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MARVEL Nuclear Fuel Performance

MARVEL Program Technology Review

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Jordan A. Evans, Ph.D.

Nuclear Materials Scientist

Department of Irradiated Fuels and Materials

Idaho National Laboratory

Overview

1. Fuel Performance Assessment Status Update
2. Background – Historical Experiences with U-ZrH_x Fuel
3. Fuel Element Properties, Effects, and Relationships
4. Results - Marvel Fuel Element Performance Analysis Under Extreme Conditions
5. Conclusions and Future Work

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MARVEL Fuel Performance Assessment Status Update

1. Establish fuel element properties, behavior, and quantitative thermophysical relationships (relevant reports, NUREGs, publications, etc.)
   - ***Complete**, Spring 2022*
2. Fuel performance analysis during beyond design basis accident (BDBA)
   - Determine fuel element hermeticity, stability/predictability of geometry, and mechanical integrity under equilibrium (steady state) BDBA conditions
   - For all variables/uncertainties, assume least favorable conditions for conservatism
   - ***Complete**, Spring 2022*
3. Produce fuel performance assessment report of Steps #1 and #2 (per NUREG-1537)
   - ***Rev.1 Complete**, Summer 2022*
4. Develop capability to perform high fidelity fuel performance computational modeling of MARVEL fuel using BISON
   - ***Complete**, Summer 2022*
5. Simulate MARVEL fuel performance using BISON to check calculations from Step #2
   - ***Complete**, Summer 2022*

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MARVEL Fuel Qualification Strategy

MARVEL fuel qualification strategy follows **NUREG-1537** (*Guidelines for Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*) guidelines

- Describe history of fuel type (previous tests, qualifications, etc.)
- Describe geometries, composition, thermophysical properties, etc.
- Describe irradiation performance relationships
- Determine operational limits
- Assess risk of reaching performance limits
    - Our strategy -- analyze fuel performance under the most extreme "bounding case" conditions
- *Information and analyses "should be current"*

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MARVEL Fuel Selection and Background

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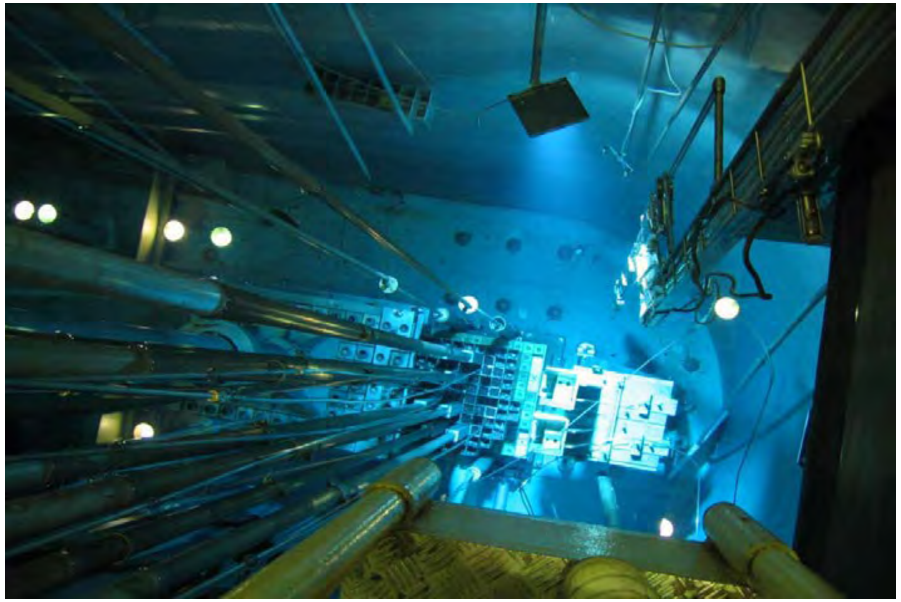
Background – MARVEL Fuel Selection

- Selecting materials that are already known/developed/licensed and commercially available facilitates rapid design, assessment, and construction of the MARVEL reactor
- The **304 SS-clad U-ZrH$_x$ fuel system** has been selected for MARVEL (aka “TRIGA” fuel)
- Fuel will be fabricated and purchased from TRIGA International
  - Same materials, same fabrication processes, etc.
- Qualified and licensed by US NRC for (and still used in) TRIGA reactors since the 1950s
- Used previously in several NASA reactors



[1] *History, Development and Future of TRIGA Research Reactors*, International Atomic Energy Agency, Vienna, 2016.

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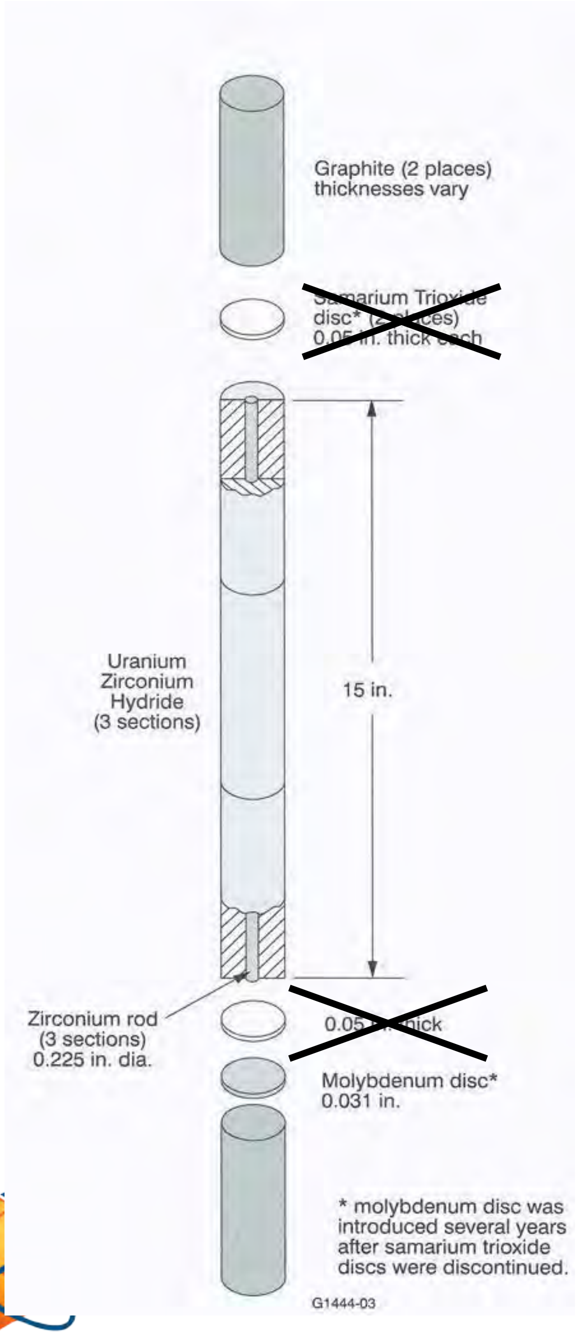
Background – The MARVEL Fuel Element

- (**5**x) Annular U-$ZrH_{1.6}$ Fuel Pellets
  - 30 wt% uranium
  - 19.75% enrichment
  - *No erbium or samarium BAs*
- (2x) Graphite neutron reflectors
- (1x) Mo diffusion barrier disc
- Axial zirconium rod
- 304 SS cladding
- Top and bottom 304 SS end plugs
- Ambient air gas gap

- Fuel meat contains fissile and neutron-moderating species
- Excellent chemical stability in water (TRIGA reactor coolant… we'll discuss NaK in a moment…)
- High fission product retentivity and high-temperature stability
- Fuel meat and cladding retain integrity under large reactivity insertions and frequent power cycling

[1] *History, Development and Future of TRIGA Research Reactors*, International Atomic Energy Agency, Vienna, 2016.



NUREG-1282: Fuel Limits in TRIGA Reactors

- TRIGA fuel limits described in **NUREG-1282** [1]
  - The safety limits of the standard TRIGA element are dominated by overpressurization of gas inside the element (*vide infra*)
  - For rapid transients (ex. reactor pulses), fuel meat temp of 1150 °C precludes loss of cladding integrity
  - For extended transients (cladding temp reaches steady state), fuel meat temp of **950 °C** precludes loss of cladding integrity
- *Note*: Recommended temperature limits lower than 950 °C can be found in literature, but those are for different systems/conditions (ex. PWRs, higher fission rates, coolant pressure, etc.) [2,3]

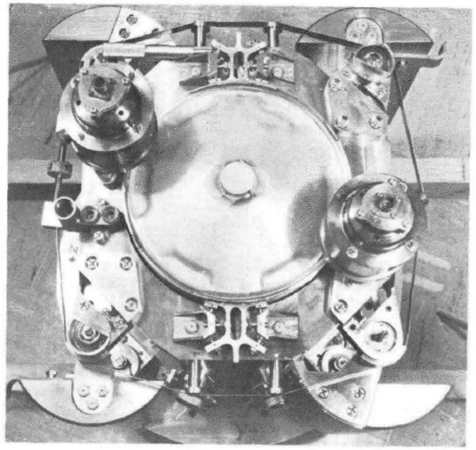
[1] *Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors*, NUREG-1282, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, (1987).
[2] D. Olander, E. Greenspan, H.D. Garkisch, B. Petrovic, *Uranium-zirconium hydride fuel properties*, Nucl Eng Des 239(8) (2009) 1406-1424.
[3] D.R. Olander, M. Ng, *Hydride fuel behavior in LWRs*, J Nucl Mater 346(2-3) (2005) 98-108.

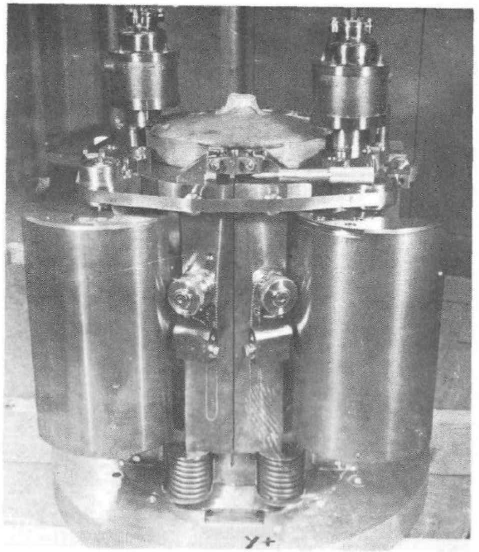
Background – Space Nuclear Auxiliary Power (SNAP) Program

- NASA's SNAP program developed nuclear reactors and RTGs for space missions in the 1950s and 1960s
- Post-irradiation examination following the SNAP-10A "extended BDBA test" (conditions held for 10,000 hours) showed no evidence of incipient failure

	MARVEL	SNAP-10A
Fuel Type	U-ZrH	U-ZrH
wt% U	30	10
Enrichment (%)	19.75	93
Gas gap	Air (1 atm)	He (0.1 atm)
Cladding	304 SS	Hastelloy-N [a]
# Fuel Elements	36	37
Coolant	NaK	NaK
Fuel Temp (°C)	565	585
Power (kW_{th})	85	34
Control	BeO + poison	Be wedges

(a) Included a thin film (internal, 2-4 mils thick) of Solaramic (glassy coating BA)





[1] H. Dieckamp, *Nuclear Space Power Systems*, Atomics International, Canoga Park, California, 1967.

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MARVEL Fuel Properties and Performance

A Few Fuel Performance Phenomena to Consider

- Hydrogen redistribution and dissociation (fuel)
- Internal gas pressure
  - From as-fabricated air in gas gap, fission gas, hydrogen
- Coolant-cladding corrosion
- Oxygen interactions (with fission products, with graphite, coolant impurity)
- Geometry changes (Zr rod, fuel meat, cladding, and graphite reflectors)
  - Thermal expansion (all), fission/void growth (fuel), hydrogen expansion (fuel), radiation-induced swelling (all), radiation-enhanced creep (all)
- Radiation effects
  - Hardening, embrittlement, etc.
- Fuel-cladding mechanical interactions (FCMI)
- Fuel-cladding chemical interactions (FCCI)
- Hydrogen embrittlement (cladding)

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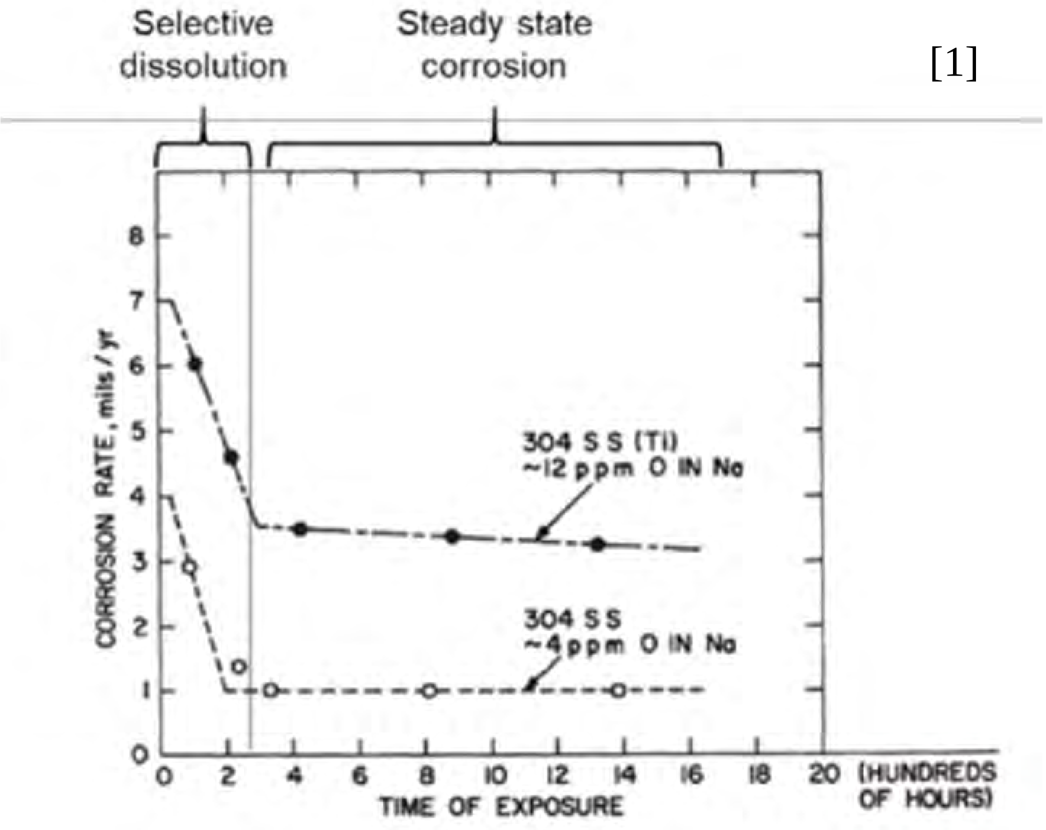
MARVEL Cladding Compatibility with Hot NaK Coolant

- 304L SS was used as cladding in EBR program
- Corrosion rates are strongly dependent upon salt impurity content (as well as temperature and flow rate)
- Corrosion is characterized by rapid, brief, selective dissolution followed by slower steady state corrosion
- For conservatism, the forthcoming analysis assumes a constant cladding corrosion rate of **4 mils/yr** (high O impurity)
  - 1 mil = 25.4 μm

[1]



Corrosion rates of 304 SS in high-velocity sodium at 760 °C

[1] M. Romedenne, B. Pint, *Corrosion in Sodium Fast Reactors*, ORNL/ SPR-2020/1580, Oak Ridge National Laboratory, 2021.

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MARVEL Fuel Meat Compatibility with Hot NaK Coolant

- High temp ZrH, U, and δ-U-ZrH compatibility tests in NaK were performed in the 1950s during the SNAP program
- Fuels *were irradiated*, then submerged in hot NaK
- No physical changes or release of radioactive species were detected in NaK up to ~540 °C
- Above 540 °C, a visibly apparent black/brown surface film manifests on the fuel meat
- Far above 540 °C, surface dissolution occurs on the order of a few mils/month

[1] J. Vetrano, *Delta-Phase Zirconium Hydride as a Solid Moderator*, BMI-1243, Battelle Memorial Institute, Columbus, Ohio, 1957.
[2] J. Katz, E. Rabinowitch, *The Chemistry of Uranium*, p. 177, Division VIII, Vol. 5, National Nuclear Energy Series, McGraw-Hill Book Company, Inc., New York, 1951.
[3] T.B. Douglas, *A Cryoscopic Study of the Solubility of Uranium in Liquid Sodium at 97.8-Degrees-C*, J Res Nat Bur Stand 52(5) (1954) 223-226.
[4] J. Stang, E. Simons, J. DeMastry, J. Genco, *Compatibility of Liquid and Vapor Alkali Metals with Construction Materials*, DMIC Report 227, Battelle Memorial Institute. Defense Metals Information Center, Columbus, Ohio, 1966.

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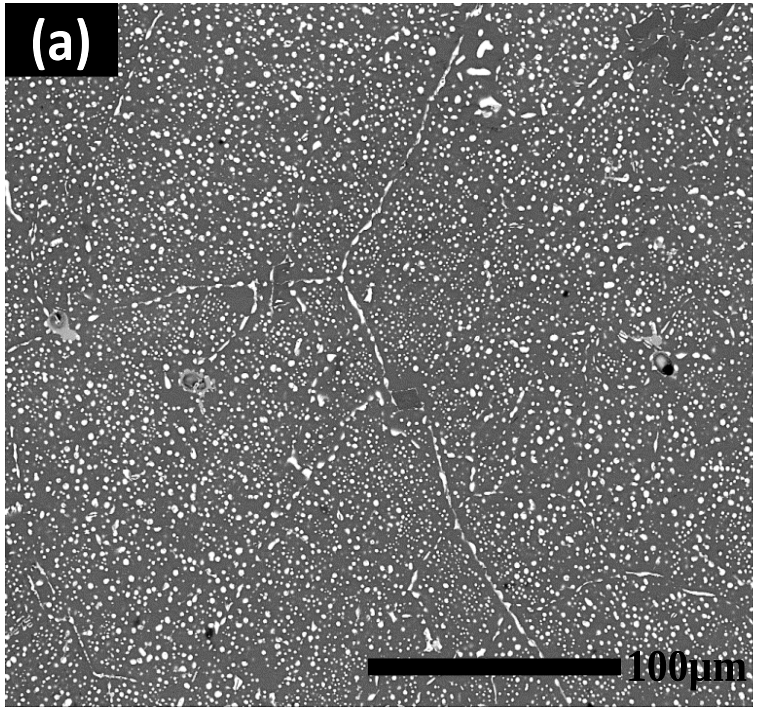
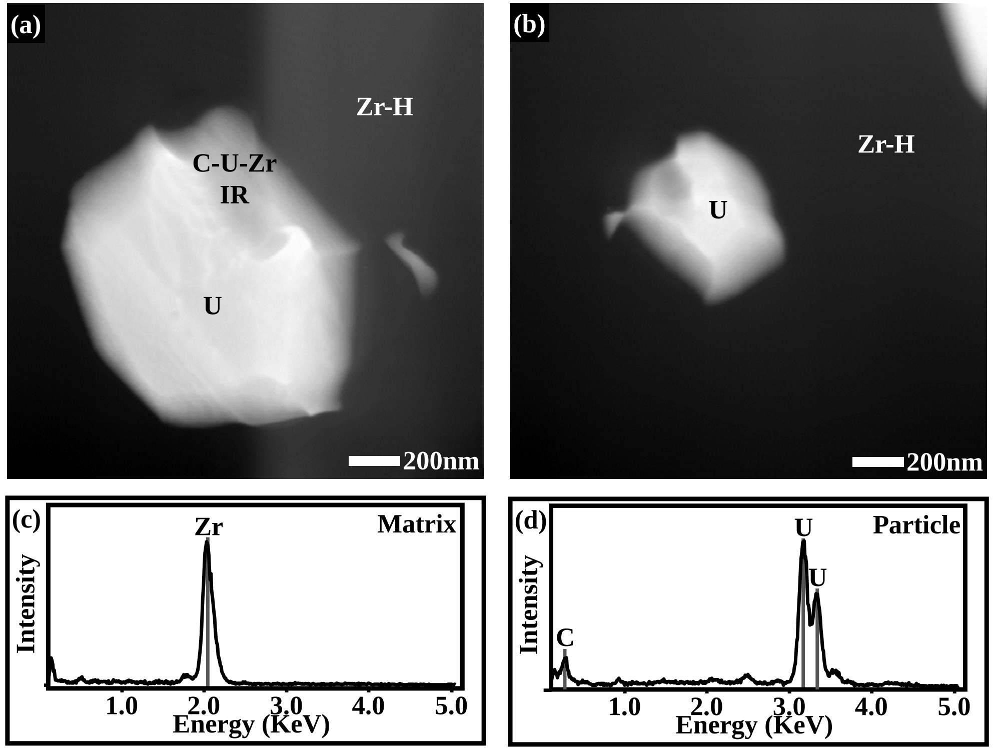
MARVEL Fuel Meat Microstructure

- Microstructure of 30 wt% U-$ZrH_{1.6}$ confirmed
- Uranium micro-particles embedded in ZrH matrix





[1] D. Keiser, Jr., E. Perez, J. Jue, F. Rice, E. Woolstenhulme, *Microstructural Characterization of Uranium Zirconium Hydride Fuel in an As-Fabricated TRIGA Fuel Element*, J Nucl Mater. *In Review*.

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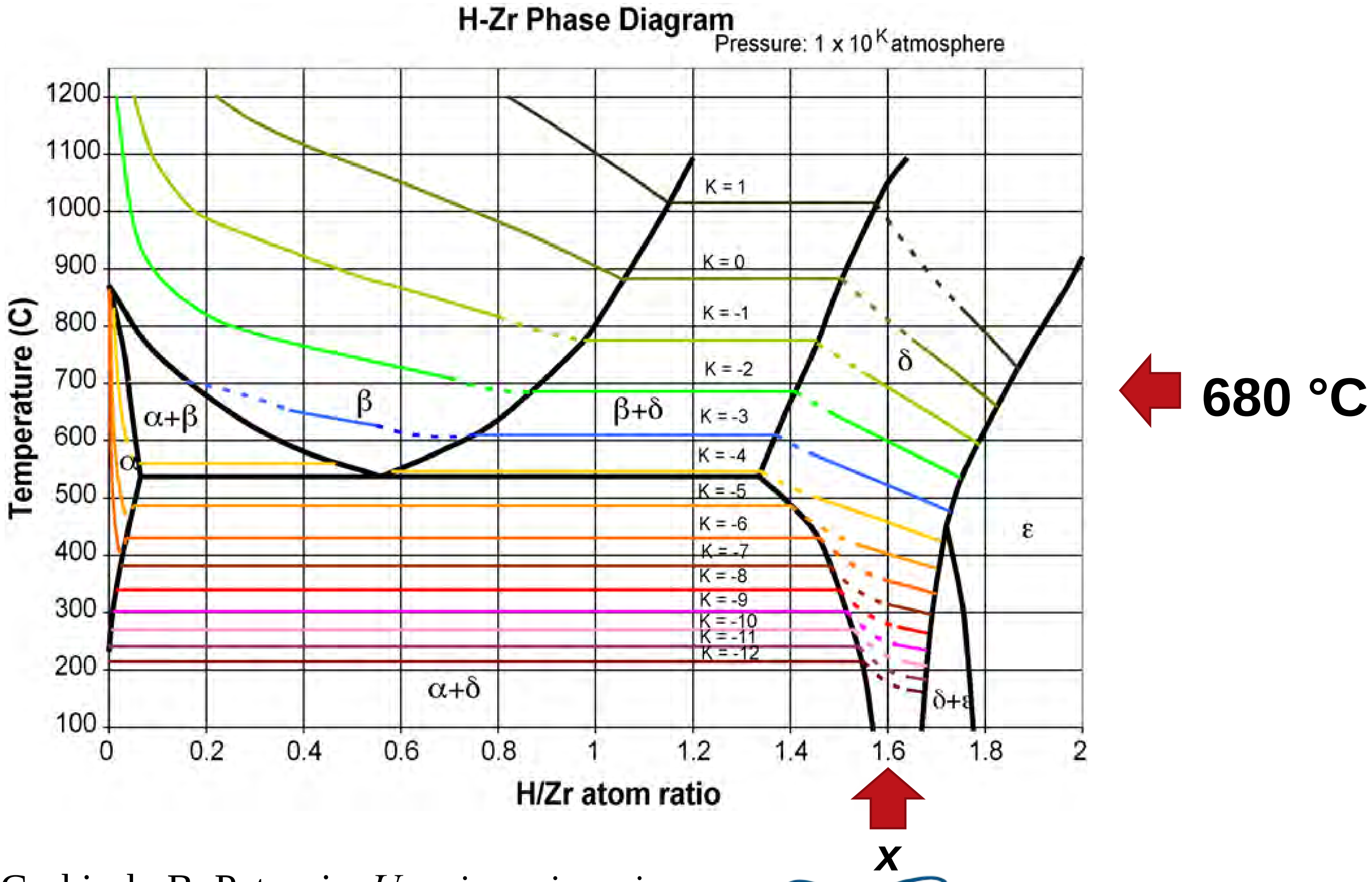
U-ZrH$_x$ Thermophysical Properties

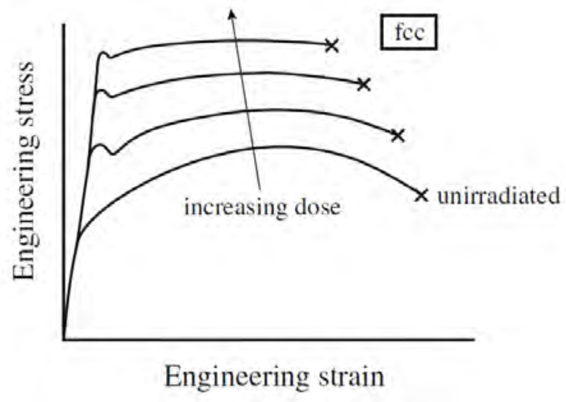
- Peak U-ZrH$_{1.6}$ fuel temperature during BDBA **680 °C** (953 K)
- Matrix remains δ-phase
  - Geometry is stable and predictable
- Design limits are based on *cladding stability*

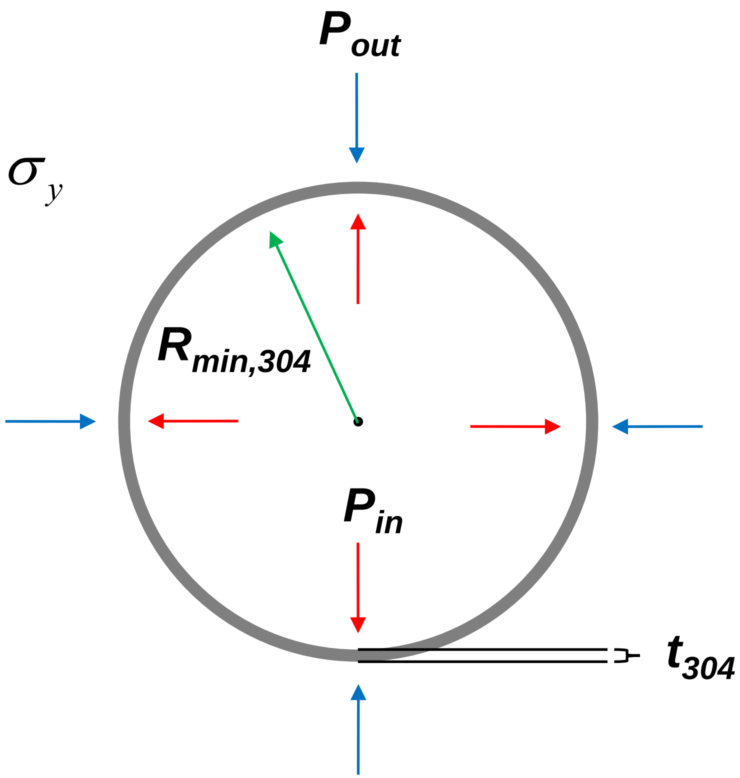


[1] D. Olander, E. Greenspan, H.D. Garkisch, B. Petrovic, *Uranium-zirconium hydride fuel properties*, Nucl Eng Des 239(8) (2009) 1406-1424.

MARVEL Fuel Element Limits – Cladding Stresses

- We define the MARVEL fuel design limit as the conditions in which unrecoverable structural deformation occurs to the cladding
- This occurs when the hoop stress reaches the material's yield stress: $\sigma_{circ} = \sigma_y$
- Hoop stress calculated using Barlow's formula
- Each variable is a function of burnup/radiation damage, temperature, temperature distribution, corrosion, etc.





$$\Delta P = \frac{\sigma_{circ} t_{304}}{R_{min,304}}$$

Where

$\Delta P = (P_{in} - P_{out})$ = differential pressure (MPa)

σ_{circ} = circumferential (hoop) stress (MPa)
t_{304} = average thickness of the cladding
$R_{min,304}$ = average internal radius of the cladding (same units as t_{304})
$P_{inside} = P_g$, only if there is no FCMI (MPa)
$P_{inside} = P_g + \sigma_{r,FCMI}$, where there is FCMI (MPa)

$$P_g = \sum_i P_i$$

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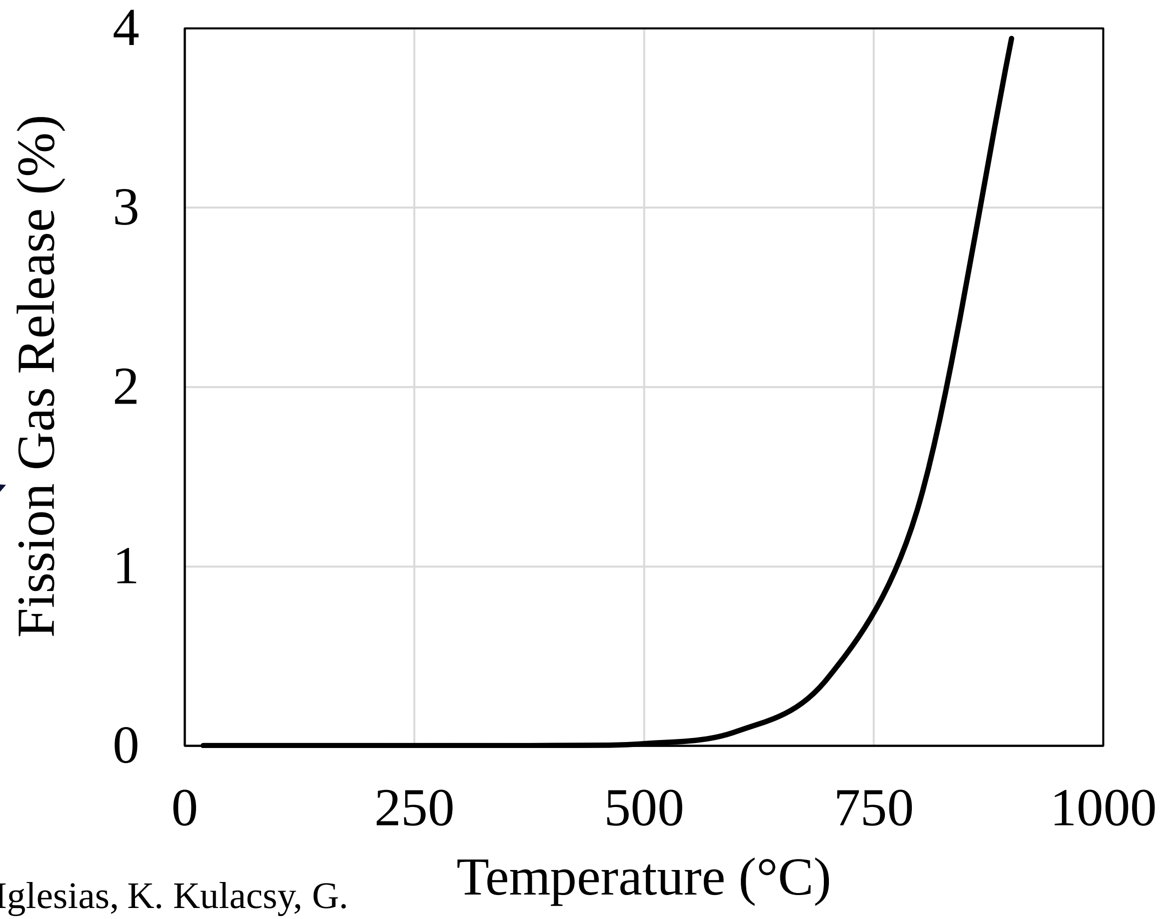
Fuel Element Internal Gas Gap Pressure

- Air → $PV = nRT$
- Fission gas
  - Produced → [1] $\Omega_{FGP} \approx \dfrac{0.3FV_{fm}}{N_A}$
  - Released

[2] $\Gamma_{FGR} = 1.5\cdot10^{-5} + 3600e^{\frac{-1.34\cdot10^{-4}}{T_K}}$

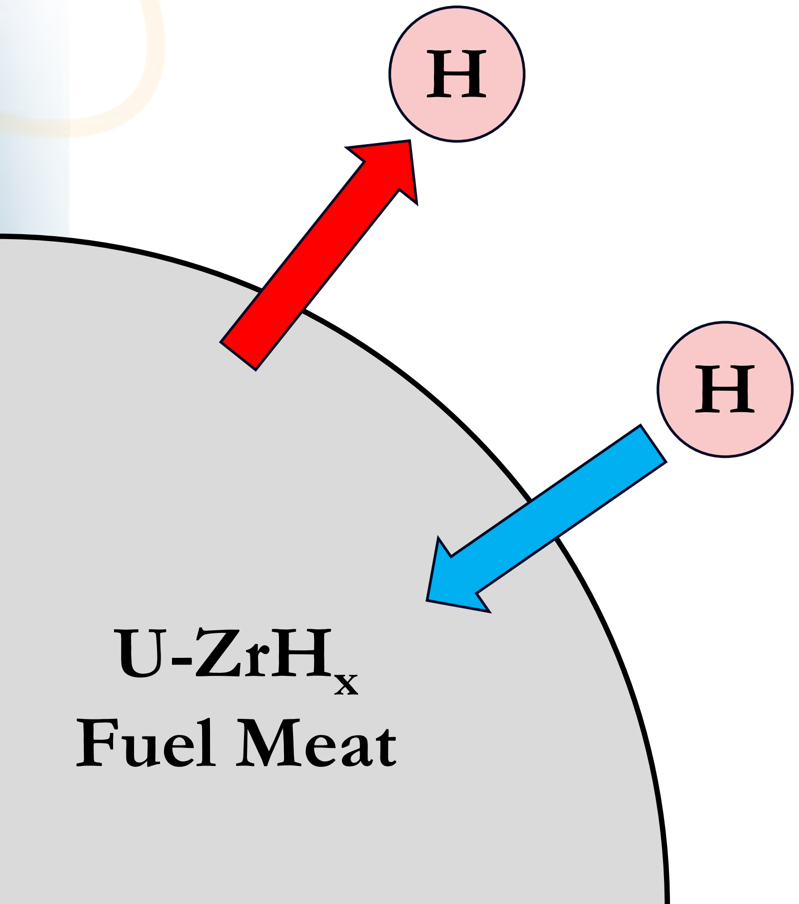
- Hydrogen Dissociation…



[1] M. Tonks, D. Andersson, R. Devanathan, R. Dubourg, A. El-Azab, M. Freyss, F. Iglesias, K. Kulacsy, G. Pastore, S.R. Phillpot, M. Welland, *Unit mechanisms of fission gas release: Current understanding and future needs*, J Nucl Mater 504 (2018) 300-317.

[2] M.T. Simnad, F.C. Foushee, G.B. West, *Fuel Elements for Pulsed TRIGA Research Reactors*, Nucl Technol 28(1) (1976) 31-56.

Hydrogen Dissociation



- Hydrogen is constantly dissociating (escaping) and re-entering the fuel meat
- Dissociated hydrogen enters the gas gap, *increasing internal pressure*
- Steady state equilibrium occurs when the hydrogen escape rate equals the hydrogen reabsorption rate
- Hydrogen dissociation dynamics are defined by
  - Temperature
  - Hydrogen concentration in the gas gap
    - i.e., hydrogen gas pressure
  - Hydrogen concentration at fuel meat surface
    - i.e., H/Zr ratio (x)

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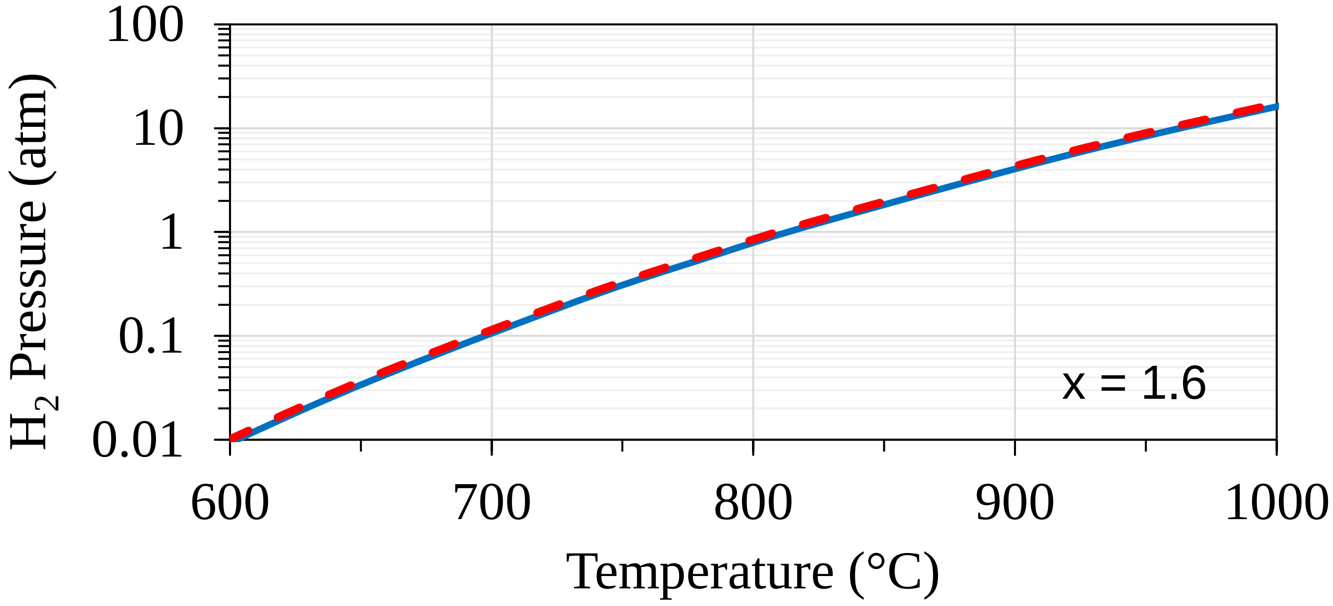
Hydrogen Dissociation Equilibria

[1]
$$\log(P) = K_1 + \frac{K_2 \cdot 10^3}{T_K}$$

$$\ln(P) = 2\ln\left(\frac{x}{2-x}\right) + 8.01 + 5.21x - \frac{20700}{T_K}$$
[2]

P = hydrogen gas equilibrium dissociation partial pressure (atm)
$\boldsymbol{T_K}$ = Temperature of the U-ZrH$_x$ fuel pellet's outer surface (K)
x = H/Zr ratio

$$K_1 = -3.8415 + 38.6433x - 34.2639x^2 + 9.2821x^3$$
$$K_2 = -31.2982 + 23.5741x - 6.0280x^2$$



[1] M.T. Simnad, F.C. Foushee, G.B. West, Fuel Elements for Pulsed TRIGA Research Reactors, Nucl Technol 28(1) (1976) 31-56.
[2] W.E. Wang, D.R. Olander, Thermodynamics of the Zr-H system, J Am Ceram Soc 78(12) (1995) 3323-3328.

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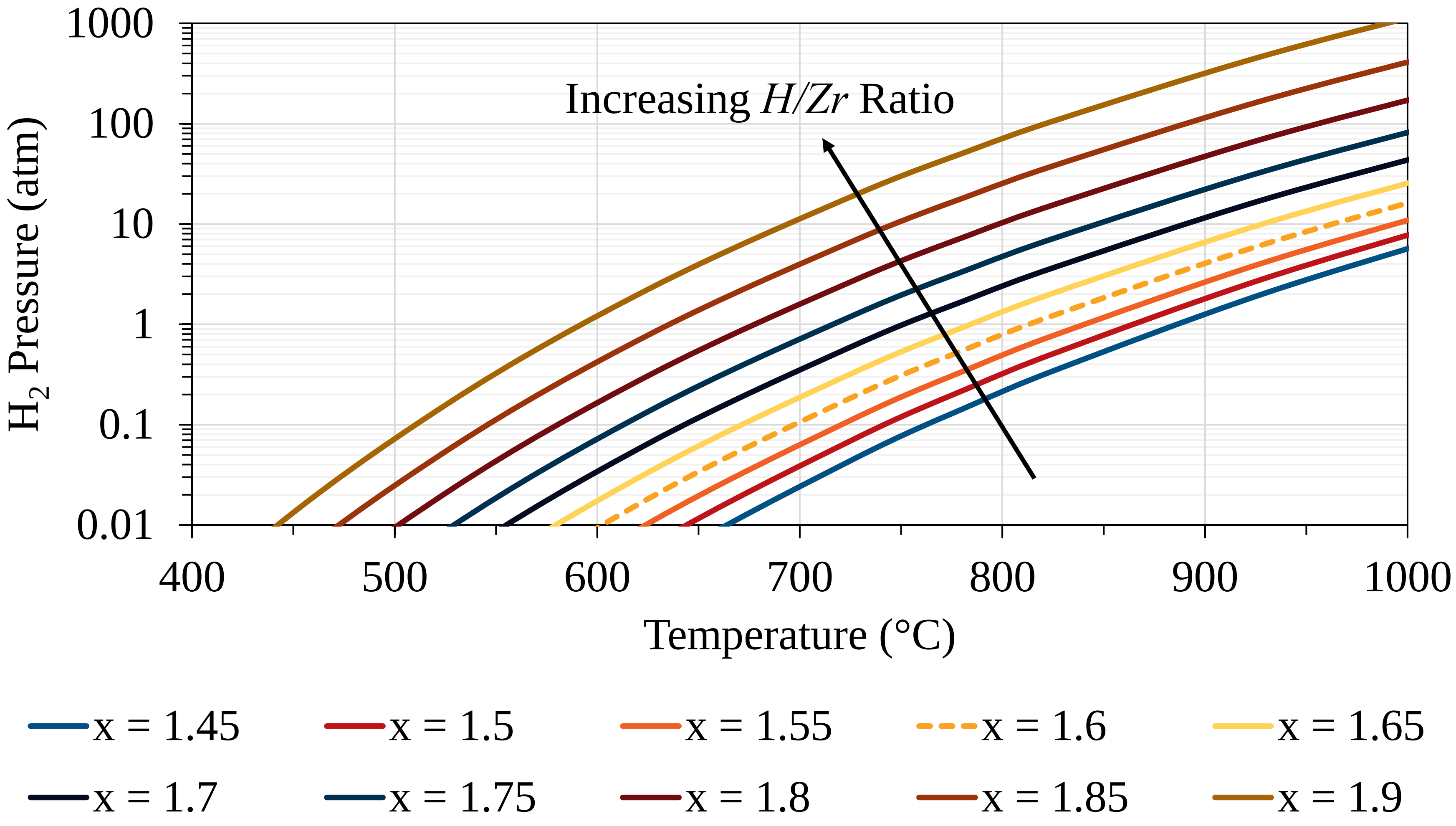
Hydrogen Dissociation Equilibria

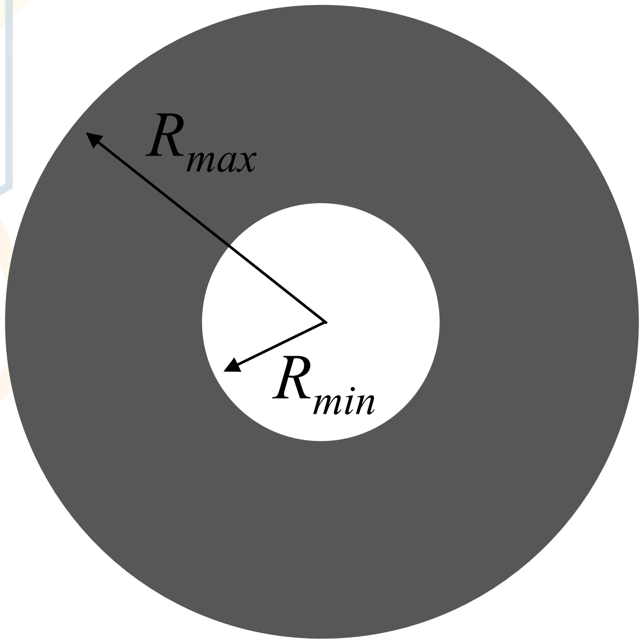
For Marvel,

H/Zr_{nom} = 1.6
T_{surf} = 664 °C

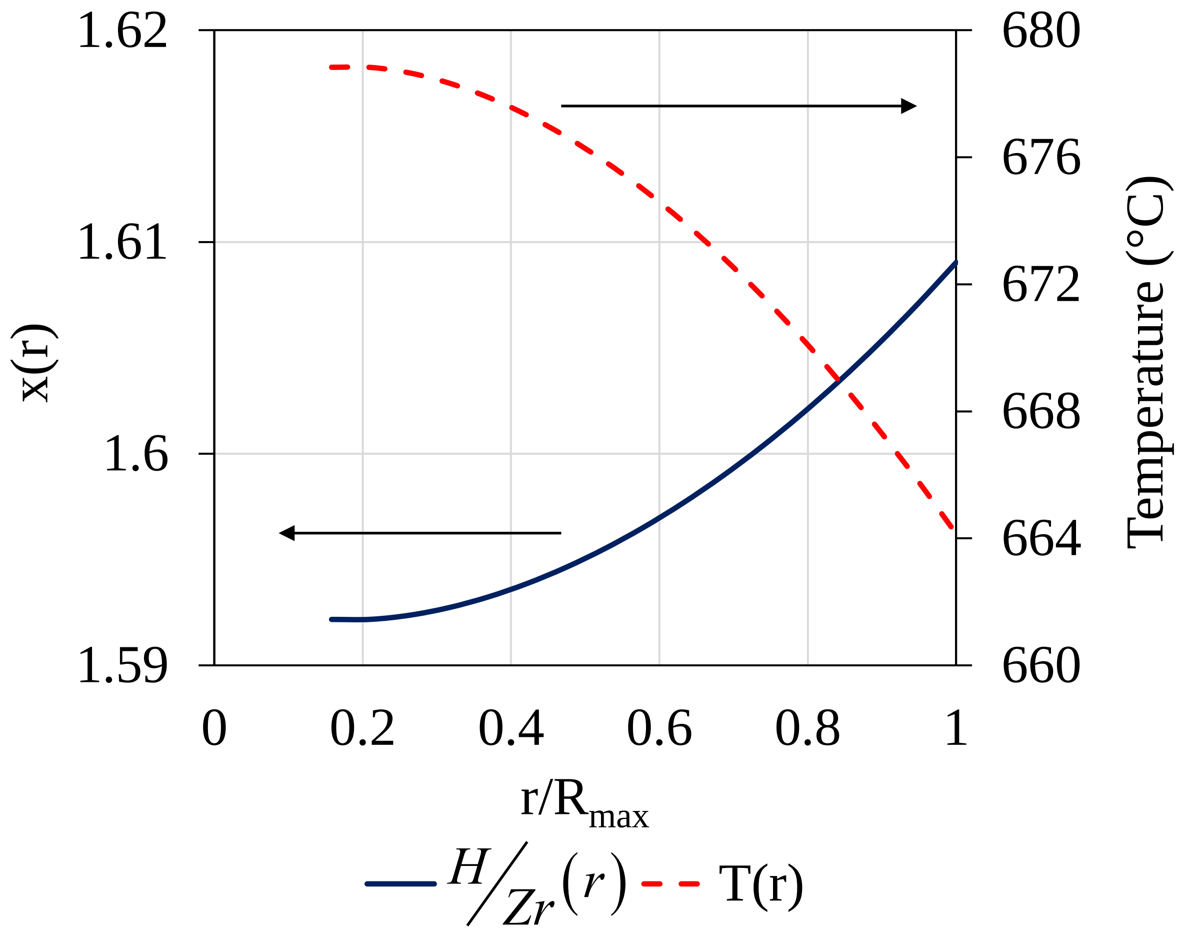
But the hydrogen dissociation pressure is dependent upon the temperature and H/Zr ratio ***at the fuel meat surface***…



Radial Hydrogen Redistribution – Steady State



$$x(r) = Ae^{T_Q / T_K(r)}$$



$$T(r) = T_{peak} - \gamma r^2\left(1 - \frac{R_{\min}{}^2}{r^2}\right) + 2\gamma R_{\min}{}^2 \ln\left(\frac{r}{R_{\min}}\right)$$

$$\gamma = \frac{T_{peak} - T_s}{R_{\max}{}^2\left[1 - \left(\frac{R_{\min}}{R_{\max}}\right)^2 - \left(\frac{R_{\min}}{R_{\max}}\right)^2 \ln\left(\frac{R_{\max}}{R_{\min}}\right)^2\right]}$$

Hydrogen Redistribution – How Long Does it Take?

Exact solution requires solving the multi-dimensional non-linear partial differential equation…

$$\frac{\partial}{\partial t}x(r,t) = D_s \frac{\rho_{Zr}}{M_{Zr}} \frac{\partial}{\partial r}\left(\frac{\partial}{\partial r}x(r,t) + \frac{T_Q}{T_K(r)} \frac{x(r,t)}{T_K(r)} \frac{dT_K(r)}{dr} \right)$$

It can be solved numerically via a finite difference method…

$$x(i,t+\Delta t) = x(i,t) + \left[J(i,t)S(i) - J(i+1,t)S(i+1) \right]\Delta t$$

D_s = diffusion coefficient
ρ_{Zr} = mass-density of zirconium
M_{Zr} = atomic weight of Zr
T_Q = heat of transport of H in ZrH_x
$S(i)$ = area of the inner surface of the i^{th} shell (cm^2)
$x(i,0) = x_{nom}$
$J(R_{min},t) = J(R_{max},t) = 0$

First order approximation…

$$t_{radial} \approx \frac{\pi\left(R_{\max}{}^2 - R_{\min}{}^2\right)}{2D_s} \cdot \frac{T_Q}{T_{peak} - T_s}$$

$$t_{axial} \approx \frac{L^2}{2D_s} \cdot \frac{T_Q}{T_{hot} - T_{cold}}$$

Under steady state MARVEL conditions…

$t_{radial} \approx$ 2.3 months
$t_{axial} \approx$ 16.6 years

BDBA Analysis Input Parameters

- Fission rate = $2.363 \cdot 10^{11}$ fis·cm^{-3}·s^{-1}
- 304 SS damage rate = $5.01 \cdot 10^{-9}$ dpa·s^{-1}
- Coolant-cladding corrosion rate = 4 mils·yr^{-1} (*vide infra)*
- Peak fuel meat temp during BDBA = 680 °C
- Fuel meat surface temp during BDBA = 664 °C
  - Assume cladding and graphite have time to reach equilibrium
- Effective full-power years (EFPY) = 2
  - Burnup = 2.5 MWd·kgU^{-1}
  - Uranium consumed = 0.27%
  - U^{235} consumed = 1.39%
- Fuel meat oxygen impurity content = 1400 ppm$_{wt}$
  - In the form of ZrO_2 (in fresh fuel)
  - Per ASTM B349/B349M – 16

Peak values chosen for conservatism

MARVEL Fuel Performance– Results

Conservative Scenario

Parameter	BOL	EOL
BU ($MWd \cdot kgU^{-1}$)	0	2.5
U Consumed (%)	0	0.27
U^{235} Consumed (%)	0	1.39
Peak 304 SS Dmg (dpa)	0	0.32
304 SS Corrosion (mm)	0	0.2
Fuel T_{peak} (°C)	680	680
BDBA Peak Hoop Stress (MPa)	5	9
304 SS σ_y (MPa)	130	190

- Strains (for all materials) are dominated by thermal expansion
- Gas gap gets smaller at higher temps/burnups *but remains open*
  - No FCMI
- Peak hoop stress is much less than yield

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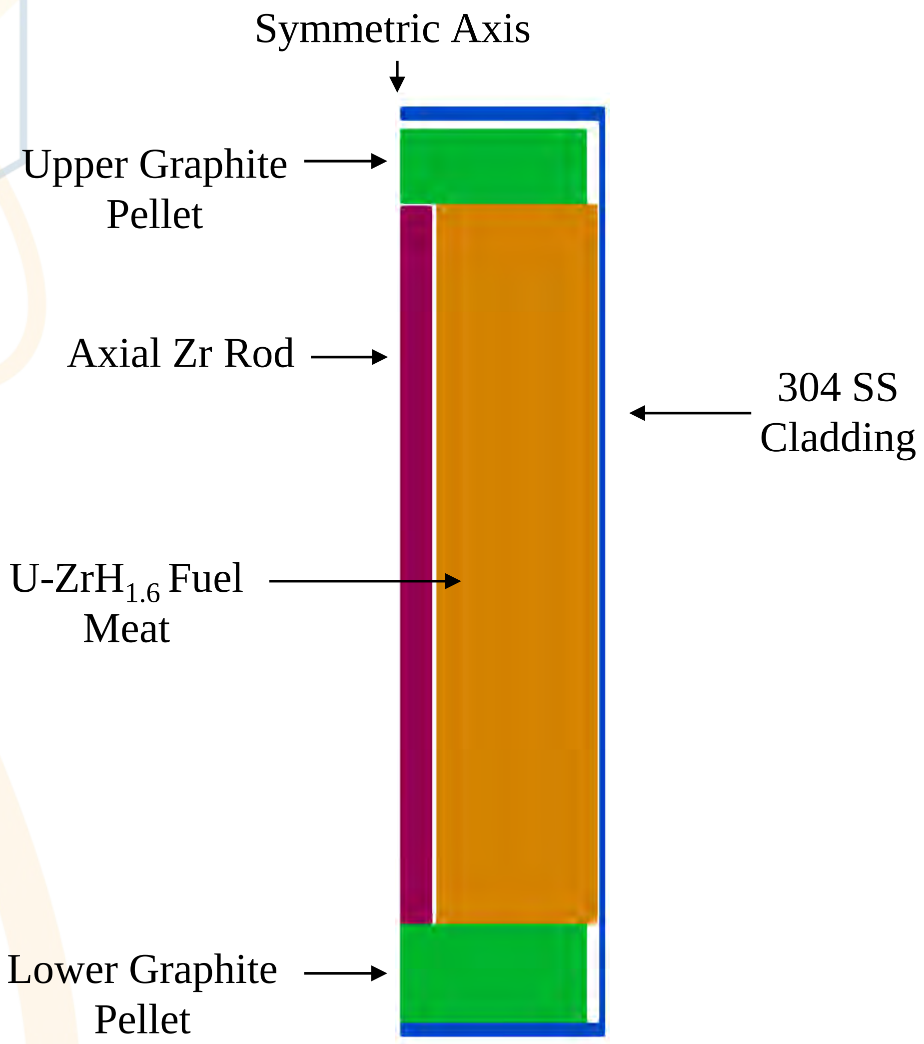
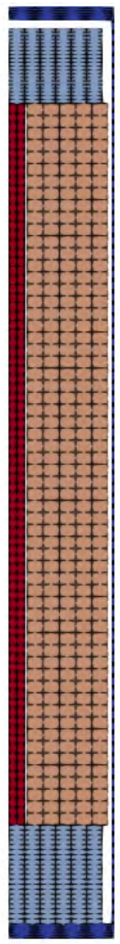
MARVEL Fuel Performance Verification in BISON





BISON simulation of hoop stress during BDBA is in good agreement with hand calculations

Conclusions (and Future Work)

- Fuel system qualification is based on NUREG-1537 guidance
- Risk of damage to the fuel element under bounding scenario conditions has been analyzed
- This fuel system is qualified for use in the MARVEL reactor because:
  - MARVEL fuel element behavior is understood
  - Maintains structural integrity, geometric stability, and behavior is stable *and* predictable under bounding BDBA conditions
  - Bounding BDBA conditions (burnup, radiation damage, temperatures, and pressures, etc.) are well below the damage limits of the fuel element

- Publications highlighting results and new BISON capability

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