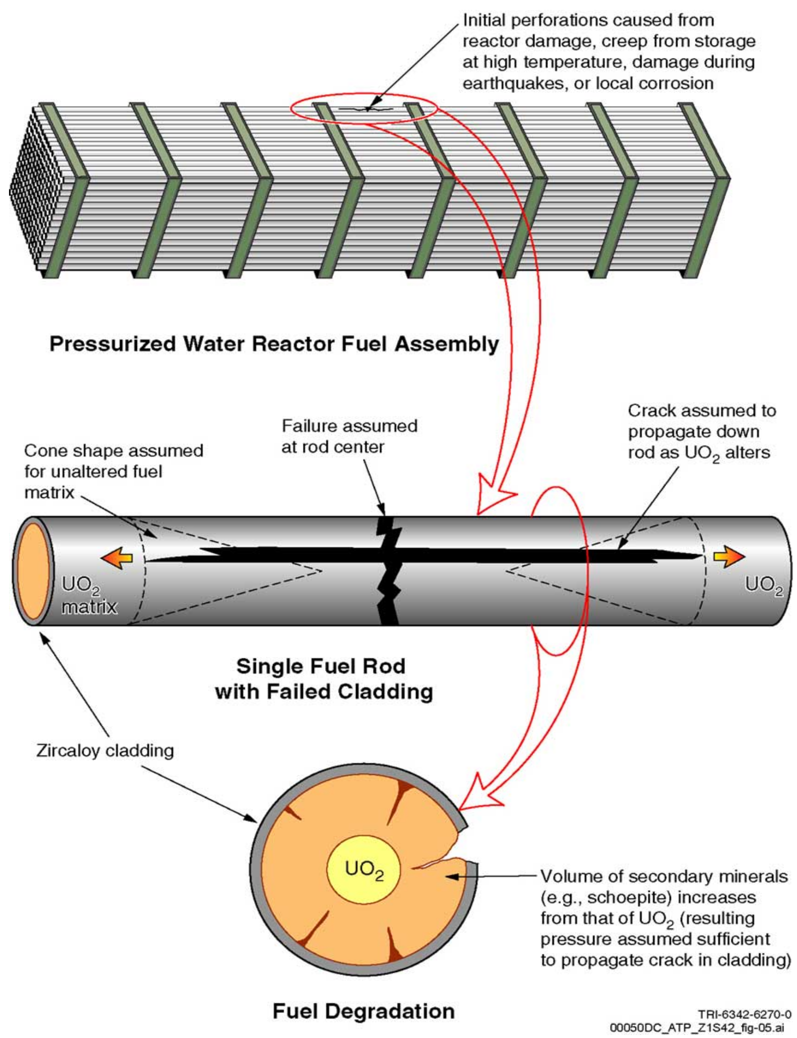
  The grain boundary and gap radionuclide inventories are considered to be mobilized rapidly, whereas the matrix inventory is considered to be mobilized at the slow SNF dissolution rate.

- Effects of pin-holes and hairline cracks on release

  No partial credit is given to cladding with pin-holes and hairline cracks or ruptured in DOE's TSPA – assumed to be bare SNF pellets without cladding

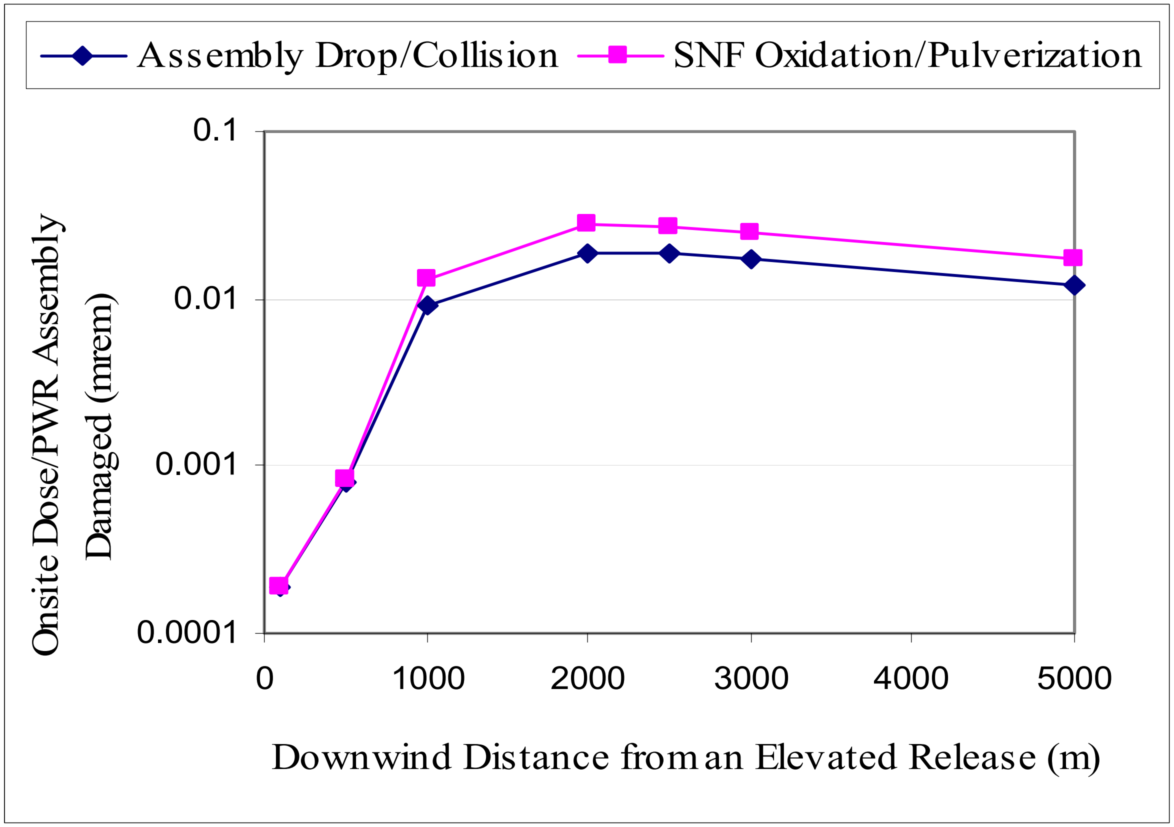
- SNF rods with pinholes and hairline cracks or ruptured may lead to reconfiguration for potential criticality.

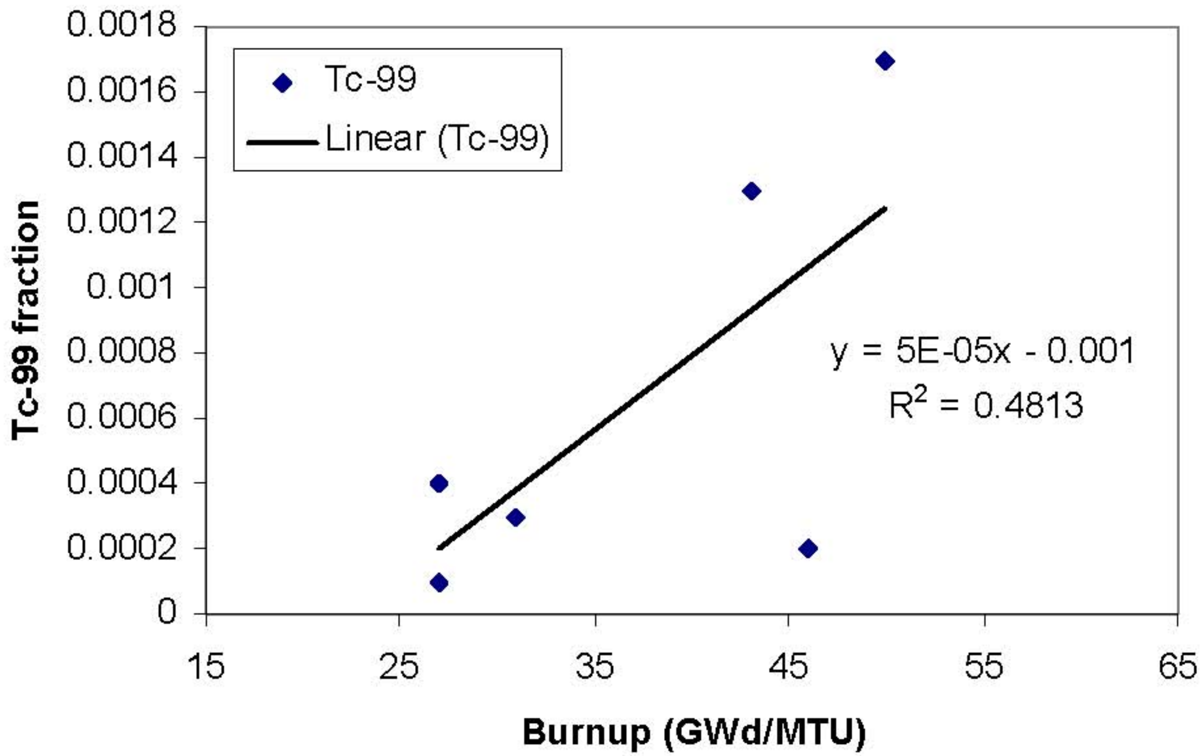
Conceptual Model of Commercial Spent Nuclear Fuel Cladding Degradation (DOE, 2002)



Example Dose Calculation Per SNF Assembly

(Kamas et al., 2005)





Instant Release Fraction for Tc-99 As a Function of Burnup. Data Are Extracted from Johnson and Tait (1997). (50GWd/MTU = 3.64×10^{12} Btu/ton)

(From Jain et al., 2004)

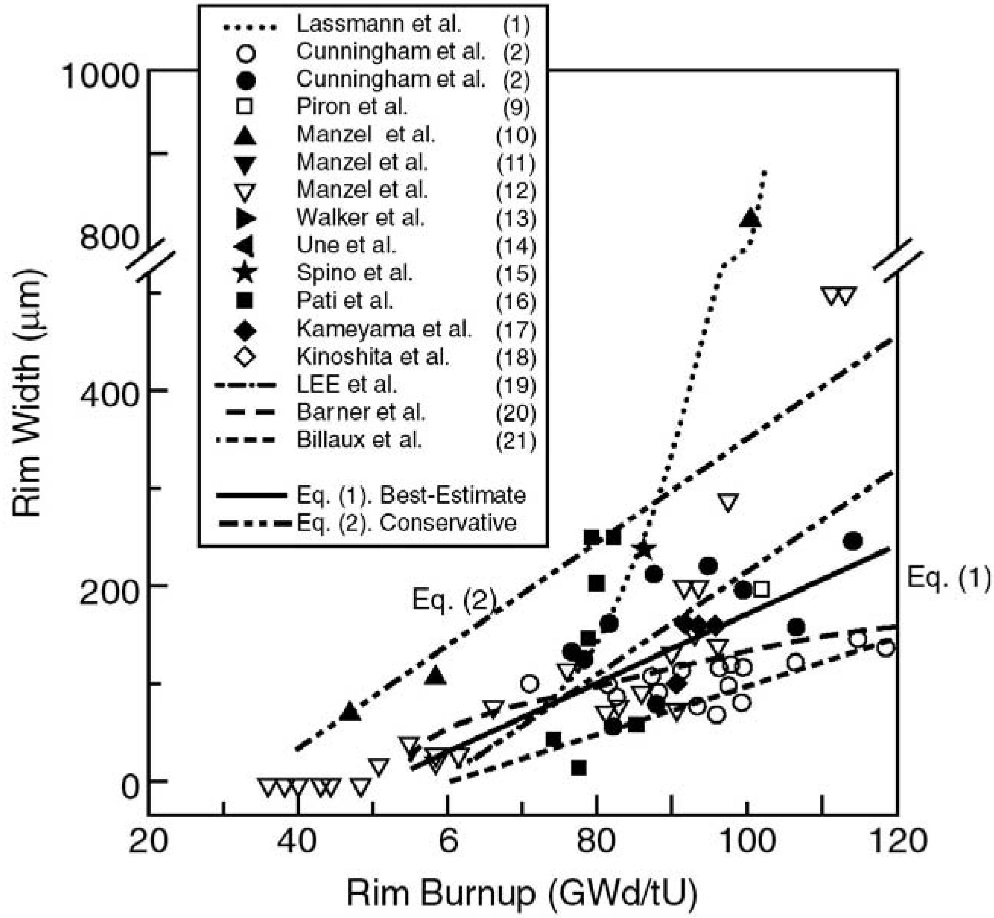
Matrix Dissolution

- **Groundwater Contact Mode**
- **Cation**
- **pH**
- **Oxygen and Iron Concentration**
- **Temperature**
- **Prior Oxidation/Hydration**
- **Grain Boundary Inventory**

- **Partial Cladding Protection**

- Slit (~0.015 cm [5.9x10^{-3} in] dia. X ~2.54 cm [1 in] length) - or Hole (~0.02 cm [7.9x10^{-3} in] dia.) – Defective SNF rods Immersion Tests in J-13 Well Water at 85 °C (Wilson, 1990)

- release decreased by a factor of ~ 140 for Tc-99, ~7x10^{5} for I-129 and ~65 for Sr-90

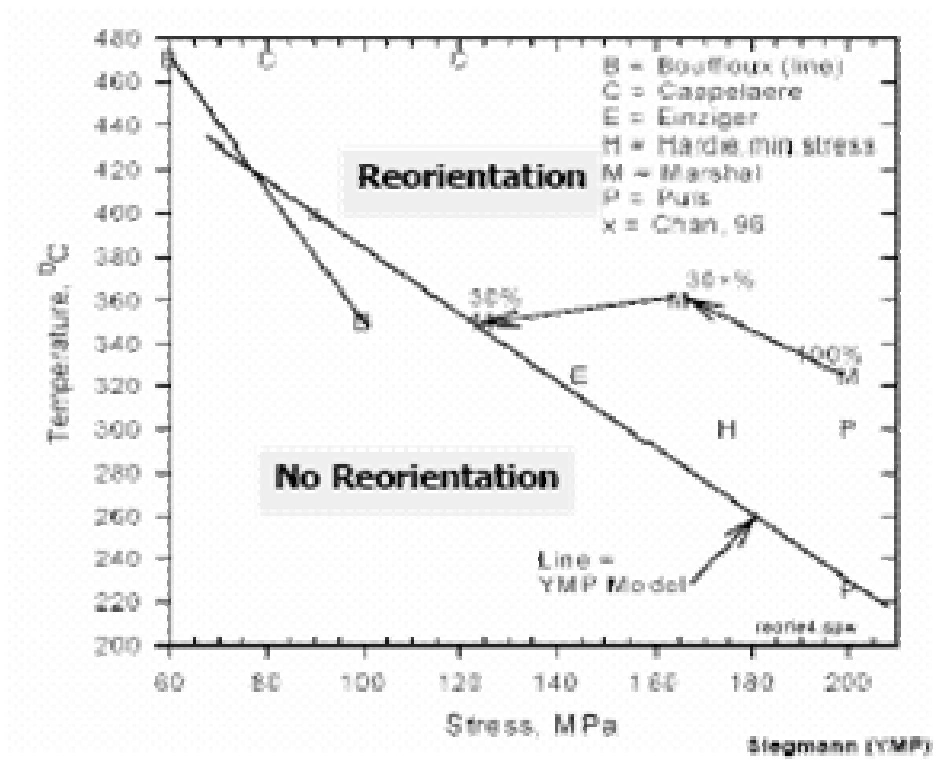


Measured Rim Width Both by Electron Microprobe Analyzer and Optical Microscopy (Koo, et al., 2001). (Reproduced with Permission from Copyright Clearance Center). NOTE: Refer to Koo, et al. (2001) for Information on References Cited in the Figure.

(From Jain et al., 2004)

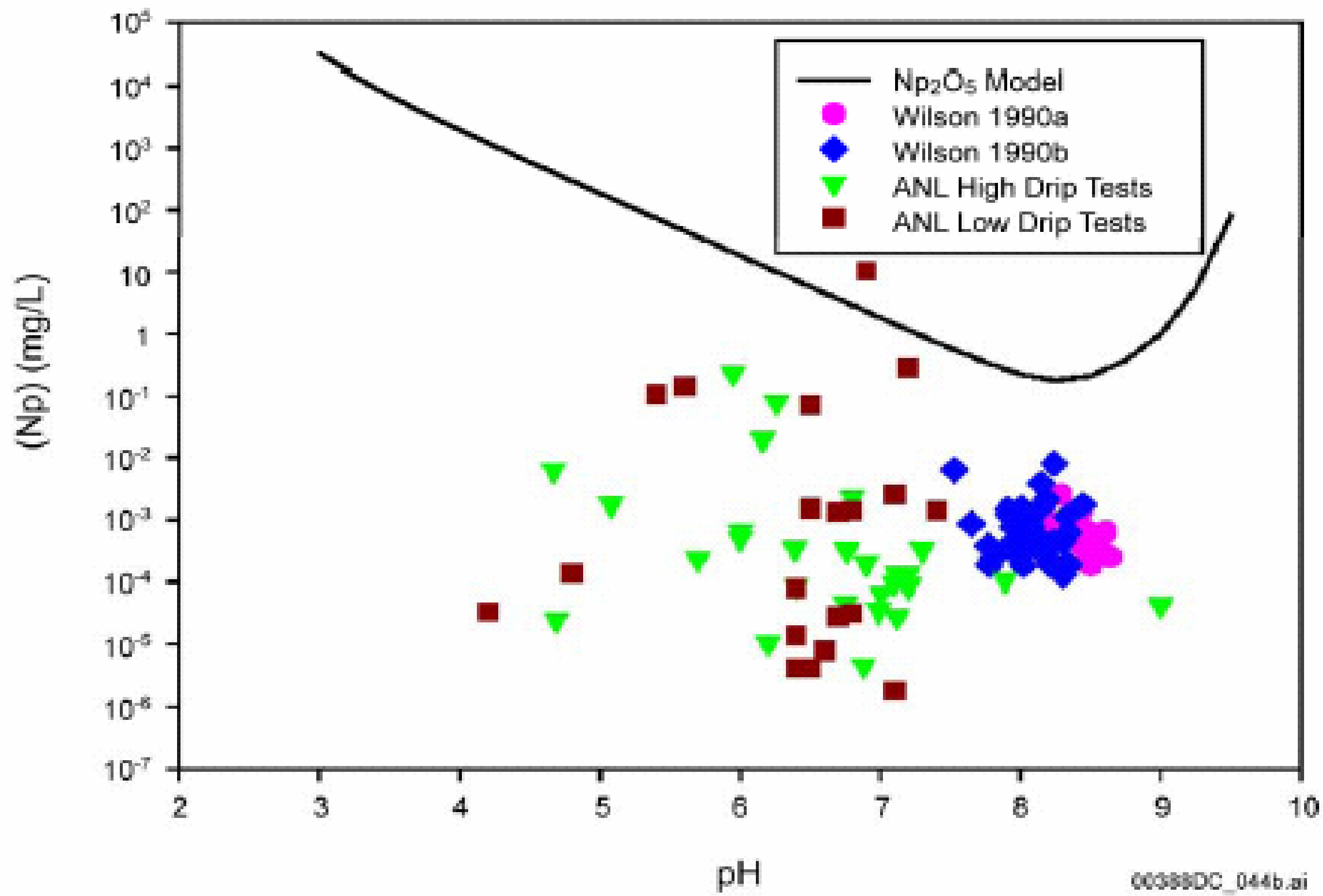
Hydride Reorientation – Creep Tests

- **Radial hydrides, as little as 40 wppm, can significantly degrade cladding's mechanical properties.** **(Marshall)**
- **Stress, temperature, cool-down rate, microstructure, H content, etc., all play important roles.** **(Einziger)**
  - **Threshold hoop stress for 400°C is ~ 100 MPa.**

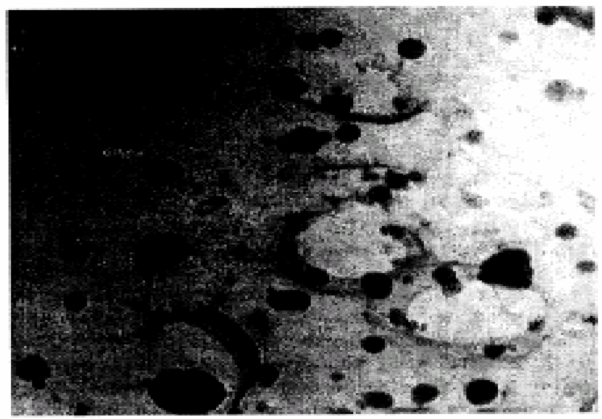


From Tsai, 2003 **(1 MPa = 0.145 ksi; 200°C = 392 °F)**

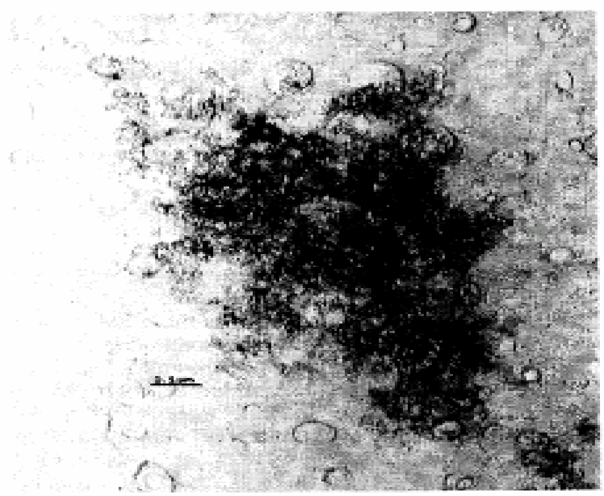
Np Solubility (DOE, 2004)



Transmission Electron Microscopy (TEM) Micrographs of Particulate Material Isolated on a Holey Carbon TEM Grid: (a) Colloids Formed from Solution and (b) Material in Liquid Spalled from the Glass Surface (Bates et al., 1992)



(a)



(b)

Cladding Protection in Release (Mohanty et al., 2004)

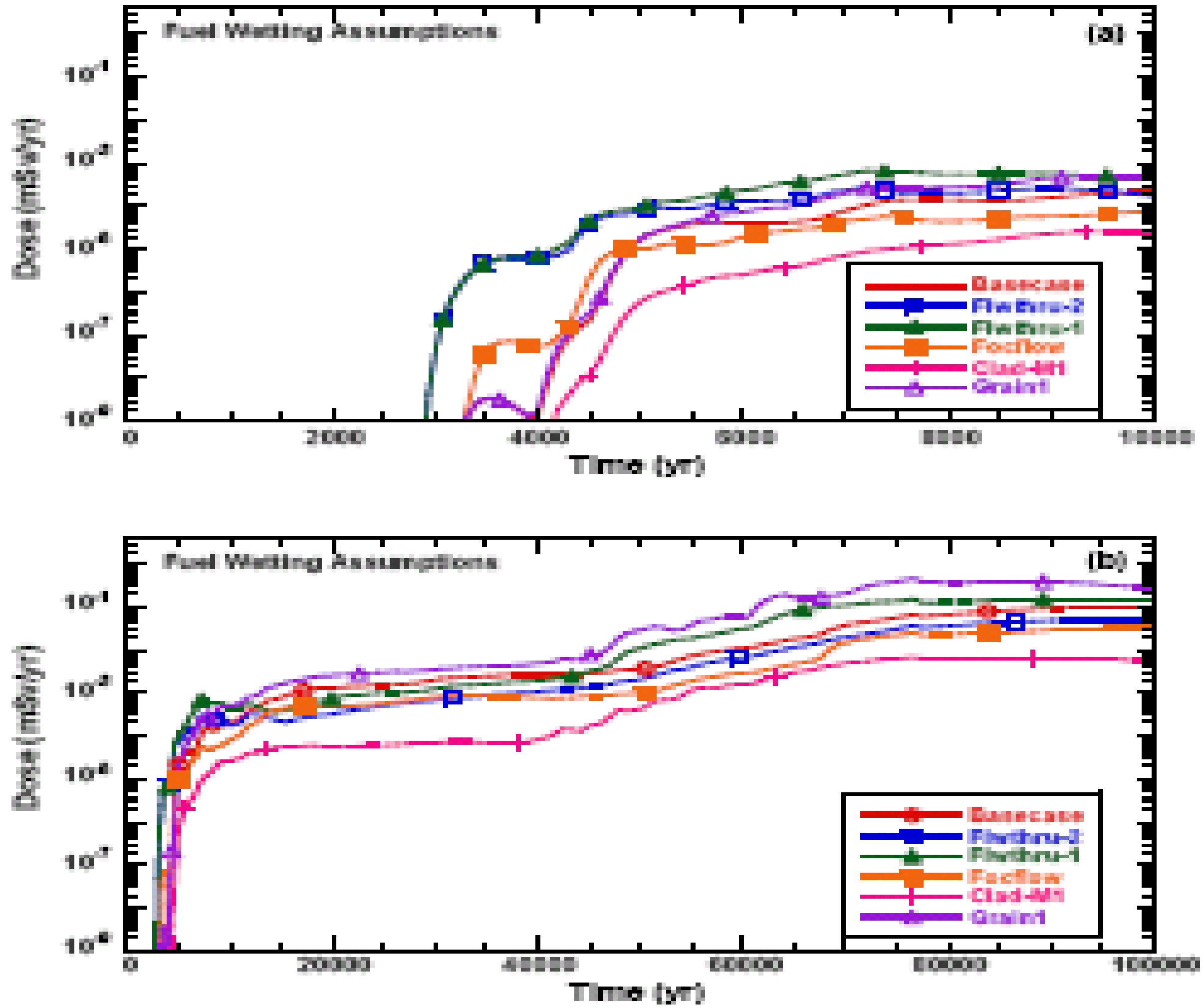


Figure 3-37. Average Groundwater Dose from the Basecase and the Fuel-Wetting Alternative Conceptual Models in (a) 10,000 and (b) 100,000 Years, for 350 Realizations

TAD Canister

DOE's Recently Proposed Conceptual Design using

Standardized Transportation, Aging and Disposal (TAD) Canisters

Waste Package

(Alloy 22, Ni-22Cr-13.5Mo-3W-4Fe)

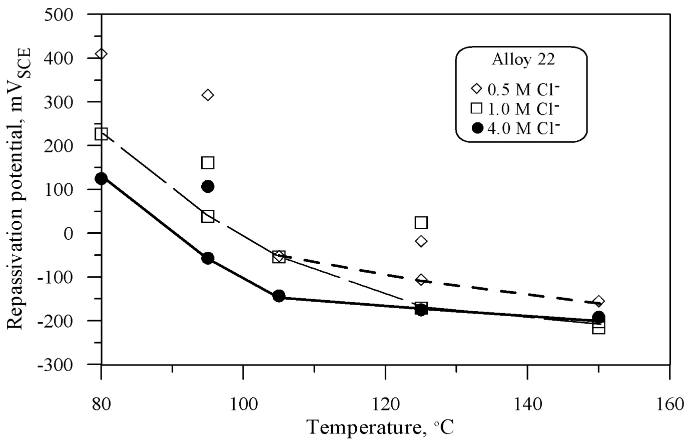
- Long-Term Passivity in Uniform Corrosion
- Localized Corrosion
- Stress Corrosion Cracking
- Long-Term Phase Stability
- Mechanical Failure
- Fabrication and Reliability

Drip Shields

(Ti-7 and Ti-24, Pd addition)

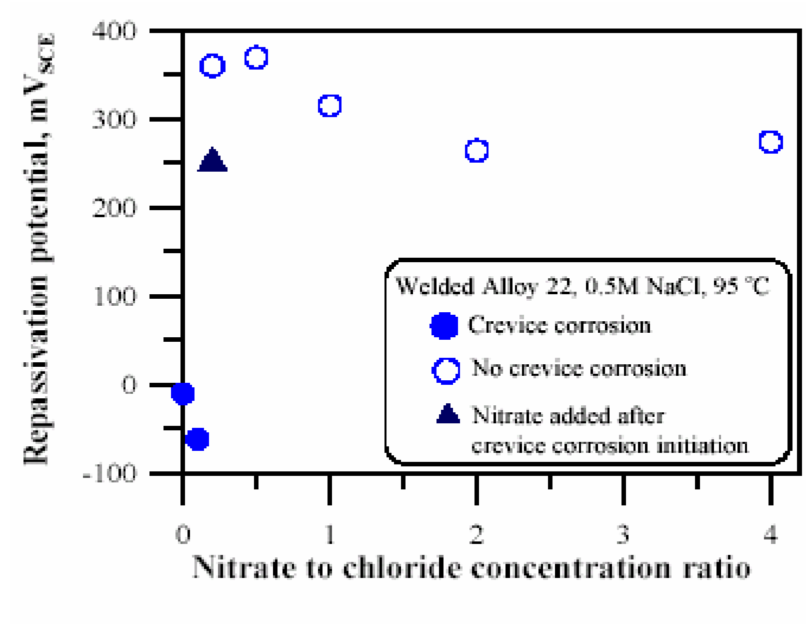
- **Long-Term Passivity in Uniform Corrosion**
- **Fluoride Uniform Corrosion**
- **Hydride Embrittlement**
- **Stress Corrosion Cracking**
- **Creep**

Localized Corrosion



Effect of temperature on the repassivation potential crevice corrosion of Alloy 22 in Cl$^-$ solutions (Brossia et al., 2001)

(100 °C = 212 °F)



(Dunn et al., 2003)

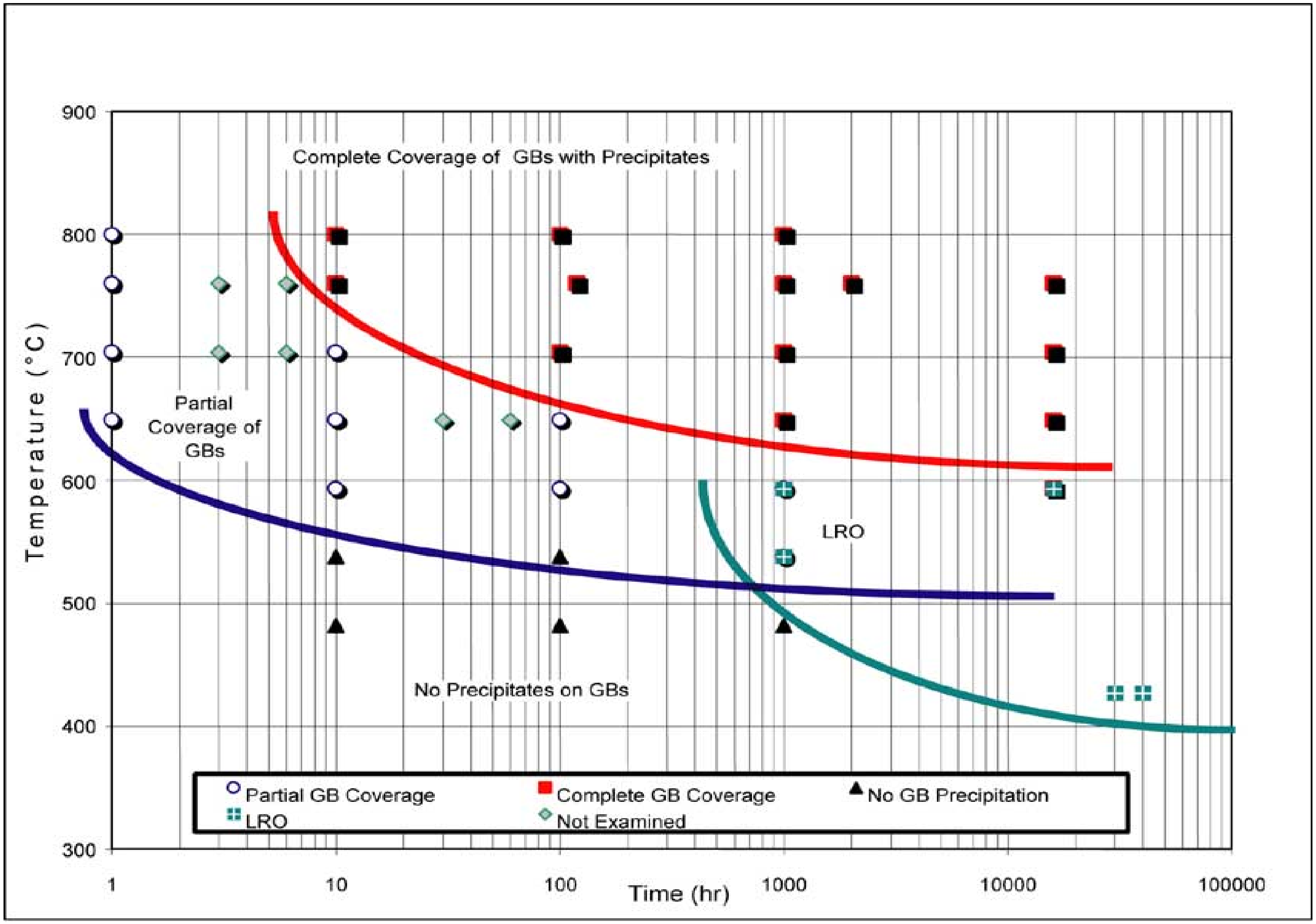
Uniform Corrosion

- Data from DOE, CNWRA, industries and international community (e.g., long-term German tests in rocksalts) point out similarities of uniform corrosion rates. A container lifetime of greater than 10,000 years can be estimated.

- A long-term integrity of passive film is suggested by various models for point defects, chemistry segregation, and passive film growth.

- Analogue studies suggests that modern electrochemical theories for corrosion may explain the analogue observation: void formation, stoichiometric dissolution of meteorites and josephinite, possible passivity of Indian Pillar, and long-term passivity of carbon and stainless steel over half a century (Sridhar and Cragnolino, 2002).

Time-Temperature-Transformation Diagram for Alloy 22 Base Metal (DOE, 2002)



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Low-Temperature Drip Shield Creep
(Neuberger et al., 2002)

- **Expected drip shield creep after rockfall loadings near material yield stress while drift operating temperatures are approximately 100 – 150° C**

Metal Fabrication Reliability
(Jain et al., 2003)

Table 4-1. Observed Weld Flaw Frequencies*†

Vessel	Location	Size of Cracks	Weld Volume	Number of Flaws	Flaws/m^3 [flaws/ft^3]
Pressure Vessel Research User Facility	Near Surface Zone {25 mm [0.98 in]}	< 3 mm [< 0.12 in]	0.014 m^3 [0.49 ft^3]	191	13,571 [384.3]
	Near Surface Zone {25 mm [0.98 in]}	> 3 mm [> 0.12 in]	0.014 m^3 [0.49 ft^3]	13	929 [26.3]
	Remaining thickness	< 5 mm [< 0.20 in]	0.20 m^3 [0.7 ft^3]	653	3,625 [102.6]
	Remaining thickness	> 5 mm [> 0.20 in]	0.20 m^3 [0.7 ft^3]	27	135 [3.8]
Shoreham Reactor Pressure Vessel	Inner 25 mm [0.98 in] surface	< 4 mm [< 0.16 in]	0.0226 m^3 [0.8 ft^3]	459	20,309 [574.5]
	Inner 25 mm [0.98 in] surface	> 4 mm [> 0.16 in]	0.0226 m^3 [0.8 ft^3]	9	398 [11.3]
	Outer 25 mm [0.98 in] surface	< 4 mm [< 0.16 in]	0.0241 m^3 [0.85 ft^3]	639	26,515 [750.8]
	Outer 25 mm [0.98 in] surface	> 4 mm [> 0.16 in]	0.0241 m^3 [0.85 ft^3]	19	788 [22.3]

*Doctor, S.R., G.J. Schuster, and F.A. Simonen. NUREG/CP–0166, Vol. 1, "Fabrication Flaws in Reactor Pressure Vessels." Proceedings of the Twenty-Sixth Water Reactor Safety Information Meeting, Bethesda, Maryland, October 26–28, 1998. Washington, DC: NRC. pp. 85–103. June 1999.

†Schuster, G.J., S.R. Doctor, S.L. Crawford, and A.F. Pardini. NUREG/CR–6471, Vol. 3, "Characterization of Flaws in U.S. Reactor Vessels—Density and Distribution of Flaw Indications in Shoreham Vessel." Washington, DC: NRC. November 1999.

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