Pacific Northwest
NATIONAL LABORATORY

PNNL-29773

Fresh Fuel Transportation of Accident Tolerant Fuel Concepts

Chromium Coated Zirconium Alloy Cladding

March 2020

Kenneth J Geelhood

U.S. DEPARTMENT OF ENERGY

Prepared for the U.S. Department of Energy
under Contract DE-AC05-76RL01830

PACIFIC NORTHWEST NATIONAL LABORATORY
operated by
BATTELLE
for the
UNITED STATES DEPARTMENT OF ENERGY
under Contract DE-AC05-76RL01830

Printed in the United States of America

Available to DOE and DOE contractors from the
Office of Scientific and Technical Information,
P.O. Box 62, Oak Ridge, TN 37831-0062;
ph: (865) 576-8401
fax: (865) 576-5728
email: reports@adonis.osti.gov

Available to the public from the National Technical Information Service
5301 Shawnee Rd., Alexandria, VA 22312
ph: (800) 553-NTIS (6847)
email: orders@ntis.gov <https://www.ntis.gov/about>
Online ordering: http://www.ntis.gov

PNNL-29773

Fresh Fuel Transportation of Accident Tolerant Fuel Concepts

Chromium Coated Zirconium Alloy Cladding

March 2020

Kenneth J Geelhood

Prepared for
the U.S. Department of Energy
under Contract DE-AC05-76RL01830

Pacific Northwest National Laboratory
Richland, Washington 99354

Abstract

The U.S. Nuclear Regulatory Commission (NRC) is preparing for anticipated licensing applications and commercial use of accident tolerant fuel (ATF) in United States power reactors. Pacific Northwest National Laboratory (PNNL) has been tasked with providing technical assistance to the NRC related to the proposed new fuel and cladding designs. This report focuses specifically on the transportation of fresh (unirradiated) fuel with chromium metal and chromium compound coatings being investigated for the outer surface of Zr-alloy cladding. The U.S NRC is specifically concerned about metallic chromium coatings (8-30 μm) and a proprietary chromium ceramic coating known as ARMOR. This report provides the current state of industry information on material properties and fuel performance considerations for Cr-coated cladding concepts in fresh fuel transportation conditions. To support the agency's readiness efforts, this report will identify and discuss the implications of Cr-coated cladding on the material properties of the cladding at the relevant conditions to fresh fuel transportation. This report will also discuss any characteristics of Cr-coated cladding that may not be addressed within existing regulatory documents.

A previous report (Geelhood & Luscher, 2019) has already provided an overview of the Cr-coating concepts that are currently being developed both in the United States and around the world as well as an overview of various coating techniques that could be used. The previous report also discussed the impact that various Cr and Cr-based ceramic coatings could have on the cladding material properties and cladding safety limits. This current report will build on the prior report and extend the discussion of material properties to those relevant to fresh fuel transportation and will also discuss the impact of Cr-coatings on other safety concerns for fuel rods under fresh fuel transportation conditions.

Acknowledgments

This work was funded by the U.S. Nuclear Regulatory Commission under contract NRC-HQ-25-14-D-0001.

Acronyms and Abbreviations

ATF	Accident Tolerant Fuel
BCC	body-centered-cubic
BWR	boiling water reactor
CFR	Code of Federal Regulations
DOE	U.S. Department of Energy
FCC	face-centered cubic
GNF	Global Nuclear Fuels
HAC	Hypothetical Accident Conditions
LWR	Light Water Reactor
MOX	Mixed Oxide Fuel (U, PuO_2)
NCT	Normal Conditions of Transportation
NRC	U.S. Nuclear Regulatory Commission
PNNL	Pacific Northwest National Laboratory
PWR	pressurized water reactor
SRP	Standard Review Plan

Contents

Abstract.....ii
Acknowledgments.....iii
Acronyms and Abbreviations.....iv
Contents.....v
1.0 Introduction.....1
  1.1 Background.....2
  1.2 Existing Regulations and Regulatory Guidance.....3
    1.2.1 Regulations.....3
    1.2.2 Standard Review Plan.....6
2.0 Impact of Cr-Coated Zr Cladding on Fresh Fuel Transportation.....7
  2.1 Normal Conditions of Transportation.....7
  2.2 Hypothetical Accident Conditions.....10
3.0 Criticality.....13
4.0 Material Properties for Fresh Fuel Transportation.....14
  4.1 Cladding Thermal Properties.....14
    4.1.1 Thermal Conductivity.....14
    4.1.2 Thermal Expansion.....15
    4.1.3 Emissivity.....16
  4.2 Cladding Mechanical Properties.....17
    4.2.1 Elastic Modulus.....17
    4.2.2 Yield Stress and Ultimate Tensile Strength.....18
    4.2.3 Ductility.....19
  4.3 Cladding Fatigue.....21
5.0 Conclusions.....22
6.0 References.....23

Figures

Figure 1. Thermal Conductivity of Zircaloy, Chromium, and Various Chromium-Based Ceramics.....15
Figure 2. Emissivity of Zircaloy, Chromium, and Various Chromium-Based Ceramics.....17

Tables

Table 1. Comparison of Cr-coated concepts being pursued by U.S. Nuclear Fuel Vendors. 2
Table 2. Requirements on normal conditions of transportation and impact on fresh fuel transportation due to changing from Zr-alloy cladding to Cr-coated Zr cladding. 8
Table 3. Requirements on hypothetical accident conditions and impact on fresh fuel transportation due to changing from Zr-alloy cladding to Cr-coated Zr cladding. 11
Table 4. Coating Adhesion for Various Coating Concepts 16
Table 5. Elastic Modulus for Cr-coated Cladding 18
Table 6. Yield Stress and Ultimate Tensile Strength for Cr-coated Cladding....18
Table 7. Ductility for Cr-coated Cladding....19
Table 8. Interdiffusion Layer for Cr-coated Zr during NCT and HAC 21
Table 9. Fatigue for Cr-coated Cladding 21

1.0 Introduction

The U.S. Nuclear Regulatory Commission (NRC) is preparing for anticipated licensing applications and commercial use of accident tolerant fuel (ATF) in United States power reactors. Several fuel vendors, in coordination with the U.S. Department of Energy (DOE), have announced plans to develop and seek approval for various fuel designs with enhanced accident tolerance (i.e., fuels with longer coping times during loss of cooling conditions). The designs being considered by industry and DOE include chromium (Cr) coated claddings, chromium trioxide (Cr_2O_3)-doped uranium dioxide (UO_2) pellets, iron-chromium-aluminum (FeCrAl) cladding, silicon carbide (SiC) cladding, uranium disilicide (U_3Si_2) pellets, and metallic fuels. These designs represent evolutions and deviations from the de facto standard zirconium alloy clad, uranium dioxide fuel form. Most of the NRC's regulatory framework for transportation of fresh nuclear fuel was developed specifically for zirconium-alloy clad, UO_2 fuel and is primarily applicable to this system. Therefore, a review of the technical challenges associated with new fuel designs would assist the NRC in reviewing upcoming applications for transport of fresh fuel.

Pacific Northwest National Laboratory (PNNL) has been tasked with providing technical assistance to the NRC related to the proposed new fuel and cladding designs. This report and others like it provide the agency with expert technical assistance to enhance the staff's knowledge base of specific ATF concepts and supports the agency's efforts to develop and review the required regulatory infrastructure to support the development of ATF.

This report provides current state of the industry information on material properties and fuel performance considerations for Cr-coated cladding concepts in fresh fuel transportation conditions. To support the agency's efforts, this report identifies and discusses the implications of Cr-coated cladding on the material properties of the cladding at the relevant conditions to fresh fuel transportation. This report also discusses any characteristics of Cr-coated cladding that may not be addressed within existing regulatory documents.

The scope of this report includes metallic coatings of chromium as well as any ceramic coatings that are in development for ATF claddings. The U.S NRC is specifically concerned about metallic chromium coatings (8-30 μm) and a proprietary chromium ceramic coating known as ARMOR. This entire class of concepts will be generically referred to as "Cr-coated Zr" throughout this report. Table 1 provides a high-level overview of the U.S. Cr-coated cladding concepts. This report provides an assessment of the impact of the substitution of typical Zr-alloy cladding with Cr-coated Zr cladding on the requirements that have been placed on the transport of fresh nuclear fuel. The remainder of this section discusses the applicable regulations and standard review plan for the transportation of fresh fuel. Section 2.0 describes the impact of Cr-coated Zr cladding on fresh fuel transport. Section 3.0 describes criticality considerations for Cr-coated Zr cladding relative to Zr-alloy cladding. Section 4.0 discusses material property differences that should be considered for Cr-coated Zr relative to Zr-alloy cladding. Overall conclusions are given in Section 5.0.

Table 1. Comparison of Cr-coated concepts being pursued by U.S. Nuclear Fuel Vendors.

Vendor	Coating	Application Process	Coating Thickness*
Westinghouse	Cr-coated ZIRLO™	Cold spray and polishing	20-30 μm
Framatome	Cr-coated M5®	PVD	8-22 μm
GNF	ARMOR[1] coated Zircaloy-2	*proprietary*	*Proprietary*

*May change by the time of application. Typical cladding thickness is 600-750 μm.

1.1 Background

The 2011 Great East Japan Earthquake and Tsunami, and the events that followed at the Fukushima Daiichi power plant led to a worldwide interest in development of fuels with enhanced performance during such rare events. In response, ATF development programs were started in many research institutions and industry teams. A new fuel in combination with other systems may provide some margin under accident conditions and provide additional benefits during anticipated operational occurrences and normal operations.

For light water reactors (LWRs) the cladding has historically been fabricated from zirconium alloys. For boiling water reactors (BWRs) the alloy Zircaloy-2 has been used. For pressurized water reactors (PWRs) the alloy Zircaloy-4 has been used. PWR and BWR cladding is typically between 0.56 and 0.75 mm thick. As demand for higher burnup levels came for LWR fuels, nuclear fuel vendors have developed proprietary, Zr-based cladding alloys that have mostly replaced the use of traditional Zircaloy alloys. Westinghouse now uses the alloys ZIRLO™ and Optimized ZIRLO™[2] for their PWR fuel, while retaining Zircaloy-2 for BWR fuel. Framatome uses M5®[3] for their PWR fuel, while also retaining Zircaloy-2 for BWR fuel. Global Nuclear Fuels (GNF) only supplies BWR fuel and has recently received approval for GNF-Ziron cladding.

ATF cladding is being developed primarily to give an advantage during high temperature oxidation that may occur following an in-reactor design basis accident or in a situation considered to be beyond the fuel design basis. In addition to providing this advantage, ATF cladding must meet the general set of requirements placed on nuclear fuel cladding during the transport of fresh fuel. For example, the fresh fuel package has requirements for containment, shielding, and maintaining sub-critical geometry under normal conditions of transportation and hypothetical accidents.

PNNL has reviewed existing regulations and guidance related to transportation of fresh nuclear fuel and have found them to be adequate for the transportation of fresh fuel with Cr-coated Zr cladding. These regulations and guidance are discussed in the following section.

[1] ARMOR coating is a proprietary ceramic coating. The thickness and ceramic material are proprietary, so this report includes discussion of several ceramic coatings or ARMOR when stated to be ARMOR.
[2] ZIRLO™ and Optimized ZIRLO™ are trademarks or registered trademarks of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States and may be registered in other countries throughout the world.
[3] M5® is a trademark or registered trademark of Framatome or its affiliates, in the USA or other countries.

1.2 Existing Regulations and Regulatory Guidance

The regulations related to the transportation of fresh nuclear fuel are contained in 10 CFR Part 71 (U.S. Nuclear Regulatory Commission, 2015). The regulations specify several types of packages that may be used to transfer radioactive material. There are two types of packages that can be used to transport material with a significant concentration of radioactivity. These are Type A and Type B packaging.

Type A packaging is used to transport limited amounts of radioactive material, which do not exceed specific activity limits defined in 10 CFR Part 71 (U.S. Nuclear Regulatory Commission, 2015). The limit of a Type A quantity is given in 10 CFR 71.4 and Appendix A of 10 CFR Part 71. Type A packaging and its radioactive contents must meet standard testing requirements designed to ensure that the package retains its containment integrity and shielding under normal transportation conditions.

Type B packaging is designed to transport material with greater than a Type A quantity of radionuclides. These package designs must withstand all Type A tests, and a series of tests that simulate severe or “worst-case” accident conditions. Accident conditions are simulated by performance testing and engineering analysis.

Except for MOX fuel, the transport of light water reactor fuel assemblies is performed using Type A packages. However, since light water reactor fuel assemblies contain fissile materials in excess of those designed in 10 CFR Part 71.15, these must be shipped in a Type A fissile material package, Type AF. The following section discusses the additional requirements for a Type AF package.

1.2.1 Regulations

10 CFR 71 (U.S. Nuclear Regulatory Commission, 2015) describes the regulations that govern the transport of radioactive material. The following sections are relevant to the transportation of fresh fuel.

> ***71.41 Demonstration of compliance:*** *The effects on a package of the tests specified in § 71.71 ("Normal conditions of transport"), and the tests specified in § 71.73 ("Hypothetical accident conditions"), and § 71.61 ("Special requirements for Type B packages containing more than 105 A2"), must be evaluated by subjecting a specimen or scale model to a specific test, or by another method of demonstration acceptable to the Commission, as appropriate for the particular feature being considered.*

This section describes the general types of analysis that should be performed:

> ***71.43 General Standards for all packages:*** *A package must be designed, constructed, and prepared for shipment so that under the tests specified in § 71.71 ("Normal conditions of transport") there would be no loss or dispersal of radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging.*

For Type A packages, only the tests specified in 10 CFR 71.71 are required to ensure that there is no loss or dispersal of the radioactive material within the package. However, for Type AF packages such as will be used to transport fresh fuel, the package must also be subjected to the accident tests from 10 CFR 71.73 as will be discussed under 10 CFR 71.55 below.

> ***71.51 Additional requirements for Type B packages:***

This section is not applicable to Type AF packages.

> ***71.55 General Requirements for Fissile Material Packages (in part):*** *(a) A package used for the shipment of fissile material must be designed and constructed in accordance with §§ 71.41 through 71.47. When required by the total amount of radioactive material, a package used for the shipment of fissile material must also be designed and constructed in accordance with § 71.51.*
>
> *(b) Except as provided in paragraph (c) or (g) of this section, a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained.*
>
> 1. *The most reactive credible configuration consistent with the chemical and physical form of the material;*
> 2. *Moderation by water to the most reactive credible extent; and*
> 3. *Close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging.*
>
> *(c) The Commission may approve exceptions to the requirements of paragraph (b) of this section if the package incorporates special design features that ensure that no single packaging error would permit leakage, and if appropriate measures are taken before each shipment to ensure that the containment system does not leak.(d) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in § 71.71 ("Normal conditions of transport") --*
>
> 1. *The contents would be subcritical;*
> 2. *The geometric form of the package contents would not be substantially altered;*
> 3. *There would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under § 71.59(a)(1), it has been assumed that moderation is present to such an extent as to cause*

maximum reactivity consistent with the chemical and physical form of the material; and

4. *There will be no substantial reduction in the effectiveness of the packaging, including:*

   a. *No more than 5 percent reduction in the total effective volume of the packaging on which nuclear safety is assessed;*

   b. *No more than 5 percent reduction in the effective spacing between the fissile contents and the outer surface of the packaging; and*

   c. *No occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm (4 in) cube.*

(e) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in § 71.73 ("Hypothetical accident conditions"), the package would be subcritical. For this determination, it must be assumed that:

1. *The fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents;*

2. *Water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents; and*

3. *There is full reflection by water on all sides, as close as is consistent with the damaged condition of the package*

This section specifies that because fresh fuel contains fissile materials it is transported in a Type AF package and some extra analyses must be performed for hypothetical accident conditions to ensure that the package will be subcritical under normal conditions of transport and hypothetical accident conditions. Therefore, for fresh fuel transportation, both normal conditions of transport and hypothetical accident conditions must be considered from a criticality perspective.

71.71 Normal Conditions of Transport

This section defines the conditions and tests used to represent normal conditions of transport (NCT). These are described later in Section 2.1.

71.73 Hypothetical Accident Conditions

This section defines the conditions and tests used to represent hypothetical accident conditions (HAC). These are described later in Section 2.2.

1.2.2 Standard Review Plan

The NRC has provided a standard review plan (SRP) , (U.S. Nuclear Regulatory Commission, 1999), to assist NRC staff in the review and approval of applications for packages used to transport radioactive material (other than irradiated nuclear fuel). This guidance is also used by applicants in producing these applications. The SRP summarizes 10 CFR Part 71 requirements for package approval, describes the procedures by which the NRC staff determines that these requirements have been satisfied, and documents the practices developed by the staff in previous reviews of package applications.

Section 4.5.2.1 of NUREG-1609 provides general considerations for Type A Fissile Packages (Type AF) and Appendix A3 of NUREG-1609 is particularly relevant as it describes unirradiated fuel packages.

The regulations in 10 CFR 71 and the review guidance in NUREG-1609 will be used in the following sections to determine what data or analytical needs there are for the transport of fresh fuel with Cr-coated Zr cladding beyond what has been previously been done for Zr-alloy cladding.

2.0 Impact of Cr-Coated Zr Cladding on Fresh Fuel Transportation

In a previous report (Geelhood & Luscher, 2019), the impact of changing from Zr-alloy cladding to Cr-coated Zr cladding was examined from an in-reactor perspective. The requirements and data needs for fresh fuel transport are different from those needed for in-reactor performance. Because the fuel has not been irradiated, irradiated material properties for fuel and cladding are not needed as they are for in-reactor performance. However, some additional testing is required to account for the different requirements for fresh fuel transport. This section will examine the normal conditions of transport and hypothetical accident conditions specified by 10 CFR 71.71 and 71.73. For each condition or requirement, the impact of changing the cladding to Cr-coated cladding will be evaluated. Following this evaluation, it can be determined what data or modeling needs are required for the analysis of transportation of fresh fuel with Cr-coated cladding.

One general conclusion from the previous report is that for both elastic modulus, and yield stress, the mechanical properties are essentially the same for unirradiated coated and uncoated cladding. (Brachet, et al., 2017) (Kim, et al., 2015) (Shahin, Petrik, Seshadri, Phillips, & Shirvan, 2018). This conclusion should be confirmed by each applicant but is used in the assessments made in the following section. Data comparisons to support these assessments are shown in Section 4.0.

2.1 Normal Conditions of Transportation

NUREG-1609 specifies that for normal conditions of transport the following analyses should be performed.

- A structural analysis to ensure no loss or dispersal of radioactive material.
- A criticality analysis to ensure subcriticality.

Table 2 lists the requirements on a fresh fuel package for normal conditions of transport. Also included in this table is an assessment of the impact of changing the cladding from Zr-alloy to Cr-coated Zr. The conclusions of this table are that for the analysis of normal conditions of transport, a fatigue lifetime curve from representative cladding should be used in place of the standard Zr fatigue lifetime curve. Additionally, if the Cr-coating impacts the cladding yield stress, this yield stress should be used in the cask free drop and corner drop to ensure no damage to the fuel rods.

Table 2. Requirements on normal conditions of transportation and impact on fresh fuel transportation due to changing from Zr-alloy cladding to Cr-coated Zr cladding.

Requirement	Impact of changing Zr-alloy to Cr-coated Zr (valid for coatings in Table 1)
Initial conditions: ambient temperature preceding and following the tests remains constant at a value between -29°C (-20°F) and +38°C (+100°F) whichever is most unfavorable for the feature under consideration.	Fuel analysis not expected to be impacted. Mechanical testing of unirradiated coated cladding shows negligible impact of the coating on the mechanical properties of coated cladding relative to uncoated cladding. See Section 4.2.
Heat: An ambient temperature of 38°C (100°F) in still air.	Fuel analysis not expected to be impacted. Mechanical testing of unirradiated coated cladding shows negligible impact of the coating on the mechanical properties of coated cladding relative to uncoated cladding. See Section 4.2.
Cold: An ambient temperature of -40°C (-40°F) in still air and shade.	Fuel analysis not expected to be impacted. Mechanical testing of unirradiated coated cladding shows negligible impact of the coating on the mechanical properties of coated cladding relative to uncoated cladding. See Section 4.2. Typical concern is regarding ductile to brittle transition temperature. Zr does not exhibit a ductile to brittle transition. Cr metal is brittle below about 150°C. Ceramic Cr-coatings are brittle at all temperatures. Thin coatings of both brittle metallic (eg. Cr) and ceramic materials can accommodate greater strain than bulk samples of the coating materials. See Section 4.2.3
Reduced external pressure: An external pressure of 25 kPa (3.5 lbf/in^2) absolute.	No impact of this requirement on the fuel since it is inside a sealed cask.
Increased external pressure: An external pressure of 140 kPa (20 lbf/in^2) absolute.	No impact of this requirement on the fuel since it is inside a sealed cask.
Vibration: Vibration normally incident to transport.	Cr-coating may reduce cladding fatigue lifetime. Fatigue lifetime curve for representative Cr-coated Zr should be developed and used in this assessment. See Section 4.3.
Water spray. A water spray that simulates exposure to rainfall of approximately 5 cm/h (2 in/h) for at least 1 hour.	No impact of this requirement on the fuel since it is inside a sealed cask.

Requirement	Impact of changing Zr-alloy to Cr-coated Zr (valid for coatings in Table 1)
Free drop. Between 1.5 and 2.5 hours after the conclusion of the water spray test, a free drop through the distance specified below onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected.	For an analysis using a stress-based approach, fuel cladding performance is not expected to be impacted. Mechanical testing of unirradiated coated cladding shows negligible impact of the coating on the mechanical properties of coated cladding relative to uncoated cladding. For a specific application, if this is not the case, yield stress from representative Cr-coated Zr should be used in this analysis to ensure the fuel rods are not damaged. See Section 4.2. Analyses using a strain-based approach have not been fully qualified, but it appears that special mechanical tests are necessary to certify each cladding material. These tests should be performed on Cr-coated cladding to make assessments regarding the acceptability of Cr-coated cladding in a strain-based approach.
Corner drop. A free drop onto each corner of the package in succession, or in the case of a cylindrical package onto each quarter of each rim, from a height of 0.3 m (1 ft) onto a flat, essentially unyielding, horizontal surface. This test applies only to fiberboard, wood, or fissile material rectangular packages not exceeding 50 kg (110 lbs) and fiberboard, wood, or fissile material cylindrical packages not exceeding 100 kg (220 lbs).	For an analysis using a stress-based approach, fuel cladding performance is not expected to be impacted. Mechanical testing of unirradiated coated cladding shows negligible impact of the coating on the mechanical properties of coated cladding relative to uncoated cladding. For a specific application, if this is not the case, yield stress from representative Cr-coated Zr should be used in this analysis to ensure the fuel rods are not damaged. See Section 4.2. Analyses using a strain-based approach have not been fully qualified, but it appears that special mechanical tests are necessary to certify each cladding material. These tests should be performed on Cr-coated cladding to make assessments regarding the acceptability of Cr-coated cladding in a strain-based approach.

Requirement	Impact of changing Zr-alloy to Cr-coated Zr (valid for coatings in Table 1)
Compression. For packages weighing up to 5000 kg (11,000 lbs), the package must be subjected, for a period of 24 hours, to a compressive load applied uniformly to the top and bottom of the package in the position in which the package would normally be transported. The compressive load must be the greater of the following: • The equivalent of five times the weight of the package; or • The equivalent of 13 kPa (2 lbf/in^2) multiplied by the vertically projected area of the package.	No impact of this requirement on the fuel. This is a cask requirement.
Penetration. Impact of the hemispherical end of a vertical steel cylinder of 3.2 cm (1.25 in) diameter and 6 kg (13 lbs) mass, dropped from a height of 1 m (40 in) onto the exposed surface of the package that is expected to be most vulnerable to puncture. The long axis of the cylinder must be perpendicular to the package surface.	If the loads on the package are not significant enough to cause deformation in the fuel, there will be no impact of this requirement on the fuel. If a package design is such that these loads cause deformation in the fuel, mechanical testing of unirradiated coated cladding shows negligible impact of the coating on the mechanical properties of coated cladding relative to uncoated cladding. See Section 4.2.

2.2 Hypothetical Accident Conditions

NUREG-1609 Appendix A3 specifies that for hypothetical accident conditions the following analyses should be performed.

- A structural analysis should address possible damage to the package, fuel assembly, and neutron poisons to ensure the fuel assemblies and neutron poisons are maintained in a fixed position relative to each other and confirm the minimum spacing between fuel assemblies for criticality concerns
- A thermal analysis should evaluate the effect of fire on neutron poisons and other temperature-sensitive materials for criticality concerns
- A criticality analysis to ensure subcriticality.

Table 3 lists the requirements on a fresh fuel package for hypothetical accident conditions. Also included in this table is an assessment of the impact of changing the cladding from Zr-alloy to Cr-coated Zr. The conclusions of this table are that for the analysis of hypothetical accident conditions if the Cr-coating impacts the cladding yield stress, this yield stress should be used in the package free drop test to assess the impact on the fuel geometry for the criticality assessment. Also, the impact of the Cr-coating should be included in the criticality assessment but may be dispositioned if it can be shown that the Cr-coated Zr has the same or greater neutron absorption cross section as Zr.

Table 3. Requirements on hypothetical accident conditions and impact on fresh fuel transportation due to changing from Zr-alloy cladding to Cr-coated Zr cladding.

Requirement	Impact of changing Zr-alloy to Cr-coated Zr (valid for coatings in Table 1)
Initial conditions: except for the water immersion tests, ambient temperature preceding and following the tests remains constant at a value between -29°C (-20°F) and +38°C (+100°F) whichever is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be the maximum normal operating pressure, unless a lower internal pressure, consistent with the ambient temperature assumed to precede and follow the tests, is more unfavorable.	Fuel analyses are not expected to be impacted. Mechanical testing of unirradiated coated cladding shows negligible impact of the coating on the mechanical properties of coated cladding relative to uncoated cladding. See Section 4.2.
Free Drop: A free drop of the specimen through 9 m (30 ft) onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected.	For an analysis using a stress-based approach, fuel cladding performance is not expected to be impacted. Mechanical testing of unirradiated coated cladding shows negligible impact of the coating on the mechanical properties of coated cladding relative to uncoated cladding. For a specific application, if this is not the case, yield stress from representative Cr-coated Zr should be used in this analysis to ensure the fuel rods are not damaged. See Section 4.2. Analyses using a strain-based approach have not been fully qualified, but it appears that special mechanical tests are necessary to certify each cladding material. These tests should be performed on Cr-coated cladding to make assessments regarding the acceptability of Cr-coated cladding in a strain-based approach.
Crush. Subjection of the specimen to a dynamic crush test by positioning the specimen on a flat, essentially unyielding horizontal surface so as to suffer maximum damage by the drop of a 500-kg (1100-lb) mass from 9 m (30 ft) onto the specimen.	No impact of this requirement on the fuel. This is a cask requirement.
Puncture. A free drop of the specimen through 1 m (40 in) in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface.	No impact of this requirement on the fuel. This is a cask requirement.

Requirement	Impact of changing Zr-alloy to Cr-coated Zr (valid for coatings in Table 1)
Thermal. Exposure of the specimen fully engulfed, except for a simple support system, in a hydrocarbon fuel/air fire of sufficient extent, and in sufficiently quiescent ambient conditions, to provide an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 800°C (1475°F) for a period of 30 minutes, or any other thermal test that provides the equivalent total heat input to the package and which provides a time averaged environmental temperature of 800°C.	Fuel analyses are not expected to be impacted. There is no negative impact of Cr-coating at this temperature. The Cr-coating is designed to provide more corrosion resistance at this temperature. Additionally data from coated cladding shows improved high temperature creep and ballooning behavior relative to uncoated tubes (Dumerval, et al., 2018) (Delafoy, et al., 2018) (Brachet, et al., 2017)
Immersion--fissile material. For fissile material subject to § 71.55, in those cases where water inleakage has not been assumed for criticality analysis, immersion under a head of water of at least 0.9 m (3 ft) in the attitude for which maximum leakage is expected.	Criticality assessment performed for Zr-alloy cladding should be acceptable if it can be determined that the k_{eff} of the fuel system will not increase with the addition of the Cr-coating. Because the neutron cross section of Cr is greater than Zr this will be the case. Even though the neutron cross section of Cr is greater than Zr, the effective cross section of the coated cladding wil be about the same as the uncoated cladding due to the coating being very thin (8-30 μm). See Section 3.0.
Immersion--all packages. A separate, undamaged specimen must be subjected to water pressure equivalent to immersion under a head of water of at least 15 m (50 ft). For test purposes, an external pressure of water of 150 kPa (21.7 lbf/in^2) gauge is considered to meet these conditions.	No impact of this requirement on the fuel. This is a cask requirement.

3.0 Criticality

A criticality assessment is performed for transport packages containing fissile material for normal conditions of transport and for hypothetical accident conditions. A criticality assessment depends on the materials present and the geometry of the materials. The structural and thermal analyses show if the geometry will change or remain the same during normal conditions of transport and hypothetical accident conditions. If the mechanical response of the Cr-coated Zr cladding is the same as the base Zr-alloy cladding, it is possible that existing criticality assessments may be applicable. If the mechanical response is not the same, the applicant should perform new criticality assessments for cases involving deformation of the fuel.

Existing criticality assessment for fuel with Zr-alloy cladding may be applicable to fresh fuel with Cr-coated Zr in the same package if the geometrical response is the same and if the k_{eff} of the fuel system will not increase with the addition of the Cr-coating. Because the neutron absorption cross section of Cr is greater than Zr, it is expected that this will be the case. Even though the neutron absorption cross section of Cr is greater than Zr, the change in the effective cross section of the coated cladding will likely be negligible due to the coating being very thin (8-30 μm).   This would also be true for ceramic coatings of chromium nitride (CrN) and Cr_2O_3. Hydrogen and carbon are both neutron moderators, so applicants should re-perform criticality assessments for ceramic coatings of chromium(I) hydride (CrH) and chromium(II) carbide (CrC).

4.0 Material Properties for Fresh Fuel Transportation

PNNL-28437 Revision 1 (Geelhood & Luscher, 2019) describes an in-depth review of the impact of Cr-coatings on the material properties of the cladding. Specifically, this review addressed thermal conductivity, thermal expansion, emissivity, enthalpy, specific heat, elastic modulus, yield stress, creep rate, axial irradiation growth, oxidation rate, hydrogen pickup, high temperature ballooning, and high temperature steam oxidation. Many of these properties are not relevant to fresh fuel transportation. The properties relevant to fresh fuel transportation are; thermal conductivity, thermal expansion, emissivity, elastic modulus, yield stress, ductility, and fatigue. This section will summarize the data that are available for each of these properties. Many of these properties are not impacted by a thin Cr or Cr-ceramic coating as will be discussed below.

4.1 Cladding Thermal Properties

This section describes the thermal properties of the cladding including thermal conductivity and thermal expansion.

4.1.1 Thermal Conductivity

There are no data available regarding the thermal conductivity of Cr-coated Zr cladding. It is likely that the overall cladding thermal conductivity will not be strongly impacted by a metallic or ceramic coating as the coatings in question are relatively thin. An applicant may choose to treat the cladding as a single material, and if so, should justify the use of Zr thermal conductivity for the coated cladding. Alternatively, the applicant can treat the Zr-substrate and Cr-coating separately and calculate temperature drop across each one separately based on their individual thicknesses and thermal conductivity. This would be similar to the in-reactor treatment of the ZrO_2 that evolves on the surface of the Zr-alloy cladding. Figure 1 shows the thermal conductivity of Zr (Geelhood, et al., 2020) as well as Cr metal (Ho, Powell, & Liley, 1972) and several Cr-ceramics[1][2].

[1] For Cr_2O_3: http://www.globalsino.com/EM/page1828.html

[2] For CrN: https://thermtest.com/materials-database#C

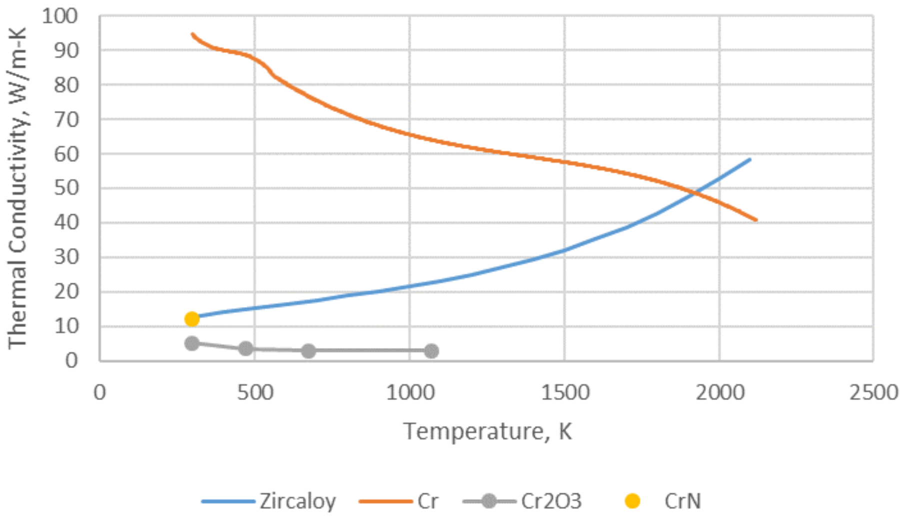

Figure 1. Thermal Conductivity of Zircaloy, Chromium, and Various Chromium-Based Ceramics

For the coating thicknesses being considered (Table 1) it is reasonable to use the thermal conductivity of Zr to model the Cr-coated Zr cladding for fresh fuel transport analyses.

4.1.2 Thermal Expansion

There are no data available regarding the thermal expansion of Cr-coated Zr cladding. Typically, the thermal expansion of a coated part will be the same as that of an uncoated part if the coating is relatively thin. However, thermal expansion data from representative cladding tubes would be useful to justify the correlation and to demonstrate that there has not been a change in behavior with the coating due to thermal expansion mismatch between the substrate and the coating. Thermal expansion mismatch between a coating and substrate typically results in plastic strain in the thin coating which is weaker than the substrate because of its thickness. This is particularly true for the Zr-Cr system since the Zr textured hexagonal crystal structure leads to different thermal expansion in different directions, while the cubic Cr or Cr-ceramic coatings will have similar thermal expansion in all directions. Many ceramics have a limited strain capability. A ceramic coating with a significant thermal expansion mismatch strain may exhibit cracking upon heating and cooling due to the inability of that coating to tolerate plastic strain.

Application methods may also lead to different thermal expansion mismatch. For example, electroplated coatings can usually not tolerate large strains, PVD coatings are usually dense and adherent, and plasma spray coatings can result in anisotropic mechanical properties due to the spray direction, i.e., in plane versus out of plane property differences. The effects of thermal expansion mismatch and their inherent interface strains can be mitigated by processing conditions. For instance, surface treatments that enhance surface area, strain tolerant microstructures, and higher ductility compliant layers can be utilized to reduce interface strains.

For Cr-coatings being considered, the available data indicate that these coatings are not observed to exhibit issues with thermal expansion mismatch. However, this may not be the case with all Cr-coatings as the coating being developed are applied by processes optimized to achieve dense and adherent coatings (Shah, et al., 2018) (Lin, et al., 2018) (Rebeyrolle, Vioujard, Scholer, Kliewer, & Reed, 2019). Table 4 shows specific observations for various Cr coated Zr systems currently under development.

Table 4. Coating Adhesion for Various Coating Concepts

Vendor	Coating	Test Performed	Results
Westinghouse (Lyons, et al., 2019)	Cr-coated ZIRLO™	General Observations	"Optimization was performed in such a way to achieve the necessary coating adherence."
Framatome (Rebeyrolle, Vioujard, Scholer, Kliewer, & Reed, 2019)	Cr-coated M5®	General Observations	"This entire set of variables was combined in a proprietary recipe, optimized to achieve an adherent, dense and uniform coating."
KAERI (Kim, et al., 2015)	Cr-coated Zircaloy-4 by 3D laser coating	Ring compression and ring tensile tests	No cracks in coating observed at 2% or 4% strain. Cracks observed at 6% strain
GNF (Lin, et al., 2018)	ARMOR coated Zircaloy-2	Thermal cycling between 20°C and 350°C and water quench	No cracking or delamination of the coating

For the coating thicknesses being considered (Table 1) it is reasonable to use the thermal expansion of Zr to model the Cr-coated Zr cladding for fresh fuel transport analyses.

4.1.3 Emissivity

The emissivity of a surface is important to determine the heat transfer that occurs from a body due radiation (as opposed to contact conductance or gas conductance). Radiation heat transfer usually has a significant contribution to the overall heat transfer between two surfaces above around 700°C but can be more significant at lower temperatures if there is no contact or in vacuum conditions. Because of this and because fresh fuel does not produce heat, it is not likely that the radiation heat transfer will significantly contribute to thermal analyses on fresh fuel packages. Nevertheless, this section shows the expected emissivity for various surfaces.

Figure 2 shows the emissivity of Zircaloy (Geelhood, et al., 2020), chromium metal (MIKRON), chromium oxide (Burgess & Waltenberg, 1914) and chromium nitride (Douard, Samelor, Delclos, Tendero, & Maury, 2009). The reason for the increase in emissivity in chromium metal is likely the formation of chromium oxide on the surface at higher temperature. Very thin layers of chromium oxide are transparent and do not significantly alter the emissivity of the surface. However, as temperature increases, and the oxide gets thicker the emissivity increases to the values observed for chromium oxide.

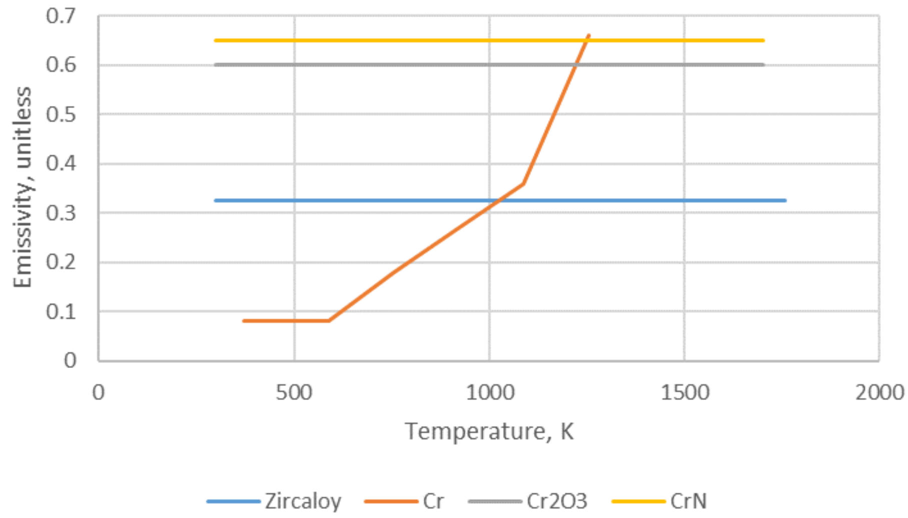

Figure 2. Emissivity of Zircaloy, Chromium, and Various Chromium-Based Ceramics

The emissivity of cladding is not likely necessary for thermal analyses of fresh fuel transport, but if it is used, the applicant should use an emissivity value representative of the surface of the cladding, such as those shown in Figure 2.

4.2 Cladding Mechanical Properties

PNNL-28437 Revision 1 (Geelhood & Luscher, 2019) concluded that for in-reactor performance, recent data on unirradiated Cr-coated Zr indicate the yield stress and elastic modulus of a coated part will be the same (within existing data variability, <10%) as that of an uncoated part (Brachet, et al., 2017) (Kim, et al., 2015) (Shahin, Petrik, Seshadri, Phillips, & Shirvan, 2018). This conclusion is also valid for fresh fuel transport as most of the data was taken at room temperature and on unirradiated material. Since the publication of (Geelhood & Luscher, 2019) more data have been published to corroborate this conclusion.

This section will describe the data for elastic modulus, yield stress, ultimate tensile strength, and ductility that are available for Cr-coated Zr cladding and make conclusions for each property based on those data.

The following discussion applies to an applicant using a stress-based approach. Analyses using a strain-based approach have not been fully qualified, but it appears that special mechanical tests are necessary to certify each cladding material. These tests should be performed on Cr-coated cladding to make assessments regarding the acceptability of Cr-coated cladding in a strain-based approach.

4.2.1 Elastic Modulus

Table 5 shows recent tests performed on various Cr-coated Zr cladding. In general, the coating was found to have no significant impact on the elastic modulus. In one case where ARMOR was noted to possibly decrease the modulus, it was also noted that the coated part was still within the specification for uncoated Zircaloy-2, so this difference would not impact the safety analysis.

Table 5. Elastic Modulus for Cr-coated Cladding

Vendor	Coating	Test Performed	Results
Framatome (Brachet, et al., 2017)	Cr-coated M5®	Tensile tests at 20°C and 400°C	Elastic modulus similar for coated and uncoated cladding
GNF (Lin, et al., 2019)	ARMOR coated Zircaloy-2	Tensile tests at 20°C and 315°C	Tensile test results show that with ARMOR coating, the tensile properties satisfy the requirements for uncoated Zircaloy-2 ARMOR coating appeared to lower the modulus at 20°C relative to uncoated Zircaloy-2, there was no significant difference at 315°C relative to uncoated Zircaloy-2.
KAERI (Kim, et al., 2015)	Cr-coated Zircaloy-4 by 3D laser coating	Ring tensile and ring compression tests at 20°C	Elastic modulus similar for coated and uncoated cladding.

For the coating thicknesses and compositions being considered (Table 1) it is reasonable to use the elastic modulus of Zr to model the Cr-coated Zr cladding for fresh fuel transport analyses. However, since mechanical properties can be impacted by the application of a coating some representative data should be used to confirm this conclusion.

4.2.2 Yield Stress and Ultimate Tensile Strength

Table 6 shows recent tests performed on various Cr-coated Zr cladding. In general, the coating was found to have no significant impact on the yield stress or ultimate tensile strength. In one case where Cr-coated ZIRLO™ was noted to increase the yield stress and ultimate tensile strength it was also noted that the coated part was still within the specification for uncoated ZIRLO™, so this difference would not impact the safety analysis.

Table 6. Yield Stress and Ultimate Tensile Strength for Cr-coated Cladding

Vendor	Coating	Test Performed	Results
Westinghouse (Lyons, et al., 2019)	Cr-coated ZIRLO™	Axial tension tests at 20°C	Yield stress and ultimate tensile strength meet specification for uncoated cladding Yield stress and ultimate tensile strength 4-5% greater than reference uncoated tubes

Vendor	Coating	Test Performed	Results
Framatome (Brachet, et al., 2017)	Cr-coated M5®	Tensile tests at 20°C and 400°C	Yield stress and ultimate tensile strength similar for coated and uncoated cladding.
GNF (Lin, et al., 2019)	ARMOR coated Zircaloy-2	Tensile tests at 20°C and 315°C	Tensile test results show that with ARMOR coating, the tensile properties satisfy the requirements for uncoated Zircaloy-2 ARMOR coating appeared to exhibit no significant difference in yield stress or ultimate tensile strength from uncoated tubes at 20°C and 315°C.
KAERI (Kim, et al., 2015)	Cr-coated Zircaloy-4 by 3D laser coating	Ring tensile and ring compression tests at 20°C	Yield stress and ultimate tensile strength similar for coated and uncoated cladding.
MIT (Shahin, Petrik, Seshadri, Phillips, & Shirvan, 2018)	Cr-coated Zry-4 by cold spray	Burst tests at room temperature	Ultimate tensile strength and burst pressure about the same for coated and uncoated cladding

For the coating thicknesses and compositions being considered (Table 1) it is reasonable to use the yield stress and ultimate tensile strength of Zr to model the Cr-coated Zr cladding for fresh fuel transport analyses. However, since mechanical properties can be impacted by the application of a coating some representative data should be used to confirm this conclusion.

4.2.3 Ductility

Table 7 shows recent tests performed on various Cr-coated Zr cladding. In general, the coating was found to have no significant impact on the measures of ductility such as uniform elongation, total elongation, or burst strain. In one case where Cr-coated ZIRLO™ was noted exhibit 25% lower total elongation than the reference uncoated ZIRLO™ it was also noted that the coated part was still within the specification for uncoated ZIRLO™, so this difference would not impact the safety analysis.

Table 7. Ductility for Cr-coated Cladding

Vendor	Coating	Test Performed	Results
Westinghouse (Lyons, et al., 2019)	Cr-coated ZIRLO™	Axial tension tests at 20°C	Total elongation meet specification for uncoated cladding Total elongation for Cr-coated ZIRLO™ was 25% lower than for uncoated ZIRLO™

Vendor	Coating	Test Performed	Results
Framatome (Brachet, et al., 2017)	Cr-coated M5®	Tensile tests at 20°C and 400°C	Uniform elongation similar for coated and uncoated cladding.
GNF (Lin, et al., 2019)	ARMOR coated Zircaloy-2	Tensile tests at 20°C and 315°C	Tensile test results show that with ARMOR coating, the tensile properties satisfy the requirements for uncoated Zircaloy-2
KAERI (Kim, et al., 2015)	Cr-coated Zircaloy-4 by 3D laser coating	Ring tensile and ring compression tests at 20°C	Uniform elongation and total elongation similar for coated and uncoated cladding.
MIT (Shahin, Petrik, Seshadri, Phillips, & Shirvan, 2018)	Cr-coated Zry-4 by cold spray	Burst tests at 20°C	Burst strain about the same for coated and uncoated cladding

For the coating thicknesses and compositions being considered (Table 1) it is reasonable to use the uniform elongation or other measure of cladding ductility to model the Cr-coated Zr cladding for fresh fuel transport analyses. However, since mechanical properties can be impacted by the application of a coating some representative data should be used to confirm this conclusion.

For some metals, there is a concern regarding ductile to brittle transition, in which the ductility of a metal is markedly reduced or eliminated below a certain temperature. Ductile to brittle transition is typically observed in body-centered-cubic (BCC) metals, but typically face centered cubic (FCC) metals remain ductile even at low temperature. Chromium metal is a BCC metal, and the ductile to brittle temperature transition is between 150°C and 250°C (Harada & Ohmori, 2004). Therefore, at room temperature and for most conditions of transportation, chromium is already in a brittle state, so there is not a concern regarding further loss of ductility at lower temperature. All of the coatings in question are applied at cold conditions which leads to little to no Zr-Cr interface layer which may also be brittle.

Previous analysis performed for in-reactor analysis (Geelhood & Luscher, 2019) indicated that after 2000 days at 300°C to 350°C the interdiffusion layer would only be 0.1 to 0.3μm thick and that this layer would not be enough to embrittle the entire thickness (650-700μm) of the cladding. The interdiffusion that is expected to occur during normal conditions of transport (-40°C to 38°C) will be even lower and likely could not be measured. Even for the highest temperature for hypothetical accident conditions (800°C for 30 minutes) the previous analysis indicates that the interdiffusion layer would only be 0.15μm thick. Calculations for these two conditions are shown in Table 8.

Table 8. Interdiffusion Layer for Cr-coated Zr during NCT and HAC

Condition	CrZr layer thickness
NCT: 38°C for 100 days[1]	3x10^{-7} μm
HAC: 800°C for 30 minutes	0.15μm

4.3 Cladding Fatigue

Cladding fatigue is necessary to evaluate the impact of vibration during NCT on Cr-coated Zr cladding. The cladding fatigue limit is typically based on the sum of the damage fractions from all the expected strain events being less than 1.0. The damage fractions are typically found relative to the O'Donnell and Langer unirradiated Zircaloy fatigue design curve (O'Donnell & Langer, 1964). It is currently unknown if the O'Donnell and Langer unirradiated fatigue design curve would be applicable to Cr-coated Zr. It has been noted (Kvedaras, Vilys, Ciuplys, & Ciuplys, 2006) that in steels, Cr-coating can improve or significantly worsen the fatigue lifetime based on the microstructures of the coating. Lowering of fatigue life was observed in the case of Cr-coated Zr where the fatigue life went down with the application of a coating (Sevecek, et al., 2018). See Table 9.

The process parameters used to apply the coating can strongly influence the microstructure of the coating and possibility the overall cladding fatigue life. Because of this, fatigue data from unirradiated cladding that was produced using a representative process is recommended to either confirm the O'Donnell and Langer unirradiated fatigue design curve or to develop a new fatigue design curve. New fatigue design curves should include a safety factor of two on stress amplitude or a safety factor of 20 on the number of cycles.

Table 9. Fatigue for Cr-coated Cladding

Vendor	Coating	Test Performed	Results
CTU & MIT (Sevecek, et al., 2018).	Cr-coated Zry-4 by cold spray	Cyclic fatigue tests in water at 300°C	Fatigue life for Zircaloy 176,000 to >500,00 cycles. Fatigue life for Cr-coated Zr 50,000 to 166,000 cycles

[1] There is no time specified for NCT, so 100 days was chosen as a typically long period time

5.0 Conclusions

This report provides an assessment of the shipment of fresh UO_2 fuel with Cr-coated Zr cladding. The U.S NRC is specifically concerned about this metallic chromium coatings (8-30 μm) and a proprietary chromium ceramic coating known as ARMOR. This assessment concludes:

- Fresh UO_2 fuel with Cr-coated Zr cladding may be shipped in a Type A fissile package because the Cr-coating doesn't increase the fissile content of the fuel.[1]

- The existing regulations (10 CFR 71) and guidance (NUREG-1609) are sufficient for shipment of fresh UO_2 fuel with Cr-coated Zr cladding because there are no new degradation or failure modes not captured by existing regulations.

- Cladding material properties needed to ensure no loss or dispersal of radioactive material during normal conditions of transportation and to ensure subcriticality during normal conditions of transportation and hypothetical accident conditions are cladding fatigue lifetime, cladding yield stress, and cladding elastic modulus for a stress-based performance analysis[2].

    - Fatigue data on unirradiated, representative Cr-coated Zr cladding can be used to develop a fatigue lifetime or confirm the use of an existing Zr fatigue lifetime.

    - The application of Cr-coatings is not expected to impact the yield stress elastic modulus, or ductility but some representative data should confirm this expectation.

    - The application of Cr-coatings is not expected to impact thermal properties such as thermal conductivity or thermal expansion. If emissivity is used in a thermal analysis, the applicant should use an emissivity value representative of the surface of the cladding.

- Existing criticality assessments for fuel with Zr-alloy cladding may be applicable to fresh fuel with Cr-coated Zr in the same package if the geometrical response is the same and if it can be determined that the k_{eff} of the fuel system will not increase with the addition of the Cr-coating.

    - Cr, CrN, or Cr_2O_3 coatings on Zr are not expected to increase the k_{eff} of the fuel system.

    - Hydrogen and carbon are both neutron moderators, so applicants should re-perform criticality assessments for ceramic coatings of CrH and CrC.

---

[1]A Type A Fissile package is not acceptable for transport of fresh MOX fuel.

[2]Analyses using a strain-based approach have not been fully qualified, but it appears that special mechanical tests are necessary to certify each cladding material. These tests should be performed on Cr-coated cladding to make assessments regarding the acceptability of Cr-coated cladding in a strain-based approach.

6.0 References

Brachet, J., Dumerval, M., Lazaud-Chaillioux, V., Le Saux, M., Rouesne, E., Urvoy, S., . . . Pauillier, E. (2017). Behaviour of Enhanced Accident Tolerant Chromium Coated Zirconium Alloys Claddings. *Enlarged Halden Programme Group 2017* (p. F1.3). Lillehammer, Norway: OECD Halden Reactor Project.

Burgess, G., & Waltenberg, R. (1914). The Emissivity of Metals and Oxides II Measurement with the Micropyrometer. *Bulletin of the Bureau of Standards*, 591-605.

Delafoy, C., Bischoff, J., Larocque, J., Attal, P., Gerken, L., & Nimishakavi, K. (2018). Benefits of Framatome's E-ATF Evolutionary Solution: Cr-Coated Cladding with Cr2O3-Doped Fuel. *TopFuel 2018* (p. A0149). Prague, Czech Republic: European Nuclear Society.

Douard, D., Samelor, D., Delclos, S., Tendero, C., & Maury, F. (2009). Online ncontrol by IR pyrometry of nanostructured multilary CrCx/CrN coatings grown by MOCVD. *Transactions of the Electrochemical Societym 25(8)*, 273-280.

Dumerval, M., Houmaire, Q., Brachet, J., Palancher, H., Bischoff, J., & Pouillier, E. (2018). Behavior of Chromium Coated M5 Claddings upon Thermal Ramp Tests under Internal Pressure (Loss-of-Coolant Accident Conditions). *TopFuel 2018* (p. A0102). Prague, Czech Republic: European Nuclear Society.

Geelhood, K., & Luscher, W. (2019). *Degradation and Failure Phenomena of Accident Tolerant Fuel Concepts, PNNL-28437, Revision 1.* Richland, WA: Pacific Northwest National Laboratory.

Geelhood, K., Luscher, W., Porter, I., Kryiazidis, L., Goodson, C., & Torres, E. (2020). *MatLib-1.0: Nuclear Material Properties LIbrarym PNNL-29728.* Richland, WA: Pacific Northwest National Laboratory.

Harada, Y., & Ohmori, M. (2004). Ductile-brittle transition behavior of rolled chromium. *Journal of Materials Processing Technology*, 93-99.

Ho, C., Powell, R., & Liley, P. (1972). Thermal Conductivity of the Elements. *Journal of Physical Chemistry Reference Data*, 279.

Kim, H.-G., Kim, I.-H., Jung, Y.-I., Park, D.-J., Park, J.-Y., & Koo, Y.-H. (2015). Adhesion property and high-temperature oxidation behavior of Cr-coated Zircaloy-4 cladding tube prepared by 3D laser coating. *Journal of Nuclear Materials*, 531-539.

Kvedaras, V., Vilys, J., Ciuplys, V., & Ciuplys, A. (2006). Fatigue Strength of Chromium-Plated Steel. *Materials Science*, 16-18.

Lin, Y., DeSilva, D., Lutz, D., Connor, M., Michael, S., Yin, L., . . . Rebak, R. (2019). Coated Fuel Rod for Accident Tolerance and Debris Fret Resistance. *TopFuel 2019* (pp. 15-19). Seattle, WA: American Nuclear Society.

Lin, Y., Faucett, R., Desilva, S., Lutz, D., Yilmaz, M., Davis, P., . . . Satterlee, N. (2018). Path Toward Industrialization of Enhanced Accident Tolernat Fuel. *TopFuel 2018* (p. A0141). Prague, Czech Republic: European Nuclear Society.

Lyons, J., Partezana, J., Byers, W., Wang, G., Parsi, A., Walters, J., . . . Shah, H. O. (2019). Westinghouse Chromium-Coated Zirconium Alloy Cladding Development and Testing. *TopFuel 2019* (pp. 8-14). Seattle, WA: American Nuclear Society.

MIKRON. (n.d.). *Table of Emissivity of Various Surfaces.* MIKRON Vertretung Schweiz.

O'Donnell, W., & Langer, B. (1964). Fatigue Design Basis for Zircaloy Components. *Nuclear Science and Engineering*, 1-12.

Rebeyrolle, V., Vioujard, N., Scholer, A., Kliewer, R., & Reed, J. (2019). PROtect Fuel: The Leading E-ATF Solution Delivered by Framatome. *TopFuel 2019* (pp. 1-7). Seattle, WA: American Nuclear Society.

Sevecek, M., Krejci, J., Shahin, M., Petrik, J., Ballinger, R., & Shirvan, K. (2018). Fatigue Behavior of Cold Spray-Coated Accident Tolerant Cladding. *TopFuel 2018* (p. A0126). Prague, Czech Republic: European Nuclear Society.

Shah, H., Romero, J., Xu, P., Oelrich, R., Walter, J., Wright, J., & Gassmann, W. (2018). Westinghouse-Exelon EnCore® Fuel Lead Test Rod (LTR) Program including Coated cladding Development and Advanced Pellets. *TopFuel 2018* (p. A0145). Prague, Czech Republic: European Nuclear Society.

Shahin, M., Petrik, J., Seshadri, A., Phillips, B., & Shirvan, K. (2018). Experimental Investigation of Cold-Spray Chromium Cladding. *TopFuel 2018* (p. A0193). Prague, Czech Republic: European Nuclear Society.

U.S. Nuclear Regulatory Commission. (1999). *Standard Review Plan for Transportation Packages for Radioactive Materials, NUREG-1609.* Washington DC: U.S. Nuclear Regulatory Commission.

U.S. Nuclear Regulatory Commission. (2015). *Title 10 Code of Federal Regulations PART 71—Packaging and Transportation of Radioactive Material.* Washington DC: U.S. Nuclear Regulatory Commission.

Pacific Northwest
National Laboratory

902 Battelle Boulevard
P.O. Box 999
Richland, WA 99354
1-888-375-PNNL (7665)

www.pnnl.gov