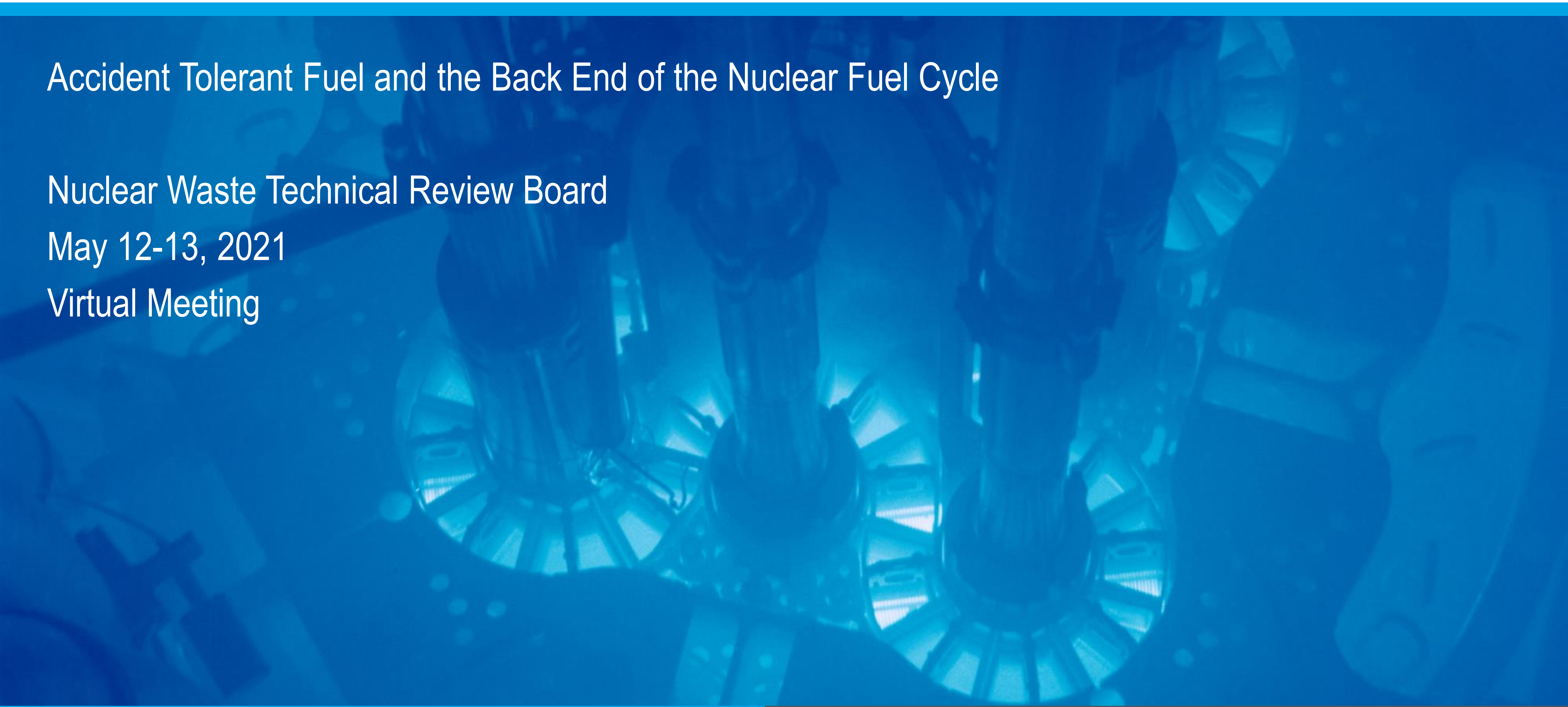


Accident Tolerant Fuel and the Back End of the Nuclear Fuel Cycle

Nuclear Waste Technical Review Board
May 12-13, 2021
Virtual Meeting

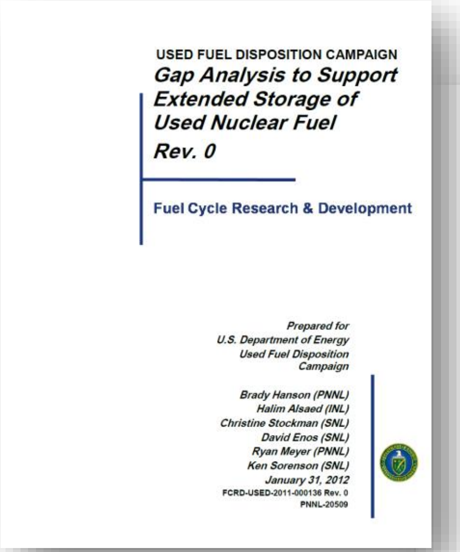
Accident Tolerant and Advanced Nuclear Fuels: Back End Key Findings and Plans

Sylvia Saltzstein, Philip Honnold, Geoff Freeze: Sandia National Laboratories
Rose Montgomery: Oak Ridge National Labs
Brady Hanson: Pacific Northwest National Laboratory
Mike Billone: Argonne National Labs
Ned Larson: Department of Energy: Nuclear Energy

Sandia National Laboratories is a multimission laboratory managed and operated by National Technology & Engineering Solutions of Sandia, LLC, a wholly owned subsidiary of Honeywell International Inc., for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-NA0003525. SAND2021_5169 PE

Planning Long-Term

The DOE R&D is driven by peer-reviewed Gap Analyses



USED FUEL DISPOSITION CAMPAIGN

Gap Analysis to Support Extended Storage of Used Nuclear Fuel

Rev. 0

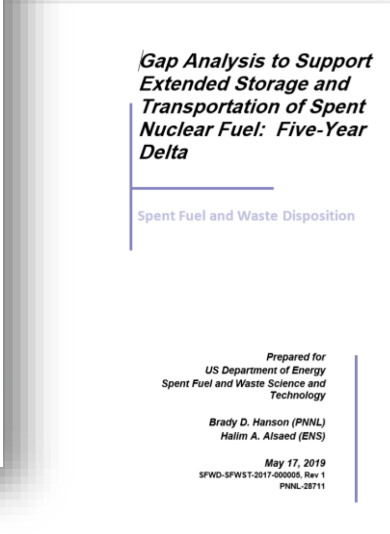
Fuel Cycle Research & Development

Prepared for
U.S. Department of Energy
Used Fuel Disposition
Campaign

Brady Hanson (PNNL)
Halim Alsaed (INL)
Christine Stockman (SNL)
David Enos (SNL)
Ryan Meyer (PNNL)
Ken Sorenson (SNL)
January 31, 2012
FCRD-USED-2011-000136 Rev. 0
PNNL-20509

2012 Gap Analysis

- Established the structure and process
- Identified Technical Categories
- Ranked R&D Gaps for R&D Funding priority.



Gap Analysis to Support Extended Storage and Transportation of Spent Nuclear Fuel: Five-Year Delta

Spent Fuel and Waste Disposition

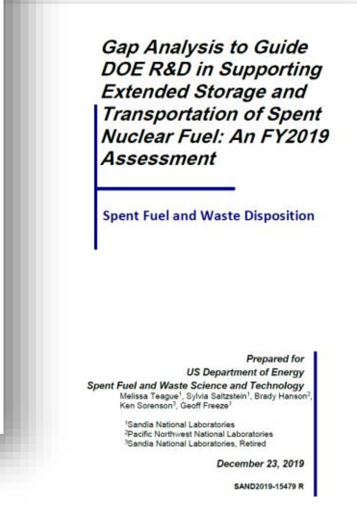
Prepared for
US Department of Energy
Spent Fuel and Waste Science and
Technology

Brady D. Hanson (PNNL)
Halim A. Alsaed (ENS)

May 17, 2019
SFWD-SFWST-2017-000005, Rev 1
PNNL-28711

2017 Five-Year Delta report

- Updated the 2014 Gap Analysis
- Covers R&D results through FY17



Gap Analysis to Guide DOE R&D in Supporting Extended Storage and Transportation of Spent Nuclear Fuel: An FY2019 Assessment

Spent Fuel and Waste Disposition

Prepared for
US Department of Energy
Spent Fuel and Waste Science and Technology
Melissa Teague[1], Sylvia Saltzstein[1], Brady Hanson[2],
Ken Sorenson[3], Geoff Freeze[1]

[1] Sandia National Laboratories
[2] Pacific Northwest National Laboratories
[3] Sandia National Laboratories, Retired

December 23, 2019

SAND2019-15479 R

FY2019 Assessment report

- Adds R&D results from FY18 & 19
- Main priorities remain the same. Some rankings have changed based on recent R&D results

Current R&D Priorities

Priority 1

- Thermal Profiles
- Stress Profiles
- Welded Canister –
  Atmospheric Corrosion
  (Priority increased)

Priority 2

- Drying Issues

Priority 3

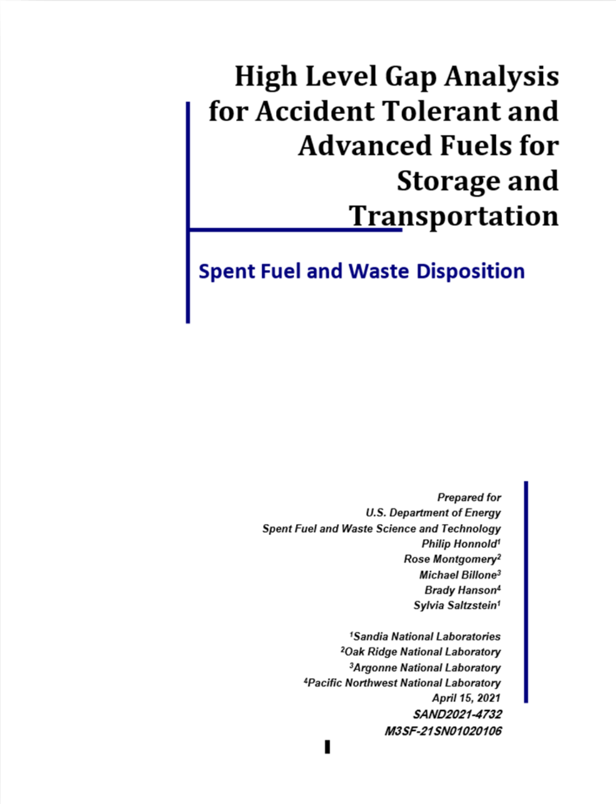
- External Monitoring
- Cladding – H_2 Effects
- Consequence of Canister Failure
- Fuel Transfer Options

DOE ranking and priorities may differ from those of industry and NRC to follow an approach that includes maintaining options for final disposition (e.g., reprocessing, geologic disposal, etc.)

The DOE:NE Accident Tolerant and Advanced Fuels Gap Analysis

- Fuels that are in the typical PWR or BWR rod-in-assembly configuration can be **tested using the same test plan** used for the High Burnup Demo Sibling Pins
- **The current storage and transportation hardware can accommodate** the above fuels with modifications for their increased weight, radioactivity, and temperature.

- **Triso and other designs will need a different test plan** to quantify their mechanical properties, if further testing is warranted.
- **Triso and other designs will not fit into the current storage and transportation container designs.** New containers will need to be designed, tested, and manufactured.



High Level Gap Analysis for Accident Tolerant and Advanced Fuels for Storage and Transportation

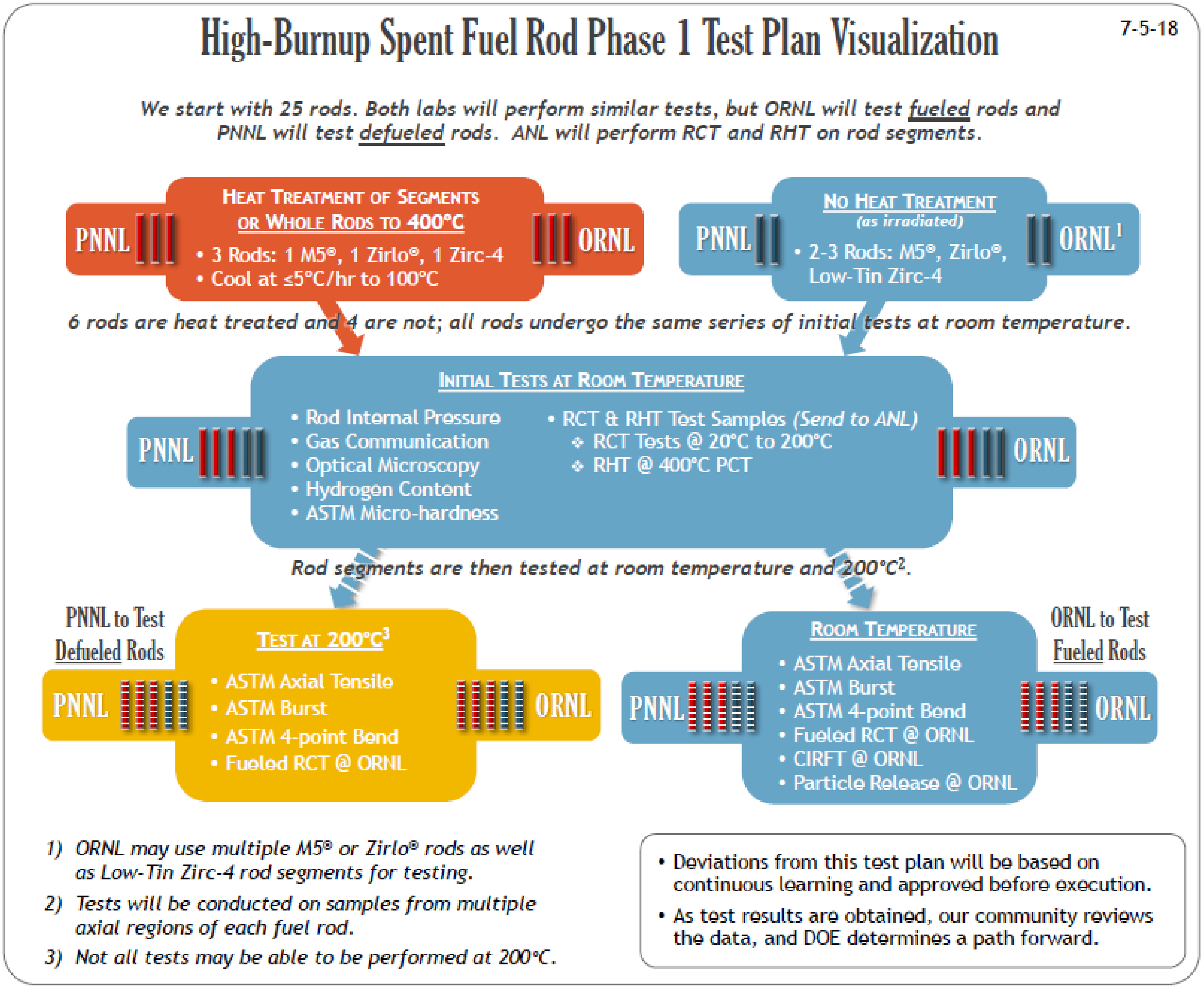
Spent Fuel and Waste Disposition

Prepared for
U.S. Department of Energy
Spent Fuel and Waste Science and Technology
Philip Honnold[1]
Rose Montgomery[2]
Michael Billone[3]
Brady Hanson[4]
Sylvia Saltzstein[1]

[1]Sandia National Laboratories
[2]Oak Ridge National Laboratory
[3]Argonne National Laboratory
[4]Pacific Northwest National Laboratory
April 15, 2021
SAND2021-4732
M3SF-21SN01020106

ATF/AF Fuels will Follow the Sister Pin Test

This allows direct comparison of mechanical properties to the currently used zircaloy-based cladding



High-Burnup Spent Fuel Rod Phase 1 Test Plan Visualization

7-5-18

We start with 25 rods. Both labs will perform similar tests, but ORNL will test fueled rods and PNNL will test defueled rods. ANL will perform RCT and RHT on rod segments.

PNNL | ORNL

HEAT TREATMENT OF SEGMENTS OR WHOLE RODS TO 400°C

- 3 Rods: 1 M5®, 1 Zirlo®, 1 Zirc-4
- Cool at ≤5°C/hr to 100°C

PNNL | ORNL[1]

NO HEAT TREATMENT *(as irradiated)*

- 2-3 Rods: M5®, Zirlo®, Low-Tin Zirc-4

6 rods are heat treated and 4 are not; all rods undergo the same series of initial tests at room temperature.

PNNL | ORNL

INITIAL TESTS AT ROOM TEMPERATURE

- Rod Internal Pressure
- Gas Communication
- Optical Microscopy
- Hydrogen Content
- ASTM Micro-hardness
- RCT & RHT Test Samples *(Send to ANL)*
  - RCT Tests @ 20°C to 200°C
  - RHT @ 400°C PCT

Rod segments are then tested at room temperature and 200°C[2].

PNNL to Test Defueled Rods

PNNL | ORNL

TEST AT 200°C[3]

- ASTM Axial Tensile
- ASTM Burst
- ASTM 4-point Bend
- Fueled RCT @ ORNL

ORNL to Test Fueled Rods

PNNL | ORNL

ROOM TEMPERATURE

- ASTM Axial Tensile
- ASTM Burst
- ASTM 4-point Bend
- Fueled RCT @ ORNL
- CIRFT @ ORNL
- Particle Release @ ORNL

1) *ORNL may use multiple M5® or Zirlo® rods as well as Low-Tin Zirc-4 rod segments for testing.*
2) *Tests will be conducted on samples from multiple axial regions of each fuel rod.*
3) *Not all tests may be able to be performed at 200°C.*

- Deviations from this test plan will be based on continuous learning and approved before execution.
- As test results are obtained, our community reviews the data, and DOE determines a path forward.

For the purposes of a gap analysis, the proposed fuels to be investigated are categorized as:

- **Modifications to uranium oxide fuels** to alter grain size to reduce fuel swelling, fission gas release, and fuel fragmentation.
  - Chromia-doped pellets
  - Chromia/Alumina-doped pellets
- **Modified current and new cladding designs** provide additional protection against oxidation, hydrogen production/uptake, and to reduce fretting wear
  - Cr-coated zirconium-based cladding
  - FeCrAl cladding
  - Silicide carbide-based cladding
- **Fuels with higher density than uranium oxide** to facilitate higher burnup and higher power while reducing fuel temperatures.
  - Uranium metals (including alloys such as with Mo or Pu/Zr)
  - Uranium nitrides
  - Uranium silicides
- **Fuels (UO_2-based or other) with enrichment 5-20% (HALEU)** to increase burnup, cycle length, and power.

TRISO fuel is physically unique from the traditional pellets or slugs contained in cladding and therefore can't be tested like the traditional PWR or BWR fuel rods. However, it will have its own data gaps related to storage, transportation, and disposal. An alternate testing plan needs to be developed for unique fuels such as TRISO.

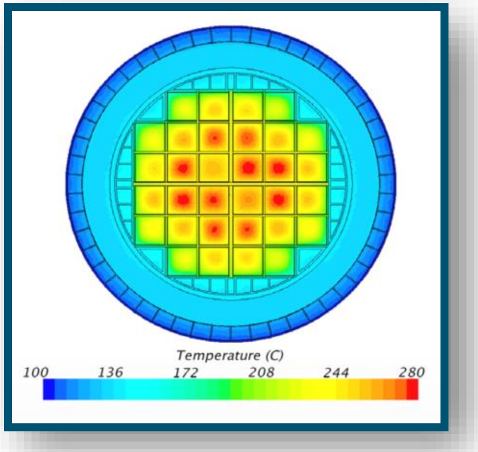
Differences in Accident Tolerant Fuels

- Zircaloy-based cladding
  - Focus on radial hydride formation.
- ATF/AF based cladding
  - The coated cladding designs are designed specifically to reduce waterside oxidation and thus reduce hydrogen production and cladding uptake.
  - The gaps related to cladding oxidation, hydride reorientation/cladding embrittlement, and cladding delayed hydride cracking are assumed to be of lesser importance.
- The lack of data with respect to higher temperatures and associated elevated internal rod pressures require a testing program, similar to the EPRI/DOE High Burnup Demonstration program Spent Fuel Rod Test Plan.
- Some ATF/AF fuel and cladding properties and characteristics, especially at the proposed higher burnups, will be obtained from examination of lead test assemblies as currently planned by DOE and the fuel vendors.
- Additional testing for the conditions of drying, storage, transportation (with emphasis on potential for fatigue failure), and accident conditions (especially in terms of potential release and dose consequences) are also needed

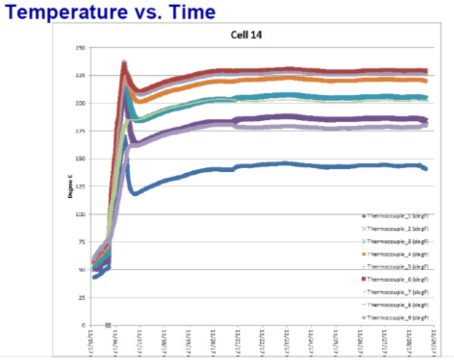
Tier 1 Gaps
Cross-Cutting Gaps

Thermal Profiles

Higher burnup fuels will be thermally hotter at reactor discharge and for longer periods and therefore it is likely that higher burnup fuels will experience higher temperatures during drying and early in dry storage.



Will hotter fuel affect the mechanical integrity of the fuel, cladding, and other structures, systems, and components (SSCs) over long-term storage and subsequent transportation?



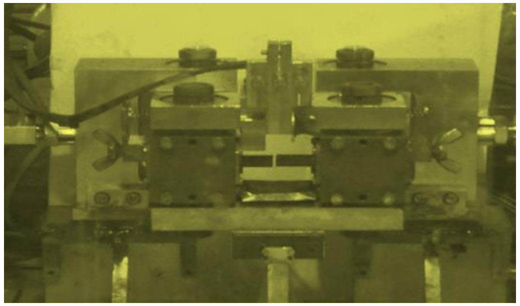
Thermal modeling and simulations can explore the extents of expected performance, while mechanical testing on representative irradiated ATF/AF materials can provide validating data for the models.

Stress Profiles

The impact of storage/transportation loadings should be evaluated for individual ATF/AF concepts.

Multi-modal Transportation Test

Can the new ATF/AF concepts withstand the impact of storage/transportation loadings?



Cyclic Integrated Reversible–
bending Fatigue Tester (CIRFT)
located at ORNL

Test articles should be subjected to **fatigue testing** (e.g., the ORNL CIRFT test apparatus) and **4-point bend tests to assess ATF/AF response to realistic mechanical loadings**. Modeling and simulations can explore the extents of expected performance.

4-point bend test–apparatus located at PNNL

Drying Issues

While ATF/AF cladding concepts are designed to limit oxidation, and with thinner oxide layers, in theory, the quantity of trapped and chemically-bound water will be reduced, the impact of residual water in dry storage on ATF/AF cladding and fuel in failed rods still should be quantified. The reaction of residual water with some of the ATF/AF fuels, especially uranium metals and nitrides, is much more aggressive than with UO_2-based fuels.

Is the effectiveness of the water removal processes during drying different for the ATF/AF concepts?

Component testing first on the retention of water in the fuel and hardware and then on the effects of water on ATF/AF concepts over the extended periods associated with storage can be performed.

SNL Dry Storage Cask Thermal Hydraulic Testing Pressure Vessel and Mount.

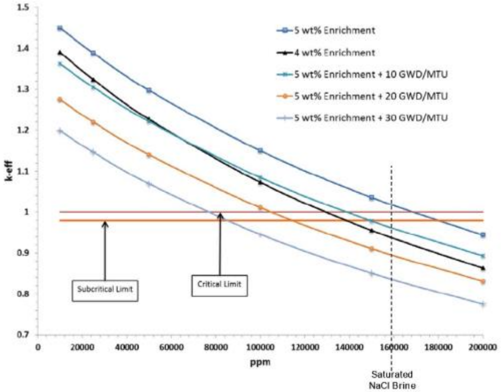
Subcriticality

The higher density and higher enrichment ATF/AF fuels may increase the risk of criticality if the fuel relocates within a cask or if the cask floods, as is typically assumed for the hypothetical accident condition during transportation.

Are the current cask/canister designs adequate to address criticality potential?

Criticality codes and generation and depletion codes (e.g., ORIGEN) can be used to assess the criticality potential of ATF/AF under transportation conditions to determine if different cask design features (e.g., increased neutron absorber material and longer-term performance of those materials) are necessary.

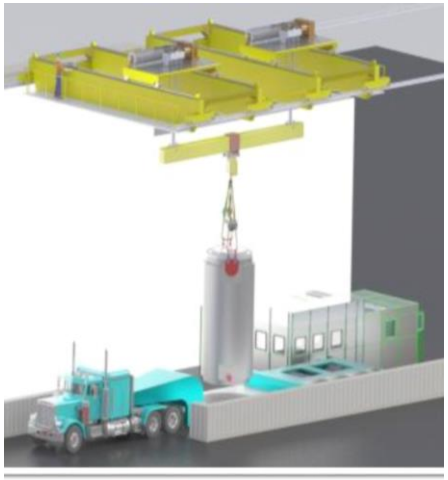
Detailed radionuclide inventory studies can be performed to validate codes and to facilitate the use of burnup credit.



Review of Criticality Evaluation for Direct Disposal of DCs and Recommendations, SFWD-SFWST-2018

Fuel Transfer Options

ATF/AF fuels, especially those with higher density, will be heavier, more radioactive, and have a higher temperature. Those changes need to be assessed relative to plant and transportation infrastructure and operational constraints.



Can facilities handle the heavier loads and can potentially more brittle cladding types sustain the handling loads?

Conduct studies to assess loaded canister weights relative to utility crane lift and railcar limits.

Tier 2 Gaps
SSC-specific gaps

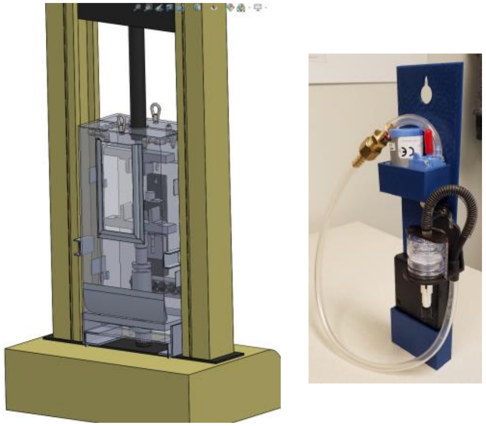
Consequences of Canister Failure

The canister is the primary confinement barrier during SNF dry storage. A failure, such as via stress corrosion cracking, could potentially release radionuclides in crud, oxide layers, fuel particulates, or fission gas.

Are there differences in the form, size distribution, and radioactivity of the ATF/AF fuel designs?

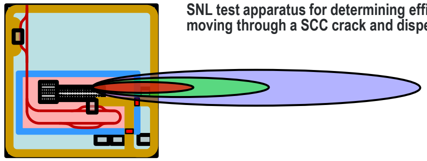
Determine fuel particle size and fractional release under simulated burst and rod-breakage experiments and then conduct engineering tests using these particle sizes to quantify potential release scenarios. Similarly, determine the extent of oxide and crud layer spallation and their particle sizes to determine their potential contribution to dose upon release.

PNNL gas communication and particle release equipment



ONRL Illustration of a load frame aerosol collection enclosure.

SNL test apparatus for determining efficiency of particles moving through a SCC crack and dispersion modeling.



Fuel Fragmentation

In reactor, fragmentation of UO_2 fuel results in the formation particles that can be dispersed upon cladding breach, with finer particles produced as the burnup surpasses a threshold between 60 and 75 GWd/MTU.

In UO_2 fuel, a high burnup rim is produced where the fuel grains have subdivided into submicron grains and its thickness increases with increasing burnup.

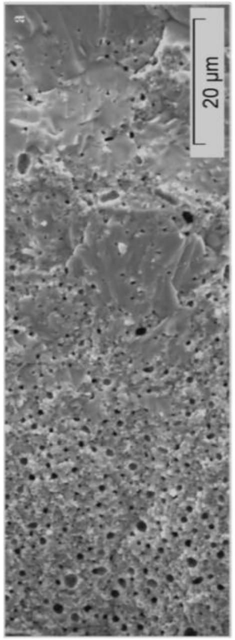
While Cr-doped fuels increase grain size and are expected to reduce fuel fragmentation, this has not been confirmed at high burnup.

Fragmentation of all fuel types may result in fuel relocation that could impact gas communication within a rod and could affect release fractions upon breach of the cladding.

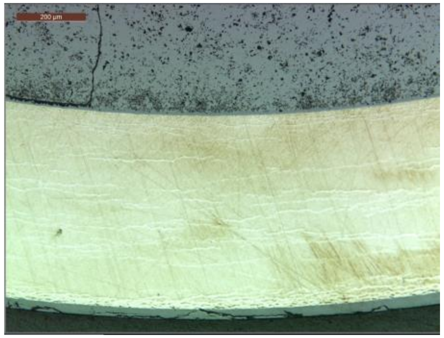
Are fuel particle release fractions different for ATF/AF designs?

Ceramography/metallography of samples from different axial locations in the rod can be performed to determine the extent of fuel fragmentation and thickness of the high burnup rim.

Gamma scanning and neutron radiography can be performed to determine relocation within a rod.



Example of High-burnup Rim Structure



ORNL Sister Rod Destructive Examination, M2SF-19OR010201026

Fuel Restructuring/Swelling

Higher burnup results in increased fission gas production and will also result in increasing pressurization over time by generation of helium from alpha decay. The Cr-doped pellets are expected to release less fission gas, but the retained gas could result in pellet swelling.

?

Will pellet swelling may result in localized stress risers in the cladding, which in turn could facilitate other degradation mechanisms such as creep or stress corrosion cracking?

Ceramography/metallography of samples from different axial locations in the rod can be performed to determine the extent of fuel swelling and the possibility of pellet clad mechanical interaction (PCMI).

Fission Product Attack on Cladding

Pellet-cladding chemical interaction (PCCI) is a fairly well understood mechanism that has been reduced by changes to pellet geometry and then the introduction of barrier fuels (e.g., Zr liner). The compatibility of ATF/AF cladding with the potential release of iodine, cesium, and cadmium from non-oxide fuels must be understood for long term cladding performance.

Will fission products in contact with the cladding facilitate cracking and breach of the cladding via PCCI?

Detailed examination of the pellet/clad interface with high resolution tools such as TEM or EMPA will be performed to determine if fission products have been released and are present in this boundary. High resolution microscopy will be used to look for signs of cracking at the cladding inner diameter.

Fuel Oxidation

Increasing burnup in uranium oxide fuels increases its resistance to oxidation to U_3O_8, which can in turn rupture the cladding and result in fuel release and relocation, the long-term, lower temperature oxidation kinetics of the various ATF/AF and the susceptibility of ATF/AF cladding to rupture are not known.

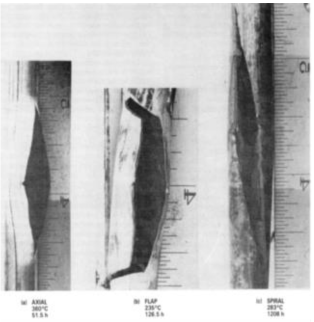
Will the kinetics of oxidation for ATF/AF fuels under the conditions of drying and storage (from retained water, water vapor, or radiolytically-generated oxidants) cause detrimental effects during long-term storage conditions?

Could these changes produce the stress necessary to initiate and propagate a crack in the various ATF/AF cladding types?

Thermogravimetric analysis of fuel samples under relevant temperature conditions in dry and humid air will determine the oxidation rates. Expanded ring testing and burst testing of irradiated cladding can be used to measure ATF/AF cladding properties.

U_3O_8 formation

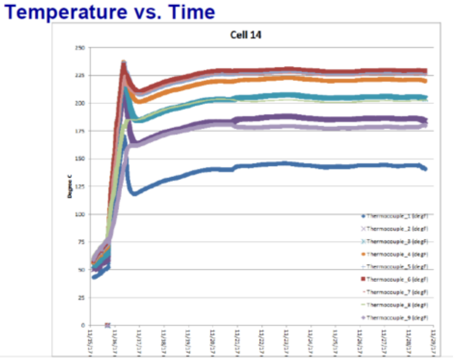
Clad oxidation leading to unzipping at ANL



Clad unzipping

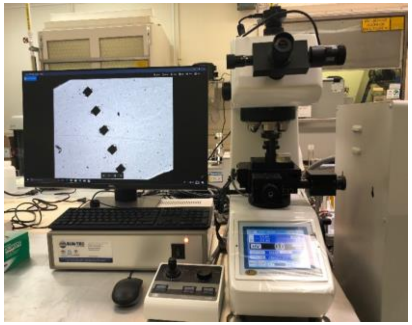
Annealing of Radiation Damage

Radiation damage is a function of the fast neutron fluence and irradiation temperature, and higher burnup is expected to result in increased radiation damage. In zirconium-based alloys, radiation damage tends to increase the hardness and strength of the cladding, with an associated reduction in ductility. The strength and ductility of ATF/AF cladding at high burnup will be well-documented for reactor operation. However, in dry storage, the radiation damage can be annealed, resulting in strength and ductility characteristics that are more consistent with unirradiated cladding.



Will annealing during drying cause changes in cladding strength and ductility?

Microhardness and ring compression tests on both as received and then thermally annealed samples can be performed.



Microhardness of cladding
at PNNL

Creep

The main driving force for cladding creep is the hoop stress caused by internal rod pressure. The end-of-life rod internal pressure of the higher burnup fuels needs to be measured and then modeled for the temperatures relevant to drying and early storage. Cr, SiC, and FeCrAl will creep differently than zircaloy and the limits before failure must be understood.

Creep results in the thinning of the cladding wall, which could then subsequently fail upon external loading or with increased rod internal pressure related to higher burnup.

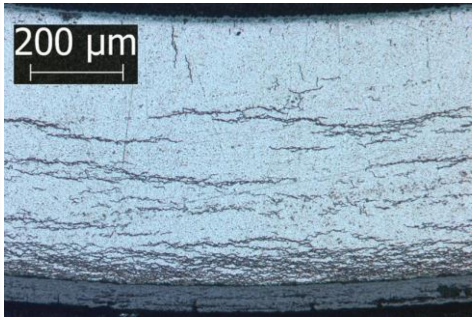
Will Cr, SiC, and FeCrAl creep differently than zircaloy? What are the limits before failure?

End of life rod internal pressures can be measured by puncturing rods and collecting the gas. Creep tests (pressurized tube tests) as a function of time and temperature can be conducted to determine if thermal creep is an issue for storage of the new cladding types.

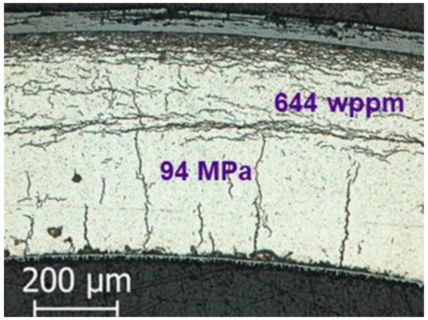
Embrittlement

Cr diffusion into the underlying zircaloy may result in embrittlement of the zircaloy. SiC claddings are known to be brittle.

Cladding embrittlement reduces the positive performance of the fuel cladding under internal pressure, mechanical, and thermal loadings.



Hydride reorientation at ANL



What is the extent of cladding embrittlement under realistic internal pressure, mechanical, and thermal loadings?

Ring compression tests, axial tube tensile, and burst tests of as-irradiated and heat-treated irradiated samples can be conducted on defueled cladding to determine cladding properties and establish if embrittlement is an issue.

CIRFT fatigue tests and ring compression tests conducted on irradiated composite fuel segments can be performed combined with metallographic imaging and chemical composition studies.

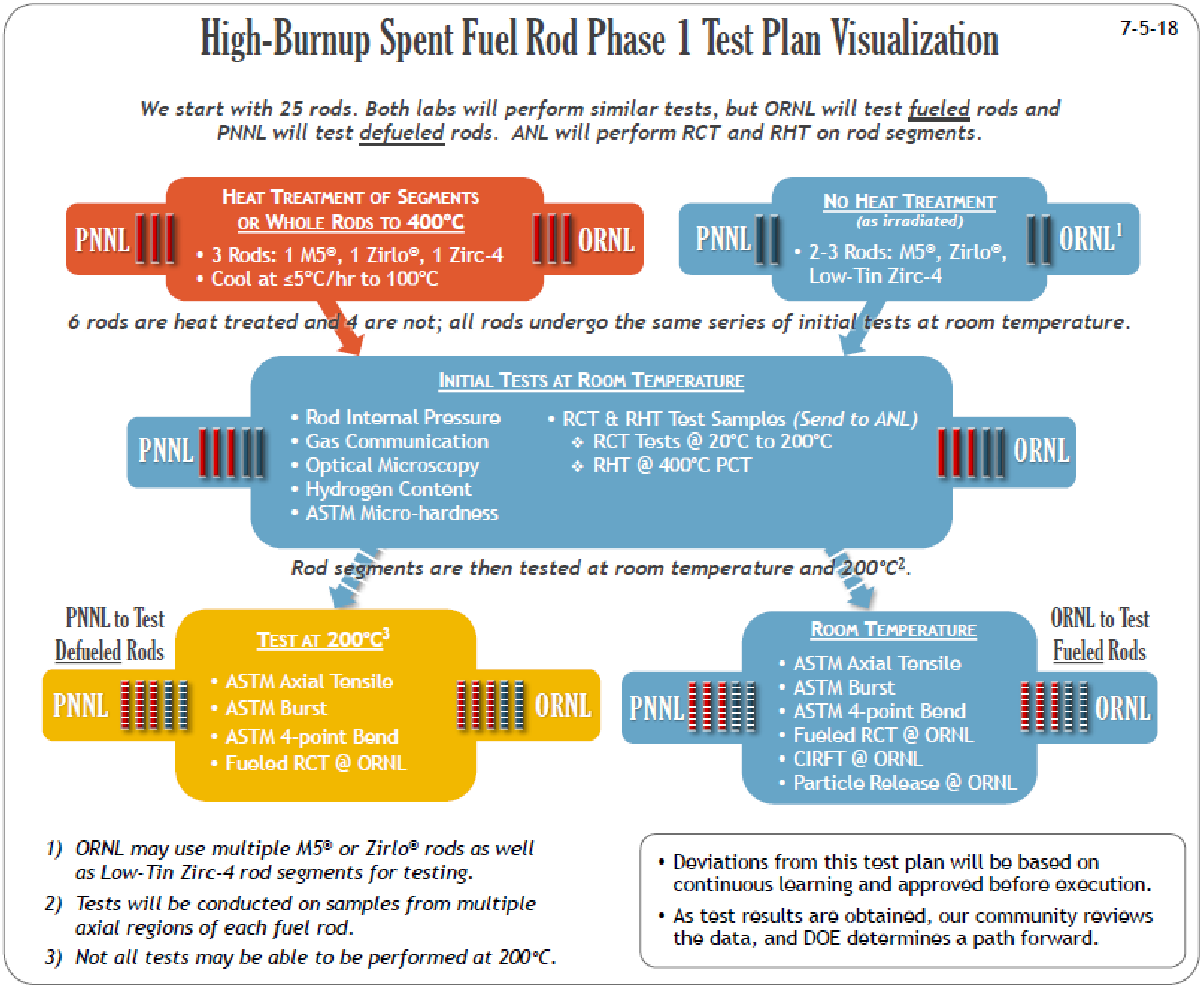
Path Forward

Path Forward

- The lack of data with respect to potential storage and transportation degradation mechanisms for ATF/AF, especially for the expected higher burnups, higher temperatures and higher internal rod pressures, require a testing program similar to the EPRI/DOE High Burnup Demonstration Program Sibling Pin Test Plan to ensure that the NRC requirement for preventing gross rupture is met.

- Attention should focus on
  - damaged spent fuel particulate size and quantity
  - clad coating robustness
  - potential corrosion and hydride potential in areas of damaged clad coatings
  - and logistical challenges from increased container weight, temperatures, and radiation levels.

ATF/AF fuels can follow the EPRI/DOE High Burnup Demo Cask Sibling Rod Test Plan



High-Burnup Spent Fuel Rod Phase 1 Test Plan Visualization

7-5-18

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- Gas Communication
- Optical Microscopy
- Hydrogen Content
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- RCT & RHT Test Samples *(Send to ANL)*
  - ❖ RCT Tests @ 20°C to 200°C
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Rod segments are then tested at room temperature and 200°C[2]*.*

PNNL to Test Defueled Rods

TEST AT 200°C[3]
- ASTM Axial Tensile
- ASTM Burst
- ASTM 4-point Bend
- Fueled RCT @ ORNL

ORNL to Test Fueled Rods

ROOM TEMPERATURE
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- ASTM Burst
- ASTM 4-point Bend
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- CIRFT @ ORNL
- Particle Release @ ORNL

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- As test results are obtained, our community reviews the data, and DOE determines a path forward.

This allows direct comparison of mechanical properties to the currently used zircaloy-based cladding.

Procedures have been completed, and equipment has been purchased and validated.

Storage and Transportation Canister/Cask Systems

- The current storage and transportation canister/cask systems should be able to manage the rod/assembly ATF/AF fuel systems that are very similar to existing commercial fuel systems.
- Modifications to handle the fuel’s increased weight and greater thermal and ionizing radiation loads may be required.
- Significantly different fuel systems (e.g., Triso, metal uranium) will require a new storage, transportation, and disposal container design.

Discussion

Thank You