PNNL-32222

Assessment of FAST for Spent Fuel Calculations on High Burnup and High Enriched Fuel

October 2021

Ken Geelhood
Beric Wells

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Contract DE-AC05-76RL01830
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Pacific Northwest National Laboratory
Richland, Washington 99354

Summary

A literature review was performed on data availability and needs for assessing high burnup (up to 75 GWd/MTU rod average) and increased enrichment (up to 19.75 wt. % U-235) for standard and accident tolerant fuel (ATF) rods in FAST for spent fuel storage and dry storage applications. The literature review investigated the following end-of-life fuel characteristics:

- Fission gas release
- End-of-life rod internal pressure
- Cladding oxide thickness and hydrogen content
- Ex-reactor creep testing

This report provides an overview of the existing FAST assessment with regard to these characteristics and shows FAST predictions for new data discovered in the literature review.

This expanded assessment did not result in the need for new models for standard fuel and previous work has already been done to include models for the analysis of near-term ATF concepts. This report summarizes the ATF model additions.

The updated FAST assessment data is shown, and ranges of applicability are determined for FAST predictions. Finally, new additional data that could be used to improve the updated FAST assessment are identified.

Acronyms and Abbreviations

ATF	Accident Tolerant Fuel
BWR	Boiling Water Reactor
CrN	Chromium Nitride
Cr_2O_3	Chromium Oxide
DOE	Department of Energy
FAST	Fuel Analysis under Steady-state and Transient
FGR	Fission Gas Release
GWd/MTU	Giga Watt Days per Metric Ton of Uranium
HBU	High Burnup
IFA	Instrumented Fuel Assembly
IFBA	Integral Fuel Burnable Absorber
LTA	Lead Test Assembly
NRC	Nuclear Regulatory Commission
PIE	Post-Irradiation Examination
PNNL	Pacific Northwest National Laboratory
PWR	Pressurized Water Reactor
RXA	Recrystallized Annealed
SNF	Spent Nuclear Fuel
SRA	Stress-Relief Annealed
UO_2	Uranium Dioxide
UO_2-Gd_2O_3	Uranium Dioxide Mixed with Gadolinium Oxide
wt%	Weight Percent

Contents

Figures

Tables

1.0 Introduction

There has recently been discussion within the nuclear industry to increase uranium enrichment and fuel burnup to support longer cycles and improve fuel utilization. Additionally, industry has developed and is currently in the process of testing several accident tolerant fuel (ATF) concepts in commercial reactors. Most typically when the industry requests approval for a new range of operation or for new fuel or cladding materials, the review of these requests focuses on what data are available to validate both existing and new analytic tools that will be used to perform the safety analyses for fuel containing these new materials.

For increased burnup and increased enrichment, most of the work performed on existing codes to date has focused on validating the in-reactor performance in these ranges. However, there are some questions that remain as to whether existing codes and methods for spent fuel storage will be applicable to the storage and transportation of spent fuel assemblies that had initially higher enrichment and were irradiated to higher burnup. Similarly, for ATF technology, there has been little effort to evaluate if existing methods and limits that have been developed for storage and transportation of traditional spent fuel will be applicable to the storage and transportation of spent ATF.

This report focuses on NRC's fuel performance code, Fuel Analysis under Steady-state and Transients (FAST) (Geelhood, et al., 2021). This code is often used to provide end-of-life conditions such as fission gas release fraction and rod internal pressure for spent fuel analyses. Additionally, FAST, and its predecessors, FRAPCON (Geelhood, Luscher, Raynaud, & Porter, 2015) and FRAPTRAN (Geelhood, Luscher, Cuta, & Porter, 2016) have been used to perform ex-reactor creep predictions and accident analyses for spent fuel. Because of these uses of FAST, it is important to identify the range of applicability of FAST, particularly if burnup levels and enrichment levels are expected to increase in the future and if ATF technology is deployed to commercial reactors.

This report will provide an overview of the current assessment range of FAST-1.0 as it relates to modeling both the base irradiation behavior that impacts spent fuel conditions as well as modeling the dry storage period of spent fuel (Section 2.0). A literature review was performed to identify any relevant data that has been published in the last 5-7 years that could be used to improve the coverage of the assessment of FAST. The new data will be described in Section 3.0. New models that have been included in FAST will be described in Section 4.0. Section 5.0 will show the code predictions of the new data in addition to existing assessment data.

For the context of this report, high burnup will refer to fuel with a rod-average burnup up to 75 GWd/MTU, and high enrichment will refer to U-235 content up to 19.75 wt% U-235. For ATF technologies, industry has identified several design features as "near-term" ATF concepts. These include Cr-coated zirconium-alloy cladding, UO_2 pellets doped with Cr_2O_3 and various other oxides that result in large (40-60 μm) grain size, and FeCrAl cladding. The industry is pursuing coated cladding and doped pellets for deployment by the mid-2020s; however, licensing or deployment dates for FeCrAl have not been provided to the NRC at this time. FeCrAl cladding will not be included in this report due to anticipated deployment being more than 5 years away. Other ATF concepts such as SiC/SiC composite cladding or uranium silicide and uranium nitride fuel are longer term ATF concepts and will not be included in this report due to a lack of a mature design. Standard fuel that is considered by FAST are all approved US cladding alloys; Zircaloy-2, ZIRON, Zircaloy-4, M5®, ZIRLO™, Optimized ZIRLO™, and

AXIOM™, and UO_2 pellets, Gd_2O_3 doped pellets, and UO_2 pellets with ZrB_2 coating (Integral Fuel Burnable Absorber, IFBA).

The parameters that FAST predicts that are most relevant to spent fuel modeling are:

- Fission gas release
- End-of-life rod internal pressures
- Cladding oxide thickness and hydrogen content
- Ex-reactor cladding creep of irradiated cladding[1]

The overall assessment of FAST includes data beyond these including fuel centerline temperature, in-reactor cladding creep and fission gas release and cladding strain following power ramps. These parameters are not a major concern to the storage and transportation of spent fuel and will not be included in this report.

It is noted that U-235 loading (i.e., enrichment level) does impact the decay heat of spent fuel. However, during dry storage the cladding temperature is affected by the cask design and the conditions of the neighboring rods and assemblies. FAST is a single-rod code, and therefore not suitable for determining the temperature evolution axially and with time during dry storage. For this reason, the cladding temperature for the dry storage period is typically input into FAST. The ANS-5.1 standard (American Nuclear Society, 2019) gives a well validated method for determining decay heat in used nuclear fuel as a function of burnup and fuel enrichment and these values can be used as input into a cask model to determine expected cladding temperature histories.

[1] Ex-reactor cladding creep is typically a concern for temperatures starting at 330°C and up to the recommended limit of 400°C and for cladding hoop stress between 30 and 250 MPa.

2.0 Current Assessment of FAST-1.0

One of the most important aspects of FAST is the assessment that is performed relative to applicable data. This assessment consists of two parts. The first is assessment of individual material property models relative to data. This assessment is described in (Geelhood, Goodson, Luscher, Corson, & Kyriazidis, 2021). This assessment provides confirmation that the individual models in FAST are representative of the available data. The second part of the assessment is a code integral assessment (Geelhood, et al., 2021). This assessment provides confirmation that the overall FAST code predictions are representative of the available data. Integral assessment data include fuel centerline temperature, fission gas release, cladding oxide thickness, and cladding strain following a power ramp. The overall assessment of FAST is used to identify the range over which FAST is validated and the expected uncertainty in the FAST code predictions.

The following sections will describe the assessment of FAST-1.0 in areas relevant to spent fuel modeling for standard fuel and ATF. No formal assessment of FAST has been performed against ATF fuel given the developmental nature of these concepts and significant lack of data. However, some discussion of the expected applicability of FAST to ATF concepts will be discussed in this section. The limited recent data and model updates will be discussed in Sections 3.0 and 4.0.

2.1 Fission Gas release

This section describes the assessment of fission gas release predictions of FAST-1.0.

2.1.1 Standard Fuel

For UO_2 FAST has been assessed to provide reasonable predictions of fission gas release (FGR) from a rod-average burnup of 0-80 GWd/MTU. Comparisons of the measured and predicted data are shown in Figure 1. The standard deviation for the steady-state predictions is 2.6% absolute FGR.

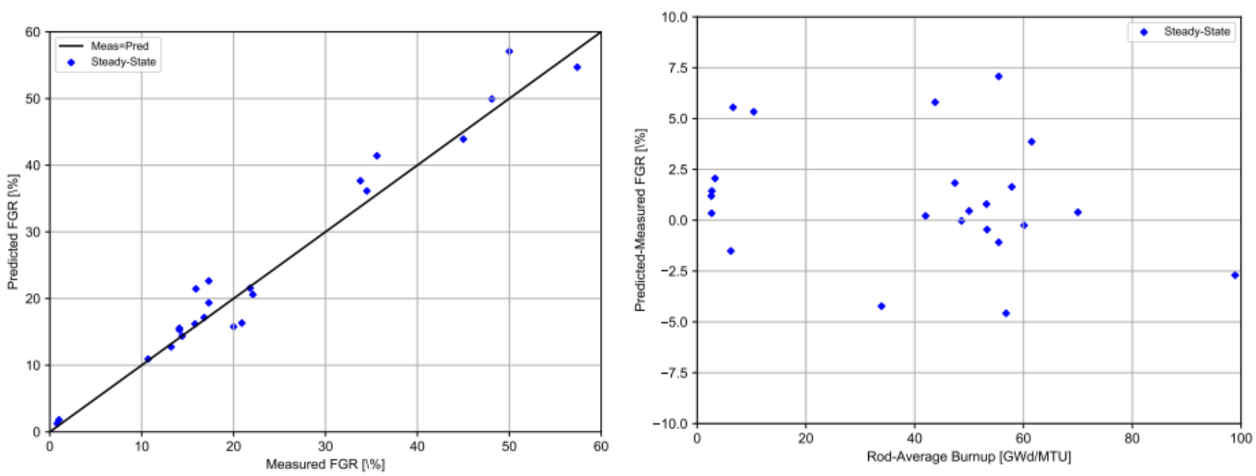


Figure 1. FAST fission gas release assessment for UO_2 rods (measured vs. predicted and predicted minus measures vs. burnup)

For UO_2-Gd_2O_3 FAST has been assessed to provide reasonable predictions of fission gas release from a rod-average burnup of 0-40 GWd/MTU. Comparisons of the measured and predicted data are shown in Figure 2. The standard deviation for these four predictions is 0.3% absolute FGR. FAST uses the same fission gas release model used for UO_2-Gd_2O_3 as it uses for UO_2. It is expected that this model will continue to be applicable to higher burnup UO_2-Gd_2O_3 fuel with the same standard deviation of 2.6% absolute FGR.

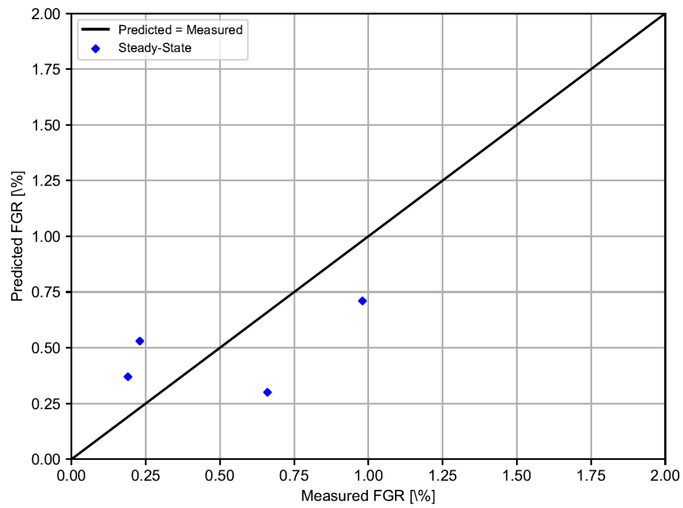


Figure 2. FAST fission gas release assessment for UO_2-Gd_2O_3 rods

There is no fission gas release data taken from IFBA fuel. However, the IFBA coating is very thin and is not expected to impact the fission gas release behavior from the pellets other than changing the power level in the pellets. The power level is input into FAST based on predictions from a core simulator. Given the appropriate power inputs, FAST is expected to provide reasonable predictions of fission gas release from IFBA fuel between 0-80 GWd/MTU with the same standard deviation of 2.6% absolute FGR.

2.1.2 ATF

The two most near-term ATF concepts are Cr_2O_3-doped UO_2 fuel and Cr-coated cladding. No assessment or measurements have been made of fission gas release from fuel with Cr-coated cladding. The coating does not impact the fuel temperature, so there are no known mechanisms whereby the use of Cr-coated cladding would impact fission gas release behavior from the pellets.

Limited measurements were made on Cr_2O_3 doped fuel that was irradiated in Halden prior to these pellets being declared to be an ATF concept. These data indicate that the steady-state fission gas release from doped UO_2 pellets is similar to that of standard UO_2 pellets and can be adequately predicted using existing models.

Halden has tested fuel with various additives that are similar to current ATF fuel concepts. These irradiations occurred under IFA-677 and IFA-716. For IFA-716 no measurements of fission gas release were made. IFA-677 had one rod with additives (doped fuel) and one rod without additives (standard fuel) that were irradiated to a rod-average burnup of around 26 GWd/MTU with measured fission gas release. FAST has been shown to provide reasonable predictions of fission gas release from the limited data on Cr_2O_3-doped fuel over a rod-average burnup of 0-30 GWd/MTU (Richmond & Geelhood, 2018). Data above this burnup level is lacking. Data should become available as some of the lead-test rods being irradiated reach high burnup and are examined in hot cells. The FAST predictions for the two rods from IFA-677 are shown in Figure 3. Both of these rods are reasonably well predicted using the standard fission gas release model in FAST.

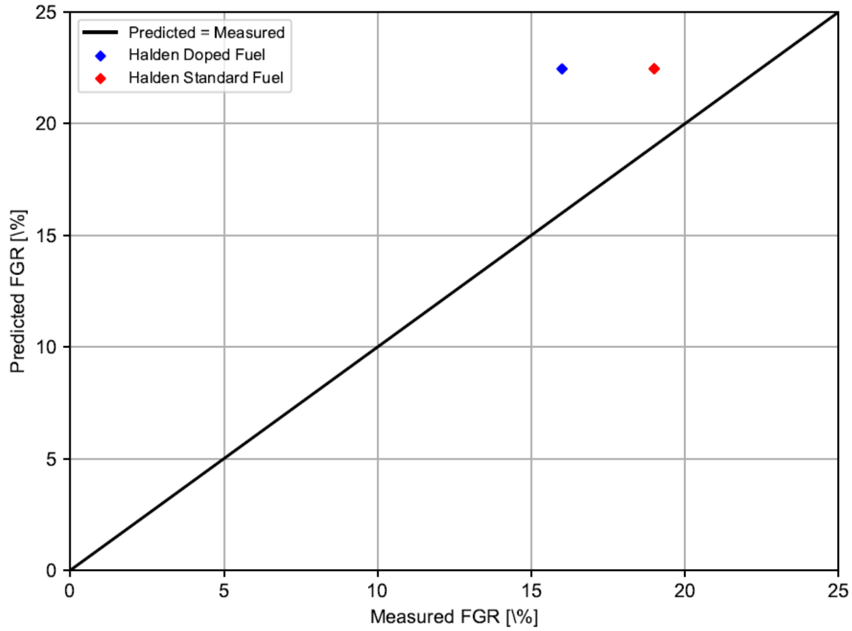


Figure 3. FAST fission gas release assessment for Cr_2O_3-doped UO_2 rods

2.2 End-of-Life Rod Internal Pressure

This section describes the assessment of rod-internal pressure predictions of FAST-1.0.

2.2.1 Standard Fuel

For UO_2 FAST has been assessed against void volume and has been shown to provide reasonable predictions of void volume from a rod-average burnup of 0-63 GWd/MTU. The standard deviation for the predictions is ±1.7 cm^3. If void volume and fission gas release are accurately predicted, then the pressure at room temperature can also be reasonable predicted. The uncertainty in void volume corresponds to an uncertainty in pressure of approximately ±1 MPa (145 psi).

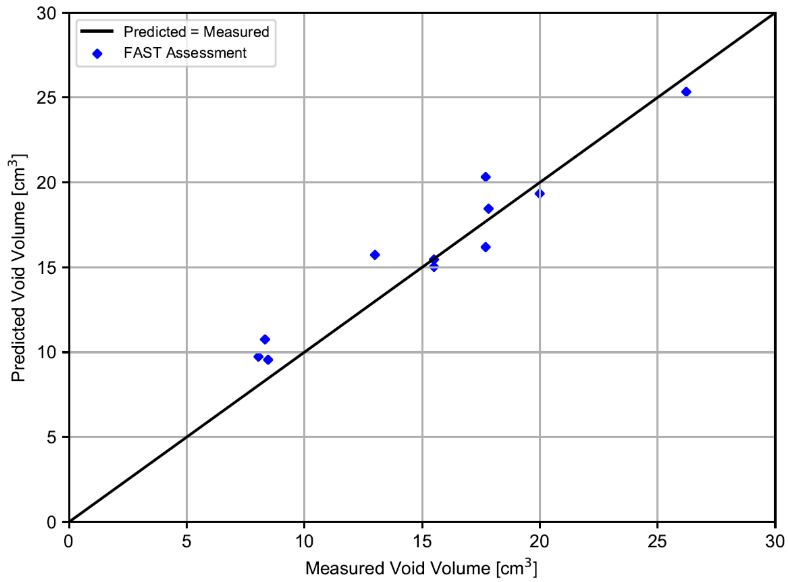


Figure 4. FAST void volume assessment for UO_2 rods

Data has not been made available to assess the void volume predictions of UO_2-Gd_2O_3 or IFBA fuel. However, the void volume and rod internal pressure are driven by pellet and cladding deformation and fission gas release, all of which have been adequately modeled in FAST (Geelhood, Goodson, Luscher, Corson, & Kyriazidis, 2021) (Geelhood, et al., 2021). For IFBA fuel there is considerable helium production and release from the IFBA layer and this production and release is adequately modeled in FAST (Geelhood, et al., 2021). The increase in fission gas inventory will have a small impact on the overall void volume, but the FAST creep model will predict that change. Based on these models, the predictions of rod internal pressure and void volume for these fuel types are considered valid over the same burnup range as for UO_2. The uncertainty in these predictions is also expected to be ±145 psi.

2.2.2 ATF

No high burnup fuel from ATF irradiation currently exists. However, the Cr-coating applied to ATF cladding is not expected to significantly alter the cladding deformation behavior (and resulting void volume) of the cladding relative to uncoated cladding. Additionally, current indications are that the doped pellets perform the same as UO_2 pellets with regard to swelling and gas release (see Section 2.1.2). Therefore, it is reasonable to assume that the FAST predictions of rod internal pressure for the identified near-term ATF fuel will be reasonable up to 63 GWd/MTU with a standard error of ±145 psi

2.3 In-reactor corrosion and hydrogen pickup

This section describes the assessment of end-of-life cladding oxide thickness and hydrogen content. Standard fuel is categorized as either pressurized water reactor (PWR) and boiling water reactor (BWR) cladding. Neither cladding oxide thickness nor cladding hydrogen content is affected by fuel enrichment.

2.3.1 Standard Fuel (BWR)

For BWRs FAST has been assessed to provide reasonable predictions of oxide thickness for Zircaloy-2 cladding with a rod-average burnup of 0-52 GWd/MTU and of hydrogen content for Zircaloy-2 cladding with a rod-average burnup of 0-70 GWd/MTU. There is currently a gap in BWR oxide thickness predictions between 52 and 75 GWd/MTU. Oxide thickness in some alloys has been observed to accelerate at high burnup, however, BWR fuel vendors have been employing Zircaloy-2 with tighter composition specifications to control high burnup oxide thickness. The standard deviation for the oxide thickness predictions is 7.6 μm. For hydrogen pickup there is a model for vintage Zircaloy-2 (pre-1998) that has a standard deviation of 10 ppm below 45 GWd/MTU and 54 ppm above 45 GWd/MTU. For Zircaloy-2 since 1998 the model has a standard deviation of 13 ppm below 45 GWd/MTU and 60 ppm above 45 GWd/MTU. Figure 5 shows the FAST assessment for Zircaloy-2 oxide thickness (Geelhood, et al., 2021), (Geelhood & Beyer, 2008) and Figure 6 shows the FAST assessment for Zircaloy-2 hydrogen content (Geelhood & Beyer, 2011).

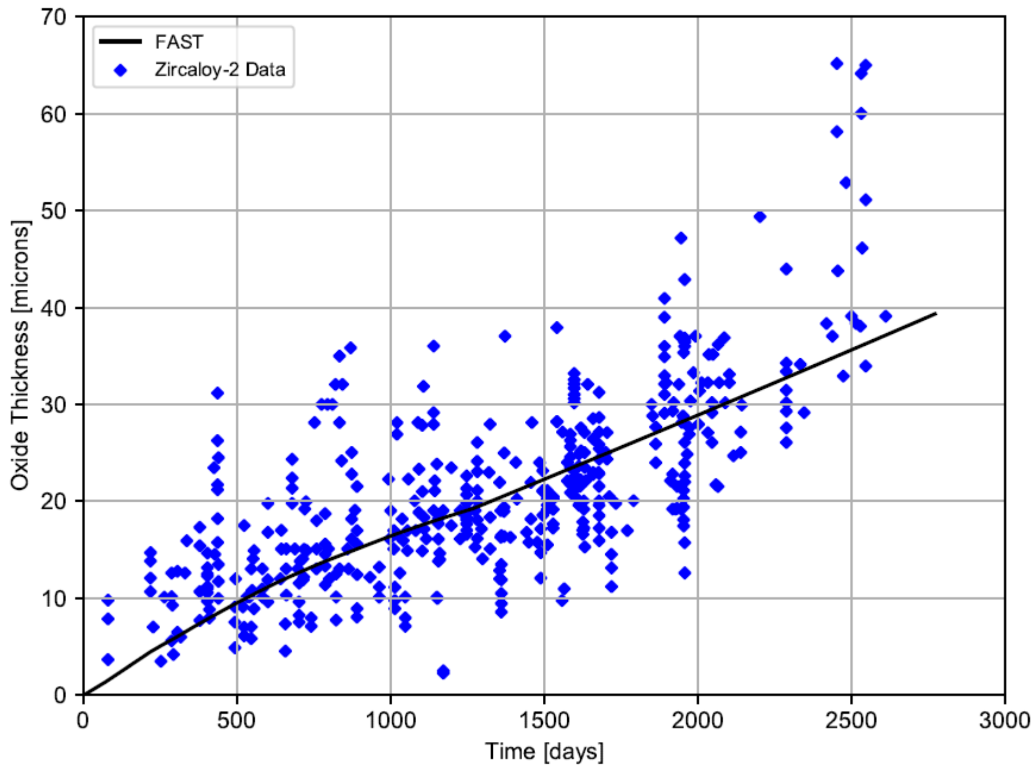
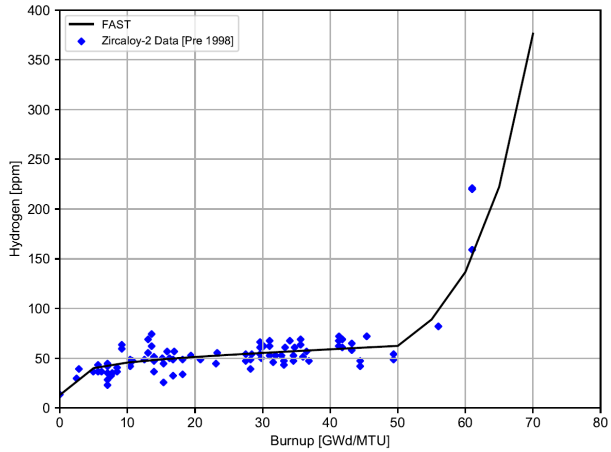
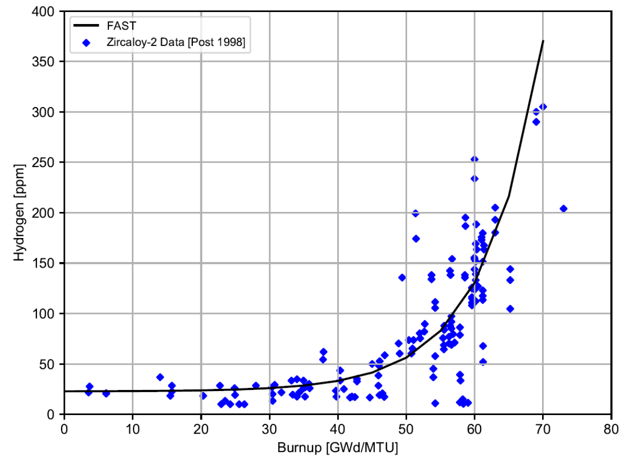


Figure 5. FAST Zircaloy-2 oxide thickness assessment



Prior to 1998



Since 1998

Figure 6. FAST Zircaloy-2 hydrogen content assessment

2.3.2 Standard Fuel (PWR)

There are a number of cladding alloys that have been approved for use in the US for PWR application. These include, Zircaloy-4, ZIRLO™, Optimized ZIRLO™, M5®, and AXIOM™. AXIOM™ had not been approved for use at the time of the FAST-1.0 assessment and was not included in that assessment. AXIOM™ will be discussed in Section 3.3.1. Table 1 shows the range of assessment of the oxide thickness and hydrogen content as well as the standard deviation for the predictions. There is currently a gap in Zircaloy-4 oxide thickness predictions between 53 and 75 GWd/MTU, however, burnup extensions beyond 62 GWd/MTU will likely not be approved for this alloy given superior performance of other alloys. For other alloys, data have been provided to higher burnup and will be discussed in Section 3.3.

Table 1. FAST PWR oxide thickness and hydrogen content assessment

Cladding	Oxide Assessment Range	Oxide Standard Deviation	Hydrogen Assessment Range	Hydrogen Standard Deviation
Zircaloy-4	0-53 GWd/MTU	15.3 μm	0-70 GWd/MTU	94 ppm
ZIRLO™	0-66 GWd/MTU	15 μm	0-70 GWd/MTU	110 ppm
Optimized ZIRLO(a)	0-66 GWd/MTU	15 μm	0-70 GWd/MTU	110 ppm
M5®	0-68 GWd/MTU	5 μm	0-70 GWd/MTU	23 ppm

(a) Assumed to be the same as ZIRLO

Typically oxide thickness data is presented as a function of rod burnup or some other parameter such as fuel duty index. Because each oxide thickness data point comes from a specific axial location and each reactor may have small differences in coolant temeperature, models in FAST are developed by creating representative rod inputs and turing the parameters in the oxide model such that the representative rods give results within the oxide vs burnup data. As such, the oxide models are inherently assessed to the burnup levels of the data. These assessments are shown in Figure 7 through Figure 9. Further assessment may be performed on axial oxide data taken at end of life. These data show that FAST provides reasonable predictions of the axial distribution of the oxide thickness. Assessment data for Zircaloy-4, ZIRLO™, and M5® are shown in Figure 10 through Figure 12 (Geelhood & Beyer, 2008), (Geelhood & Beyer, 2011), (Geelhood, et al., 2021).

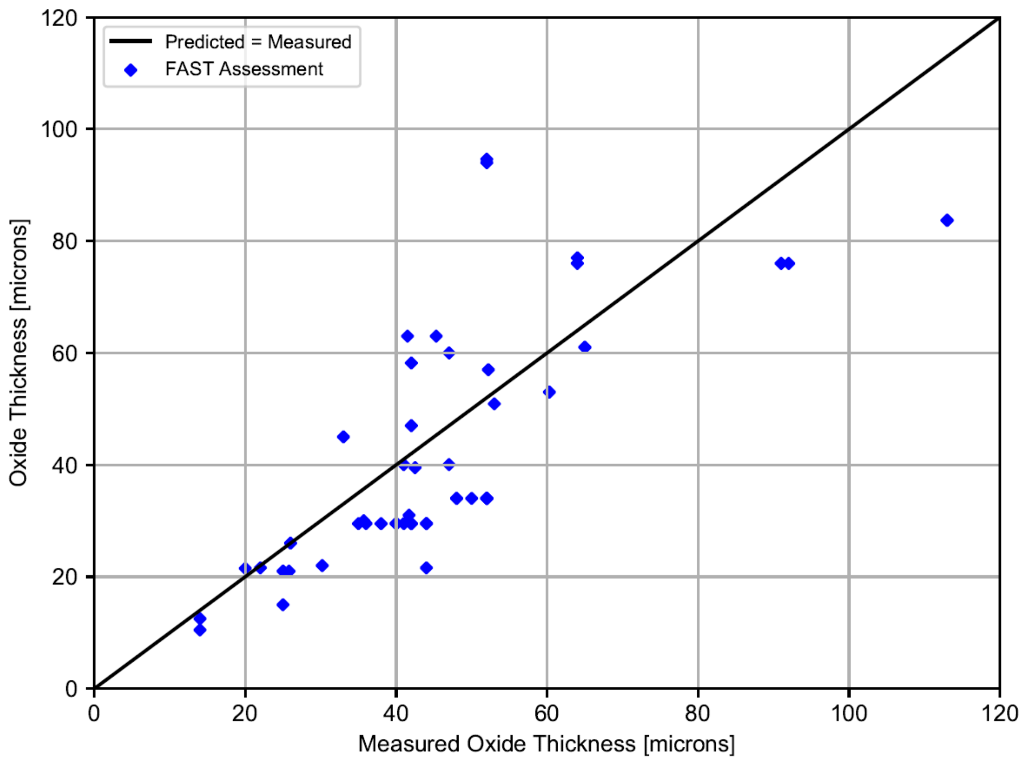


Figure 7. FAST Zircaloy-4 oxide thickness assessment

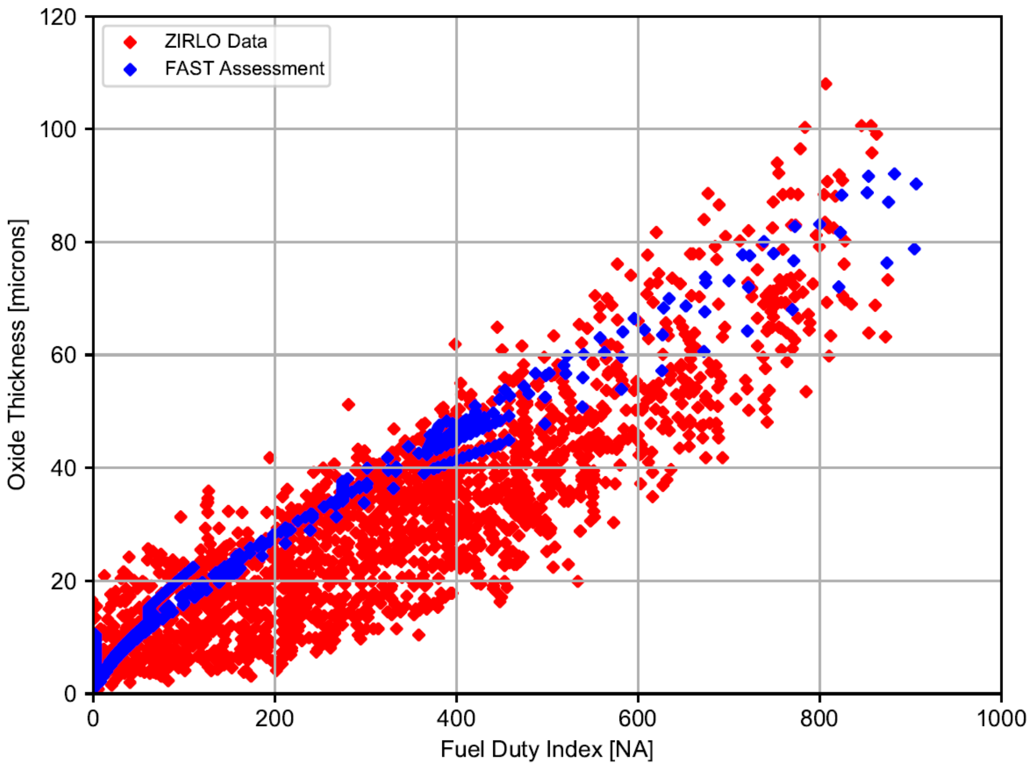


Figure 8. FAST ZIRLO™ oxide thickness assessment

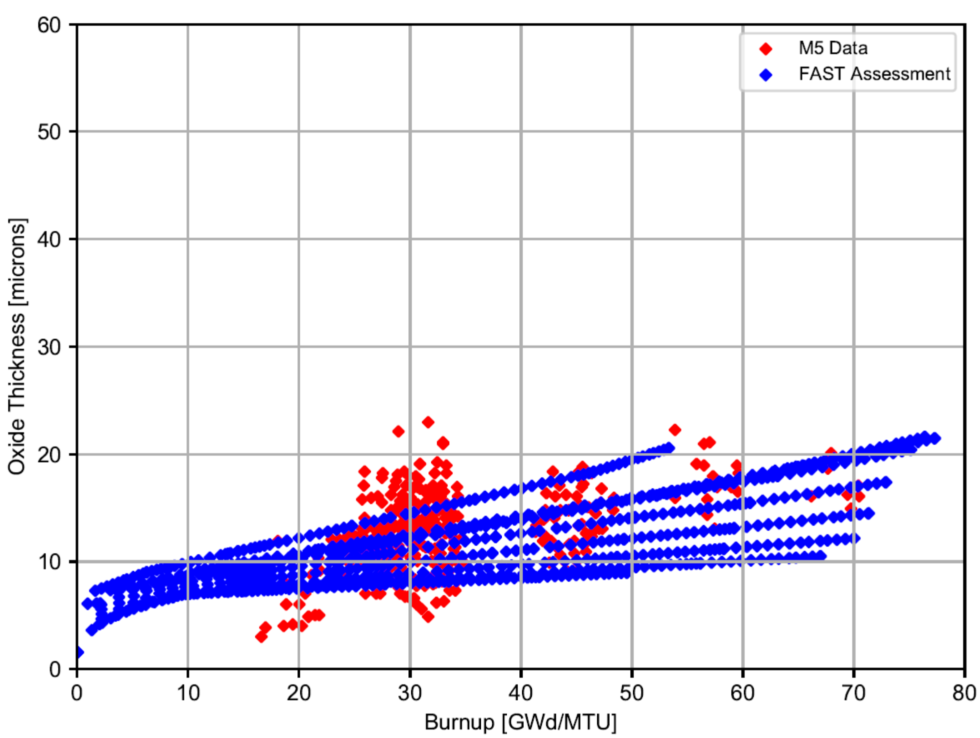


Figure 9. FAST M5® oxide thickness assessment

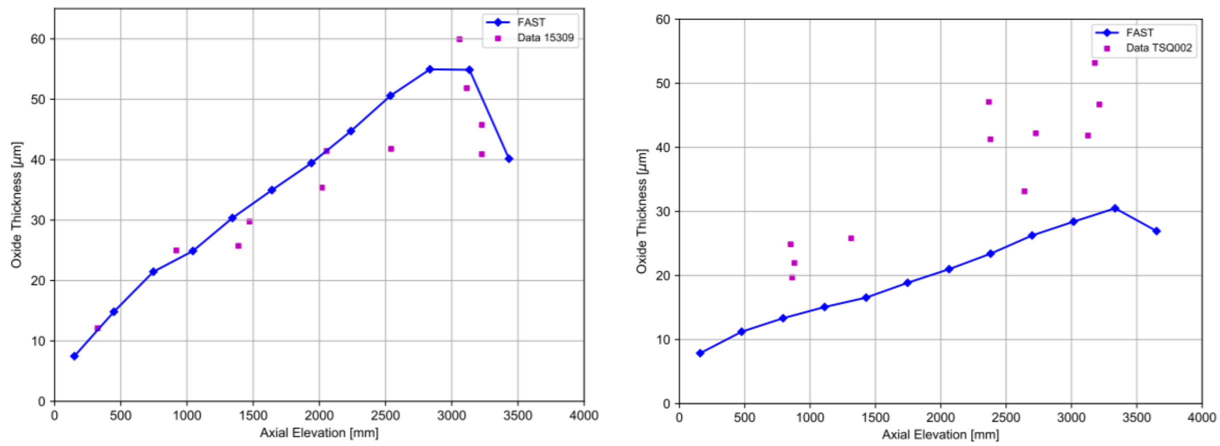


Figure 10. FAST Zircaloy-4 axial oxide thickness assessment

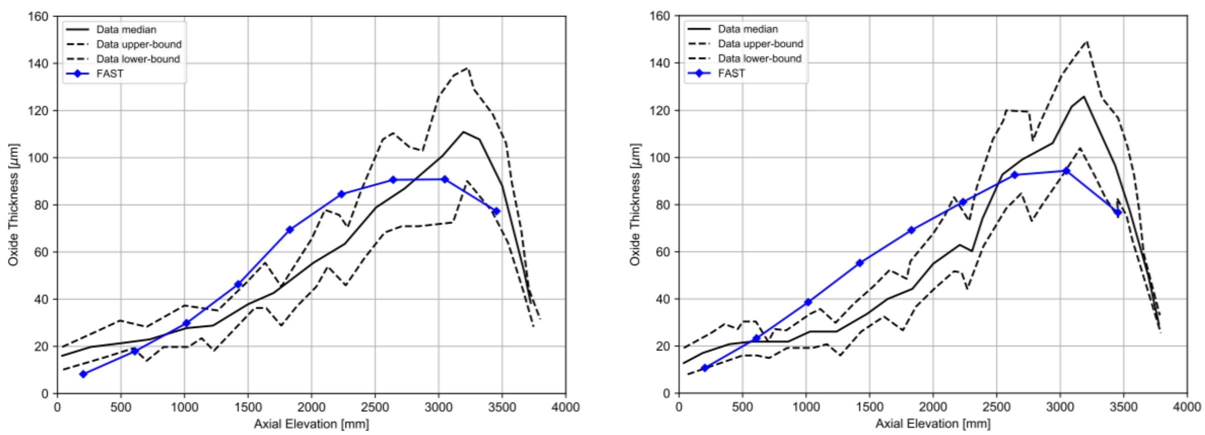


Figure 11. FAST ZIRLO™ axial oxide thickness assessment

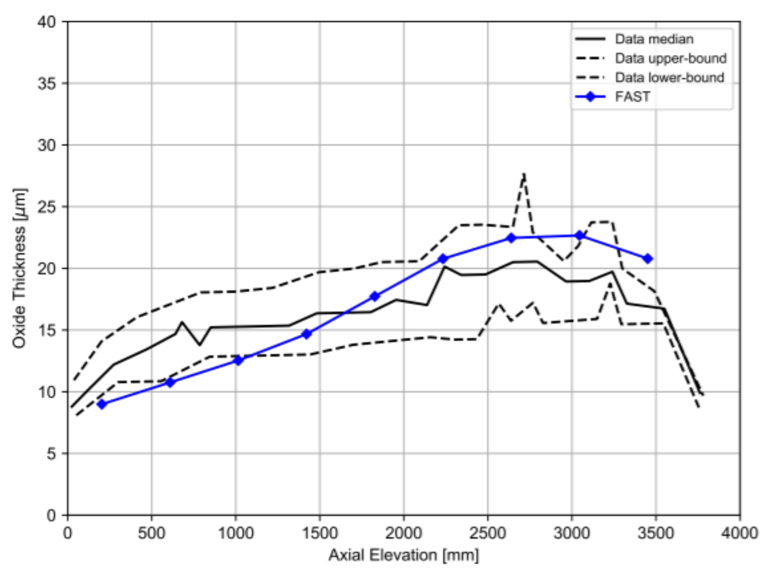


Figure 12. FAST M5® axial oxide thickness assessment

Assessment data for Zircaloy-4, ZIRLO™, and M5® hydrogen content are shown in Figure 13.

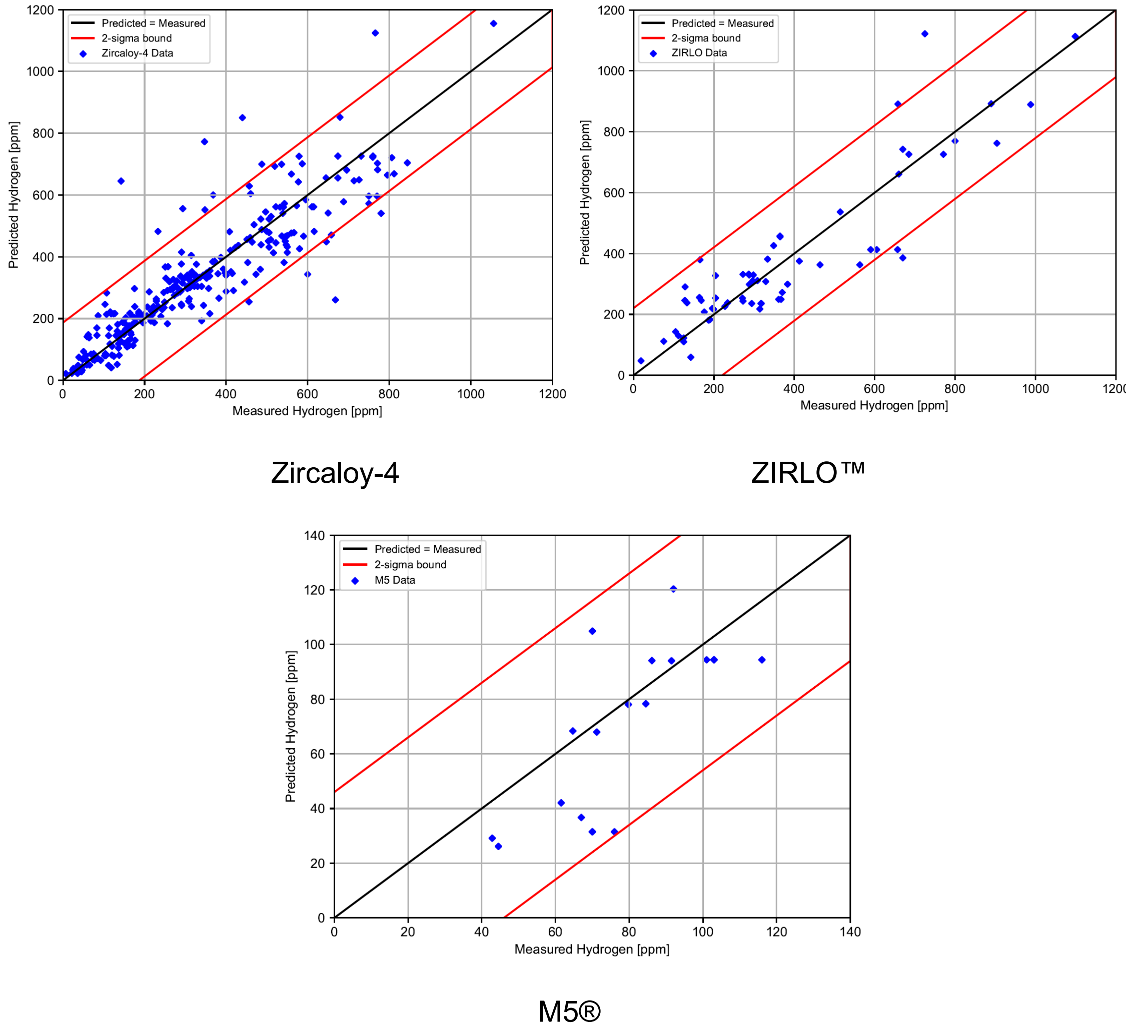


Figure 13. FAST Zircaloy-4, ZIRLO™, and M5® hydrogen content assessment

2.3.3 ATF

No high burnup fuel from ATF irradiation currently exists. However, the Cr-coating applied to ATF cladding is expected to almost completely eliminate any build-up of oxide during irradiation. Likewise, the cladding is expected to not absorb any hydrogen during irradiation as this absorbed hydrogen is attributed to hydrogen liberated from the metal-water reaction, and this reaction is eliminated when the cladding surface is Cr-coated. There are no known mechanisms that would cause doped pellets to affect the cladding oxide thickness or hydrogen content.

2.4 Ex-reactor cladding creep

This section describes the assessment of ex-reactor cladding creep predictions of FAST-1.0. The formal assessment of FAST-1.0 does not focus on ex-reactor cladding creep. However, work was recently performed to identify and implement an ex-reactor cladding creep model into FAST to enable to use of FAST for spent fuel creep calculations. The assessment of the selected model is described in this section.

2.4.1 Standard Fuel

During the recent work performed to identify an ex-reactor creep model, a number of different models were compared to a significant database of creep data. After a review of these creep models for spent fuel applications, it was decided to retain the current FAST in-reactor cladding creep model for application to spent fuel. The in-reactor cladding creep model has two terms, a thermal creep term and an irradiation creep term. For ex-reactor applications, FAST will neglect the irradiation creep term. To assess the applicability of this model, the FAST correlation was compared to a dataset of creep tests (Nuclear Power Engineering Corporation, 2003) (Einziger, Tsai, Billone, & Hilton, 2003) (Tsai & Billone, 2006) that includes approximately 200 data points for both stress relief annealed (SRA) and fully recrystallized annealed cladding (RXA) and a range of tests that spanned from 30 MPa to 250 MPa constant hoop stress and 330°C - 420°C. Fluence of the samples ranged from unirradiated cladding to $1.4\text{x}10^{22}$ n/cm^2. This fluence corresponds to a burnup of approximately 75 GWd/MTU. This dataset covers the expected conditions of normal and off-normal storage and presents an excellent comparison of the creep correlation.

Figure 14 and Figure 15 show the FAST predictions of ex-reactor creep for RXA and SRA cladding respectively. The data is plotted up to 2% strain although it can be noted that there are a small number of tests that experienced creep outside these ranges. However, FAST predicts creep rupture beyond this strain level. The predicted and measured strains match well for the RXA data (average bias of -0.002% strain and a standard error of 0.25% strain). In the case of the SRA cladding there is a noticeable conservative bias in the predictions (average bias of 0.32% strain and a standard error of 0.27% strain). If the data are separated into those with temperature between 330-380°C and those from 390-420°C, it is observed that the bias exists mostly in the high temperature data. It is noted that the RXA data only goes up to a temperature of 390°C, so future work may focus on improving the FAST creep predictions in the 390°C to 420°C, although these temperatures are only expected to be seen during the initial vacuum drying operation and not for long periods of time. This bias is a conservative bias and not large enough to cause concern when applied for best estimate calculations except at strain levels outside the plotted range where FAST predicts failure.

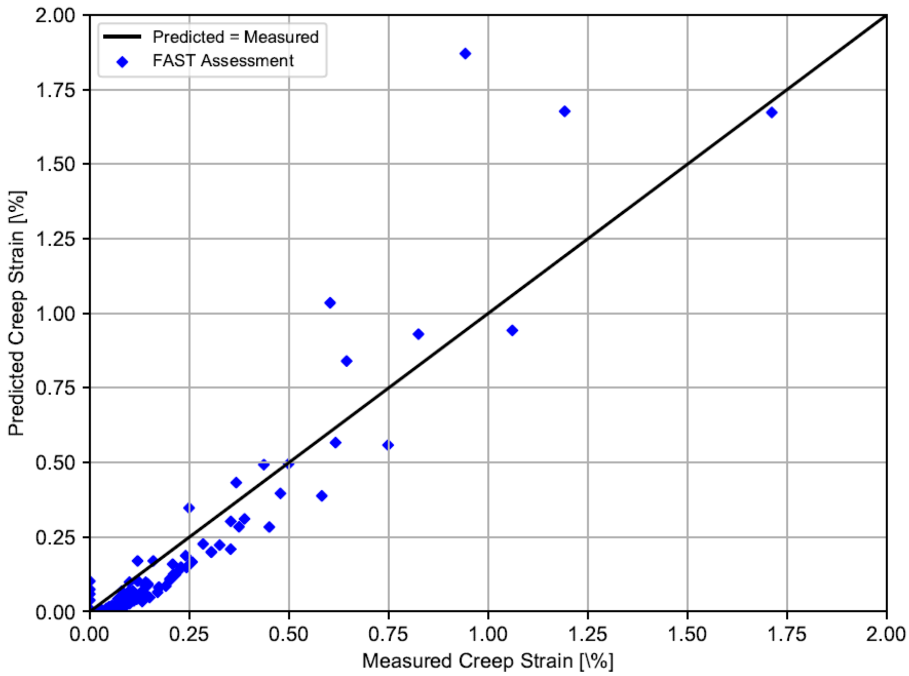


Figure 14. FAST ex-reactor creep comparisons for RXA cladding (Zircaloy-2, M5®)

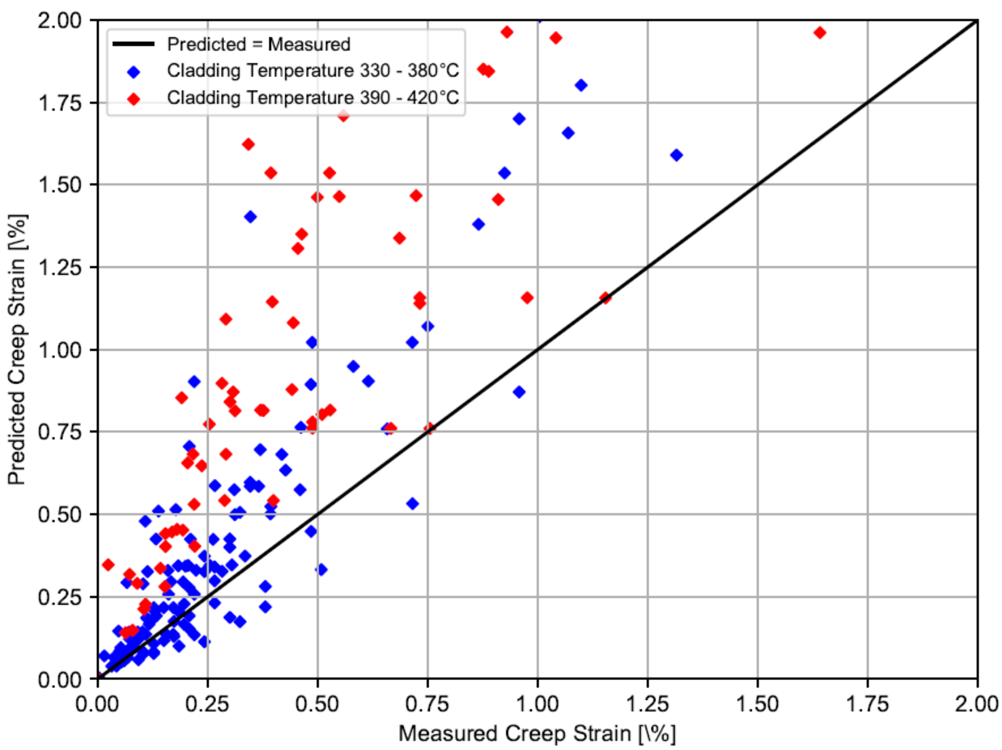


Figure 15. FAST ex-reactor creep comparisons for SRA cladding (Zircaloy-4, ZIRLO™)

2.4.2 ATF

The creep model in FAST has not been assessed for irradiated ATF cladding given the lack of data, although in general the thin coatings on the cladding in the coated cladding concepts are not expected to affect the creep rate. The coatings are expected to reduce the in-reactor corrosion and hydrogen release, so concerns regarding hydrogen embrittlement can be expected to be mitigated for ATF coated cladding. There are no known mechanisms that would cause doped pellets to affect the cladding ex-reactor creep rate.

3.0 New Data

A review of recent literature was performed to determine if any data are available from commercial fuel rod irradiation of standard or ATF fuel at high burnup (up to 75 GWd/MTU) and/or high enrichment.

There are no recent data from high enriched (5-20 wt% U-235) fuel. No commercial reactor in the U.S. is licensed to irradiate fuel with enrichment greater than 5 wt%. Historically some test reactor fuel was irradiated in reactors such as Halden and BR3, and these are the data that have been used to certify that FAST is validated from 2-12 wt% U-235. It is noted that the FAST predictions are not significantly impacted by enrichment. The only in-reactor model in FAST affected by enrichment is the radial power profile model which results in slightly different temperature profiles across the fuel pellet. This model (TUBRNP) has been explicitly validated between 2 and 8.25 wt% enrichment and 21-75 GWd/MTU, and temperature and gas release predictions have been reasonable between 2-12 wt% enrichment. There is no reason to assume that it would cease to be valid beyond these levels.

Regarding dry storage, the initial U-235 enrichment impacts the helium generation after discharge from the reactor. Higher initial U-235 enrichment will result in lower helium generation. The model that predicts this extra helium generation is based on ORIGEN calculations and is valid between 1-20% U-235.

For high burnup, there were several data sets that had not been included in the formal assessment of FAST-1.0. These data are primarily fission gas release data, although some rod internal pressure data was also found. No ex-reactor creep data on irradiated cladding have been found, although some ex-reactor creep data on unirradiated ATF data are available.

3.1 Fission Gas release

This section shows FAST predictions of new fission gas release data.

3.1.1 Very high burnup rods

Some rods with very high burnup up to 100 GWd/MTU were irradiated in a European reactor and fission gas release measurements were reported with sufficient information on power history to simulate these rods in FAST (Manzel & Walker, 2002). In particular, there are two rods irradiated to a rod-average burnup of 90 and 95 GWd/MTU that have matching fission gas release measurements. Other lower burnup representative rods were modeled and are shown relative to all the data from similar rods. These data and FAST predictions are shown in Figure 16. It can be seen that FAST provides adequate predictions of these rods even up to 95 GWd/MTU.

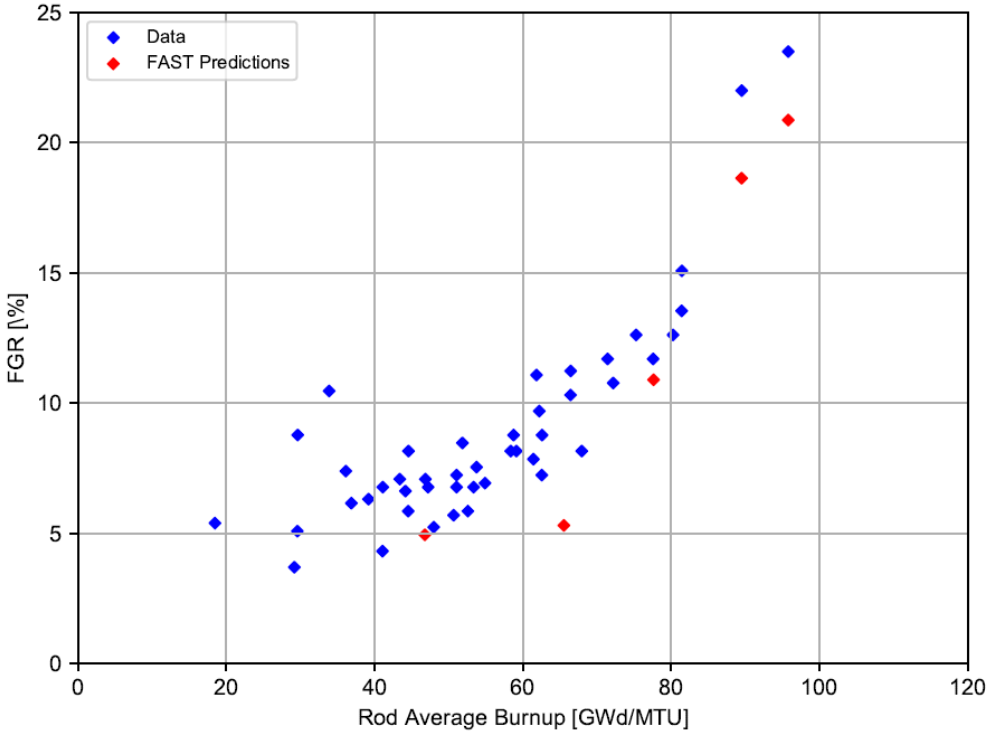


Figure 16. FAST fission gas release prediction of high burnup rods from Manzel and Walker

3.1.2 Sibling Rods

The U.S. Department of Energy (DOE) Office of Nuclear Energy, Spent Fuel & Waste Disposition under the Spent Fuel & Waste Disposition Campaign instituted the High Burnup Spent Fuel Data Project. This project obtains data to support the enhancement of the technical basis for the extended storage and transportation of high burnup (HBU) (>45 GWd/MTU) spent nuclear fuel (SNF). Under this project, a U.S. Nuclear Regulatory Commission (NRC)-licensed storage cask was loaded with 32 high burnup SNF assemblies. The cask, referred to as the Research Project Cask, was modified to allow radial and axial temperature profiles to be measured using thermocouple lances inserted through the lid. After a period of at least 10 years, the cask will be transported to a facility to be opened so the SNF can be examined and tested to provide confirmation of laboratory data that will be collected under this test plan.

In parallel with the 10-year storage of HBU SNF in the Research Project Cask, 25 HBU fuel rods have been removed either from assemblies going into the Research Project Cask or from assemblies with similar irradiation histories. These 25 “sibling rods,” have been characterized and tested (Montgomery, et al., 2018) (Montgomery, et al., 2019) (Shimskey, et al., 2019). Prior to testing the 25 “sibling rods” were modeled (Geelhood K. , 2018) using the NRC fuel performance code, FAST (Geelhood, et al., 2021). The blind FAST code predictions were later compared to the measurements of fission gas release that was been performed on the sibling rods.

The sibling rods have a rod average burnup between 50 and 60 GWd/MTU and include rods with Zircaloy-4, ZIRLO™ and M5® cladding. Figure 17 shows the FAST predictions of the sibling rods. FAST predicts the fission gas release from these rods well within the expected uncertainty range calculated based on the larger assessment database.

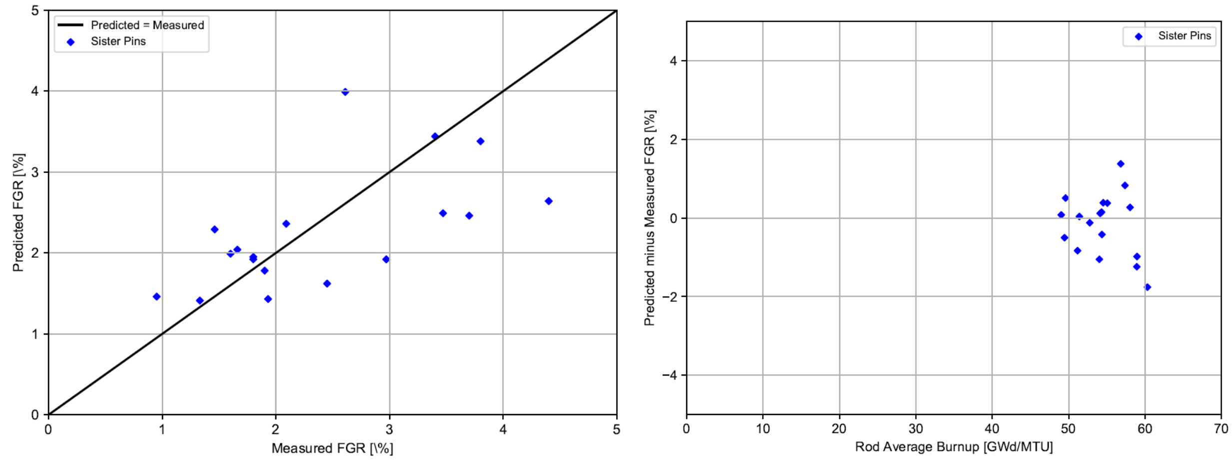


Figure 17. FAST fission gas release assessment for sibling rods

3.1.3 Ringhals 2 & 3 Data

Several rods irradiated in the Ringhals 2 & 3 reactor with measured fission gas release and power histories were provided to PNNL from Vattenfall for use in the void volume assessment. These rods are included in the void volume assessment but have not been included in the fission gas release assessment. These rods have rod-average burnup between 28 and 63 GWd/MTU and have Optimized ZIRLO™ and M5® cladding. Figure 18 shows the FAST predictions of these rods. FAST slightly underpredicts the fission gas release from these rods, but the results are within the expected uncertainty range calculated based on the larger assessment database.

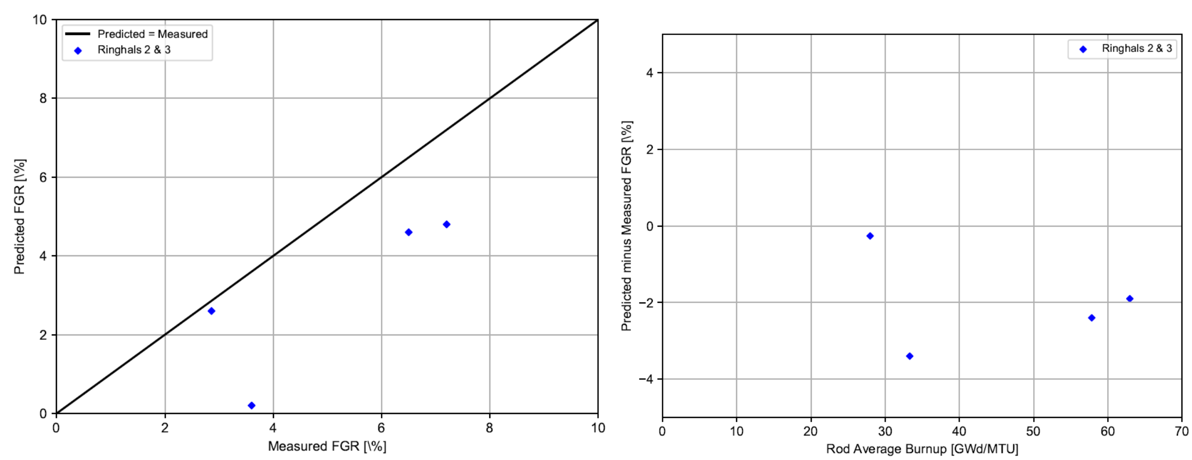


Figure 18. FAST fission gas release assessment for Ringhals 2 & 3 data

3.1.4 Additional Data

The literature search identified other data that did not include sufficient information to include in an assessment of FAST. For example, (Brankov, et al., 2016) presented high burnup BWR data irradiated to a burnup of 51 to 65 GWd/MTU. However, no power history information or other cycle data that could be used to estimate power histories was included so it is not possible to perform FAST calculations for these rods.

As previously mentioned, there is no high burnup puncture data from ATF fuel irradiated in commercial reactors. The most near-term data is expected to come from Gösgen nuclear power plant in Switzerland (Rebeyrolle, Vioujard, & Scholer, 2019). Beyond that, low burnup ATF data is being collected from US irradiations, and high burnup ATF data is still several years out.

3.2 Rod Internal Pressure

This section shows FAST predictions of new rod internal pressure. For analysis of spent fuel, rod internal pressure is the key parameter that drives cladding hoop stress as well as cladding creep strain.

3.2.1 Sibling Rods

The DOE sibling rods that were previously discussed for fission gas release (Section 3.1.2) also have measurements of end-of-life void volume and end-of-life rod internal pressure. The sibling rods have a rod average burnup between 50 and 60 GWd/MTU and include rods with Zircaloy-4, ZIRLO™ and M5® cladding. Figure 19 shows the FAST predictions of void volume and rod-internal pressure for the sibling rods. FAST predicts the fission gas release from these rods well within the expected uncertainty range calculated based on the larger assessment database.

Overall, the predictions of void volume are biased above the measurements (2.2 cm^3 on average) with a standard deviation of 2.6 cm^3. The expected uncertainty in void volume is around ±3 cm^3, so this bias is not outside the overall expected uncertainty, but it may indicate a systemic bias in the FAST predictions. It is noted that FAST predicts about 2.5 cm^3 remaining in the pellet dishes and 1.4 cm^3 in the pellet cracks. FAST does not currently have any model to predict dish filling or crack healing.

In contrast, the predictions of rod internal pressure are biased below the measurements (-0.53 MPa on average) with a standard deviation of 0.67 MPa. The expected uncertainty in void volume is around ±1 MPa, so this bias is not outside the overall expected uncertainty. This bias is very much in agreement the previous observations that the total moles of gas are predicted well, and the void volume is somewhat overpredicted. These two results in combination would lead to the expectation of rod internal pressure being somewhat underpredicted.

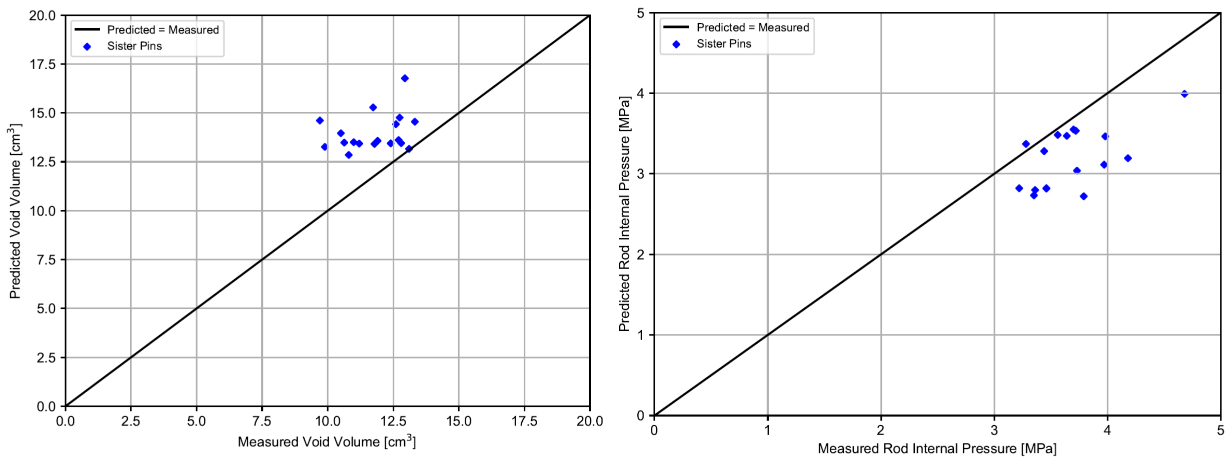


Figure 19. FAST void volume and rod internal pressure assessment for sibling rods

3.2.2 Additional Data

The literature search identified other data that did not include sufficient information to include in an assessment of FAST. For example, (Machiels, Rashid, Lyon, & Sunderland, 2017) presented a collection of rod internal pressure from a large database taken at a burnup between 25 and 75 GWd/MTU and attempted to derive correlations in this data relative to burnup and design. However, no power history information was included so it is not possible to perform FAST calculations for these rods.

There is no high burnup puncture data from ATF fuel irradiated in commercial reactors. The most near-term data is expected to come from Gösgen nuclear power plant in Switzerland (Rebeyrolle, Vioujard, & Scholer, 2019). Beyond that, low burnup ATF data is being collected from US irradiations, and high burnup ATF data is still several years out.

3.3 In-reactor Corrosion and Hydrogen Pickup

This section shows FAST predictions of new oxide thickness and hydrogen content data.

3.3.1 Optimized ZIRLO™ and AXIOM™

Recent oxide and hydrogen content data was published for Optimized ZIRLO™ and AXIOM™ (Pan, et al., 216). These data can be seen in Figure 20 and Figure 21. FAST-1.0 assumes that the oxide thickness and hydrogen content for Optimized ZIRLO™ is the same as for ZIRLO™ and these figures demonstrate that this is a reasonable assumption. For AXIOM™, these data indicate lower oxide thickness and similar hydrogen pickup fraction as for ZIRLO™. There is a proprietary version of FAST that includes appropriate models for AXIOM™.

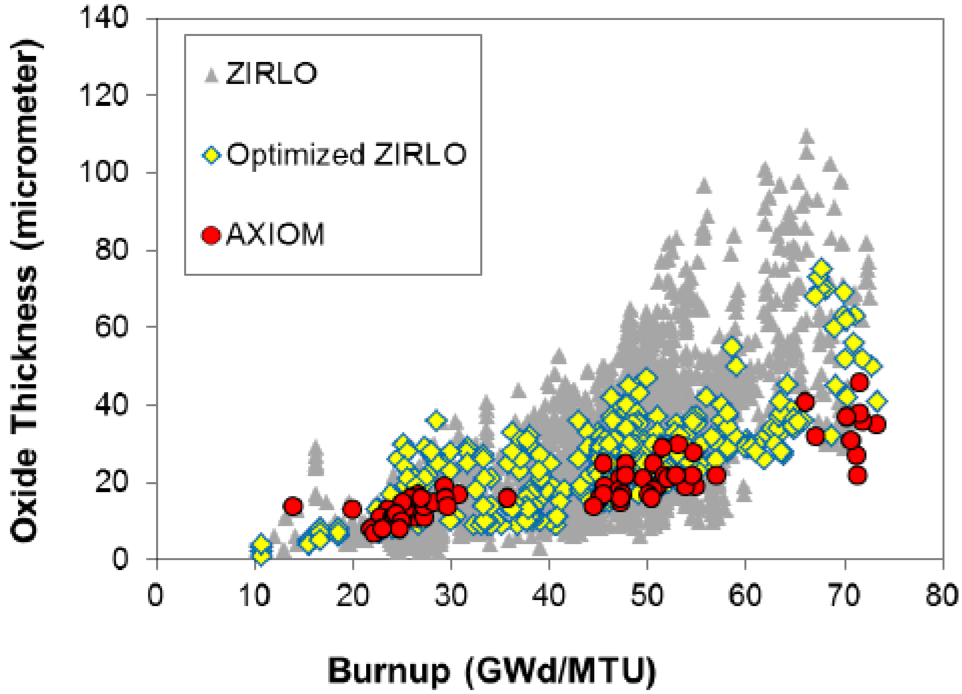


Figure 20. Oxide thickness data for Optimized ZIRLO™ and AXIOM™ (Pan, et al., 216)

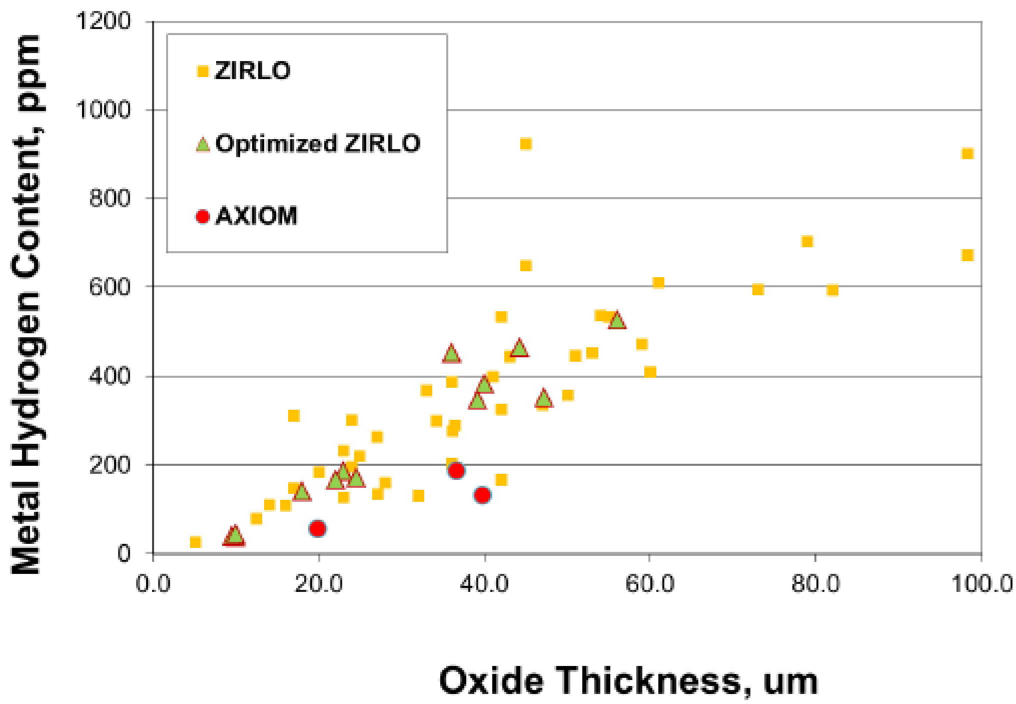
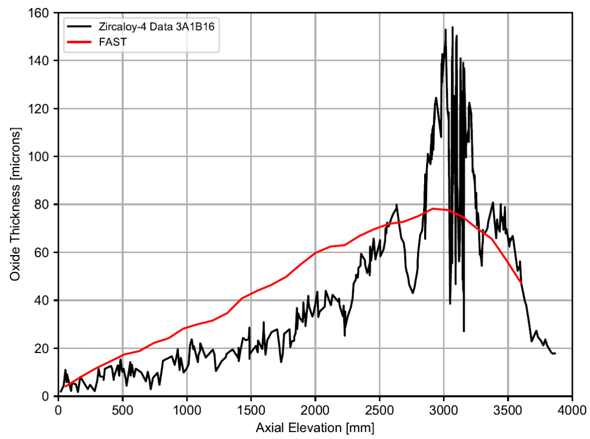


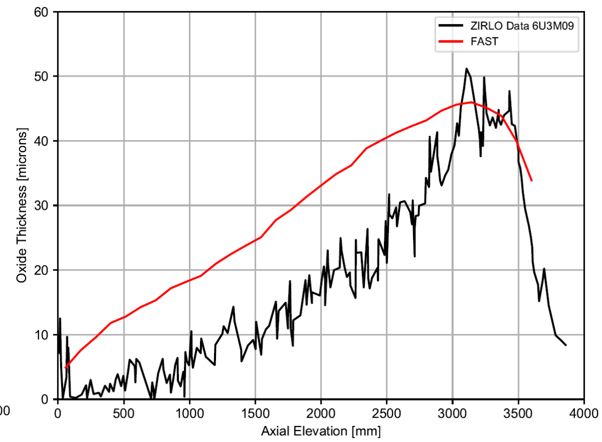
Figure 21. Hydrogen pickup data for Optimized ZIRLO™ and AXIOM™ (Pan, et al., 216)

3.3.2 Sibling Rods

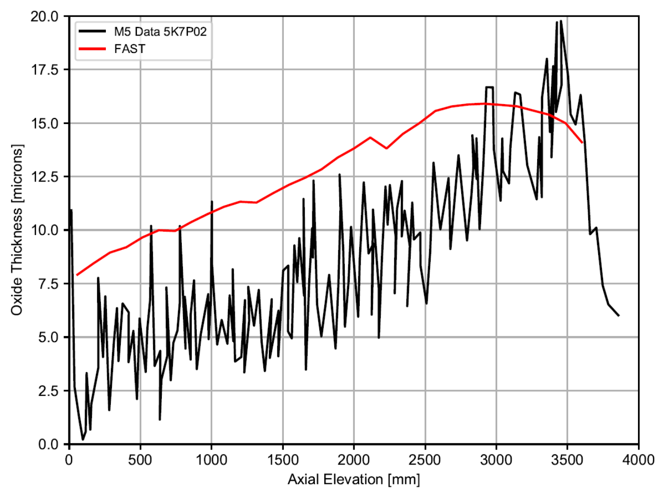
The DOE sibling rods that were previously discussed for fission gas release (Section 3.1.2) also have measurements of end-of-life oxide thickness. Hydrogen content measurements are currently in the process of being made. The sibling rods have a rod average burnup between 50 and 60 GWd/MTU and include rods with Zircaloy-4, ZIRLO™ and M5® cladding. Figure 22 shows the FAST predictions of oxide thickness for a representative Zircaloy-4, ZIRLO™, and M5® clad rod from the sibling rods. FAST predicts the oxide thickness from these rods well within the expected uncertainty range calculated based on the larger assessment database.



Zircaloy-4 (48 GWd/MTU)



ZIRLO™ (55 GWd/MTU)



M5® (51 GWd/MTU)

Figure 22. FAST comparison to oxide thickness from sibling rods

3.4 Creep Testing

No new data on ex-reactor creep testing of irradiated cladding have been found for either standard cladding or ATF cladding. However, some data have been found on unirradiated ATF coated cladding.

Recent data on unirradiated Cr-coated Zr indicate the thermal creep behavior of a coated part will be the same as that of uncoated cladding (Brachet, et al., 2017)[1]. Similar data have been taken for other mechanical properties such as elastic modulus, yield stress and burst pressure (Brachet, et al., 2017) (Kim, et al., 2015) (Shahin, Petrik, Seshadri, Phillips, & Shirvan, 2018).

Based on these preliminary observations it is reasonable to use the same ex-reactor creep correlation for coated and uncoated irradiated cladding.

[1] Creep tests on as-received samples were performed for 240 hours 400°C. The hoop stress was not reported, but significant strain (0.4% to 1.3%) was reported

4.0 New Models

Part of the ongoing maintenance that is performed on FAST is new model development or modifications to existing models when new data indicate that the FAST predictions may no longer be adequate for a new fuel or cladding type or a new range of operating condition such as burnup or enrichment. This section will describe how the models in FAST for standard fuel and for ATF have been updated to account for additional new data.

4.1 Standard Fuel

For standard fuel, the current models in FAST have been found to be applicable to high burnup and high enrichment. New models for AXIOM™ cladding have been added to a proprietary version of FAST. Beyond this, no new model development was performed. FAST has a choice of two fission gas release models and for burnup greater than 65 GWd/MTU the FRAPFGR fission gas release model continues to be the recommended model. As previously discussed, the only model in FAST that is dependent on uranium enrichment is the radial power profile which has been explicitly validated between 2 and 8.25% enrichment and 21-75 GWd/MTU and the FAST temperature and gas release predictions have been reasonable between 2-12% enrichment. There is no reason to assume that it would cease to be valid beyond these levels.

4.2 ATF

Since the release of FAST-1.0 new models have been developed and implemented into FAST to model Cr-coated cladding. These models will be available in the next release, FAST-1.1. In this version, if the user selects either Cr coating, or CrN coating, the code will use revised models for oxide thickness and hydrogen pickup that effectively eliminates both the growth of oxide and cladding hydrogen uptake. In addition, the code will consider the small temperature rise across the coating thickness.

In the new version for FAST the code will use the creep and axial growth rates for the base alloy that was selected assuming that the coating does not impact these rates. This is consistent with ex-reactor observations and can be confirmed during post-irradiation examination (PIE) of current lead-test assembly (LTA) irradiations.

To-date no modifications have been made to FAST to account for the use of Cr_2O_3 doped pellets, and the recommendation to use the standard UO_2 models still holds (Richmond & Geelhood, 2018).

5.0 FAST Assessment

This section will show the FAST predictions of the new data that have been identified in the context of the existing FAST assessment data. In general, the new data continue to confirm the applicability of the FAST predictions to high burnup and high enriched LWR fuel rods.

5.1 Fission Gas Release

Figure 23 shows the updated comparison of all the fission gas release data discussed in this report. The literature review resulted in 27 new end-of-life FGR comparisons. It can be seen that the additional data falls within the expected range of the original assessment data and provides more confidence in the FAST fission gas release predictions.

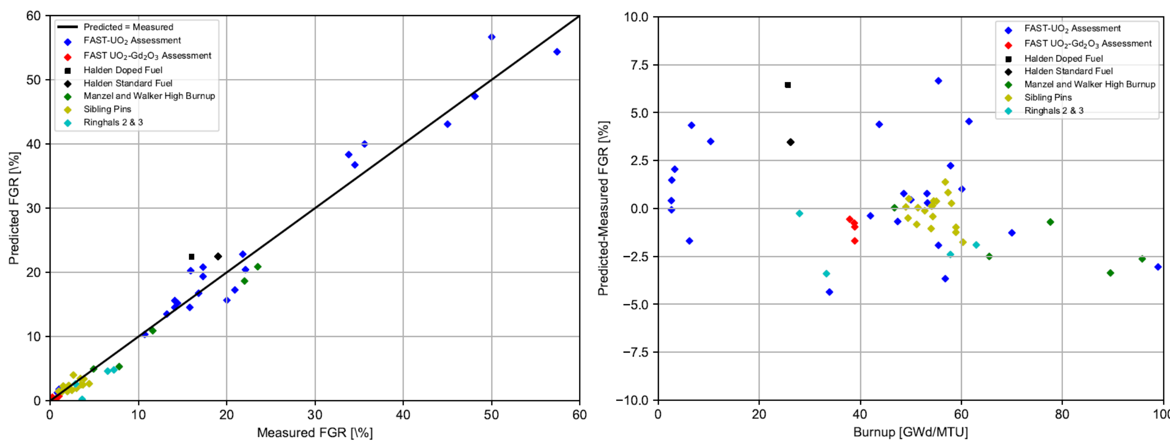


Figure 23. FAST updated fission gas release assessment for standard and ATF fuel

5.2 Rod Internal Pressure

Figure 24 shows the updated comparison of all the void volume data discussed in this report. The literature review resulted in 18 new end-of-life void volume and rod internal pressure comparisons.It can be seen that the additional data falls within the expected range of the original assessment data and provides more confidence in the FAST void volume predictions. Rod internal pressure was not part of the original FAST assessment but is driven primarily by fission gas release and void volume. Figure 24 also show the void volume predictions for the new data.

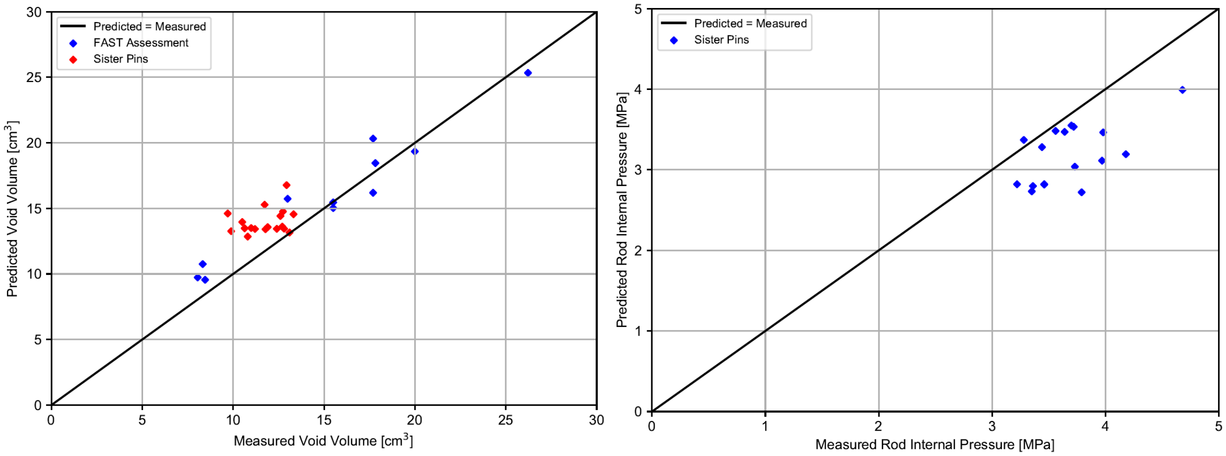


Figure 24. FAST updated void volume and rod internal pressure assessment for standard fuel

5.3 In-Reactor Corrosion and Hydrogen Pickup

The addition of new corrosion and hydrogen pickup data did not significantly alter the original assessment of FAST. There were nine M5™, twelve ZIRLO™, and five Zircaloy-4 new end-of-life axial oxide comparisons that were added, but they showed similar agreement to previous comparisons. Data showing oxide thickness and hydrogen pickup as a function of burnup to 75 GWd/MTU were added for Optimized ZIRLO™ and AXIOM™.

New models for AXIOM™ cladding have been added to a proprietary version of FAST. Additionally, data was provided to confirm assumptions that oxide thickness and hydrogen content for Optimized ZIRLO™ can be appropriately modeled using the ZIRLO™ models.

5.4 Ex-reactor Creep

No new data on ex-reactor creep of irradiated cladding have been found. Figure 25 and Figure 26 show the FAST predictions of creep for the existing database.

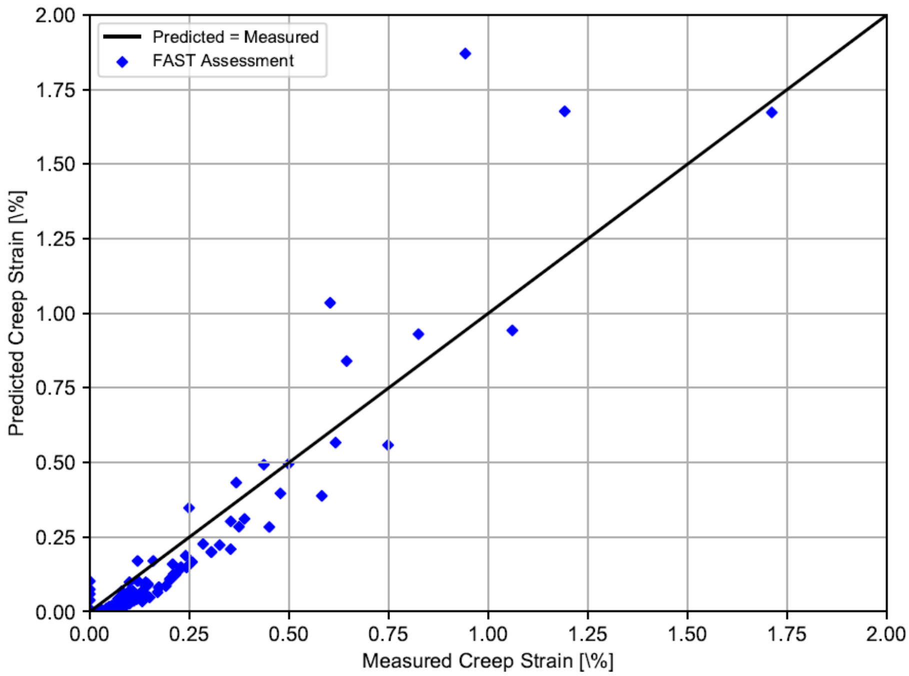


Figure 25. FAST ex-reactor creep comparisons for RXA cladding (Zircaloy-2, M5®)

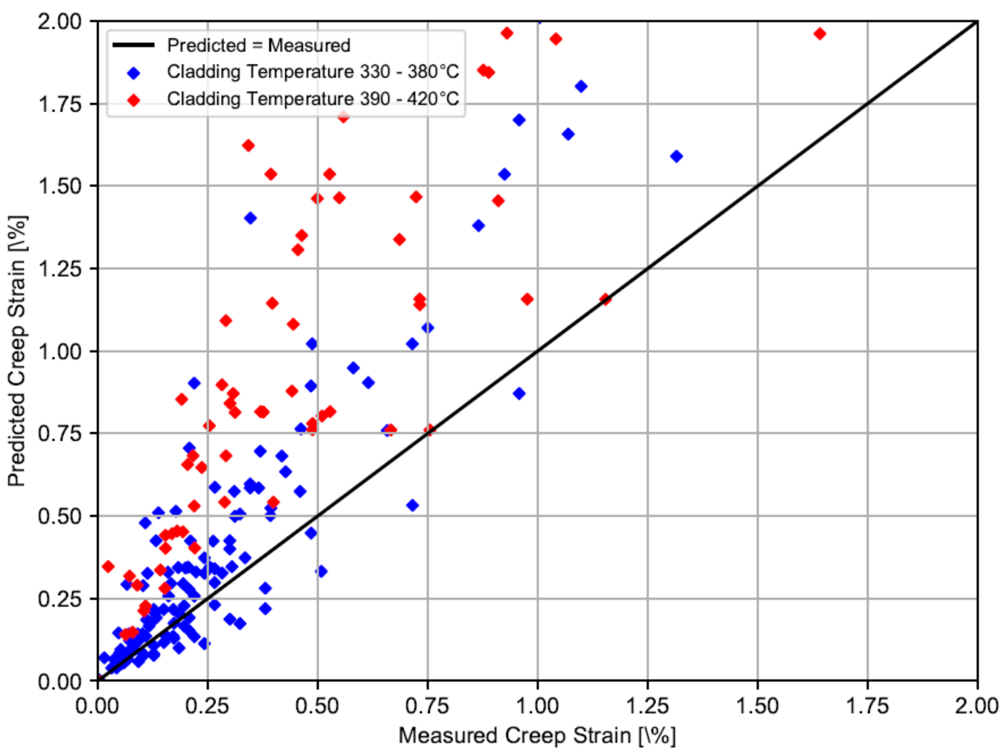


Figure 26. FAST ex-reactor creep comparisons for SRA cladding (Zircaloy-4, ZIRLO™)

6.0 Conclusion

A literature review was conducted to expand the assessment range of FAST-1.0. Although no data were found that challenge the existing FAST predictions, these data provide additional confidence in the FAST predictions.

6.1 FAST Validation for spent fuel analysis

The FAST fuel performance code is often used to provide end-of-life conditions such as fission gas release fraction and rod internal pressure for spent fuel analyses. Additionally, FAST is being developed to improve its ability to perform ex-reactor creep predictions and accident analyses for spent fuel. Because of these uses of FAST, it is important to identify the range of applicability of FAST, particularly if burnup levels and enrichment levels are expected to increase in the future and if ATF technology is deployed to commercial reactors. Based on the review of existing and new assessment data, the assessment ranges shown in Table 2 have been established for FAST in the use of spent fuel analysis.

Table 2. FAST burnup and enrichment assessment ranges for spent fuel analysis.

Parameter	*Standard Fuel[a]*	*UO_2-Gd_2O_3 Fuel*	*IFBA Fuel*	*Near-Term ATF Fuel[b]*
Fission Gas Release	75 GWd/MTU 2-12 wt% U-235	*Limited data to 40 GWd/MTU, expected same as UO_2*	*No data, expected same as UO_2*	*Limited data to 30 GWd/MTU, expected same as UO_2*
End-of-Life Room Temperature Rod Internal Pressure	60 GWd/MTU 2-12 wt% U-235	*No data, expected same as UO_2*	*No data, expected same as UO_2*	*No data, expected same as UO_2*
Cladding Oxide Thickness	52-68 GWd/MTU depending on alloy (See Table 1) Not impacted by enrichment	*Cladding oxide thickness not impacted by fuel*	*Cladding oxide thickness not impacted by fuel*	*No data. Oxide thickness expected to be minimal*
Cladding Hydrogen Content	70 GWd/MTU for all alloys Not impacted by enrichment	*Cladding hydrogen content not impacted by fuel*	*Cladding oxide thickness not impacted by fuel*	*No data. Hydrogen content expected to be minimal*
Ex-reactor Cladding Creep	75 GWd/MTU Not impacted by enrichment	*Cladding creep not impacted by fuel*	*Cladding creep not impacted by fuel*	*No data, expected same as standard cladding*

(a) Standard fuel is UO_2 with Zircaloy-2, Zircaloy-4, M5®, ZIRLO™, or Optimized ZIRLO™ cladding
(b) Near-term ATF fuel is Cr-coated Zr-alloy cladding with or without Cr_2O_3 doped UO_2 fuel. FeCrAl Cladding is more than 5 years from deployment and was not considered

6.1.1 Impact of Burnup on spent fuel behavior

Fuel burnup has a significant impact on spent fuel behavior. It can have a significant impact on the initial conditions of the spent fuel after irradiation. Fission gas release, which drives rod internal pressure and cladding stress exhibits an exponential increase above around 80 GWd/MTU (See Figure 16). Additionally, BWR cladding shows an exponential increase in hydrogen content above 50 GWd/MTU (See Figure 6) and some PWR cladding shows exponential increase in oxide thickness and hydrogen content above 70 or 80 GWd/MTU, although more modern alloys have been designed to reduce this oxide thickness at high burnup and would probably be those alloys irradiated to very high burnup.

With regard to additional changes to fuel behavior that occur during dry storage, fuel burnup has a smaller impact. Although fast neutron fluence (related to burnup) is observed to have some impact on cladding creep rate, the effect typically saturates after fluence corresponding to 40 to 50 GWd/MTU. The initial conditions such as oxide thickness, hydrogen content, and rod internal pressure are impacted by burnup as discussed above, beyond that decay heat and helium production from plutonium decay are impacted by burnup, but are well qualified as discussed below.

6.1.2 Impact of Enrichment on spent fuel behavior

Fuel enrichment does not greatly impact spent fuel behavior. It can have a small impact on the initial conditions of the spent fuel after irradiation. Enrichment impacts the radial power profile during irradiation which in turn can slightly impact the fuel temperature distribution and fission gas release. FAST has been assessed to provide adequate predictions up to 12% U-235 and there is no concern that FAST would not be able to provide adequate predictions up to 19.75% U-235.

With regard to additional changes to fuel behavior during dry storage, enrichment also has a small impact. The initial enrichment of the fuel impacts the helium production from plutonium decay and the overall decay heat. The decay heat should be determined using the ANS-5.1 standard (American Nuclear Society, 2019) which is well qualified over a wide range of burnup and initial enrichment and input into a cask thermal analysis to calculate rod temperature histories that are input into FAST. For helium production and release, recent improvements have been made to FAST including a model to predict helium production during dry storage. This model is based on ORIGEN calculations and there is no concern that it will not be able to provide adequate predictions up to 19.75% U-235.

6.2 Recommendations for additional data to support FAST validation for spent fuel analysis

Part of the ongoing maintenance that is performed on FAST is identification of new assessment data sets that can be used to improve the validation of FAST. The following section identifies areas relevant to spent fuel analysis where assessment data are lacking.

6.2.1 High Burnup

To improve the assessment of FAST at high burnup, puncture data (fission gas release, void volume, and pressure) are the key data needed. Along with puncture measurements it is important to include rod design information and at a minimum cycle lengths and cycle average power level (or burnup accumulation during each cycle). Data for the following fuel types are needed the most.

- UO_2 greater than 65 GWd/MTU
- UO_2-Gd_2O_3 at any burnup level
- IFBA at any burnup level
- Near-term ATF fuel at any burnup level

In the absence of new data, it is reasonable to continue to rely on FAST predictions of rod-internal pressure, hydrogen content and ex-reactor cladding creep. Rod internal pressure is driven primarily by FGR, fuel swelling, and cladding creep which are validated to greater than 75 GWd/MTU. There is no indication that the hydrogen pickup fraction will change from the constant value observed up to 70 GWd/MTU with the extension to 75 GWd/MTU. Although fast neutron fluence (related to burnup) is observed to have some impact on ex-reactor cladding creep rate, the effect typically saturates after fluence corresponding to 40 to 50 GWd/MTU, so little change to the creep rate is expected from 50-75 GWd/MTU.

6.2.2 High Enrichment

To improve the assessment of FAST at high enrichment, puncture data (fission gas release, void volume, and pressure) are the key data needed. Along with puncture measurements it is important to include rod design information and at a minimum cycle lengths and cycle average power level (or burnup accumulation during each cycle). Although FAST is qualified beyond 5% enrichment, there have been little commercial irradiations to support higher enrichments. The following fuel types would be useful to confirm FAST predictions.

- UO_2, UO_2-Gd_2O_3, IFBA, and Near-term ATF at greater than 5% enriched

6.2.3 ATF Fuel

The near-term ATF concepts under consideration for deployment within the next five years are small deviations from the current fuel system. Therefore, there is not a major concern that FAST cannot be used to predict the behavior of these fuel types. However, data confirming this assumption is critical. In addition to the puncture data mentioned in the previous two sections, ex-reactor creep tests performed at representative stress levels for any Cr-Coated cladding concept would provide excellent confirmation data for ATF fuel.

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